Safety Evaluation Report

NUREG-75/075

U. S. Nuclear
Regulatory Commission

related to construction of

South Texas Project, Units 1 and 2

Houston Lighting and Power
Company, et al

Office of Nuclear
Reactor Regulation

Docket No. STN 50-498
STN 50-499

August 1975

POOR ORIGINAL

1547 006

7911150520

NUREG-75/075

August 1, 1975

SAFETY EVALUATION REPORT

BY THE

DIVISION OF REACTOR LICENSING

OFFICE OF NUCLEAR REACTOR REGULATION

U.S. NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

HOUSTON LIGHTING & POWER COMPANY, ET AL

SOUTH TEXAS PROJECT UNITS 1 AND 2

DOCKET NOS. - STN-50-498 AND STN-50-499

Available from National Technical Information Service
Springfield, Virginia 22161

Price: Paper Copy $9.50
Microfiche $2.25

1547 007

TABLE OF CONTENTS

1547 008

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

1547 011

TABLE OF CONTENTS (Continued)

1547 012

TABLE OF CONTENTS (Continued)

1547 013

TABLE OF CONTENTS (Continued)

1547 014

APPENDICES

1547 015

LIST OF FIGURES

1547 016

LIST OF TABLES

1.0 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

The Houston Lighting & Power Company, the City Public Service of San Antonio, the Central Power and Light Company, and the City of Austin (hereinafter referred to as the applicants) filed with the Nuclear Regulatory Commission (NRC or Commission) an application, docketed on July 5, 1974, for licenses to construct and operate its proposed South Texas Project Units 1 and 2 (South Texas Project or facility). The applicants have designated Houston Lighting & Power Company as Project Manager responsible for the technical adequacy of the design, construction, and operation of the South Texas Project Units 1 and 2. The facility will be located in Matagorda County, Texas, approximately 89 miles southwest of Houston, Texas. The South Texas Project will utilize a Westinghouse Electric Corporation standard nuclear steam supply system.

A Preliminary Safety Analysis Report (PSAR) was submitted with the South Texas Project application. This report describes the design of the balance-of-plant structures, systems and components and incorporates by reference the Westinghouse Electric Corporation (Westinghouse) report "Reference Safety Analysis Report" (RESAR-41). RESAR-41 describes the design of the standard nuclear steam supply system.

The initial Commission policy statement on standardization of nuclear power plants was issued on April 28, 1972. It provided the impetus for both industry and the Commission to initiate active planning in their respective areas in order to realize the benefits of standardization while maintaining protection of the health and safety of the public and of the environment. In a subsequent statement issued on March 5, 1973, the Commission announced its intent to implement a standardization policy for nuclear power plants. WASH-1341, "Programmatic Information for the Licensing of Standardized Nuclear Power Plants" was issued August 20, 1974. 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 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).

RESAR-41 was submitted by the Westinghouse Electric Corporation in the form of an application for a Preliminary Design Approval from the Commission and was in response to Option 1 of 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 March 11, 1974, the application for RESAR-41 was docketed.

Our evaluation for RESAR-41 is presented in our Report To The Advisory Committee On Reactor Safeguards, a copy of which is attached as Appendix A to this report. Where we have made use of this evaluation, we have referenced the appropriate sections of RESAR-41 in this report.

The information in the PSAR was supplemented by Amendments 1 through 26. Copies of the PSAR and RESAR-41, as amended, are available for public inspection at the U.S. Nuclear Regulatory Commission Public Document Room, 1717 H Street, N.W., Washington, D.C. 20555 and at the Matagorda County Court House, 1700 Seventh Street, Bay City, Texas.

This Safety Evaluation Report summarizes the results of the technical evaluation of the proposed South Texas facility 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, Implementation of the National Environmental Policy Act of 1969, are discussed in the Commission's Final Environmental Statement issued March 1975.

Upon favorable resolution of the outstanding issues discussed herein and summarized in Section 1.8 of this report, we will be able to conclude that the South Texas facility 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 facility 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 South Texas facility. Construction will be accomplished under the surveillance of the Commission's staff. Prior to issuance of an operating license, 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 license and the Commission's regulations under the continued surveillance of the Commission's staff.

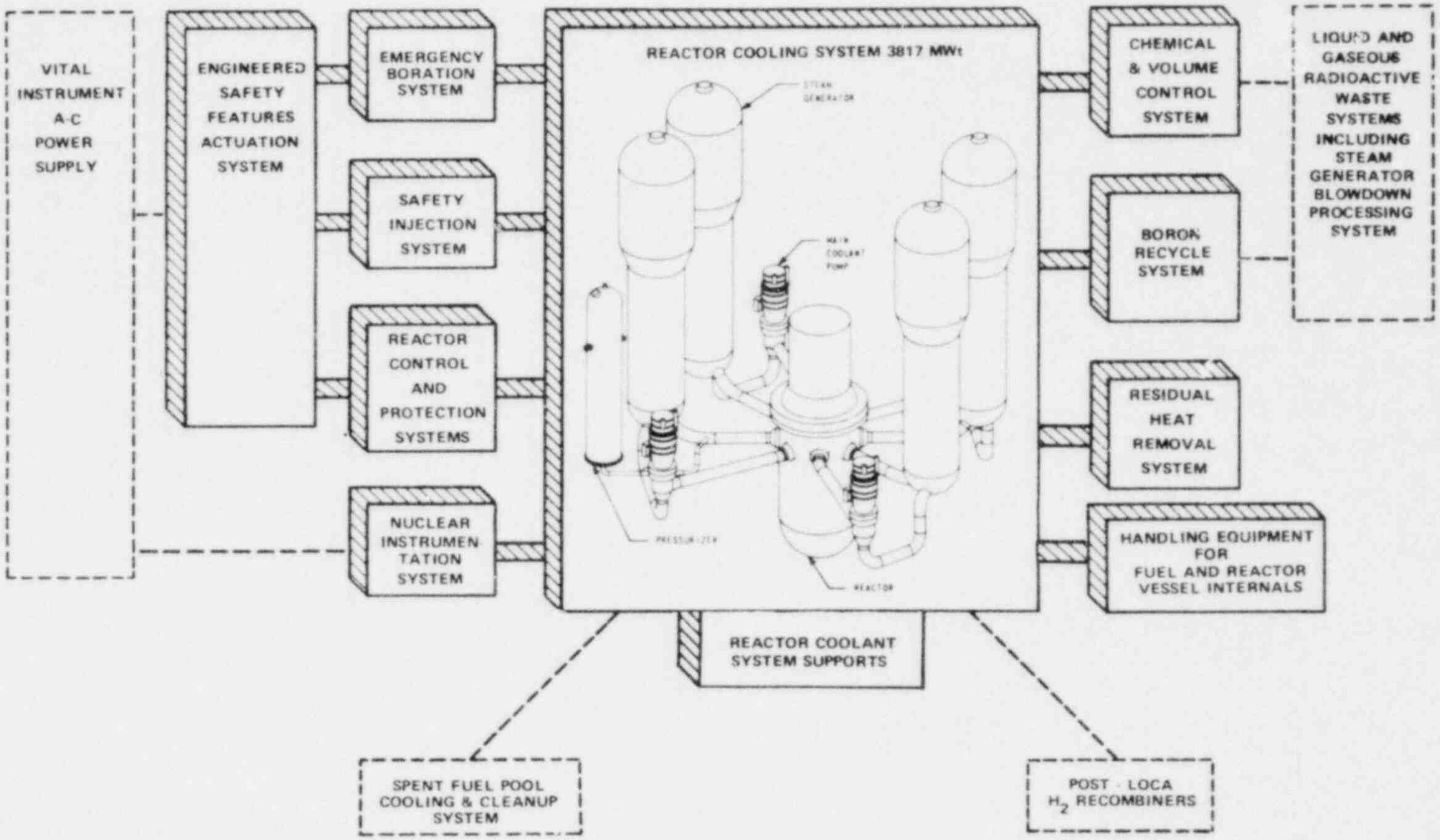
1.2 General Plant Description

The proposed nuclear steam supply system design, as described in RESAR-41, incorporates a pressurized water reactor in a four-loop reactor coolant system. Preliminary designs for control and instrumentation systems, safety systems and power systems which will support the reactor coolant system under normal and accident conditions are also included. Figure 1.1 graphically shows the design scope of RESAR-41.

The RESAR-41 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.

1.2.1 Reactor

The proposed reactor core will consist of fuel rods made from uranium-dioxide pellets contained in slightly cold worked Zircaloy-4 tubing which will be plugged and seal welded at the ends to encapsulate the fuel. The fuel pellets consisting of slightly enriched uranium-dioxide powder will be compacted by cold pressing and then sintered to the desired density. Shifting of the fuel within the cladding prior to fuel



STANDARD SCOPE OF RESAR - 41

NON STANDARD SCOPE OF RESAR - 41 DISCUSSED IN APPENDIX A TO THIS REPORT

Figure 1-1 RESAR - 41 NUCLEAR STEAM SUPPLY SYSTEM

NOTE: THIS FIGURE IS NOT INTENDED TO SHOW ALL INTERCONNECTIONS BETWEEN SYSTEMS loading will be prevented by a stainless steel spring which bears on top of the fuel. All fuel rods will be internally pressurized with helium during the welding process. The design height of the fuel pellets within each rod is 164 inches, while the overall fuel rod length will be 173.3 inches.

The fuel rods will be combined in a 17 x 17 array to form fuel assemblies. These fuel assemblies will have nine spacer grids and contain guide thimble channels for the neutron absorber rods, burnable poison rods or neutron source assemblies. The core will be formed of 193 fuel assemblies divided into three regions, each utilizing fuel of a different enrichment of U-235. The new, highest enrichment fuel will be introduced into the outer core regions, moved inward at successive refuelings, and ultimately removed from the inner region to spent fuel storage. The 164 inch fuel is often referred to as "14-foot fuel" and is designated 17 x 17 XLR by Westinghouse. The proposed fuel enrichment for the core regions are 2.10 weight percent uranium-235 for the inner region, 2.60 weight percent for the middle region, and 3.10 weight percent for the outer region.

The reactor design provides for reactivity control by means of full and part length rod cluster control assemblies, a burnable poison assembly and regulation of boric acid concentration in the reactor coolant. The burnable poison assembly will normally only be used for the initial core because of this core's higher reactivity. The design of the mechanical control rods consists of clusters of stainless-steel clad silver-indium-cadmium alloy absorber rods for insertion into the guide tubes in the fuel assemblies. There are two categories of full length control rod assemblies. Control assemblies will compensate for reactivity changes due to variations in operating conditions of the reactor, and the shutdown assemblies will have the necessary negative reactivity to provide an adequate shutdown margin. The control system for the full length control assemblies will allow the plant to accept step load changes of 10 percent and ramp changes of five percent per minute over the range of 15 to 95 percent of full power under normal operating conditions. The function of the part length control assemblies will be to control axial neutron flux shape and axial xenon oscillations, should they occur.

Water circulating through the reactor vessel and core will serve as a neutron moderator, radiation shield, and coolant. The reactor vessel design is basically the same as that of current 3411 thermal magawatts Westinghouse plants except that the design provides for removal of control assemblies with the vessel head during refueling and a Roto-Lok closure stud design has been incorporated as part of the "Rapid Refueling" concept.

1.2.2 <u>Reactor Coolant System</u>

In the reactor coolant system, primary coolant will be circulated through the reactor vessel and core by four vertical, single stage, centrifugal pumps, one in each of the four cold legs. Significant proposed system operating parameters are listed below.

1547 021

Normal Operating Pressure, pounds per square inch, gauge	2235
Reactor Power, megawatts, thermal	3800
Reactor Vessel Inlet Temperature, degrees Fahrenheit	559.8
Reactor Vessel Outlet Temperature, degrees Fahrenheit	623.8
Total Reactor Flow Rate, pounds per hour	144,700,000
Steam Pressure, pounds per square inch, gauge	1100
Total Steam Flow, pounds per hour	16,960,000

After being heated in the core, the coolant will be circulated through the four U-tube steam generators. It is here that heat will be transferred to the secondary system to form steam to be used to drive the turbine-generator. This coolant system design does not include loop stop valves.

Reactor coolant pressure will be established and maintained by an electrically-heated pressurizer connected to the hot leg piping of one of the loops. The pressurizer will be designed to maintain a saturated steam bubble at the saturation temperature of the existing reactor coolant pressure. This will provide a surge volume to accommodate reactor coolant volume changes. Reactor coolant system overpressure protection will be provided through motor-operated relief valves and self-activated safety valves connected to the pressurizer vapor space.

1.2.3 Facility Structures

The nuclear steam supply system for each unit will be housed in a containment structure. The containment will consist of a steel-lined, prestressed concrete structure. The prestressed concrete structure, including its penetrations, will be designed to safely confine within the leakage limit of the containment, the radioactive material that could be released in the event of an accident. A separate mechanical and electrical auxiliaries building for each unit, to be located adjacent to and abutting the containment structure, will house the radioactive waste processing system, engineered safety features systems and various related auxiliary systems for each unit. A separate fuel handling building for each unit will contain the spent fuel pool and new fuel storage facilities. The fuel handling building will also house the safety injection pumps and the containment spray pumps.

1.2.4 Engineered Safety Features

The RESAR-41 engineered safety features will consist of accumulator tanks, high head and low head safety injection systems, provisions for recirculation of the borated coolant after the end of the injection phase, and the emergency boration system. These systems will assure core cooling and protection for the complete range of postulated primary and secondary coolant pipe break sizes.

The accumulators and the high and low head safety injection systems will provide core protection for both large and small reactor coolant system ruptures. The RESAR-41 design consists of three independent safety injection trains. Each train will be connected to the refueling water storage tank and the containment sump. Each train will include one high head and one low head safety injection pump located external to the containment, one accumulator, and one residual heat removal heat exchanger located inside containment, and will be connected to the hot and cold legs of only one primary loop. For long term cooling following a loss-of-coolant accident, the low head safety injection pumps will recirculate the water collected in the containment sumps through the residual heat removal heat exchangers, through the core and out the break and back to the sumps.

Separate residual heat removal pumps, to be located inside the containment, will be used in conjunction with the residual heat removal heat exchangers for normal plant cooldown.

The emergency boration system will be designed to provide sufficient negative reactivity for safe shutdown capability in the event of any single steam pipe rupture or spurious lifting of a pressure relief valve. This design consists of a source of highly borated water and two parallel boron injection pumps. The pumps will inject the highly borated water into a common header which will connect to the cold leg of all four primary loops. As water is injected, excess water will be discharged from the reactor coolant system and circulated back through the emergency boration system.

Other important engineered safety features which are within the scope of the balance-of-plant are the containment heat removal systems which will consist of the containment spray system, and the reactor containment fan cooler subsystem which will be a part of the reactor containment heating, ventilation and air conditioning system. The containment spray system will provide borated water containing sodium hydroxide to remove heat and radioactive iodine in the event of a postulated loss-of-coolant accident. The reactor containment heating, ventilation and air conditioning system including the reactor containment fan cooler subsystem which will consist of six containment fan coolers located within the containment vessel will serve to maintain normal plant operation. During accident conditions the containment fan cooler subsystem in conjunction with the containment spray system will be capable of maintaining the containment pressure below the containment design pressure even in the event of a single active failure in either system.

1.2.5 Protection Systems

Plant protection systems designs are provided that will automatically initiate appropriate action whenever a monitored condition approaches pre-established limits. These protection systems will act to shut down the reactor, close isolation valves, and initiate operation of the engineered safety features should any or all of these actions be required.

The reactor trip system will shut down the reactor whenever unsafe operating limits are approached. It will consist of sensors which, when connected with analog circuitry consisting of two to four redundant channels, will monitor various plant parameters, and digital circuitry, consisting of two redundant logic trains, which will receive inputs from the analog protection channels to complete the logic necessary to automatically drop the control rod assemblies into the core and shut the reactor down.

The engineered safety features actuation system will consist of adequate instrumentation and controls to sense accident situations and initiate operation of the necessary engineered safety features. The system will consist of:

(1) Three to four redundant analog channels per plant parameter being monitored, and

(2) Two redundant digital logic trains which will receive inputs from the analog protection channels and actuate the engineered safety features.

The functions initiated by this system are:

(1) Reactor trip
(2) Safety injection
(3) Auxiliary feedwater flow
(4) Emergency boration flow
(5) Containment cooling
(6) Containment isolation
(7) Emergency diesel operation
(8) Containment spray
(9) Auxiliary supporting systems

1.2.6 Power Sources

The South Texas Project will be capable of being supplied with electrical power from two independent offsite power circuits and each unit will be provided with independent and redundant offsite emergency power supplies capable of supplying power to shut down the plant safely or to operate the engineered safety features in the event of an accident and a loss-of-offsite power sources.

1.2.7 Refueling

The RESAR-41 nuclear steam supply system will incorporate several new design features intended to reduce the time required for refueling. Westinghouse refers to the combination of these features as the "Rapid Refueling" concept. Significant aspects include quick disconnect head bolts (Roto-Lok), internals and control rods that will be removed with the head, an integral control rod drive mechanism cooling system and missile shield which will be removed with the head, and control rod drive mechanism power cables and instrumentation cables that do not need to be disconnected for head removal. In addition, the shutdown reactivity margin required for refueling will be reduced to 5 percent.

1.3 Comparison with Similar Facility Designs

Some features in RESAR-41 represent new Westinghouse designs. However, many design 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 the RESAR-41 design. Where this has been done, the appropriate sections of Appendix A to this report identify the specific safety evaluation reports involved. These

1547 024

safety evaluation reports are available for public inspection at the Nuclear Regulatory Commission's Public Document Room at 1717 H Street, N.W., Washington, D.C. 20555.

To assist in better understanding the relationship of the RESAR-41 design to other Westinghouse designs, Westinghouse has presented a comparison of principal design features of RESAR-3 Consolidated Version with those for the RESAR-41 nuclear steam supply system in Tables 1.3 and 4.1-1 of RESAR-41. A listing of principal parameters and features is presented in Table 1-1 of Appendix A to this report. Some of the applications which reference RESAR-3 are those for the Catawba plant (docket numbers 50-413 and 414), the Vogtle plants (docket numbers 50-424 through 427), the Millstone 3 plant (docket number 50-423), the Comanche Peak plant (docket numbers 50-445 and 446), and the Seabrook plant (docket numbers 50-443 and 444). Our safety evaluation reports for these other applications are available for public inspection in the Public Document Room at 1717 H Street, N.W. Washington, D.C. 20555.

1.4 Identification of Agents and Contractors

The Houston Lighting & Power Company will act as Project Manager for the applicants and is responsible for the design, construction and operation of the South Texas Project Units 1 and 2.

The applicants have retained Brown & Root, Incorporated (B&R) to perform architectural engineering and construction services. Westinghouse has been contracted to design, manufacture and deliver to the site the nuclear steam supply system and initial cores for the South Texas Project units. Westinghouse will also provide technical assistance during the erection of the nuclear steam supply system, core loading, startup, and pre-operational testing.

The applicants will also utilize consultants, as required, in specialized areas; for example, NUS Corporation is assisting in environmental engineering; EDS Nuclear is assisting in quality assurance; S. M. Stoller Corporation is assisting in fuel management; and Woodward-Clyde Consultants is assisting in seismology and geology studies.

1.5 Summary of Principal Review Matters

Our technical review and evaluation of the information submitted by the applicants considered the principal matters summarized below.

We reviewed the population density and use characteristics of the site environs, and the physical characteristics of the site, including seismology, meteorology, geology and hydrology to determine that these characteristics had been determined adequately and had been given appropriate consideration in the plant design, and that the site characteristics were in accordance with the Commission's siting criteria (10 CFR Part 100, "Reactor Site Criteria") taking into consideration the design of the facility including the engineered safety features proposed.

We reviewed the design, fabrication, construction, and testing criteria, and expected performance characteristics of the facility 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 codes and standards, and that any departures from these criteria, codes and standards have been identified and justified.

We considered the response of the facility to certain anticipated operating transients and postulated accidents. We considered the potential consequences of a few highly unlikely postulated accidents (design basis accidents). We performed conservative analyses of these design basis accidents to determine that the calculated potential offsite 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.

We evaluated the applicants' plans for the conduct of plant operations, including the organizational structure and the general qualifications of operating and technical support personnel, the measures taken for industrial 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 applicants will be technically qualified to operate the plant and will have established effective organizations and plans for continuing safe operation of the facility.

We evaluated the design of the systems provided for control of the radioactive effluents from the facility to determine that these systems can control 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 applicants in such a manner as to reduce radioactive releases to levels that are as low as practicable within the contemplation of the Commission's regulations, 10 CFR Part 50.

We are evaluating the financial data and information provided by the applicants 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 applicants are financially qualified to design and construct the proposed facility. We will report the results of our evaluation in a supplement to the Safety Evaluation Report prior to the commencement of the public hearing.

1.6 Facility Modifications as a Result of Staff Review

During the review of the South Texas Project application, numerous meetings were held with representatives of the applicants, its contractors, and its consultants to discuss the design of the facility and the technical material submitted. A chronological listing of the meetings and other significant events in our review of the application is given in Appendix B to this report. During the course of the review the applicants proposed or we requested a number of technical and administrative changes. These are described in various amendments to the original application.

1547 026

We have listed below the more significant modifications that have resulted from our review. Included are references to the sections of this report where each matter is discussed more fully.

(1) Modification of the air intakes for the control room to include detectors for acetaldehyde and vinyl acetate (Sections 2.2 and 6.5).

(2) Incorporation of an acceptable loose parts monitoring system (Section 5.4.7).

(3) Modification of the fuel handling system to preclude travel of the fuel cask over safety related systems (Section 9.1.4).

(4) Modification of the design of the two outside control room air intake to withstand tornado missiles (Section 9.4.1).

(5) Modification of the quality group classification of certain radwaste system components (Section 11.2).

(6) Modifications related to the nuclear steam supply system (Section 1.6 of Appendix A to this report).

1.7 Requirements for Future Technical Information

The applicants have identified in Section 1.5 of the PSAR and in Section 1.5 of the RESAR-41, certain development programs applicable to the South Texas Project facility. These programs, that are aimed at verifying the nuclear steam supply system design and confirming the design margins, are all being conducted by Westinghouse. The objectives, schedules for completion, and current results are summarized in RESAR-41. Our evaluation of this information is presented in Section 1.4 of Appendix A to this report.

In summary, the verification programs have been reviewed and we have concluded that (1) the test programs outlined in RESAR-41, if carried out as stated, will provide in a timely manner the necessary information to verify the design and safe operation of RESAR-41 nuclear steam supply systems, and (2) in the event any of the programs provide unexpected results, appropriate restrictions on operation can be used and/or modifications in designs can be made to protect the health and safety of the public.

1.8 Outstanding Issues

We have identified certain outstanding issues in our review many of which will require that the applicants provide additional information to confirm that the proposed design will meet our requirements. Items 1 through 4 are issues that require additional information. Items 5 through 10 are issues where we are currently reviewing information provided by the applicants, and where our review is not yet complete. These items are listed below and are discussed further in the sections of this report as indicated.

(1) We have evaluated the interface information contained in RESAR-41 and the South Texas Project and find it to be inadequate. Westinghouse and Houston Lighting & Power Company acknowledge that additional interfaces should be provided. Westinghouse has undertaken to conduct an accelerated short-term program to supplement the interface information already provided in RESAR-41 in an effort to identify essentially all of the safety related interfaces. The major technical information

transmitted by Westinghouse to its customers includes the safety analysis, transient analyses, and normal operations which is contained in a set of documents referred to as a Standard Information Package. Westinghouse has reviewed this information in detail to identify and define additional interface information for submittal to the staff. We have reviewed the additional interface information developed by Westinghouse in its short-term program and have conducted an audit of selected portions of the information used by Westinghouse in its program including its Standard Information Package. On the basis of this review and audit, we have recognized that in the near term our objectives of resolving the interface problem on the nuclear steam supply system standard design applications can not be completed satisfactorily to preclude a schedule slippage of utility applications referencing these standard designs. However, since the RESAR-41 and South Texas Project applications are being reviewed by the staff at the same time, we will review and resolve the outstanding interface matters on the South Texas Project without requiring the same degree of interface definitions as is believed to be necessary for issuance of a Preliminary Design Approval for RESAR-41. Therefore, we will review those outstanding interface issues associated with RESAR-41 to confirm that the South Texas Project balance-of-plant is compatible with the RESAR-41 nuclear steam supply system design. We will report the results of our review in a supplement to the Safety Evaluation Report.

(2) The issues identified as Items 2 through 11 of Section 1.7 of Appendix A to this report must be resolved for the South Texas Project prior to a decision for issuance of construction permits. We will report the results of our review in a supplement to the Safety Evaluation Report.

(3) We require an analysis to determine the minimum containment pressure in accordance with the requirements of Appendix K to 10 CFR Part 50 (Section 6.3).

(4) We require an analysis which demonstrates that monitors in the control room and fuel handling building ventilation systems should not be considered in the list of variables associated with engineered safety features actuation and for monitoring during and after an accident (7.5.1).

(5) Evaluation of the subsidence portion of the monitoring program including the design criteria for long term settlements, differential settlement and tilting of seismic Class I structures (Sections 2.5.2 and 2.5.5).

(6) Evaluation of the grid frequency decay rate trip and frequency set point specification (Section 7.2).

(7) Evaluation of the control room smoke detection system (Section 9.4.1).

(8) Evaluation of the diesel generator building design and the heating, ventilation and air conditioning system regarding the possibility of a long lasting fire resulting in loss of building integrity and/or spread of the fire to other fuel tanks and diesel generators in the building (Sections 9.4.4 and 9.5.1).

(9) Evaluation of the radioactive waste systems regarding Appendix I of 10 CFR Part 50 (Section 11.1).

(10) Evaluation of the applicants' financial qualifications (Section 20.0).

(11) We have also identified certain issues where we have stated our position and the applicants have orally agreed to conform with these positions. Subject to confirmatory documentation we consider that these items, listed below, are resolved.

(a) We require that the South Texas Project facility be designed to the spectrum of missiles and impact velocities as described in either Table 3.1 or Table 3.2 of this report (Section 3.5.1).

(b) We require that the applicants commit to comply with the requirements of the Institute of Electrical and Electronics Engineers (IEEE) Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations - IEEE Std 323-1974 (Section 3.11).

(c) We require automatic isolation of the control room heating, ventilation and air conditioning system by signals from radiation detectors located within the outside air intake for the control room (Section 6.5).

(d) We require that the fuel handling building exhaust subsystem be designed to seismic Category I requirements (Section 6.6 and 9.4.3).

1547 029

2.0 SITE CHARACTERISTICS

2.1 Geography and Demography

The proposed South Texas Project site is located in Matagorda County, Texas, 12 miles south-southwest of Bay City, Texas, and 89 miles southwest of Houston, Texas. Figure 2.1 shows the site location with respect to Jackson County, Wharton County, Brazoria County and Calhoun County lines. Figure 2.2 shows the site with respect to nearby communities.

The proposed site is situated on 12,352 acres of flat, rural land within the Coast Prairie region which extends in a broad band parallel to the Texas Gulf Coast. The Colorado River flows along the eastern boundary of the site. A cooling reservoir will cover 7000 acres of the site property. Figure 2.3 shows the principal features of the site including the site property limits and exclusion area. The exclusion area, which is completely within the site property limits, has a minimum boundary distance of 4,692 feet (1430 meters). The applicants own all of the surface and mineral rights within the exclusion area. No public transportation routes will traverse the exclusion area at the time of facility operation. At present, a county highway (FM 521) crosses the southern part of the exclusion area. However, this highway will be rerouted so that it will lie just outside the northern boundary of the exclusion area. The applicants have made preliminary arrangements with the Matagorda County Highway Department to reroute county highway FM 521.

The only portion of the exclusion area intended for use by the public will be the visitors center, the plant access road leading to it, and a picnic area located near the visitors center.

The population in the region surrounding the South Texas Project site is low. Table 2.1 shows the 1970 census residence population and future projected populations as a function of distance out to 50 miles from the site.

TABLE 2.1

POPULATION DATA

Radius Miles	1970	1980	2020
0-5	217	1,173	1,354
0-10	3,025	3,621	5,483
0-30	42,594	55,040	124,822
0-50	176,234	263,691	795,974

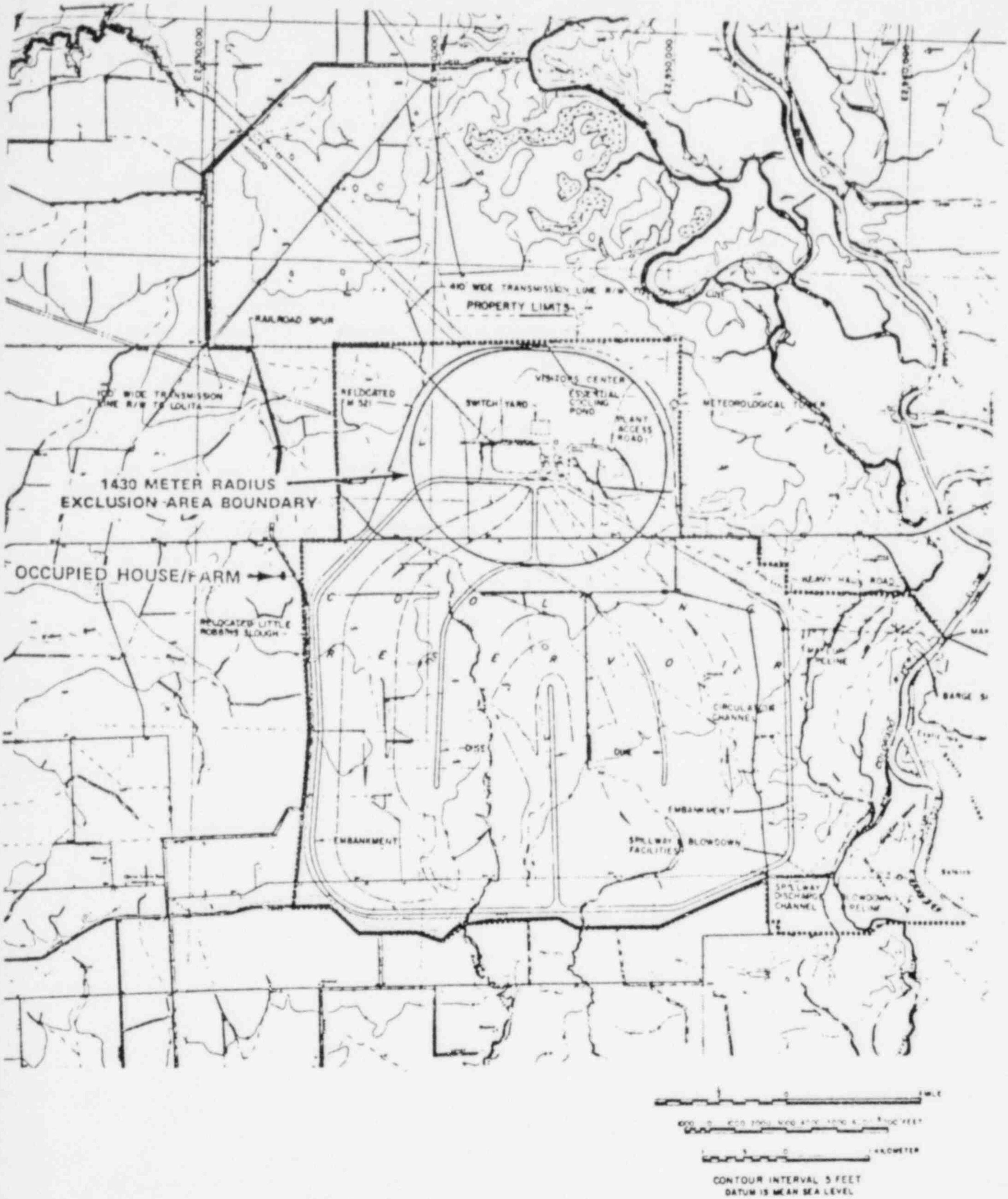
1547 030

GENERAL SITE LOCATION
FIGURE 2.1

1547 031

1547 032

LOCAL SITE AREA
FIGURE 2.2



SITE AND EXCLUSION AREA BOUNDARY
FIGURE 2.3

1547 033

Figure 2.4 shows the 1980 cumulative resident population as a function of distance from 0-50 miles. For reference, the cumulative population corresponding to a moderately populated area of 500 people per square mile is also shown. The data in Figure 2.4 illustrate that the population at all distances out to 50 miles of the site is much less than 500 people per square mile.

We obtained an independent estimate of the 1970 population within 50 miles of the site from Bureau of the Census data and found that this population figure (173,989) agreed closely with the applicants' value. The applicants' projected population growth rate for the area within 50 miles was compared to the population projections of the Bureau of Economic Analysis for Economic Area No. 141, an area comprising Houston, Texas, and the surrounding counties including Matagorda County. This comparison indicated that the applicants' growth projections of 35 percent per decade are higher than the Bureau of Economic Analysis' projections of 14 percent per decade. The applicants have specified a low population zone with an outer radius of three miles. The 1970 resident population within the low population zone was estimated to be approximately 55 persons. There are no significant transient population movements within the low population zone. The retirement and recreational communities of Selkirk Island and Exotic Isle are located between 3.5 and five miles southeast of the site. These communities are projected to have a maximum of 367 seasonal and permanent dwellings when fully developed.

Our review of the preliminary emergency planning for the site has confirmed the practicability of taking protective measures, including evacuation, within the low population zone and that the retirement communities southeast of the plant will not pose any unusual emergency planning problems (see Section 13.3 of this report).

Bay City, Texas, whose nearest corporate boundary is 12 miles north-northeast of the site, has been selected by the applicants as the population center based on population growth estimates. Bay City had a 1970 population of 11,733 and the applicants project that the population will be about 24,000 persons in 2020. We agree with the applicants that it is reasonable to consider Bay City as the population center for purposes of comparison with the guidelines of 10 CFR Part 100. The population center distance of 12 miles is greater than the minimum distance of one and one-third times the distance from the center of the site to the outer boundary of the low population zone as required by 10 CFR Part 100.

On the basis of the 10 CFR Part 100 definitions of the exclusion area, low population zone and population center, and the calculated potential radiological dose consequences of postulated design basis accidents presented in Section 15.0 of this report, we have concluded that the exclusion area, low population zone, and population center distances specified for the South Texas Project site meet the requirements of 10 CFR Part 100 and are acceptable.

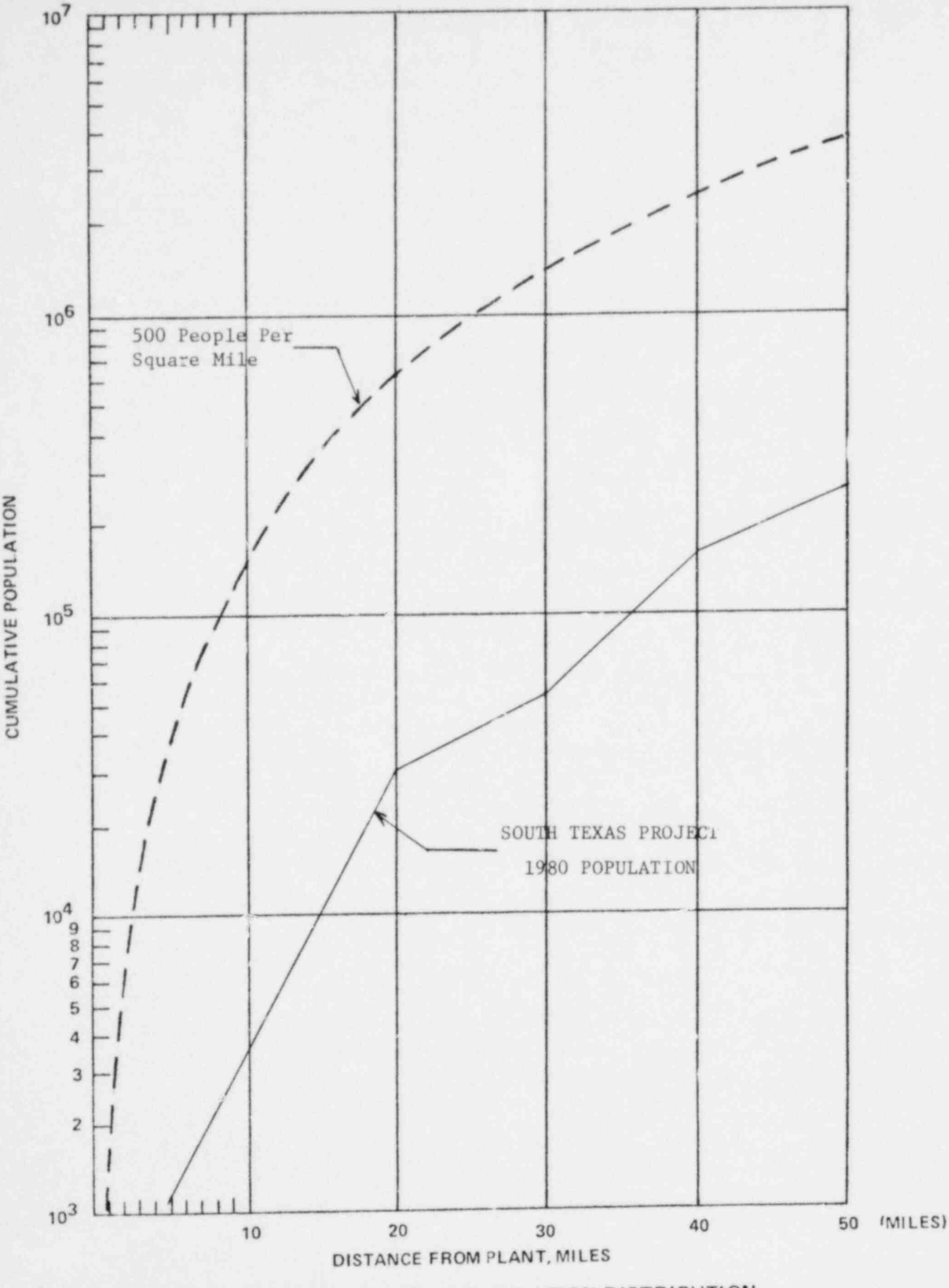


FIGURE 2.4 CUMULATIVE POPULATION DISTRIBUTION

1547 035

2.2 Nearby Industrial, Transportation, and Military Facilities

The closest highway to the proposed site will be county highway FM 521 which will pass around the outside of the northern boundary of the exclusion area. The nearest major highway is State Route 60 which is located about seven miles east of the proposed site.

There are no railroad lines within five miles of the proposed site other than two industrial spur lines which terminate about five miles north-northeast of the site.

The Colorado River runs in a generally north-south direction east of the proposed site with its closest point of approach being about three miles. The river is used for barge transportation between the Gulf Intracoastal Waterway which is about ten miles south of the proposed site and a turning basin on the river which is located about five miles north-northeast of the proposed site. The barges carry chemicals, petroleum products, and oyster shells to the industrial facilities located along the river.

The largest industrial facility in the vicinity of the site is the Celanese Chemical Company plant located on the Colorado River 4.8 miles north-northeast of the proposed site. The plant employs between 400 and 500 workers and produces a variety of chemicals including four which the applicants have identified as being hazardous, namely acetaldehyde, cyclohexane, vinyl acetate, and anhydrous ammonia.

A gasoline and fuel oil terminal facility is also located on the Colorado River 4.8 miles north-northeast of the site. The terminal has the capacity to store 75,000 barrels of gasoline.

The nearest transmission pipeline to the site is a 16-inch natural gas pipeline which is 2.1 miles northwest at its closest point of approach. A four-inch natural gas gathering line is 1.6 miles west of the site and a 30-inch natural gas pipeline is located 4.5 miles north of the proposed site. No liquid petroleum gas or liquid natural gas lines are located within five miles of the proposed site. There are several oil and gas (primarily gas) production fields within five miles of the proposed site. The South Duncan Slough and Petrucha fields, the closest gas producing fields, are located 1.7 miles and 3.5 miles from the proposed site, respectively.

Liquid petroleum gas is stored in underground salt dome formations in the region. The nearest underground storage facility is the Markham salt dome which is 16 miles from the proposed site. At this distance, the underground storage of liquified petroleum gas presents no hazard to the proposed site. Two other salt domes are located in Matagorda County, the closest of which is ten miles from the proposed site. However, no liquified petroleum gas is stored in either of these salt domes. Previously performed extensive underground exploration of Matagorda County indicates that there is little likelihood of the existence of other salt domes in the vicinity of the site.

A small airport with a 3,700 foot grass runaway located 9.5 miles west-northwest is the closest airport to the proposed site. Two low level federal airways are located at distances of five and nine miles northeast of the site. A low level military airway (OB-19) passes over the proposed site area. The applicants state that flight route OB-19 was a special purpose training route which was last used in 1971 and that there are no current plans for its reactivation. Furthermore, the U. S. Air Force has indicated in a letter to the applicants that OB-19 will be modified to assure a minimum clearance distance of five miles from the site. Other than flight route OB-19, there are no military bases or facilities within five miles of the site.

The applicants have evaluated and we have reviewed the potential consequences of postulated explosions on the transportation routes and in the gas production fields near the site, the postulated release of hazardous chemicals in the site vicinity, and the delayed ignition of flammable vapor clouds from a postulated pipeline accident and a gas well blowout.

The applicants have evaluated postulated explosions involving a gasoline truck on county highway FM 521, a truck carrying alpha trinitrotoluene (TNT) on county highway FM 521, a 15,000 barrel barge on the Colorado River 2.75 miles from the site carrying a 10 percent gasoline - 90 percent air mixture, and a 75,000 barrel storage tank containing a 10 percent gasoline - 90 percent air mixture at the petroleum terminal 4.8 miles north-northeast of the site. On the basis of our review we conclude that the applicants have acceptably demonstrated that none of these postulated explosive accidents will affect the safe operation of the facility.

The applicants have also analyzed postulated accidental releases from the largest single storage vessel for the four identified toxic chemicals (acetaldehyde, cyclohexane, vinyl acetate and anhydrous ammonia) stored at the chemical plant 4.8 miles north-northeast of the site and determined the effects of these postulated releases on the habitability of the control room. The methods and assumptions used in the analysis were consistent with those given in Regulatory Guide 1.78. The results of this evaluation showed that the control room is adequately protected with the addition of detectors and automatic isolation for acetaldehyde and with detectors only for vinyl acetate. Since the concentrations of anhydrous ammonia or cyclohexane do not exceed their toxicity limits, no detection instrumentation or protective action is needed for these chemicals. The applicants also analyzed postulated accidental releases of toxic chemicals shipped by barge on the Colorado River and by truck on county highway FM 521. The results of these analyses showed that no additional protection is required for the control room beyond that specified for accidental releases from the chemical plant. Since no chlorine will be stored onsite, and there is no identifiable reason for it being transported in the vicinity of the site, accidental releases of chlorine are not considered to represent a hazard. We have concluded that the applicants have included acceptable design features in the proposed facility to protect the control room from postulated accidental releases of toxic chemicals in the vicinity of the site. Control room habitability is discussed further in Section 6.5 of this report.

1547 037

The applicants have stated that there is little or no potential for future expansion of the oil and gas production fields within five miles of the proposed site. This conclusion was based on production data from the existing fields and geological data for the site vicinity. We agree with the applicants that it appears unlikely that there would be further development of the gas and oil fields in the vicinity of the proposed site; however, the possibility of such development cannot definitely be ruled out. Therefore, we requested that the applicants evaluate the potential hazard to the site in the event that sometime over the lifetime of the plant successful drilling operations might be conducted closer to the proposed site than is now indicated. The applicants provided an analysis of a "worst case" type of drilling accident which was assumed to occur at the site boundary.

The applicants postulated that a well blowout occurred and gas was continuously released at the maximum flow rate which could be delivered from the gas bearing strata in the vicinity of the proposed site. Back flow from a connecting pipeline was also included in the gas flow rate. Five percentile (accident) meteorology conditions were assumed and the downwind (toward the site) extent of the flammable limits of the gas cloud were determined. It was conservatively assumed that the cloud remained at ground level and no credit was taken for the inherent buoyancy of the natural gas cloud. The unconfined gas-air mixture was assumed to detonate at the approximate cloud centroid, a point 1460 feet downwind of the source, and the resultant blast overpressures and ground accelerations as well as missile trajectories were calculated for the nearest proposed safety related structures. This analysis indicated that the consequences of the gas cloud explosion would not adversely affect the safety of the facility. We have reviewed the applicants' analysis and concur with their conclusions.

In addition to the analysis of a postulated drilling accident, the applicants held discussions with State and Federal government authorities and industrial representatives and reviewed reports on well blowouts. These discussions and reports did not produce any indications of potential damage occurring beyond about 1600 feet from the well.

The applicants have analyzed postulated pipeline accidents involving releases from the 16-inch natural gas line 2.1 miles from the proposed site and the 30-inch natural gas line 4.5 miles from the proposed site. The evaluation included a detonation at the rupture point and the delayed detonation of a flammable gas cloud downwind from the release point conservatively assuming the gas to be non-buoyant. The applicants concluded that natural gas pipeline accidents in the vicinity of the site do not present a hazard which could affect the safe operation of the plant. We concur with the applicants' conclusion.

We have concluded that with regard to potential accidents resulting from activities at nearby industrial, transportation, and military facilities, the proposed facility design is acceptable.

2.3 Meteorology

2.3.1 Regional Climatology

The proposed South Texas Project site is located in the flat coastal plains of Southern Texas about 15 miles northwest of the Gulf of Mexico. The climate of this region of Southern Texas is predominately humid subtropical, influenced during much of the year by the anticyclonic circulation of the Azores-Bermuda high pressure system. Winters are generally short and mild, with an occasional incursion of continental polar air bringing cooler temperatures and northwest winds. Summers are long, hot, and humid, with maritime tropical air masses predominating over the area.

2.3.2 Local Meteorology

Climatological data from Victoria (located 59 miles west of the site), Galveston (located about 75 miles east-northeast of the site), Houston (located 89 miles northeast of the site), and available onsite data have been used in assessing the local meteorological characteristics of the proposed site.

Mean monthly temperatures in the area of the proposed site may be expected to range from about 55 degrees Fahrenheit in January to about 83 degrees Fahrenheit in July. Extreme temperatures in the area have been 110 degrees Fahrenheit at Victoria and 8 degrees Fahrenheit at Galveston.

Precipitation is well-distributed throughout the year. Annual average precipitation in the area ranges from about 36 inches at Victoria to about 46 inches at Houston, with most stations in the proposed site area averaging 42 to 43 inches. The maximum 24 hour rainfall in the area was 14.35 inches in Galveston, which occurred in July 1900. Snowfall is generally negligible in the area, although 15.4 inches fell in 24 hours at Galveston in February 1895.

Wind data from the 33-foot level of the onsite meteorological tower for the period July 20, 1973 through July 20, 1974 indicate prevailing winds from the southeast, south-southeast, and south, which occur about 44 percent of the time. The mean wind speed at the 33-foot level for the one year period was 10.7 miles per hour. Calm conditions were only reported 0.7 percent of the time. The "fastest mile" wind speed reported in the area was 100 miles per hour at Galveston in September 1900.

The applicants have examined reports of extreme wind speeds in the site area and have concluded that the peak gusts reported from Port Lavaca and Matagorda cannot be accepted with much confidence. However, the applicants have performed a study to determine an appropriate operating basis wind speed (defined as the "fastest mile" wind speed with a recurrence interval of 100 years) based on data from Corpus Christi, Galveston, and Victoria. The selected operating wind speed of 120 miles per hour with a peak gust value of 156 miles per hour, as identified in Section 2.3 of the PSAR, is acceptable based on the data available.

The applicants performed a study of sea breeze penetration to the site during the period July 20, 1973 through July 20, 1974. Sea breeze occurrence was confirmed on 35 days at the proposed site.

As a result of circulation patterns that bring warm, moist, unstable air from the Gulf of Mexico in all months of the year, thunderstorms can be expected on about 59 days annually. Since the proposed site is located near the center of a two-degree latitude-longitude square, tornado occurrences were examined for the two-degree square. During the period 1965-1967, 138 tornadoes were reported in this two-degree square, giving a mean annual tornado frequency of 2.7 for a comparable one-degree square containing the site. The recurrence interval for a tornado at the plant site, as computed by the method in the paper by H. C. S. Thom, "Tornado Probabilities," is 550 years. In the period 1871-1971, about 36 tropical storms, hurricanes, and depressions passed within 50 miles of the site. There were about four atmospheric stagnation cases totaling about 16 days during the period 1936-1970.

The applicants have selected a design basis tornado, consistent with the recommended tornado model presented in Regulatory Guide 1.76, which is adequately conservative for that area of the country.

2.3.3 Onsite Meteorological Measurements Program

An onsite pre-operational meteorological measurements program was initiated in July 1973, consisting of the accumulation of data from instruments installed on a 195-foot tower located about 5000 feet northeast of the proposed reactor structures. The emergency cooling pond will be located between the reactor structures and the meteorological tower. Instrumentation on the meteorology tower consists of windspeed and direction sensors at the 33-foot and 195-foot elevations, instruments to measure the vertical temperature gradients between the 33-foot and 195-foot elevations, ambient dry bulb and dewpoint temperature sensors at the 33-foot elevation, and instruments to measure the precipitation and solar radiation near the ground surface.

Data are recorded in analog form on strip charts (since July 1973) and digitally on magnetic tape (since December 1973). The applicants have performed a correlation study between the two recording systems for the period November 5, 1974 through January 6, 1975. The correlation between the two systems appears reasonably good.

Based on the above considerations, we have determined that the onsite meteorological program complies with the recommendations of Regulatory Guide 1.23 and, therefore, is acceptable.

The applicants submitted one full year (July 20, 1973 through July 20, 1974) of onsite joint frequency distributions of wind speed and direction by atmospheric stability (defined by the vertical temperature gradient between the 33-foot and 195-foot elevations) from the 33-foot level. Data recovery for this period was about 97 percent. These data have been used to evaluate atmospheric dispersion conditions at the proposed site.

The applicants have committed to collect a second year of onsite data. They further committed that a decision to continue the onsite meteorological program during construction will be made after analyses of the variability of diffusion conditions from year to year and further discussions with us concerning the effects of the 7000 acre cooling reservoir on local meteorological conditions. We find this commitment acceptable.

2.3.4 Short-Term (Accident) and Long-Term (Routine) Diffusion Estimates

Utilizing standard staff practices discussed below, we have evaluated the meteorological diffusion characteristics of the site for both accident analysis and routine releases analysis purposes.

The evaluation of the calculated offsite doses resulting from radioactive releases due to postulated accidents required calculations of the relative concentration (X/Q) for the first 30 days following an assumed accident. The impact of routine radioactive releases required calculations of an annually averaged relative concentration. These relative concentrations were then incorporated into dose analyses.

Accident dose analyses utilize calculated relative concentration values which vary with time. We use our most conservative assumptions when calculating the relative concentration values for the first eight hours following an assumed accident. Additional credit is given for diffusion and spread of the gaseous plume for time periods beyond the first eight hours.

The calculated dose at the minimum exclusion distance at the end of the first two hours and the 30-day dose at the low population zone boundary must be within 10 CFR Part 100 limits.

In our evaluation of diffusion rates for short term (0-2 hours at the minimum exclusion distance and 0-8 hours at the outer boundary of the low population zone) accidental releases from the buildings and vents, a ground level release with a building wake factor of 1320 square meters was assumed. Using the model described in Regulatory Guide 1.4 (Revision 2, June 1974), our calculated results are: (1) the relative concentration for 0-2 hours that is exceeded five percent of the time was found to be 1.7×10^{-4} seconds per cubic meter at the minimum exclusion distance of 4692 feet (this relative concentration is equivalent to dispersion conditions produced by Pasquill Type F stability with a wind speed of 1.5 meters per second); and (2) the relative concentration for 0-8 hours was found to be 2.1×10^{-5} seconds per cubic meter at the outer boundary of the low population zone (three miles).

The calculated relative concentration values, in seconds per cubic meter, t the low population zone for long-term accidental ground release were found to be 1.4×10^{-5} for the 8-24 hour period, 5.8×10^{-6} for the 1-4 day period, and 1.6×10^{-6} for the 4-30 day period.

1547 041

For our evaluation of routine effluent releases, long-term diffusion calculations were made using the procedures described in Regulatory Guide 1.42 (Revision 1, March 1974). We have determined the highest offsite annual average relative concentration for vent releases, assuming a ground-level release, to be 1.6×10^{-6} seconds per cubic meter at the minimum exclusion distance of 4692 feet north of the reactor complex.

We have concluded that the meteorological data presented by the applicants for the period from July 20, 1973 to July 20, 1974 provide an acceptable basis for determining conservative estimates of atmospheric dispersion for calculating accidental and routine gaseous releases from the South Texas facility.

2.3.5 Meteorological Heat Dissipation

The applicants have examined only seven years (1951-1957) of data for determining the design conditions of the emergency cooling pond for maximum evaporative loss, although 20 years (1951-1970) were examined for determining the design conditions for minimum cooling. Although the examination of only a seven year period of data, particularly a period that ended 18 years ago, is not acceptable, the conditions selected by the applicants are adequately conservative for two units considering the design of the emergency cooling pond as discussed in Section 2.4.3. of this report.

2.4 Hydrology

2.4.1 Hydrologic Description

The proposed site is located in South Central Matagorda County, Texas, west of the Colorado River. Plant grade is to be at elevation 28 feet above mean sea level (MSL). Access to safety related facilities will be at or above 28 feet above mean sea level. All safety related facilities subject to the design basis flood are to be protected by waterproof doors.

The principal surface hydrology features of this site are the Colorado River, the main cooling reservoir, and the emergency cooling pond. Cooling water for normal operation will be provided by a 7000 acre cooling reservoir impounded by a non-seismic Category I earthen and soil-cement embankment. There will be several dividing dikes within the cooling reservoir to enhance cooling of heated effluents.

Emergency cooling water will be provided by a separate, excavated emergency cooling pond, eight feet deep with an area of approximately 40 acres. The emergency cooling pond will be surrounded by a dike. That portion of the dike which protects the intake, discharge, and pumping structures from the design basis flood damage will be designed to seismic Category I requirements as shown in Figure 2.5. All pumping machinery for the emergency cooling pond will be protected behind waterproof doors in seismic Category I buildings.

Within the emergency cooling pond, an interior seismic Category I dividing dike will prevent possible thermal short-circuiting between the intake and discharge. All service water for normal shutdown will be provided by this pond.

Makeup water for the cooling reservoir will be provided by the Colorado River. Blowdown from the cooling reservoir will be discharged downstream from the intakes. Makeup and blowdown may be intermittent, and will depend on the flow rate in the Colorado River being above 300 cubic feet per second. Makeup water for the emergency cooling pond will also be provided from the Colorado River. Wells will be provided as a backup supply in the event surface water is unavailable.

The Colorado River near the proposed site is used for intermittent irrigation. The nearness of the site to the Gulf of Mexico causes the Colorado River to be estuarine, and brackish during much of the year, thereby limiting its use. The applicants have identified several users of the Colorado River near the proposed site. The nearest user is 4.5 miles downstream from the proposed South Texas Project site.

2.4.2 Flood Potential

The potential for site flooding from precipitation events, hurricanes, tsunamis, and dam and embankment failures has been investigated by the applicants and evaluated by us.

The applicants have estimated that the probable maximum flood, as defined by Regulatory Guide 1.59, on the Colorado River would cause a stillwater level of about 30 feet mean sea level which is about two feet above plant grade. This water level is well below the design level of safety related structures. We have reviewed the applicants' analysis of the probable maximum flood and concur with the applicants' estimate.

The applicants have analyzed the flood due to a combination of postulated failures of upstream dams on the Colorado River coupled with the standard project flood. This event would produce a stillwater level of about 33 feet above mean sea level. The added effect of wind-induced waves would cause a water level of about 39 feet above mean sea level. This level is used as the design basis for safety related structures on the north side of the reactor buildings. We have reviewed the applicants' analysis of the probable maximum flood and concur with the applicants' analysis for this event, and have concluded that the applicants' analysis is conservative.

The applicants have also analyzed the storm surge from the probable maximum hurricane, as defined in Regulatory Guide 1.59, coupled with the 100 year flood in the Colorado River, and found that the stillwater level at the site would be about 30 feet above mean sea level. High winds associated with the hurricane could cause additional runup of about 11 feet on the south embankment of the cooling reservoir. Safety related structures of the plant itself, however, would be protected by the cooling reservoir embankments. Therefore, this event would not produce the design basis flood at the site. We have reviewed the analysis for this event and concur with the applicants' conclusions.

The design basis flood for all safety related structures, except the north walls of the reactor buildings, could be caused by a failure of part of the embankment surrounding the main cooling reservoir. The applicants have analyzed several hypothetical

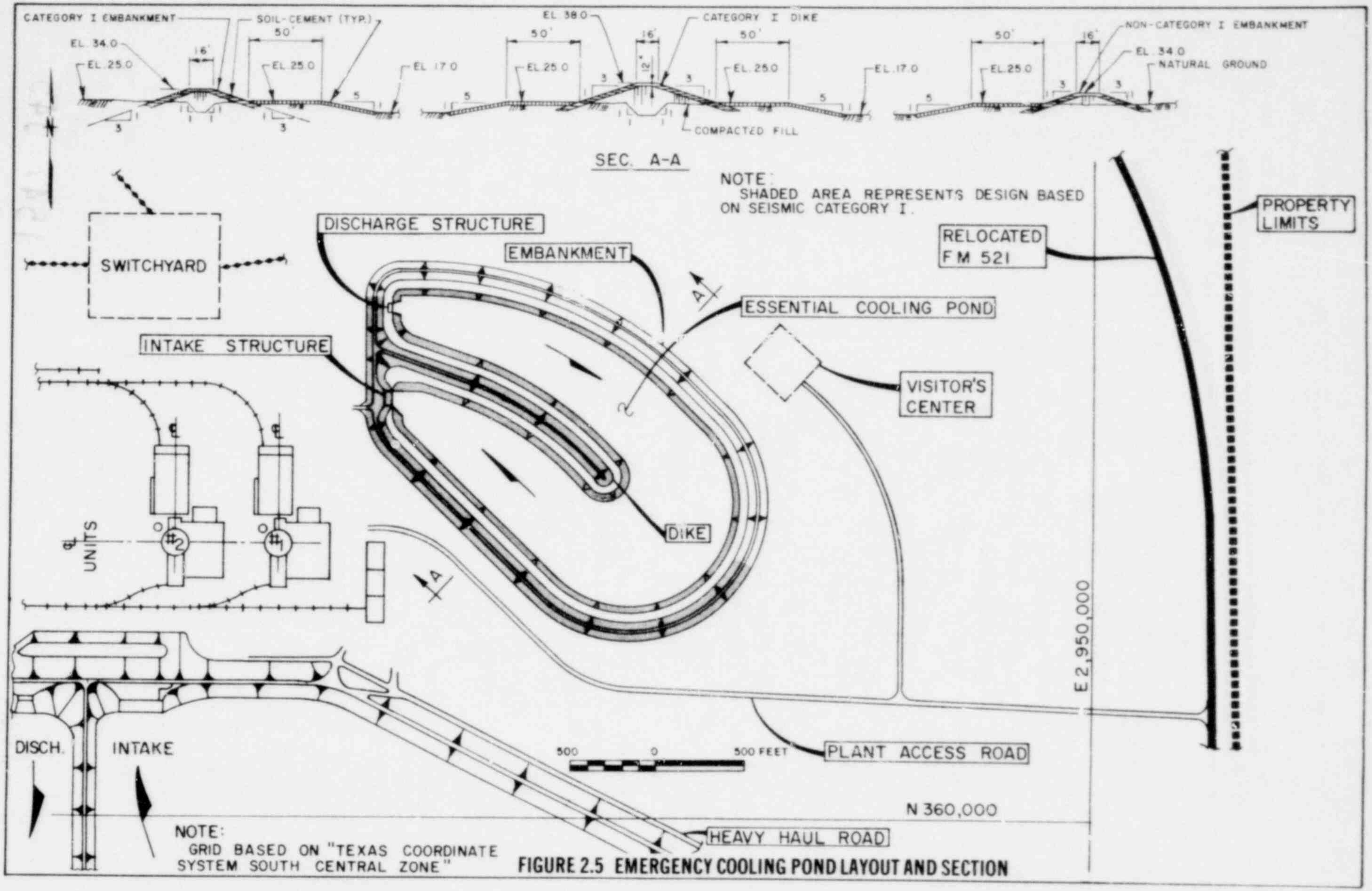


FIGURE 2.5 EMERGENCY COOLING POND LAYOUT AND SECTION

1547 044

failures of the cooling reservoir embankment, utilizing a two-dimensional finite difference computer code. In the computer analysis, sections of the embankment facing the reactor buildings and the emergency cooling pond were assumed to arbitrarily and instantaneously fail. In the analysis, a 2000-foot long embankment is assumed to fail. This failure is estimated to result in a water level between the proposed Units 1 and 2 of 50.6 feet above mean sea level. A 2000-foot removal is estimated to result in a maximum runup on the south face of the reactor buildings of 50.2 feet above mean sea level.

The postulated removal of a 2000-foot section of the embankment facing the emergency cooling pond is estimated to result in a water level at the safety related intake structure of 40.8 feet above mean sea level. Hydrostatic and hydrodynamic loadings were evaluated using the velocities and water levels calculated in these simulations. The facility will be designed to withstand the effects of the above-mentioned estimated extreme loads and water levels. Because of the intricate nature of these flood analyses, there is no one design water level applicable at every safety related structure. Rather, the individual worst water levels from the above-mentioned mechanisms apply to different points at the proposed site.

We consider the failure of the main cooling reservoir embankment to be remote (see Section 2.5.5 of this report). The maximum length of an embankment failure was estimated by the applicants to be 400 feet. We agree with this estimate. Water levels calculated from assuming a 400-foot embankment failure are well below the design levels calculated from the 2000-foot embankment failure previously mentioned.

We have evaluated the analytic procedures and models used by the applicants. We concur with the applicants that the design basis water levels determined from the instantaneous failure of the 2000-foot section of reservoir embankment results in acceptable design basis flood elevations at the proposed site.

2.4.3 Low Water Considerations

There are no safety related features which would be affected by low water in the Colorado River. Normal cooling for the plant will be provided by the main cooling reservoir which is capable of being used for several months without any makeup or blowdown.

Water for emergency shutdown will be provided by a separate emergency cooling pond. This pond will be used to supply the service water systems during normal operation. We will require the applicants to verify their estimate of the minimum water level for safe operation at the operating license stage of review (see Section 2.5.5 of this report).

The applicants have shown that the emergency cooling pond will contain enough water and heat dissipation capability to allow (1) both units to shut down under normal conditions, or (2) emergency shutdown of one unit with simultaneous safe shutdown of the other unit, and maintain them in a safe shutdown condition for 30 days without

1547 045

the need for any makeup water. This conforms with the recommendations of Regulatory Guide 1.27. Design basis temperatures were determined by the applicants to be about 116 degrees Fahrenheit.

We have evaluated the applicants' analyses and performed independent analyses. Based on these considerations, we have concluded that the proposed design bases for the emergency cooling pond are acceptable in regard to low water considerations.

2.4.4 Groundwater

Groundwater aquifers in the proposed site vicinity are included in the thick and widespread composite of deltaic sediments of the lower Gulf Coastal Plain. These deposits extend to depths of as much as 2,600 feet or more in this area and have been designated as the Gulf Coast Aquifer. They are composed of discontinuous interfingering beds of clay, silt, sand and gravel.

The groundwater consists of a shallow, low quality aquifer which occurs above depths ranging from 90 to 150 feet in the site area and a deep, high quality aquifer which lies below depths of 200 to 300 feet in the site area. These two aquifers are separated by a thick aquiclude which is composed predominately of clay materials, usually 150 feet thick. The Beaumont formation supplies most of the usable groundwater in the site vicinity.

Groundwater usage in the area is almost totally from the deep, high quality aquifer and is estimated to be an average of 1900 gallons per minute within ten miles of the proposed site. The applicants estimate that the facility will withdraw a peak of 560 gallons per minute from this zone with an expected average of 130 gallons per minute from a well about 4000 feet from the facility buildings. This water will be used for the fire protection system, demineralizers, and potable water systems.

The deep aquifer is confined by an impermeable clay aquiclude at least 150 feet thick at the proposed site area. There are no nearby recharge areas for this aquifer zone. Therefore, we have concluded that there is virtually no potential for contamination of the deep aquifer from a normal or accidental release of radioactivity at the site.

Because of its high dissolved solids, only minor amounts of water from the shallow aquifer zone are used. Its primary use is for domestic and stock watering purposes. Usage from this zone is estimated to be 130 acre feet per year within a ten mile radius of the proposed site. Facility dewatering during construction will draw only on the shallow aquifer zone. The gradient of the shallow aquifer is to the south. Groundwater flow is to the south from the facility site in the direction of the Colorado River and Matagorda Bay.

Since the radioactive waste treatment system storage tanks (the maximum concentration of their contents will be limited by the technical specifications) will be housed in seismic Category I structures, accidental releases of radioactive waste from this system will be very unlikely. However, if an accidental liquid spill from the radioactive waste treatment system were to occur in or around the proposed facility, conditions are unlikely for transport of contamination in the shallow aquifer. This is because of the low permeability (1.5×10^{-7} centimeters per second) of the soil.

The applicants have estimated that it would take more than 2000 years for contamination from the postulated rupture of the boron recycle tanks to reach the Colorado River. We estimate a somewhat shorter time of about 1100 years, and further dilution by the groundwater by a factor of 7300. The applicants' and our analyses were conservative since they assume total instantaneous release of the contents of the tanks and neglect decay and ion exchange.

We also performed a similar analysis that indicates that the travel time to the nearest domestic or stock watering well from a postulated spill is about 2500 years with an estimated dilution by the groundwater by a factor of 11,000. There are no wells for potable water which could reasonably be affected by contamination from the facility. Accidental releases to the groundwater would not be expected to contaminate the cooling reservoir since it will be built above plant grade. The applicants have conservatively analyzed the potential for contamination of the cooling reservoir by normal releases through the circulating water system. The contaminated water would reach the closest domestic well in 185 years, after being diluted in the large volume of the cooling reservoir, and further diluted by a factor of 100 by the groundwater. We determined a total groundwater and river water dilution factor of 1,400,000,000.

The radwaste tank that will contain the highest total quantity of activity is the waste holdup tank. This tank will have a volume of 10,000 gallons and will have a primary coolant activity concentration of approximately 6 microcuries per milliliter. Considering dilution and radioactive decay over the above transit times, a rupture of the waste discharge tank will give a concentration of less than 10^{-10} microcuries per milliliter at a potable water source. This value is a small fraction of the limits of 10 CFR Part 20, Appendix B, Table II, Column 2 for unrestricted areas.

The basis for acceptance in our review is that the postulated failure should not result in radionuclide concentrations in excess of 10 CFR Part 20 limits at the nearest potable water supply. Based on the foregoing evaluation we have concluded that the provisions incorporated in the applicants' design to mitigate the effects of component failures involving contaminated liquids are acceptable.

The design basis for subsurface hydrostatic loadings proposed by the applicants assumes that the design groundwater levels are at plant grade. The actual levels are estimated to range from two to 15 feet below plant grade. We consider the effects of the cooling reservoir on the design basis groundwater levels to be relatively small; however, the adopted design basis accounts for any likely increase.

2.4.5 Conclusions

On the basis of our review and evaluation of the hydrologic information presented by the applicants in Section 2.4 of the PSAR as amended, we have concluded that the analyses of the flood design bases for the proposed South Texas Project site are

1547 047

conservative, that an adequate safety related water supply will be available, and that the facility can be operated without impact on the regional ground and surface waters.

2.5 Geology and Seismology

2.5.1 Regional Geology

The proposed site is located within the Coastal Plain physiographic province on the northwest flank of the Gulf Coast geosyncline. Quaternary and older sediments, perhaps as much as 45,000 feet thick, underlie the site.

Structurally the site region is bounded on the north and west by the Ouachita Tectonic Belt and on the south and east by the arcuate axis of the Gulf Coast geosyncline. Subsidence of the Gulf Coast geosyncline began between Jurassic and late Cretaceous and was greatly accelerated during the Tertiary. True geosynclinal proportions were attained by the Oligocene.

The major near-surface structures in the West Gulf Coastal Plain are normal faults, some of which extend for many tens of miles. These have been called down-to-the-coast faults because the hanging wall is typically on the coastward side. Some of them have well-developed antithetic faults.

Faulting in the West Gulf Coastal Plain can be divided into two groups (1) older and (2) younger, based on the age of their formation. The older faults, located along the inner periphery of the province boundary (the Ouachita Tectonic Belt), form a belt or zone approximately 65 miles wide at the nearest approach (85 miles northwest) to the proposed South Texas Project site. The faulting is reportedly (Flawn, 1964) associated with the Ouachita belt. The older, peripheral faulting is, in the site region, comprised of four fault zones (1) the Balcones, (2) Luling, (3) Mexia-Talco, and (4) Charlotte-Jordanton. The Balcones is the most distant of these zones while the Charlotte-Jordanton, is the closest to the proposed facilities. Coastward of the older faults is a second, younger group having similar characteristics. They are predominantly faults with a history of low normal confining stress, termed growth faults. Such faults are found within the proposed site vicinity. They are of non-tectonic origin and are characterized by steep near surface dips, which become less steep with depth and eventually pass into bedding planes at great depth. Sediment accumulation has occurred simultaneously with fault movement, resulting in thicker strata on the downthrown (coastward) side.

Movement on most known growth faults ceased in the Tertiary. Subsequent deposition has resulted in their now being under high lithostatic stress and, therefore, they pose no threat of surface displacement. Some however have continued to move during deposition, and are undergoing active movement currently which cause offsets of the ground surface. These are most evident in the greater Houston area where reactivation of nearsurface faults is thought to be largely a consequence of extensive ground water withdrawal. Where active displacements are occurring at the surface, these faults can be easily delineated. They can be delineated in the subsurface by seismic reflection profiling and analysis of borehole geophysical logs.

Extensive ground water extraction, primarily as a result of demand for irrigation, is taking place in Matagorda County. Numerous growth faults have been defined in this area as a result of hydrocarbon exploration. Although considerable subsidence due to ground water extraction is reported in the Bay City - Francitas area at least 12 miles north of the site, a comprehensive ground-truth survey by the applicants indicated no evidence of reactivation of growth faults in the form of scarps, unexplained pavement failure or other phenomena either in the area of most pronounced subsidence (up to 1.7 feet at Francitas between 1951 and 1973) or at the proposed site and near vicinity. Hydrocarbon exploitation, in the form of oil and gas, is taking place at numerous fields (61) within a 15 mile radius of the site. No reactivation of growth faults as a result of these operations is known.

On the basis of our review, we have concluded that +here are no regional geologic hazards such as capable surface faulting, faulting eactivated by fluid extraction (ground water or hydrocarbon) or other phenomena presenting a potential hazard to the proposed South Texas Project.

2.5.2 Site Geology

The proposed South Texas Project site is located immediately adjacent to the Colorado River in Matagorda County approximately 12 miles southwest of Bay City, Texas, within the essentially featureless West Gulf Coastal Plain section of the Coastal Plain physiographic province (Fenneman, 1969). The Gulf of Mexico is nearly 15 miles southeast of the proposed facilities.

A Pleistocene deltaic sequence of interbedded, lenticular clays, silty clays and sands and gravels with clay interbeds exceeding 2,600 feet in thickness underlies the critical structures area. The uppermost Pleistocene formation, the Beaumont, consists predominantly of stiff to hard clay with some silty clay layers as well as silty sand and some fine to medium sand. Total thickness of the Beaumont underlying the site is estimated at 1,400 feet. Based upon available information, additional Pleistocene formations, with an estimated thickness of at least 1,200 feet, consist, in descending order of the Montgomery, Bentley and Willis.

Subsurface investigations, specifically shallow and deep reflection surveys and induction electric logs, were utilized extensively by the applicants to define the geologic structure in the site area. Nine seismic reflection profiles were made within the site boundaries, four of them within the critical plant structure area. Two additional lines lie immediately to the west and the south of the site boundary. These data show the geologic structure beneath the site to consist of nearly horizontal to gently dipping layers.

Growth faults have been interpreted beneath the site area below the 6,200- foot depth. A seismic reflection survey across the plant area shows several high quality reflectors which have no apparent offset above a depth of about 6,200 feet, thus indicating lack of definable fault movement since Oligocene time. Any faults which may be interpreted to exist below the 6,200 foot depth would present no safety hazard to the site. The lithostatic stress is too large at that depth of burial to permit renewed movement of these faults by any of the postulated mechanisms for reactivation.

One of the growth faults in the site area projects to within about 1,000 feet of the surface more than two miles north of the plant site. An antithetic fault which intersects it at a depth of approximately 11,000 feet and terminates at a depth of more than 6,200 feet would project surfaceward into the site area. The applicants utilized state-of-the-art seismic reflection methods to determine that this fault does not project above a depth of 6,200 feet. Several high quality reflection horizons were identified at and above this depth in the geologic section. These horizons are continuous with no apparent offsets indicative of fault displacement. Based on this we have concluded that faults identified in the subsurface below the site have a maximum upward extent that is approximately 6,200 feet below the ground surface immediately beneath the site. In response to our request, the applicants projected the plane of this fault to the ground surface. Assuming a conservative range of dips for the fault plane, the projection showed the surface intersection would be several hundred feet south of the nearest seismic Category I structure (Emergency Cooling Pond).

General subsidence has occurred and is continuing to occur in Matagorda County. This subsidence, based upon leveling surveys (1951-1973) conducted through Bay City (12 miles northeast of the site), northern Matagorda County and portions of adjacent Brazoria and Jackson counties, is attributed to concentrated ground water usage. Subsidence in the site area, extrapolated from the 1951-1973 leveling surveys is estimated to be approximately 1.3 feet, based upon a piezometric level decline beneath the plant site of 34 feet. The applicants estimate the aquifer underlying the structures area may be subjected to an additional piezometric level decline of 87 feet, resulting in a maximum subsidence of three feet. Uniform stratigraphic conditions, coupled with the lack of discernible structure and the apparent vertical and horizontal continuity of permeable layers within the underlying utilized aquifer, rule out the possibility of significant differential movement due to ground water extraction. We have concluded that the investigations performed by the applicants demonstrate that there is little likelihood of intolerable differential subsidence occurring at the site.

Considering the uncertainties associated with projections of future ground water withdrawals, attendant gradients and directional variability of the gradients as well as the effect of onsite usage of ground water upon subsidence, we require the applicants to establish a monitoring system, capable of detecting both vertical and horizontal movements. These are necessary to document subsidence and any tensional strain caused by subsidence during the life of the facility. The applicants have committed to monitor both vertical and horizontal movements. The complete monitoring program has been recently submitted for our review. This program consists of a subsidence monitoring program and a heave and settlement monitoring program and includes the instrumentation description, sequence of installation and layout of the horizontal and vertical movement detectors. As discussed in Section 2.5.5 of this report, we have reviewed and approved the heave and settlement monitoring program. We have not completed our review of the subsidence monitoring program. We will review the information pertaining to the subsidence monitoring program and report the results of our review in a supplement to the Safety Evaluation Report. We also require that this instrumentation be installed prior to pouring the slabs of any seismic Category I structure. The monitoring of both vertical and horizontal movements associated with subsidence due to fluid withdrawal is an effective means by which any harmful effects caused by this subsidence could be detected in sufficient time to allow remedial measures to be taken or to allow a safe shutdown of the proposed facilities (Hendron, 1975).

We have concluded that subsidence resulting from withdrawal of petroleum is unlikely within the site boundaries. Extensive subsurface investigations, both geophysical and test holes, conducted by the applicants and previously by numerous oil exploration firms within the site boundaries have not found conditions, either stratigraphic or structural, favorable for the accumulation of commercially-attractive hydrocarbon deposits. Nine exploratory wells, ranging from 4,056 to 16,154 feet in depth, have been drilled at the site. All were nonproductive. Additionally, numerous exploratory wells, all unsuccessful, have been drilled south, east and west of the site. At least nine seismic reflection lines have been run within the site boundaries.

The producing field nearest the site is the South Duncan Slough. The nearest well in this field is 1.7 miles to the northwest of the facility structures. It produces no oil and only a minor amount of natural gas. Production zones in this field are below a depth of 11,000 feet. The nearest oil production area is the Collegeport-North Collegeport-Citrus Grove field approximately seven miles south west of the plant structures. As of August 1973, less than 50,000 barrels of oil have been produced from this field. Based on the exploration history at the site, it would appear that hydrocarbon production within the site boundary is highly unlikely. The applicants have obtained all the mineral rights within the approximately 1800 acre exclusion area.

Petroleum production at great depth (in excess of 11,000 feet) cannot be totally discounted in the immediate site area. Due to the extreme depth, apparent limited size of the potentially producing field, reduced permeability and porosity of any potential hydrocarbon yielding strata, we do not consider subsidence due to fluid or gas extraction from these depths to constitute a potential subsidence hazard at the proposed South Texas Project site. In the unlikely event oil were to be extracted in the site vicinity, the subsidence instrumentation system to be installed by the applicants will be capable of detecting any small movements that may result.

The Texas Bureau of Economic Geology has conducted extensive studies of photolinears in the Texas Coastal Plain. The linears extend over tens of miles and several lines of evidence in the Houston area indicate that some of the linears are related to geologic structure. The linears are usually not uniformly identifiable along their entire extent and in some segments are visible as zones up to a few thousand feet

1547 051

wide rather than narrow lines. Because linears have been observed to be co-extensive over portions of their extent with faulting, we emphasized in our review the determination of their safety implications for the South Texas Project. Five northeast-trending linears identified previously by the Texas Bureau of Economic Geology pass through the site boundaries to the north and south of the seismic Category I structures area. Two linears of limited extent have been identified passing through the plant structures area. One trends northeast while the other has a northwest trend.

These two linears, as well as many others, both within and beyond the site boundaries have been thoroughly investigated by the applicants. The surface investigations included detailed airphoto studies (conventional as well as remote sensing photos), studies of soil and surface geologic maps, comparison of the linears with the surface intersection of deep faults projected to the surface, comparison of the photolinears with geomorphic features and trenching coupled with geologic mapping both at the site and along the nearby Colorado River. Subsurface investigations included borings, oil and gas well logs and geophysical techniques.

These studies revealed no evidence of structural control of the linears in the plant site area. The two linears within the plant structures area have no structural control and are related to soil-vegetation changes and old as well as recent cultural features. Subsurface investigations conducted by the applicants and obtained and compiled from others confirms that the linears passing through the plant structures area are not related to the faulting at great depths beneath the site. Only a short segment of the surface projection of the northwest-dipping antithetic fault which terminates below the 6,200-foot depth coincides with the northeast-trending linear in the plant structure area. The linear continues in the northeast direction while the fault trend changes abruptly to the south and southwest. Moreover, subsurface investigations indicate no continuation of the fault plane above the 6,200-foot level.

The cause of linears is not known. While some have structural control elsewhere in the Texas Gulf Plain over part of their extent, state-of-the-art investigations have been utilized by the applicants showing no relationship between discernible subsurface geologic structure and photolinears at the proposed South Texas site.

The applicants have agreed to geologically map and photograph in detail geologic features exposed in the excavations for all seismic Category I structures. In addition, similar mapping will be conducted within other excavations related to non-safety structures, if required, to adequately interpret the site geology. We shall visit the site during, or shortly after, mapping of the excavations is complete in order to confirm the site geologic conditions.

Subject to the consideration identified above, we have concluded that there are no geologic structures or other hazards at the site and near vicinity that represent a threat to the facility or that would localize an earthquake in the proposed area. However, we have further concluded that our knowledge is not absolute relative to certain geological and hydrologic phenomena, such as the projected amount of piezometric decline across the site, the gradient associated with this withdrawal, tensional strains resulting from subsidence, abrupt lateral soil facies changes, and the significance and origin of photolinears. To accommodate these uncertainties and to provide conservative margins of safety, the applicants have committed to monitor both horizontal and vertical movements associated with subsidence throughout the life of the proposed facility. To acquire sufficient background information, the applicants have agreed to initiate the subsidence monitoring program prior to placement of the foundation slab for any safety related structure.

2.5.3 Vibratory Ground Motion

King (1969) in his discussion of the tectonic map of North America defines the Atlantic and Gulf Coastal Plains as platform deposits (Mesozoic and younger) that were laid over the deformed Paleozoic and older rocks of the Appalachian and Ouachita fold-belts. The platform deposits thicken and slope seaward from the exposed parts of these fold-belts, the basement descending beneath them. From New Jersey to the Llano uplift in central Texas, the landward border of the platform deposits on Paleozoic basement is drawn at the edge of the Cretaceous and/or Tertiary deposits of the coastal plains, where they overlap on older rocks. These limits define the Coastal Plain. The Gulf Coastal Plain Tectonic Province, in which the proposed South Texas Project site is located, is that part of the Coastal Plain extending from west Florida westward and southward into Mexico (Eardley, 1962).

For the purposes of establishing the safe shutdown earthquake for nuclear power plants, we recognize that different regions of this large province exhibit vastly different levels of seismicity. In particular, to arrive at the appropriate choice of the safe shutdown earthquake for the proposed South Texas Project site, we recognized four seismic zones: (1) the Mississippi Embayment Earthquake Zone, (2) the Southern Cordilleran Front Zone, (3) the zone at the intersection of the Ouachita Tectonic Belt and the Wichita Structural System, and (4) a Gulf Coast Seismic Zone.

The Mississippi Embayment Earthquake Zone is a region of much higher seismic activity than the remainder of the eastern United States. It has also been the source region of the largest earthquakes in the eastern United States, the 1811-1812 New Madrid earthquakes. The closest approach to the site of the Mississippi Embayment Earthquake Zone was established at the Monroe Uplift during our review of the Grand Gulf Nuclear Station, Units 1 and 2 site (Docket Nos. 50-416 and 50-417) and the closest approach of earthquakes similar to the 1811-1812 series is considered to be near Memphis, Tennessee, over 500 miles from the proposed site.

The Southern Cordilleran Front consists of a belt of Laramide folds and thrust faults extending southward from New Mexico and Texas into central and eastern Mexico (King, 1969; Eardley 1962). Several earthquake epicenters are located along this zone including the Valentine, Texas, earthquake of 1931 which had an epicentral intensity of Modified Mercalli VIII. The epicentral intensity of the largest reported historical earthquake in this zone would be less than Modified Mercalli X. This zone apparently interrupts the Gulf Coastal Plain and has its closest approach to the proposed South Texas Project approximately 300 miles southeast of the site.

Within the remainder of the Gulf Coastal Plains (the region between west Florida and where the Gulf Coastal Plain is narrowed and partly interrupted by the outer folds of the Cordillera in Mexico) there is very little seismic activity. Few small earthquakes, none larger than Modified Mercalli VII, have been recorded.

One of the two Modified Mercalli VII earthquakes that have occurred in the general area of interest, is the 1882 earthquake located near Paris, Texas. This earthquake was recently relocated by Docekal (1970) based on a reevaluation of its effects and characteristics.

The region of maximum intensity is located at the intersection of the Ouachita Tectonic Belt and the Wichita Structural System. This is a complex region where various complex tectonic forces have acted (King, 1969). The Ouachita Tectonic Belt is recognized in this area as a region of intense folding and thrust faulting which developed principally in Pennsylvanian time. The Wichita Structural System includes a number of block uplifts and fault-bounded basins and strikes northwest-southwest in southern Oklahoma and north-central Texas. The segmentation of the Wichita system into the various crustal blocks of its present configuration came during several stages of Pennsylvanian orogeny. Further adding to the tectonic complexity of this area is the Nemaha Uplift, a nearly north-south structure of sharply uplifted and faulted Precambrian basement material which also formed during the Pennsylvanian orogeny (Eardley, 1962). The Nemaha Uplift apparently trends into the Wichita System and terminates in the vicinity of the Arbuckle Mountains. These three tectonic units are penecontemporaneous, and apparently interfere structurally in the area of their intersection.

Numerous earthquake epicenters, none larger than epicentral intensity Modified Mercalli VIII, coincide with each of these three tectonic units. Therefore, we consider the 1882 earthquake to be located in a tectonic province separate from the remainder of the Gulf Coastal Plain as suggested by Docekal (1970) who relates the earthquake to the buried structures associated with the Arbuckle Mountains. The closest approach of these structures to the proposed site is about 300 miles.

The seismicity of the Gulf Coastal Plain except for the area already noted is relatively uniform. In terms of historical earthquakes, this portion of the Gulf Coastal Plain is one of the least active areas of the United States. This area includes the source region of the 1891 Rusk, Texas, earthquake which had a reported epicentral intensity of Modified Mercalli VII. This earthquake was felt over an extremely small area indicating that it was a very small, shallow earthquake. Indeed, the "felt area" suggests that it was of magnitude much smaller than even a typical Modified Mercalli VI earthquake. This assessment is further discussed in detail below. Other historical earthquakes in this zone have produced intensities no greater than Modified Mercalli VI.

Active surface faults are recognized in the Gulf Coast. In the Gulf of Mexico, active slump faulting or growth faulting is also occurring. As discussed in Sections

2.5.1 and 2.5.4 of this report, there are various models proposed for the mechanisms of this faulting. However, in view of the low level of seismicity for the region, we have concluded that the typical movement on these faults is a fault creep process and does not release significant seismic energy in the form of earthquakes. We have, therefore, not considered such faults to be capable of generating significant earthquakes.

As discussed previously, we recognize four seismic zones within the large Gulf Coastal Plains Tectonic Province: the Mississippi Embayment Earthquake Zone, the zone in which the 1882 Paris, Texas, earthquake occurred, the Southern Cordilleran Front Zone, and the remainder of the Gulf Coastal Plain (the Gulf Coast Seismic Zone) which includes the South Texas Project site. The former three zones are so remote from the South Texas Project site, that the resulting intensity at the site from the largest historical earthquakes located in these zones is less than would occur at the site from a random earthquake located in the Gulf Coast Seismic Zone, assuming a conservative relation between intensity and epicentral distance (Nuttli, 1973; Gupta and Nuttli, 1975).

The largest historical earthquakes which have occurred in the Gulf Coast Seismic Zone were at Rusk, Texas, in 1891, at Donaldsonville, Louisiana, in 1930, and near Mexia and Wortham, Texas, in 1932. The damage reports from all three of these events were very sparse. The "felt areas" were 18,500 square miles for the Donaldsonville earthquake, 1000 square miles for the Mexia-Wortham earthquake, and the Rusk earthquake was felt only in the town of Rusk. A relation developed by Nuttli, et al. (1974) between earthquake magnitude and "felt area" would indicate that the Donaldsonville earthquake had a "felt area" no greater than that for a typical intensity Modified Mercalli V-VI. Based on this analysis the Mexia-Wortham and the Rusk earthquakes were much smaller than a typical Modified Mercalli VI earthquake; i.e., the energy released by these events was less than that released by an average event of intensity Modified Mercalli VI. The reported intensities for these events were Modified Mercalli VII for the Rusk earthquake, intensity Modified Mercalli VI for the Donaldsonville earthquake, and intensity Modified Mercalli V-VII for the Mexia-Wortham earthquake. These data support the observation that small magnitude earthquakes may cause higher intensities over a limited area than would be indicated by their energy because of very shallow focal depths. In the past we have utilized "far field" assumptions and directly related intensity to the seismic energy release in assessing the safe shutdown earthquake. In this context, a typical intensity Modified Mercalli VI earthquake is an appropriate safe shutdown earthquake for the South Texas Project site.

Consistent with our past practices, we have assumed that the acceleration level associated with this intensity corresponds to the mean value obtained from intensity-peak acceleration relations. Using the relation developed by Trifunac, et al. (1975) or Neumann (1954), this acceleration is 0.07 g. Based on this analysis we concur with the applicants that 0.1 g is an appropriate acceleration level to be used as the high frequency input to the spectra recommended in Regulatory Guide 1.60. This is to be applied at foundation level in the free field.

2.5.4 Surface Faulting

Non-capable faulting underlies the site at great depth. We have concluded that the potential for surface faulting in the site area is remote. Therefore, surface faulting is not considered a potential problem at the proposed South Texas Project site.

2.5.5 Foundation Engineering

The site soils are composed of discontinuous lenses of silts, sands, and clays of varying consistency, typical of low-lying deltaic deposits. Some thirteen significant strata of importance to foundations (within the uppermost 300 feet of sediments) have been identified during the site exploration program.

The seismic Category I emergency cooling pond will have a water level of 25 feet above mean sea level, and will be formed by excavating an eight-foot deep hole to elevation 17 feet above mean sea level in the alluvium. A 13-foot high dividing dike (crest elevation 38 feet above mean sea level) will bisect the pond and a nine-foot high dike surrounding the emergency cooling pond (crest elevation 34 feet above mean sea level) will retain wave runup within the pond. To assure that an adequate supply of water will be available for emergency conditions, we will require periodic monitoring of leakage from the emergency cooling pond.

The excavation slopes for the emergency cooling pond are five horizontal to one vertical. The interior slopes of the dike surrounding the emergency cooling pond will be 2-1/2 horizontal to one vertical, and the dike exterior slopes will be three horizontal to one vertical. The dividing dike slopes will be 2-1/2 horizontal to one vertical. Compaction of these dikes to at least 95 percent of Standard Proctor will assure that the assumed embankment strength values are attained. Dike slope stability analyses indicate a static safety factor of 1.6 and a pseudostatic (seismic) safety factor of 1.1.

As discussed in Section 2.4.1 of this report, the earthen and soil embankments (dikes) surrounding the main cooling reservoirs will not be seismic Category I earthworks. However, the limited failure of these embankments discussed in Section 2.4.2 of this report under seismic loadings is predicated on acceptable engineering practices and design features. Such practices and features include embankment compaction to at least 95 percent of Standard Proctor densities and a system of pressure relief wells along the outer toe of these embankments. Borrow material suitable for the construction of the extensive dike system is generally plentiful in the plant area.

Foundation soils supporting seismic Category I dikes and the main cooling reservoir dikes are composed of interbedded silts, sands, and slickensided clays. These clays are rather brittle and peak strengths are reached at low strains and residual strengths are much less than peaks. The effective strength of the interbedded silts and sand lenses may be somewhat reduced immediately following a seismic disturbance. The applicants have investigated classic liquefaction of the site soils, and have determined that such conditions are unlikely to be developed during the postulated

1547 056

earthquake. They have assessed the post-earthquake stability of safety related earthwork associated with the emergency cooling pond and have found no change in static safety factors due to seismic effects. We have estimated the post-earthquake stability of these dikes in a different manner, and have found static factors of safety exceeding 1.3. It was necessary to evaluate the post-earthquake stability of these dikes because liquefaction analyses do not properly assess post-earthquake soil strengths. Thus, we find that, in the event of an earthquake, there would be no hazard to the circulation of essential water within the emergency cooling pond and no possibility of an earth slide blocking the intake structure for the emergency cooling pond.

The containment structures will be founded on a 10-foot thick silty sand lens about sixty feet below plant grade. Because the near surface soils near the containment structure are unsuitable for founding other plant structures, these soils will be excavated to minus ten to minus fifteen feet, below mean sea level, and replaced with structural backfill to support seismic Category I structures which will be located at higher elevations. The essential cooling water intake structure and essential cooling water discharge structures, located at the emergency cooling pond, will be founded on stiff clay, approximately 15 feet and eight feet below plant grade.

Select material for filters, drainage blankets, and structural backfill are critical to the safe support and stability of site facilities. Specifications and construction control of these materials must receive appropriate attention. The applicants have proposed the use of screenings from a rock crushing plant, a well graded medium to coarse sand, compacted to at least 95 percent of Modified Proctor or to at least 80 percent relative density, whichever method yields the greatest dry unit weight. On this basis, we have concluded that these backfill criteria are adequate for the support of safety related structures and facilities.

Because of the compressible clays underlying the proposed site, perhaps the most difficult aspect of foundations for the South Texas Project involves the long term settlement of facility structures. These settlements vary from building to building, depending on load and foundation design, but are expected to range to about one-half foot. Unless appropriate design measures are made for connecting piping and conduits, severe distress to these components may be induced as long term settlement, differential settlement, and tilting of structures occurs. Therefore, we advised the applicants that we will require that design criteria for such movement be established. The applicants have recently provided the design criteria for our review. We will report the results of our review in a supplement to the Safety Evaluation Report.

The applicants have committed to the heave and settlement portion of the monitoring program and will install appropriate devices, prior to the excavation, which will aid in interpreting total and differential settlement of buildings within the facility complex. We have reviewed this portion of the monitoring program and find it acceptable. As discussed previously, we have not completed our review of the subsidence portion of the monitoring program.

We have concluded that because (1) the containment buildings will be founded at depth on competent soil, (2) the essential water intake and discharge structures will be founded on stiff clay, and (3) other seismic Category I structures will be founded on competent structural backfill, which in turn will rest on a seismically stable hard silt strata at an elevation of about minus ten to minus fifteen feet, below mean sea level, there will be inconsequential degradation of foundation support capability caused by soil-structure interaction effects during the postulated safe shutdown earthquake. Buried pipes will be designed to withstand the soil strain caused by this earthquake.

Subject to the establishment of acceptable design criteria regarding long term settlement, differential settlement and tilting of structures, we have concluded that the soil and foundation conditions at the South Texas Project site are acceptable for the proposed facility structures. We have also concluded that the proposed preliminary foundation and earthwork designs are acceptable.

With the provision that conservative designs, responsible construction planning and control measures, and planned confirmatory measurements of foundation and emergency cooling pond performance are carried out, we have concluded that foundations associated with the South Texas Project are acceptable.

3.0 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS AND COMPONENTS

3.1 Conformance with General Design Criteria

The applicants have stated that the South Texas Project Units 1 and 2 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 have concluded 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 RESAR-41 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 system is presented in Section 3.2.2 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 shutdown 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 facility will be 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. 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 Table 3.2-1 of the PSAR.

The basis for our acceptance has been conformance of the applicants' 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, staff positions, and industry standards.

We have concluded 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 guide, staff positions, and industry standards and are considered acceptable.

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 system 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.

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 applicants have applied the classification system identified in Regulatory Guide 1.26 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 the safe shutdown conditions, and (3) to contain radioactive material. These fluid systems have been classified in an acceptable manner in Tables 3.2-1 and 3.2-3 of the PSAR and on system piping and instrumentation diagrams in the PSAR.

The basis for our acceptance has been conformance of the applicants' design, 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 Criterion 1 of the General Design Criteria, the requirements of the Codes specified in Section 50.55a of 10 CFR Part 50, and to Regulatory Guide 1.26, staff positions, and industry standards.

We have concluded that fluid system pressure-retaining components important to safety, which are within the scope of the balance-of-plant that are designed, fabricated, erected, and tested to quality standards in conformance with the Commission's regulations, the applicable Regulatory Guide, staff positions and industry standards are acceptable. Conformance with these requirements provides reasonable assurance that the facility will perform in a manner providing adequate safeguards to the health and safety of the public.

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 therefore be protected from the effects of the design wind. The design wind specified for the South Texas Project site has a velocity of 120 miles per hour and is based on a recurrence interval of 100 years.

The procedures that will be used to transform the wind velocity into pressure loadings on structures, and the associated vertical distribution of wind pressures and gust factors, will be 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 such as the Comanche Peak Steam Electric Station Units 1 and 2 (Docket Nos. 50-445 and 50-446). 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 tornado. The design basis tornado is in accordance with the recommendations of Regulatory Guide 1.76 which specifies a tangential wind velocity of 290 miles per hour and a translational velocity of 70 miles per hour. The pressure drop associated with the tornado is three pounds per square inch in 1.5 seconds.

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

All the facility structures that are not to be 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 and systems. The safety function and structural integrity of seismic Category I structures will thereby be assured.

We have concluded that the procedures to be utilized to determine the loadings on seismic Category I structures induced by the design wind and the design basis tornado specified for the facility 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 occurrence of the design wind or the design basis tornado, the structural integrity of seismic Category I structures will not be impaired and, therefore, seismic Category I systems and components located within these structures will be adequately protected and will perform their intended safety functions if needed. Conformance with these procedures is an acceptable basis for satisfying the applicable requirements of Criterion 2 of the General Design Criteria.

1547 061

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 is 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.

All safety related components will be located inside buildings except the condensate storage tank which will be designed to withstand the maximum flood levels and associated effects. All exterior building openings which can communicate with safety related components will be located above the calculated maximum surge level or be provided with waterproof double doors.

The exterior walls of buildings containing safety related components will be waterproofed to grade level for protection from ground water. All construction joints in exterior walls and slabs will be provided with water stops above the maximum surge level of the design basis flood. Major tanks containing liquids in the auxiliary building will be housed in watertight compartments which will retain the contents of the tank. The plant drain system is discussed in Section 9.3.3 of this report.

We have concluded that the procedures utilized to determine the loadings on seismic Category I structures induced by the design basis flood or highest groundwater level specified for the South Texas Project 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 floods or high groundwater, the structural integrity of seismic Category I structures will not be impaired and, therefore, seismic Category I systems and components located within these structures will, thereby, be adequately protected and will perform their intended safety functions, as required. Conformance with these design procedures is an acceptable basis for satisfying the applicable requirements of Criterion 2 of the General Design Criteria.

3.5 Missile Protection

3.5.1 Missile Protection Criteria

The South Texas Project Units 1 and 2 will be designed so that missiles from internal sources and from outside of containment do not cause or increase the severity of an accident.

The applicants have considered missiles generated by pressurized components and rotating components which have the potential of being subjected to an overspeed in excess of design limitations. To protect the essential systems and components from the damaging effects of these missiles, compartmentation, restraints, separation,

1547 062

orientation and/or missile barriers will be provided. The criteria used to design structures for missile impact are described in Section 3.5 of the PSAR. The commitments of the applicants regarding the protection of essential structures and vital equipment from internally generated missiles are in accordance with the applicable requirements of Criterion 4 of the General Design Criteria and the recommendations of Regulatory Guides 1.13 and 1.27. Based on the above, we have concluded that the applicants can develop an acceptable facility design to prevent missiles from damaging structures and equipment required for the safe shutdown of the facility.

We have reviewed the information supplied in the PSAR concerning analysis of the tornado missile velocities and trajectories. In Amendment 17 of the PSAR, the applicants have stated that the tornado missile velocities shown in Table 3.5-4 were calculated using methods described in Brown & Root, Inc. Topical Report 001 "An Analysis of Tornado Generated Missiles." The applicants have recently submitted this topical report for our review. However, we have determined that the tornado missile velocities presented in Table 3.5-4 of the PSAR are considerably lower than those estimated by other methods which we have reviewed and found acceptable in other recently approved applications and, therefore, are unacceptable.

On the basis of our previous evaluations we required that the South Texas Project be designed to withstand the impacts of missiles and impact velocities listed in Table 3.1 (the missile spectrum is discussed in WASH-1361 "Safety Related Site Parameters for Nuclear Power Plants") or in Table 3.2.

The applicants have orally agreed to design the South Texas Project to withstand the impacts of missiles and impact velocities described in Table 3.1. Subject to confirmatory documentation, we have concluded that the proposed design is acceptable.

3.5.2 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, 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 Standard 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 such as the Comanche Peak Steam Electric Station Units 1 and 2 (Docket Nos. 50-445 and 50-446). Furthermore, secondary missiles will be prevented by fixing the target thickness well above that determined for penetration. 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.

TABLE 3.1

TORNADO MISSILE SPECTRUM

Missile*	Dimensions	Weight	Velocity
A - Wood plank	4 inches x 12 inches x 12 feet	200 pounds	420 feet per second
B - Steel pipe	3 inch diameter, 10 feet long, schedule 40	78 pounds	210 feet per second
C - Steel rod	1 inch diameter x 3 feet long	8 pounds	310 feet per second
D - Steel pipe	6 inch diameter, 15 feet long, schedule 40	285 pounds	210 feet per second
E - Steel pipe	12 inch diameter, 15 feet long, schedule 40	743 pounds	210 feet per second
F - Utility pole	13.5 inch diameter x 35 feet long	1490 pounds	210 feet per second
G - Automobile	20 square feet frontal area	4000 pounds	100 feet per second

* Missiles A through E are to be considered at all altitudes, and missiles F and G at altitudes up to 30 feet above the highest terrain within 1/2 mile of the safety related structures. Protection of both vertical and horizontal surfaces is required.

1547 064

TABLE 3.2

ALTERNATE TORNADO MISSILE SPECTRUM

Missile*	Dimensions	Weight	Horizontal Velocity	Vertical Velocity
A - Wood plank	4 inches x 12 inches x 12 feet	200 pounds	368 feet per second	294 feet per second
B - Steel pipe	3 inch diameter, 15 feet long, schedule 40	115 pounds	268 feet per second	214 feet per second
C - Steel rod	1 inch diameter, 3 feet long	8 pounds	259 feet per second	207 feet per second
D - Steel pipe	6 inch diameter, 15 feet long, schedule 40	300 pounds	230 feet per second	184 feet per second
E - Steel pipe	12 inch diameter, 30 feet long, schedule 40	1500 pounds	205 feet per second	164 feet per second
F - Utility pole	14 inch diameter x 35 feet long	1500 pounds	241 feet per second	193 feet per second
G - Automobile	20 square feet frontal area	4000 pounds	100 feet per second	80 feet per second

* Missiles A through F are to be considered at all altitudes, and missile G at altitudes up to 30 feet above the highest terrain within 1/2 mile of the safety related structures.

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 have concluded 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 provide a conservative basis for engineering design to assure that the structures or barriers are adequately resistant to the effects of missile impacts.

3.5.3 Turbine Missiles

The South Texas Project Units 1 and 2 turbine generators will be arranged in a peninsular orientation in relation to their respective containment buildings. We have concluded that the turbine orientation is acceptable and that no additional provisions for protection against turbine missiles are necessary.

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

3.6.1 Protection Against Dynamic 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 3.4 of Appendix A to this report. In addition, the applicants have incorporated provisions in the design of the piping systems which are within the scope of balance-of-plant that are consistent with Regulatory Guide 1.46 for piping inside of containment.

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.

1547 066

Pipe motion subsequent to rupture and the pipe restraint dynamic interaction will be analyzed by the use of an elastic-plastic lumped mass beam element model sufficiently detailed to reflect the structural characteristics of the piping system.

The protection against dynamic effects associated with the postulated rupture of piping within the primary loop is provided in RESAR-41, which we have found acceptable contingent upon the requirements that each referencing plant demonstrate that the specific reactor coolant system component support designs be within the design envelope of Westinghouse Topical Report WCAP-8082, "Pipe Breaks for the LOCA Analysis of the Westinghouse Primary Coolant Loop," and that the interface responsibilities between the nuclear steam supply system designed portion of the primary loop and the portion provided by the architect/engineer be clearly identified. The applicants have identified these interface responsibilities for the South Texas Project and have provided adequate assurance that their component support designs do lie within the design envelope of WCAP-8082.

On the basis of our review, we have concluded 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 Protection Against Dynamic Effects Associated with the Postulated Rupture of High Energy Piping Outside Containment

The proposed South Texas Project 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 South Texas Project 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 the Commission's letter of July 12, 1973, "Protection Against Postulated Events and Accidents Outside Containment."

The applicants 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 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.

1547 067

The applicants will describe the protective features against dynamic effects associated with postulated pipe failures for individual systems, and provide preliminary piping layout drawings, and other pertinent information. The applicants have agreed to provide this information in late 1975.

Based on our review, we have concluded 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 is acceptable.

3.7 Seismic Design

Our evaluation of the seismic design of systems and components within the scope of the nuclear steam supply system is presented in Section 3.5 of Appendix A to this report. Therefore, 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 input design response spectra and damping values to be applied in the design of seismic Category I structures, systems and components, comply with the recommendations of Regulatory Guide 1.60 and Regulatory Guide 1.61. The synthetic time history to be used for the seismic design of Category I plant equipment is adjusted in amplitude and frequency to envelop the design response spectra specified for the proposed site.

Conformance with these requirements provides reasonable assurance that, for an earthquake of intensity 0.05g for the operating basis earthquake and 0.10g for the safe shutdown earthquake, the resulting accelerations and displacements to be imposed on seismic Category I structures, systems and components are adequately defined to assure a conservative basis for the design of such structures, systems and components to withstand the consequent seismic loadings.

We have concluded that the seismic input criteria proposed by the applicants are acceptable for seismic design.

3.7.2 Seismic System Analysis

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

The system and subsystem analyses will be performed by the applicants on an elastic basis. Modal response spectrum multi-degree-of-freedom and time history methods will form the basis for the analyses of all major seismic Category I structures, systems

1547 068

and components. When the modal response spectrum method is used, governing response parameters are combined by the square root of the sum of the squares rule. However, the absolute sum of the modal responses are 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 are considered.

The finite element approach will be used to evaluate soil-structure interaction and structure-to-structure interaction effects upon seismic responses. For the finite element analysis, appropriate nonlinear stress-strain and damping relationships for the soil will be considered in the analysis.

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

3.7.3 Seismic Instrumentation Program

The proposed 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 Regulatory Guide 1.12.

On this basis, we have concluded that the seismic instrumentation program proposed by the applicants is acceptable.

3.8 Design of Category I Structures

3.8.1 Concrete Containment

The reactor coolant system will be enclosed in a concrete containment as described in Section 3.8.1 of the PSAR. The containment structure will be designed in accordance with applicable subsections of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Division 2, to resist various combinations of dead loads, live loads, environmental loads including those due to wind, tornadoes, one-half the safe shutdown earthquake, safe shutdown earthquake, and loads generated by the design basis loss-of-coolant accident, including pressure, temperature and associated pipe rupture effects.

Analysis of the containment shell and the liner design for the containment will employ methods similar to those previously reviewed and accepted for previously licensed plants such as the Trojan Nuclear Plant (Docket No. 50-344). The choice of materials, the arrangement of anchors, the design criteria and design methods will be similar to those evaluated for previously licensed plants such as the Trojan Nuclear Plant (Docket No. 50-344). Materials, construction methods, and quality control measures are, in general, similar to those used for previously accepted facilities.

Prior to operation, the containment will be subjected to an acceptance structural test during which the internal pressure will be 1.15 times the containment design pressure in accordance with the provisions of Regulatory Guide 1.18.

The criteria used in the analysis, design, and construction of the concrete containment structure, to account for anticipated loadings and postulated conditions that may be imposed upon the structure during its service lifetime, are in conformance with established criteria, codes, standards, applicable Regulatory Guides, and specifications acceptable to the Commission's staff.

The use of these criteria as defined by applicable codes, standards, guides 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 inservice surveillance requirements provide reasonable assurance that, in the event of winds, tornadoes, earthquakes and various postulated accidents occurring within and outside the containment, the structure will withstand the specified design conditions without impairment of structural integrity or safety function. Conformance with these criteria, codes, specifications and standards constitutes an acceptable basis for satisfying the applicable requirements of Criteria 2, 4, 16, and 50 of the General Design Criteria.

3.8.2 Other Category I Structures

Our review of seismic Category I structures other than the containment structure included (1) the containment interior structures consisting of the shield wall around the reactor, the secondary shield walls and interior walls, compartments and floors, and (2) the mechanical and electrical auxiliaries building, fuel handling building, diesel generator building and the essential cooling water intake and discharge structures including their foundations.

The principal code that will be used in the design of concrete seismic Category I structures is the American Concrete Institute (ACI) 318-71 Code, "Building Code

Requirements for Reinforced Concrete." For seismic Category I steel structures, the American Institute of Steel Construction (AISC), "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," will be used.

The concrete and steel seismic Category I structures will be designed to resist various combinations, as applicable, of dead loads, live loads, environmental loads including those associated with extreme winds, tornadoes, and earthquakes, and pipe rupture induced loads including loads associated with reaction and jet impingement forces, compartment pressures, and impact effects of whipping pipes.

The proposed design and analysis procedures that will be used for these Category I structures are the same as those approved in previously licensed applications and are in accordance with the applicable 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 established by the staff under the various load combinations. These limits are, in general based on the ACI 318-71 Code and the AISC Specification for concrete and steel structures, respectively.

The materials of construction, their fabrication, construction and installation, will be in accordance with the ACI 318-71 Code and the AISC Specification for concrete and steel structures, respectively.

The criteria to be used in the analysis, design and construction of the plant 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 acceptable to the staff.

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 provides reasonable assurance that, in the event of winds, tornadoes, earthquakes and various postulated accidents occurring within the structures, these structures can be expected to withstand the specified design conditions without impairment of their structural integrity or their safety function.

We have concluded that conformance with these criteria, codes, specifications, and standards in designing seismic Category I structures other than the containment structure constitutes an acceptable basis for satisfying the requirements of Criteria 2 and 4 of the General Design Criteria.

3.9 Mechanical Systems and Components

3.9.1 Dynamic System 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 nuclear steam supply system is presented in Section 3.6 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 applicants will perform a preoperational vibration dynamics effects test program to check the performance of piping important to safety. The preoperational vibration dynamic effects test program that will be conducted on safety related ASME Class 1 and 2 piping systems and their restraints during startup and initial operating conditions constitutes an acceptable program.

This program will provide adequate assurance that the piping and piping restraints of the system have been designed to withstand vibrational dynamic effects due to valve closures, pump trips, and other operating modes associated with the design operational transients. The tests, as planned, will develop loads similar to those experienced during reactor operation. Compliance with this test program constitutes an acceptable basis, in fulfillment of the applicable requirements of Criterion 15 of the General Design Criteria.

The applicants have proposed dynamic testing and analysis procedures to confirm that all seismic Category I mechanical equipment will function during and after an earthquake of magnitude up to and including the safe shutdown earthquake, and that all equipment support structures will be adequately designed to withstand seismic disturbances.

Subjecting the equipment and its 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 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 gross loss of structural integrity.

The applicants will perform dynamic analyses of piping to determine the effects of turbine stop valve closure and relief valve operation on the main steam piping. The piping system will be modeled as a lumped mass model and a dynamic time-history response of the system will be determined. This response will be combined with the longitudinal stresses produced by the internal pressure, live and dead loads and operating basis earthquake.

The applicants have also committed to submit at the operating license stage of review, a program to determine flow-induced vibrations of piping systems. This program will consist either of tests or analysis. We find this commitment to be acceptable for the construction permit stage of review.

We have concluded that implementation of these dynamic testing and analysis procedures constitutes an acceptable basis for satisfying the applicable 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 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.

The specified design basis combinations of loading 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 (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 will not exceed the allowable design stress and strain limits for the materials of construction.

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 will be provided by limiting the stresses under such loading combinations. We have concluded that the applicants' design load combinations and associated stress and deformation limits specified for all ASME Code Class 2 and 3 components, constitute an acceptable basis for design, in satisfying the applicable requirements of Criteria 1, 2 and 4 of the General Design Criteria and are acceptable.

The applicants will develop and conduct component test programs, supplemented by analytical predictive methods that will provide adequate assurance and confirmation of the capability of ASME Code Class 2 and 3 active valves and pumps to (1) withstand the imposed loads associated with normal, upset, emergency and faulted plant conditions without loss of structural integrity, and (2) perform their "active" function (i.e., valve closure or opening and pump operation) under conditions and combinations of conditions comparable to those expected in effecting a safe plant shutdown or in mitigating the consequences of an accident.

On the basis of our review, we have concluded that the component test program proposed by the applicants will provide an acceptable basis for providing reasonable assurance that the ASME Code Class 2 and 3 active valves and pumps will perform their design safety function.

The criteria used in developing the design and mounting of the safety and relief valves of ASME Code Class 2 systems will 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 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 have concluded that the use of the proposed criteria for the design and installation of overpressure relief devices in ASME Code Class 2 systems constitutes an acceptable design basis in meeting the applicable requirements of Criteria 1, 2, 4, 14 and 15 of the General Design Criteria, and is consistent with the recommendations of Regulatory Guide 1.67.

3.10 Seismic Qualification of Category I Instrumentation and Electrical Equipment

Our evaluation of the design of seismic Category I instrumentation and electrical equipment within the scope of the nuclear steam supply system is presented in Section 3.7 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.

For the balance-of-plant, the applicants have proposed a seismic qualification program that will be implemented for seismic Category I instrumentation and electrical equipment and the associated supports for this equipment to provide assurance that such equipment can be expected to function properly and that structural integrity of the supports will not be impaired during the excitation and vibratory forces imposed by the safe shutdown earthquake and the conditions of post-accident operation.

The applicants have stated in the PSAR that their seismic qualification program is in accordance with Institute of Electrical and Electronics Engineers Standard 344, 1971, "Guide for Seismic Qualification of Class 1 Electric Equipment for Nuclear Power Generating Stations." We have advised the applicants that conformance to Institute of Electrical and Electronic Engineers Standard 344, 1971 is not acceptable and that their qualification program must be supplemented by the procedures and requirements stated in the staff technical paper, "Electrical and Mechanical Seismic Qualification Program." This technical paper is included as Appendix C to this report. The applicants have committed to meet our requirements. We have concluded that this commitment is acceptable.

3.11 Environmental Design of Electrical Equipment

Our evaluation for equipment within the scope of the nuclear steam supply system is presented in Section 7.6.1 of Appendix A to this report. Therefore, the discussion below is limited to equipment within the scope of the balance-of-plant.

Originally, the applicants committed to conform with the requirements of the Institute of Electrical and Electronic Engineers (IEEE) Standard 323, 1974. This standard includes a requirement that acceptable aging of the insulation is completed prior to type testing. However, in Amendment 23 of the PSAR, the applicants deleted their commitment regarding this matter. We have determined that the applicants' proposed modifications of their commitment are unacceptable. Therefore, we required that all Class IE equipment be qualified to satisfy the requirements of IEEE Std 323, 1974. The applicants have orally agreed to conform with our requirements. Subject to confirmatory documentation, we have determined that this commitment is acceptable.

We recognize that the applicants may encounter some problems in implementing the aging requirements of IEEE Std 323, 1974. If, during the construction phase, the applicants encounter difficulty in meeting this standard, we will require them to submit any proposed deviation for our review and resolution prior to installation of the equipment in question.

The applicants have also committed to conform with IEEE Std 383, 1974. The applicants have also stated that they will comply with IEEE Stds 317, 1972, 334, 1971 and 382, 1972, as modified by Regulatory Guide 1.73.

With respect to the proposed environmental qualification of the component cooling water pumps and motors, the applicants originally proposed to qualify these pumps and motors for a temperature of 105 degrees Fahrenheit. The pumps will be located in an area of the mechanical auxiliary building which will be cooled mainly by the supplementary cooler subsystem. This subsystem will receive its cooling from the emergency cooling pond, which under the most severe design basis loss-of-coolant accident conditions could reach a temperature of 115 degrees Fahrenheit. Therefore, we required the applicants to determine the temperature environment of the component coolant pump area. The applicants have determined this temperature to be compatible with a pump environmental qualification temperature of 50 degrees Centigrade (122 degrees Fahrenheit) and have agreed to qualify the pumps and motors accordingly. We have concluded that this is acceptable.

Subject to confirmatory documentation, we have concluded that the proposed environmental design for electrical equipment vital to plant safety is acceptable.

1547 075

4.0 REACTOR

4.1 Introduction

Our evaluation of the reactor is presented in Section 4.0 of Appendix A to this report. Therefore, our discussion below is limited to the fuel surveillance program for the 17x17 XLR fuel assemblies described in RESAR-41 which will be performed on the South Texas Project Unit 1 by the applicants.

The section numbering system used in this section is based on the numbers in Section 4.0 of Appendix A to this report that deal with the same subject matter.

4.2 Mechanical Design

4.2.1 Fuel

4.2.1.4 Fuel Surveillance

Since the South Texas Project Unit 1 is expected to be the first reactor to employ the 17x17 XLR fuel assemblies, a supplemental fuel surveillance program was proposed by the applicants in Amendment 25 of the PSAR. The three important aspects of the program are (1) the visual inspection of all peripheral fuel rods in all of the first core fuel assemblies whenever they are moved to the spent fuel pool, (2) the commitment for destructive examination of fuel rods if deemed necessary from the observation of anomalies, and (3) the timely reporting of results and conclusions to the Commission. We have determined that the proposed fuel surveillance program will provide the final verification of the reliable performance of the 17x17 XLR fuel assemblies and, therefore, is acceptable.

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 South Texas Project Units 1 and 2 nuclear steam supply system and 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 1 components. 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 will provide reasonable assurance that the resulting quality standards are commensurate with the importance of the safety function of the reactor coolant pressure boundary, and is acceptable.

In response to our request, the applicants have made a commitment that no code cases considered unacceptable to the Commission 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). The applicants have also indicated their intent to comply with Regulatory Guides 1.84 and 1.85. In the event the use of new ASME Council approved code cases are planned, authorization will be requested of the Commission prior to their application in the construction of Section III, Class 1 components. We have concluded that compliance by the applicants with the Commission's regulations on the use of approved code cases will result in a component quality level commensurate with the importance of the safety function of the reactor coolant pressure boundary and is acceptable.

1547 077

5.2.2 Overpressurization Protection

The criteria to be used in developing the design and mounting of the safety and relief valves of ASME Code Class 1 systems will 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 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 have concluded that the criteria to be used for the design and installation of overpressure relief devices in ASME Code Class 1 systems meet Regulatory Guide 1.67 and constitute an acceptable design basis in meeting the applicable requirements of Criteria 1, 2, 4, 14 and 15 of the General Design Criteria.

5.2.3 Material Considerations

The applicable code, code edition and addenda regarding fracture toughness and operating limitations for the South Texas Project reactor pressure vessels will be the ASME Boiler and Pressure Vessel Code, Section III, 1971 Edition including Addenda through Summer 1972. The applicable code, code edition and addenda for the reactor pressure vessels comply with the rules of 10 CFR Part 50, Section 50.55a, Codes and Standards and, therefore, are acceptable.

We have reviewed the material selection, toughness requirements and extent of materials testing proposed by the applicants to provide assurance that the ferritic materials used for pressure retaining Code 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. All ferritic materials for Class 2 and Class 3 components will meet the toughness requirements of the ASME Boiler and Pressure Vessel Code, Section III as supplemented by our requirements.

The fracture toughness tests and procedures required by Section III of the ASME Code and as supplemented by our requirements provide reasonable assurance that adequate safety margins against the possibility of nonductile behavior or rapidly propagating fracture can be established for ferritic materials used for pressure-retaining Code Class 2 and Class 3 components, both within and outside the reactor coolant system.

5.2.4 Inservice Inspection Program

To assure that no deleterious defects develop in the reactor coolant system during service, selected welds and weld heat-affected zones will be inspected periodically. The design of the reactor coolant system incorporates provisions for access for inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code. In addition, means 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 will be in accordance with the requirements of Section XI of the ASME Boiler and Pressure Vessel Code. This will provide 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.

We have concluded that compliance with the inservice inspections required by Section XI of the ASME Code constitutes an acceptable basis for satisfying the requirements of Criterion 32 of the General Design Criteria.

To assure 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 in accordance with Section XI of the ASME Code. ASME Code Class 2 systems and Code Class 3 systems will receive visual inspections with the systems pressurized in order to detect evidence of structural degradation or loss of leak-tight integrity.

Examples of Code Class 2 systems are: (1) residual heat removal systems, (2) portions of chemical and volume control systems, and (3) those portions of the engineered safety features not part of the ASME Code Class 1 systems. Examples of ASME Code Class 3 systems are: (1) component cooling water systems, and (2) portions of radwaste systems. All of these systems transport fluids. The ASME Code Class 2 and 3 systems will be in conformance with the inservice inspections required by Section XI of the ASME Code and our requirements. We have concluded that for ASME Code Class 2 and 3 systems and components compliance with the inservice inspections required by Section XI of the ASME Code and our requirements constitutes an acceptable basis for satisfying the applicable requirements of Criteria 36, 39, 42, and 45 of the General Design Criteria.

5.2.5 Detection of Leakage Through the Reactor Coolant Pressure Boundary

We have reviewed the leakage detection systems proposed by the applicants as described in Section 5.2-7 of the PSAR.

The proposed systems for detection of coolant leakage to containment will provide (1) diverse leak detection methods, (2) sufficient sensitivity to measure small leaks, (3) identification of the leakage source to the extent practical, and (4) suitable control room alarms and readouts.

The major components of this system will be the containment atmosphere gas and particulate radioactivity monitors, the containment drains sump level monitor, and the containment air recirculation fan cooler temperature monitor. Indirect indication of leakage will also be possible from observation of the reactor coolant system makeup flow rate monitor, containment drains transfer tank and pressurizer relief tank monitor, pressure and temperature indicators.

The design criteria for leakage detection systems proposed to detect leakage from components and piping of the reactor coolant pressure boundary, are in accordance with Regulatory Guide 1.45, and provide reasonable assurance that any material degradation resulting in leakage during service will be detected in time to permit corrective actions.

1547 079

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 systems are acceptable.

5.3 Reactor Vessel and Appurtenances

The applicable code, code edition and addenda for the South Texas Project reactor pressure vessels will be the ASME Boiler and Pressure Vessel Code, Section III, 1971 Edition including Addenda through Summer 1972.

5.4 Component and Subsystem Design

5.4.7 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 such reasons, we have encouraged applicants over the past several years to participate in programs designed to develop an effective, on-line loose parts monitoring system. For the past few years we have required many applicants 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 being installed at several plants.

As a result of our review, the applicants have indicated that an analysis of available systems for loose parts monitoring will be initiated and have made a commitment to install an appropriate available system.

We have concluded that this commitment is acceptable for the construction permit stage of review.

1547 080

6.0 ENGINEERED SAFETY FEATURES

6.1 Design Considerations

The purpose of the various engineered safety features is to provide a complete and consistent means of assuring that the plant personnel and the public will be protected from excessive exposure to radioactive materials should a major accident occur in the plant. In this section we discuss the engineered safety feature systems proposed for the South Texas facility. Certain of these systems or parts of these systems will have functions for normal plant operation as well as serving as engineered safety features.

We have reviewed the proposed systems and components designated as engineered safety features. These systems and components 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. They will be designed, therefore, to seismic Category I requirements 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. These design requirements are in accordance with the General Design Criteria, Appendix A to 10 CFR Part 50.

6.2 Containment Systems

The containment systems for each of the South Texas Project units will include a reactor containment structure, containment heat removal systems, a containment isolation system, a containment combustible gas control system, and provisions for containment leakage testing.

6.2.1 Containment Functional Design

1547 081

The containment will consist of a steel-lined, prestressed concrete structure with a net free volume of 3,300,000 cubic feet. The containment structure will house the nuclear steam supply system, including 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 56.5 pounds per square inch gauge and a temperature of 312 degrees Fahrenheit.

The applicants have analyzed the containment pressure responses for postulated loss-of-coolant accidents in the following manner. Mass and energy release rates from the postulated reactor coolant system and secondary system pipe breaks to the containment were based upon the data provided in RESAR-41. Our evaluation of the mass and energy release rates provided in RESAR-41 is presented in Section 6.2.1 of Appendix A to this report. These data were used as input to the CONTEMPT computer code which performs transient thermodynamic calculations with appropriate consideration of containment heat removal systems and structural heat sinks to calculate the containment pressure. The CONTEMPT computer code used by the applicants is basically the same code used by us to perform containment pressure analyses.

The applicants have described the methods used to determine the containment design pressure in the PSAR. A spectrum of postulated loss-of-coolant break locations and sizes and steam line breaks were considered. The results of analyses have shown that the consequences of a postulated double-ended pipe rupture at the pump suction of the reactor coolant system will result in the highest containment pressure and is the design basis loss-of-coolant accident. The single active failure applied to this design basis is the loss of one of three containment fan cooler system trains (one fan cooler train consists of two fan cooler units) resulting from a failure to start, with an additional fan cooler from one of the other two trains out of service for maintenance. The minimum containment heat removal capability then will consist of the containment spray system, a fan cooler system train, and one of the fan cooler units of a second train. Full safety injection of the emergency core cooling system is correspondingly assumed. Additional heat removal is provided by heat transfer to the containment structures.

The mass and energy release rate data given in RESAR-41 for the design basis loss-of-coolant accident are based on a containment back pressure of 37 pounds per square inch gauge, which is substantially lower than the containment design pressure of 56.5 pounds per square inch gauge for the South Texas Project. Sensitivity studies on other plants have indicated that the release rates will increase as the containment pressure increases. Therefore, we requested the applicants to provide justification for using a containment back pressure of 37 pounds per square inch gauge. The applicants have adjusted the mass and energy release rates for the design basis loss-of-coolant accident by using the results of a sensitivity study described in the Westinghouse Topical Report WCAP-8312, "Westinghouse Mass and Energy Release Data for Containment Design." This report has been previously accepted by the staff. Based on this adjusted mass and energy release data, the applicants showed that the containment pressure will increase by about one psi for the design basis loss-of-coolant accident.

The applicants have calculated a peak containment pressure for the design basis loss-of-coolant accident of 50.7 pounds per square inch gauge which includes one psi for the increase in mass and energy release rates due to a higher containment back pressure than that used in RESAR-41 as discussed previously. We have also analyzed the containment response to the design basis loss-of-coolant accident, and we have calculated a peak containment pressure of 50 pounds per square inch gauge which also includes the increase in mass and energy release rates due to a higher containment back pressure. The containment design pressure provides an 11 percent margin above the peak calculated pressure. Therefore, we have concluded that the applicants' containment analysis for the South Texas Project is acceptable and that the containment design pressure is adequate.

The applicants have analyzed the containment pressure response to a postulated failure of a main steam line using the RELAP-3 computer code to calculate the mass and energy release rates. The applicants conservatively assumed the failure of the steam line isolation valve in the ruptured line to close and that only dry steam will be released through the break. The applicants calculated a peak containment pressure of 41.0 pounds per square inch gauge which is considerably less than the pressure calculated for the design basis loss-of-coolant accident. Based on our review, we have concluded that the applicants have conservatively calculated the containment pressure response for the postulated steam line break and that the containment design pressure of 56.5 pounds per square inch gauge is acceptable.

The applicants have analyzed the containment pressure response assuming inadvertent actuation of the containment spray system. The applicants' calculations indicate that the maximum external pressure that the containment would be subjected to is 2.75 pounds per square inch differential.

We have performed a similar analysis and concur with the applicants' result. Therefore, we have concluded that the proposed external design pressure of 3.5 pounds per square inch differential for the containment is acceptable.

The applicants have analyzed the pressure response of various containment interior compartments to postulated high energy line breaks. The compartments investigated include the reactor cavity and inspection toroid, pressurizer compartment, pressurizer surge line compartment, steam generator compartments, and the steam and feedwater line compartments outside the secondary shield wall. The SATAN-V computer code was used to calculate the mass and energy release rates for the worst credible breaks assignable to each compartment.

The applicants used the RELAP-3 computer code to calculate the peak subcompartment pressure differentials. The subcompartment pressures calculated by the applicants are summarized in Table 6.1. The applicants have conservatively used the maximum calculated absolute pressure to determine the maximum differential pressures for the subcompartments; i.e., the applicants neglected increases in the pressures on the outside surfaces of structures. Design differential pressures for the subcompartments will be established by applying a 40 percent margin above the maximum calculated value, which is consistent with our current practice.

We have performed confirmatory analyses of the pressure response in the subcompartments using the RELAP-3 computer code, and our results are in agreement with the applicants' results.

1547 083

TABLE 6.1

RESULTS OF SUBCOMPARTMENT ANALYSIS

Compartment	Postulated Piping Failure	Peak Differential Pressure (psid)*
Steam Generator Compartment	Double-Ended Cold Leg Double-Ended Hot Leg	23 19
Pressurizer Compartment	Spray Line	33
Surge Line Compartment	Surge Line	73
Pressurizer Skirt	Surge Line	167
Reactor Vessel Nozzle Safe End Inspection Toroid	150 square inches Cold Leg Limited Displacement Rupture	136
Reactor Cavity	150 square inches Cold Leg Limited Displacement Rupture	61
Steam Line Compartment (Between Elevation 52 feet and 68 feet)	Double-Ended Steam Line	39
Feedwater Line Compartment (Between Elevation 37 feet 3 inches and 50 feet)	Double-Ended Feedwater Line	8

*Maximum pressure minus initial containment pressure (14.7 pounds per square inch absolute).

1547 084

We have evaluated the proposed containment system functional design for conformance with the General Design Criteria, in particular Criteria 16 and 50. We have concluded that the proposed containment design pressure of 56.5 pounds per square inch gauge provides an acceptable margin when compared to the maximum calculated containment pressure of 50.7 pounds per square inch gauge. In addition, based on our confirmatory calculations and the 40 percent margin specified for the subcompartment design pressure differentials, we find that the proposed subcompartment design pressure differentials are acceptable. Therefore, we have concluded that the containment functional design conforms with the requirements of Criteria 16 and 50 of the General Design Criteria and is acceptable.

6.2.2 Containment Heat Removal Systems

The containment heat removal systems will consist of the containment fan cooler system and the containment spray system. The containment fan cooler system and the containment spray system will reduce the containment pressure following postulated high energy line break accidents. The containment fan cooler system will also be used during normal plant operation, whereas the containment spray system will serve only as an engineered safety feature and will perform no normal operating function.

The containment spray system will consist of three separate spray trains of equal capacity. Each train will have its own individual containment emergency sump. All active components of the system will be located outside the containment vessel to facilitate maintenance operations. Missile protection will be provided by direct shielding or physical separation of equipment. The system will be designed to seismic Category I requirements. The containment spray pump recirculation intake from the containment emergency sump will be enclosed by a screen assembly designed to prevent debris from entering that could clog the spray nozzles. The protective screen assembly design will be consistent with the recommendations of Regulatory Guide 1.82.

A high containment pressure signal from the safety features actuation system will automatically actuate the containment spray system. The system design will permit manual operation of pumps and valves from the control room. The spray pumps will initially take suction from the refueling water storage tank. When the water in the tank reaches a low level, a switchover from injection to recirculation will be initiated automatically.

The applicants have provided an analysis which demonstrates that sufficient net positive suction head will be available to the spray pumps for both the injection and recirculation modes of operation. This analysis was performed consistent with the recommendations of Regulatory Guide 1.1.

The containment fan cooler system will consist of three separate fan cooler trains of equal capacity. Each train of the containment fan cooler system will include two fan cooler units. The system components and equipment required to remain operable

1547 085

following an accident will be located outside the secondary concrete shield for missile protection at an elevation that precludes flooding, and will be designed to seismic Category I requirements. The system will be designed to withstand the dynamic conditions following a postulated loss-of-coolant accident.

A high containment pressure signal or a low reactor coolant system pressure signal from the safety features actuation system will automatically actuate the containment air cooling system. The system design will permit manual operation from the control room.

Based on our review of the containment heat removal systems, we have concluded that the containment heat removal systems will be designed to meet the requirements of Criteria 38, 39, and 40 of the General Design Criteria and, therefore, are acceptable. The applicants have committed to provide in the Final Safety Analysis Report a dynamic analysis of the differential pressure imposed on the components and equipment of the containment fan cooler system to show that the calculated differential pressure will not exceed the design differential pressure. We find this commitment acceptable.

6.2.3 Containment Air Cleanup System

In addition to the containment spray system's heat removal functions, the system will be used for iodine removal from the containment atmosphere following a postulated design basis loss-of-coolant accident. Sodium hydroxide will be added to the containment spray solution to enhance the iodine scrubbing function of the system. The system will be designed to raise the pH, or hydrogen ion concentration, of the spray to 10.5 during the injection phase of operation of the spray system. A sufficient quantity of sodium hydroxide will be injected to raise the equilibrium pH in the containment sump to a minimum value of 8.5.

We have calculated the removal coefficients for elemental iodine used in the dose analysis presented in Section 15.0 of this report. We have determined that the first order removal coefficients for elemental and particulate iodine are 10 and 0.45 inverse hours, respectively, in an estimated effective volume of 2,750,000 cubic feet. The minimum sump pH of 8.5 is considered adequate to achieve and maintain a decontamination factor of 100 for the elemental iodine. We have evaluated the containment spray and containment spray additive subsystem and find them effective for removal of elemental iodine and iodine absorbed on airborne particulate matter.

6.2.4 Containment Isolation System

The containment isolation system will be designed to automatically isolate piping systems that penetrate the containment to prevent outleakage of the containment atmosphere following postulated loss-of-coolant accidents. 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. Containment penetration piping 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.

1547 086

Containment isolation will occur automatically upon receipt of containment high pressure signals or reactor coolant system low pressure signals from the safety features actuation system. High radiation signals will also be used to isolate the containment vessel purge system lines.

Based on our review, we have concluded that the design of the containment isolation system conforms to Criteria 54, 55, 56 and 57 of the General Design Criteria and the recommendations of Regulatory Guide 1.11, and is acceptable.

6.2.5 Combustible Gas Control System

Following a postulated loss-of-coolant accident, hydrogen may accumulate inside the containment as a result of (1) a 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 spray solution, and (3) radiolytic decomposition of the cooling water in the reactor core and containment sump.

The combustible gas control system will be designed to control the concentration of hydrogen within the containment vessel following a loss-of-coolant accident. The system will consist of the containment mixing system (containment fan cooler system), hydrogen recombiner system, hydrogen monitoring system and supplementary containment purge system.

Redundant, portable hydrogen recombiners will be provided for combustible gas control following a loss-of-coolant accident. One of the recombiner units will be located at the South Texas Project site, while the other unit will be located at the applicants' Allens Creek site. The distance between the South Texas Project site and the Allens Creek site is approximately 50 miles. The recombiner units will be designed such that they can be transported between the Allens Creek site and the South Texas Project site.

The hydrogen recombiner units will be compatible with the South Texas facility design and will be capable of maintaining the hydrogen concentration in the containment following a postulated loss-of-coolant accident below the limits specified in Regulatory Guide 1.7. The hydrogen recombiner system will be designed to seismic Category I requirements. In addition, separate supply and exhaust piping penetrations will be provided for each recombiner.

The supplementary containment purge system will serve as a backup to the hydrogen recombiner system. The system will consist of a single exhaust train having two 100 percent capacity fans. It will release the containment structure atmosphere through the plant main exhaust duct. The supplementary containment purge system will not be designed to seismic Category I requirements.

Based on our review of the systems provided for combustible gas control following a postulated loss-of-coolant accident, we have concluded that the systems will conform to Criteria 41, 42 and 43 of the General Design Criteria and the recommendations of Regulatory Guide 1.7, and are acceptable.

6.2.6 Containment Leakage Testing Program

The containment design will include provisions and features which satisfy the testing requirements of Appendix J to 10 CFR Part 50. The design of the containment penetrations and isolation valves will permit individual periodic leakage rate testing at the pressure specified in Appendix J to 10 CFR Part 50. Included will be those penetrations that have resilient seals and expansion bellows, such as personnel airlocks, equipment hatch, refueling tube blind flange, hot process line penetrations 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 integrity can be verified throughout the service lifetime of the plant and that the leakage rates will be periodically checked on a timely basis to assure that they are within specified limits.

Maintaining containment leakage rates within such limits provides reasonable assurance that, in the event of any radioactivity releases within the containment structure, the loss of containment atmosphere through potential leakage paths will not be in excess of acceptable limits specified for the site, i.e., the doses will be well within 10 CFR Part 100 limits.

We have concluded that compliance with the leakage testing requirements of Appendix J to 10 CFR Part 50 constitutes an acceptable basis for satisfying the requirements of Criteria 52, 53 and 54 of the General Design Criteria.

6.3 Emergency Core Cooling Systems

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

The applicants have agreed to submit the minimum containment pressure analysis as required by Appendix K to 10 CFR Part 50 for use in the evaluation of emergency core cooling system performance in conjunction with the RESAR-41 emergency core cooling system evaluation model. We will review this information and our evaluation and conclusions will be reported in a supplement to the Safety Evaluation Report.

6.4 Emergency Boration System

Our evaluation of the emergency boration system is presented in Section 6.4 of Appendix A to this report.

6.5 Control Room Habitability

The applicants have proposed to meet the control room habitability requirements of Criterion 19 of the General Design Criteria by use of concrete shielding and by installing three 1000 cubic feet per minute once through charcoal filter units and three 5000 cubic feet per minute recirculating charcoal filters in the control room ventilation system. Two-out-of-three of the emergency filter trains are assumed to operate upon receipt of a safety injection signal, thus providing 2000 cubic feet per minute of filtered outside air for pressurization and, 10,000 cubic feet per minute of filtered control room recirculation air.

In Amendment 19 to the PSAR, the applicants have proposed to eliminate the provision for initiation of the automatic isolation of the control room ventilation system by a signal from radiation detectors located within the outside air intakes for the control room.

We reviewed the applicants' proposed design and have determined that it is unacceptable. We have advised the applicants that it is our position that this type of automatic actuation, in addition to other signals (such as the Safety Injection Signal), is necessary for adequate protection of control room personnel against airborne radioactive contaminants from various potential sources on or in the vicinity of the site. The applicants have orally agreed to provide this type cf automatic actuation. Subject to confirmatory documentation, we have concluded that this commitment is acceptable.

A portion of the 2000 cubic feet per minute of outside air will be used to pressurize the electrical portion of the mechanical and electrical auxiliaries building which houses the control room. A pressure test will be performed to assure that both of the areas serviced by the 2000 cubic feet per minute pressurization flow will be maintained at the design positive pressures of 1/4 inch water gauge for the control room, and 1/8 inch water gauge for the electrical portion of the mechanical and electrical auxiliaries building. We have independently calculated the potential radiation doses to control room personnel following the design basis loss-of-coolant accident in consideration of the habitability systems proposed for the control room and have concluded that the resultant thyroid and whole body gamma doses will be within the guidelines of Criterion 19 of the General Design Criteria.

We have also considered the potential for toxic gas release with respect to control room habitability in accordance with the recommendations of Regulatory Guide 1.78. A number of sources of toxic gases were identified as potential hazards, namely, storage at and shipment to and from the nearby Celanese Chemical Company of acetaldehyde, anhydrous ammonia, cyclohexane, and vinyl acetate. The applicants have performed analyses of the effects of postulated releases of these four chemicals on the control room habitability. As a result of these analyses, the applicants have indicated that provisions for detection and automatic control room isolation will be provided for acetaldehyde, and for detection only for vinyl acetate. Since the concentrations of anhydrous ammonia or cyclohexane do not exceed their toxicity limits, no detection instrumentation or protective action is needed for these chemicals.

1547 089

We have evaluated the potential effects of these chemicals, taking a conservative and realistic surface area of spillage. Our independent calculations show that given the proposed protective provisions, the release of these four chemicals at the various source locations would not present a hazard to the control room personnel provided that adequate breathing apparatus is supplied. The applicants have committed to provide an appropriate quantity of self-contained breathing apparatus with sufficient (six-hour) bottled air supply and the ability to replenish the air supply as needed.

Subject to confirmatory documentation regarding automatic isolation of the control room on a high radiation signal, we have concluded that on the basis of our review and independent calculation the control room habitability system is acceptable. The control room ventilation system is further discussed in Section 9.4 of this report.

6.6 Engineered Safety Features Air Filtration Systems

The engineered safety features air filtration systems proposed for the South Texas facility include the fuel handling building exhaust subsystem and the control room emergency ventilation system. We have evaluated the design of these systems with respect to the recommendations in Regulatory Guide 1.52 and have determined that the control room emergency ventilation system conforms with the recommendations of this guide. We also determined that the proposed design of the fuel handling building exhaust subsystem did not conform with all of the recommendations of this guide.

The exception pertained to the seismic classification of the fuel handling building exhaust subsystem. We advised the applicants that the design of this system to non-seismic Category I requirements is unacceptable. Because the calculated potential doses from fuel handling accidents are significant fractions of 10 CFR Part 100 guideline doses, we required that the fuel handling building exhaust subsystem be designed as a seismic Category I system (see Section 9.4.3 of this report). The applicants have orally agreed to design the fuel handling building exhaust subsystem as a seismic Category I system. Subject to confirmatory documentation, we have concluded that the design of the engineered safety features air filtration systems is acceptable.

6.7 Engineered Safety Features Materials

Our evaluation of the materials that will be used in engineered safety features which are within the scope of the nuclear steam supply system is presented in Section 6.5 of Appendix A to this report. Therefore, the discussion below is limited to materials which are within the scope of the balance-of-plant.

We have reviewed the mechanical properties of materials that will be used for the balance-of-plant engineered safety features and have determined that they will satisfy Appendix I of Section III of the American Society of Mechanical Engineers (ASME) Code, or Parts A, B, and C of Section II of the Code, and the staff position that the yield strength of cold-worked stainless steels shall be less than 90,000 pounds per square inch.

The controls on the use and fabrication of the austenitic stainless steel in the systems 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 will provide added assurance that stress-corrosion cracking will not occur during the postulated accident time interval.

The controls on the pH, or hydrogen ion concentration, of the reactor containment sprays and the emergency core cooling water following a postulated loss-of-coolant accident are adequate to assure freedom from stress-corrosion cracking of the austenitic stainless steel components and welds of the containment spray and emergency core cooling systems throughout the duration of the postulated accident to completion of cleanup. In addition, the control of the pH of the sprays and cooling water, in conjunction with controls on selection of containment materials, in accordance with recommendations of Regulatory Guide 1.7, provides assurance that the sprays and cooling water will not give rise to excessive hydrogen gas evolution by corrosion of containment metal, or 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 features are in accordance with the recommendations of Regulatory Guide 1.36.

Conformance with the Codes and Regulatory Guide recommendations mentioned above, our requirements on the allowable maximum yield strength of cold-worked austenitic stainless steel, and our requirements on the minimum level of pH 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.

We have reviewed the selection of materials proposed for the engineered safety features for the balance of plant, in conjunction with the expected chemistry of the cooling and containment spray system water. The applicants have shown that the use of sensitized stainless steel will be avoided. We have concluded that the proposed controls on material and cooling water chemistry will provide assurance that the integrity of components of these systems will not be impaired by corrosion or stress corrosion.

1547 091

7.0 INSTRUMENTATION AND CONTROLS

7.1 Introduction

7.1.1 General

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

The South Texas Project Preliminary Safety Analysis Report references appropriate portions of the Westinghouse Reference Safety Analysis Report (RESAR-41) for systems that will be identical to that design. Our review of the South Texas Project protection and control systems took cognizance of this RESAR-41 reference material and was limited to those systems and equipment which were within the balance-of-plant scope of supply and those that interface with RESAR-41 requirements. As discussed in Section 1.8 of this report, our review of interfaces is not complete. We will report the results of our review in a supplement to the Safety Evaluation Report.

Our evaluation of the instrumentation and controls of the equipment within the scope of the nuclear steam supply system is presented in Section 7.0 of Appendix A to this report. Therefore, the discussion below is limited to the instrumentation and controls of equipment within the scope of the balance-of-plant.

7.1.2 Design Criteria

We have reviewed the design criteria specified in RESAR-41. The applicants have agreed to revise the South Texas Project PSAR to be consistent with the criteria specified in RESAR-41.

Subject to receipt of documentation, we have concluded that the design criteria for the South Texas Project are acceptable.

7.2 Reactor Trip System

The proposed reactor trip system consists of the initiating circuits, logic, bypasses, redundancy, diversity and actuated devices utilized to initiate reactor shutdown.

The majority of the reactor trip system input signals will be within the Westinghouse scope of supply and are identified in RESAR-41. The applicants have also proposed to include the options of containment and control room radioactivity detection and high steam generator water level. The applicants will provide three inputs to the reactor trip system namely, the turbine trip, and the reactor coolant pump underpower and underfrequency trips. They also committed to design these inputs to meet the requirements specified in RESAR-41.

We have determined that the applicants' implementation of RESAR-41 underfrequency trip design features to be inadequate and unacceptable in that they did not provide a description of the instrumentation to be employed for the grid-frequency decay rate trip. The applicants have recently submitted this information. In addition, the underfrequency set point specification of "not higher than 57 HZ" is not consistent with that required by RESAR-41. We will evaluate the information submitted by the applicants and report the resolution of this matter in a supplement to the Safety Evaluation Report.

On the basis of our review, we have concluded that, with the satisfactory resolution of the above considerations, including interface design items, the reactor trip system will conform to applicable regulations, guides, technical positions, and industry standards, and is acceptable.

7.3 Engineered Safety Features Actuation Systems

The engineered safety features actuation systems include the instrumentation and controls used to detect a plant condition requiring operation of an engineered safety features system, to initiate action of the engineered safety features, and to control its operation. The actuation logic of all engineered safety features and engineered safety features support systems with the exception of the fuel handling building exhaust subsystem and the containment combustible gas control system will be within the scope of the nuclear steam supply system. The applicants have referenced Section 7.3 of RESAR-41 for detailed information on the actuation logic and on those engineered safety features system within the scope of the nuclear steam supply system. The engineered safety features systems within the scope of the nuclear steam supply system will be the (1) emergency core cooling system, and (2) emergency boration system.

The engineered safety features systems and engineered safety features support systems within the scope of the balance-of-plant will be the (1) containment heat removal system, (2) containment spray system, (3) containment isolation system, (4) essential cooling water system, (5) containment combustible gas control system, (6) auxiliary feedwater system, (7) main feedwater and steam line isolation system, (8) control room ventilation system, (9) component cooling water system, (10) containment penetration exhaust subsystem, and (11) fuel handling building exhaust subsystem.

Our evaluation of the engineered safety features within the scope of the nuclear steam supply system is presented in Section 7.3 of Appendix A to this report.

We have reviewed the information provided by the applicants including the design bases, analyses of conformance to applicable criteria and simplified logic diagrams defining the interface and scope of equipment between the nuclear steam supply system and the balance of plant. We will review the detailed system design and the manner of implementation of all applicable design criteria committed to in the PSAR at the operating license stage of review.

We have concluded that, with the satisfactory resolution of the interface items and subject to confirmatory documentation of items discussed in Sections 7.3.1 and 7.3.3 of this report, the design of the instrumentation and controls associated with the engineered safety features systems satisfy the Commission's requirements identified in Section 7.1 of this report and will be acceptable.

7.3.1 Control Room Heating, Ventilation, and Air Conditioning System

In Amendment 19 of the PSAR, the applicants propose to eliminate the provisions for initiation of the automatic isolation of the control room ventilation system by a signal from radiation detectors located within the outside air intakes for the control room. As discussed in Section 6.5 of this report, we required that the applicants provide this type of automatic isolation. The applicants have orally agreed to this type of automatic isolation. Subject to confirmatory documentation we have concluded that the design of the control room heating, ventilation and air conditioning system is acceptable.

7.3.2 Isolation Devices

The design of the South Texas Project engineered safety feature systems design will include analog, alarm and trip signals which will be provided to the facility computer and annunciation systems through isolation devices. The applicants have stated that these isolation devices will be identical to those employed in the RESAR-41 design. We find this committment to be acceptable.

7.3.3 Component Cooling Water System

The proposed component cooling water system design provides a single cooling water supply line and a single return line for all four reactor coolant pumps. The inlet line contains one motor operated valve and one check valve for containment isolation. The return line contains two motor operated containment isolation valves in series. Inadvertent closure of any one of the three motor operated valves would terminate the coolant flow for the reactor coolant pump seals and bearings. As discussed in Section 9.2.2 of this report, it is our position that this could result in reactor coolant pump failure without coastdown. This part of the system, therefore, does not meet the single failure criterion for a safety related function and is, therefore, not acceptable. We required the applicants to provide system redundancy so that a single active failure will not cause loss of coolant flow to the reactor coolant pumps. As discussed in Section 9.2.2 of this report, the applicants have conformed with our requirements. Therefore, we have concluded that the design is acceptable.

7.4 Systems Required for Safe Shutdown

The applicants have referenced Section 7.4 of RESAR-41 for information on systems required for safe shutdown. In addition, they have identified and described balance-of-plant systems required for safe shutdown and the features of the auxiliary shutdown control panel proposed for maintaining a safe shutdown condition from outside the control room.

We have reviewed the descriptive information relating to those systems including preliminary logic diagrams, design criteria, design bases, and the applicants' analysis of the adequacy of these criteria and bases.

Subject to the resolution of the interface items, we have concluded that the design of the instrumentation and control systems required for safe shutdown conforms to the Commission's requirements identified in Section 7.1 of this report and is acceptable.

7.5 Safety Related Display Instrumentation

The safety related display instrumentation will provide the operator with information on the status of the plant to allow manual safety actions to be performed whenever necessary and for post-accident and incident monitoring. The applicants have referenced Section 7.5 of RESAR-41 for information on safety related display instrumentation not within the balance-of-plant scope.

Our review of the safety related display instrumentation included the descriptive information provided on monitoring plant variables, safety related systems, post-accident and incident monitoring and the applicants' proposed design criteria and design bases, including that for indication of bypassed and inoperable safety related systems.

We have concluded that with the satisfactory resolution of the item discussed in Section 7.5.1 of this report, the design of the safety related display instrumentation conforms to the Commission's requirements identified in Section 7.1 of this report and will be acceptable. We will review the details and implementation of the post-accident and incident monitoring system to assure its conformance to IEEE 279-1971 requirements at the operating license review stage.

7.5.1 Monitoring of Heating, Ventilation, and Air Conditioning Systems

The applicants had originally included monitors in the control room and fuel handling building ventilation systems in the list of plant variables associated with engineered safety features actuation and for monitoring during and after an accident. These variables were removed from the list in later amendments to the PSAR but with no justification for their deletion. We will require the results of an analysis which justifies this action. We will report the resolution of this item in a supplement to the Safety Evaluation Report.

7.6 Other Systems Required For Safety

The applicants have identified the instrumentation and control power supply system, the residual heat removal isolation valves interlocks, the refueling interlocks and monitoring of combustible gas in containment as systems required for safety. With the exception of the monitoring of combustible gas in containment, the applicants have referenced Section 7.6 of RESAR-41 for information on these systems required for safety. Our evaluation of these systems is presented in Section 7.6 of Appendix A to this report. Therefore, we have limited the discussion below to the combustible gas monitoring system.

Our review of the combustible gas monitoring system included the descriptive material, the proposed design criteria and design bases of this system.

Subject to the satisfactory resolution of the interface items, we have concluded that the proposed designs of the combustible gas monitoring systems satisfies the Commission's requirements identified in Section 7.1 of this report and will be acceptable.

7.7 Control Systems Not Required for Safety

The applicants have included documentation of the features of the proposed South Texas Project bypass status indication system including identification of the systems requiring bypass indication on a system level. They have also provided a commitment that the design will satisfy the requirements of Section 4.13 of IEEE Std 279-1971 and the recommendations of Regulatory Guide 1.47, as supplemented by our position on application of this Regulatory Guide.

On this basis, we have concluded that the control systems not required for safety are acceptable.

1547 096

8.0 ELECTRIC POWER

8.1 General

The following served as the basis for evaluating the adequacy of the electric power systems:

(1) Criteria 17 and 18 of the Commission's General Design Criteria.

(2) Institute of Electrical and Electronics Engineers (IEEE) Standards, including IEEE Criteria for Class IE systems for Nuclear Power Generating Stations (IEEE 308-1974).

(3) The applicable Commission's Regulatory Guides.

8.2 Offsite Power System

The South Texas Project Units 1 and 2 will feed power through two transmission lines into a 345 kilovolt (kV) switchyard which will tie into the Texas Interconnected System. The facility's two main generators will provide power at 25 kilovolts through isolated phase buses to the two main transformer banks. Eight 345 kilovolt transmission circuits will connect to the switchyard and terminate at six points in the Texas Interconnected System grid over three separate rights-of-way as follows: (1) Velasco substation (2 circuits), (2) a) W. A. Parish substation, b) Hill Country substation (2 circuits), c) Glidden substation, and d) Lon Hill substation, and (3) Rincon substation. The normal power supply to the unit's balance-of-plant auxiliary loads will be provided through the unit auxiliary transformer connected to the generator bus. Upon tripping of the generator these loads will be automatically transferred to the unit's standby transformer which will furnish offsite power. The engineered safety features auxiliary buses will be normally served from the unit's standby transformer which will be connected to the 345 kilovolt switchyard. Upon trip of a standby transformer, the engineered safety features auxiliary buses will be automatically transferred to the standby transformer of the other unit. Unit 1 and Unit 2 standby transformers will be individually supplied by separate and independent overhead 345 kilovolt ties from the switchyard. Each standby transformer will have sufficient capacity to provide power for startup and full load operation of either unit and will be capable of providing normal offsite power for simultaneous normal shutdown of both units or concurrent full auxiliary loads of one unit and design basis accident loads in the other unit.

A 138 kilovolt system will provide an alternate, separate and independent supply of offsite power to the South Texas Project from a tap off the Central Power and Light Company's Blessing-Bay City 138 kilovolt circuit via a 138 kilovolt emergency transformer. Two transformers, i.e., the nearby unit's standby transformer and the 138 kilovolt emergency transformer will provide the means of furnishing offsite electrical power to the engineered safety features buses when the normal source is not available. The 138 kilovolt emergency transformer will have sufficient capacity to provide power for simultaneous shutdown of one unit and design basis accident loads in the other unit, in addition to other loads it normally serves. The 138 kilovolt emergency transformer may be used as a source of engineered safety features power for any or all engineered safety features buses for Units 1 and 2 by manual transfer from the control room.

The generator lines and transmission lines at the 345 kilovolt switchyard will be arranged in a breaker-and-a half scheme. There will be no crossings of the transmission circuits from the South Texas Project which would affect the availability of the offsite power to the switchyard. Protective relay systems will be utilized in line operation, including primary and backup relaying on each circuit out of the plant along with breaker failure backup tripping. The switchyard will be provided with two independent 125 volt direct-current systems, each with its own battery and battery charger for operation of the high voltage circuit breakers. The primary and backup relaying protection schemes will be connected so that failure of any component in either scheme will not affect the relaying protection of the other scheme. Periodic inspection and testing of the switchyard breakers and protective relaying will be provided to assure the operability and functional performance of the components.

The applicants have conducted electrical grid stability and availability studies of the transmission system. The analysis included steady state and transient stability and transient and sustained outage conditions. The results of the stability studies demonstrated that the loss of both units at the South Texas Project or the loss of one unit with the other unit online would not impair the ability of the system to supply power to the engineered safety features system. The availability studies demonstrated that offsite power to the engineered safety features system will be highly reliable even if the generating units for Units 1 and 2 are not operating and no improvement in line outage rate over present levels is experienced.

As a result of our review and in consideration of the applicants' commitments, we have concluded that the design of the offsite power system will meet the Commission's requirements identified in Section 8.1 of this report and is acceptable.

8.3 Onsite Power System

8.3.1 Alternating Current Power System

The proposed alternating current onsite power system of each South Texas Project unit will consist of four major subsystems, (1) the main auxiliary power distribution system, (2) the normal power distribution system, (3) the emergency power distribution system and (4) the onsite-standby power generation and distribution system. The main auxiliary power distribution system supplies power to the non-Class IE loads and the other systems serve the Class IE loads. During normal operation, each unit's electrical power is supplied by the unit auxiliary transformer, with the exception of all 4.16 kilovolt engineered safety features loads which are supplied by the standby transformer. The preferred sources of emergency power are either of the two unit standby transformers or the 138 kilovolt emergency transformer, all of which are fed from the system grid.

1547 098

An alternate onsite emergency power source for the Class IE engineered safety features loads will be provided by the standby diesel generator units. Controls, instruments and alarms will monitor the status of the system and provide controls and intelligence for routine operation of the plant and emergency operation.

There will be three redundant power supply Class IE load groups for each reactor unit. Each load group will be provided with an onsite standby power source, electrical bus distribution cables, and other devices physically and electrically separated from the other load groups. Equipment for each of these redundant load groups will be located in a separate room of the seismic Category I structure and qualified to withstand the seismic requirements and environment in which it will be located.

Each Class IE load group will have its own standby diesel generator for a source of power and its own 125 volt direct current system. (An additional 125 volt direct current system for the fourth reactor protection system channel is provided using train A as the power supply). Although we find this design acceptable, we will require a more restrictive plant technical specification on maintenance and operations with regard to non Class IE loads connected to this and associated systems to assure that minimum redundancy requirements for the reactor trip system are maintained. We will review the manner of coordinating the maintenance methods and procedures with operations for proper implementation of this requirement at the operating license stage of review.

Each diesel generator unit will be rated for continuous operation at 4500 kilowatts with margin in excess of design requirements. The applicants have not yet selected the diesel generators for the plant. However, to satisfy our requirements, they have stated that they will select diesel generator sets that have been previously qualified at the power level necessary for the South Texas Project. Furthermore, if the plant essential loads require a diesel generator set not previously qualified, the applicants have committed to qualification tests which demonstrate a 99 percent reliability of starting and accepting loads. In addition, the diesel generators will conform to the recommendations of Regulatory Guide 1.9 and with the requirements of IEEE 387-1972, "Criteria for Diesel Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations." We have concluded that these design commitments are acceptable.

Each diesel generator fuel supply will have sufficient capacity to permit its associated engine to operate at full rated load for at least seven days. The original plant layout proposed for the South Texas Project showed the fuel storage tanks in a separate structure remote from the diesel generator building. However, in a later amendment to the PSAR, these tanks have been relocated into the diesel generator building above the diesel generators. The fire potential of this building layout is discussed in Section 9.5.1 of this report. As discussed in Section 9.5.1 of this report, the applicants have recently submitted an analysis to demonstrate the adequacy of the diesel generator building design. We will review this information and report the results of our review in a supplement to the Safety Evaluation Report.

Each engineered safety features diesel generator will be located in a separate room of the diesel generator building. The associated Class IE electrical equipment will be so located and protected within the room as to minimize the possibility of damage due to internally generated missiles, pipe ruptures, and fires. No two trains of Class IE equipment or cables will be located in or routed through any of the other engineered safety features diesel generator rooms. Automatic and manual control of each of the engineered safety features diesel generators and associated engineered safety features equipment requiring automatic sequencing will be provided. Control power for the engineered safety features diesel generator systems is obtained from the associated engineered safety features 125 volt direct current systems. There is no electrical interconnection of redundant standby diesel generators.

On the basis of our review, and subject to the satisfactory resolution of the diesel generator building design as discussed above and interface items, we have concluded that the alternating current onsite power system satisfies the applicable criteria outlined in Section 8.1 of this report and is acceptable.

8.3.2 Direct Current Power Systems

Onsite direct current power will be provided by seven battery systems for each unit. Three non-class IE battery systems, consisting of two 125 volt direct current buses and one 250 volt direct current bus with two battery chargers and a battery for each bus, will provide power for the turbine generator auxiliary direct current loads and for switchyard control. There will be four safety-related Class IE 125 volt direct current battery systems per unit consisting of four independent and physically separated buses, each energized by two battery chargers and one battery. These batteries will provide emergency power for plant protection, control and emergency lighting when alternating current sources are unavailable. Each battery system will also supply power to an inverter which converts the direct current power to 120 volt alternating current power for vital instrumentation and protection systems.

The capacity of each battery will be sufficient, for a minimum of four hours, to provide power required by (1) emergency direct current controls and vital alternating current instrumentation, and (2) protection systems to shut down the reactor and maintain it in a safe shutdown condition until the supply of power from alternating current sources to the battery chargers is restored.

Each of the four 125 volt direct current Class IE batteries will be located in a separate room in a seismic Category I building. Battery chargers and distribution panels associated with a given battery will be located outside of the battery room. Each battery room will be ventilated through separate intake and exhaust ducts by fans which will be energized by the engineered safety features bus.

Subject to the satisfactory resolution of interface items, we have concluded that the direct current onsite power system satisfies the applicable criteria outlined in Section 8.1 of this report and is acceptable.

8.4 Physical Independence of Electric Systems

The applicants have provided criteria for physical separation of electrical equipment and circuits to preserve the independence of redundant equipment. In addition, they have stated that the design would meet the recommendations of Regulatory Guide 1.75.

We have reviewed the proposed design criteria and have concluded that the applicants' criteria for physical independence of electric systems will provide adequate assurance that no single event will negate the redundant safety features and are acceptable.

1547 101

9.0 AUXILIARY SYSTEMS

The auxiliary systems for the South Texas Project Units 1 and 2 are described in Section 9.0 of the PSAR. Those systems necessary to assure safe plant shutdown include the chemical and volume control system, the essential cooling water system, the component cooling water system, the reactor makeup water system, the fire protection system, the diesel generator fuel storage and transfer system, the diesel generator auxiliary systems, and the heating, ventilation, and air conditioning systems which serve the control room, electrical portion, and mechanical portion of the mechanical and electrical auxiliaries building, diesel generator building, and the central cooling water pump building.

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

We have reviewed those auxiliary 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 systems include the boron recycle system and the equipment and floor drain system.

We have also reviewed the design of other auxiliary systems whose failures would neither prevent safe shutdown nor result in potential radioactive release. These include the auxiliary cooling water system, the makeup demineralized water system, the compressed air system, and the turbine building heating, ventilation, and air conditioning system. We have determined that failure of these systems will not affect the capability of safety related systems to effect safe shutdown. On this basis, we have concluded that these systems are acceptable.

Our evaluation of the spent fuel pool cooling and cleanup system, the chemical and volume control system, and the boron recycle system is presented in Appendix A to this report.

9.1 Fuel Storage and Handling

South Texas Project Units 1 and 2 will each include a separate fuel handling building. The fuel handling buildings will be designed as seismic Category I structures and the design will provide tornado missile protection. Each building will house a new fuel area, spent fuel pool, shipping cask area, and fuel pool cooling and cleanup system. Each building will also house the safety injection pumps and containment spray pumps.

9.1.1 New Fuel Storage

The new fuel will be stored dry. The new fuel storage racks will be designed to store approximately one third of a core (66 new fuel assemblies), and will be designed to preclude the possibility of a fuel assembly being incorrectly placed. The racks will be designed with sufficient spacing to maintain an effective multiplication factor equal to or less than 0.95 even if completely flooded with non-borated water.

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

9.1.2 Spent Fuel Storage

Each fuel handling building will contain a spent fuel pool, providing storage for about one and two-thirds cores (322 fuel assemblies). In addition, a refueling pool located inside containment will provide storage for about one-fifth of a core. Both fuel pools will be of reinforced concrete construction with stainless steel liners, designed to seismic Category I requirements. The 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, and the uplift force 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 (See Section 9.1.4 of this report). The racks will be designed with sufficient center-to-center distance to maintain an effective multiplication factor equal to or less than 0.95 even if completely flooded with unborated water.

The capability to supply makeup to the pool will be provided by permanently installed connections from the demineralized water system, the reactor makeup system, and the refueling water storage tank. Of these, the reactor makeup water storage tank will provide assured seismic Category I makeup.

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 on seismic design, missile protection, design compatibility with the maximum crane loads that can travel over the pool, and availability of assured makeup systems, and are, therefore, acceptable.

9.1.3 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, are described in Section 9.1.4 of RESAR-41. Our evaluation of this portion of the fuel handling system is presented in Section 9.1.3 of Appendix A to this report. The equipment within the scope of the nuclear steam supply system includes a manipulator crane, spent fuel pool bridge, fuel transfer system, rod cluster control changing fixture, new and spent fuel handling tools, reactor vessel head and upper internals lifting device, reactor internals lifting device and the reactor vessel stud tensioner. The spent fuel cask handling system, the fuel handling building overhead crane, and the new fuel handling area overhead crane are within the scope of the balance-of-plant and are described in the PSAR.

The spent fuel cask handling system will consist of the following major components: the spent fuel cask transporter with cask tank, the inner and outer bellows seal assemblies, the transporter drive unit, and the spent fuel cask crane. This system is unique to the South Texas Project and the Allens Creek Nuclear Generating Station.

The cask will be placed in a water tight tank and positioned under the cask loading pool. A water tight connection will be made between the tank and a port at the bottom of the cask loading pool. The cask handling system will include redundant leakage barriers, each capable of retaining the water from the cask loading pool. During the cask loading operation the spent fuel cask will be held in position by four seismic Category I restraints. The spent fuel cask, bellows assemblies, cask adapter, and other components needed to form a leak tight envelope during spent fuel cask loading will be designed to seismic Category I requirements. The spent fuel cask loading area will be designed to be flood proof, and to contain any water, including ground water, which may enter it without leaking into adjacent portions of the building which contain safety related equipment (See Section 9.3.1 of this report).

The transporter will be designed such that a collision with the spent fuel pool wall will not occur assuming a single failure. Limit switches will be provided to de-energize the transporter drive unit to prevent such a collision. A snubber will be provided to prevent the collision should the limit switches fail. Thus, a cask handling system malfunction should not result in any fuel pool damage.

The spent fuel cask crane will be designed so that the cask vertical lift will be less than 30 feet above a restraining surface during any moving sequence. The crane will not be designed to accommodate safe shutdown earthquake loading without failure. The cask lift and travel over safety related systems (e.g., the fuel pool cooling system) will be physically precluded. Cask damage to the fuel pool will be precluded by adequate separation between the pool and the overhead crane system.

The fuel handling building overhead crane will be used for movement of new fuel assemblies, for removal of safety injection pumps, and for transporting the spent fuel shipping cask head. Crane design and building arrangement will preclude travel over the spent fuel pool. Since this crane will travel over the new fuel storage area, the crane design will accommodate safe shutdown induced dynamic loading such that the crane bridge or trolley will not fall into the new fuel storage pit as a result of the safe shutdown earthquake.

Based on our review we have concluded 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 are, therefore, acceptable.

9.2. Water Systems

9.2.1 Essential Cooling Water System

The essential cooling 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 essential cooling water system will recirculate water from the emergency cooling pond. The essential cooling water system will be completely separated from the auxiliary cooling water system. The auxiliary cooling water system will be utilized to provide cooling water to the non-safety related plant systems, and will recirculate water from the main cooling reservoir.

The essential cooling water system will consist of three separate and identical trains. Each train will contain one essential cooling water pump, component cooling heat exchanger, diesel generator heat exchanger, and miscellaneous heat exchangers required for operation of safety related systems. Each train will be powered by an independent engineered safety features bus. Normal operation will require utilization of one essential cooling water system train. Cooldown can be accomplished with two essential cooling water system trains. Two trains will be required to provide cooling for equipment required to be operable in the event of a design basis loss-of-coolant accident. The system will be designed as a seismic Category I system, and will meet the single failure criterion.

The essential cooling water intake structure will be common to both units and will be designed as a seismic Category I structure and for protection against tornado missiles and the design basis flood. Each essential cooling water pump will be located in a separate waterproof compartment within this intake structure. The separating walls between compartments will be designed to withstand internal missiles and will have a three-hour fire rating. The pump casings will be located in a sump beneath the pond bottom to assure adequate submergence. A seismic Category I discharge structure will be utilized for returning the essential cooling water to the pond.

Based on our review we have concluded that the essential cooling water 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 and to meet the single failure criterion, and Criteria 45 and 46 of the General Design Criteria as regards to system design that allows performance of periodic tests and inspections, including functional testing and confirmation of heat transfer capabilities. The essential cooling water system design criteria and bases also meet the RESAR-41 system requirements. We have concluded that the system is acceptable.

1547 105

9.2.2 Component Cooling Water System

The component cooling water system is a closed cooling water system which transfers heat to the essential cooling water system from components which process potential radioactive fluids. The component cooling water system will be designed to function during normal plant operation, safe cold shutdown, and postulated accident conditions. The component cooling water system will consist of three identical trains. Each train will contain a component cooling pump and heat exchanger, and will provide cooling for one residual heat removal train, one high head safety injection pump, one low head safety injection pump, one containment spray pump, one emergency boration pump, and one reactor containment fan cooler train. The other cooling loads, including the lines to the reactor coolant pumps, chemical and volume control system heat exchangers, and fuel pool coolers, will be supplied by a common header.

The safety related portion of the component cooling water system will be designed to seismic Category I requirements. The non-seismic portion of the system will be remotely isolable in the event of a malfunction. Assured makeup will be provided by the reactor makeup water storage system. Each component cooling water train will be powered by an independent engineered safety features bus. Normal operation will require utilization of one component cooling water system pump and heat exchanger.

Cooldown can be accomplished with two component cooling water system trains. Two trains will be required to provide cooling for the design basis loss-of-coolant accident.

The proposed system design will provide a single component cooling water system supply line and a single return line for all four reactor coolant pumps. The inlet line will contain one motor operated valve and one check valve for containment isolation. The return line will contain two motor operated containment isolation valves in series. Inadvertent closure of any one of the three motor operated valves would terminate the coolant flow for the reactor coolant pump seals and bearings. It is our position that this could result in reactor coolant pump failure without coastdown. This part of the system, therefore, does not meet the single failure criterion for a safety related function and is, therefore, not acceptable. We have advised the applicants that we require system redundancy so that a single active failure will not cause loss-of-coolant flow to the reactor coolant pumps. The applicants have conformed with our requirements and have incorporated dual parallel remote isolation valves in the component cooling water supply and discharge lines for the equipment. We have also determined that redundancy will be provided in the power supplies to these valves. Therefore, we have concluded that the design is acceptable.

We have concluded that the component cooling water 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 have further concluded that the system design criteria and bases meet the requirements of Criteria 45 and 46 of the General Design Criteria as regards to system design that allows performance of periodic inspections and tests, including functional testing and confirmation of heat transfer capabilities. The component cooling water system design criteria and bases also meet the RESAR-41 system requirements.

9.2.3 Ultimate Heat Sink

The emergency cooling pond will serve as the ultimate heat sink. The emergency cooling pond will be a seismic Category I pond with an approximate volume of 111,000,000 gallons. Further discussion of the ultimate heat sink is provided in Sections 2.4, 3.4, and 9.2.1 of this report.

The applicants' analysis of the ultimate heat sink is based on the assumption that one unit experiences a design basis loss-of-coolant accident while the other unit is placed in normal shutdown and cooldown. The essential cooling water is recirculated to the emergency cooling pond for a period of 30 days, assuming no makeup water is available. The applicants have submitted values for the heat rate and total integrated heat rejected due to fission product and heavy element decay heat, rejected heat from station auxiliary systems, containment sensible heat, and the summation of the above. We have reviewed these values and find them acceptable.

Based on these heat inputs and conservative meteorology, the applicants have calculated that the emergency cooling pond will contain enough water and heat dissipation capability to maintain both units in a safe shutdown condition for a period of 30 days. Assuming the most conservative recorded 30-day period with respect to meteorological conditions, the maximum intake temperature was calculated by the applicants as 116 degrees Fahrenheit assuming two units at a safe shutdown condition, and 115 degrees Fahrenheit assuming one unit is at a safe shutdown condition, and the other unit is in a condition following a design basis loss-of-coolant accident. Assuming a seepage loss of two cubic feet per second, the applicants calculated the total water loss for the 30-day period to be 63 percent. We have performed independent analyses, and concur with the applicants' results. We have concluded that the emergency cooling pond volume and heat dissipation capabilities are in accordance with Item 1 of Regulatory Guide 1.27, and are, therefore, acceptable.

The applicants have further demonstrated that the ultimate heat sink is in accordance with Item 2 of Regulatory Guide 1.27, with respect to the (1) capability of the ultimate heat sink to withstand the most severe natural phenomena expected taken individually, (2) site related events that historically have occurred or may occur during plant lifetime, (3) reasonably probable combinations of less severe natural phenomena and/or site related events, and (4) a single failure of man-made structural features.

Based on our review, we have concluded that the ultimate heat sink design criteria and bases comply with the recommendations of Regulatory Guide 1.27, and are, therefore, acceptable.

1547 107

9.2.4 Reactor Makeup Water System

The reactor makeup water system will provide nonborated makeup water for the chemical and volume control system and assured makeup for the component cooling water system and the spent fuel pool. The safety related part of the reactor makeup water systems is composed of one 150,000 gallon storage tank, pumps, valves, and piping designed to seismic Category I requirements. The system pump capacity will be 250 gallons per minute. We have concluded that the system storage and pumping capacities are acceptable.

Based on our review, we have concluded that the reactor makeup water system design criteria and bases can meet their designated safety function, and are, therefore, acceptable.

9.3 Process Auxiliaries

9.3.1 Equipment and Floor Drain System

The equipment and floor drain system will accommodate drains from the containment, auxiliary building, fuel handling building, turbine generator building, and diesel generator building. Drains from potentially radioactive sources will be processed in the liquid waste system. The portions of drain systems serving areas or compartments containing engineered safety features equipment will be separated from the rest of the system.

The high head and low head safety injection pumps and the containment spray pumps will be located in the fuel handling building, in three individual water tight compartments, one for each train. These compartments will be protected from flooding from other areas in the fuel handling building by the seismic Category I design of the building, including the fuel pool, the integrity of the fuel pool stainless steel liner, and separation from areas in which accidental major spills could occur, including the spent fuel cask loading area (see Section 9.1.3 of this report) and the fuel pool cooling system area.

Each auxiliary feedwater pump will be located in an individual watertight compartment. Major tanks within the auxiliary building, including the refueling water storage tank, reactor makeup water storage tank, volume control tank, and boric acid storage tanks, will be housed in watertight compartments which will retain the contents of the tank. Sumps and sump pumps will be located in various areas of the containment, auxiliary building, fuel handling building, and diesel generator building, including all compartments containing engineered safety features equipment.

The equipment and floor drain leak detection instrumentation serving engineered safety features equipment will be designed to seismic Category I requirements. Dual detectors with automatic high level sump pump starts will be provided for all safety related equipment. Except for the detection instrumentation, the engineered safety features and fuel handling building leak detection and drain system will be designed as non-seismic Category I. The sump pumps for the safety injection and containment spray pump compartments will be designed to accommodate the leakage from a pump seal failure.

The majority of the drain lines utilize dual check valves to prevent backflow from the systems, tanks, or sumps which collect the effluent. The inspection and maintenance accessibility of the check valves will be reviewed during the operating licensing stage of review.

Based on our review of the design, design criteria and bases of the equipment and floor drain system, we have concluded that this system can meet its designated safety function and is, therefore, acceptable.

9.4 Heating, Ventilation, and Air Conditioning Systems

9.4.1 Control Room and Electrical Auxiliary Building Heating, Ventilation and Air Conditioning System

The control room and electrical auxiliary building heating, ventilation, and air conditioning system will be designed to maintain the control room, essential switchgear rooms, cable spreading rooms, battery rooms, relay room, miscellaneous electrical equipment room, offices and living quarters inside the electrical auxiliary building, 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 control room heating, ventilation and air conditioning system will consist of three 50 percent capacity redundant trains, each powered by a separate engineered safety features bus. Normally two trains will be operating and the third will be kept on standby. Ventilation of the other rooms in the electric auxiliary building will be accomplished by exhausting air from the control room and recirculating it to the control room heating, ventilation, and air conditioning supply ducts. The criteria for protection of the systems against dynamic effects associated with postulated rupture of piping are discussed in Section 3.6.2 of this report. The system will be designed as seismic Category I.

The heating, ventilation, and air conditioning system will be designed to maintain the control room and the balance of the building under positive pressure. Two outside air intakes will be provided. To meet our requirements, the applicants have revised their design of these air intakes to withstand tornado missiles.

The two intakes will be cross-connected by an unisolable duct. Therefore, credit cannot be given for the dual intake feature. One air intake will be located only 50 feet away from the diesel fuel oil storage tank according to the proposed design. The adequacy of the diesel fuel oil storage tank location is discussed in Section 9.5.1 of this report. As stated in Amendment 13 of the PSAR, dual smoke detectors will be placed in the common inlet duct rather than the individual air intakes. We have informed the applicants that we will require demonstration that the transport time from the air intake to the detector location is compatible with system isolation capabilities. The applicants have recently provided this analysis. We will review this information and report the results of our review in a supplement to the Safety Evaluation Report.

In the event of a safety injection actuation, the makeup air will be automatically diverted to two of three 1000 cubic feet per minute makeup filtration units, and the recirculated control room air supply will be automatically diverted to two of three 5000 cubic feet per minute control room filtration units. Also, in the event of a safety injection signal, the fans that normally exhaust air from the battery rooms and living quarters will be tripped, reducing the required control room air makeup from 6000 cubic feet per minute to 2000 cubic feet per minute. The heating, ventilation, and air conditioning system will be designed to maintain the hydrogen concentration at less than three percent when the exhaust fans are turned off.

Three 50 percent capacity seismic Category I control room and electrical auxiliary building heating, ventilation and air conditioning chilled water trains will be provided to maintain the control room at a maximum temperature of 80 degrees Fahrenheit. The essential cooling water system will remove the heat from the chiller units.

Subject to the exception identified above, we have concluded that the system design criteria and bases are in conformance with the requirements of Criterion 19 of the General Design Criteria as regards the capability to operate the plant from the control room during normal and accident conditions discussed above. We have also concluded that the system's acceptability is subject to the satisfactory resolution of the diesel fuel oil fire concern discussed in Section 9.5.1 of this report.

9.4.2 Mechanical Auxiliary Building Heating, Ventilation, and Air Conditioning System

The mechanical auxiliary building heating, ventilation, and air conditioning system will be designed to provide suitable environmental conditions for personnel and equipment, maintain the building under negative pressure to minimize outleakage, and limit the concentration of airborne activity. Only the component cooling water pump supplementary coolers are safety related, and therefore are designed to seismic Category I requirements, and powered from independent engineered safety features buses to meet the single active failure criterion. The effectiveness of these units to maintain the component cooling water pump environmental design temperature is disccused in Section 9.2.2 of this report.

Based on our review of the design, design criteria and bases of the mechanical auxiliary heating, ventilation and air conditioning system, we have concluded that this system is acceptable.

9.4.3 Fuel Handling Building Heating, Ventilation, and Air Conditioning System

The fuel handling building heating, ventilation, and air conditioning system will be designed to control the fuel building thermal environment within acceptable design limits for personnel and equipment, to maintain the building under negative pressure, and to mitigate the consequences of a fuel handling accident by filtration of the exhaust air. The system will consist of the supply subsystem, the supplementary coolers subsystem and the exhaust subsystem.

The supplementary coolers subsystem will be designed to maintain the ambient design temperature of the safety injection pumps and containment spray pumps within their design value of 120 degrees Fahrenheit. This subsystem will be designed as a seismic Category I system. The supplementary coolers will be powered from independent engineered safety features buses and will meet the single failure criterion. The essential cooling water system will remove the heat load.

Based on our review, we have concluded that the supplementary coolers subsystem design criteria and bases are acceptable.

With respect to the fuel handling building exhaust subsystem, the applicants have proposed a filtration system to mitigate the consequences of postulated fuel handling accidents which meets the recommendations of Regulatory Guide 1.52, except that this system will be designed as a non-seismic Category I system. We have computed the radiological consequences of a fuel handling accident without taking credit for the iodine filtration of the fuel handling building exhaust subsystem. We find that a dose of 96 rem to the thyroid would result at the exclusion boundary for a two-hour period following the design basis accident. Because fuel handling accidents are not of extremely low probability and because the calculated potential doses are significant fractions of the 10 CFR Part 100 guideline values, we required, as discussed in Section 6.6 of this report, that a high quality system (engineered safety feature grade) be provided to mitigate the consequences of such an event.

In addition, the South Texas Project emergency core cooling system and containment spray system equipment area will be serviced by the fuel handling building exhaust subsystem. As discussed in Section 15.7.5 of this report, in order to mitigate the possible consequences of fission product leakage from these systems during the recirculation phase of a postulated loss-of-coolant accident, we required that the fuel handling building exhaust subsystem conform to the requirements of an engineered safety features system and meet all the recommendations of Regulatory Guide 1.52.

The quality assurance provisions associated with Regulatory Guide 1.52 and seismic Category I design are applicable to this system. In this regard, Item 2(C) of Regulatory Guide 1.52 states that, "all components of an engineered safety features atmosphere cleanup system should be designated as seismic Category I if failure of a component would lead to the release of significant quantities of fission products to the working or outdoor environments."

The applicants have orally agreed to design the fuel handling building exhaust subsystem in accordance with all the recommendations of Regulatory Guide 1.52. Subject to confirmatory documentation, we have concluded that the proposed design of this system is acceptable.

9.4.4 Diesel Generator Building Heating, Ventilation, and Air Conditioning System

The diesel generator building heating, ventilation, and air conditioning system will be designed to provide the diesel engines with the required combustion air, maintain the thermal environment inside the building within equipment design limits, and protect the diesel engine rooms from the adverse effects of fire eruption in the adjacent rooms. The system will consist of three individual trains with no interconnections. To meet the single active failure criterion, the system fans will be powered by the engineered safety features bus connected with the corresponding diesel engine. The system will be designed as a seismic Category I system, and will be protected from tornado missiles.

As discussed in Section 9.5.1 of this report, the applicants have not demonstrated the adequacy of the building design for the case of a postulated fuel oil fire. A specific area of concern regarding the diesel generator building heating, ventilation and air conditioning system is the absence of fire dampers. Therefore, we have concluded that the system design is unacceptable. The applicants have recently provided additional information regarding these fire dampers. We will reevaluate the adequacy of the heating, ventilation, and air conditioning system in conjunction with the evaluation of a postulated diesel fuel oil fire, discussed in Section 9.5.1. We will report the results of our evaluation in a supplement to the Safety Evaluaton Report.

9.4.5 Main Steam Valve Structure Ventilation System

The main steam valve structure ventilation system will be designed to maintain a thermal environment in the main steam valve and auxiliary feedwater pump area compatible with the component design criteria. Two redundant 100 percent capacity supply and exhaust fans will be provided, powered by separate engineered safety features buses. The system will meet the single active failure criterion, and will be designed as a seismic Category I system.

Based on our review of the design, design criteria and basis of the main steam valve structure ventilation system, we have concluded that this system can meet its designated safety function and is, therefore, acceptable.

9.4.6 Essential Cooling Water Pump Building Heating, Ventilation, and Air Conditioning System

The essential cooling water heating, ventilation, and air conditioning system will be designed to maintain the thermal environment in the essential cooling water pump building compatible with component design criteria. Each pump compartment will be furnished with two inlet and outlet ducts and two 50 percent capacity exhaust fans. The system motors will be powered by the same engineered safety features bus that powers the associated essential cooling water pumps. The system will be designed as seismic Category I, and will meet the single failure criterion.

Based on our review of the design, design criteria and bases of this system, we have concluded that the system can meet its designated safety function, and is, therefore, acceptable.

9.5 Other Auxiliary Systems

9.5.1 Fire Protection System

To minimize the fire potential throughout the plant, flame retardant materials of construction will be used for safety related areas. Within the electrical auxiliary building, safety related rooms will be protected by three-hour fire walls and floors. Openings in the fire walls and floors will be protected by fire doors and dampers. In general, cabling for each engineered safety features train will be run on a different level. The exception to this is the cabling within the cable spreading rooms and the power cable vault areas. For each of these, two trains will be located on one level and in one room, with the trains separated by either a barrier or by distance. The third train will be located in another level and room. Class IE electric equipment will be segregated from non-Class IE equipment. Cable insulation will be flame-retardant or noncombustible. Fire stops of non-combustible sealing material will be placed where cable runs pass through walls and floors, and at approximately 15-foot intervals on vertical cable runs. Combustible material storage control will be maintained in safety related areas, and in construction areas. Warehouses will be located remotely from safety related areas.

The fire protection system will be designed to provide automatic or manual fire extinguishing capability, to provide fire detection equipment in the plant essential areas, and to comply with the applicable standards of the National Fire Protection Association. The fire protection system will include a water system, an automatic carbon dioxide system, and a fire detection system.

The fire protection water system will be shared by South Texas Project Units 1 and 2. The system will utilize two 300,000 gallon water storage tanks and will include one 2500 gallons per minute motor driven pump and one 2500 gallons per minute diesel-engine driven pump. The water storage and pumping capacities are in accordance with Nuclear Engineering Property Insurance Association requirements.

The system will supply water to South Texas Project Units 1 and 2 by means of a fire loop. The safety related areas served by this system will be the diesel generator rooms (automatic pre-action sprinklers), the diesel generator fuel oil storage tank rooms (automatically actuated foam-water sprinkler system), and the mechanical and electrical auxiliary building, (pre-action standpipes and hose racks, with remote manual actuation). Fire mains will not be located in auxiliary building rooms containing engineered safety features equipment. Adequate drainage will be provided by the equipment and floor drain system to preclude flooding of engineered safety features equipment in the event of a fire protection water pipe rupture. The equipment within each diesel engine room will be designed to remain operable in the event of sprinkler actuation.

An automatically actuated total flooding carbon dioxide system will be provided for the electrical auxiliary building cable vault penetration areas, power cable vault areas, cable spreading rooms, and electrical cable chases. The carbon dioxide storage unit will be maintained at 300 pounds per square inch gauge and 0 degrees Fahrenheit. Approximately 50 percent concentration of carbon dioxide will be required to extinguish a fire in the above areas. Activation of any carbon dioxide subsystem will automatically close the fire dampers in the affected room to seal it. To protect personnel working in these areas, system lockout valves will be installed in the supply pipes, and there will also be a time delay between the sounding of an alarm and system actuation. These rooms will not contain any diodes, relays, or other electric equipment other than cabling.

We have advised the applicants that, since the fire protection system will not be designed as a seismic Category I system, a seismic event could result in an inadvertent release of carbon dioxide, which may result in incapacitating engineered safety features systems due to freezing or other adverse effects. The applicants have submitted vendor information and test data in order to indicate that cabling would not be adversely affected in the event of an inadvertent carbon dioxide actuation. We have examined this information and found it acceptable. Therefore, we have concluded that the system is acceptable.

The fire detection system will utilize fixed temperature heat detectors, rate-of-rise heat detectors, and ionization smoke detectors. A fire protection annunciator located in the control room will alarm upon actuation of a fire detector and actuation of any automatic extinguishing system.

The engineered safety features switchgear rooms, the engineered safety features portion of the relay rooms, the battery rooms and the essential cooling water pump rooms will be protected by portable fire extinguishers and nearby hose stations. With the exception of the essential cooling water pump rooms, the above areas will be located in the electrical auxiliary building and will thus be near manned areas. Each of these rooms will contain the components of only one train, and will be provided with three-hour fire walls and floors, to assure separation of redundant equipment. The fire potential for the essential cooling pump rooms will be low, and we, therefore, accept the fire protection for these rooms.

Since the fire protection of the switchgear, battery, and relay rooms requires entry and manual action by the operators, a detection system is required that covers the entire area and provides early warning. Separate fire detection systems with a minimum of two detectors will be provided for each of these rooms or areas. Failure of any detector will not affect the operability of any other detector. The ionization detectors utilized will be designed to detect small fires in a maximum of two minutes. On this basis, we have determined that the fire protection for these rooms is acceptable.

We have determined that the maximum credible fire that could affect safety related equipment would be a fire in a fuel oil storage tank room within the diesel generator building. In accordance with the present facility design, the diesel generator building will contain three emergency diesel generators in the lower level and three

60,000 gallon fuel oil tanks in the upper level. Each fuel oil storage tank will be located in an individual room with three-hour fire walls providing separation between rooms. The diesel building heating, ventilation and air conditioning system design does not presently show the utilization of fire dampers, and, therefore, the rooms will not be isolable from the outside in case of a fire.

The applicants have performed an analysis regarding the effects of a fire in one train on the combustion air for the diesel generators of the other two trains. The assumptions made in this analysis include storage tank rupture and an open heating, ventilation and air conditioning system. No credit was taken for the fire protection system. The applicants' conclusion was that the evolved combustion products would result in a reduction of the oxygen concentration in the diesel engine intake of less than four percent and would not affect the load carrying capabilities of the diesel generators. We have advised the applicants of our concerns regarding the possibility of a long lasting fire resulting in loss of building integrity and/or spread of the fire to the other fuel oil tanks and diesel generators in the building. Specific areas of concern include the effects of release of large quantities of soot on the operability of the diesel engines due to plugging of filters, the possibility of explosion, the possibility of spalling of the concrete walls, and floor collapse. We also advised the applicants that their assumptions for the maximum fire duration, maximum temperature and air flow rate were not necessarily conservative. Therefore, we have concluded that the diesel generator building design is not acceptable on the basis of the information reviewed to date.

The applicants have recently submitted an analysis that would consider each of our concerns and also consider certain design changes that may lessen the likelihood and effects of a fire in order to demonstrate the adequacy of the present building layout. We will review this information and report the result of our review in a supplement to the Safety Evaluation Report.

Based on our review, we have concluded that the fire protection system design criteria and bases are not in conformance with Criterion 3 of the General Design Criteria in the following areas: (1) the design of the safety related areas and systems to minimize the probability and effect of fire, and (2) design of fire fighting systems to assure that their rupture or inadvertent operation will not significantly impair the safety capability of structures and systems. The unsatisfactory area is the diesel fuel oil fire potential. Therefore, we have concluded that the proposed fire protection system design criteria and bases are not acceptable. As discussed above, we required the applicants to locate the diesel fuel oil tanks at a distance where a fire will not affect safety related equipment, and for which tornado protection is provided, or demonstrate that their present location is acceptable by an analysis which proves that the most severe diesel fuel oil tank fire will not incapacitate more than one diesel generator and thus allow safe shutdown. As stated previously, the applicants have recently provided additional information to justify their designs. We will review this information and report the results of our review in a supplement to the Safety Evaluation Report.

9.5.2. Diesel Generator Fuel Storage and Transfer System

The diesel generator fuel storage and transfer system will be designed to provide sufficient storage of fuel oil to allow operation of each emergency diesel generator for a minimum of seven days. The diesel generator fuel storage and transfer system will include a 60,000 gallon fuel oil storage tank and an engine driven pump. We have concuded that the system fuel oil storage capacity is adequate. The diesel generator fuel storage and transfer system will be designed as a seismic Category I system.

The PSAR originally stated that the tanks would be separated by a missile proof wall so that tornado missiles could not damage more than one tank. We advised the applicants that the individual storage tanks must be protected against tornado missiles. As a result, the applicants revised the location of the tanks and remainder of the system within the missile proof diesel generator building. The fire hazard concern for this layout is discussed in Section 9.5.1 of this report.

Based on our review, we have concluded that the system capacity and design criteria are acceptable. However, due to the fire hazard concern, the location of the diesel fuel oil tanks is not acceptable without further justification as discussed in Section 9.5.1 of this report.

9.5.3 Diesel Generator Auxiliary Systems

The diesel generator auxiliary systems will consist of the diesel generator closed cooling water system, the diesel generator starting system, and the diesel generator lubrication system.

The diesel generator closed cooling water system will be designed to remove the heat from the air coolers, lube oil coolers and engine water jackets. Three independent trains will be provided, one for each diesel generator. Heat removal will be provided by the essential cooling water system. The system will be designed as a seismic Category I system and will meet the single active failure criterion. The expansion tanks will be sized to allow system operation for at least seven days with no makeup.

Each diesel generator will be provided with two redundant compressed air starting trains, each consisting of an air compressor and receiver. Each air receiver will have sufficient capacity for five engine starts. We have concluded that the number of trains and air receiver capacity are acceptable. The starting system, with the exception of the air compressors, will be designed to seismic Category I requirements. The system will meet the single active failure criterion.

Each diesel generator system will be provided with a lubrication system designed as seismic Category I and will meet the single active failure criterion.

Based on our review of the design, design criteria and bases of the diesel generator auxiliary systems, we have concluded that these systems can meet their designed safety functions, and are, therefore, acceptable.

1547 116

10.0 STEAM AND POWER CONVERSION SYSTEMS

10.1 Summary Description

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 four 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.

10.2 Turbine-Generator

The turbine generator will be a tandem compound type with one double flow high pressure turbine and three low pressure turbines. 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 control the steam flow to the turbine.

The turbine control system will be designed to trip the turbine under the following conditions: turbine overspeed, condenser low vacuum, excessive thrust bearing wear, reactor trip, generator electric trip, low bearing oil pressure, low hydraulic fluid pressure, or manual trip.

The turbine generator will be provided with two overspeed protection systems: an electro-hydraulic control system and a mechanical overspeed protection system. The electro-hydraulic control system will rapidly close the governor and interceptor valves if 103 percent of rated speed is exceeded. If 111 percent of rated speed is reached the mechanical overspeed sensor will trip all steam valves (throttle, governor, reheat stop and interceptor valves) to maintain the speed below 120 percent of rated speed. As a backup, an electro-magnetic speed sensor (separate from the normal speed sensor) will also trip all valves at 111 percent of rated speed.

Based on our review of the design, design criteria and bases of the turbine-generator overspeed protection system, we have concluded that this system can meet its designated safety functions and is, therefore, acceptable.

10.3 Main Steam Supply System

The steam generated in each of four steam generators will be routed to the turbine through a main steam header by each of the four steam lines. Each main steam line will contain six safety valves, one air operated relief valve and one main steam isolation valve. The main steam supply system will be designed to seismic Category I requirements up to and including the main steam isolation valves.

1547 117

The main steam isolation valves will be designed to close within ten seconds after a major steam line break. Since the closure signal will reach the actuator within five seconds, the main steam isolation valves will be designed to close in five seconds upon receipt of a signal indicating low steam line pressure, high-high containment pressure or low primary loop cold leg temperature. The valves will be designed to close for the condition of the maximum mass flow rate in the event of a double-ended steam line break in either direction. Failure of one main steam isolation valve coincident with a steam line break will not result in uncontrolled flow from more than one steam generator, based on proposed design main steam isolation valve 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 assured by designing the cubicles that contain the relief valves and main steam isolation valves with access to the relief valves connected to the unaffected steam lines. The relief valves will then be manually operated to decrease primary and secondary plant pressure at a rate that is compatible with initiation of the residual heat removal system within eight hours after the accident.

Based on our review, we have concluded that the main steam supply system design criteria and bases are in conformance with the single failure criterion, the position related to seismic design of Regulatory Guide 1.29, and valve closure time requirements, and are, therefore, acceptable. As discussed in Section 3.6.2 of this report, the applicants will provide system layout drawings in late 1975.

10.4 Circulating Water System

The circulating water system will furnish the main steam condenser with cooling water from the reservoir at a flow rate of 907,000 gallons per minute.

We have reviewed the consequences of flooding as a result of failure of an expansion joint assuming that the circulating water pumps keep running for ten minutes after rupture occurs. Flooding would be detected by a high level alarm in the condenser pit. There is no safety related equipment in the turbine building, and no safety related equipment would be affected due to flooding of passageways, pipe chases or cableways at or below the maximum level reached inside the turbine building at the end of ten minutes after rupture.

Based on our review of the design, design criteria and bases of the circulating water system, we have concluded that this system can meet its designated safety function and is therefore, acceptable.

10.5 Auxiliary Feedwater System

The auxiliary feedwater system will be designed to supply water to the steam generators for sensible and decay heat removal when the main feedwater system is not available. The auxiliary feedwater system will be utilized during certain periods of normal

1547 118

1547 117

startup and shutdown, in the event of malfunctions such as loss of offsite power, and in the event of accidents.

The auxiliary feedwater system will be designed as a seismic Category I system and will be protected from tornado missiles. The system will consist of four 100 percent trains, one for each steam generator. Three of the trains will each include one 500 gallons per minute motor operated pump. Each pump will be powered from a separate alternating current engineered safety features bus. The fourth train will include a steam turbine driven 600 gallons per minute pump. The turbine steam supply will be provided by a line from one of the main steam lines upstream of the main steam isolation valve. The control valves for this train will be operated by direct current power. One train will be operable that is independent of alternating current power. Therefore, the required power diversity will be provided.

The system piping will be designed so that the design function of each train is independent of other trains when considering a piping failure, a component failure, a power supply or control malfunction. Crossover lines between trains will contain two normally closed, fail closed valves. Each auxiliary feedwater pump and associated piping will be located in an individual water tight compartment. The applicants have provided the results of an analysis that demonstrates that adequate decay heat removal will be obtained with a minimum of one pump and one steam generator. We have reviewed this analysis and concur with the applicants conclusions. The system will be designed to assure that two pumps and two steam generators will be available assuming the failure of a single active component concurrent with a high energy line failure.

Each auxiliary feedwater pump will automatically start in the case of a safety injection signal or loss of all main feedwater pumps. All motor driven pumps will automatically be started by the diesel generator automatic loading sequence signal. The turbine driven auxiliary feedwater pump will automatically start on loss of offsite power. Each auxiliary feedwater pump will automatically start by activation of two-out-of-three low-low level signals for its corresponding steam generator. Manual control will be possible both from the control room and the auxiliary shutdown panel.

The auxiliary feedwater will be supplied from the condensate storage tank, a 500,000 gallon capacity, concrete, stainless steel lined tank designed to seismic Category I requirements and protected from tornado missiles. This tank will also supply normal makeup to the condenser. All tank nozzles, with the exception of the auxiliary feedwater supply nozzles, will be at a sufficient elevation to assure that a minimum of 250,000 gallons reserve will be available for the auxiliary feedwater system. This quantity of water is sufficient to maintain the plant at hot shutdown for two hours, followed by cooldown at 50 degrees Fahrenheit per hour down to a condition at which the residual heat removal system can be initiated.

Events causing damage to the feedwater system piping such as that experienced at

Indian Point 2, and at other plants, could originate as a consequence of uncovering of the feedwater sparger in the steam generator or uncovering the steam generator feedwater inlet nozzles. Uncovering of the steam generator feedwater nozzles could cause a pressure wave that is propagated through the pipes. The applicants have agreed to provide additional information regarding the system design to demonstrate that unacceptable pressure wave propagation (water hammer) such as that experienced at Indian Point 2 would not result at the South Texas Project.

Based on our review, we have concluded that the system design criteria and bases are in accordance with our positions including diversity of power sources, system flexibility, and redundancy including the combination of single active and high energy line failures. The system design criteria and bases also meet the RESAR-41 requirements regarding minimum delivered flow rate, pump head, and actuation logic. We have concluded that the system design criteria and bases are acceptable.

10.6 Material Considerations

We have reviewed the mechanical properties of materials selected for Class 2 and 3 components of the steam and feedwater systems. We have determined that the mechanical properties satisy Appendix I of Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, or Parts B and C of Section II of the ASME Boiler and Pressure Vessel Code. We have also determined that the fracture toughness properties of the ferritic materials will satisfy the requirements of the ASME code, as supplemented by our requirements.

The controls imposed upon the austenitic stainless steel are in conformance with 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 design life of the plant. The controls 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.

The welding procedures that will be used in limited access areas satisfy the recommendations of Regulatory Guide 1.71. The onsite cleaning and cleanliness controls during fabrication satisfy recommendations given in Regulatory Guide 1.37, and the requirements of American National Standard Institute Standard N45.2.1-1973, "Cleaning of Fluid Systems and Associated Components for 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.

We have concluded that conformance with the codes, standards, and applicable Regulatory Guides constitutes an acceptable basis for assuring the integrity of the 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

The radioactive waste (radwaste) systems will consist of the liquid waste system, the gaseous waste system, and the solid waste system. The liquid waste system will process waste liquid streams such as equipment drains, coolant leakage, condensate demineralizer regenerant liquids, decontamination and laboratory waste liquids, and laundry and shower waste water. The treated liquid waste will be recycled for reuse if the plant water balance requires makeup and if the water quality is adequate. The liquid waste system will utilize evaporation, demineralization, and filtration for removal of radioactive material, chemical impurities, and particulates.

Gaseous wastes will be generated during the operation of the plant from degassing primary coolant, from vents for equipment handling radioactive materials, and due to leakage from systems and components containing radioactive material. The gaseous waste system will treat gaseous streams for radioactive material removal by filtration, adsorption, and holdup for radioactivity decay. The treated gas streams will be released to the environment.

Solid wastes will be generated during plant operation. The wastes will consist of waste materials such as contaminated clothing, evaporator bottoms, demineralizer resins and discarded radioactive components and tools. Treatment will consist of solidification, packaging, and shipping to a licensed burial site.

South Texas Project Units 1 and 2 will have separate waste management systems.

In the Final Environmental Statement dated March 1975 for South Texas Project Units 1 and 2, we performed an evaluation to determine the quantities of radioactive materials that will be released in the liquid and gaseous plant effluents, and that will be shipped offsite as solid wastes for burial. In that evaluation we considered waste flows, waste activities, and equipment operating performance that are consistent with normal plant operation, including anticipated operational occurrences over the life of the facilities. In Amendment 12 to the PSAR, the applicants proposed design modification to the main condenser mechanical pump gaseous treatment system subsequent to our evaluation in the Final Environmental Statement. The modification removed the high efficiency particulate air filters and charcoal adsorbers from the vacuum pump exhaust and our evaluation considers this modification.

The parameters used in the Final Environmental Statement evaluation, along with their bases, are given in Appendix B to WASH-1258, "Final Environmental Statement Concerning Proposed Rule Making 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." Modified versions of the ORIGEN and STEFFEG Codes, which were the liquid and gaseous calculational models used in our evaluation, are given in Appendix C to WASH-1258.

Our evaluation presented below was performed in accordance with the design objectives of our report "Concluding Statement of Position of the Regulatory Staff" Docket No. RM-50-2 dated February 20, 1974. We have not completed our review of the radioactive waste systems to meet the dose design objectives of Appendix I to 10 CFR Part 50 (effective June 4, 1975) and the required cost-benefit analysis. We will report the results of our review in a supplement to the Safety Evaluation Report.

11.2 Liquid Waste Treatment Systems

The liquid waste treatment system will consist of three subsystems: (1) the liquid waste processing system, (2) the turbine building drains, and (3) the condensate polishing regeneration waste system. In addition to these subsystems, the Boron Recovery System is considered in our evaluation.

The liquid radwaste system will be designed to collect and process wastes based on the chemical purity, relative to the primary coolant, as determined by the origin of the waste in the plant. The boron recovery system will process shim bleed and equipment drain waste, collected inside the reactor containment, by means of evaporation and demineralization. The liquid waste processing system recycle portion will process equipment drain wastes and tank overflow wastes, from components outside reactor containment, by evaporation and demineralization. The liquid waste processing system's waste portion will also process detergent wastes should radiation measurements indicate higher than expected radioactivity levels. Detergent wastes and turbine building floor drain wastes will normally be released without treatment after monitoring for radioactivity. The liquid waste processing system will also process the condensate polishing regeneration waste resulting from regeneration of the secondary loop volatile chemistry condensate demineralizers. All steam generator blowdown will be recycled. The waste management systems and the condensate polishing regeneration waste system will be separate for each unit. The principal components making up each of these systems, along with their principal design criteria, are listed in Table 11.1.

In our evaluation of the liquid radwaste system we have considered (1) the system's capability to reduce radioactive releases to "as low as practicable" levels based on expected radwaste inputs over the life of the plant, (2) the system's capability to maintain releases below the limits in 10 CFR Part 20, Appendix B, Table 2, Column II during periods of fission product leakage at design levels from the fuel, (3) the system's capability to meet the processing demands of the station during anticipated operational occurrences, (4) the quality group classification and seismic category applied to the system design, and (5) the design features incorporated to preclude uncontrolled releases of radioactive materials due to tank overflows. The process and effluent monitoring design capabilities are considered in Section 11.5 of this report.

TABLE 11.1

DESIGN PARAMETERS OF PRINCIPAL COMPONENTS CONSIDERED IN LIQUID RADWASTE EVALUATION

Component	Number Per Unit	Capacity, each	Quality Group*
Boron Recovery System			
Recycle Holdup Tanks	2	75,000 gallons	C
Evaporator Feed Demineralizer	2	120 gallons per minute (30 cubic feet of resin)	C
Evaporator Condensate Demineralizer	1	60 gallons per minute (20 cubic feet of resin)	D (Augmented)**
Evaporator	1	15 gallons per minute	C
Liquid Waste Processing System			
Waste Holdup Tank	1	10,000 gallons	D (Augmented)
Recycle Evaporator Condensate Tank	2	5,000 gallons	D (Augmented)
Waste Evaporator Condensate Demineralizer	1	35 gallons per minute (30 cubic feet of resin)	D (Augmented)
Recycle Evaporator Condensate Demineralizer	1	35 gallons per minute (30 cubic feet of resin)	D (Augmented)
Waste Evaporator	1	10 gallons per minute	D (Augmented)
Recycle Evaporator	1	30 gallons per minute	D (Augmented)
Chemical Drain Tank	1	600 gallons	D (Augmented)
Laundry Tank	1	10,000 gallons	D (Augmented)
Floor Drain Tank	1	10,000 gallons	D (Augmented)
Waste Monitor Tanks	3	5,000 gallons	D (Augmented)
Condensate Polishing Regeneration Waste System			
Collection Tank	1	15,000 gallons	D (Augmented)
Miscellaneous			
Spent Resin Storage Tank	1	350 cubic feet	D (Augmented)

*Quality Group C components will be of seismic Category I design and Quality Group D (Augmented) components will be of non-seismic design.

**Quality Group D (Augmented) components will be designed to meet the quality assurance provisions of the staff's technical position presented in Attachment 010-2 to the January 10, 1975 letter from Mr. D. B. Vassallo to Mr. G. W. Oprea, Jr.

Our evaluation of the liquid radwaste treatment system for normal operation is given in the Final Environmental Statement for the South Texas Project. In the Final Environmental Statement, we have determined that the proposed liquid radwaste treatment systems will be capable of reducing the release of radioactive materials in liquid effluents to approximately 0.5 Curies per year per reactor, excluding tritium and dissolved gases, and 350 Curies per year per reactor for tritium. An isotopic listing of our calculated liquid source term is given in Table 3.6 of the Final Environmental Statement.

Based on that evaluation, we have found that the release of radioactive materials in liquid effluents will not result in whole body or critical organ doses in excess of 5 millirem per hour at or beyond the site boundary, and radioactive materials released in liquid effluents, exclusive of tritium and dissolved gases, will not exceed five Curies per year per reactor.

We have reviewed the effects of reactor operation with one percent of the operating fission product inventory in the core being released to the primary coolant. We have determined that under these conditions, the concentrations of radioactive materials in liquid effluents will be a small fraction of the limits in 10 CFR Part 20, Table 2, Column II.

The design capacities of the evaporators for the boron recovery system and the recycle and waste portions of the liquid waste processing system are approximately 21,000 gallons per day, 42,000 gallons per day and 14,000 gallons per day, respectively. We calculate the average expected waste flow to the boron recovery system and the recycle and liquids waste processing systems to be 3000 gallons per day, 760 gallons per day, and 610 gallons per day, respectively. The difference between the expected flows and design capacity provides adequate reserve for processing surge flows. The system design will allow wastes to be processed interchangeably between the three systems in the event of equipment downtime. We consider the system capacity and system design to be adequate for meeting the demands of the South Texas Project during anticipated operational occurrences.

The liquid radwaste systems will be located in a seismic Category I structure. The system components which could result in radionuclide concentrations in excess of the limits in 10 CFR Part 20, in the nearest potable water supply or at the nearest surface water supply will be designed to Quality Group C standards. The remainder of the system will be designed to Quality Group D (Augmented) standards in accordance with the quality assurance provisions of the staff's technical position presented in Attachment 010-2 to the staff's January 10, 1975 letter from D. B. Vassallo to G. W. Oprea, Jr. The quality group designations of the equipment are listed in Table 11.1. We find the applicants' proposed system design to be acceptable. The system will also be designed to preclude the uncontrolled 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.

The liquid radwaste system includes the equipment and instrumentation to control the release of radioactive materials in liquid effluents. The scope of our review included the system's capability to reduce releases of radioactive materials in liquid effluents to "as low as practicable" levels in accordance with 10 CFR Parts 20 and 50.36a, considering normal operation and anticipated operational occurrences, the design provisions incorporated to preclude uncontrolled release of radioactive materials in liquids due to leakage or overflows in accordance with Criterion 60 of the General Design Criteria, and the quality group and seismic design criteria.

Our review has included an evaluation of effluent releases based on the proposed treatment processes. Included in the review were piping and instrumentation diagrams, schematic diagrams, and descriptive information from the PSAR.

We have determined that the applicants' designs, design criteria, and design bases for the liquid radwaste system conform with the design objectives of our report "Concluding Statement of Position of the Regulatory Staff" Docket No. RM-50-2 dated February 20, 1974.

We have not completed our review of the liquid waste treatment system to meet the dose design objectives of Appendix I to 10 CFR Part 50 (effective June 4, 1975) and the required cost-benefit analysis. We will report the results of our review in a supplement to the Safety Evaluation Report.

11.3 Gaseous Waste Systems

The gaseous radwaste treatment system will be designed to process wastes based on the origin of the wastes in the plant and their expected activity levels. The gaseous waste processing system will process gases stripped from the primary coolant by means of a chilled water cooler, air dryers and charcoal decay beds. Each reactor will have a separate gaseous waste processing system. Radioactive gases from the main condenser vacuum pump exhaust will not be treated prior to release. Ventilation exhausts from the fuel handling building will be processed through high efficiency particulate air filters and charcoal adsorbers prior to release. The containment building atmosphere will be recirculated through filters and charcoal adsorbers prior to purging. The turbine building ventilation exhausts will not be treated prior to release. Ventilation air from the auxiliary building laboratory and sample areas will be processed through high efficiency particulate air filters prior to release. The principal components in the gaseous waste processing system, along with their principal design criteria, are listed in Table 11.2.

In our evaluation of the gaseous radwaste system we have considered (1) the system's capability to reduce radioactive releases to "as low as practicable" levels based on expected gaseous waste inputs and radioactive leakage rates over the life of the plant, (2) the system's capability to maintain releases below the limits in 10 CFR

TABLE 11.2

DESIGN PARAMETERS OF PRINCIPAL COMPONENTS CONSIDERED IN GASEOUS RADWASTE EVALUATION

Component	Number Per Unit	Capacity, each	Quality Group**
Gaseous Waste Processing System			
Charcoal Decay Tanks*	4	145 cubic feet (3500 pounds of charcoal)	D (Augmented)
Decay Tanks (shutdown)*	2	600 cubic feet	D (Augmented)
Dryer (Twin Unit)*	1	7 standard cubic feet per minute	D (Augmented)
Compressor*	1	7 standard cubic feet per minute	D (Augmented)

*Design Pressure - 150 pound per square inch gauge

**Quality Group D (Augmented) components will be of non-seismic design.

***Quality Group D (Augmented) components will be designed to meet the quality assurance provisions of the staff's technical position presented in Attachment 010-2 to the January 10, 1975 letter from Mr. D. B. Vassallo to Mr. G. W. Oprea, Jr.

1547 126

Part 20, Appendix B, Table 2, Column I during periods of fission product leakage at design levels from the fuel, (3) the system's capabilities to meet the processing demands of the facility during anticipated operational occurrences, (4) the quality group classification and seismic category applied to the system design, and (5) the potential for gaseous releases due to hydrogen explosions. The process and effluent monitoring design capabilities are considered in Section 11.5 of this report.

Our evaluation of the gaseous radwaste treatment system for normal operation is given in the Final Environmental Statement. In the Final Environmental Statement 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 1550 Curies per year per reactor of noble gases and 0.17 Curies per year per reactor of iodine 131. An isotopic listing of calculated gaseous source term is given in Table 3.7 of the Final Environmental Statement. Based on that evaluation, the release of radioactive materials in gaseous effluents will not result in an annual air dose, at or beyond the minimum exclusion distance, in excess of 10 millirad for gamma radiation and 20 millirad for beta radiation, the annual thyroid dose to an individual will not exceed 15 millirem considering the location of the nearest cow, seven miles east of the reactor complex and the annual quantity of iodine-131 released will not exceed one Curie for each reactor at the site.

We have reviewed the effects of reactor operation with one percent of the operating fission product inventory in the core being released to the primary coolant. We have determined that under these conditions the concentrations of radioactive materials in gaseous effluents will be a small fraction of the limits in 10 CFR Part 20.

The gaseous waste processing system will process hydrogen-nitrogen gas mixtures with trace quantities of fission gases from the primary coolant. These gases will be processed through a chiller, a twin dryer package and four (145 cubic feet each; design pressure, 150 pounds per square inch gauge) charcoal adsorber beds containing a total of seven tons of charcoal. One dryer bed may be regenerated while the other is in operation to provide a design dew point of less than 0 degrees Fahrenheit. In the Final Environmental Statement, we have determined that holdup time provided by the charcoal beds will be approximately 69 days for xenon and four days for krypton radionuclides prior to release or recycle to the volume control tank. The system also includes a compressor and two storage tanks (600 cubic feet each; design pressure, 150 pounds per square inch gauge) for shutdown use. These tanks contain nitrogen for use in the final stages of primary system degassing during shutdowns. This gas may be recycled or processed through the charcoal adsorber beds to maintain a constant gas inventory. The effect of back-to-back plant shutdowns on the gaseous holdup time will be minor. Waste gas releases from the gaseous waste processing system will be filtered through a high efficiency particulate air filter on the discharge of the charcoal decay beds. We consider the system capacity and the system design to be adequate for meeting the demands of the station during anticipated operational occurrences.

The gaseous waste processing system will be located in a seismic Category I structure. The gaseous waste processing system will be designed to quality group classification and seismic category consistent with the quality assurance provisions of the staff's technical position presented in Attachment 010-2 to the staff's January 10, 1975 letter from D. B. Vassallo to G. W. Oprea, Jr.. We find the design criteria acceptable. The quality group design criteria and seismic category are listed in Table 11.2.

Gaseous waste from the main condenser will not be treated prior to release according to the applicants' design modification in Amendment 12 to the PSAR. We calculate Iodine-131 and Iodine-133 release rates of 0.09 and 0.06 Curies per year per reactor, respectively, for the mechanical vacuum pump releases. The system releases will be proportional to the rate of primary to secondary system leakage and the primary coolant activity. In the event of excessive primary to secondary leakage, the affected steam generator will be isolated before radioactive material concentrations in the main condenser offgas releases exceed the limits in 10 CFR Part 20.

The auxiliary building will be ventilated with a once-through system, and the ventilation air from the laboratory and sample areas will be exhausted through high efficiency particulate air filters to the environment.

Ventilation air from the fuel handling building will be exhausted through high efficiency particulate air filters and characoal adsorbers. The containment building will be purged without treatment during release. The containment building will be provided with a recirculation system with a capacity of 20,000 standard cubic feet per minute, containing high efficiency particulate air filters and charcoal adsorbers. The turbine building ventilation exhausts will not be treated. The charcoal adsorbers used in the ventilation and recirculation systems will provide a decontamination factor of 10 for radioiodine, and the recirculation system for the containment building will provide a decontamination factor of approximately 80 for radioiodine after 16 hours of operation. The charcoal adsorbers will not reduce the noble gas activities.

The plant ventilation systems will be designed to induce air flows from potentially less radioactively contaminated areas to areas having a greater potential for radioactive contamination. Potentially contaminated building areas will be maintained at a slightly negative pressure with respect to the exterior pressure to promote collection of radioactive materials by the ventilation system and allow dispersion through roof and plant vent exhausts while reducing exfiltration. The ventilation system will have adequate capacity to limit radioactive material concentrations in areas within the plant that are accessible during operation to below the limits in 10 CFR Part 20.

The gaseous radwaste system includes the equipment and instrumentation to control the release of radioactive materials in gaseous effluents. The scope of our review included the system's capability to reduce releases of radioactive materials in gaseous effluents to "as low as practicable" levels in accordance with 10 CFR Parts 20 and 50.36a considering normal operation and anticipated operational occurrences, the quality group and seismic design criteria and the design provisions incorporated to reduce the potential for hydrogen explosions. The review has included an evaluation of effluent releases based on the proposed treatment processes and considering pathways due to process vents and due to leakage affecting building ventilation systems. Included in the review were piping and instrumentation diagrams, schematic diagrams, and descriptive information from the PSAR.

We have determined that the applicants' designs, design criteria, and design bases for the gaseous radwaste system conform with the design objectives of our report "Concluding Statement of Position of the Regulatory Staff" Docket No. RM-50-2 dated February 20, 1974.

We have not completed our review of the gaseous radwaste system to meet the dose design objectives of Appendix I to 10 CFR Part 50 (effective June 4, 1975) and the required cost-benefit analysis. We will report the results of our review in a supplement to the Safety Evaluation Report.

11.4 Solid Waste System

The 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 cement-vermiculite mixture to form a solid matrix and sealed in 55-gallon drums. 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 drums using an industrial baling machine. Each reactor will have its separate solid waste system.

In our evaluation of the solid radwaste treatment system we have considered (1) the system design objectives in terms of expected types, volumes, and activities of wastes processed for shipment offsite, (2) the design capacities of system components, method of operation, and capability of meeting the demands of the station due to anticipated operational occurrences, (3) waste packaging and conformance to applicable Federal packaging regulations, (4) provisions for controlling potentially radioactive airborne dusts during baling operations, (5) seismic design and quality group classification, and (6) provisions for onsite storage prior to shipping.

Our evaluation of the solid radwaste treatment system for normal operation is given in the Final Environmental Statement. In the Final Environmental Statement we determined that the expected solid waste volumes and activities shipped offsite will be 600 drums per year per reactor of wet solid waste containing an average of 10 Curies per drum and 350 drums per year per reactor of dry solid waste containing less than 5 Curies total. Storage facilities to accommodate approximately 600 drums will be provided within the auxiliary building.

Wastes will be packaged in 17H 55-gallon steel drums that meet Department of Transportation requirements and shipped to a licensed burial site in accordance with Nuclear Regulatory Commission and Department of Transportation regulations. Drum shields consisting of a two-piece cylindrical lead shield with steel jackets will be used, as required, to maintain a contact dose rate at the container (shield) surface of less than 10 milliroentgen per hour. We consider the packaging provisions to meet the requirements of 10 CFR Part 20, 10 CFR Part 71 "Packaging of Radio-active Material for Transport and Transportation of Radioactive Material Under Certain Conditions", and 49 CFR Parts 170-178. Dry wastes will be compacted using an industrial hydraulic baler. During compaction, drums will be enclosed in a dust shroud which will be vented to the plant vent to preclude releasing dusts to the operating area. An interlock to prevent operation of the baler with the shroud door open will be incorporated in the system design. The components containing spent resins, filter sludges, and evaporator bottoms are considered in the liquid waste system review in Section 11.2 of this report.

Drums will be evacuated by means of a vacuum pump. The drum will be isolated from the pump and the drum vacuum will be used to draw wastes into the drums for filling. This will preclude overfilling drums and prevent radioactive spills. Waste transfer piping will be designed to Quality Group D and non-seismic Category I standards. Since the quantity of radioactive materials in the piping will not have a significant potential for uncontrolled release to the environs, we consider this design to be acceptable.

The solid radwaste system includes the equipment and instrumentation for solidifying and packaging radioactive wastes prior to shipment offsite for burial. The scope of our review included the system's capability of processing the types and volumes of wastes expected during normal operation and anticipated operational occurrences in accordance with Criterion 60 of the General Design Criteria, the provisions for handling wastes with regard to the requirements of 10 CFR Parts 20 and 71, 49 CFR Parts 170-178, and the quality group classification and seismic design criteria.

Our review has included the provisions for controlling airborne dusts during dry waste compaction. Included in the review were piping and instrumentation diagrams, and descriptive information from the PSAR.

The basis for acceptance in our review has been conformance of the applicants' designs, design criteria, and design bases for the solid radwaste system to the Commission's Regulations, as referenced above, as well as staff technical positions and industry standards.

Based on the foregoing evaluation, we have concluded that the proposed solid radwaste system is acceptable.

11.5 Process and Effluent Radiological Monitoring

The process and effluent radiological monitoring system will be designed to provide information concerning radioactivity levels in systems throughout the plant, indicate radioactive leakage between systems, indicate equipment performance, and monitor and control radioactivity levels in plant discharges to the environs. Scintillation detectors will be used for particulate monitoring in gaseous effluents and for monitoring liquids. Geiger-Mueller detectors will be used for particulate monitoring of radioactive gases in vent effluents. Gaseous iodine will be collected on replaceable, impregnated adsorbers which will be continuously monitored while in use by scintillation detectors and counted weekly in the plant laboratory. Systems which are not amenable to continuous monitoring or for which detailed isotopic analyses are required will be periodically sampled and analyzed in the plant laboratory.

Tables 11.4-1, 11.4-2 and Section 9.3.2 of the PSAR indicate the proposed location and type of continuous monitors and the proposed sampling locations, sampling frequencies, and parameters measured. Monitors on the containment vent, plant vent, turbine building floor drain discharge line, and the liquid waste effluent line will automatically terminate discharges should radiation levels exceed a predetermined value.

In our evaluation of the process and effluent monitoring system, 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 material leakage between systems.

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 intermittent sampling locations listed in Table 11.4-1 and 11.4-2 of the PSAR, we have concluded that all normal and potential release pathways, excluding the turbine building vent, will be monitored. Due to the high potential for exfiltration from the turbine building which is a relatively open structure, we do not require monitoring of the low level gaseous releases from the turbine building.

The design will include provisions for automatically terminating effluent releases in the event radiation levels in discharge lines exceed a predetermined level. 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 the monitoring and sampling provisions to meet the requirements of Criteria 13, 60 and 64 of the General Design Criteria and the guidelines of Regulatory Guide 1.21.

The provisions for process and effluent radiological monitoring will include the instrumentation and controls for monitoring and controlling the releases of radioactive materials in plant effluents and monitoring the level of radioactivity in process streams. The scope of our review included the provisions for monitoring and controlling the release of radioactive materials in plant effluents in process streams. The scope of our review included the provisions for monitoring and controlling the release of radioactive materials in plant effluents in accordance with Criteria 60 and 64 of the General Design Criteria and Regulatory Guide 1.21, and for monitoring radioactivity levels within the plant in process streams in accordance with Criterion 13 of the General Design Criteria.

The basis for acceptance in our review has been conformance of the applicants' design, design criteria, and design bases for the process and effluent monitoring systems to the Commission's Regulations as set forth in the General Design Criteria and to applicable Regulatory Guides, as referenced above, as well as staff technical positions and industry standards.

Based on the foregoing evaluation, we have concluded that the proposed provisions for monitoring process and effluent streams are acceptable.

1547 132

12.0 RADIATION PROTECTION

12.1 Shielding

We have evaluated the proposed radiation protection program as described in the PSAR for the South Texas Project Units 1 and 2. The design objectives for the facility shielding will be to assure that radiation exposure to operating personnel will be within the required limits of 10 CFR Part 20 and 10 CFR Part 50, and that radiation exposures to operating personnel during refueling, maintenance, inservice inspections and other plant activities will be maintained as low as practicable. In conformance with Regulatory Guide 8.8, the applicants have committed to design the plant in such a manner as to maintain occupational radiation exposure to as low as practicable, and, in response to our requests for additional information, to having the facility design reviewed by a competent radiation protection specialist.

The facility design will include measures for reducing the need for maintenance of equipment and measures for reducing radiation levels and the time spent in radiation zones when maintenance is required in accordance with the recommendations of Regulatory Guide 8.8. These measures will include such features as careful selection of component materials for use in radiation zones, the enclosure in separate cubicles of components and support equipment such as pumps, and the use of thresholds to openings and special surfaces to control contamination and facilitate decontamination. Also, the applicants stated plans for permanent provisions for access to the steam generators, coolant pumps, and pressurizer for maintenance and inservice inspection. The design will include a system of radiation dose rate zones and access controls which are in compliance with 10 CFR Part 20. The radiation dose rate zones and access controls are described in Section 12.1 of the Preliminary Safety Analysis Report.

Calculations of source terms to be used for shielding design are based on (1) a core power of 3800 megawatts thermal, (2) a failed fuel rate of one percent, (3) a worst case choice of normal operating conditions, shutdown conditions, or design bases conditions, and (4) a set of estimated leakage rates and partition factors. These bases are maximum design conditions for shielding in accordance with the recommendations of Regulatory Guides 8.8 and 1.42. We have made calculations to confirm these shielding design source terms and we find the source terms to be acceptable.

Using the source terms discussed above, the radiation shielding will be designed to meet the shielding requirements of the radiation dose rate zones as defined by the applicants. Shield wall thickness calculations have been based on basic accepted shielding data and equations obtained from such references as the "Reactor Shielding Design Manual," edited by Theodore Rockwell III. In most cases, the computer shielding codes used were (1) QAD-SQ, a point kernel shielding program, (2) ANISN2, a one-dimensional multigroup code, and (3) CYLDOSE, a simplified code. Using the same basic references and source terms as the applicants, we performed calculations to confirm the shield wall thickness, with the SDC computer code (a simplified code based on the "Reactor Shielding Design Manual.") Based on our calculations, we find the shield wall thicknesses to be acceptable. The applicants have also committed to construct the shield walls in accordance with the recommendations of Regulatory Guide 1.69.

The applicants described several design objectives of the area radiation monitoring system including the objective of warning plant operators of unauthorized or inadvertent movement of radioactive material in the plant, and the indication of a substantial increase in radiation levels at all points where such an increase might be of immediate importance to personnel frequenting or working in the area. These design objectives lead to three criteria for selection of areas for placement of area radiation monitors: (1) areas where personnel perform regular duties in radiation fields; (2) areas where personnel perform infrequent duties, but where there is a high probability that significant changes in radiation levels could occur; and (3) areas where surveillance is desired. We find these design objectives and placement criteria to be acceptable. Area radiation monitors will be provided in 18 locations within the facility. We have determined that the objectives and criteria are met by the proposed radiation monitoring system, and the necessary monitors will be located in conformance with 10 CFR Part 50 and 10 CFR Part 70. We, therefore, find the location of the area radiation monitors to be acceptable. Acceptable design features of the above monitoring system will include ease of access for maintenance and calibration, audible local alarms for detector failure and high radiation, variable alarm setpoints, and five decade dose rate range coverage.

The applicants also described numerous practices to be used in operating the plant in such a manner as to maintain occupational radiation exposure to as low as practicable. They also indicate that these as low as practicable practices will be included in the plant radiation protection manual and will be updated constantly to reflect operating experience. These practices will include such important measures as draining and flushing components before maintenance, pre-job planning and mock-ups, efficient use of manpower, adequate supervision, transfer of components to be maintenanced to lower radiation fields, and the use of man-rem goals. We have determined that these procedural features are in accordance with the recommendations of Regulatory Guide 8.8 and are, therefore, acceptable.

Estimates of exposure to the plant personnel and to the construction force during the period when Unit 1 is in operation and Unit 2 is under construction were made. Included in the plant exposure estimates are routine patrols, tests, operations, and jobs occurring more than once-per-year; control room operations; and refueling. Doses were estimated to be 104.4 man-rem per unit per year to plant personnel from these sources. It is our position that such man-rem estimates should be based on operating experience and as low as practicable improvements in design and operating procedure, and should include expected dose rates, plant manpower, occupancy times, and the jobs that are expected to be performed. The applicants' estimate is based on such information. Our investigations show that operating light water reactor plants presently average roughly 400-500 man-rem per unit annually. Thus, the applicants can expect larger man-rem doses if major component failure or other unanticipated problems are encountered. We find the applicants' estimate to be consistent within the scope provided with our as low as practicable policy and acceptable.

The dose estimate to the construction force is based on several assumptions: (1) the main contributions to dose will be direct radiation from the operating unit, containment and exposure to the gaseous effluents from that unit; (2) workers will be located near the control room for Unit 2 and unshielded except for the Unit 1 containment; and (3) 1,930,000 man-hours of work will be expended in the construction of Unit 2 after the startup of Unit 1 (18 months). The applicants estimate 1.6 man-rem as the dose to the construction force. We estimate that the dose may be higher by as much as a factor of two. However, our estimate places the dose within the range of construction worker man-rem doses experienced to date. Therefore, we find their estimate to be as low as practicable and acceptable. Based on our review of the information presented in the PSAR and amendments, including the facility layout, selected shielding calculations, equipment design, and dose assessment, we have concluded that the consideration given by the applicants to shielding design and facility layout to keep exposures within applicable limits and to reduce unnecessary exposure during normal operation is acceptable.

12.2 Ventilation

The South Texas Project Units 1 and 2 plant ventilation systems will be designed to maintain a suitable environment for personnel and equipment. Among the design objectives of these systems are the protection of operating personnel from possible airborne radioactivity and the assurance that maximum expected airborne radioactivity concentrations will be maintained within the limits of 10 CFR Part 20 and as low as practicable. We have determined that these design objectives are acceptable. To meet these objectives, several design criteria are used including: (1) air-flow from areas of least radioactive contamination to areas of progressively greater radioactive contamination followed by exhaust to ventilation ducts; (2) maintenance of slight negative pressures in selected areas; and (3) careful selection of airflow rates to the various cubicles to maintain as low as practicable airborne concentrations of radioactivity. These design criteria are in accordance with the recommendations of Regulatory Guide 8.8 and, therefore, are acceptable.

The normal containment purge subsystem will provide a means of reducing the airborne contamination inside the containment to allow personnel access, and in the fuel building, part of the ventilation will be exhausted around the edge of the fuel pool to reduce airborne radioactivity concentrations in that area. We have determined that the ventilation system, as described in Section 9.4 of the PSAR, meets the design objectives and is, therefore, acceptable. The atmospheric clean-up filter trains will be designed in accordance with the applicable recommendations of Regulatory Guide 1.52 to maintain radiation exposure to as low as practicable during maintenance.

The bases and methods of estimating airborne radioactivity in the plant and expected levels of airborne concentrations in various portions of the plant are described in Section 12.2 of the PSAR. These bases, including rates of leakage and partition factors, are in accordance with the recommendations of Regulatory Guide 1.42. We have performed calculations to confirm the airborne concentrations for certain portions of the plant, and based on these calculations, we find the airborne source terms acceptable.

The design objectives of the airborne radioactivity monitoring system located in the plant ventilation system include: (1) compliance with 10 CFR Part 20, 10 CFR Part 50, and Regulatory Guide 8.8; (2) early warning of increasing radioactivity levels indicative of equipment failure; and (3) continuous surveillance of in-plant airborne radioactivity levels. As a result of our requests for additional information, the fixed airborne radioactivity monitoring systems were changed to give the necessary sensitivity to detect one maximum permissible concentration in air, based on a 40 hour work week, of the most restrictive particulate and iodine isotopes (strontium-90 and iodine-131) in the area or cubicle of lowest ventilation flowrate within one hour after the maximum permissible concentration in air level occurs. Two small compartments in the containment will not meet these monitoring requirements. However, these areas will have very low occupancy and the sensitivity of the system for these areas will allow detection of one maximum permissible concentration in air in two hours. Portable continuous air monitors and portable air samplers will be used to supplement the fixed monitoring system. Therefore, we find the airborne radioactivity monitoring systems to be acceptable.

Several practices which complement the ventilation and fixed monitoring systems and help maintain exposures to airborne contamination as low as practicable are described in Section 12.2 of the PSAR. Some of these practices are training of plant personnel in respiratory hazards and protection, routine airborne radioactivity surveys, access control of areas susceptible to airborne contamination, periodic bioassays, and the use of equipment such as respirators and portable air filtering units. These practices are in accordance with the recommendations of Regulatory Guide 8.8 and, therefore, are acceptable.

Section 12.2 of the PSAR presents an estimate of possible inhalation doses and doses due to submersion in gaseous activity. Using the airborne source terms discussed previously, the applicants calculated representative values of expected doses. These doses represent worst case exposures in which one worker would spend some 4300 hours per year in the areas of highest expected concentrations of airborne contamination and would receive a whole body dose of 4.23 rem per year and a thyroid dose of 3.37 rem per year. We have performed calculations to confirm the applicants' dose estimates. Since a normal work year is approximately 2000 hours, we find the estimate to be in keeping with as low as practicable policy and, therefore, acceptable.

12.3 Health Physics Program

The objectives of the South Texas Project Units 1 and 2 radiation protection program are to provide radiation protection for plant personnel in accordance with 10 CFR Part 20 and to maintain occupational exposures to as low as practicable. To meet these objectives, the program will include training tailored to assignment, radiation zone access control, posting of radiation areas, radiation work permit system, special tools, exposure records and various radiation measuring and monitoring equipment. We have determined that the objectives of the program are acceptable, and the program is in accordance with the recommendations of Regulatory Guide 8.8.

The radiation protection facilities will include an access control checkpoint, change room, personnel decontamination area, radiochemical laboratory, chemical laboratory, instrument calibration room, counting room, and laundry. We consider these facilities to be acceptable for the maintenance of an as low as practicable radiation protection program.

The radiation protection equipment will include protective clothing, respiratory equipment, air sampling equipment, portable radiation measuring instruments, calibration sources, counting room instrumentation, area monitors, airborne activity monitors, laboratory equipment, and special shielding. We consider this equipment to be acceptable for the maintenance of an as low as practicable radiation protection program. Also, to be included are several types of personnel dosimeters. Either thermoluminescent dosimeters or film badges will be used. Film badges and self-reading dosimeters will be used in accordance with the reommendations of Regulatory Guides 8.3 and 8.4, respectively. Neutron film badges, alarming dosimeters, and extremity dosimeters will also be used. Bioassay in the form of whole-body counting and urinalysis will be performed on a periodic basis. All radiation exposure information will be processed and recorded in compliance with 10 CFR Part 20. We have determined that the radiation protection related equipment and facilities are in accordance with the recommendations of Regulatory Guide 8.8.

The Radiation Protection Supervisor will have the responsibility of administering the radiation protection program and for assuring that the requirements and guidance of 10 CFR Part 19, "Notices, Instructions and Reports to Workers; Inspections," 10 CFR Part 20, and those Regulatory Guides concerned with radiation protection will be implemented. In response to our request for additional information, the applicants have stated:

> "The Radiation Protection Supervisor will have a direct line of communication with the Plant Superintendent and the Radiation Protection Supervisor will have

a bachelor's degree or its equivalent in a science or engineering subject including some formal training in health physics. He will have approximately six years of professional experience in health physics (a master's degree may be considered equivalent to a year of experience and a doctor's degree to two years, where courses or work related to radiation protection are involved). At least three years of this professional experience will be in applied radiation protection work dealing with problems similar to those expected in operation of a nuclear power plant."

Based on the above, we have determined that the administrative organization of the radiation protection program and the qualifications of the Radiation Protection Supervisor are in accordance with the recommendations of Regulatory Guide 8.8 and are acceptable. On the basis of our review of the applicants' radiation protection equipment and facilities, qualifications of the radiation protection supervisor, and plans for implementation of the radiation protection program, we have concluded that the overall preliminary radiation protection program is acceptable.

1547 138

13.0 CONDUCT OF OPERATIONS

13.1 Organization and Qualifications

The Houston Lighting & Power Company will act as the project manager for the applicants and will be responsible for the design, construction and operation of the South Texas Project Units 1 and 2. Brown and Root, Inc. has been selected to perform the architect-engineering and construction management services. Westinghouse Electric Corporation will design and fabricate the nuclear steam supply system.

The Houston Lighting & Power Company has established under the General Manager, Power Plant Engineering and Construction, an organization to implement Houston Lighting & Power Company's responsibility for the South Texas Project. Additional technical support for the project will be provided by the Houston Lighting & Power Environmental Protection and Engineering Departments. The Energy Production Department will be responsible for the operation and maintenance of the South Texas Project Units 1 and 2. Quality assurance aspects of the South Texas Project are discussed in Section 17.0 of this report.

The station organization for the operation of the South Texas Project Units 1 and 2 will consist of a technical staff of approximately 70 persons for one unit operation and 107 persons for two unit operation under the direction of a plant superintendent and an assistant plant superintendent, both of whom will have all the qualifications required for a senior operator license except for the license examination itself. Reporting to the assistant superintendent will be: a results engineer (one for each unit) who is responsible for plant performance and who will provide technical assistance; a technical supervisor who directs the activities of the technical staff of approximately 16 persons; a maintenance supervisor, with a staff of approximately 20 persons, responsible for the performance of all work activities in accordance with established procedures and safety standards; an instrumentation and control supervisor, with a staff of about 14 persons, who will supervise the instrument and controls maintenance program; an office manager who is responsible for the maintenance of all plant records and files; and an operating supervisor with a staff of 48 persons for plant operation. This will be a conventional type of plant organization to provide an onsite operating and technical support staff for plant operations. The shift crew for one unit operation will consist of six persons, one of who will hold a senior operators license and two of whom will hold operators licenses. The shift crew for two unit operation will consist of eleven persons, two of whom will hold senior operator licenses and four of whom will hold operator licenses.

The applicants have stated that the qualification requirements of all plant supervisory, operating, technical, and maintenance support personnel will meet or exceed the minimum requirements set forth in American National Standard Institute N18.1-

1971, "Selection and Training of Nuclear Power Plant Personnel" which is consistent with the recommendations of Regulatory Guide 1.8.

Technical support for the plant staff during plant operation will be provided by the General Manager of the Power Plant Engineering and Construction Department and his staff, and by the staff of the Engineering Department. The Houston Lighting & Power Company has an overall management and professional complement in excess of 900 personnel.

We have concluded that the applicants have established an acceptable organization to manage the South Texas Project, and that the proposed plant organization and plans for offsite technical support of plant operations are acceptable. The proposed qualifications for plant staff personnel meet Regulatory Guide 1.8 and are acceptable.

13.2 Training Program

The overall conduct and administration of the plant training program for the station staff is the responsibility of the Plant Superintendent. The Plant Training Coordinator may be delegated the responsibility of development and implementation of the program.

The applicants have stated that a training program will be established to provide plant personnel with sufficient knowledge and operating experience to startup, operate, and maintain the plant in a safe and efficient manner. The training program is to be developed by Houston Lighting & Power Company with principal assistance from the Westinghouse training staff. Training for the station personnel to be licensed will include: basic nuclear training, research reactor training and operation, observation at an operating pressurized water reactor, a plant system lecture series, simulator training and practical on-the-job training. The applicants have committed to simulator training on a simulator that is similar in design and completeness to that of the Westinghouse Zion simulator. The training program at the simulator will be similar in scope and content to the Westinghouse simulator program. Maintenance and technical staff personnel will receive specialized training in their particular fields. Station personnel will also receive training in security and emergency plans, administrative procedures and radiation protection, as appropriate.

On the basis of our review, we have concluded that the training program proposed for the South Texas Project Units 1 and 2 will provide an acceptable number of trained personnel for operation of the facility and is acceptable at the construction permit stage of review.

13.3 Emergency Planning

The applicants have 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 emergencies considered in the emergency plan will include fire or explosion, injury and illness, radiation and contamination accidents, and other emergency conditions that may result from operational malfunctions, natural disasters and civil disturbances. The ranking member of the operating staff present will be designated the Emergency Director and will direct the implementation of the Emergency Plan in accordance with detailed emergency procedures. These procedures will include instructions for the notifications of plant personnel by telephone, alarm or public address systems, and the notification of offsite company management and offsite agencies by telephone or radio.

Initial contacts and arrangements have been made with the following agencies: Texas State Department of Health, Texas Department of Public Safety, Matagorda County Sheriff, US Coast Guard, Matagorda General Hospital, University of Texas M. D. Anderson Hospital, and Bay City Fire Department. The Texas State Department of Health, Division of Occupational Health and Radiation Control, has been identified as having primary responsibility for radiological emergency planning in the environs of the proposed facility.

Responses to emergency situations have been categorized into four levels, depending on the severity of the situation. In-plant monitors will provide the first indication of a radiological emergency. Provisions will be made for surveys by portable meters and air sampling devices on a timely basis. The onsite Visitors Center has been designated as the Emergency Operations Center. One alternate center will be designated. Decontamination facilities and a first aid room will be provided. Arrangements have been initiated with area hospitals to treat contaminated injury cases. All plant personnel will receive training in emergency procedures and periodic drills will be conducted. Training will also be provided for those offsite agencies who may be called upon in emergency situations.

We have made an independent assessment of the population distribution and evacuation routes in the area of the proposed site and have determined that it is feasible and practicable to take protective measures, including evacuation on a timely basis within and beyond the site boundary during the expected lifetime of the plant. We have also determined that appropriate criteria have been identified for the design of an acceptable emergency plan.

We have reviewed the applicants' preliminary emergency plans for coping with emergencies and find that they meet the requirements of 10 CFR Part 50, Appendix E, "Emergency Plans for Production and Utilization Facilities", and are acceptable.

13.4 Review and Audit

The applicants have described the preliminary plans for the review and audit of South Texas Project Units 1 and 2 operations. We have reviewed these preliminary plans and have concluded that they generally meet those provisions described in American National StandardsInstitute N18.7-1972, "Administrative Controls for Nuclear Power Plants," and are acceptable.

13.5 Plant Procedures

Plant procedures will be performed in accordance with written and approved operating and emergency procedures. Areas that will be covered include normal startup, operation and shutdown, maintenance, and abnormalities in operation. American National Standards Institute N18.7,-1972, "Administrative Controls for Nuclear Power Plants," will be used as a guide in preparation of these procedures. The administrative and operating procedures will be completed and reviewed at least three months prior to fuel loading. The applicants have committed to developing procedures for the review, change and approval of all plant operating, maintenance and testing procedures.

We have concluded that the applicants' proposed program for preparation, review, approval and use of written procedures, and the commitment to document operating and maintenance activities is acceptable at the construction permit stage of review.

13.6 Plant Records

The applicants have committed to keeping plant records in accordance with American National Standards Institute N18.7-1972, "Administrative Control for Nuclear Power Plants" and N45.2.9, "Requirements for Collection, Storage and Maintenance of Quality Assurance Records." On this basis, we have concluded that these record keeping provisions are acceptable.

13.7 Industrial Security

The applicants have 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 and conform to Regulatory Guide 1.17.

On the basis of our review, we have concluded that the applicants' arrangements for protection of the plant against acts of industrial sabotage are acceptable for the construction permit stage of review.

1547 142

14.0 INITIAL TESTS AND OPERATIONS

We have reviewed the applicants' planned test program for the South Texas Project Units 1 and 2, as described in Section 14.1 of RESAR-41 and as supplemented in Section 14.1 of the PSAR.

The initial test program for South Texas Units 1 and 2 will be conducted by the applicants who will receive technical direction and support from the nuclear steam supply system vendor, Westinghouse Electric Corporation, and architect-engineer-constructor, Brown and Root, Inc. The applicants have committed to develop and execute the test program in accordance with Regulatory Guide 1.68. We have determined that the applicants' plans for the test program are acceptable and will, when implemented, provide for verification of the functional adequacy of the facility.

Our review has identified two potential problem areas in the applicants' plans for preoperational testing. The issues involve the planned preoperational testing of the emergency core cooling system and the planned testing of the instrument air system. We have advised the applicants that testing of these systems should be conducted in accordance with methods described in Regulatory Guides 1.79 and 1.80 or that suitable alternative testing should be performed to demonstrate that these systems will meet design requirements. We believe that acceptable testing can be developed for these systems and we will require the applicants to develop acceptable testing methods at the operating license stage of review.

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

1547 143

15.0 ACCIDENT ANALYSES

15.1 Introduction

Our evaluation of the capability of the RESAR-41 nuclear steam supply system to withstand abnormal operational transients and postulated accidents is presented in Section 15.0 of Appendix A to this report. Therefore, the discussion below is limited to radiological consequences of accidents specifically related to the South Texas Project Units 1 and 2.

The section numbering system used in this section is based on the numbers in Section 15.0 of Appendix A to this report that deal with the same subject matter. This correspondence is valid in all cases to the first decimal (e.g., 15.7) but does not necessarily follow to the second decimal.

15.7 Radiological Consequences of Accidents

15.7.1 General

The postulated design basis accidents analyzed by the applicants for the offsite radiological consequences are the same as those analyzed for previously licensed pressurized water reactor plants such as the Comanche Peak Steam Electric Station Units 1 and 2 (Docket Nos. 50-445 and 50-446). These include a design basis loss-of-coolant accident, a steam line break accident, a steam generator tube rupture, a fuel handling accident, and a control rod ejection accident.

We have reviewed these accidents and further evaluated the design basis loss-of-coolant accident and the fuel handling accident. The offsite doses we calculated for these accidents are presented in Table 15.1 and the assumptions we used are listed in Tables 15.2, 15.3 and 15.4 and of this report.

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 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 such that any single failure (such as a relief valve lifting and sticking open) would not result in doses that are more than a small fraction of the 10 CFR Part 100 guideline values.

1547 144

The control rod ejection accident will be evaluated at the operating license stage of our review. This may require a technical specification which limits the allowable operational leakage of reactor coolant into the steam generator secondary side to assure that the radiological consequences of this accident will be within the dose guidelines of 10 CFR Part 100.

15.7.2 Loss-of-Coolant Accident

The South Texas Project pressurized water reactors will each be housed in a low leakage containment structure. The containment building spray system will be equipped with a sodium hydroxide additive injection system. The purpose of the additive is to increase the iodine removal capability of the containment spray following the postulated design basis loss-of-coolant accident. Sections 6.2.2 and 6.2.3 of this report discuss the operation of the containment spray system. The design basis loss-of-coolant accident dose values given in Table 15.1 include credit for iodine removal by the containment sprays. The calculated design basis loss-of-coolant accident doses meet the guideline values given in Regulatory Guide 1.4 for a plant at the construction permit review stage. The assumptions used in evaluating the consequences of the accident are given in Table 15.2. The dose model and dose conversion parameters used in the design basis loss-of-coolant accident analysis are consistent with those given in Regulatory Guide 1.4.

15.7.3 Fuel Handling Accident

For the analysis of the fuel handling accident, we have assumed that a fuel assembly was dropped in the 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 handling exhaust subsystem. As discussed in Sections 6.6 and 9.4.3 of this report, we required that the fuel handling building exhaust subsystem meet the recommendations of Regulatory Guide 1.52 including seismic requirements. On this basis, in our calculations of the radiological consequences of accidents, we have credited the fuel handling building exhaust subsystem with an adsorption efficiency of 95 percent for both elemental and organic iodine removal. Our assumptions are listed in Table 15.3, and the calculated doses are listed in Table 15.1.

15.7.4 Hydrogen Purge Dose Analysis

The applicants 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. For use in the event of the failure of both recombiners, the applicants will provide a backup purging mode. We have evaluated the additional dose an individual might receive due to purging the containment following the postulated design basis loss-of-coolant accident. Our assumptions are listed in Table 15.4, and the calculated doses are listed in Table 15.1. We find that the calculated low population zone doses from purging, when added to the design basis loss-of-coolant accident doses, are well within the guideline of 10 CFR Part 100.

TABLE 15.1

RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS ACCIDENTS

Accident	Exclusion Area 1430 meters (4692 feet) 2-Hour Dose (Rem)		Low Population Zone 4830 meters (three miles) 30-Day Dose (Rem)	
	Thyroid	Whole Body	Thyroid	Whole Body
Loss-of-Coolant	141	8	68	3
Hydrogen Purge Dose	-	-	20	<1
Fuel Handling	5	3	-	-

1547 146

TABLE 15.2

ASSUMPTIONS USED IN THE ESTIMATE OF

DESIGN BASIS LOSS-OF-COOLANT ACCIDENT DOSES

Power Level	4100 megawatts thermal
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
Containment Leak Rate	
0-24 hours	0.3 percent per day
greater than 24 hours	0.15 percent per day
Containment Volume	
Sprayed Volume	2,860,000 cubic feet
Unsprayed Volume	500,000 cubic feet
Containment Mixing Rate Between Sprayed and Unsprayed Volume	150,000 cubic feet per minute
Containment Spray System	
Maximum Elemental Iodine Decontamination Factor	100
Removal Coefficients	
Elemental Iodine	10 inverse hours
Particulate Iodine	0.45 inverse hours
Organic Iodine	0
Minimum Exclusion Area Boundary Distance	1430 meters (4692 feet)
Low Population Zone Distance	4830 meters (3 miles)
Relative Concentration Values (seconds per cubic meter)	
0-2 hours at 1430 meters (4692 feet)	1.7×10^{-4}
0-8 horrs at 4830 meters (3 miles)	2.1×10^{-5}
8-24 hours at 4830 meters (3 miles)	1.4×10^{-5}
24-96 hours at 4830 meters (3 miles)	5.8×10^{-6}
96-720 hours at 4830 meters (3 miles)	1.6×10^{-6}

1547 147

TABLE 15.3

REFUELING ACCIDENT CALCULATION ASSUMPTIONS

AND INPUT PARAMETERS

Power Level	4100 megawatts thermal
Number of Fuel Rods Damaged	264
Total Number of Fuel Rods in Core	50,952
Radial Peaking Factor of Damaged Rods	1.65
Shutdown Time	20 hours
Inventory Released from Damaged Rods (Iodines and Noble Gases)	10 percent
Pool Decontamination Factors	
Iodines	100
Noble Gases	1
Iodine Fractions Above Pool	
Elemental	75 percent
Organic	25 percent
Filter Efficiencies for Iodine Removal	
Elemental	95 percent
Organic	95 percent
0-2 Hour Relative Concentration Value at 1430 meters (4692 feet)	1.7×10^{-4} seconds per cubic meter

1547 148

TABLE 15.4

HYDROGEN PURGE DOSE INPUT PARAMETERS

Power Level	4100 megawatts thermal
Containment Volume	3,300,000 cubic feet
Holdup Time in Containment Prior to Purge Initiation	22 days
Purge Duration	30 days
Purge Rate	48 standard cubic feet per minute
4-30 day Relative Concentration Value at 4830 meters (3 miles)	1.6×10^{-6} seconds per cubic meter

15.7.5 Post-Loss-of-Coolant Accident Recirculation Leakage

A potential source of fission product leakage following a design basis loss-of-coolant accident is leakage from the emergency core cooling system and containment spray system which will be located outside of containment in the fuel handling building.

In our review of RESAR-41 we evaluated the possible consequences of leakage of these systems following a design basis loss-of-coolant accident. As a result of our review of RESAR-41 we determined that if the emergency core cooling system and containment spray system equipment area is served by filters effective in removing iodine under accident conditions, the offsite doses from possible pump and valve leakage in this area would be within the guidelines of 10 CFR Part 100, even for substantial amounts of leakage (see Section 15.7.2 of Appendix A to this report). Therefore, to assure that substantial leakage can be accommodated, we will require that the emergency core cooling system and spray system equipment area be serviced by a filter system which conforms to the requirements of an engineered safety features system and meets the recommendations of Regulatory Guide 1.52.

The South Texas Project emergency core cooling system and containment spray system equipment area will be serviced by the fuel handling building exhaust subsystem. Therefore, as discussed in Sections 6.6 and 9.4.3 of this report, we required that this subsystem meet the recommendations of Regulatory Guide 1.52 including seismic requirements.

1547 150

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 final design.

We have reviewed the proposed technical specifications presented in Section 16 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 a similar design to South Texas Project Units 1 and 2. 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.

1547 151

17.0 QUALITY ASSURANCE

17.1 General

The description of the quality assurance (QA) program for South Texas Project Units 1 and 2 is contained in Section 17 of the PSAR as amended. The applicants have designated Houston Lighting & Power Company as Project Manager responsible for the technical adequacy of the design and construction of the South Texas Project Units 1 and 2.

Our evaluation of the description of the South Texas Project QA program is based on our review of this information and detailed discussions with the applicants' Project Manager, Houston Lighting & Power Company, to determine the qualification and capability of the Houston Lighting & Power Company and the principal contractors, Brown & Root, Inc. and Westinghouse Electric Corporation, to comply with the requirements of Appendix B to 10 CFR Part 50 and applicable Regulatory Guides and industry standards.

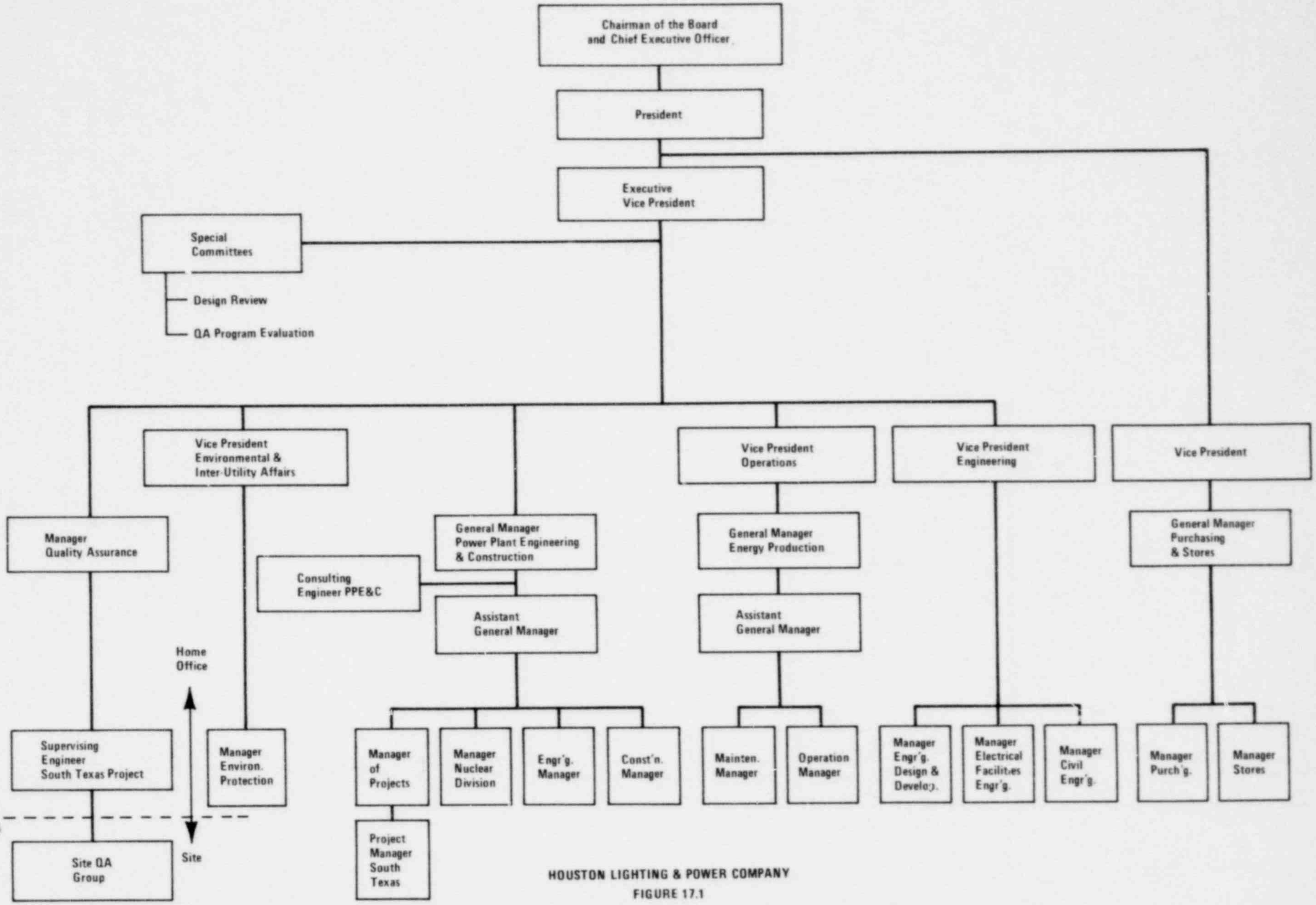
17.2 Houston Lighting & Power Company

Houston Lighting & Power Company has contracted with Westinghouse to supply the nuclear steam supply system and with Brown & Root to supply the architect-engineering and construction services and to be responsible for the QA program for the balance of the plant. Houston Lighting & Power Company is responsible for the total South Texas Project QA program and is organized to oversee and control the efforts of the principal contractor's QA programs.

Houston Lighting & Power Company consists of two major organizational elements reporting to a President (as shown in Figure 17.1) with a Vice-President responsible for administration activities and an Executive Vice-President responsible for technical activities. The Manager of QA reports to the Executive Vice-President who, in addition to QA, is responsible for power plant engineering and construction, operations (energy production), environmental and inter-utility affairs, and engineering.

Figure 17.1 shows the Manager of QA on the same organizational level as those whose work he oversees. We find that with this corporate organization structure, QA has adequate independence and reports at a sufficiently high management level to accomplish its objectives.

The President of Houston Lighting & Power Company has delegated, through the Executive Vice-President to the QA Manager, authority and responsibility for establishing and implementing a QA program. The QA Manager has established, well-defined responsibilities and authorities for implementing the QA program in documented procedures and instructions. The President has issued a written statement, which is included in the QA manual, stating that the QA program requirements are mandatory for all persons performing quality related activities. We find that Houston Lighting & Power Company has clearly defined the responsibilities and authorities of its QA organization.



HOUSTON LIGHTING & POWER COMPANY
FIGURE 17.1

1547 153

Houston Lighting & Power Company implements its QA functions by means of a staff QA group, a site QA group, and two corporate level committees. The two committees are the Design Review Committee and the QA Program Evaluation Committee.

The Manager of QA is a member of the Design Review Committee and is Chairman of the QA Program Evaluation Committee. The Design Review Committee is composed of manager level personnel who meet at least quarterly to assure the technical adequacy of plant design by means of design reviews. The QA Program Evaluation Committee is composed of executive level management personnel who meet semiannually to assess the status and adequacy of the overall QA program.

The South Texas Project Manager of QA and the Supervising Engineer have stop work authority. Stop work authority can be exercised during construction by personnel of the Houston Lighting & Power Company Site QA Group.

We find that Houston Lighting & Power Company's provisions for implementing their QA program; with corporate level management involvement, authority from the President to enforce QA requirements, and QA stop work authority; are acceptable.

The Houston Lighting & Power Company has developed a detailed indoctrination, training, and continuing education program to ensure that QA personnel are qualified to a level commensurate with their responsibility and will meet the requirements of American National Standard Institute N45.2.6. The training program includes indoctrination of new employees, on-the-job training, training courses and sessions, industry seminars, and continuing education funded by the Houston Lighting & Power Company.

Our evaluation of the Houston Lighting & Power Company QA organization is that it is independent of the organizations whose activities it verifies; it has clearly defined authorities and responsibilities; it has adequately defined qualification and training requirements for its staff; it is so organized that it can identify quality problems in the other organizations performing quality related work; it can initiate, recommend or provide solutions; and it can verify implementation of solutions. We therefore conclude that the Houston Lighting & Power Company organization complies with Appendix B to 10 CFR Part 50 and is acceptable.

The initial information in the PSAR did not provide sufficient detail on the QA program to allow us to complete our evaluation. In response to our requests, Houston Lighting & Power Company provided, in amendments to the PSAR, additional information on the QA program for the design, procurement and construction of the South Texas Project, which is described below.

A list of the QA policies and procedures used to administer the QA program has been provided in Section 17 of the PSAR. These policies and procedures form the Houston Lighting & Power Company QA program and the South Texas Project QA plan. The QA program specifies the quality requirements to which the South Texas Project will comply. The South Texas Project QA plan provides the procedures, methods, techniques and instructions necessary to implement the requirements. Both documents are prepared by the QA organization and are approved by the Manager of QA and the Executive Vice-President. The PSAR includes a listing of the QA program requirements and the South Texas Project QA plan procedures plus a matrix of these requirements and procedures cross referenced to the criterion of Appendix B to 10 CFR Part 50. The structures, systems, and components comprising the safety items which are subject to this program have been identified in the PSAR. Based on our review of this information, we have concluded that each criterion of Appendix B to 10 CFR Part 50 has been adequately included in both of these documents.

The QA program and QA plan for the South Texas Project are structured in accordance with the Regulatory Guides and industrial standards that are addressed by the Nuclear Regulatory Commission in "Guidance on QA Requirements During Design and Procurement Phase of Nuclear Power Plants," (Revision 1) May 24, 1974 (WASH 1283) and "Guidance on QA Requirements During the Construction Phase of Nuclear Power Plants," May 10, 1974 (WASH 1309). Based on this and Houston Lighting & Power Company's definition of their policies and procedures, we have determined that the QA program to which the Houston Lighting & Power Company has committed is acceptable.

The Houston Lighting & Power Company by surveillance, will assure that its principal contractors and subcontractors have adequate QA programs, that inspections will be performed to documented inspection instructions by qualified personnel, and that results will be recorded. Houston Lighting & Power Company will assure by surveillance and audits that personnel performing inspections are free from undue cost and schedule pressures of the project.

Houston Lighting & Power Company has established program requirements on itself and on its contractors which assure there will be a documented system of records attesting to quality.

A system of planned and documented audits, described in the PSAR, will be used by the Houston Lighting & Power Company to verify compliance with all aspects of the QA program and to assess its effectiveness. The Houston Lighting & Power Company has committed in the PSAR that the auditing system used by itself and its contractors, subcontractors, and vendors will meet Houston Lighting & Power Company's QA program requirements. This authorization includes manpower, funding, and facilities to implement the system of audits. Houston Lighting & Power Company's audit results will be reviewed and corrective action taken by responsible management. Follow-up action is taken to assure corrective action. We have determined that the audit system to which the Houston Lighting & Power Company has committed is acceptable.

Houston Lighting & Power Company's executive level management regularly assesses the scope, implementation, and effectiveness of the QA program by means of the QA Program Evaluation Committee, described previously, which meets semiannually and submits a written report of its findings to the Executive Vice-President.

Based on our review of the description of the QA program contained in Section 17 of the PSAR, we find that there are adequate and well-defined procedures, a commitment to the Nuclear Regulatory Commission 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 acceptable management assessment of the status and adequacy of the QA program.

We have concluded that Houston Lighting & Power Company's QA program for the South Texas Project includes 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.

17.3 Brown & Root, Inc.

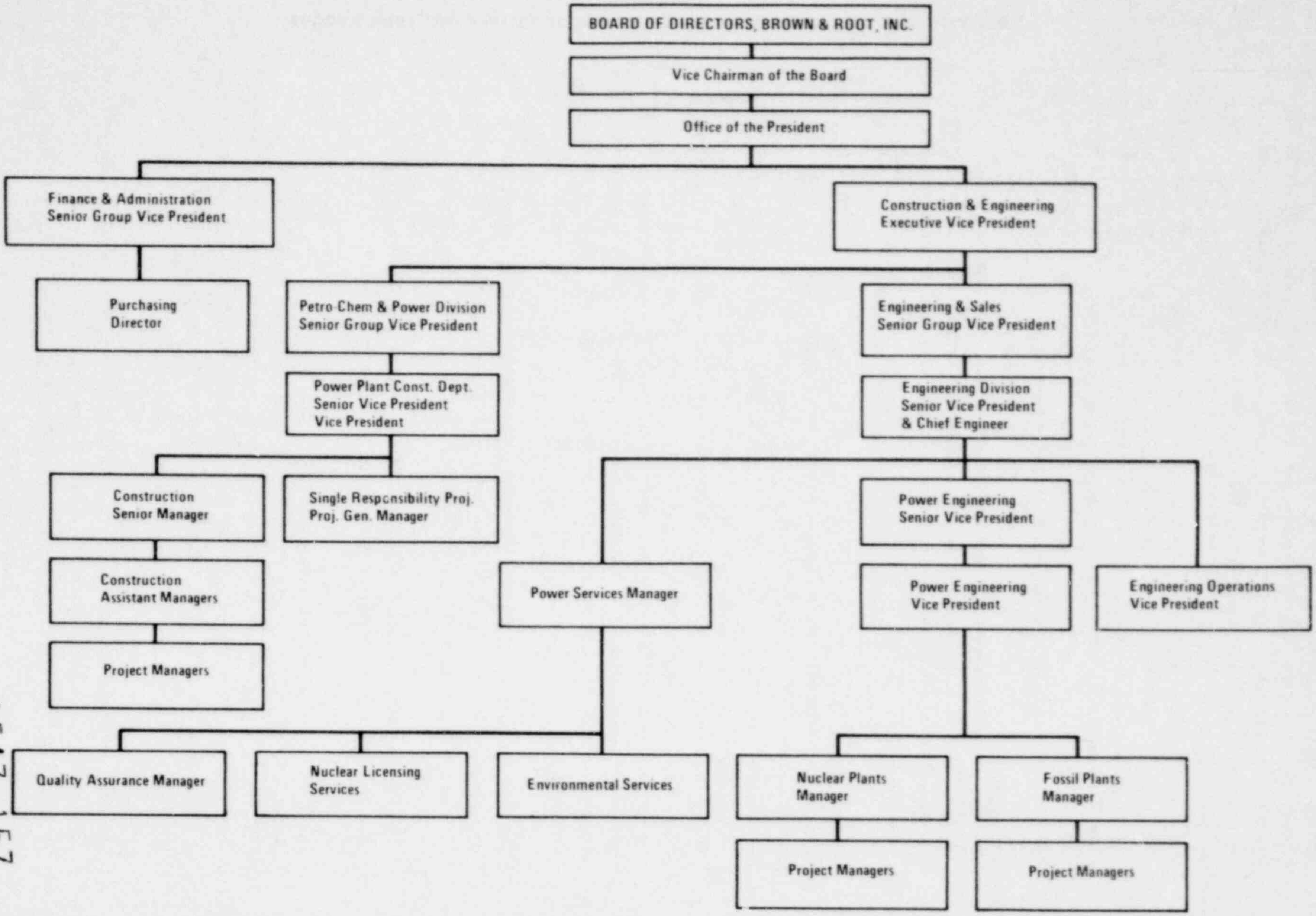
Brown & Root Inc. has been designated engineer-constructor responsible for the design, engineering, equipment and materials procurement, and construction of the South Texas Project. This includes all plant structures, systems, and components except those provided by Westinghouse. Figure 17.2 shows the Brown & Root organization as it relates to engineering, procurement, construction, and QA. The Manager of Power Services, responsible for the QA program, reports to the Engineering Division Senior Vice-President and Chief Engineer. The Manager of Power Services is on a comparable organization level and technically and administratively independent of the engineering, construction and project organizations whose work the QA organization oversees.

Although the purchasing organization is outside the construction and engineering organization, QA controls of procurements are well defined in the PSAR. Therefore, we find the organizational independence shown in Figure 17.2 acceptable.

The Executive Vice-President of Brown & Root has issued a management statement of policy which requires mandatory implementation of the QA program.

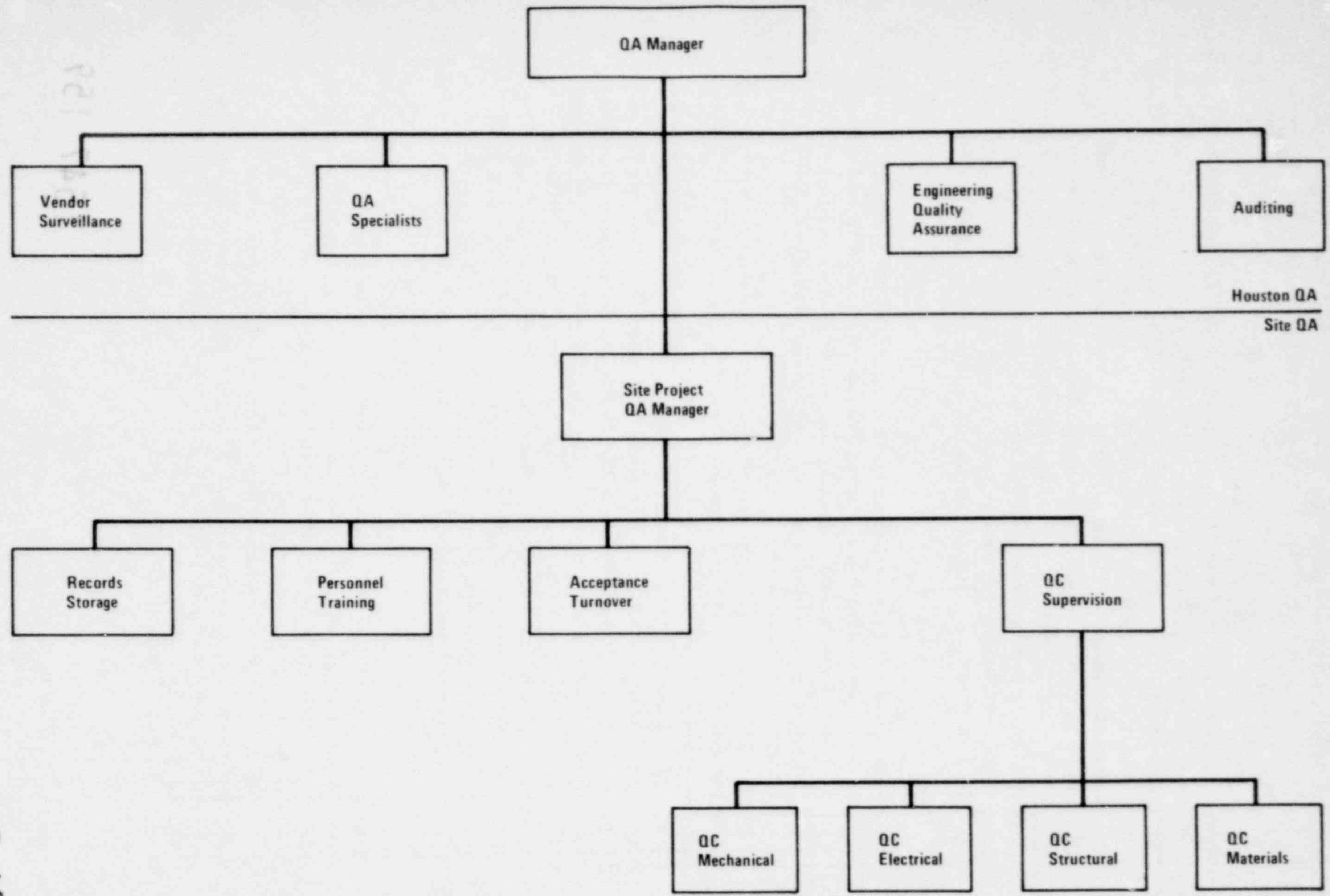
The Brown & Root QA Manager, reporting to the Manager of Power Services, issues the quality assurance/quality control (QA/QC) procedures for the South Texas Project. Engineering procedures and purchasing procedures for South Texas Project are reviewed and audited by Brown & Root QA. Quality verification activities such as inspection, audits, and surveillance are conducted by personnel in the QA organization (see Figure 17.3).

The PSAR includes matrices of QA/QC procedures (including inspection procedures for construction), engineering procedures, and purchasing procedures for the South Texas Project. These procedures are cross referenced to the criteria of Appendix B to 10



BROWN & ROOT, INC. ORGANIZATION
FIGURE 17.2

1547 157



BROWN & ROOT, INC. QUALITY ASSURANCE/QUALITY CONTROL (QA/QC) ORGANIZATION
FIGURE 17.3

1547 158

CFR Part 50. Based on our review of this information, we have concluded that each criterion of Appendix B to CFR Part 50 has been adequately included in the QA program for the South Texas Project.

In response to our request, Brown & Root, Inc. has committed to follow the guidance provided by the Commission in "Guidance on Quality Assurance During Design and Procurement Phase of Nuclear Power Plants," (Revision 1) May 24, 1974 (WASH 1283) and "Guidance on Quality Assurance Requirements During the Construction Phase of Nuclear Power Plants," May 10, 1974 (WASH 1309). Based on this, Brown & Root, Inc. has committed to the essential requirements for a QA program in compliance with Appendix B to 10 CFR Part 50. Brown & Root, Inc. has identified the safety related structures, systems, and components that are subject to the Brown & Root QA program in the PSAR. These safety related items, and those listed in RESAR-41 supplied by Westinghouse, will fall within the Brown & Root QA program upon receipt at the South Texas Project site.

The PSAR describes a training and indoctrination program committed to assuring that personnel performing quality affecting activities understand, implement, and enforce the Brown & Root QA policies and procedures. The program assures adequate training and qualification in the principles and techniques of quality related activities.

The Brown & Root QA organization is shown in Figure 17.3. Functions of Engineering QA include assurance that QA program requirements are implemented during the design and procurement phase of the project. This office also coordinates QA technical matters through the Brown & Root Houston QA office during the construction phase of the project. The Brown & Root site QA organization is supervised by the Site Project QA Manager who is directly responsible to the QA Manager. He coordinates QA project administration and policy with the Construction Project Manager. His staff performs the needed QA/QC functions at the South Texas Project site. In addition, testing laboratories performing QC functions will report to the Site Project QA Manager. QA inspection personnel report to QC supervision which, in turn, reports to the Site Project QA Manager. We have concluded that the site QC inspectors have sufficient authority and organizational freedom to perform their functions effectively and without reservation.

Brown & Root, Inc. has described a system of planned and documented audits with provision for corrective and followup actions. Audits will be performed in accordance with written checklists by appropriately trained personnel having no direct responsibility in the area audited. Audit schedules are based on the status and safety importance of the activities being performed. Audit report distribution includes management personnel of the audited area, Houston Lighting & Power Company and Brown and Root. The audit reports will be a part of the QA record files at the project.

1547 159

Brown & Root, Inc. has established a QA Review Board which includes the Senior Vice-President of Power Construction and Power Engineering, the Director of Purchasing, and the QA Manager. This board meets at least semiannually to review and discuss the administrative activities of the QA Department to determine and evaluate the effectiveness of the corporate QA program.

Based on our review and evaluation of the QA program described in Section 17 of the PSAR, we have concluded that Brown & Root's QA program for the South Texas Project demonstrates an acceptable QA and QC 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 Westinghouse Electric Corporation

Our evaluation of the Westinghouse QA program is presented in Section 17.0 of Appendix A to this report.

17.5 Implementation of Quality Assurance Program

The Commission's Office of Inspection and Enforcement has inspected the quality assurance program implementation for South Texas Project Units 1 and 2. On the basis of this and past inspection results and the time available to implement the necessary corrective actions, the Office of Inspection and Enforcement has concluded that the QA program implementation is consistent with the status of the project.

17.6 Conclusions

In our review, we have evaluated the QA program of Houston Lighting & Power Company, Westinghouse, and Brown & Root for compliance with the Commission's regulations and applicable Regulatory Guides and industry standards.

Based on our review, we have concluded that the QA program complies with Appendix B to 10 CFR Part 50 and applicable Regulatory Guides and industry standards and is acceptable for the design, procurement, and construction of the South Texas Project. 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.

1547 160

18.0 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The application for the South Texas Project Units 1 and 2 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.

1547 161

19.0 COMMON DEFENSE AND SECURITY

The applicants state 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 applicants are citizens of the United States.

The applicants are 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 applicants have agreed to safeguard any such data that might become involved in accordance with the requirements of 10 CFR Part 50. The applicants 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.

1547 162

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. We are continuing our review of the financial qualifications of the applicants and will report the results of our evaluations in a supplement to this report.

1547 163

21.0 CONCLUSIONS

Based on our analysis of the proposed design of South Texas Project Units 1 and 2 and upon favorable resolution of the outstanding matters set forth in Section 1.8 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 applicants have described the proposed design of the facility, including, but not limited to, the principal architectural and engineering criteria for the design, and have 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 reasonably be 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 by the applicants and the applicants have identified, 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 applicants are technically qualified to design and construct the proposed facility;

(6) The applicants have reasonably estimated the costs and are financially qualified to design and construct the proposed facility; and

(7) The issuance of permits for construction of the facility will not be inimical to the common defense and security or to the health and safety of the public.

1547 164

APPENDIX A

1547 165

July 3, 1975

REPORT

TO THE

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

BY THE

OFFICE OF NUCLEAR

REACTOR REGULATION

U. S. NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

WESTINGHOUSE ELECTRIC CORPORATION

REFERENCE SAFETY ANALYSIS REPORT

RESAR-41

DOCKET NO. STN 50-480

1547 166

TABLE OF CONTENTS

1547 167

1547 168

TABLE OF CONTENTS (Continued)

1547 168

TABLE OF CONTENTS (Continued)

1547 169

TABLE OF CONTENTS (Continued)

1547 170

TABLE OF CONTENTS (Continued)

1547 171

LIST OF TABLES

1547 172

LIST OF FIGURES

1547 173

1.0 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

The Westinghouse Electric Corporation (hereinafter also referred to as Westinghouse) filed on December 3, 1973, with the Nuclear Regulatory Commission (then known as the Atomic Energy Commission), a proposed preliminary standard design for a nuclear steam supply system. This submittal was in the form of an application for a Preliminary Design Approval by the Nuclear Regulatory Commission and was in response to Option 1 of the Nuclear Regulatory Commission's (the Commission) standardization policy WASH-1341 "Programmatic Information for the Licensing of Standardized Nuclear Power 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. On March 11, 1974, the application was docketed.

The initial Commission policy statement on standardization of nuclear power plants was issued on April 28, 1972. It provided the impetus for both industry and the Commission to initiate active planning in their respective areas in order to realize the benefits of standardization while maintaining protection of the health and safety of the public and of the environment. In a subsequent statement issued on March 5, 1973, the Commission announced its intent to implement a standardization policy for nuclear power plants. WASH-1341 was issued August 20, 1974. 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 Part 50 and Section 2.110 of Part 2 of Title 10 of the Code of Federal Regulations (CFR).

A Standard Safety Analysis Report was submitted with the application in the form of a Westinghouse Reference Safety Analysis Report, RESAR-41. The information in RESAR-41 has been supplemented by Amendments 1 through 17. RESAR-41 and copies of these amendments are available for public inspection at the Nuclear Regulatory Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C.

This Report to the Advisory Committee on Reactor Safeguards (report) summarizes the results of the technical evaluation of the proposed RESAR-41 design performed by the Commission's staff and delineates the scope of the technical matters considered in evaluating the radiological safety aspects of RESAR-41. Aspects of the environmental impact were not considered in the review of RESAR-41 but will be addressed in each utility application for a construction permit which references RESAR-41.

Upon the favorable resolution of the outstanding issues discussed herein and summarized in Section 1.7 of this report, we will be able to conclude that the proposed preliminary design of the nuclear steam supply system can be incorporated by reference in construction

1547 174

permit and standard balance-of-plant design applications and can be constructed without endangering the health and safety of the public when it is referenced in a reviewed and approved construction permit application. Our detailed conclusions are presented in Section 19 of this report.

Future utility applicants referencing RESAR-41 will retain their own architect-engineers, constructors, turbine-generator vendor, and consultants as needed. We will need to conclude for each application referencing RESAR-41 that the applicant, along with his contractors, is technically competent to manage, design, construct and operate a nuclear power plant prior to issuance of a Construction Permit.

The review and evaluation presented in this report is only the first stage of a continuing review by the Commission's staff of the design, construction, and operating features of the RESAR-41 nuclear steam supply system. Prior to the issuance of an operating license for any application referencing RESAR-41 we will review the final design of the RESAR-41 nuclear steam supply system to determine that all of the Commission's safety requirements have been met in accordance with Appendix O to 10 CFR Part 50. The facility may then be operated only in accordance with the terms of the operating license and the Commission's regulations under the continued surveillance of the Commission's staff.

Designs for systems and components contained in RESAR-41 which are outside the standardized scope of nuclear steam supply system applications as defined in Amendment 1 to WASH-1341 are not discussed in the main body of this report. Our evaluation of these systems and components are described in Appendix A to this report. In addition, Westinghouse offers several of its systems and components as options. We have appropriately identified these options in each section.

In the course of our safety review of the material submitted, we held numerous meetings with representatives of Westinghouse to discuss the plant design and performance. During our review, we requested Westinghouse to provide additional information that we needed for our evaluation. This additional information was provided in amendments to RESAR-41. As a result of our review, a number of changes were made in the facility design. These changes are described in the amendments and are discussed in appropriate sections of this report. Section 1.6 provides a listing of the principal design changes which were made. A chronology of the principal actions relating to the processing of the application is attached as Appendix B to this report. A bibliography is also included in Appendix C.

Since the RESAR-41 nuclear steam supply system does not cover the entire facility, it is necessary to specifically and extensively describe the safety-related interfaces between the nuclear steam supply system and the balance-of-plant. Interface information addresses the pertinent safety-related design requirements including the operating environment, inputs to transient and accident analysis, and the layout, structural and performance requirements necessary to assure the compatability of the nuclear steam supply system to its mating portion of the plant and site. The status of our review of the interface information in RESAR-41 is contained in Section 1.7 of this report.

1.2 General Description

As can be seen from Figure 1-1, RESAR-41 includes only systems and components which will be closely associated with the operation of the reactor coolant system. Not included in the RESAR-41 scope are plant buildings and structures, the turbine generator and its auxiliaries, and the main steam system beyond the steam generators. System piping and layout will only be included for the reactor coolant system. The RESAR-41 nuclear steam supply system is designed to operate at a core thermal power level of 4100 megawatts. In keeping with the guidelines of Regulatory Guide 1.49, RESAR-41 is an application for a Preliminary Design Approval for a core thermal power of 3800 megawatts.

The proposed nuclear steam supply system design in RESAR-41 incorporates a pressurized water reactor in a four-loop reactor coolant system. Preliminary designs for control and instrumentation systems, safety systems and power systems which will support the reactor coolant system under normal and accident conditions are also included. Figure 1-1 graphically shows the design scope of RESAR-41.

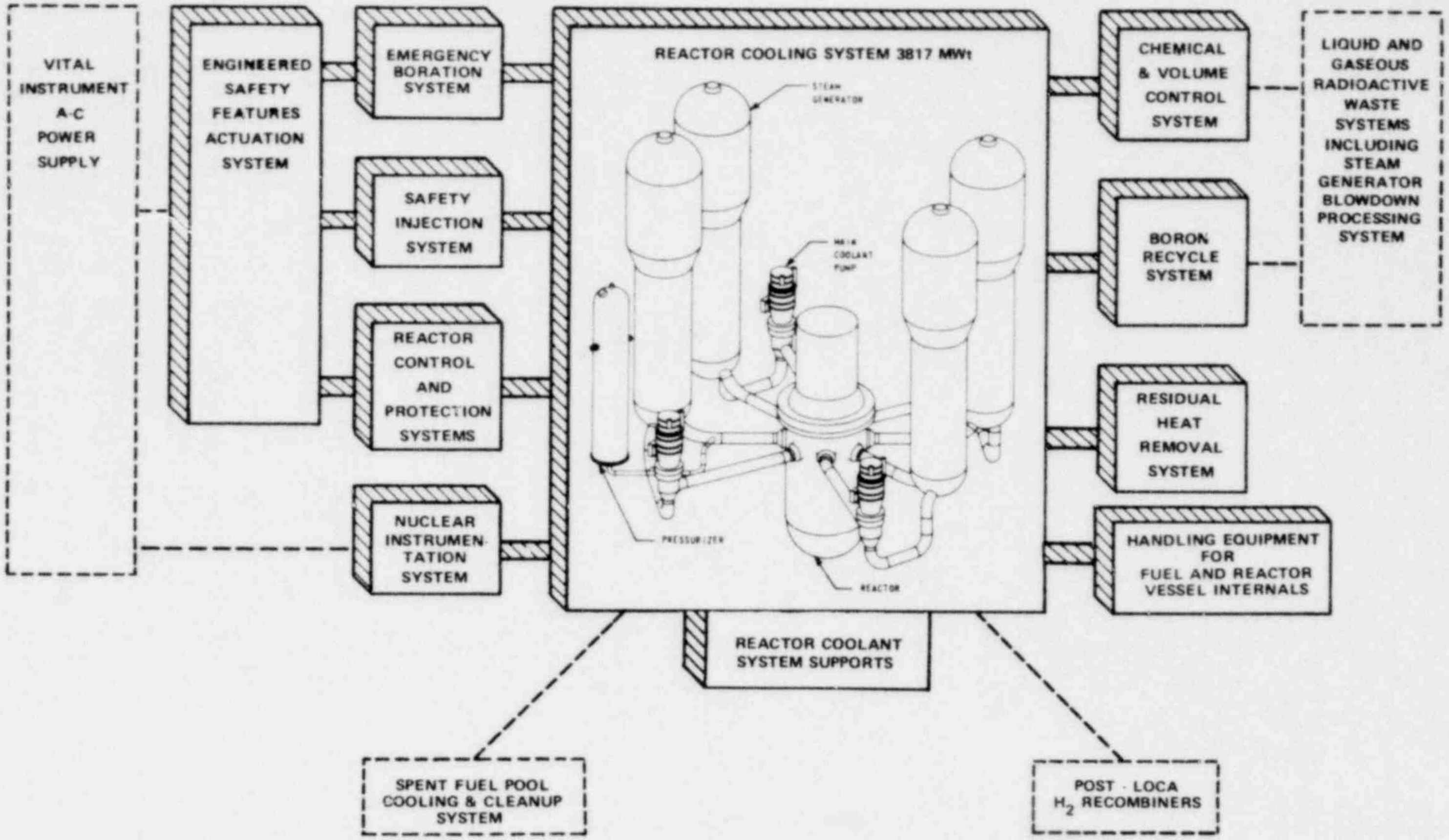
The RESAR-41 nuclear steam supply system is a design for a single unit. Systems and components within the nuclear steam supply system that are important to safety will not be shared.

1.2.1 Reactor

The proposed reactor core will consist of fuel rods made from uranium-dioxide pellets contained in slightly cold worked Zircaloy-4 tubing which will be plugged and seal welded at the ends to encapsulate the fuel. The fuel pellets consisting of slightly enriched uranium-dioxide powder will be compacted by cold pressing and then sintered to the desired density. Shifting of the fuel within the cladding prior to fuel loading will be prevented by a stainless steel spring which bears on top of the fuel. All fuel rods will be internally pressurized with helium during the welding process. The design height of the fuel pellets within each rod is 164 inches, while the overall fuel rod length will be 173.3 inches.

The fuel rods will be combined in a 17x17 array to form fuel assemblies. These fuel assemblies will have nine spacer grids and contain guide thimble channels for the neutron absorber rods, burnable poison rods or neutron source assemblies. The core will be formed of 193 fuel assemblies divided into three regions, each utilizing fuel of a different enrichment of U-235. The new, highest enrichment fuel will be introduced into the outer core regions, moved inward at successive refuelings, and ultimately removed from the inner region to spent fuel storage. The 164 inch fuel is often referred to as "14-foot fuel" and is designated 17x17 XLR by Westinghouse. The proposed fuel enrichment for the core regions are 2.10 weight percent uranium-235 for the inner region, 2.60 weight percent for the middle region, and 3.10 weight percent for the outer region. Because of the rapid refueling capability discussed in Section 1.2.6

1547 176



STANDARD SCOPE OF RESAR - 41

NON STANDARD SCOPE OF RESAR - 41 DISCUSSED IN APPENDIX A TO THIS REPORT

Figure 1-1 RESAR - 41 NUCLEAR STEAM SUPPLY SYSTEM

NOTE: THIS FIGURE IS NOT INTENDED TO SHOW ALL INTERCONNECTIONS BETWEEN SYSTEMS

1547 177

below, utility applicants referencing RESAR-41 will have the option of refueling on an equilibrium six month cycle. This will allow approximately a 0.3 weight percent reduction in the required fuel enrichment.

The reactor design provides for reactivity control by means of full and part length rod cluster control assemblies, a burnable poison assembly and regulation of boric acid concentration in the reactor coolant. The burnable poison assembly will normally only be used for the initial core because of this core's higher reactivity. The design of the mechanical control rods consists of clusters of stainless-steel clad silver-indium-cadmium alloy absorber rods for insertion into the guide tubes in the fuel assemblies. There are two categories of full length control rod assemblies. Control assemblies will compensate for reactivity changes due to variations in operating conditions of the reactor, and the shutdown assemblies will have the necessary negative reactivity to provide an adequate shutdown margin. The control system for the full length control assemblies will allow the plant to accept step load changes of 10 percent and ramp changes of five percent per minute over the range of 15 tc 95 percent of full power under normal operating conditions. The function of the part length control assemblies will be to control axial neutron flux shape and axial xenon oscillations, should they occur.

Water circulating through the reactor vessel and core will serve as a neutron moderator, radiation shield, and coolant. The reactor vessel design is basically the same as that of current 3411 thermal megawatts Westinghouse plants except that the design provides for removal of control assemblies with the vessel head during refueling and a Roto-Lok closure stud design has been incorporated as part of the "Rapid Refueling" concept.

1.2.2 Reactor Coolant System

In the reactor coolant system, primary coolant will be circulated through the reactor vessel and core by four vertical, single stage, centrifugal pumps, one in each of the four cold legs. Significant proposed system operating parameters are listed below.

Normal Operating Pressure, pounds per square inch, gauge	2235
Reactor Power, megawatts, thermal	3800
Reactor Vessel Inlet Temperature, degrees Fahrenheit	559.8
Reactor Vessel Outlet Temperature, degrees Fahrenheit	623.8
Total Reactor Flow Rate, pounds per hour	144,700,000
Steam Pressure, pounds per square inch, gauge	1100
Total Steam Flow, pounds per hour	16,960,000

After being heated in the core, the coolant will be circulated through the four U-tube steam generators. It is here that heat will be transferred to the secondary system to form steam to be used to drive the turbine-generator. This coolant system design does not include loop stop valves.

1547 178

Reactor coolant pressure will be established and maintained by an electrically-heated pressurizer connected to the hot leg piping of one of the loops. The pressurizer will be designed to maintain a saturated steam bubble at the saturation temperature of the existing reactor coolant pressure. This will provide a surge volume to accommodate reactor coolant volume changes. Reactor coolant system overpressure protection will be provided through motor-operated relief valves and self-activated safety valves connected to the pressurizer vapor space.

1.2.3 Engineered Safety Features

The engineered safety features will consist of accumulator tanks, high head and low head safety injection systems, provisions for recirculation of the borated coolant after the end of the injection phase, and the emergency boration system. These systems will assure core cooling and protection for the complete range of postulated primary and secondary coolant pipe break sizes.

The accumulators and the high and low head safety injection systems will provide core protection for both large and small reactor coolant system ruptures. RESAR-41 consists of three independent safety injection trains. Each train will be connected to the refueling water storage tank and the containment sump. Each train will include one high head and one low head safety injection pump located external to the containment, one accumulator, and one residual heat removal heat exchanger located inside containment, and will be connected to the hot and cold legs of only one primary loop. For long term cooling following a loss-of-coolant accident, the low head safety injection pumps will recirculate the water collected in the containment sumps through the residual heat removal heat exchangers, through the core and out the break and back to the sumps.

Separate residual heat removal pumps, to be located inside the containment, will be used in conjunction with the residual heat removal heat exchangers for normal plant cooldown.

The emergency boration system will be designed to provide sufficient negative reactivity for safe shutdown capability in the event of any single steam pipe rupture or spurious lifting of a pressure relief valve. This design consists of a source of highly borated water and two parallel boron injection pumps. The pumps will inject the highly borated water into a common header which will connect to the cold leg of all four primary loops. As water is injected, excess water will be discharged from the reactor coolant system and circulated back through the emergency boration system.

1.2.4 Protection Systems

Plant protection systems designs are provided that will automatically initiate appropriate action whenever a monitored condition approaches pre-established limits. These protection systems will act to shut down the reactor, close isolation valves, and initiate operation of the engineered safety features should any or all of these actions be required.

The reactor trip system will shut down the reactor whenever unsafe operating limits are approached. It will consist of sensors which, when connected with analog circuitry consisting of two to four redundant channels, will monitor various plant parameters, and digital circuitry, consisting of two redundant logic trains, which will receive inputs from the analog protection channels to complete the logic necessary to automatically drop the control rod assemblies into the core and shut the reactor down.

The engineered safety features actuation system will consist of adequate instrumentation and controls to sense accident situations and initiate operation of the necessary engineered safety features. The system will consist of:

(1) Three to four redundant analog channels per plant parameter being monitored, and

(2) Two redundant digital logic trains which will receive inputs from the analog protection channels and actuate the engineered safety features.

The functions initiated by this system are:

(1) Reactor trip
(2) Safety injection
(3) Auxiliary feedwater flow
(4) Emergency boration flow
(5) Containment cooling
(6) Containment isolation
(7) Emergency diesel operation
(8) Containment spray
(9) Auxiliary supporting systems

1.2.5 Power Sources

The RESAR-41 design will require a minimum of three independent emergency on-site power supplies, each of which can supply the power requirements of one of the redundant sets of engineered safety features. The normal and emergency power supplies will be described in applications which reference RESAR-41.

1.2.6 Refueling

The RESAR-41 nuclear steam supply system will incorporate several new design features intended to reduce the time required for refueling. Westinghouse refers to the combination of these features as the "Rapid Refueling" concept. Significant aspects include quick disconnect head bolts (Roto-Lok), internals and control rods that will be removed with the head, an integral control rod drive mechanism cooling system and missile shield which will be removed with the head, and control rod drive mechanism power cables and instrumentation cables that do not need to be disconnected for head removal. In addition, the shutdown reactivity margin required for refueling will be reduced to 5 percent.

1.3 Comparison with Similar Designs

Some features in RESAR-41 represent new Westinghouse designs. However, many design 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 the RESAR-41 design. Where this has been done, the appropriate sections of this report identify the specific safety evaluation reports involved. These safety evaluation reports are available for public inspection at the Nuclear Regulatory Commission's Public Document Room at 1717 H Street, N.W., Washington, D.C. 20545.

To assist in better understanding the relationship of the RESAR-41 design to other Westinghouse designs, Westinghouse has presented a comparison of principal design features of RESAR-3 Consolidated Version with those for the RESAR-41 nuclear steam supply system in Tables 1.3 and 4.1-1 of RESAR-41. A listing of principal parameters and features is presented in Table 1-1 of this report. Some of the applications which reference RESAR-3 are those for the Catawba plants (docket numbers 50-413 and 414), the Vogtle plants (docket numbers 50-424 through 427), the Millstone 3 plant (docket number 50-423), the Comanche Peak plants (docket numbers 50-445 and 446), and the Seabrook plants (docket numbers 50-443 and 444). Our safety evaluation reports for these other applications are available for public inspection in the Public Document Room at 1717 H Street, N.W., Washington, D. C. 20545.

1.4 Requirements for Future Technical Information

Westinghouse has identified in Section 1.5 of RESAR-41 the verification test programs applicable to the RESAR-41 nuclear steam supply system. These programs are aimed at verifying the nuclear steam supply system design and confirming the design margins. The objectives and schedules for completion of these verification programs are given. A listing of the programs, their objectives, and their status is reproduced in Table 1-2 herein.

All test programs listed in Table 1-2, with the exception of the lower internals vibration test, are to be completed by the end of 1976. The test program schedule should provide adequate time for design changes should any of these test programs produce unexpected results. The final test to be performed is the reactor lower internals vibration test which is to be conducted on the first RESAR-41 plant to be completed.

In summary, the verification programs have been reviewed and we have concluded that (1) the test programs outlined in RESAR-41, if carried out as stated, will provide in a timely manner the necessary information to verify the design and safe operation of RESAR-41 nuclear steam supply systems and (2) in the event any of the programs provide unexpected results, appropriate restrictions on operation can be used and/or modifications in designs can be made to protect the health and safety of the public.

1.5 Summary of Principal Review Matters

Our technical review and evaluation of the RESAR-41 information submitted by Westinghouse included the principal matters discussed below.

We evaluated the design and expected performance of the systems and components important to safety to determine whether they are in accord with the Nuclear Regulatory Commission's General Design Criteria and Quality Assurance Criteria, and other applicable guides, codes and standards, and whether any departures from criteria, codes and standards have been identified and justified.

TABLE 1-1

COMPARISON OF

PRINCIPAL PARAMETERS AND DESIGN FEATURES

OF RESAR-41 AND RESAR-3

Parameter or Feature	RESAR-41	RESAR 3 (Consolidated Version)
Core Power Level (megawatts, thermal)	3800	3411
Number of Loops	4	4
Steam Flow from NSSS (pounds per hour)	16.96×10^6	15.16×10^6
Steam Pressure at Steam Generator Outlet (pounds per square inch, absolute)	1100	1000
Total Coolant Flow Rate (pounds per hour)	144.7×10^6	142.2×10^6
Net Electrical Output (megawatts, electric)	1295	1161
Average Linear Power (kilowatts per foot)	5.33	5.45
Maximum Linear Power for Normal Operation (kilowatts per foot)	13.3	13.6
Heat Flux Hot Channel Factor, Fq	2.50	2.50
No. of Safety Injection Trains	3	2
Injection Design Flow Rate (gallons per minute)		
High Head Pumps, ea.	800 @ 1225 psig*	150 @ 2800 psig*
Low Head Pumps, ea.	1400 @ 267 psig*	3000 @ 600 psig*
No. RHR Trains and Pumps	3	2
No. of Accumulators	3	4
Emergency Boration System Injection Pump Flow Rate, ea. (gallons per minute)	450 @ 215 psig*	425 @ 1500 psig*
Number of Fuel Assemblies	193	193
UO_2 Rods Per Assembly	264	264
Fuel Rod Array	17 x 17	17 x 17
Number of Grids Per Assembly	9	8
Fuel Weight as UO_2 (pounds)	243,765	222,739
Fuel Rod Clad Thickness (inches)	0.0225	0.0225
Number of Rod Cluster Control Assemblies Full/Part Length	61/8	53/8
Number of Absorber Rods Per Cluster	24	24
Core Diameter (inches)	132.7	132.7
Core Average Active Fuel Height (inches)	164	144
Fuel Enrichment (weight percent)		
Region 1	2.10	2.10
Region 2	2.60	2.26
Region 3	3.10	3.10

*Pounds per square inch, guage

TABLE 1-2

VERIFICATION TEST PROGRAMS

Test	Purpose	Status
I. Verification Tests (17 x 17 XLR)		
Rod Cluster Control Spider Tests	Verify structural adequacy	Completed
Grid Tests	Verify structural adequacy	Completed
Departure from Nucleate Boiling	Determine effect of 17 x 17 geometry on departure from nucleate boiling heat flux	Complete in 1975
Single Rod Burst Test	Determine maximum flow blockage	Completed
Fuel Assembly Structural Tests	Determine mechanical strength of assembly	Completion in 1976
Prototype Assembly Tests	Demonstrate performance of 17 x 17 XLR fuel assembly	Completion in 1976
Lower Internals 1/7 Scale Tests	Determine vibration and flow forces on support columns	Completion in 1975
Lower Internals Vibration Tests	Verify 1/7 scale test results	Completion in 1980
II. Inpile Fuel Densification	Define material characteristics and manufacturing processes	In Progress
III. Loss-of-Coolant Accident Heat Transfer Tests		
G-Loop Tests	Simulate blowdown in fuel assembly	Completion in 1975
IV. Rapid Refueling Hardware Tests		
Prototype Closure System Test	Verify design	Completed
Prototype Control Rod Drive Mechanism with Holdout System Device Test	Verify safety and reliability of system	Completion in 1975

1547 183

We evaluated the expected response of the nuclear steam supply system to anticipated operating transients and to a broad spectrum of postulated accidents and 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 and determined that the calculated potential offsite doses that might result in the very unlikely event of their occurrence would be within the Commission's guidelines for site acceptability, as given in 10 CFR Part 100, for typical sites when the RESAR-41 nuclear steam supply system design is combined with an acceptable balance-of-plant design at an acceptable site.

1.6 Design Modifications as a Result of Staff Review

During the review of RESAR-41, numerous meetings were held with representatives of Westinghouse and its consultants to discuss the proposed design and the technical material submitted in the application. A chronological listing of the meetings and other significant events in our review of RESAR-41 is given in Appendix B to this report. During the course of the review Westinghouse proposed, or we requested, a number of technical and administrative changes. These changes are described in various amendments to the application. We have listed below the more significant modifications that have resulted from our review. Included are references to the sections of this report where each matter is discussed more fully.

(1) Modification to the emergency core cooling system to lockout power to certain motor operated valves to protect against spurious movement of the valves. (Sections 6.3.4, 7.3.1, and 7.6.3)

(2) Modification to the emergency core cooling system to provide completely automatic transfer from the injection mode to the recirculation mode. (Sections 6.3.2 and 7.3.3)

(3) Modification to the source term assumptions to include a more conservative fraction of the fuel assumed to be releasing fission products. (Section 11.1)

(4) Modification to the waste gas processing system to control releases by a monitor in the system discharge line. (Appendix A Section 11.4)

(5) Modification to the spent fuel pool cooling system to meet seismic Category I requirements. (Appendix A Section 9.1)

1.7 Outstanding Issues

We have identified certain outstanding issues in our review most of which will require that Westinghouse provide additional information to confirm that the proposed design will meet our requirements. These items are listed below and are discussed further in sections of this report as indicated. We are continuing to review these issues and expect to have most of them resolved by mid-August and will require all of them to be resolved prior to issuance of the Preliminary Design Approval.

(1) We have evaluated the interface information contained in RESAR-41 through Amendment 16 and find it to be inadequate. Westinghouse acknowledges that additional interfaces should be provided. They have undertaken to conduct an accelerated short-term program to supplement the interface information already provided in an effort to identify essentially all (i.e., greater than 95%) of the safety related interfaces. The Westinghouse program will reassess the safety analyses, transient analyses, normal operations and review the major technical information transmitted by Westinghouse to its customers which is contained in a set of documents referred to as a Standard Information Package. Westinghouse agreed to review this information in detail to identify and define additional interface information for submittal to the staff. We will review the additional interface information developed by Westinghouse in its short-term program and conduct an audit of selected portions of the information used by Westinghouse in its program including its Standard Information Package. The results of our evaluation and audit will be reported in a subsequent report.

On a longer-term basis, Westinghouse will complete the upgrading of the interface information on a more comprehensive basis. We will complete our review of this information prior to issuance of the Preliminary Design Approval.

(2) Westinghouse must submit the loss-of-coolant analysis in accordance with the acceptance criteria of Appendix K of 10 CFR Part 50. This analysis is presently scheduled to be submitted by August 8, 1975. (Section 6.3.5)

(3) Westinghouse has submitted the analysis of the effect of anticipated transients without scram on the RESAR-41 design for review by the staff. Our generic review of this information is underway and we will require that any design changes that are required as a result of our review, when it is completed, be implemented in the RESAR-41 design. (Section 15.4.7)

(4) We require that Westinghouse modify the design of the chemical and volume control system to allow adequate time for operator action during a boron dilution accident while refueling or during startup. (Section 15.3)

(5) We require that Westinghouse provide an evaluation of the effect on plant safety of a postulated accident in which the reactor vessel head is dropped while it is being moved for refueling or maintenance operations. (Section 5.4.6)

(6) Westinghouse submitted, on June 24, 1975, an analysis of the consequences of a postulated failure of a single Roto-Lok head closure stud. We are presently reviewing this analysis. (Section 5.4.6)

1547 185

(7) We require that the residual heat removal system be modified to meet our requirements. These requirements include isolation valve interlocks and alarms, cooldown capacity, and electrical and instrumentation design criteria. (Section 5.4.3 and 7.4.1)

(8) We require that Westinghouse clarify the methods and criteria that will be used in the seismic analysis of systems and components supported from two or more locations with relative displacements and different response spectra. (Section 3.5.2)

(9) We require that Westinghouse commit to a satisfactory program for demonstrating the seismic and environmental testing and qualification of instrumentation and electrical equipment which will meet the requirements of IEEE Std 344-1975 and IEEE Std 323-1974 in the near future in a manner acceptable to the staff. (Sections 3.7, and 7.6.1)

(10) We require that Westinghouse provide adequate design criteria to show how the proposed temperature monitoring system for the emergency boration system will meet the single failure criterion requirements for safety systems. (See Section 7.3.4)

(11) We require that the nuclear instrumentation neutron detectors be qualified for the worst case environment in the containment for which they are expected to operate, in accordance with the requirements of IEEE Std 323-1974. (See Section 7.6.1 (5))

1547 186

2.0 SITE CHARACTERISTICS

Since the RESAR-41 application does not include any specific site location, the specific site parameters have not been discussed by Westinghouse. These will be addressed by applications which reference RESAR-41.

As discussed in Section 3.6.1, we have found that, of those sites previously evaluated by the staff, approximately 90 percent had a design safe shutdown earthquake of 0.4g or less which is the seismic design value for the RESAR-41 nuclear steam supply system.

Applicants referencing RESAR-41 must show that the site seismic design parameters for their site are within the design envelope of RESAR-41 and that the system safety related equipment is adequately protected from site related hazards.

1547 187

3.0 DESIGN CRITERIA FOR SYSTEMS AND COMPONENTS

3.1 Conformance with the General Design Criteria

Westinghouse has stated that the RESAR-41 nuclear steam supply system will be designed in accordance with the Commission's General Design Criteria for Nuclear Power Plants. On the basis of our review of the documentation supporting this commitment, we have concluded that the RESAR-41 nuclear steam supply system can be designed to meet the requirements of the General Design Criteria. Discussions regarding compliance with each criterion are presented in Section 3.1 of RESAR-41.

3.2 Classification of Components and Systems

3.2.1 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 Westinghouse'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 Westinghouse of quality groups to those sections of systems required to perform safety functions.

Westinghouse has applied the classification system of the American Nuclear Society (Safety Classes 1, 2, 3 and Non-Nuclear Safety), which corresponds to the Commission's Quality Groups A, B, C and D in Regulatory Guide 1.26, 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 the safe shutdown condition, and (3) to contain radioactive material. These fluid systems have been classified in an acceptable manner in Table 3.2-1 and on system piping and instrumentation diagrams in RESAR-41.

The basis for our acceptance has been conformance of Westinghouse'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, staff technical positions and industry standards.

We conclude that 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.

3.2.2 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 necessary safety functions. These plant features are those that will be necessary to assure (1) the integrity of the reactor coolant pressure boundary (2) the capability to shutdown 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 information presented by Westinghouse identifying those systems and components (including their supports) which are within the scope of RESAR-41 and are important tc safety and are designed to withstand, without loss of function, the effect of a safe shutdown earthquake.

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 Table 3.2-1 of RESAR-41. 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.

We conclude that systems and components important to safety will be designed in accordance with seismic Category I requirements which provides reasonable assurance that in the unlikely event of a severe seismic event, the plant will perform in a manner providing adequate safeguards of the health and safety of the public.

The basis for acceptance in our review has been conformance of the Westinghouse 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, staff technical positions, and industry standards.

3.3 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. We have reviewed the RESAR-41 systems and components to be protected from missiles. The review included missile sources and internally generated missiles associated with component overspeed failures and missiles that could originate from high-pressure system ruptures. Also, we reviewed the acceptability of the design analysis and criteria used for structures or barriers that protect essential systems and components from missiles.

Section 3.5 of RESAR-41 describes the characteristics of postulated missiles which may occur inside the containment from failure of equipment within the scope of the RESAR-41 nuclear steam supply system. These missiles include control rod drive mechanism missiles, valve missiles, piping temperature sensing element assembly missiles, reactor coolant pump temperature element missiles, pressurizer instrument well missiles, and pressurizer heater missiles. Characteristics of these postulated missiles are identified as interface information to be used by balance of plant designers in providing adequate missile protection. Our evaluation of the control rod drive mechanism missile shield design is contained in Section 5.4.8.4.

RESAR-41 does not include the design analysis and criteria used for other structures or barriers that will protect essential systems and components from missiles generated internally or outside the containment structure. We did not include turbine missiles in our review since the turbine placement and its design and operating characteristics, as well as overall plant layout and structural characteristics have to be considered in assessing turbine missile damage hazards. Similarly, we did not include tornado-generated missiles in our review of RESAR-41. Therefore, applicants referencing RESAR-41 must consider the effects of postulated missiles and provide the necessary protection to safety related components.

We conclude that the RESAR-41 design with regard to protection from missiles conforms to the Commission's Regulations and to applicable Regulatory Guides, staff technical positions, and industry standards. Conformance to these requirements constitutes an acceptable basis for satisfying the applicable requirements of Criteria 2 and 4 of the General Design Criteria.

3.4 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 RESAR-41 to determine that the design will accommodate the effects of postulated pipe breaks and jet impingement from piping systems. Westinghouse states that the criteria to be employed for determination of the systems to be evaluated, the locations and types of piping breaks which will be postulated and the protection measures against pipe whip for the reactor coolant system piping, will be in accordance with Westinghouse Topical Report, WCAP-8082, "Pipe Breaks for the LOCA Analysis of the Westinghouse Primary Coolant Loop."

We have reviewed and accepted WCAP-8082 by letter to Westinghouse dated May 22, 1974 for purposes of specifying pipe break locations in the reactor coolant system piping. Our approval was based on the finding that implementation of the criteria specified in WCAP-8082 provides a level of protection equivalent to that resulting from the application of the criteria of Regulatory Guide 1.46.

The validity of the criteria contained in WCAP-8082 will be dependent on the dynamic response of the overall reactor coolant system as mounted and constrained by the component supports. The detailed design of the reactor coolant system layout and component supports may vary in actual plants incorporating the RESAR-41 nuclear steam supply system. Therefore, we will require that each applicant referencing RESAR-41 supplement the information provided in RESAR-41 on the determination of the type of breaks postulated for the reactor coolant system piping. Each such applicant will be required to demonstrate that its specific reactor coolant system layout and component support designs lie within the design envelope of WCAP-8082.

We have determined that the structural characteristics of the system will be appropriately considered by the analytical methods and procedures that will be used to determine the most probable type of pipe break at a particular location and to determine pipe motions subsequent to rupture and the pipe-whip restraint dynamic interactions. The pipe-whip restraints will be designed to withstand the resultant loadings and remain intact to assure the protection of essential structures, systems and components.

The system layout and component support provisions of RESAR-41 for protection against the dynamic effects associated with pipe ruptures and the resulting discharging coolant provide acceptable assurance that the conditions and safety functions described below will be accommodated and assured when combined with an approved balance of plant design.

(1) The magnitude of the design basis loss-of-coolant accident cannot be aggravated by potential multiple failures of piping.
(2) The emergency core cooling systems can be expected to perform their intended function.
(3) Systems and components important to safety will be appropriately protected.

This assurance will apply 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.

Westinghouse has provided as interface information the pressures and temperatures of the fluids in systems within their scope. In addition, they have identified the RESAR-41 nuclear steam supply system safety-related equipment which must be protected from the effects of postulated pipe ruptures. The criteria and design bases that will be used to preclude the consequences of postulated pipe ruptures will be reviewed on applications which reference RESAR-41.

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.

3.5 Seismic Design

Criterion 2 of the General Design Criteria requires that systems and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes. We reviewed the RESAR-41 systems and components important to safety to determine their ability to withstand the effects of earthquakes.

3.5.1 Seismic Input

We have reviewed and evaluated the seismic design input criteria that will be employed by Westinghouse with respect to all Category I systems and components. The seismic design will be based on a maximum horizontal ground acceleration of 0.4 times normal gravitational acceleration at zero period for the safe shutdown earthquake. The ground response spectra and damping values specified in RESAR-41 are consistent with the recommendations of Regulatory Guides 1.60 and 1.61. These design response spectra will be normalized to a specific site by scaling to the proper maximum site ground acceleration. In the event that Westinghouse later proposes to use higher damping values in RESAR-41 than those recommended by Regulatory Guide 1.61, they must be justified by testing programs and reviewed by the staff.

In addition to the site interface of the maximum horizontal ground acceleration, the building designs must meet certain criteria. The seismic design of the RESAR-41 nuclear steam supply system Category I systems and components assumes that the plant buildings will be designed to the requirements described in Figure 3-1 below. We have reviewed these requirements and conclude that if they are met by an applicant referencing RESAR-41, the seismic design of the RESAR-41 nuclear steam system will be adequate to assure safe operation during the safe shutdown earthquake specified above. Of those sites previously evaluated by the staff, approximately 90 percent had a design safe shutdown earthquake characterized by a maximum horizontal ground acceleration equal to or less than 0.4 times normal gravitational acceleration.

The synthetic time history used for Category I plant component design will be adjusted in amplitude and frequency to envelope the design response spectra of Regulatory Guide 1.60. This will be provided in applications referencing RESAR-41.

Conformance with the recommendations of Regulatory Guides 1.60 and 1.61 provides reasonable assurance that earthquake accelerations imposed on Category I systems and components are adequately defined to assure a conservative basis for the design of such systems and components to withstand the consequent seismic loadings. Compliance with these guides constitutes an acceptable basis for satisfying the provisions of Criterion 2 of the General Design Criteria.

3.5.2 Seismic System And Subsystem Analysis

Modal response spectrum and time history methods for multi-degree-of-freedom systems will form the bases for analyses of all major Category I systems and components. Governing response parameters will be combined by the square root of the sum of squares when the modal response spectrum method is used. Corrective terms involving double summation of products of responses will be used for modes with closely spaced frequencies.

Three components of seismic motion will be considered, two horizontal and one vertical. The total response will be obtained by the square root of the sum of squares of the three components for the modal response spectrum method or by algebraic combination at each time step for the time history method.

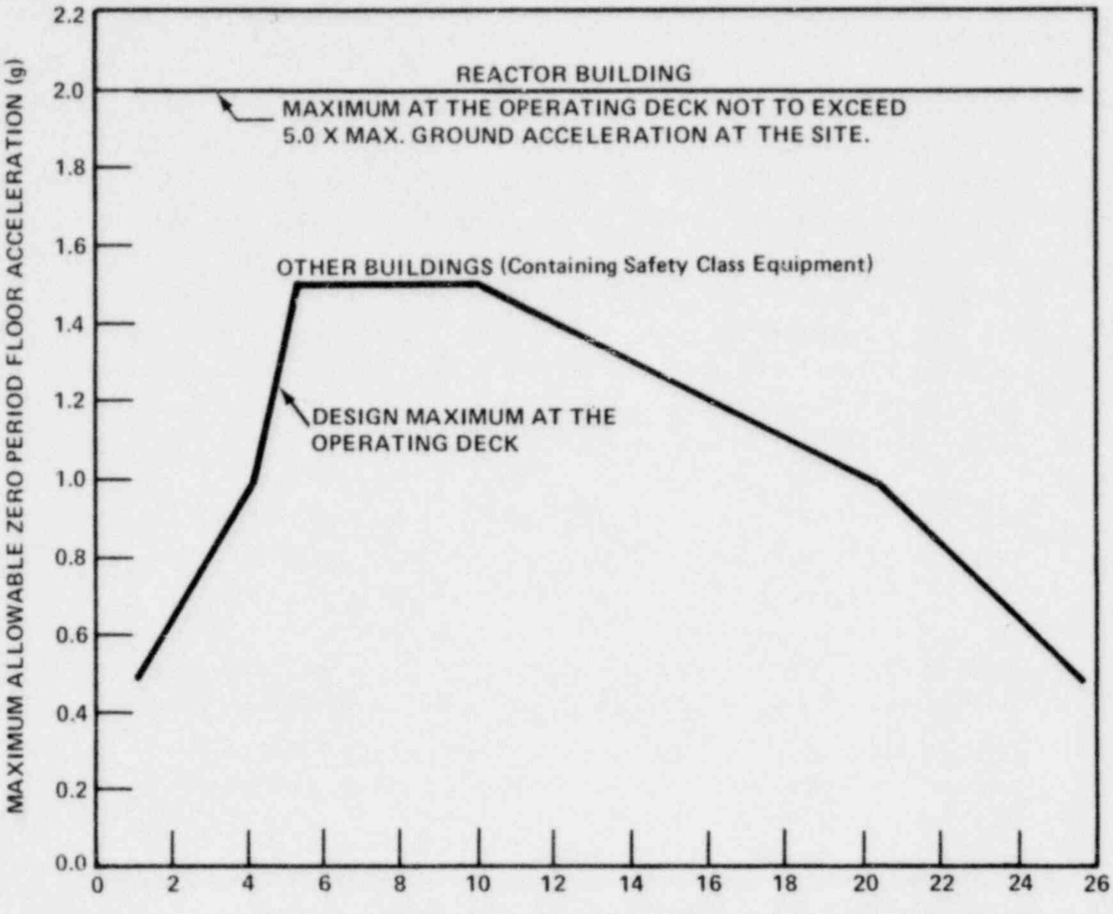


Figure 3-1 INTERFACE INFORMATION FOR BOP BUILDING SEISMIC DESIGN

1547 193

Floor response spectra inputs to be used for design and test verification of structures, systems and components will be described in applications referencing RESAR-41. Dynamic analysis of vertical seismic systems will be employed for all systems and components where dynamic amplifications in the vertical direction are significant. System and subsystem analyses will be performed on an elastic basis.

However, for the case where a component or system is supported from two or more locations with relative displacements and different response spectra, Westinghouse has not clearly committed to comply with the staff position.

We require that, where the response spectrum method is used, the procedure involves two steps. First, a static analysis must be made by considering the maximum relative displacement between support points; i.e., the design displacement is obtained by adding in an absolute manner. Second, a dynamic analysis must be made assuming no relative displacement between support points by using the worst floor response spectrum when the support points are in the same structure or the enveloped floor response spectrum when the support points are in separate structures. Results from these two steps, static and dynamic, are to be combined in an absolute manner. (For piping components, these results should be used in accordance with Section III of the Code, Paragraphs NB-3652 and NB-3653-1.)

For interconnected components and piping systems, Westinghouse mentions "proper phasing" for relative displacements instead of adding in an absolute manner. When the response spectrum method is used, the phase relationship of responses is lost and the statement of "proper phasing" becomes confusing. We require that the relative displacement be obtained by adding in an absolute manner. We also require that the applicable paragraph or subsection of Section III of the Code be clearly specified.

For piping systems, we also require that Westinghouse comply with Paragraph NB-3652 as well as NB-3653 of Section III of the Code, which they have already committed to.

We conclude that upon satisfactory resolution of the methods used to analyze components and systems supported from two or more locations with relative displacements and different response spectrum, the dynamic methods and procedures for seismic systems analyses proposed by Westinghouse will provide an acceptable basis for siesmic design.

3.5.3 Seismic Instrumentation Program

RESAR-41 does not include a seismic instrumentation program. A description of this program must be provided in applications referencing RESAR-41.

1547 194

3.6 Mechanical Systems and Components

3.6.1 Dynamic Analysis and Testing

3.6.1.1 Evaluation

Criterion 1 of the General Design Criteria requires that structures, 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 reviewed the RESAR-41 criteria, testing procedures, and dynamic analyses to be employed to ensure structural and functional integrity of piping systems, mechanical equipment, and reactor internals under vibratory loadings, including those due to fluid flow and postulated seismic events.

We reviewed the preoperational piping vibrational and dynamic effects testing program to be conducted during startup functional testing on all safety related piping components and component supports classified as American Society of Mechanical Engineers (ASME) Class 1. The purpose of these tests will be to confirm that these components and supports have been designed to withstand the dynamic loadings from operational transient conditions that will be encountered during services as required by Section III, NB-3622.3 of the ASME Boiler and Pressure Vessel Code (hereafter the Code) code. These code requirements 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. Westinghouse has committed to perform a preoperational piping vibrational and dynamics testing program in accordance with Section III, NB-3622.3 of the Code. The preoperational vibrational testing of Class 1 auxiliary piping and Class 2 piping will be reviewed in applications referencing RESAR-41.

Westinghouse had provided no guidelines to determine where and how the visual observations should take place, including the methods and procedures to determine whether the observed vibration intensity is excessive. We required that this information be provided by Westinghouse or that they commit to providing it for the final design review. As a result, by Amendment 16 to RESAR-41, Westinghouse committed to provide this information in the final design application. We find this commitment to be acceptable.

The testing programs should include development of loads similar to those experienced during reactor operation and must be consistent with staff positions concerning preoperational piping dynamics effects test programs. Selected locations in the piping system that will be subjected to visual inspection and measurements (if needed) as performed by the piping designer during these tests must be provided. For each of these selected locations, the allowable deflection (peak-to-peak) criteria that will be applied to establish that the stress and fatigue limits are within the design levels must be provided. If vibration is noted beyond the acceptance levels set by the criteria discussed above, corrective restraints will be designed, incorporated in the piping system analysis and installed upon approval by the staff. If, during the test, the piping systems restraints are determined to be inadequate or damaged, corrective restraints will be installed and another test performed to determine that the vibrations have been reduced to an acceptable level. These corrective designs will also require the approval of the Nuclear Regulatory Commission.

We have reviewed the analysis of the dynamic responses of structural components within the reactor vessel that will be caused by operational flow transients and the safe shutdown earthquake. The purpose of this analysis is to predict the vibration behavior of the dynamic responses, such that the input forcing functions and the level of response can be estimated before conducting the preoperational vibration test of a prototype (first of a design) reactor. Our review included the method of analysis, the specific locations for response calculation, the considerations to define the mathematical model, the interpretation of analytical results, the acceptance criteria, and the verification of predictions via tests.

The integrity of the reactor internals in service is essential to assure the proper positioning of reactor fuel assemblies and unimpaired operation of the control rod assemblies to permit safe reactor operation and shutdown. With regard to flow-induced vibration testing of reactor internals, Westinghouse stated that they will designate the first RESAR-41 plant incorporating a RESAR-41 nuclear steam supply system to feature the modified internals design for the XLR (14 ft) fuel as the prototype plant to be tested in accordance with Regulatory Guide 1.20. The purpose of this test will be to demonstrate that flow-induced vibrations similar to those expected during operation will not cause large unanticipated flow-induced vibrations or structural damage. We will review the details of the testing program at the Final Design Approval review stage including a list of flow modes, a description of test procedures, methods used to process and interpret the measured data, the scheme to implement the visual inspection, and a comparison of the test results with the analytical predictions.

After a satisfactory prototype is established, additional confirmatory vibration testing and monitoring programs with subsequent visual inspection will be conducted for subsequent RESAR-41 plants 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.

3.6.1.2 Conclusion

Except for the guidelines for conducting the visual observations which we will review at the final design review, the preoperational vibration test program which will be conducted during startup and initial operation on all safety-related piping systems, restraints, components, and component supports classified as ASME Class 1, is an acceptable program. Implementation of the test program will provide adequate assurance that the piping and piping restraints of the system have been designed to withstand vibrational dynamic effects due to valve closures, pump trips, and other operating modes associated with the design basis operational transients. The planned tests will develop loads similar to those experienced during reactor operation. Compliance with this test program constitutes an acceptable basis for fulfilling the applicable requirements of Criterion 15 of the General Design Criteria.

The preoperational vibration program planned for the reactor internals provides an acceptable basis for verifying the design adequacy of these internals under test loading conditions comparable to those that will be experienced during operation. The combination of tests, predictive analysis, and post-test inspection provides adequate assurance that the reactor internals will, during their service lifetime, withstand the flow-induced vibrations of reactor operation without loss of structural integrity. The conduct of the preoperational vibration tests will be in conformance with the provisions of Regulatory Guide 1.20 and therefore will constitute an acceptable basis for demonstrating design adequacy of the reactor internals, and satisfy the applicable requirements of Criteria 1 and 4 of the General Design Criteria.

The dynamic system analysis to be performed provides an acceptable basis for confirming the structural design adequacy of the reactor internals and unbroken piping loops to withstand the combined dynamic loads of postulated loss-of-coolant accidents and the safe shutdown earthquake. Westinghouse is aware of the need to properly interpret all potential dynamic loads for the design including those shock-type loads that can be developed for specific pipe rupture loads at specific locations. The analysis will provide adequate assurance 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, and that the resulting deflections or displacements of any structural elements of the reactor internals will not distort the reactor internals geometry to the extent that core cooling may be impaired.

The methods to be used for component analysis have been found to be compatible with those used for the systems analysis. The proposed combinations of component and system analyses are, therefore, acceptable. The assurance of structural integrity of the reactor internals under loss-of-coolant accident conditions for the most adverse postulated loading event provides added confidence that the design will withstand a spectrum of lesser pipe breaks and seismic loading events. Accomplishment of the dynamic system analysis constitutes an acceptable basis for satisfying the applicable requirements of Criteria 2 and 4 of the General Design Criteria.

3.6.2 Analysis Methods for Seismic Category I Components

3.6.2.1 Evaluation

Criterion 2 of the General Design Criteria requires that structures, systems and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, etc.

We have reviewed the RESAR-41 information concerning design transients and methods of analysis for seismic Category I components, including both those designated as Class 1, 2, 3, or component supports under the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III (hereafter "the Code"), and component supports, reactor internals, and other components not covered by the Code.

We reviewed the list of transients to be used in the design and fatigue analysis of all Code Class 1 components, and of component supports and reactor internals within the reactor coolant pressure boundary. The number of events for each transient is included in RESAR-41 along with assurance that the number of load and stress cycles per event have been and will be properly taken into account. All design transients such as startup and shutdown operations, power level changes, emergency and recovery conditions, switching operations (i.e., startup or shutdown of one or more coolant loops), control system or other system malfunctions, component malfunctions, transients resulting from single operator errors, inservice hydrostatic tests, seismic events, etc., that are contained in the Code-required "Design Specifications" for the components of the reactor coolant pressure boundary are specified. All transients or combinations of transients are categorized with respect to the plant operating conditions identified as "normal," "upset," "emergency," or "faulted."

The RESAR-41 transient conditions selected for equipment fatigue evaluation are based upon a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from those conditions. To a large extent the selection of these specific transient operating conditions is based upon engineering judgment and experience. Partial guidance on the selection of these transients can be found in Regulatory Guide 1.48.

We find that the design transients, plant conditions, and loading combinations specified provide a complete basis for the design of the reactor coolant pressure boundary for all conditions and events expected over the service lifetime of the plant and satisfy the requirements of Criteria 14 and 15 of the General Design Criteria.

We reviewed the computer programs that will be used in dynamic and static analyses to determine the structural and functional integrity of seismic Category I Code and non-Code items and the analyses to determine stresses. The design control measures were reviewed to determine compliance with Appendix B of 10 CFR Part 50. These computer programs are provided in RESAR-41 including a brief description of each program and the extent of its application.

As required by Appendix B of 10 CFR Part 50 we determined that the applicability and validity of the above computer programs has been shown by one of the following methods:

(1) The computer program is recognized and widely used with a sufficient history of successful use to justify its applicability and validity without further demonstration by the applicant.

(2) The computer program's solutions to a series of test problems, with accepted results, have been demonstrated to be substantially identical to those obtained by a similar program which meets the criteria of (1) above.

(3) The program's solutions to a series of test problems are substantially identical to those obtained by hand calculations or from accepted experimental test or analytical results published in the technical literature.

We reviewed the inelastic stress and deformation design limits specified by Westinghouse for Code Class 1 components, and for component supports, reactor internals, and other non-Code items, and the methods of analysis used to calculate the stresses and deformations resulting from faulted condition loadings.

1547 198

Westinghouse employs an inelastic method of analysis to evaluate the design of safety related Code Class 1 components, component supports, reactor internals and other non-Code items for the faulted plant condition (NB-3225 and Appendix F of the Code).

The design analysis or test methods and associated stress or load allowable limits that will be used in evaluation of faulted conditions are those that are defined in Appendix F of the Code.

3.6.2.2 Conclusion

The criteria used in the methods of analysis that Westinghouse will employ in the design of all seismic Category I Code Class 1, 2, 3, and component supports, and, components, component supports, reactor internals, and other non-Code items are in conformance with established technical positions and criteria which are acceptable to the staff.

The use of these criteria in defining the applicable design transients, computer codes used in analyses, analytical methods, and experimental stress analysis methods will provide assurance that the stresses, strains, and displacements calculated for the above-noted items will be as accurate as the current state-of-the-art permits and will be adequate for the design of these items.

3.6.3 Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

3.6.3.1 Discussion

Criterion 1 of the General Design Criteria requires that structures, 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 the RESAR-41 information concerning the structural integrity and operability of pressure-retaining components, their supports, and core support structures which are designed in accordance with the rules of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III (hereafter "the Code").

We reviewed the plant and component operating conditions, design transients, and design loading combinations considered for each system that provides the basis for the design of Code Class 1, 2, 3 and component support items for all conditions and events expected over the service lifetime of the plant.

The acceptability of the combination of loading conditions and design transients applicable to the design of Code constructed items within a system, including the categorization of the appropriate plant and component operating condition for each initiating event (i.e., loss-of-coolant accident, safe shutdown earthquake) used with each loading combination, are judged by comparison with the positions stated in Regulatory Guide 1.48, and with appropriate standards acceptable to the staff developed by professional societies and standards organizations. The corresponding stress limits applied to the design of Code-constructed items are specified in the appropriate subsections of Division 1 of

1547 199

Section III of the Code. The need for more conservative stress limits for active components and their supports is considered in the context and with the other features of the operability assurance program.

The objective in reviewing the loading combinations and stress limits employed by Westinghouse in the design of Code Class 1, 2, 3, and component support items were to confirm that each of the plant operating conditions have been included, that the loading combinations and design transients applicable to the design of Code constructed items and the categorization of proposed operating conditions are appropriate, that the design stress levels associated with each imposed loading combination are low enough to provide adequate margins with respect to the structural integrity of the item, and that for active components and their supports, stress levels are considered in the operability assurance program.

3.6.3.2 Code Class 2 and 3 Components

All safety-related Code Class 2 and 3 systems and components, 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. The specified design basis combinations of loadings (of the safety-related Code Class 2 and 3 pressure-retaining components in systems classified as Category I) provide reasonable assurance that in the event that 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 would not be expected 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 RESAR-41 design load combinations and associated stress and deformation limits specified for all Code Class 2 and 3 components constitute an acceptable basis for the design satisfying Criteria 1, 2 and 4 of the General Design Criteria and are consistent with staff positions.

3.6.3.3 Analytical and Empirical Methods for the Design of Pumps and Valves

The operation of certain pumps and valves is relied upon to shut down the plant or mitigate the consequences of an accident. These are termed "active" pumps and valves. Certain of these active pumps and valves may be required to function coincidentally with the postulated accident or event. Other active pumps and valves may be required to function only after a postulated accident or event has occurred. We reviewed the procedures for demonstrating the operability of active pumps and valves during or after postulated accidents or natural events.

The objective of the review of the pump and valve operability assurance program was to determine whether the program submitted will assure the operability of a component which is required to function to shut down the plant or mitigate the consequences of an accident. A commitment to adopt a program which satisfactorily meets the acceptance criteria was submitted by Westinghouse.

1547 200

The proposed operability assurance program applies to active pumps and valves in seismic Category I systems including those which may be classified as ASME Code Class 1, 2 and 3. The program will demonstrate the ability 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 component operability assurance procedures specified by Westinghouse constitute an acceptable basis for meeting the requirements of Criteria 1, 2 and 4 of the General Design Criteria as related to operability of ASME Code Class 1, 2 and 3 active pumps and valves.

3.6.3.4 Pressure Relieving Devices

The design criteria for the installation of the RESAR-41 pressure-relieving devices are not within the scope of RESAR-41 and therefore, findings as to acceptability will be made for specific applications during individual review of application referencing RESAR-41. We will require that a level of protection equivalent to that resulting from implementation of Regulatory Guide 1.67 be maintained.

3.6.3.5 Component Support Design

The primary system component support designs must provide adequate margins of safety under all plant operating conditions.

The acceptability of the combinations of loading conditions and design transients applicable to the design of component supports within a system, including the categorization of the appropriate plant and component support operating condition for each initiating event, (e.g., loss-of-coolant accident and safe shutdown earthquake) used with each loading combination, was judged by comparison with the positions stated in Regulatory Guide 1.48, and with appropriate standards developed by professional societies and standards organizations that are acceptable to the staff. The corresponding stress limits applied to the design of component supports will be as specified in Subsection NF of Division 1 of Section III of the Code. The need for more conservative stress limits for active component supports was considered in context with the other features of the operability assurance program.

In addition, for the component support that affects the operability requirements of the supported component, deformation limits were also specified. The deformation limits for active component supports will be compatible with the operability requirements of the components supported. In establishing allowable deformations, the possible movements of the support base structures were taken into account.

The objective in the review of component supports was to determine that adequate attention has been given the various aspects of design and analysis, so that there is assurance as to support structural integrity and as to operability of active components that interact with component supports.

The specified design basis loading combinations used for the design of safety-related ASME Code Class 1, 2, and 3 component supports in systems classified as seismic Category I provide assurance that in the event of an earthquake or an upset, emergency, or faulted plant transient, the resulting combined stresses imposed on system components will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative basis for the design of support components to withstand the most adverse combination of loading events without loss of structural integrity or supported component operability. The design load combinations and associated stress and deformation limits specified for ASME Code Class 1, 2, and 3 component supports constitute an acceptable basis for satisfying applicable portions of Criteria 1, 2 and 4 of the General Design Criteria.

3.6.3.6 Interfaces

Westinghouse has delineated the responsibilities between the architect-engineer and Westinghouse for ASME Code Class 1 branch lines and certain ASME Code Class 2 and 3 systems. In the case of Class 1 branch lines the Westinghouse reactor coolant loop design and analysis groups will transmit drawings of the reactor coolant loop and also deflections at the nozzle centerline intersection of Class 1 branch lines with respect to the reactor coolant loop to the balance of plant designer for a specific plant or standardized balance of plant. The deflections are for all loading cases, which are derived from both static and dynamic analysis. Westinghouse also states that the balance of plant designer will provide Westinghouse with the loads exerted on all reactor coolant loop piping branch nozzles for all loading cases. Other systems and classes are similarly covered. To complete the process, Westinghouse will then review the loads furnished by the balance of plant designer to assure they are within the RESAR-41 standardized design envelope, thus achieving compatibility between nuclear steam supply system and components designed by the balance of plant designer at the connection interface and still retaining the validity of the standardized plant concept. We find these procedures acceptable and consistent with what is done for custom plants.

3.6.3.7 Inservice Testing of Pumps and Valves

To ensure that all ASME Code Class 1, 2 and 3 pumps and valves within Westinghouse scope of responsibility will be in a state of operational readiness to perform the necessary safety functions throughout the life of the plant, Westinghouse has committed to design the pumps and valves such that a test program will provide baseline preservice testing information and periodic testing schedule.

Westinghouse has stated that the inservice test program for all Code Class 1, 2 and 3 pumps and valves meets the requirements of the ASME Code, Section XI, Subsections IWP and IWV, respectively.

Compliance with these Code requirements constitutes an acceptable basis for satisfying the applicable portions of Criteria 37, 40, 43 and 46 of the General Design Criteria.

1547 202

3.7 Seismic Qualification of Category I Instrumentation and Electrical Equipment

Criterion 2 of the General Design Criteria requires that structures, systems and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes, etc.

Proper functioning of certain instrumentation and electrical equipment in the event of a safe shutdown earthquake is essential to assure the capability of such equipment to initiate required protective actions including, for example, operation of engineered safety features and standby power systems. The information presented in RESAR-41 relative to equipment which has been previously tested to the requirements of IEEE Std 344-1971, "Seismic Qualification of Class I Electrical Equipment for Nuclear Power Generating Stations," is not acceptable as presented. This equipment will be acceptable if it can be characterized by the following:

(1) The nature of the equipment is such that the resonance frequencies can be validly identified by testing.
(2) None or not more than one of the resonance frequencies is below 33 Hertz, and
(3) The absence of significant directional coupling (from the standpoint of response or failure mode) can be determined by inspection or by comparison with tests of similar equipment.

Since equipment not qualified under the above criteria may not remain functional under seismic excitation due to multi-frequency response or because of directional coupling, we require that Westinghouse's qualification program contain these criteria, or other consistent criteria which we find acceptable, prior to issuance of the Preliminary Design Approval.

We require that prior to issuance of a Preliminary Design Approval, Westinghouse commit to a satisfactory program for demonstrating the seismic qualification of Category I instrumentation and electrical equipment within the near future.

When the above requirements have been satisfied, the seismic qualification testing program which will be implemented for seismic Category I instrumentation and electrical equipment will provide adequate assurance that such equipment may be expected to function properly during the excitation from vibratory forces imposed by the safe shutdown earthquake and under the conditions of post-accident operation. This program will constitute an acceptable basis for satisfying the applicable requirements of Criterion 2 of the General Design Criterion.

3.8 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.

1547 203

4.0 REACTOR

4.1 Summary

Criterion 10 of the General Design Criteria requires that the reactor core and associated systems shall be designed to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. We have reviewed the information provided in RESAR-41 in support of the proposed reactor design. Our evaluation is contained in the follcwing sections.

The RESAR-41 nuclear steam supply system will operate at a licensed thermal power rating of 3817 megawatts with sufficient margin to allow for transient operation and instrument error without causing damage to the core and without exceeding the pressure settings of the safety valves in the coolant system. The core thermal power level will be 3800 megawatts. The 17 megawatts difference is the net contribution of heat to the reactor coolant system from the reactor coolant pumps.

The core will be cooled and moderated by light water at a pressure of 2250 pounds per square inch, absolute, in the reactor coolant system. The reactor coolant will contain boron as a neutron poison. The concentration of the boron will be varied as required to control relatively slow reactivity changes including the effects of fuel burnup. Additional boron, in the form of burnable poison rods, will be employed in the first cycle to establish the desired initial reactivity.

Except for differences introduced by substitution of the 14 foot 17x17 fuel assemblies and the rapid refueling modifications to the control rod drives, the fuel assemblies and control rod assemblies and control rod drives of the proposed reactor design will be comparable to the assemblies and drives of a number of nuclear power plants now operable. Additional analyses and test programs are both in process and planned for the immediate future to demonstrate both the integrity of the modified designs and the validity of applying experience with the 15x15 assemblies to the evaluation of the conceptually similar 17x17 fuel assemblies.

4.2 Mechanical Design

4.2.1 Fuel

4.2.1.1 Description

The fuel assemblies will consist of 264 fueled rods, 24 guide thimbles, and one instrumentation thimble arranged in a 17x17 array. The instrumentation thimble will be at the center of the assemblies and will facilitate the insertion of neutron detectors, while the guide thimbles will provide channels for inserting various reactivity controls. The fuel rods will contain uranium dioxide ceramic pellets hermetically clad in Zircaloy-4 tubes supported at both ends by stainless steel nozzles. Alignment and transverse spacings will be maintained by nine spacer grids separated uniformly along the axis of the assembly.

All fuel rods will be internally pre-pressurized with helium during final welding to minimize cladding compressive stresses during service. The level of pre-pressurization is designed both to preclude any cladding tensile stresses due to a net internal pressure even during anticipated transients and to preclude clad flattening. The specific level of pre-pressurization will be dependent upon the planned fuel burnup and will be determined for the final design.

The fuel assembly design (17x17 array) is mechanically similar to the previously used Westinghouse fuel assembly (15x15 array). Those mechanical aspects which differ are indicated in Table 4-1 of this report. The differences are essentially geometric and will result in a lower linear power density and other increased safety margins for the 17x17 fuel assembly. The proposed assembly will be further distinguished from the 144 inch 17x17 assembly by three features (1) the fuel column height of 164 inches, (2) the use of two instead of no alignment pins, and (3) the use of nine instead of eight spacer grids.

The evaluation of the Westinghouse fuel mechanical design is based upon mechanical tests, in-reactor operating experience and engineering analyses. Additionally, the in-reactor performance of the fuel design will be subject to the continuing surveillance programs of Westinghouse and individual utilities. These programs provide confirmatory and current design performance information.

4.2.1.2 Thermal Performance

In our evaluation of the fuel thermal performance we assume that densification of the uranium fuel pellets may occur during irradiation in power reactors. 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 the stored energy, the linear thermal output, the probability for local power spikes, and the thermal resistance of the radial gap.

The primary effects of densification on the fuel rod mechanical design analysis are manifested in calculations of fuel-cladding gap conductance and time-to-collapse of the cladding. Time-to-collapse calculations predict the time required for unsupported cladding to become dimensionally unstable and to flatten into an axial gap caused by fuel pellet densification. Gap conductance calculations predict the increase in thermal resistance due to opening of the fuel-clad radial gap.

The engineering methods to be used by Westinghouse to analyze the densification effects on fuel thermal performance have been previously submitted to the Commission and reviewed. The methods addressed include testing, mechanical analyses, thermal and hydraulic analyses and accident analyses. The results of our review were reported in "Technical Report on Densification of Westinghouse PWR Fuel" issued by the Commission on May 14, 1974, and in our evaluation of Westinghouse Topical Report WCAP-8185, "Reference Core Report 17x17" in a letter to Westinghouse dated July 26, 1974. On the

TABLE 4-1

FUEL MECHANICAL DESIGN COMPARISON

Design Parameter	Westinghouse RESAR-41	Westinghouse Typical Operation Fuel
FUEL ASSEMBLY		
Rod Array	17x17	15x15
Number of Fueled Rods	264	204
Number of Spacer Grids	9*	7
Number of Guide Thimbles	24	20
Inter-rod Pitch	0.496 inches	0.563 inches
Alignment Pins	2*	0
Average Thermal Output (4 loop)	5.4 kilowatts per foot	7.0 kilowatts per foot
FUEL PELLETS		
Fuel Column Height	164 inches*	144 inches
Density (theoretical)	95 percent	94 percent
Fuel Weight/Unit Length (per rod not assembly)	0.364 pounds per foot	0.462 pounds per foot
FUEL CLADDING		
Outside Radius	0.187 inches	0.211 inches
Thickness	0.0225 inches	0.0243 inches
Radius/Thickness Ratio	8.31	8.68

*Different from 17x17, 144 inch long 8-grid fuel assemblies.

1547 206

basis of our review we conclude that the methods to be employed by Westinghouse will consider the effects of densification on the reactor fuel assemblies in a manner which adequately describes the fuel behavior.

4.2.1.3 Mechanical Performance

Although there exists no direct operating experience on 164 inch cores, substantially all of the in-reactor operating experience with Westinghouse fuel rods and assemblies is applicable to the RESAR-41 fuel design since the 17x17 fuel assembly is a slight mechanical extrapolation from the 15x15 fuel assembly. The current use of similar fuel rods and assemblies has yielded operating experience that provides confidence in the acceptable performance of the RESAR-41 fuel assembly design. The range in design parameters for which in-reactor experience is specifically applicable has been tabulated in Table 4-2. By the time a RESAR-41 nuclear steam supply system has been constructed there will be significant additions to this experience. The assemblies referred to in Table 4-2 have been irradiated for up to six years and have had peak exposures of 30 gigawatt days per metric tonne, totaling more than 70 million megawatt hours of power generation.

TABLE 4-2

RANGE OF DESIGN PARAMETER EXPERIENCE

Parameter	Range of Power Reactor Experience
Fuel Rod Array	14 x 14 and 15 x 15
Rods Assembly	179 to 204
Guide Thimbles/Assembly	16 to 20
Assembly Envelope	7.76 to 8.43 inches
Inter-rod Pitch	0.556 to 0.463 inches
Plenum Length	3.27 to 6.69 inches
Pre-pressurization	14.7 to > 400 pounds per square inch, absolute
Diametral Gap	0.0065 to 0.0075 inches
Spacer Grids/Assembly	7 to 9
Fuel Column Height	120 to 144 inches

During this power reactor service a small fraction of the fuel rods have experienced defects. However, there has been no instance where cladding defects have threatened either the plant or the public safety. Cladding defects were caused by excessive manufacturing impurities, excessive coolant cross-flow velocities and fuel pellet densification. Excessive manufacturing impurities have been eliminated by modifications to the manufacturing procedures and cross-flow velocities were reduced by modifications to baffle joints. Densification effects are discussed earlier in this section. The fuel related modifications required adjustments of design limits rather than a mechanical redesign of the fuel assembly.

1547 207

Confidence that the mechanical characteristics of the RESAR-41 fuel assemblies are predictable is enhanced by the results of out-reactor mechanical tests. Most of the current results are from tests on typical 15x15 fuel assemblies. Since the 17x17 fuel assembly is a slight mechanical extrapolation from the 15x15 fuel assembly, we expect the mechanical behavior of the two assemblies to be similar. Table 4-3 lists the tests and/or analyses that have been performed by Westinghouse and reported in the cited topical reports.

TABLE 4-3

GENERIC DESIGN EVALUATION TOPICALS

Tests and Analysis Topical Report Titles	Topical Report Number
"Hydraulic Flow Test of the 17x17 Fuel Assembly"	WCAP-8278
"An Evaluation of Fuel Rod Bowing"	WCAP-8346
"Effect of a Bowed Rod on DNB"	WCAP-8176
"17x17 Design Fuel Rod Behavior During Simulated Loss-of-Coolant Accident Conditions"	WCAP-8289

We have reviewed the topical reports listed in Table 4-3 above and have accepted them for reference in license applications using the 17x17 array, 144 inch, 7-grid fuel assemblies. Our acceptance letters to Westinghouse are dated July 21, 1975 for WCAP-8346, December 2, 1974 for WCAP-8176 and May 12, 1975 for WCAP-8289. We have found the test results described in WCAP-8278 to be acceptable for the 7-grid assemblies and are presently reviewing additional information related to the 8-grid design. These topical reports describe test results and analytical methods which are also pertinent to RESAR-41 fuel assemblies. However, the RESAR-41 fuel assemblies have three features which may slightly modify the applicability of the tests and analytics described in the approved topical reports. The features of interest are (1) the 164 inch length of the core, (2) the nine axial spacer grids, and (3) the 2-alignment pins on the upper end fitting.

The specific effects of these features on fuel rod bowing, hydraulic pressure drop, fretting and fuel column gap size are being reviewed by the staff on a generic basis. While the status of our present evaluation is adequate for issuance of a Preliminary Design Approval, we do plan to continue our generic review program in this area. We will confirm the acceptability of the fuel design at the Final Design Approval stage on the basis of the then current methods of analysis.

4.2.1.4 Fuel Surveillance

Performance of the fuel during operation will be indirectly monitored by measurement of the activites of both the primary and the secondary coolant for compliance with technical specification limits. Onsite surveillance normally includes examination of fuel rod integrity, fuel rod and fuel assembly dimensions and alignment, and surface deposits. RESAR-41 also includes, as an option, a failed fuel detection system which is discussed in Section 9.2.1 of this report.

For new fuel designs for which there is no operating experience we require that a supplemental fuel surveillance program be conducted. This program will be conducted on two of the first plants using the new design and must include the capability to perform destructive fuel rod tests.

We will require Westinghouse to commit to (1) develop a supplemental fuel surveillance program that includes the capability for destructive examination of fuel rods, and (2) arrange to have this supplemental surveillance program conducted by the utility applicants for two of the first plants using the 14 foot fuel. Commitments for the above requirements are sufficient for the Preliminary Design Approval. We will review the details of the supplemental surveillance program during the Final Design Approval review. We will report the status of this commitment in a supplement to this report.

4.2.1.5 Conclusion

Subject to a favorable resolution to the supplemental fuel surveillance program, we conclude that the cladding integrity of the RESAR-41 fuel will be maintained, that significant amounts of radioactivity will not be released, and that neither accidents nor earthquake induced loads will result in either an inability to cool the fuel or interference with control rod insertion. Our conclusion is based on (1) the operating experience with similar fuel, (2) the results of out-reactor tests on an assembly of similar design, (3) the increased thermal margins which the 17x17 fuel has, (4) the technical specification requirements that will be in effect to monitor and limit offgas and effluent activity, and (5) the existence of a continuing fuel rod surveillance program and non-destructive post irradiation examination requirements.

4.2.2 Reactor Pressure Vessel Internals

The reactor internals design has been modified from previous designs to accommodate the 14 foot long 17x17 fuel. In addition, the RESAR-41 design will allow for the removal of the upper core support assembly as a unit with the vessel closure head during refueling as part of the rapid refueling concept. The closure head will be connected to the upper core support structure by three lifting rods, which will be attached by threading to the upper support plate and penetrate the closure head in adapters similar to those provided for the control rod drive mechanism pressure housing.

This arrangement will permit the head and upper core support structure to be removed and inserted as a unit. When necessary, the upper core support structure can be disconnected from the head for separate removal. A discussion of the complete rapid refueling system is found in Section 5.4.8 of this report.

We have reviewed the information presented in RESAR-41 on:

(1) The physical and design arrangements of all reactor internals structures, components, assemblies, and systems, including the manner of positioning and securing such items within the reactor pressure vessel, the manner of providing for axial and lateral retention and support of the internals assemblies and components, and the manner of accommodating dimensional changes due to thermal and other effects.

(2) The design loading conditions that will provide the basis for the design of the reactor internals to sustain normal operation, anticipated operational occurrences, postulated accidents, and seismic events. All combinations of design loadings that will be accounted for in the design of the core support structure are listed (e.g., operating pressure differences and thermal effects, seismic loads, and transient pressure loads associated with postulated loss-of-coolant accidents).
(3) Each combination of design loadings categorized with respect to the "normal," "upset," "emergency," or "faulted" condition as defined in Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (hereafter, the Code) and the associated design stress intensity or deformation limits. The design loadings include the safe shutdown earthquake and operating basis earthquake loads.
(4) The design bases for the mechanical design of the reactor vessel internals including limits such as maximum allowable stresses, deflection, cycling, and fatigue limits, and core mechanical and thermal restraints for positioning and holddown purposes.

We determined that Westinghouse will perform a dynamic system analysis of the reactor internals and of the broken and unbroken piping loops. This analysis will be provided with the final design. The dynamic system analysis will be performed to provide an acceptable basis for confirming the structural design adequacy of the reactor internals and the unbroken piping loops to 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 described in RESAR-41 and find that they will provide adequate assurance that the combined stresses and strains in the components of the reactor coolant system and reactor internals will not exceed the allowable design limits for the materials of construction as specified in Appendix F to Section III of the Code. We also find that the resulting deflections or displacements of any structural elements of the reactor internals will not distort the reactor internals geometry 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 result in an acceptable structural design for the reactor internals, and constitutes an acceptable basis for satisfying the requirements of Criteria 2 and 4 of the General Design Criteria.

The design procedures and criteria that Westinghouse will use for the reactor internals are in conformance with established technical procedures, positions, standards, and criteria which are acceptable to the staff.

1547 210

The use of the specified design transients, design loadings, and combinations of loadings as applied to the design of the reactor internals structures and components will provide reasonable assurance that, in the event of an earthquake or of a system upset or faulted condition transient during normal plant operation, the resulting deflections and associated stresses imposed on the structures and components involved will not exceed allowable stresses and deformation limits for the materials of construction. Limiting the stresses and deformations under such loading combinations provides an acceptable basis for the design of these structures and components to withstand the most adverse loading events which have been postulated to occur during the service lifetime without loss of structural integrity or impairment of function. In addition, the design procedures and criteria to be used by Westinghouse in the design of the reactor internals constitutes an acceptable basis for satisfying the applicable requirements of Criteria 1, 2, 4 and 10 of the General Design Criteria.

4.2.3 Materials Considerations for Reactor Vessel Internals

The maintenance of the integrity of the reactor vessel internals in service is essential to assure that all reactor fuel assemblies remain in place. Proper placement of fuel assemblies is necessary to permit unimpaired operation of the control rod assemblies for safe reactor operation and shutdown. To evaluate the adequacy of the proposed design, we reviewed the materials selection, compatability, fabrication controls, and the extent of testing proposed by Westinghouse.

We have reviewed the material specifications to be used for major components of the reactor internals. The major material that will be used is Type 304 stainless steel. The bolts and dowel pins will be fabricated from Type 316 stainless steel, except for the radial support key bolts, which will be fabricated from Inconel-750. All the materials that will be used in the reactor vessel internals are in conformance with the requirements of Appendix I of Section III of the Code.

The materials for the reactor vessel internals that will be exposed to the coolant are compatible with the expected environment, as proven by extensive testing and satisfactory performance. The specified controls on the reactor coolant chemistry provide reasonable assurance that the reactor vessel internals will be adequately protected during operation from an environment which could lead to stress-corrosion of the materials and loss of component structural integrity.

We have reviewed the controls that will be imposed on the fabrication of the reactor vessel internals and conclude that they are in conformance with the recommendations of Regulatory Guide 1.31 and Regulatory Guide 1.44 (See Section 5.2.3 of this report for more information).

Laboratory stress-corrosion tests and service experience provide the basis for the staff criterion that cold-worked austenitic stainless steels used in the reactor coolant pressure boundary should have an upper limit on the yield strength of 90,000 pounds per square inch. The only stainless steel materials that will be used in the reactor core support structures which have yield strengths greater than 90,000 pounds per square inch are the 403 series used for holddown springs. These materials will be used in accordance with the 1971 Boiler and Pressure Vessel Code, Code Case number 1337. We conclude that these materials will be compatible with the reactor coolant and are acceptable for this use.

The materials selection, fabrication practices, and examination and protection procedures will be performed in accordance with the recommendations of the Code and the Nuclear Regulatory Commission. This provides reasonable assurance that the materials used for the reactor vessel internals will not be susceptible to stress-corrosion during service.

The use of materials proven to be satisfactory by actual service experience and conforming to Nuclear Regulatory Commission and Code recommendations constitutes an acceptable basis for compliance with the requirements of Criteria 1 and 14 of the General Design Criteria.

4.2.4 Reactivity Control System

4.2.4.1 Evaluation

Reactor power in the RESAR-41 nuclear steam supply system will be controlled by (1) permanent devices such as the full and partial length rod cluster control assemblies, (2) temporary devices such as the burnable poison assemblies used only in the initial core, and (3) a soluble chemical neutron absorber (boric acid). The reactor control system will direct the control rod drive mechanisms to insert, hold, withdraw or trip the rod cluster control assemblies. The chemical and volume control system will provide another means of reactivity control by varying the concentration of boric acid in the coolant to effect relatively slow reactivity changes.

The control rod system will consist of 61 clusters of full length rods and eight clusters of part length rods to shape the reactor power distribution and to compensate for changes in reactivity resulting from fuel burnup. Each cluster will have 24 absorber rods fastened at the top end to a common spider assembly. The absorber material that will be used in the control rods is a silver-indium-cadmium alloy which is "black" to thermal neutrons and in addition, has a resonance absorption capability which increases its worth. The alloy will be in the form of extruded rods sealed in stainless steel tubes.

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, that is, power and temperature variations. The control and shutdown groups will provide adequate shutdown margin (1.75 percent reactivity) in the event of a reactor trip. Shutdown margin is defined as the amount by which the core will be subcritical at hot shutdown if all rod cluster control assemblies are tripped, assuming that the highest worth assembly remains fully withdrawn and assuming no changes in xenon or boron concentration or part length rod cluster control assembly position.

1547 212

The manually controlled part length rods will be designed to control the axial neutron flux shape and axial xenon oscillations should they occur. Restriction on their use is discussed in Section 4.3.1 of this report.

The proposed control rod drive mechanisms will include a provision whereby the control rods can be mechanically locked in the fully withdrawn position. This will be included as part of the rapid refueling system which requires that all rod cluster control assemblies be completely withdrawn and stored in the reactor upper core support structure guide tubes during head removal for refueling. This is discussed in detail in Section 5.4.8.

The soluble boric acid neutron absorber will be varied to control long term reactivity changes. These long term reactivity changes result from (1) fuel depletion and fission product buildup, (2) cold to hot, zero power reactivity change, (3) reactivity changes produced by intermediate-term fission products, such as xenon and samarium, and (4) burnable poison depletion.

For the RESAR-41 accident analyses, a rod drop time of 2.4 seconds to 85 percent insertion has been used. This time is based on tests for the 12-foot 17x17 assembly but with an allowance for the additional two feet of travel at constant velocity in the 14-foot core. These tests were conducted at the Westinghouse Test Engineering Laboratory in the D-loop hydraulic test facility.

The prototype assembly tests listed in Table 1-2 of this report, which are to be completed in 1976, will provide data on the integrated fuel assembly guidance system and rod cluster control assembly performance. The data to be taken will include control rod drop time and stall velocity. The rod drop time used in the accident analyses submitted at the Final Design Approval stage will be required to be based on the outcome of these tests.

The objectives of our review were to determine that the design, fabrication, and construction of the control rod drive mechanisms will provide structural adequacy and that suitable life cycle testing programs have been utilized to prove operability under service conditions.

We evaluated the design criteria for both the internal pressure containing portions and other portions of the control rod drive mechanisms.

The design stress limits, including fatigue limits, and deformation limits as appropriate to the components of the control rod drive mechanism were compared with those of specified codes, previously designed and successfully operating systems, or with the results of scale model and prototype testing programs.

Loading combinations are defined as those loadings associated with plant operations which are expected to occur one or more times during the lifetime of the plant and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power, combined with loadings caused by natural or accident events. These load combinations were compared with those specified for each of the plant operating condition as defined in Paragraph NB-3113 of the Code.

The control rod drive mechanisms of this new design will be subjected to a life cycle test program to determine the ability of the drives to function over the full range of temperatures, pressures, loadings, and misalignments expected in service. The tests will include functional tests to determine times of rod insertion and withdrawal, latching operation, scram operation and time, ability to overcome a stuck rod condition, and wear. Rod travel and the number of trips expected during the mechanism's operational life will be duplicated in the tests.

The design criteria and the testing program to verify the mechanical operability and life cycle capabilities of the reactivity control system conform to established criteria, codes, standards, and specifications that we find acceptable. The use of these criteria and programs provide reasonable assurance that the system will function reliably when required, and form an acceptable basis for satisfying the mechanical reliability requirements of Criterion 27 of the General Design Criterion.

4.2.4.2 Materials Considerations

The integrity of the control rod system is essential to assure unimpaired operation of the control rod assemblies for safe reactor operation and shutdown, and to maintain the integrity of the reactor coolant pressure boundary. To evaluate the adequacy of the design proposed in RESAR-41, we reviewed the materials information relating to mechanical properties, the methods to control sensitization of stainless steel, compatability, testing, and cleaning and cleanliness control.

We reviewed the selection of the reactivity control system materials for compatibility in a pressurized water reactor environment, for adequate mechanical properties at room and operating temperature, for resistance to adverse property changes in a radioactive environment, and for compatibility with interfacing components. The major materials used will be austenitic and martensitic types of stainless steel, and Inconel-X and cobalt base alloys. The martensitic steel is specified in accordance with Code Case 1337-8 to be tempered at a minimum temperature of 1125 degrees Fahrenheit. The yield strength of the cold worked austenitic stainless steel is specified not to exceed 90,000 pounds per square inch in accordance with staff requirements.

The controls imposed upon the austenitic stainless steel of the system will conform to the recommendations of Regulatory Guide 1.31. Fabrication and heat treatment practices performed in accordance with these recommendations will provide added assurance that stress corrosion cracking will not occur during the design life of the components. The compatibility of all materials used in the reactivity control system in contact with the reactor coolant satisfies the criteria of Articles NB-2160 and NB-3120 of Section III of the Code. Both martensitic and precipitation hardening stainless steels will be given tempering or aging treatments in accordance with staff positions. Hard chromium plating will provide wear surfaces on the parts and prevent galling between mating components.

Conformance with the Code 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 Criterion 26 of the General Design Criteria.

Based on this review, we conclude that the design, fabrication and testing of the control rods and control rod drives will be in accordance with Code and Nuclear Regulatory Commission requirements.

4.3 Nuclear Design

The core design proposed in RESAR-41 differs from previous Westinghouse four loop plant designs previously approved for construction and operation in that the active fuel length will nominally be 14 feet. Like the four-loop 12-foot cores, this core will have 193 fuel assemblies, and essentially the same X-Y plane nuclear characteristics. This Section will, therefore, focus on the changes in nuclear characteristics resulting from increasing the height of the core. We find that these characteristics will not have altered significantly from the designs already approved, and conclude that the RESAR-41 nuclear design is acceptable.

The proposed 13.9 percent fuel length increase of 20 inches in the active fuel length over prior Westinghouse designs is similar to the change made by Combustion Engineering, Incorporated from the Calvert Cliffs class of reactors to the Waterford class, Docket No. 382 (includes San Onofre 2 & 3, Docket Nos. 361 and 362, and Forked River, Docket No. 363) where the active fuel length was increased from 137 inches to 150 inches. In the case of Combustion Engineering, Incorporated case, the entire 9.5 percent increase in length was used to increase power. The 3800 megawatt core rating for the proposed RESAR-41 design represents about an 11.5 percent power increase from the RESAR-3 design.

There has also been a slight increase in flow for the RESAR-41 nuclear steam supply system. At the 3800 thermal megawatts rating, the reactor described in RESAR-41 will actually be operated at a lower average linear power density than a 12-foot reactor, 5.33 kilowatts per foot as compared to 5.44 kilowatts per foot.

4.3.1 Power Distributions

The expected power distributions, both typical and limiting, have been acceptably presented. These distributions include the radial, assembly, local rod, and axial power distributions. The associated peaking factors are also discussed. The core-average linear power density at full power will be 5.33 kilowatts per foot, compared to 5.44 kilowatts per foot in a 12-foot reactor. A design limit peaking factor of 2.50 has been established for the RESAR-41 core, whereas it is 2.32 for the 12-foot cores. This increase is a result of a difference in the effect of part-length control rods in the 14-foot core. At 102 percent of full power, the peak linear power density will be 13.6 kilowatts per foot for the 3800 megawatt 14-foot core and 12.9 kilowatts per foot for a 3411 megawatt 12-foot core. Westinghouse has calculated the power distributions expected during both steady-state and typical load-follow operations to show that the actual peaking factor can be maintained below the design value. An allowance for calculational error of 5 percent in the expected peaking factors was determined by Westinghouse from comparisons between measured and calculated distributions. The comparison between expected and design peaking factors demonstrates that the plant can be operated below the design values. Thus, the design peaking factors are appropriate for use in the accident analyses.

We have reviewed the nuclear instrumentation which will be provided to measure the core power distribution. The reactor will be provided with two types of monitoring systems, a system of movable incore fission chamber detectors and a system of fixed ion chambers located symmetrically around the core outside the reactor pressure vessel. The movable incore detectors will be capable of measuring the fuel rod peaking factor to within 5 percent and will be used to make periodic incore maps of the power distribution. The ion chambers located outside the reactor pressure vessel will provide total power as measured by neutron flux, relative power in each quadrant of the core, and the relative power in the top and bottom of the core. The axial power offset, as measured from the relative power in the top and bottom of the core, and the radial tilt will be used to maintain the core peaking factor below 2.50.

The power distribution monitoring procedure proposed is constant axial offset control using excore detectors. This is identical to the procedure currently in use at operating Westinghouse reactors and which has been reviewed and approved in several recent operating license cases. The intent of constant axial offset control is to maintain the axial power distribution and therefore, the axial xenon distribution, constant as a function of power level. This will limit the magnitude of axial xenon transient effects on the peaking factor. This will be achieved by restricting operation to a $\pm$ 5 percent band about a value of flux difference (upper minus lower excore detector readings) associated with full power operation of an essentially unrodded core. Above 90 percent of full power, the flux difference must be maintained within the operating band. Between 50 percent and 90 percent of full power, the flux difference may be out of this band no longer than one hour in any 24 hour period. Control of the flux difference within the target band will be accomplished using one of two methods, (1) the use of full-length control rods and boron control (called Mode A by Westinghouse) and (2) the use of part-length control rods (called Mode B by Westinghouse).

Westinghouse has identified potential departure from nucleate boiling problems associated with the use of part-length control rods. As a result, we have required that part-length controls not be used in currently operating Westinghouse plants. This subject is under generic review by Westinghouse and the staff. Until this item is resolved to the satisfaction of the staff, the use of part-length control rods in RESAR-41 plants will be prohibited.

1547 216

4.3.2 Reactivity Coefficients

The reactivity coefficients reflect the changes in the neutron multiplication due to varying core conditions such as power, temperature, pressure, and void changes. These coefficients vary with fuel burnup. Westinghouse has presented the values for these coefficients that have been employed in the analyses of both normal and transient reactor operation. For the current state of design of the core as described in RESAR-41, the reactivity coefficients fall in the same range as those covered by analyses made for the 12-foot cores. This type of bounding approach is normal and acceptable for a construction permit or Preliminary Design Approval analysis. When a detailed final design is completed, more precise predictions of the coefficients will be available. We will review these coefficients again at the final design review stage. These coefficients are also used in the accident analyses presented in Chapter 15 of RESAR-41.

The predicted total power coefficient is strongly negative for all reactor conditions throughout core life satisfying the requirements of Criterion 11 of the General Design Criteria. Westinghouse has agreed to measure the moderator temperature coefficient and the power coefficient during startup tests of plants referencing RESAR-41 to check the calculated values and to ensure that conservative coefficient values were used in the accident analyses.

4.3.3 Control

To allow for changes of reactivity due to reactor heatup, changes in operating conditions, fuel burnup and fission product buildup, a significant amount of excess reactivity will be built into the core. Westinghouse has provided sufficient information relating to core reactivity balance for the first core, has shown typical values for a reload core, and has shown that means are incorporated into the design to control excess reactivity at all times. Control will be achieved with movable control rods and through the variation of boron concentration in the reactor coolant. Sufficient additional control rod worth will be provided to accommodate the reactivity effects of the most limiting accident (steam line break) at any time during the core life with an allowance for the most reactive control rod assembly stuck in the fully withdrawn position. In addition, the chemical and volume control system will be capable of shutting down the reactor by adding soluble boron poison and will be able to maintain it in the cold shutdown condition at any time during the core life. This combination of control systems satisifes the requirements of Criterion 26 of the General Design Criteria.

The RESAR-41 design includes 61 full-length control rod assemblies. The 12-foot 193 fuel assembly reactor design includes 53 of these assemblies. The 14-foot RESAR-41 core is predicted to require 0.35 percent more shutdown reactivity for redistribution, 0.07 percent more for full power temperature defect and 0.08 percent more for the stuck rod worth than the 12-foot core. The remainder of the 1.2 percent increase in rod worth provided by eight extra control assemblies represents additional shutdown margin.

Core reactivity will be controlled by means of boron chemical poison dissolved in the coolant, the control rod assemblies, and burnable poison rods. The plant will be operated at steady-state full power with most of the full-length control rods withdrawn.

1547 217

Limited insertion of the full-length control rods will permit compensating for fast reactivity changes (e.g., the effects of minor variations in moderator temperature and boron concentrations) without impairing shutdown capability. (Part-length control rods will be provided to control the axial power distributions. Restrictions upon their use were previously discussed in the power distribution section.) Soluble boron poison will be used to compensate for slow reactivity changes including those associated with fuel burnup, changes in xenon and samarium concentration, buildup of long-life fission products, burnable poison rod depletion, and the large moderator temperature change from cold shutdown to hot standby. The soluble boron poison system will provide the capability to take the reactor at least 10 percent subcritical in the cold shutdown condition.

The proposed full-length control rod assemblies are divided into two groups, control and shutdown. Load changes will be made with the control rods and/or the soluble poison system. Rod insertion will be controlled by the power-dependent insertion limits that will be given in the technical specifications. These limits will (1) ensure that there is sufficient negative reactivity available to permit the rapid shutdown of the reactor with ample margin, (2) ensure that the worth of a control rod that might be ejected in the unlikely event of an ejected rod accident will be no worse than that assumed in the accident analyses, and (3) along with the power distribution control procedure, ensure that the axial peaking factor does not exceed the limiting value used for the accident analyses. The shutdown rods, which will never be inserted during operation, will be required to ensure a rapid reactor shutdown.

We have reviewed the calculated rod worths and the uncertainties in these worths, and conclude that rapid shutdown capability will exist at all times in core life assuming the most reactive control rod assembly is stuck in the fully withdrawn position. The estimate of uncertainties is based upon appropriate comparison of calculations with experiments. On the basis of our review, we have concluded that Westinghouse's assessment of reactivity control is suitably conservative, and that adequate negative reactivity worth has been provided by the control system to assure shutdown capability.

4.3.4 Stability

The stability of the reactor to xenon-induced power distribution oscillations and the control of such transients have been discussed by Westinghouse in RESAR-41. Due to the negative power coefficient, the reactor will be inherently stable to oscillations in reactor power. Also, the control system, described in Section 7.7 of this report, will provide adequate protection against total power instabilities. The core is calculated to be stable against X-Y xenon oscillations throughout core life, although, the 14-foot core will be less stable to axial xenon oscillations than the 12-foot core. Design predictions indicate beginning of life (150 megawatt days of burnup per metric tonne of uranium) stability indices of -0.020 hours^{-1} and -0.47 hours^{-1} for the 14- and 12-foot cores, respectively. The index is predicted to become zero at between 8000 to 9000 megawatt days of burnup per metric tonne of uranium for the 14-foot core and at between 11,000 to 12,000 megawatt days of burnup per metric tonne of uranium range for the 12-foot core. We do not anticipate that the slight decrease in stability predicted for the 14-foot core will have a noticeable effect on operation. Some form of additional surveillance will be required for the first core of this design, to verify the predicted stability index. This surveillance will be based on our review of the final design of the reactor. Westinghouse has provided sufficient information to show that axial oscillations will be detected and controlled before any safety limits are reached, thus preventing any fuel damage.

4.3.5 Analytical Methods

Westinghouse has described the computer programs and calculational techniques used to calculate the nuclear characteristics of the reactor design and has provided examples to demonstrate the ability of these methods to predict experimental results. We conclude that the information presented adequately demonstrates the ability of these analytical methods to calculate the reactor physics characteristics of the reactor described in RESAR-41.

4.4 Thermal and Hydraulic Design

4.4.1 Evaluation

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. Westinghouse 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 anticipated operational occurrence. This is referred to as the 95/95 criterion. The preliminary RESAR-41 core design uses a minimum allowable limit of 1.30 for the departure from nucleate boiling ratio. The 1.30 is based on the one-side confidence limits for the original data base for the W-3 departure from nucleate boiling correlation; the final departure from nucleate boiling ratio limit must be justified by appropriate statistical analysis of applicable data.
(2) Operating conditions are selected to assure hydraulic stability within the core, thereby preventing a premature departure from nucleate boiling.
(3) The peak centerline temperature of the fuel will be less than the melting point (5080 degrees Fahrenheit for unirradiated fuel) during normal operation and any anticipated operational occurrence.

The thermal and hydraulic design parameters for the reactor are listed in Table 4-4 in this report. A comparison of these parameters with those of the RESAR-3 reactor is given in the table.

The main difference between this 17x17 application and previous 17x17 submittals is an increase of 20 inches in the nominal active fuel length from 144 inches to 164 inches. The actual fuel rod will be 173.3 inches long. This increase in the active fuel length allows the average maximum linear heat generation rate to remain approximately the same for the 164-inch core with a 3800 thermal megawatt power rating as that of a twelve foot 3411 thermal megawatt core.

TABLE 4-4

COMPARISON OF THERMAL HYDRAULIC DESIGN PARAMETERS

FOR RESAR-41 AND RESAR-3 CORES[a]

	RESAR-41[c]	Consolidated Version[b] RESAR-3
Reactor Core Heat Output (MWt)	3800	3411
System Pressure, Nominal (psia)	2250	2250
Minimum DNBR for Design Transients	> 1.30	> 1.30
Total Thermal Flow Rate (lb/hr)	144.7×10^6	142.2×10^6
Effective Flow Rate for Heat Transfer (lb/hr)	138.2×10^6	135.2×10^6
Average Velocity Along Fuel Rods (ft/sec)	17.2	16.8
Average Mass Velocity (lb/hr-ft^2)	2.71×10^6	2.66×10^6
Coolant Temperature (°F)		
Design Nominal Inlet	559.8	557.3
Average Rise in Core	66.8	62.3
Active Heat Transfer Surface Area (ft^2)	68,000	59,900 (59,700)[c]
Average Heat Flux (Btu/hr-ft^2)	185,200	189,400 (189,800)[c]
Maximum Heat Flux (Btu/hr-ft^2)	463,100	454,600 (474,500)[c]
Maximum Thermal Output for Normal Operation (kW/ft)	13.3	13.0 (13.6)[c]
Fuel Central Temperature at Beginning of Life, Maximum at 100% Power (°F)	3460	3250 (3500)[c]

[a]Abbreviations:

MWt	- megawatts thermal
psia	- pounds per square inch, absolute
lb/hr	- pounds per hour
ft/sec	- feet per second
lb/hr-ft^2	- pounds per hour per square foot
°F	- degrees Fahrenheit
ft^2	- square feet
Btu/hr-ft^2	- British thermal units per hour per square foot
kW/ft	- kilowatts per foot

1547 220

[b]Without densification effects except as noted.

[c]With densification effects.

The peak to average value of the cosine axial heat flux distribution used for design departure from nucleate boiling calculations will be kept at 1.55. In addition, the thermal flow rate shows an increase over that for the 12-foot core. This results from increased pump size and increased flow measurement accuracy due to the leading edge flowmeter to be provided in each loop.

As noted in Table 1-2 of this report, 1/7 scale model flow tests are planned to verify the flow model by experimentally providing information concerning flow distributions through the lower core support plate to the fuel assemblies.

The margin to departure from nucleate boiling at any point in the core is expressed in terms of the departure from nucleate boiling ratio, which 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 to be used for the design of this core is the W-3 correlation with the "R" grid spacer factor which is described in the Westinghouse Topical Report WCAP-8296, "Effect of 17x17 Fuel Geometry on DNB." We have reviewed this method and accepted it for use in applications using 17x17 fuel by letter to Westinghouse dated December 31, 1974.

To maintain nucleate boiling as the mode of heat transfer, approximately a 10 percent margin in the departure from nucleate boiling ratio has been retained for three reasons:

(1) To incorporate the final results of the departure from nucleate boiling and mixing tests described in Section 1.4,
(2) To incorporate the final results of the experimental D-loop hydraulic tests of the 17x17 fuel assembly for the 164-inch core to confirm the pressure drop characteristics used in establishing the primary loop flow rate, and
(3) To allow for any fabrication tolerances that could occur in the manufacture of the fuel assembly.

The first part of the departure from nucleate boiling tests, using uniformly heated rods 8 and 14 feet in length (therefore applicable to the 164-inch RESAR-41 design), was completed and reported in WCAP-8296, "Effect of 17x17 Fuel Assembly Geometry on DNB." The results indicate that (1) the previously used departure from nucleate boiling correlation (W-3 with modified spacer factor) must be multiplied by 0.88 in order to show agreement with the 17x17 data, (2) the use of a thermal diffusion coefficient of 0.38 is conservative, and (3) a departure from nucleate boiling ratio of 1.275 corresponds to the 95/95 criterion. Since only data with uniformly heated rods were considered, it is uncertain at the present time whether further adjustments in the correlation or in the departure from nucleate boiling ratio corresponding to the 95/95 criterion are needed to cover the expected range in axial power shapes. Departure from nucleate boiling ratio tests with non-uniform axial heat flux have been completed and will be reviewed by the staff when the data are submitted. A 0.88 factor has been measured in uniform heat flux tests for the departure from nucleate boiling correlation. This is more restrictive than the 0.9 factor presently used by Westinghouse design calculations. Based on our review of these tests we may require reevaluation of the thermo-hydraulic analysis and adjustments in the departure from nucleate boiling ratio limits. We will require that RESAR-41 be updated to include the results of the test program upon completion of our review of these results. Since major adjustments are not expected we have concluded that this can be accomplished as a post-Preliminary Design Approval task. We will require that the final design safety analysis presented in RESAR-41 be based on a departure from nucleate boiling correlation consistent with all departure from nucleate boiling test results including the non-uniform axial heat flux tests. Should the final results be less favorable than the correlation used for the preliminary safety analysis, appropriate operating restrictions can be placed on RESAR-41 plants during the final design review stage.

Another parameter that influences the thermal-hydraulic design of the core is rod-to-rod bowing within fuel assemblies. Experimental data on the extent of bowing in the 17x17 fuel is not available but will be reported as it becomes available, in a future Westinghouse topical report. In the meantime, the design of the core is based on bowing as predicted by the method described in WCAP-8346, "An Evaluation of Fuel Rod Bowing." We had accepted this method to show that the design penalty used by Westinghouse to account for rod bowing is valid for one cycle of operation. Based on additional analyses submitted in May of 1975, we conclude that the design penalty will be valid for three full cycles of operation. As additional operating data on rod bowing is obtained, we will require adjustments in the design penalty if this is necessary.

In steady-state, two-phase, heated flow, a potential for flow instability in parallel channel exists. Westinghouse uses the HYDNA code, which has not been submitted for our review, for predicting the hydrodynamic stability of parallel closed channels. Results of HYDNA calculations predict inception of thermo-hydrodynamic instability at a power level in excess of 185 percent of rated power. Westinghouse also conducted an experimental program which demonstrated that parallel open channels are more stable than parallel closed channels. Westinghouse, therefore, concludes that the HYDNA analysis can be conservatively applied to the Westinghouse core which consists of parallel open channels. We conclude that the margin for hydrodynamic stability predicted by the HYDNA code is acceptable for the preliminary design. We will review the methods used in the HYDNA code prior to approval of the final design.

Preservation of nucleate boiling as the mode of heat transfer between the hot spot of the fuel cladding and the coolant not only assures that the cladding temperatures are only slightly greater than that of the coolant, but that the fuel centerline temperature will not reach the melting temperature. Using its thermal performance model,[1/]

[1/] Supplemental information on fuel design transmitted from R. Salvatori, Westinghouse NES, to D. Knuth, AEC, as attachments to letters NS-SL-518 (12/22/72), NS-SL-521 (12/29/72), NS-SL-524 (12/29/72) and NS-SL-543 (1/12/73) (Westinghouse Proprietary), and supplemental information on fuel design transmitted from R. Salvatori, Westinghouse NES, to D. Knuth, AEC, as attachments to letters NS-SL-527 (1/2/73) and NS-SL-544 (1/12/73).

1547 222

Westinghouse has calculated that at the beginning of core life at 100 percent power, with a linear heat generation rate of 13.3 kilowatts per foot, the fuel centerline temperature will be 3460 degrees Fahrenheit. The peak power density that would occur for a reactor trip at the 118 percent maximum overpower trip is less than 18.0 kilowatts per foot. At a linear heat generation rate of 18.0 kilowatts per foot, Westinghouse calculated a centerline temperature of 4200 degrees Fahrenheit, thus indicating no fuel melting. We have reviewed and approved the Westinghouse methods of calculating fuel temperature as reported in "Additional Testimony on Point Beach-2 Nuclear Plant in regard to Fuel Densification and its Effects," issued by the Atomic Energy Commission on February 2, 1973, and "Technical Report on Densification of Westinghouse PWR Fuel," issued by the same Commission on May 14, 1974. We conclude that the Westinghouse calculations adequately show that there will be no fuel melting.

For reactor described in RESAR-41 and recently reviewed Westinghouse designed reactors, the THINC computer code has been used to calculate core thermal-hydraulic performance characteristics. The code considers cross-flow between adjacent assemblies in the core and thermal diffusion between adjacent subchannels in the assemblies. The effect of local power distributions is considered. As a result of these considerations, the THINC code permits the computation of more realistic power shapes than those that had been available from previously used computer codes. These power shapes are especially important at the design overpower conditions.

The Westinghouse topical reports on the THINC program, WCAP-7956, "THINC-IV - An Improved Program for Thermal and Hydraulic Analysis of Rod Bundle Cores" and WCAP-8054, "Application of the THINC-IV Program to PWR Design," are still under review by the staff.

We will require, as we have in other instances wherein the design is based on this type of analysis, that the acceptability of the thermal, hydraulic, and nuclear feedback calculations be verified by confirmatory tests and analyses. We will review the results of the tests and analyses as they become available prior to the Final Design Approval stage of review. Should the test results as obtained fail to cover the anticipated range of conditions predicted for RESAR-41 core, Westinghouse will be required to perform the necessary additional tests prior to completion of our review of the final design.

4.4.2 Conclusions

On the basis of our review of the thermal-hydraulic design of the proposed RESAR-41 core including the design criteria and the steady state analysis of the core thermo-hydraulic performance, we have identified several tests that must be conducted and analyses and codes that must be reviewed for the final design. These result in requirements for:

1) Submittal and review of the complete non-uniform axial heat flux departure from nucleate boiling tests.

2) Perform statistical analysis of data to show compliance with the 95/95 criterion as stated above,
3) Confirmatory flow model tests, the 1/7 scale test and lower internals tests listed in Table 1-2 for the inlet geometry described in RESAR-41, and
4) Submittal and review of the HYDNA computer code.

On the basis of our review of the analytical techniques applied to the previously reviewed and approved 15x15 core designs, we have concluded that for the 17x17 core design, there is reasonable assurance that (1) the proposed thermal-hydraulic design will account for departure from nucleate boiling and fuel centerline temperature limitations in a satisfactory manner, and (2) the conservatism in the thermal-hydraulic design procedures can be verified. Therefore, we have concluded that with the stipulations mentioned above the preliminary thermal-hydraulic design of the RESAR-41 reactor is acceptable for the preliminary design review.

In the event that sufficient verification cannot be obtained from the test programs or that the analytical methods are determined not to be conservative, appropriate restrictions on operations can be established at the operating license stage for plants employing the RESAR-41 nuclear steam supply system.

1547 224

5.0 REACTOR COOLANT SYSTEM

5.1 Summary

Section 50.2 (v) of 10 CFR Part 50 defines the reactor coolant pressure boundary to include all those pressure-containing components of pressurized water-cooled nuclear power reactors, such as pressure vessels, piping, pumps and valves, which are:

(1) Part of the reactor coolant system, or

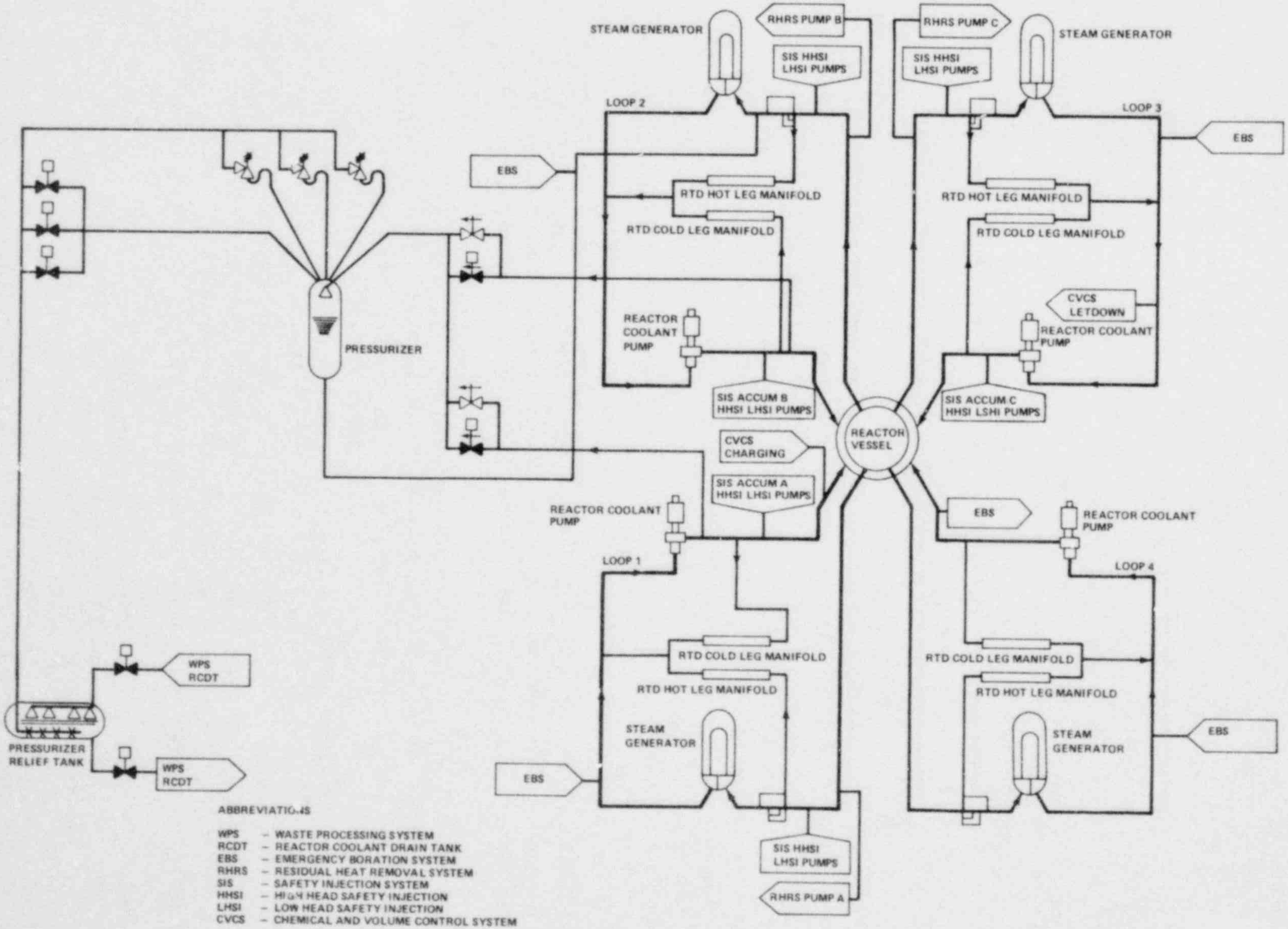
(2) Connected to the reactor coolant system, up to and including:

- (a) the outermost containment isolation valve in system piping which penetrates primary reactor containment,
- (b) the second of two valves normally closed during normal reactor operation in system piping which does not penetrate the primary reactor containment, and
- (c) the reactor coolant system safety and relief valves.

The reactor coolant system contains the reactor vessel, including the control rod drive mechanism housings, the reactor coolant side of the steam generators, the reactor coolant pumps, a pressurizer, and the interconnecting piping and valves associated with these components.

The residual heat removal system, the safety injection system, the chemical and volume control system, and the sampling system are the principal systems connected to the reactor coolant system. The proposed RESAR-41 nuclear steam supply system design incorporates a pressurized water reactor in a four-loop reactor coolant system. The 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. Each coolant loop will consist of a 29-inch inside diameter hot leg pipe between the reactor vessel outlet and the steam generator inlet and a 31-inch inside diameter cold leg pipe from the steam generator outlet to the reactor coolant pump inlet and a 27.5-inch inside diameter pipe connecting the pump discharge to the reactor vessel inlet. The pressurizer will be connected to one of the hot legs by a 14-inch schedule 160 surge line while spray lines will be connected to two cold legs. Each cold leg will contain a reactor coolant pump. The reactor coolant system will include a pressurizer relief tank, together with the interconnecting piping and instrumentation necessary for operational control, to receive, condense and cool steam discharged from the pressurizer safety valves. The entire system described above will be located within the containment building. Figure 5-1 in this report is a simplified diagram of the reactor coolant system.

During operation, the reactor coolant system will transfer the heat generated in the core to the steam generators where steam will be produced to drive the turbine generator. Borated demineralized water will be circulated in the system at a flow rate, pressure and temperature consistent with achieving the design reactor core thermal-hydraulic performance. The water will also act as a radiation shield, and neutron moderator and reflector.



ABBREVIATIONS

WPS – WASTE PROCESSING SYSTEM
RCDT – REACTOR COOLANT DRAIN TANK
EBS – EMERGENCY BORATION SYSTEM
RHRS – RESIDUAL HEAT REMOVAL SYSTEM
SIS – SAFETY INJECTION SYSTEM
HHSI – HIGH HEAD SAFETY INJECTION
LHSI – LOW HEAD SAFETY INJECTION
CVCS – CHEMICAL AND VOLUME CONTROL SYSTEM

Figure 5-1 REACTOR COOLANT SYSTEM

1547 226

The system is similar in design to that of the RESAR-3 design, with the following significant changes.

For the RESAR-41 design, the reactor vessel head closure system has been changed to facilitate rapid refueling. Also pertinent to the reactor vessel, the lower internals package has been modified due to the longer fuel length. The reactor coolant pump used in the RESAR-41 design is conceptually similar to that in the RESAR-3 systems but will have a greater capacity. To remove the additional heat in the RESAR-41 system, the steam generators will have more tubes that are longer than in the RESAR-3 system. A major difference between the RESAR-41 and RESAR-3 systems occurs in the residual heat removal system. The proposed RESAR-41 design uses three cooling trains instead of two and the residual heat removal system will be located completely within the containment and the pump functions will not be shared with the emergency core cooling system.

5.2 Integrity of the Reactor Coolant Pressure Boundary

5.2.1 Design of Reactor Coolant Pressure Boundary Components

Criterion 4 of the General Design Criteria requires that structures, systems, and components important to safety be designed to accommodate the effects of normal operation, maintenance, testing, and postulated accidents. We reviewed the design of the reactor coolant pressure boundary components to determine that component quality will be commensurate with the importance of the safety function of the reactor coolant pressure boundary.

We determined that the design loading combinations specified under Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (hereafter the Code) for Class 1 components have been appropriately categorized with respect to the plant condition identified as "normal," "upset," "emergency" or "faulted". The design limits proposed by Westinghouse for these plant conditions are consistent with the criteria recommended in Regulatory Guide 1.48. Use of these criteria will provide reasonable assurance that, in the event an earthquake should occur at the site, or other system 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 Criteria 1, 2 and 4 of the General Design Criteria.

Pressure-retaining components of the reactor coolant pressure boundary are designated as Class I components under Section III of the Code. Section 50.55a of 10 CFR Part 50 requires that these components meet the requirements for Class I components under the Code.

1547 227

We have reviewed the information provided in RESAR-41 and conclude that the components of the reactor coolant pressure boundary, as defined by the rules of Section 50.55a of 10 CFR Part 50, have been properly identified and classified as Code, Class I components. Westinghouse states that these components will be constructed in accordance with the requirements of the applicable codes and addenda as specified by the rules of Section 50.55a of 10 CFR Part 50.

We conclude that construction of the components of the reactor coolant pressure boundary in conformance with these codes 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.

5.2.2 Overpressurization Protection

Protection of the primary system against overpressurization will be provided by three power operated pressure relief valves and three safety valves. The three safety valves, in conjunction with the steam generator safety valves which are not within the scope of RESAR-41, will protect the reactor coolant system against overpressure in the event of a complete loss of heat sink, assuming that the reactor does not trip. The relief valves will be designed to limit the pressurizer pressure to a value below the high pressure trip set point for all design transients up to and including the design percentage step load decrease with steam dump, but without reactor trip.

The required capacity of the pressurizer safety valves was determined from consideration of a complete loss-of-steam flow to the turbine with credit taken for steam generator safety valve operation (assumed to have a capacity of 105 percent of rated steam flow) and maintenance of the main feedwater flow, but with no credit for reactor trip. The peak reactor coolant system pressure will be limited to 110 percent of the design value of 2500 pounds per square inch, absolute. No credit is taken for operation of the pressurizer relief valves, steam line relief valves, steam dump system, reactor control system, pressurizer level control system, or pressurizer spray.

A loss of load transient has also been analyzed for the case where the main feedwater flow is lost at the same time that steam flow to the turbine is lost. For this transient, the system will be protected against overpressurization by the pressurizer and steam generator safety valves in conjunction with the reactor protection system. The maximum pressure reached will be 2550 pounds per square inch, absolute.

The methods used by Westinghouse to analyze the overpressure protection of the reactor coolant system are presented in the topical report WCAP-7769, "Overpressure Protection for Westinghouse Pressurized Water Reactors" Revision 1, along with specific revisions pertinent to RESAR-41. We conclude that the margin for overpressure protection predicted in WCAP-7769, Revision 1, is acceptable for the preliminary design. We will review the methods used in WCAP-7769, Revision 1, prior to Final Design Approval.

1547 228

Requirements imposed on Anticipated Transients Without Scram (ATWS) (WASH-1270) could impose further requirements on the sizing of safety valves. See Section 15.5.7 for further discussions of this matter.

5.2.3 Reactor Coolant Pressure Boundary Materials

Criteria 1 and 14 of the General Design Criteria require that the reactor coolant pressure boundary be designed, fabricated, erected, and tested so as to have an extremely low probability of a rapidly propagating failure and of a gross rupture. In addition, they require that the reactor coolant pressure boundary be tested to quality standards commensurate with the importance of the safety function to be performed.

Our review included the compatibility of the materials of construction employed in the reactor coolant pressure boundary with the reactor coolant, contaminants, and radiolytic products to which the system will be exposed. The extent of the corrosion of ferritic low alloy steels and carbon steels in contact with the reactor coolant was reviewed. In addition, a review was made of the controls that will be used to prevent cracking of austenitic stainless steels and the fracture toughness and welding requirements for ferrite materials.

5.2.3.1 Material Specifications and Compatibility with Reactor Coolant

The materials proposed for use in the components of the reactor coolant pressure boundary have been identified by specification, and will be procured in accordance with the requirements of Section III of the Code, including Addenda and Code Cases appropriate to comply with Appendix B of 10 CFR Part 50. The residual elements in the ferritic material of the reactor vessel beltline will be controlled in order to reduce the sensitivity of the material to irradiation embrittlement.

Westinghouse has committed to not using any of the high strength materials represented by the unendorsed Code Cases 1358, 1412, and 1414 without prior Commission review and approval.

Austenitic stainless steels in a variety of product forms will be used for construction of pressure-retaining components in the reactor coolant pressure boundary. Unstabilized austenitic type stainless steels, which include American Iron and Steel Institute Types 304 and 316, will normally be used. Because these compositions are susceptible to stress-corrosion cracking when exposed to certain environmental conditions, process controls must be exercised during all stages of component manufacturing and reactor construction to avoid severe sensitization of the material and to minimize exposure of the stainless steel to contaminants that could lead to stress-corrosion cracking.

The materials of construction of the reactor coolant pressure boundary that will be exposed to the reactor coolant have been identified and all of the materials are compatible with the expected environment, as proven by extensive testing and satisfactory performance. General corrosion of all materials, except unclad carbon and low alloy steel, will be negligible. For these materials, conservative corrosion allowances have been provided for all exposed surfaces in accordance with the requirements of Section III of the Code.

The proposed maximum contaminant levels of the reactor coolant, as well as the proposed hydrogen ion concentration, hydrogen overpressure, and boric acid concentrations, have been shown by test and service experience to be adequate to protect against corrosion and stress corrosion problems.

The controls imposed on reactor coolant chemistry are in conformance with the recommendations of Regulatory Guide 1.44 and provide reasonable assurance that the reactor coolant pressure boundary components will be adequately protected during operation from conditions that could critically lead to stress corrosion of the materials and loss of structural integrity of a component.

The instrumentation provided for the control of reactor coolant water chemistry will provide adequate monitoring capability to detect changes on a timely basis to effect corrective actions before stress-corrosion attacks occur at an unacceptable level. The use of materials of proven performance and conformance with the recommendations of the Regulatory Guide constitutes an acceptable basis for satisfying the requirements of Criteria 14 and 31 of the General Design Criteria.

5.2.3.2 Fabrication and Processing of Ferritic Materials

The pressure-retaining components of the reactor coolant pressure boundary that are made of ferritic materials must meet requirements for fracture toughness during system hydrostatic tests and any condition of normal operation, including anticipated operational occurrences.

All materials must meet the acceptance standards of Article NB-2330 of Section III of the Code and the requirements of Appendix G of 10 CFR Part 50.

We have reviewed the materials selection, toughness requirements, and extent of materials testing proposed by Westinghouse and find them acceptable. These requirements provide assurance that the ferritic materials used for pressure retaining components of the reactor coolant pressure boundary will have adequate toughness under test, normal, and transient operation. All ferritic materials will meet the toughness requirements of Section III of the Code (1974 Edition). In addition, materials for the reactor vessel will meet the acceptance criteria of Appendix G of 10 CFR Part 50.

The fracture toughness tests and procedures required by Section III of the Code, as augmented by Appendix G, 10 CFR Part 50 for the reactor vessel, provide reasonable assurance that adequate safety margins against the possibility of nonductile behavior or rapidly propagating fracture can be established for all pressure-retaining components of the reactor coolant pressure boundary.

The use of Appendix G of Section III of the Code, and the results of fracture toughness tests performed in accordance with the Code and Commission regulations in establishing safe operating procedures, provide adequate safety margins during operating, testing, maintenance, and postulated accident conditions. Compliance with these Code provisions and Commission regulations constitutes an acceptable basis for satisfying the requirements of Criterion 31 of the General Design Criteria.

We reviewed the proposed control of preheat of ferritic steel welding for conformance with the requirements of the Code, Section III, Appendix D, Paragraph D-1200, supplemented by Regulatory Guide 1.50.

The controls imposed on welding preheat temperatures are in conformance with the recommendations of Regulatory Guide 1.50. These controls provide reasonable assurance that cracking of components made from low alloy steels will not occur during fabrication and minimize the possibility of subsequent cracking due to residual stresses being retained in the weldment.

5.2.3.3 Fabrication and Processing of Austenitic Stainless Steel

We have reviewed the information provided by Westinghouse on the criteria for testing, controlling alloy composition, and heat treatment to avoid sensitization in austenitic stainless steels.

Westinghouse has stated that the possibility of intergranular stress corrosion in austenitic stainless steel used for components of the reactor coolant pressure boundary will be minimized because sensitization will be avoided, and adequate precautions will be taken to prevent contamination during manufacture, shipping, storage, and construction.

Austenitic stainless steel is subject to hot cracking (microfissuring) during welding if the weld metal composition or the welding procedure is not properly controlled. Because cracks formed in this manner are small and difficult to detect by nondestructive testing methods, welding procedures, weld metal compositions, and delta ferrite percentages that minimize the possibility of hot cracking must be specified. The proposed welding procedures have been reviewed.

The controls imposed upon components constructed of austenitic stainless steel used in the reactor coolant pressure boundary conform to the recommendations of Regulatory Guide 1.31 and Regulatory Guide 1.44. Material 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 in part the requirements of Criteria 1 and 14 of the General Design Criteria.

5.2.4 Reactor Coolant Pressure Boundary Inservice Inspection and Testing

5.2.4.1 Evaluation

Criterion 32 of the General Design Critera requires that components which are part of the reactor coolant pressure boundary be designed to permit periodic inspection and testing of important areas and features to assess their structural and leaktight integrity. Inservice inspection programs are based on Section XI of the Code, "Rules for Inservice Inspection of Nuclear Power Components."

1547 231

We reviewed the inservice inspection program for Quality Group A components of the reactor coolant pressure boundary. These components are also Code Class 1 components.

We reviewed Westinghouse's definition of the reactor coolant pressure boundary against the inspection requirements of Section XI of the Code for all Class 1 pressure-containing components (and their supports) except for those components excluded under IWB-1220 of Section XI. The RESAR-41 reactor coolant pressure boundary includes all pressure vessels, piping, pumps, and valves which are part of the reactor coolant system, or connected to the reactor coolant system, up to and including:

(1) The outermost containment isolation valve in system piping that penetrates the primary reactor containment.

(2) The second of two valves normally closed during normal reactor operation in system piping that does not penetrate the primary reactor containment.

(3) The reactor coolant system safety and relief valves.

We reviewed the design and arrangement of system components to determine conformance with the requirements of WA-1500, "Accessibility," of Section XI of the Code. Westinghouse has stated that the design of the reactor coolant system incorporates provisions for access for inservice inspection in accordance with Section XI of the ASME Boiler and Pressure Vessel Code. Tools and equipment have been designed 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 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. Inservice inspection of the Roto-Lok assembly can be performed in accordance with the requirements of Section XI, 1974 Edition.

We reviewed Westinghouse's proposed examination techniques and procedures for inservice inspection of the Reactor Coolant Pressure Boundary. Westinghouse has stated that the inservice inspection program for Code Class 1, 2 and 3 pressure retaining components and systems within the scope of RESAR-41 will comply with the requirements of Section XI of the Code, 1974 Edition. The inservice testing programs for pumps and valves that are part of the Section III Code Class 1, 2 and 3 safety related systems will comply with the requirements of Section XI of the Code. The examinations will be conducted in accordance with Section III of the Code, Appendix IX or equivalent procedures.

We reviewed the pressure-retaining Code Class 1 component leakage and hydrostatic pressure test program.

5.2.4.2 Conclusions

To ensure that no deleterious defects develop during service, selected welds and weld heat-affected zones will be inspected periodically. Westinghouse has stated that the designs of Code Class 1 components of the reactor coolant pressure boundary incorporate provisions for access for inservice inspections in accordance with Section XI of the Code, and that methods 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 leakage and hydrostatic testing of pressure-retaining components of the reactor coolant pressure boundary in accordance with the requirements of Section XI of the 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. Compliance with the inservice inspections required by this Code constitutes an acceptable basis for satisfying the requirements of Criterion 32 of the General Design Criteria.

5.3 Reactor Vessel and Appurtenances

5.3.1 Reactor Vessel Materials

5.3.1.1 Evaluation

Criterion 31 of the General Design Criteria requires that the reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions the boundary will behave in a nonbrittle manner and the probability of rapidly propagating fracture will be minimized.

We reviewed the material specifications used for the reactor vessel and applicable appurtenances such as the shroud support, studs, control rod drive housings, vessel support skirt, stub tubes, and instrumentation housings. Their adequacy for use in the construction of such components was assessed on the basis of the material, mechanical, and physical properties, the effects of irradiation on these materials, their corrosion resistance, and fabricability.

Fracture toughness of the ferritic materials to be used for reactor vessels and appurtenances thereto were reviewed to assure that such components will behave in a nonbrittle manner and that the probability of rapidly propagating fracture will be minimized under operating, maintenance, testing, and postulated accident conditions. The review included the descriptions of the fracture toughness tests to be performed on all ferritic materials that will be used for the reactor vessel and appurtenances thereto, and considered the acceptability of the proposed transverse Charpy-V-notch impact test specimens, dropweight test specimens, and any other test specimens included by Westinghouse in its program.

The test procedures specified by Westinghouse were reviewed.

The toughness properties of the reactor vessel beltline material will be monitored throughout service life with a material surveillance program that will meet all the requirements of ASTM E 185-73 and Appendix H of 10 CFR Part 50.

The composition of ferritic materials employed for the reactor vessel were reviewed and the amount of residual elements such as copper, sulfur, and phosphorous were checked.

We determined that the composition of reactor vessel beltline material, including welds, will be controlled to minimize the copper and phosphorus content, thus ensuring that the sensitivity to radiation damage will be low.

Although the use of controlled composition material for the reactor vessel beltline will minimize the possibility that irradiation will cause serious degradation of the toughness properties, Westinghouse has stated that should results of tests indicate that the toughness is not adequate, the reactor vessel can be annealed to restore the toughness to acceptable levels.

We reviewed the adequacy of the reactor vessel material surveillance program to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline.

We reviewed the end of life fluence calculated for the vessel beltline, the maximum predicted shift in reference transition temperature, the number of capsules, and the number and types of specimens to be placed in the capsules and found them acceptable. We conclude that the program is in compliance with ASTM E 185-73 and Appendix H of 10 CFR Part 50.

The materials for the Roto-Lok fasteners used to hold the reactor vessel head were reviewed to determine their adequacy. Mechanical properties, including fracture toughness, were checked to assure that all requirements are met.

We determined that the fastener material will fulfill the acceptance levels of Paragraph IV.A.4 of Appendix G to 10 CFR Part 50.

The proposed nondestructive examination of the Roto-Lok feature is discussed in Section 5.2.4 of this report.

5.3.1.2 Conclusion

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

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

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

Special controls and special welding processes used for welding the reactor vessel and its appurtenances have been identified and found to be qualified in accordance with the requirements of Sections III and IX of the Code. The controls imposed on welding preheat temperatures will be in conformance with the requirements of Regulatory Guide 1.50, and 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 controls imposed upon austenitic stainless steel welds will be in conformance with Regulatory Guide 1.31.

The fracture toughness tests required by the Code and by Appendix G of 10 CFR Part 50 provide reasonable assurance that adequate safety margins against the possibility of nonductile behavior or rapidly propagating fracture can be established for all pressure-retaining components of the reactor coolant boundary. The use of Appendix G of the Code as a guide in establishing safe operating procedures, and use of the results of the fracture toughness tests performed in accordance with the Code and Commission regulations, will provide adequate safety margins during operating, testing, maintenance, and postulated accident conditions. Compliance with these Code provisions and Commission regulations constitute an acceptable basis for satisfying the requirements of Criterion 31 of the General Design Criteria.

5.3.2 Pressure-Temperature Limits

The following pressure-temperature limits to be imposed on the reactor coolant pressure boundary during operation and tests were reviewed to assure that they will provide adequate safety margins against nonductile behavior or rapidly propagating failure of ferritic components of the reactor coolant pressure boundary, as required by Criterion 31 of the General Design Criteria.

(1) Pressure-temperature limits for preservice hydrostatic tests.
(2) Pressure-temperature limits for inservice leak and hydrostatic tests.
(3) Pressure-temperature limits for heatup and cooldown operations.
(4) Pressure-temperature limits for core operation.

Appendices G and H of 10 CFR Part 50 describe the conditions that require pressure-temperature limits and provide the general basis for these limits. These appendices specifically require that pressure-temperature limits must provide safety margins at least as great as those recommended in Section III, Appendix G, "Protection Against Nonductile Failure," of the Code during heatup, cooldown and test conditions. Appendix G of 10 CFR Part 50 also requires additional safety margins whenever the reactor core is critical (except for low-level physics tests).

Actual operating limit curves cannot be determined at the preliminary design stage because the fracture toughness and other required tests have not been performed on the actual material that will be used. Typical curves, with temperatures shown relative to the reference transition temperature, and the basis for determining the curves were reviewed and compared with the acceptance criteria described below.

We evaluated each limit curve for acceptability by performing check calculations using the simplified methods referenced in the Code and the Welding Research Council (WRC) Bulletin 175, "PVRC Recommendations on Fracture Toughness."

1547 235

Westinghouse has stated that the reactor is capable of being operated in a manner that will minimize the possibility of rapidly propagating failure, in accordance with Appendix G to Section III of the Code (1974 Edition) and Appendix G of 10 CFR Part 50.

The use of Appendix G of the Code as a guide in establishing safe operating limitations, using results of the fracture toughness tests performed in accordance with the Code and Commission regulations, will assure adequate safety margins during operating, testing, maintenance, and postulated accident conditions. Compliance with these Code provisions and Commission regulations, constitute an acceptable basis for satisfying the requirements of Criterion 31 of the General Design Criteria.

5.3.3 Reactor Vessel Integrity

The integrity of the reactor vessel is of such importance that a special summary review of all factors relating to the integrity of the reactor vessel is warranted. All portions of RESAR-41 relating to the integrity of the reactor vessel were reviewed to assure that the information is complete, and that no inconsistencies in information or requirements exist that would reduce the certainty of vessel integrity.

We have reviewed all factors contributing to the structural integrity of the reactor vessel and conclude there are no special considerations that make it necessary to consider potential reactor vessel failure for the proposed RESAR-41 design. The bases for our conclusion are that the design, materials, fabrication, inspection, and quality assurance requirements for the plant will conform to applicable Commission regulations and regulatory guides, and to the rules of the ASME Boiler and Pressure Vessel Code, Section III. The stringent fracture toughness requirements of the regulations and ASME Code Section III will be met, including requirements for surveillance of vessel material properties throughout service life. Also, operating limitations on temperature and pressure will be established for the vessel in accordance with Appendix G, "Protection Against Nonductile Failure," of Section III of the Code and Appendix G of 10 CFR Part 50.

The integrity of the reactor vessel will be assured because the vessel:

(1) Will be designed and fabricated to the high standards of quality required by the ASME Boiler and Pressure Vessel Code and any pertinent Code Cases;

(2) Will be made from materials of controlled and demonstrated high quality;

(3) Will be subjected to extensive preservice inspection and testing to provide assurance that the vessel will not fail because of material or fabrication deficiencies;

(4) Will be required by the Commission to 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, and that the vessel will not fail under the conditions of any of the postulated accidents;

(5) Will be required by the Commission to be subjected to periodic inspection to demonstrate that the high initial quality of the reactor vessel has not deteriorated significantly under service conditions; and

(6) Can be annealed to restore the material toughness properties if this becomes necessary.

5.4 Component and Subsystem Design

5.4.1 Reactor Coolant Pumps

5.4.1.1 Description

The reactor coolant pumps will be sized to provide adequate core cooling flow to maintain a departure from nucleate boiling ratio greater than 1.3 under normal and transient operating conditions. The estimated design loop flow will be 36.19 million pounds per hour.

Sufficient pump rotational inertia (110,000 pounds-feet squared) will be provided by a flywheel, in conjunction with the impeller and motor assembly, to provide flow during coastdown which is adequate to maintain a departure from nucleate boiling ratio greater than 1.3 in the event of a loss of pump power.

The reactor coolant pump will be a vertical, single stage, centrifugal, shaft seal pump. Suction will be from the bottom and discharge will be horizontal. The pump will be composed of three areas, the hydraulics, the shaft seals, and the motor.

5.4.1.2 Pump Flywheel Integrity

Criterion 4 of the General Design Criteria requires that structures, systems, and components of nuclear power plants important 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 Westinghouse and the staff. The Electrical Power Research Institute has contracted Combustion Engineering, Incorporated to perform a 1/5 scale reactor coolant pump research program. 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. Results from the program are expected by the fall of 1976. We will be following the development and performance of this program as well as other industry analytical and experimental programs on a generic basis.

If the results of the generic investigations of this matter indicate that additional protective measures are warranted to prevent excessive pump overspeed or to limit potential consequences to safety related equipment, we will determine what modifications, if any, are necessary to assure that an acceptable level of safety is maint ined. If modifications are necessary, we will require Westinghouse to make them.

Information in RESAR-41 on materials selection and the procedures used to minimize flaws and improve mechanical properties were reviewed to establish that sufficient information is provided to permit an evaluation of the adequacy of the flywheel materials.

The fracture toughness of the materials, including materials tests, correlation of Charpy specimens to fracture toughness parameters, or the alternate use of a nilductility transition reference temperature were reviewed to establish that the flywheel materials will exhibit adequate fracture toughness at normal operating temperature. The flywheel design information including allowable stresses, design overspeed considerations, and shaft and bearing design adequacy was reviewed. The overspeed test procedures were reviewed to establish their adequacy.

Normal and anticipated transient conditions are used by Westinghouse as the basis for the design of the flywheel. The design speed of the flywheel is 125 percent of the normal synchronous speed of the motor. In addition, the completed flywheel will be subjected to 100 percent volumetric, ultrasonic inspection using procedures and acceptance criteria equivalent to those specified for Class 1 components in Section III of the ASME Boiler and Pressure Vessel Code.

Analysis has shown that pump seizure results in shaft failure in torsion below the coupling to the motor, thus disengaging the flywheel and motor from the shaft. Following such an incident the motor will continue to run without overspeed and the flywheel will maintain its integrity. The loss of flow incident is evaluated in Section 15.4.

The probability of a loss of pump flywheel integrity can be minimized by the use of suitable material, adequate design, preservice spin testing, and inservice inspection. Westinghouse's selection of materials, fracture toughness tests, design procedures, preservice overspeed spin testing program, and inservice inspection program for reactor coolant pump flywheels have been reviewed and found acceptable on the basis of conformance with Regulatory Guide 1.14, and established industry codes and standards.

The use of suitable materials with adequate fracture toughness, conservative design procedures, preservice testing, and inservice inspection for flywheels of reactor coolant pump motors provide reasonable assurance of the structural integrity of the flywheels in the event of design overspeed transients or postulated accidents. Conformance with the recommendations of Regulatory Guide 1.14 constitutes an acceptable basis for satisfying the applicable portions of Criterion 4 of the General Design Criteria.

5.4.2 Steam Generators

5.4.2.1 Description

The steam generators will be vertical shell and U-tube evaporators with integral moisture separators. The primary reactor coolant will enter the steam generator lower hemispherical head and flow through the U-tubes giving up heat to generate steam on the shell side of the unit. The U-tube and tubesheet boundary will be designed to withstand full reactor coolant side design pressure and temperature with atmospheric pressure on the secondary side so as to prevent the activity generated within the primary system from passing over to the secondary system. Since the steam generators must provide a heat sink for the primary reactor coolant system during certain shutdown conditions, they will be at a higher elevation than the core to assure natural circulation flow for decay heat removal.

A main steam line flow restrictor, consisting of a disc with seven venturi-type nozzles, will be welded inside each steam generator steam outlet nozzle. It will be designed to limit the blowdown rate of steam from the steam generators in the event of a main steam line rupture.

Feedwater flow must pass through a preheater section of the steam generator before entering the boiler section of the steam generator. In the preheater section, the feedwater will be heated almost to the saturation temperature. The steam-water mixture which flows up through the tube bundle must pass through a set of centrifugal moisture separators which will remove most of the entrained water. The remaining steam will then pass through steam dryers to raise the steam quality before leaving the steam generator. The proposed RESAR-41 steam generators will be similar to those used in RESAR-3 type plants except that the heat transfer surface area has been increased through the use of more, and longer tubes.

The secondary side overpressure protection system is not within the scope of the RESAR-41 design and is therefore not described. A relief capacity of 105 percent of the rated steam flow has been assumed by Westinghouse in the primary side pressure relief analysis. This therefore is an interface requirement that must be considered in the balance of plant design.

5.4.2.2 Steam Generator Materials

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 reviewed the selection and fabrication of materials and the steam generator design.

We determined that the materials that will be used in the Class 1 components of the steam generators are selected and will be fabricated according to codes, standards, and specifications that we find acceptable.

We determined that the steam generators will be designed to avoid crevices where the tubes pass through the tube sheet and where the tubes pass through tubing supports.

Conformance with applicable codes, standards, staff positions, and regulatory guides constitutes an acceptable basis for meeting the applicable portions of Criteria 14, 15 and 31 of the General Design Criteria.

5.4.2.3 Steam Generator Inservice Inspection

Criteria 1 and 32 of the General Design Criteria require that components which are part of the reactor coolant pressure boundary or other components important to safety be designed to permit periodic inspection and testing of critical areas for structural and leaktight integrity. The design of the steam generators as described in RESAR-41 was reviewed to establish that use of the specified inspection techniques is feasible.

We conclude that the steam generators have been designed to permit inservice inspection of all Code Class 1 and 2 components including individual tubes as recommended in Regulatory Guide 1.83 and Section XI of the Code. Conformance with Regulatory Guide 1.83 and Section XI of the Code constitutes an acceptable basis for meeting the applicable portions of Criteria 1 and 32 of the General Design Criteria.

5.4.3 Residual Heat Removal System

The residual heat removal system will be designed to remove decay heat and sensible heat from the reactor coolant system and core during the latter stages of cooldown. The system will also control the reactor coolant temperature during refueling, and provide the means for filling and draining the refueling cavity. The system will consist of three parallel flow trains each consisting of a residual heat removal heat exchanger, a residual heat removal pump, and the associated valves and instrumentation necessary for operational control. The inlet lines to the system will be connected to the hot legs of three of the reactor coolant system loops and the return lines will be connected to the cold legs of the same three reactor coolant system loops. Each residual heat removal heat exchanger and the piping joining it to a reactor coolant system cold leg will be shared with one of the emergency core cooling low pressure injection systems. The valve arrangement will be such that at all times the emergency core cooling system can inject into the reactor vessel should the need arise. This will not limit or hamper the residual heat removal function of the heat exchangers.

The residual heat removal system will be placed into operation approximately four hours after initiation of plant shutdown when the temperature and pressure of the reactor coolant system are below 350 degrees Fahrenheit and 400 pounds per square inch, guage, respectively. Assuming operation of the three pumps and three heat exchangers, and that each heat exchanger will be supplied with component cooling water at design flow and temperature, the residual heat removal system is designed to reduce the reactor coolant system temperature from 350 to 150 degrees Fahrenheit within eight hours after being placed into operation. If one or two of the three pumps or heat exchangers were not operable, safe cooldown of the plant would still be possible but the time required for cooldown would be extended.

1547 240

In compliance with Criterion 34 of the General Design Criteria, the RESAR-41 residual heat removal system will be capable of performing the normal shutdown cooling function in the event of a single active or passive failure, with or without offsite power available. In addition to the above requirement, the system will be able to perform its function by remote control from the control room in accordance with Criterion 19 of the General Design Criteria. Use of the system for normal plant cooldown will not compromise the use of the heat exchanger by the safety injection system. The valves associated with the system will normally be aligned in such a way as to allow use of the necessary portions of the system for emergency core cooling should the need arise.

However, the proposed residual heat removal system design does not meet all of our requirements for compliance with Criterion 34 of the General Design Criteria. The specific deficiencies are:

(1) Alarms have not been provided in the control room to alert the operator if either suction valve is open when the reactor coolant system pressure exceeds the residual heat removal system design pressure.

(2) The valves do not have independent diverse interlocks to provide power actuation to automatically close each suction valve if the pressure in the reactor coolant system increases above the design pressure of the residual heat removal system.

(3) The system has not been shown to be capable of bringing the reactor to a cold shutdown condition (below 200 degrees Fahrenheit) within a short period of time, such as approximately 24 hours, even with a single failure.

We require that the residual heat removal system design be modified to incorporate the design features identified in (1), (2) and (3) above or other design modifications resulting in an equivalent level of protection for the system in a manner acceptable to the staff prior to issuance of a Preliminary Design Approval. We will report resolution of this item in a subsequent report.

Our evaluation of the electrical valve interlocks is found in Section 7.4.1 of this report.

The residual heat removal system will be inspected by applicants referencing RESAR-41 periodically during normal plant operation. Recalibration of the instrumentation channels, should it be necessary, will be done during each refueling operation.

5.4.4 Pressurizer

The pressurizer will maintain the reactor coolant system pressure during steady-state operation and will limit pressure changes during transients. It will contain a water volume sized to permit the reactor system to experience a step load increase of 10 percent at full power without uncovering the electrical heaters in the pressurizer and to maintain the pressure high enough so as not to activate the high pressure injection

1547 241

system. Above the water level will be a volume of steam sized to prevent water relief through the safety valves following a loss of load with credit taken for the pressurizer high water level initiating a reactor trip and without reactor control or steam dump. The steam volume will be large enough to accommodate the surge resulting from a 50 percent reduction of full load with automatic reactor control and 40 percent steam dump without the high water level reactor trip point being reached. No reactor trip will occur if the secondary system limits the primary system to a step change of 10 percent.

Electric heater bundles, located in the lower section, and water spray nozzles in the top head of the pressurizer will maintain the steam and water at the saturation temperature which corresponds to the desired reactor coolant system pressure. During outsurges, as the system pressure decreases, some of the water will flash to steam limiting the pressure drop and the electric heaters will act to restore the normal operating pressure. During insurges, as system pressure increases, some steam will naturally condense limiting the pressure increase while the automatic water spray will condense more steam to reduce the pressure to the normal operating level. Three ASME Code safety valves will be connected to the upper pressurizer head to relieve system overpressure. Three motor operated relief valves will also be provided to limit the lifting frequency of the safety valves.

The safety and relief valves will discharge to the pressurizer relief tank, located within containment. For overpressure protection for anticipated transients and accident conditions, which is discussed in Section 5.2.2 of this report, credit will only be taken for safety valve operation.

5.4.5 Pressurizer Relief Tank

The pressurizer relief tank will condense and cool the discharge from the pressurizer safety and relief valves. The tank will normally contain water and a predominately nitrogen atmosphere. However, provision will be made to permit the gas in the tank to be periodically analyzed to monitor the concentration of hydrogen and/or oxygen.

By means of its connection to the waste processing system, the pressurizer relief tank will provide a means for removing any non-condensable gases from the reactor coolant system which might collect in the pressurizer vessel. The tank design is based on the requirement to absorb a pressurizer discharge of pressurized steam equal to 110 percent of the volume above the full-power pressurizer water level set point. The volume of water in the tank will be capable of absorbing the heat from the assumed discharge, assuming an initial temperature of 120 degrees Fahrenheit and increasing to a final temperature of 200 degrees Fahrenheit. If the temperature in the tank rises above 120 degrees Fahrenheit during plant operation, the tank will be cooled by spraying in cool water and draining out the warm mixture to the waste processing system.

1547 242

Rupture discs on the relief tank will provide sufficient relief capacity (1.6 million pounds per hour at 100 pounds per square inch, guage) to prevent tank overpressurization. The tank design pressure of 100 pounds per square inch, guage, will be equal to twice the calculated pressure resulting from absorption of the design discharge from the pressurizer. The tank and rupture discs holders will also be designed for full vacuum to prevent tank collapse if the contents cool following a discharge without the normal addition of nitrogen. Based on the analyses presented in Section 15 of RESAR-41, for any anticipated transient the pressurizer relief tank pressure will not exceed the design pressure of the rupture discs. Therefore there is no anticipated transients for which reactor coolant would be released to the containment.

We conclude that the design bases, system description, and safety evaluation for the pressurizer relief tank are acceptable.

5.4.6 Safety and Relief Valves

The pressurizer safety valves will be the totally enclosed pop type valve. The valves will be spring loaded, self activated and with back pressure compensation features. The combined capacity of the pressurizer safety valves will be designed to accommodate the maximum surge resulting from complete loss of load. The pressurizer safety valves, with a total relieving capacity of 1.26 million pounds per hour, will prevent reactor coolant system pressure from exceeding 110 percent of system design pressure of 2500 pounds per square inch, guage, in compliance with Section III of the Code. This objective will be met without reactor trip or any operator action provided that the secondary system steam safety valves, assumed to have a capacity of 105 percent of rated steam flow, open as designed when the steam pressure reaches the steam-side safety setting.

The relief valves will be quick-opening and operated automatically or by remote control. Remotely operated stop valves will be provided to isolate the power-operated relief valves if excessive leakage develops. The pressurizer power-operated relief valves, each with a relieving capacity of 210,000 pounds per hour, will be designed to limit pressurizer pressure to a value below the high pressure reactor trip setpoint for all design transients up to and including the design step load decrease with steam dump.

5.4.7 Loose Parts Monitor

Occasionally, miscellaneous items such as nuts, bolts, and other small items have become loose parts within the 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 such reasons, for the past few years we have required many applicants 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 being installed at several plants.

RESAR-41 includes, as an option, a loose parts monitoring system. We will impose the requirement for installation of an acceptable loose parts monitoring system for each applicant referencing RESAR-41.

5.4.8 Rapid Refueling System

5.4.8.1 Introduction

The rapid refueling system will be composed of five main design developments (1) a quick release reactor head called the Roto-Lok closure system, (2) el[illegible]ion of head electrical disconnects, (3) a one-lift concept in which the missile shield, reactor vessel head, upper core-support structure, and rod custer control assemblies will be removed as a single unit, (4) withdrawal of the rod cluster control assemblies into the head and upper internals package where they will be held withdrawn during refueling, and (5) modified fuel handling capability. Figure 5-2 shows how these features (except for fuel handling) will be incorporated in the reactor vessel and head.

5.4.8.2 Roto-Lok Closure System

The new system used to attach the reactor vessel head to the reactor vessel will employ 36 closure studs modified with breech-block lugs (Roto-Lok) for attachment to the vessel. The lugs on the bottom end of the Roto-Lok studs will be engaged or released from the reactor vessel flange by a 60-degree rotation (Fig. 5-3). Identical lugs on the top portion of the stud will mate with the adapter of the hydraulic stud tensioner. The tensioner will be locked to the stud by a 60-degree rotation of the tensioner adapter and hydraulic pressure will be applied to stretch the stud; a closure nut will be rotated as necessary to release or hold the tension on the stud; the hydraulic pressure will then be released and the tension removed.

Westinghouse has submitted supplemental material in WCAP-8447 described below, indicating that analysis and testing completed to the present time has been solely in support of the developmental program, and that the final design and analysis for a specific vessel is to be performed by the vessel supplier when the Roto-Lok is actually applied to production vessels according to the ASME Section III Code.

The Roto-Lok design and development testing as presented in WCAP-8447, "Roto-Lok Closure System Development," has been reviewed and accepted for reference in RESAR-41. However, as we have indicated to Westinghouse, we require additional information to analyze the consequences of a failure of a single Roto-Lok stud. The remaining units in the Roto-Lok system must be shown by suitable analyses or tests, or a combination of analyses and testing, to be structurally adequate to assure that the head will not come off following the failure of one stud.

1547 244

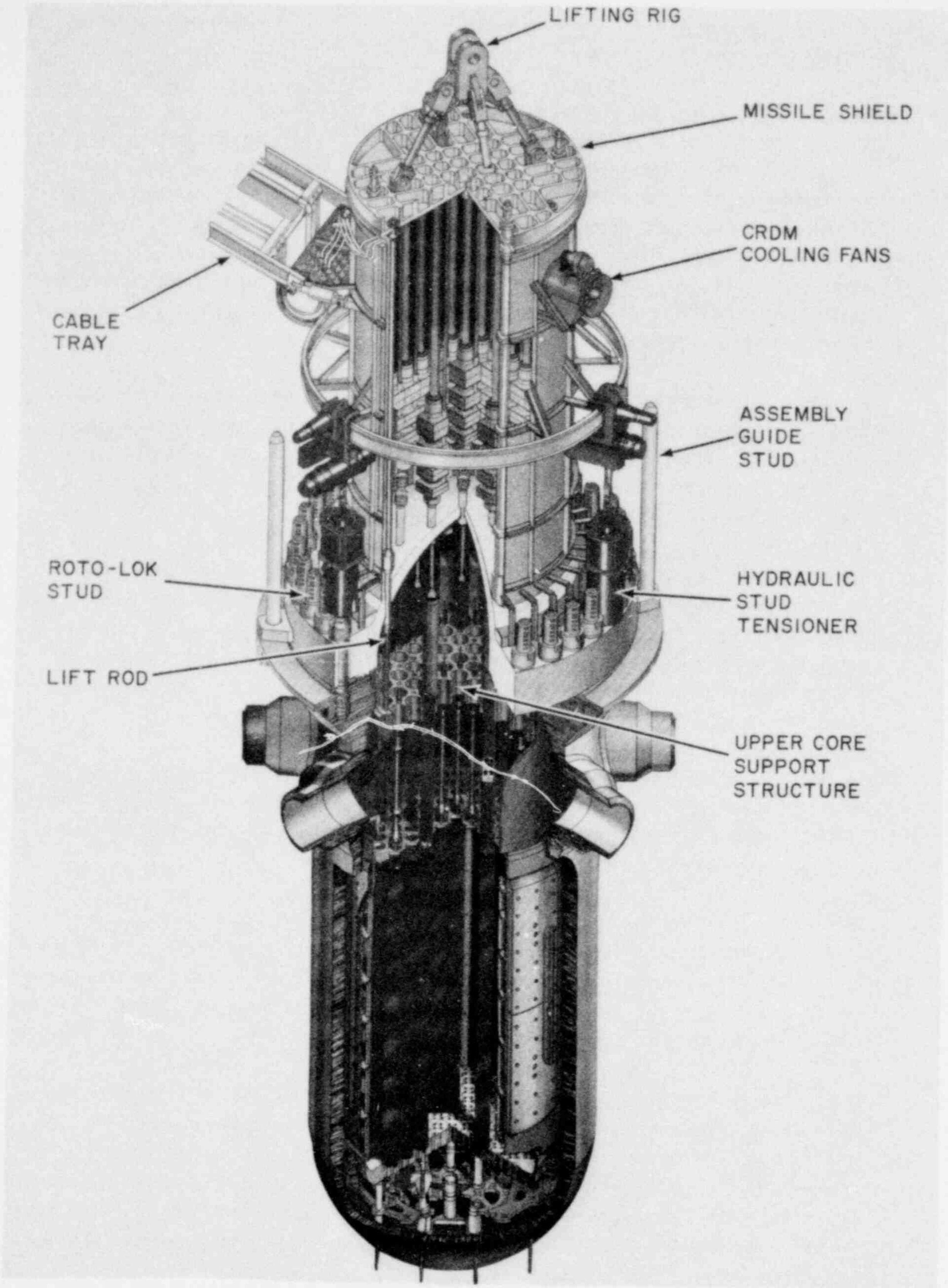


Figure 5-2 RESAR 41 REACTOR VESSEL AND INTERNALS SHOWING RAPID REFUELING HARDWARE.

1547 245

WCAP-8447 was submitted by Westinghouse in June 1975. We will report our evaluation of this submittal in a subsequent report.

5.4.8.3 Elimination of Electrical Disconnects

A cable tray (Fig. 5-3) will permit all electrical connections to remain connected when the reactor vessel head is removed, thereby eliminating the time spent in making and verifying electrical connections following refueling. The tray will be a bridge-type structure, approximately 36 feet long by eight feet wide, spanning from the cavity wall to the head cooling shroud. One end of the cable tray will be supported by a hinged connection to the head shroud and the other end will be supported by guide rollers that rest on the refueling canal wall. Utilization of this tray is graphically shown in Figure 5-4 of this report.

The cable tray will carry power cables for the control rod drive mechanisms, signal cables for the rod position indicator system, and signal cables for in-core thermocouples. The cables in the cable tray will be clamped at both ends of the tray and supported in spaces between by anchor brackets.

5.4.8.4 One Lift Concept

The rapid refueling upper package will combine the series of operations of missile shield removal, control-rod drive mechanism cooling duct removal, and upper core support structure removal into a single-lift operation as shown in Figure 5-4. This will be accomplished by connecting the missile shield to the head, connecting the upper core support structure to the head, and providing an integral control rod drive mechanism cooling system (Fig. 5-2).

The missile shield designed for the rapid refueling system will be located at the top of the head shroud structure and will be attached to the reactor closure head by four lifting rods. This design will replace the conventional concrete-and-steel shield that must be rolled back and out of the way prior to refueling.

The new missile shield design will also serve functions in addition to missile protection. It will be provided with large clearance holes to give lateral seismic support to the control rod drive mechanism assemblies. The missile shield will also serve as a spreader bar for the lifting rig, transmitting the load from the lifting rig through the lifting rods to the reactor vessel head.

The design of the missile shield structure over the control rod drive mechanisms to block missiles which might be associated with a fracture of the pressure housing of any mechanism will consist of three-inch thick steel plate stiffened by steel ribs placed orthogonally. The steel missile shield thickness required to prevent perforation was determined by the Stanford formula which is acceptable to the staff. We conclude that the design procedure used to determine the required thickness of the control rod drive mechanism missile shield is acceptable.

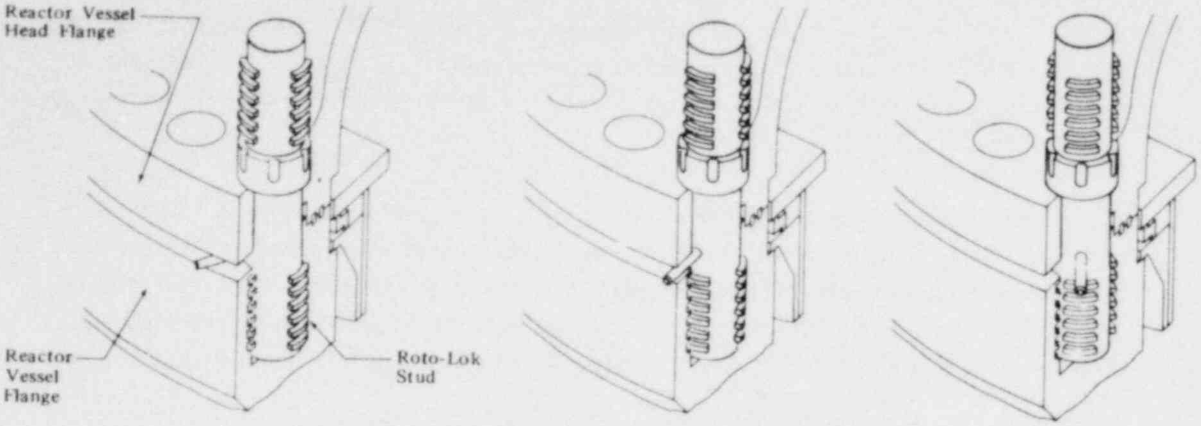


Figure 5-3 ROTO-LOK CLOSURE CONCEPT

The stud used in the Roto-Lok reactor vessel close closure system has breechblock lugs rather than conventional threads. The lugs can be engaged or disengaged by 60-degree rotation.

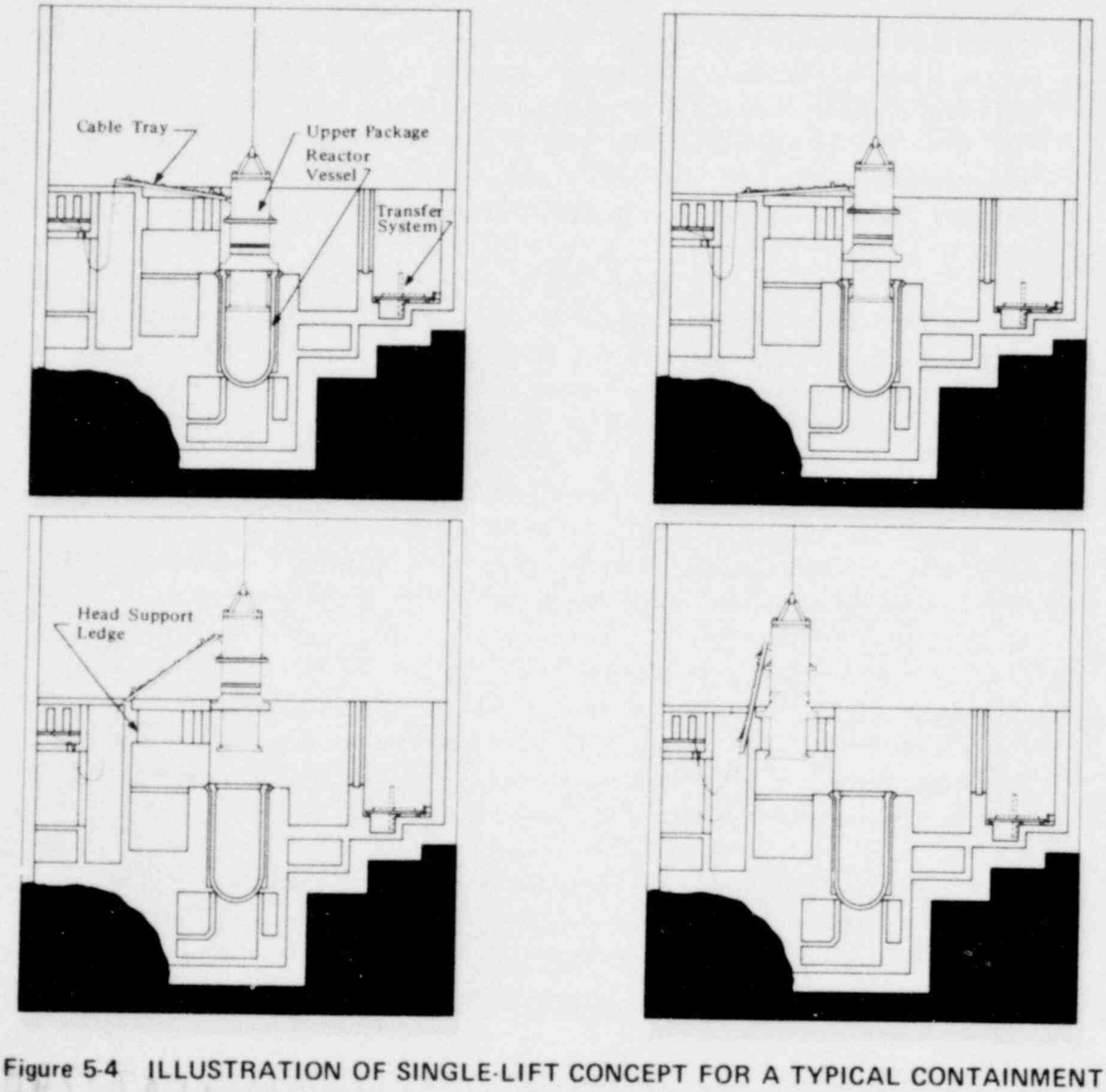


Figure 5-4 ILLUSTRATION OF SINGLE-LIFT CONCEPT FOR A TYPICAL CONTAINMENT

The closure head will be connected to the upper core support structure by three lifting rods, which will be attached by threading to the upper support plate and penetrate the closure head in adapters similar to those provided for the control rod drive mechanism pressure housings.

This arrangement will permit the head and upper core support structure to be removed and inserted as a unit. Under certain conditions, however, it might be necessary to disconnect the upper core support structure from the closure head for separate removal. This can be achieved by unthreading the lifting rods to permit the head, control-rod drive shafts, and upper core support structure to be removed in a conventional manner.

Westinghouse's proposed one-lift concept entails lifting as a unit the reactor vessel head with the upper internals and control rods to a greater distance above the reactor vessel than has been previously required. This has raised concern about the integrity of the reactor coolant system and the ability to maintain core geometry and provide adequate core cooling assuming the reactor vessel head were dropped. Accordingly, we require that Westinghouse provide an analysis of the consequences from dropping of the reactor vessel head assembly. The analysis should assess the extent of damage to the core and the reactor coolant system and should demonstrate that core cooling capability would be maintained. We will require that this analysis be submitted for our review so that any design changes required to limit damage to acceptable levels or to eliminate the dropped head accident as a design basis requirement can be approved prior to issuance of the Final Design Approval.

To avoid the time required for disconnecting the control rod drive mechanism cooling system, a forced-air cooling system will be fitted into the head shroud structure. The cooling system will consist of four fans mounted to the upper section of the cooling shroud. Ducts will be located inside the cooling shroud to carry air from below the mechanisms.

Westinghouse has proposed that the cooling fans and their associated circuitry not be designed to Class IE standards. Their basis for this is that the temperature of the control rod drive mechanisms is not of concern to the safety of the plant.

If all cooling were lost at normal operating temperature, a loss of insulation life would occur which would eventually result in shorting of the coils and tripping of the control rods. The control rod drive mechanism internals are designed to operate in 650 degrees Fahrenheit coolant indicating that cooling fans are not required for proper mechanical operation of the mechanism internals.

In addition, the control rod drive mechanism design will be tested as described in Section 1.4. These tests will determine if the new holdout device will interfere with safe operation of the assembly. See Section 4.2.4 for our evaluation of the reactivity control system.

1547 248

We have reviewed the information provided by Westinghouse on the control rod drive mechanism cooling system and conclude that the design is acceptable.

5.4.8.5 Removal of Rod Cluster Control Assemblies

The rapid refueling system requires that all rod cluster control assemblies will be completely withdrawn and stored in the reactor upper core support structure guide tubes following boration of the reactor cooling system to the required refueling-shutdown boron concentration. Our mechanical evaluation of the reactivity control system is contained in Section 4.2.4.

A backup holdout device (or fail-safe lock) will be provided on the full-length control rod drive mechanisms for this purpose. Part-length mechanisms will not require the holdout device because their design will prevent them from being released on loss of power.

The backup holdout device will be completely isolated electrically from the magnetic gripper coils during normal plant operation. When the holdout device is energized, it will raise a latch bar into position behind the stationary gripper latch arms to hold them engaged. The mechanical latch that results will no longer require electrical power.

With all control rods removed from the core during refueling, adequate shutdown margin will be maintained by the boron content of the coolant.

Prior to initiating refueling operations the reactor coolant system will be borated and cooled down to the refueling shutdown conditions of more than 5 percent subcritical and less than 150 degrees Fahrenheit. Criticality protection for refueling operations, including a requirement for daily checks of boron concentration, will be specified in the technical specifications.

Subsequent to establishing refueling shutdown conditions the control rods will be fully withdrawn and their holdout devices actuated. All of the control rods will then be removed at one time as part of the single lift of the head, upper internals and control rods.

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 content 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 rod cluster control assemblies inserted in the core. With or without these assemblies, the subcriticality of the system must be maintained using soluble boron. The technical

1547 249

specifications will require that the boron content be sufficient to maintain the core at least 5 percent subcritical including uncertainties. The consequences of inadvertent boron dilution during refueling without rod cluster control assemblies are discussed in Section 15.4.

Based on our review of the information provided by Westinghouse, we conclude that core subcriticality during refueling can be maintained in an acceptable manner.

5.4.8.6 Fuel Handling System

The fuel handling system will consist of the equipment needed for the transfer of spent fuel assemblies from the reactor vessel through the transfer canal to the spent fuel storage facility.

The equipment described in RESAR-41 includes a manipulator crane, spent fuel pit bridge, new fuel elevator, fuel transfer system, rod cluster control changing fixture, new and spent fuel handling tools, reactor vessel head and upper internals lifting device, reactor internals lifting device, and the reactor vessel stud tensioner. The new fuel handling crane, spent fuel cask crane, reactor overhead crane and the building facilities will be provided by the utility applicant. The RESAR-41 equipment will be designed for underwater handling of the spent fuel from the time it leaves the reactor vessel until it is placed in a spent fuel shipping cask.

We have reviewed the design bases, systems description, and safety evaluation for the fuel handling system and conclude that the design is acceptable.

5.5 Conclusion

We conclude that the proposed design of the reactor coolant system conforms to the Commission's regulations and to applicable regulatory guides, staff technical positions, and industry standards, and is acceptable with the following provisions:

(1) We reserve final judgment regarding anticipated transients without scram until such time as the Westinghouse generic anticipated transients without scram report has been reviewed. Depending on the outcome of that review, we may find it necessary to apply further requirements on the RESAR-41 reactor coolant system, or components thereof. See Section 15.5.7.

(2) The evaluation of the overpressure protection system is under review as part of the review of WCAP-7769. We will complete our evaluation for the final design.

(3) The residual heat removal system must meet our design criteria as stated in Section 5.4.3 of this report.

(4) The reactor vessel head cluster studs must be shown to be adequate to withstand the failure of one stud during normal operation.

(5) The consequences from dropping of the reactor vessel head assembly during refueling must be shown to be acceptable.

6.0 ENGINEERED SAFETY FEATURES

6.1 Summary

The purpose of the various engineered safety features will be to provide a complete and consistent means of assuring that the plant personnel and the public will be protected from excessive exposure to radioactive materials should a major accident occur in the plant. In this section we discuss the engineered safety feature systems proposed for the RESAR-41 nuclear steam supply system. Certain of these systems or parts of these systems will have functions for normal plant operation as well as serving as engineered safety features.

We have reviewed the proposed systems and components designated as engineered safety features. These systems and components 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 requirements and to function even with the 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 in accordance with Criterion 35 of the General Design Criteria.

6.2 Containment Systems

RESAR-41 describes a nuclear steam supply system utilizing a four-loop reactor coolant system, a 3800 thermal megawatt pressurized water reactor, and associated auxiliary systems. The containment systems for a nuclear generating station utilizing the RESAR-41 design will include a reactor containment structure, containment heat removal systems, containment isolation systems and containment combustible gas control systems. However, RESAR-41 includes only the containment isolation system for the systems within the scope of RESAR-41.

The containment will be described in the application for the balance of the plant. Westinghouse has provided mass and energy release information that would result from loss-of-coolant accident to be used in establishing the containment design conditions and designing the containment subcompartments. Containment design pressure evaluations, the design pressure evaluations of containment internal structures, i.e., subcompartment designs, and the containment response to ruptures in the secondary system will be provided by applicants referencing RESAR-41. The containment building type (i.e., dry, subatmospheric pressure suppression) is not specified within RESAR-41. For any containment type the mass and energy information provided by Westinghouse in RESAR-41 will be acceptable for containment design purposes provided the maximum calculated containment pressure is less than that assumed by Westinghouse in calculating the mass and energy release rates resulting from a loss-of-coolant accident (i.e., 52 pounds per square inch, absolute, in the containment following the initial blowdown).

6.2.1 Containment Functional Design

The containment will provide a low leakage barrier that encloses the nuclear steam supply system; i.e., the reactor, steam generators, reactor coolant pumps and pressurizer as well as certain components of the engineered safety features. RESAR-41 contains no specific information on containment design. However, the effects of operation and accident conditions of the RESAR-41 systems on the containment design must be accounted for.

Westinghouse has calculated the mass and energy release produced by a loss-of-coolant accident. Double-ended break sizes in the hot leg, and the pump suction and discharge of a reactor coolant pump were analyzed. In addition, a 60 percent double-ended break and a three square foot break at the pump suction were analyzed. This information will be utilized by the designer for the balance of plant in establishing the design requirements (pressure and temperature) for the containment building. Secondary system ruptures will be analyzed by the balance of plant designs. Based on our evaluation of the containment pressure response for reactor systems similar to that proposed in RESAR-41, we expect that the primary system break sizes provided will be sufficient to establish the containment design basis accident. In the event the containment pressure analysis for the balance of plant does not demonstrate the most severe break size to be provided within RESAR-41, we will require additional break sizes to be analyzed on an individual plant basis. For example, if the double-ended hot leg break or the double-ended pump discharge break produces the maximum containment pressure, we will require additional smaller break sizes to be analyzed so that the break size producing the highest containment pressure will be adequately bracketed.

The computer methods used by Westinghouse to evaluate mass and energy release to the containment during the blowdown and reflooding periods (SATAN-V, LOCTA and W REFLOOD) were designed for emergency core cooling system evaluation and were approved for use in meeting the Interim Acceptance Criteria established by the Commission for emergency core cooling system performance (see Section 6.3.5 of this report). The evaluation of energy sources, however, has been determined on the basis of containment design and therefore certain assumptions differ from those used for emergency core cooling system analyses to maximize the effect on the calculation of containment pressure.

The SATAN-V computer code was used by Westinghouse to determine the mass and energy addition rates to the containment during the blowdown phase of the accident. To obtain a conservatively high energy release from the core during the blowdown period a detailed calculation was made using the LOCTA core analysis code to determine core energy release. The core average channel was modeled in LOCTA using core inlet conditions calculated by the SATAN-V code. In this analysis nucleate boiling was assumed for the core heat transfer for an extended period to maximize the energy release rate from the core. This additional energy release was added to that calculated by the SATAN-V code. By this method, 80 percent of the available energy was removed from the core during blowdown which yielded an average temperature of the peak fuel pellet in the average channel of less than 520 degrees Fahrenheit for the double-ended pump suction break. Under these assumptions, the core transfers more heat to the containment than would be predicted by a calculation suitable for emergency core cooling performance evaluation. This additional energy release from the core will increase the calculated containment pressure and therefore assure a margin of conservatism in the analysis. The SATAN-V and the LOCTA computer codes have been accepted by the staff for calculating mass and energy released during a loss-of-coolant accident.

The time delay for the lower plenum to be refilled to the core bottom has not been considered for containment analysis. Westinghouse has conservatively assumed that the bottom of the core will be recovered immediately after the end of blowdown. Thus, the reflood period will begin immediately after the end of blowdown.

Westinghouse has calculated the mass and energy that will be released to the containment during the reflood phase of the accident using the W REFLOOD computer code described in WCAP-8170, "Calculational Model for Core Reflooding After a Loss-of-Coolant Accident (W REFLOOD Code)." Assumptions were made to maximize heat transfer from the core and steam generators and steam flow to the containment. The W REFLOOD code has been accepted by the staff for calculation of mass and energy release during the reflooding period as part of the emergency core cooling system evaluation model by letter to Westinghouse dated May 30, 1975.

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 will 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 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 ruptures of the cold leg at the pump suction will result in the highest mass flow through the core, and thus through the steam generators.

The W REFLOOD code calculates mass and energy release rates using a loop hydraulic resistance model and an energy balance model. The hydraulic model determines the core flooding rate whereas the energy balance model calculates the core exit conditions and the energy addition from the steam generator. The core exit flow rate is calculated from the core flooding rate and a correlation based on the results of the Full Length Emergency Cooling 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.

The heater bundle of the Full Length Emergency Cooling Heat Transfer experiments is 12 feet in length. The test results show that liquid entrainment continues until the fuel is recovered with water to about the eight-foot elevation. At this time the fuel clad temperature transient ceases (i.e., quenching occurs). For plants with 12-foot cores, Westinghouse has conservatively assumed quenching of the core at the 10-foot elevation for containment functional design calculations. The reactor proposed in RESAR-41 has a 14-foot core and Westinghouse has assumed that quenching occurs at the 12-foot elevation. We conclude this approach to be a conservative extrapolation of the experimental data.

The rate of steam flow from the reactor to the containment during the reflooding period will be dependent on the containment pressure and will increase with increasing pressure. This is due to a decrease of hydraulic resistance to steam flow in the reactor loops with increasing containment pressure. Westinghouse has selected a containment pressure of 52 pounds per square inch, absolute, for analysis of the reflood transient. The value of the mass and energy calculations will therefore be conservative for plants with a calculated containment pressure less than 52 pounds per square inch, absolute. For any application referencing RESAR-41 and for which the calculated containment pressure is higher than 52 pounds per square inch, absolute, we will require additional analyses of the mass and energy release rates.

Westinghouse 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 resulting in a two-phase mixture of steam and water in the core. The calculations performed by Westinghouse assumed that this two-phase mixture rises above the core and subsequently enters the steam generators. By this process the remainder of the available steam generator energy will be removed by boiling of the water entrained in the two-phase mixture and carried into the containment. In calculating the rate of energy removed from the steam generators, Westinghouse has used the maximum steam flow based on the hydraulic resistance of the system and steam generator heat transfer.

A sensitivity study was made in which the flow distribution of core produced steam between the one broken and three unbroken loops (flow split) was varied. A flow split of 95 percent for steam entering the broken loop was found to maximize the mass and energy release to the containment and was used in this analysis.

Data from steam-water mixing tests such as described in Combustion Engineering, Incorporated topical reports CENPD-63, Rev. 1, March 1973, "1/5 Scale Intact Loop Post-LOCA Steam Relief Tests," and CENPD-101, October 1973, "Steam Water Mixing Test Program Test D," 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. Westinghouse has conservatively accounted for steam and water mixing only during the post-reflood period when heat removal is calculated in the steam generator. During the reflood period no quenching is assumed for containment calculations.

After 1205 seconds into the postulated loss-of-coolant accident, the steam generators will have cooled to the temperature of saturated steam at the containment pressure and the primary source of heat release to the containment will be decay heat. As the containment depressurizes additional sensible heat will be released from the core primary metal and steam generators. This energy release is a function of the containment depressurization rate. Westinghouse has provided long-term mass and energy release data in RESAR-41 that assumes a conservatively short depressurization rate of one hour for release of sensible heat. No credit is taken in the analysis for steam quenching of the decay heat by emergency core cooling system water during this phase.

We have reviewed Westinghouse's calculational methods and assumptions and conclude that the calculated mass and energy release data is conservative for containment design purposes.

Westinghouse had calculated the mass and energy release to the containment for the short-term period following a loss-of-coolant accident for use in the analysis of pressure increases in the various containment building internal compartments. Typical compartments are those formed by the reactor cavity and the steam generator shield walls. The designs of these compartments will be supplied in utility applications referencing RESAR-41.

The SATAN-V code is used to calculate these mass and energy release rates. This code has been accepted by us for emergency core cooling system analysis. Westinghouse has made noding studies which demonstrate that a convergent solution is obtained and has made further conservative assumptions which act to maximize the mass and energy release rate to the containment. We conclude that the method described by Westinghouse 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 RESAR-41 may not be appropriate. For example, the subcompartment design and piping restraints may preclude occurrence of the full size piping breaks analyzed in RESAR-41. In the event that pipe restraints are utilized or other design features of a specific balance of plant cause use of the break sizes and locations analyzed in RESAR-41 to be inappropriate, we will require applicable mass and energy release analyses with appropriate justification to be presented in the particular application referencing RESAR-41.

The methodology for calculating mass and energy release from secondary system ruptures and the pressure response for both subcompartment and containment design considerations will be presented for our review in applications referencing RESAR-41.

6.2.2 Containment Isolation Systems

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 systems within the scope of RESAR-41 are evaluated herein. The detailed description of isolation provisions for the balance of plant will be supplied in application referencing RESAR-41.

Reactor building penetration piping up to and including the external isolaton 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 valve or piping failure can result in the loss of containment integrity.

The reactor building isolation signal will be activated by high reactor building pressure. Certain containment isolation valves including those in the containment ventilation lines will also isolate following low steamline pressure or low primary system pressure. Steamline and feedwater line isolation will occur on containment high pressure or low steam generator pressure.

1547 255

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 the portion of the containment isolation system within the scope of RESAR-41 and conclude that it is in conformance with Criteria 59, 55, 56, and 57 of the General Design Criteria and Regulatory Guide 1.11 and is acceptable.

6.2.3 Combustible Gas Control in Containment

Following a loss-of-coolant accident, hydrogen may accumulate inside the containment. The major sources of hydrogen generation include:

(1) A chemical reaction between the zirconium fuel rod cladding and steam.

(2) Radiolysis of aqueous solutions in the reactor core and in the containment sump.

(3) A chemical reaction between construction materials and water or reactive spray solutions.

Westinghouse has analyzed the post-loss-of-coolant accident hydrogen generation from the nuclear steam supply systems described in RESAR-41 with respect to items (1) and (2) above. This analysis is consistent with the guidelines of Regulatory Guide 1.7.

In our evaluation of applications referencing RESAR-41, we will consider any additional hydrogen source terms and assure that the assumptions used in the RESAR-41 analysis are consistent with the balance of plant design and with the resulting containment hydrogen concentration. We will also review the provisions for atmospheric mixing within the containment.

A hydrogen sampling system, hydrogen recombiners, and a backup purge system design will be described in applications referencing RESAR-41.

6.3 Emergency Core Cooling System

6.3.1 Design Bases

Criterion 35 of the General Design Criteria and Section 50.46 of 10 CFR Part 50 require that an emergency core cooling system shall be provided which can perform its safety function assuming a single failure.

The proposed design of the emergency core cooling system as given in RESAR-41 has been substantially changed from earlier Westinghouse designs. As in previous designs, the RESAR-41 emergency core cooling system will be designed to provide emergency core cooling during those postulated accident conditions where it is assumed that mechanical failures occur in the reactor coolant system piping resulting in loss of coolant from the reactor vessel greater than the available coolant makeup capacity using normal operating equipment. However, the RESAR-41 emergency core cooling system will not be designed to provide complete protection against steam line break consequences; for this postulated

1547 256

accident, the emergency boration system described in Section 6.5 of this report will be employed in conjunction with the high head safety injection subsystems of the emergency core cooling system.

The system design bases are to prevent fuel and cladding damage that would interfere with adequate emergency core cooling and to mitigate the amount of clad-water reaction for any size break up to and including a double-ended rupture of the largest primary coolant line. Westinghouse has stated that these requirements will be met even with minimum engineered safeguards available, as would be the case with the loss of one emergency power bus together with the unavailability of offsite power.

The emergency core cooling system to be provided will have the required number, diversity, reliability, and redundancy of components such that no single active failure of emergency core cooling system equipment during the short term or no single active or passive failure during the long term of an accident will result in inadequate cooling of the reactor core. Each of the proposed emergency core cooling system subsystems will be designed to function over a specific range of reactor coolant piping system break sizes, up to and including the flow area associated with a postulated double-ended break in the largest reactor coolant pipe (10.48 square feet is the double-ended area).

6.3.2 System Design

In a design basis loss-of-coolant accident, mass and energy will be released from the postulated pipe break to the containment. These releases will occur over a time period depending upon the particular loss-of-coolant accident that has been postulated. Within this time period several phases may be considered to occur in terms of blowdown, refill, reflood, and post-reflood phases. These are discussed separately below.

The blowdown phase of the accident is the time immediately following the occurrence of the postulated break during which most of the mass and energy contained in the reactor system will be released to the containment, i.e., the primary coolant, and the metal and core stored energy. The refill phase is that time during which the lower reactor vessel plenum will be refilled to the bottom of the core by the emergency core cooling system.

The reflood phase is that time during which the core will be recovered by the emergency core cooling system and, for cold leg breaks, the time period during which most of the secondary energy will be removed from the steam generators. The remaining energy in the secondary system along with decay heat from the reactor core, will be released to the containment during the post-reflood period.

For hot leg breaks the broken piping will provide a direct path for fluid from the core to travel directly into the containment without passing through the steam generators. Therefore the secondary system energy will be removed at a much slower rate.

The emergency core cooling system will be made up of three separate and independent subsystems. Each subsystem will consist of a passive accumulator, a high head safety injection pump, and a low head injection pump that will deliver borated water to one of three of the four reactor coolant system cold legs. As can be seen from Figure 6-1,

1547 257

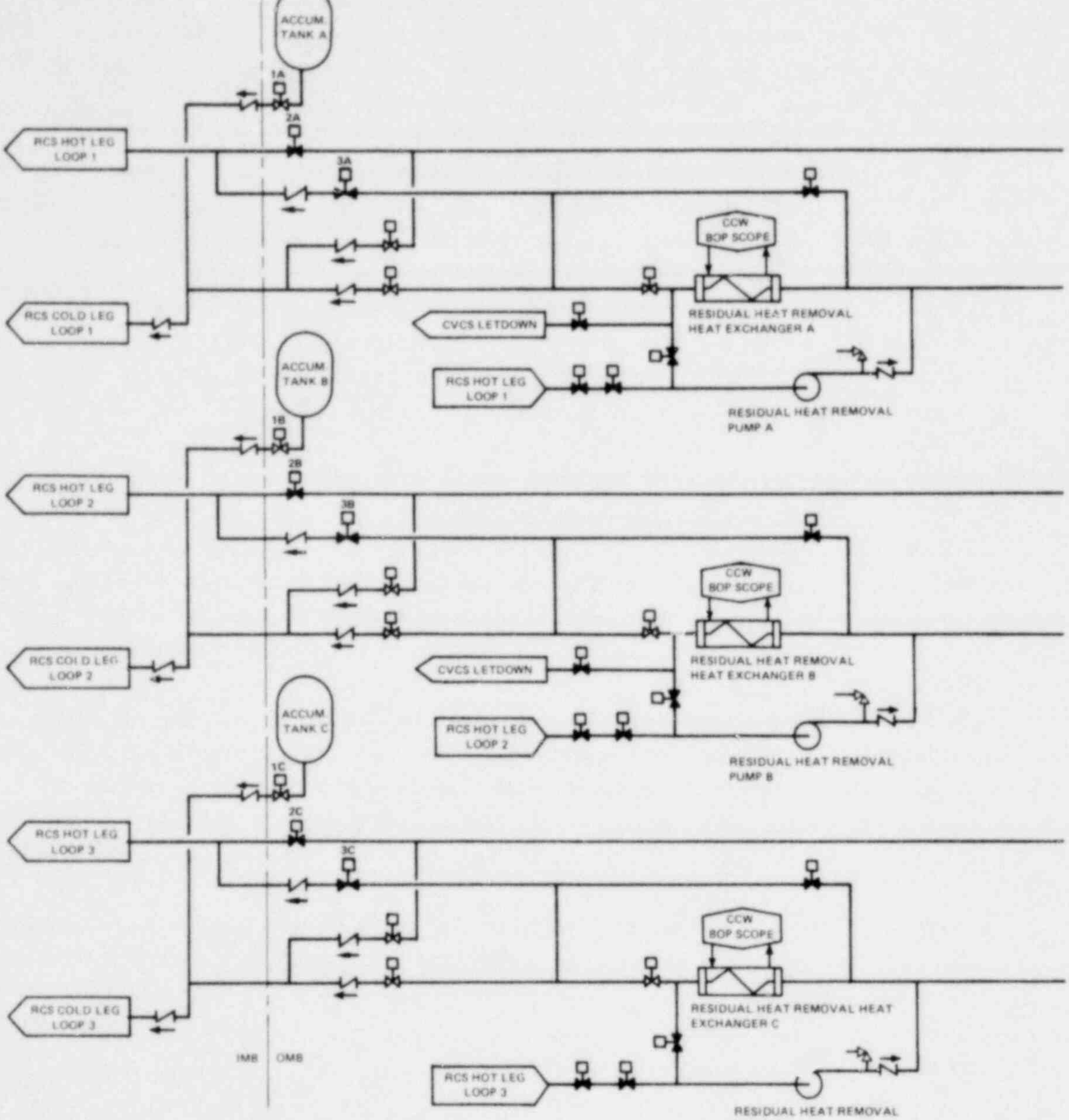
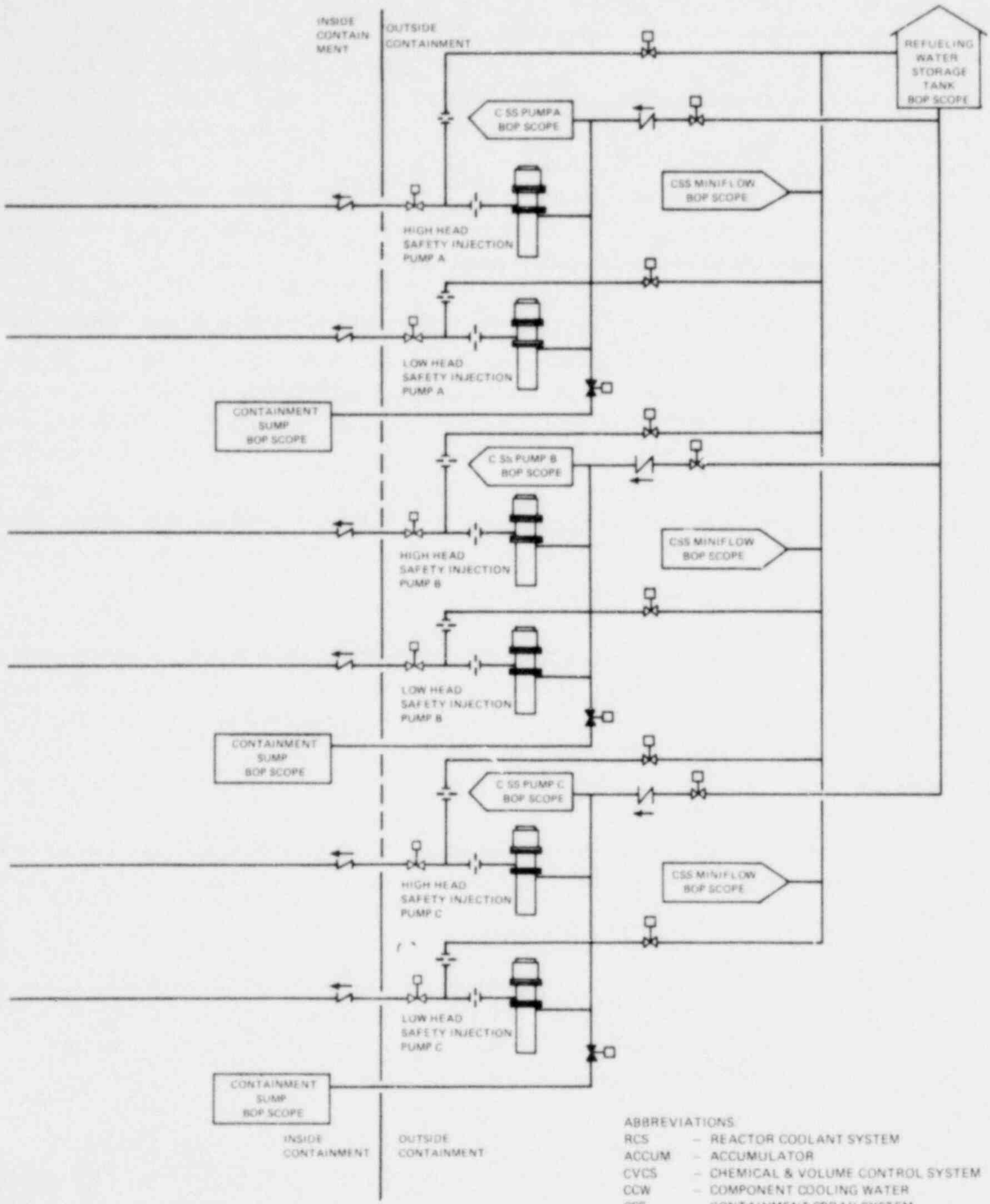


Figure 6-1 RESIDUAL HEAT REMOVAL SYSTEM/ EMERGENCY CORE COOLING SYSTEM PROCESS FLOW DIAGRAM

1547 258



ABBREVIATIONS

RCS – REACTOR COOLANT SYSTEM
ACCUM – ACCUMULATOR
CVCS – CHEMICAL & VOLUME CONTROL SYSTEM
CCW – COMPONENT COOLING WATER
CSS – CONTAINMENT SPRAY SYSTEM
IMB – INSIDE MISSILE BARRIER
OMB – OUTSIDE MISSILE BARRIER

1547 259

there will be no interconnection or headering of subsystems (trains) and thus each train can be physically separated from the other two trains either by barriers or by distance, depending upon the plant layout chosen by the balance of plant designer. An independent actuator train and an independent electrical power train will be provided for each subsystem.

Following a postulated loss-of-coolant accident, the emergency core cooling system will be actuated by the engineered safety features actuation system. All high head and low head safety injection pumps will start and take suction from the refueling water storage tank. Initially, water will be injected into the core by the high head injection pumps and the passive accumulators. This will be followed by injection from the low head injection trains and finally, for long term cooling, by recirculation using the low head injection pumps connected to the containment sumps.

Each of the three accumulators will have a total volume of 2500 cubic feet with a minimum volume of borated water of 1500 cubic feet and a maximum volume of nitrogen gas of 1000 cubic feet at a minimum pressure of 600 pounds per square inch gauge. The minimum boric acid concentration will be 2400 parts per million. Each tank will be connected to one of the reactor coolant system cold legs with two check valves in series. A normally open motor operated gate valve will also be located in the lines between each accumulator and the cold leg piping. As discussed in Section 7.6.3 these valves will be provided with appropriate interlocks to assure that the valves will be open during power operations when availability of the accumulators is required.

Upon actuation of a safety injection signal, the high pressure injection mode of operation will consist of three high head safety injection pumps (rated at 800 gallons per minute each at a design head of 2850 feet), which will take their suction from the refueling water storage tank which will contain borated water at a concentration of 2500 parts per million.

Low pressure injection will be provided by three low head pumps (rated at 1400 gallons per minute design flow rate at a design head of 620 feet) which will take their suction from the refueling water storage tank.

The original RESAR-41 design required certain manual operations to be performed to transfer from the injection mode to the recirculation mode. We required that transfer be completely automatic. As a result, Westinghouse altered the design so that all required operations will be automatic.

Upon actuation of the low-level alarm from the refueling water storage tank, suction will be transferred automatically to the containment sump for the recirculation mode of operation. Then the emergency core cooling system will provide the long-term cooling requirements by recirculating the spilled reactor coolant (from the ruptured pipe) collected in the sump, back to the reactor vessel. The return of the sump water will be through the reactor coolant cold legs for twenty-four hours after the accident and through the hot legs thereafter.

1547 260

Compared to previous Westinghouse designs, the use of safety equipment for more than one purpose is reduced in the proposed RESAR-41 design. As can be seen from Figure 6-1 and the system descriptions provided herein, the charging pumps will not be used for safety injection, and the emergency core cooling system will not provide emergency boration for the steam break accident.

The only components which will have dual functions in this design are:

(1) The residual heat removal heat exchangers will be used for both normal cooldown and for long-term (recirculation) cooling following an accident. They will be lined up for safety injection during normal operations.

(2) The low head safety injection pumps will provide both low head injection immediately following the accident and recirculation cooling to the core for long-term cooling.

The proposed RESAR-41 design will require that one safety injection train and two accumulators inject into the core following a loss-of-coolant accident. This configuration results from the assumption of one train and one accumulator spilling out of the break with the loss of offsite power and failure of emergency power for one train.

6.3.3 Containment Pressure Response for Emergency Core Cooling Evaluation

Following a loss-of-coolant accident the pressure in the containment will increase as steam, resulting from the flashing of the primary coolant at the break, is added to the containment atmosphere. Following initial blowdown of the reactor coolant system, heat flow to the emergency core cooling system water from the core, primary metal structures, and steam generators will produce additional steam. This steam, together with emergency core cooling system water being spilled from the reactor coolant system, will flow through the break to the containment.

Energy removal will occur within the containment by several means. Steam condensation on containment walls and internal structures, which serve as passive heat sinks, is an effective energy removal mechanism early in the accident. Subsequently, operation of the containment spray system will also remove energy from the containment atmosphere. When the energy removal rate exceeds the energy addition rate from the primary system, the containment pressure will decrease.

For the purpose of emergency core cooling system evaluation it is conservative to minimize the containment pressure to increase the resistance to steam flow in the reactor coolant loops and reduce the reflood rate in the core. Appendix K to 10 CFR Part 50 of the Commission's regulations requires that the effect of operation of all installed containment pressure reducing systems and processes be included in the emergency core cooling system evaluation. In calculating the containment pressure for the emergency core cooling system analysis in accordance with Appendix K, a heat removal rate was assumed for the containment for heat removal systems. If the RESAR-41 containment pressure calculations are to remain valid for use by applications referencing RESAR-41, then the actual heat removal rate must be at least as great as that assumed in RESAR-41. The adequacy of the containment pressure analysis for the emergency core cooling system evaluation will be determined by the staff as part of our evaluation of the Appendix K requirements concerning emergency core cooling system. Westinghouse has informed us orally that this information will be submitted at about August this year. We will report the results of our evaluation in a subsequent report prior to issuance of a Preliminary Design Approval.

6.3.4 Design Evaluation

We reviewed the proposed design to determine that our diversity, reliability, and redundancy requirements will be met such that no single failure of the emergency core cooling system equipment will result in inadequate cooling of the reactor core as specified by Criterion 35 of the General Design Criteria. Specifically, we evaluated the system ability to withstand a single active failure during the short term, or a single active or passive failure during the long term following a loss-of-coolant accident.

Each injection train will have its own independent valving, power supplies, control circuitry, sump line, and injection point. However, Westinghouse has identified nine motor-operated valves in the proposed emergency core cooling system design which should not move from normal alignment during certain phases of the postulated loss-of-coolant accident. These valves and the required alignments are:

(1) Accumulator isolation valves, one valve to each of three accumulators, which must remain open. (Valves marked 1A, 1B, and 1C on Figure 6-1.)

(2) High head safety injection pump discharge hot leg isolation valves, one valve to each of three injection trains, which must remain shut. (Valves marked 2A, 2B, and 2C on Figure 6-1.)

(3) Low head safety injection pump discharge hot leg isolation valves, one valve to each of three injection trains, which must remain shut. (Valves marked 3A, 3B, and 3C on Figure 6-1.)

These valves will be manually controlled from the control room. They will normally be properly aligned for functioning in the event of a postulated loss-of-coolant accident and none of them will be required to be moved from their normal position in the short term following an accident.

We determined that the design would be unacceptable in those instances in which a single failure in an electric system could result in the loss of capability to perform a specified function. As a result, Westinghouse agreed to disconnect power to the electric system for these valves in lieu of corrective design changes.

Westinghouse has specified as interface requirements that restoration of power to these valves should be readily available to the operator. The specific means of achieving this requirement will be reviewed for all applications which reference RESAR-41.

We conclude that, with power removed from the valves identified above, the design proposed in RESAR-41 will be able to perform its safety function assuming a single failure and therefore satisfy the requirements of Criterion 35 of the General Design Criteria.

6.3.5 Performance Evaluation

The emergency core cooling system has been designed to deliver fluid to the reactor coolant system in order to control the predicted cladding temperature transient following a postulated pipe break and for removing decay heat in the long-term, recirculation mode.

On June 29, 1971, the Commission issued an Interim Policy Statement containing Interim Acceptance Criteria for the performance of the emergency core cooling system for light-water cooled nuclear power reactors. The Interim Policy Statement includes a set of conservative assumptions and procedures to be used in conjunction with computer codes to analyze and evaluate the emergency core cooling system function for a pressurized water reactor incorporating a dry containment. A public rule making hearing on the Interim Acceptance Criteria for the emergency core cooling system for light-water cooled nuclear power reactors has been conducted.

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 amendments to 10 CFR Part 50 to incorporate the ruling. Subparagraphs (a)(4) of paragraph 50.34 and (a)(1) of paragraph 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors," of the amended 10 CFR Part 50, state in part:

50.34(a)(4), "Analysis and evaluation of ECCS cooling performance following postulated loss-of-coolant accidents shall be performed in accordance with the requirements of §50.46 for facilities for which construction permits may be issued after December 28, 1974."

50.46(a)(1), "...each boiling and pressurized light-water nuclear power reactor fueled with uranium oxide pellets with cylindrical Zircaloy cladding shall be provided with an emergency core cooling system (ECCS) which shall be designed such that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in paragraph (b) of this section. ECCS 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, sets forth certain required and acceptable features of evaluation models."

These provisions are applicable to RESAR-41 since a decision for issuance of the construction permits for applications which reference RESAR-41 will be after December 28, 1974.

Accordingly, Westinghouse has committed to provide its analysis satisfying the requirements of these new criteria. In partial response we have received the analysis of what Westinghouse considers to be the worst case. Westinghouse will also submit an analysis to determine the minimum containment pressure in accordance with the requirements of Appendix K. We will review this information and our evaluation and conclusions will be included in a supplement to this report prior to the issuance of a preliminary design approval.

For the purpose of this report, we have reviewed and evaluated the emergency core cooling system performance in accordance with the Interim Policy Statement, which states that the performance of the emergency core cooling system is acceptable if the course of the loss-of-coolant accident is limited as follows:

(1) The calculated maximum fuel element cladding temperature does not exceed 2,300 degrees Fahrenheit.

(2) The amount of fuel element cladding that reacts chemically with water or steam does not exceed one percent of the total amount of cladding in the reactor.

(3) The clad temperature transient is terminated at a time when the core geometry is still amenable to cooling, and before the cladding is embrittled as to fail during and after quenching.

(4) The core temperature is reduced and decay heat is removed for an extended period of time, as required by the long-lived radioactivity remaining in the core.

Westinghouse has presented in RESAR-41 the evaluation of the loss-of-coolant accident in accordance with the requirements of the Interim Policy Statement. The evaluation resulted in a peak clad temperature of 2275 degrees Fahrenheit and showed compliance with the Interim Acceptance Criteria.

6.3.6 Tests and Inspections

Westinghouse has stated that the operability of the emergency core cooling system can be demonstrated by subjecting all components to preoperational tests, periodic testing, and in-service testing and inspections. The preoperational tests that will be performed by utility applicants referencing RESAR-41 fall into three categories:

(1) System actuation tests to verify (a) the operability of all emergency core cooling system valves initiated by the safety injection signal, the phase A containment isolation signal, and the phase B containment isolation signal, and (b) the operability of all safeguard pump circuitry down through the pump breaker control circuits and the proper operation of all valve interlocks.

(2) Accumulator injection tests to check the accumulator system and injection line to verify that the lines are free of obstructions and that the accumulator check valves and isolation valves operate correctly. The utility applicant will perform a low pressure blowdown of each accumulator with the reactor head and internals removed to meet the test objective.

(3) Safety injection pump tests to evaluate the hydraulic and mechanical performance of the pumps as they deliver through the required flow paths for emergency core cooling. The tests will be divided into two parts, pump operation under miniflow conditions, and pump operation at full flow conditions. By measuring the flow in each pipe, the applicant will make the adjustments necessary to assure that no one branch has an unacceptably low or high resistance. System checks will be made to ascertain that total line resistances are sufficient to prevent excessive runout of the pump.

For preoperational testing of the emergency core cooling system, Westinghouse has stated that it can be tested in accordance with Regulatory Guides 1.1, 1.68, and 1.79. However, Westinghouse does not at this time recommend that applicants referencing RESAR-41 comply with Regulatory Guide 1.79 Section C-3B; Low Pressure Recirculation Test. This test is necessary to verify pump net positive suction head, vortex control, and pressure drop across screens, piping, and valves. The guide states that "to avoid reactor coolant system contamination, the sump water may be discharged to external drains or other systems." We will require that this test be performed by applicants referencing RESAR-41.

The utility applicant referencing RESAR-41 will be required to perform routine periodic testing of the emergency core cooling system components and all necessary support systems with the plant at power. Valves that are required to operate after a loss-of-coolant accident will be operated through a complete cycle and pumps will be operated individually in these tests on their miniflow lines.

The utility applicant will also be required to use test circuits to periodically check for leakage of reactor coolant through the accumulator discharge line check valves to ascertain that these valves seat whenever the reactor coolant system pressure is above a preset value. The periodic emergency core cooling system testing will also include a visual inspection of pump seals, valve packings, flanged connections, and relief valves to detect leakage. Westinghouse has stated that the emergency core cooling system components will be designed and fabricated to permit inspection and inservice tests in accordance with ASME Code Section XI.

6.3.7 Conclusion

On the basis of our evaluation, we have concluded that the predicted functional performance of the proposed RESAR-41 emergency core cooling system to provide protection for the full spectrum of postulated break sizes is in accord with the Commission's Interim Policy Statement and acceptance criteria. As stated earlier, Westinghouse will be required to demonstrate compliance with the emergency core cooling system criteria published in the Federal Register on January 4, 1974, and our evaluation will be included in a supplement to this report prior to issuance of a Preliminary Design Approval.

6.4 Emergency Boration System

6.4.1 Design Basis

The emergency boration system will be designed to provide shutdown capability in the event of any single steam line rupture or spurious relief valve lifting. This system will be effective for a range of postulated steam line ruptures up to and including the double ended circumferential rupture of the largest single pipe in the steam system. Since the steam generators have integral steam line flow restrictors, the system design is based upon an effective steam break area of 1.4 square feet, the cross-sectional flow area of each flow restrictor.

6.4.2 System Design

The emergency boration system will consist of the boron injection tank, the boron injection surge tank, the boron injection recirculation loop, the boron injection pumps, and the associated valves. See Figure 6-2 in this report for a flow diagram of the system. The boron injection tank will contain 1,500 gallons of 20,000 parts per million boric acid solution and will be connected to the reactor coolant system by means of a loop consisting of a four-inch pipe between one reactor coolant pump's discharge side and the boron injection tank inlet. The boron injection tank outlet will be connected to the two boron injection pumps which will discharge through a common manifold pipe to pipes connected to each of the four reactor coolant pump's suction lines. The reactor coolant pump from which the reactor coolant is drawn through the boron injection tank will be the pump in the primary coolant loop not containing a safety injection tank.

The boron injection surge tank will contain 75 gallons of the same concentration of boric acid as the boron injection tank and will be used to supply surge capacity for the boron injection tank recirculation loop. During normal operation the boric acid solution will be recirculated by the recirculation pump continuously in a closed loop consisting of the boron injection tank and boron injection surge tank. This will be done to maintain mixing and prevent stratification. The safety injection signal will automatically stop the recirculation pumps and close the valves in the recirculation lines.

The system will be provided with two boron injection pumps in parallel. However, for the main steamline break analysis one pump is assumed to be inoperable as a result of an assumed single active failure. The size of the pump is thus established to assure adequate safety margin in the event of a main steamline break.

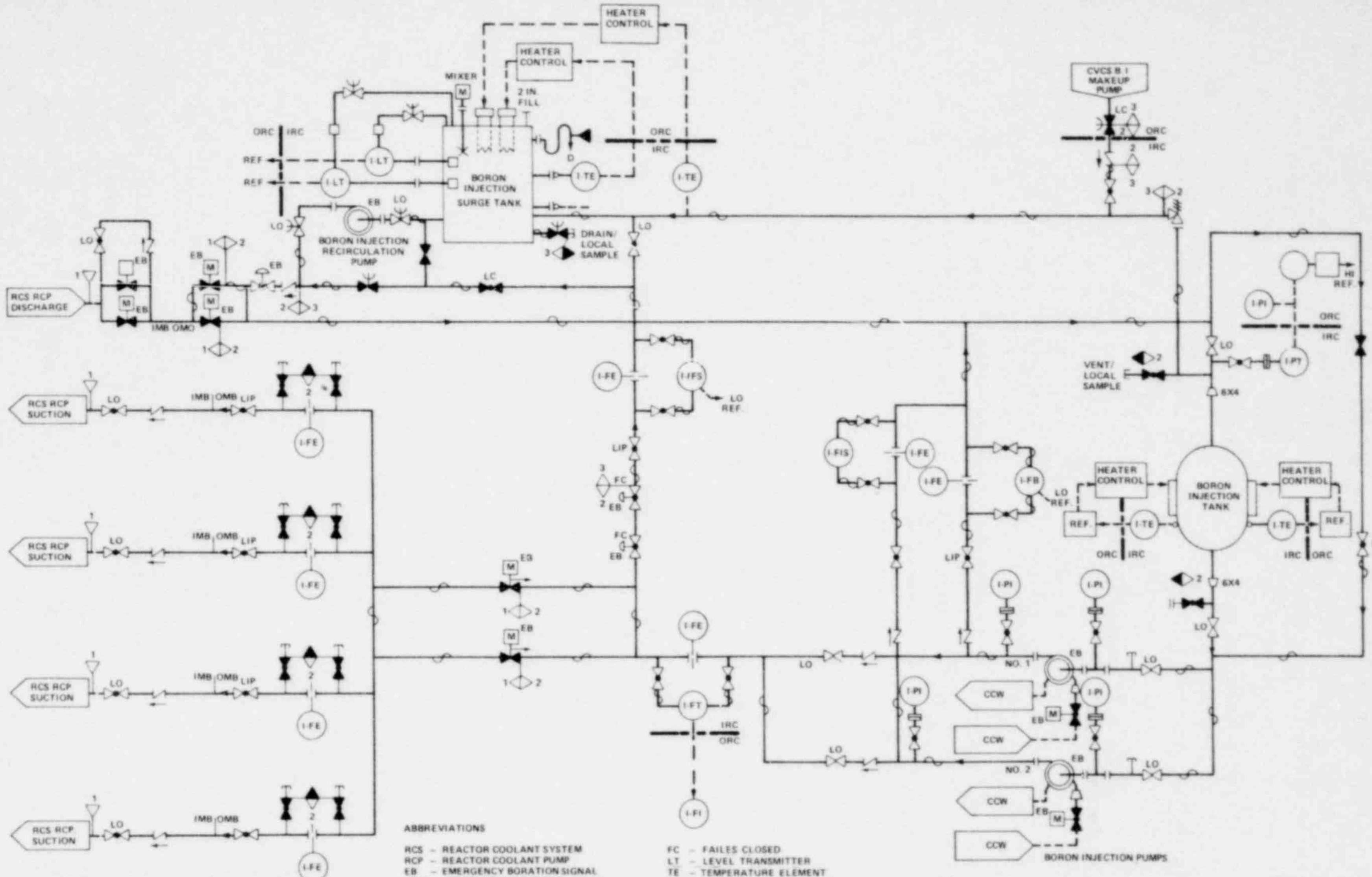
The emergency boration system will be actuated by an emergency boration signal produced by:

(1) Low temperature in one reactor coolant system cold leg, coincident with a reactor power level of less than 10 percent or a reactor trip.

(2) Manual actuation. Actuation of the boron injection system will also generate a safety injection signal.

The system has been designed to accept any single failure of any active component and still perform its intended function.

Redundant and separate heat tracing must be provided by utility applicants referencing RESAR-41. This heat tracing will be installed on all piping, valves, flanges, instrumentation lines and pump casings carrying the 20,000 parts per million boric acid



ABBREVIATIONS

RCS – REACTOR COOLANT SYSTEM
RCP – REACTOR COOLANT PUMP
EB – EMERGENCY BORATION SIGNAL
IMB – INSIDE MISSILE BARRIER
OMB – OUTSIDE MISSILE BARRIER
CCW – COMPONENT COOLING WATER SYSTEM
B.I. – BORON INJECTION SIGNAL
LO – LOCKED OPENED
LIP – LOCKED IN PLACE
IRC – INSIDE REACTOR CONTAINMENT
ORC – OUTSIDE REACTOR CONTAINMENT

FC – FAILES CLOSED
LT – LEVEL TRANSMITTER
TE – TEMPERATURE ELEMENT
FE – FLOW ELEMENT
FT – FLOW TRANSMITTER
FIS – FLOW INDICATING SENSOR
PI – PRESSURE INDICATOR
PT – PRESSURE TRANSMITTER
M – MOTOR
2 – DENOTES CHANGE OF SAFETY CLASS. IN THIS CASE FROM CLASS 2 TO NON NUCLEAR SAFETY.
– HEAT T'ACING ON PIPES

Figure 6-2 EMERGENCY BORATION SYSTEM

solution. This will minimize the potential for boric acid precipitation. As an added precaution against boric acid precipitation, the small lines which allow recirculation during normal operation will be provided with flow indication and alarms. If these lines become clogged the operator in the control room will be provided with flow indication allowing him to take the necessary corrective action.

6.4.3 Performance Evaluation

The emergency boration system has been designed to deliver concentrated boric acid solution to the reactor coolant system to control the reactivity insertion following a postulated steamline break. While the concentrated boric acid solution is being injected into the reactor coolant system, the shrinkage caused by the cooldown following a steamline break will be made up by the high head safety injection pumps. Safety injection will be terminated once the pressurizer level has been restored.

The steamline break analysis which is provided in Section 15.4 indicates that although limited fuel cladding damage is permissible for a condition IV accident, the minimum departure from nucleate boiling ratio does not go below 1.30. This indicates no fuel damage for the main steamline break accident.

6.4.4 Tests and Inspections

The operability of the emergency boration system can be demonstrated by subjecting all components to preoperational tests, periodic tests, and in-service tests and inspections for each plant employing the RESAR-41 nuclear steam supply system.

The preoperational test will be an integrated system test while the plant is at cold shutdown. The test will check the automatic actuation circuitry, valves and pumps to assure proper operation. A satisfactory test must (1) generate and transmit the emergency boration signal, (2) operate the valves properly, and (3) start and operate the pumps at the proper flow rate.

The utility applicant will perform periodic tests of each active system component during normal plant operation. The boron injection pumps will be able to be operated on their miniflow lines.

Westinghouse has stated that the emergency boration system components will be designed and fabricated to permit inspection and in-service tests in accordance with ASME Code Section XI and has stated that the emergency boration system has been designed to allow testing in accordance with the requirements of Regulatory Guides 1.68 and 1.79.

6.4.5 Conclusion

The emergency boration system will include the valves, pumps, boric acid tank, and recirculation equipment needed to provide reactivity control in the event of a steamline break. We have reviewed the drawings, component descriptions, and design criteria and have concluded that the emergency boration system will be designed to conform to the Commission's requirements as set forth in the General Design Criteria, Regulatory Guides, and staff technical positions. We conclude that the system will be capable of performing its function with only onsite electric power or with only offsite electric power, assuming the most restrictive single active failure.

We conclude that the proposed design of the emergency boration system is acceptable.

6.5 Engineered Safety Features Materials

We have reviewed the mechanical properties of materials selected for the engineered safety features and find that they will satisfy Appendix I, Section III, or parts A, B, and C, Section II of the ASME Code, and the staff position that the yield strength of cold worked stainless steels shall be less than 90,000 pounds per square inch.

The proposed controls on the use and fabrication of the austenitic stainless steel in the systems satisfy the recommendations of Regulatory Guides 1.31 and 1.44. Fabrication and heat treatment practices performed in accordance with these requirements provide added assurance that stress-corrosion cracking will not occur during the postulated accident time interval.

Applications referencing RESAR-41 must show that the controls on the hydrogen ion concentration (pH) of the reactor containment sprays following a postulated loss-of-coolant accident are adequate to assure freedom from stress-corrosion cracking of the austenitic stainless steel components and welds of the engineered safety features throughout the duration of the postulated accident to completion of cleanup. In addition, they must show that control of the acidity of the sprays provides assurance that the sprays will not give rise to hydrogen gas evolution by corrosion of the materials described in RESAR-41, in accordance with the recommendations of Regulatory Guide 1.7.

We have reviewed the selection of materials proposed for the engineered safety features, in conjunction with the expected chemistry of the cooling and containment spray system water. Westinghouse has shown that the use of sensitized stainless steel will be avoided. We have concluded that the proposed controls on material and cooling water chemistry will provide assurance that the integrity of components of these systems will not be impaired by corrosion or stress-corrosion.

Conformance with the Codes and Regulatory Guide recommendations mentioned above and with the stated position on the allowable maximum yield strength of cold worked austenitic stainless steel constitutes an acceptable basis for meeting the requirements of Criteria 35, 38, and 41 of the General Design Criteria.

1547 269

7.0 INSTRUMENTATION AND CONTROLS

7.1 General

The RESAR-41 instrumentation and control systems have been reviewed utilizing the Commission's General Design Criteria, the Institute of Electrical and Electronics Engineers (IEEE) Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations" of 1971, applicable Regulatory Guides for Power Reactors and staff technical positions as bases for evaluating their adequacy. The specific documents used in the review are listed below.

(1) RESAR-41 - Reference Safety Analysis Report through Amendment 14

(2) 10 CFR Part 50 and Appendix A to 10 CFR Part 50

(3) USAEC Regulatory Guides, Division 1, Power Reactors

(4) Institute of Electrical and Electronics Engineers (IEEE) Standards

   (a) IEE Std 279-1971 - "Criteria for Protection Systems for Nuclear Power Generating Stations"

   (b) IEEE Std 308-1971 - "Criteria for Class IE Electric Systems for Nuclear Power Generating Stations"

   (c) IEEE Std 323-1974 - "IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations"

   (d) IEEE Std 334-1971 - "Trial-Use Guide for Type Tests of Continuous Duty Class I Motors Installed Inside the Containment of Nuclear Power Generating Stations"

   (e) IEEE Std 336-1971 - "Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations"

   (f) IEEE Std 338-1971 - "Trial Use Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems"

   (g) IEEE Std 379-1972 - "Trial Use Guide for the Application of the Single Failure Criterion to Nuclear Power Generating Station Protection Systems"

   (h) IEEE Std 382-1972 - "Trial Use Guide for Type Test of Class I Electric Valve Operators for Nuclear Power Generating Stations"

   (i) IEEE Std 384-1974 - "Trial Use Standard: Criteria for Separation of Class IE Equipment and Circuits"

7.2 Reactor Trip System

The reactor trip system will be comprised of two to four redundant and independent channels per trip input. Input signals from nuclear instrumentation, process bistables or direct sensor contacts will operate miniature relays in the solid state input cabinet, whenever the conditions monitored reach a preset level. Contacts of the input relays will supply signals to the logic portion of the system, located in the adjacent logic cabinet. Electrical and physical isolation between redundant channels will be maintained through the input cabinet. The logic circuits can be connected to produce the various logic combinations such as "two-out-of-four", "one-out-of-two" etc. Two redundant logic trains will be provided for each reactor trip. Each logic train will be capable of operating a separate and independent reactor trip breaker through undervoltage release provided in the breaker. The two trip breakers in series will connect power for the control rods and when either of the trip breakers opens, power will be interrupted to the rod drive power supply, which will cause insertion of all rods by gravity. Bypass breakers will be provided to permit testing of the trip breakers.

The following is a list of reactor trips provided:

(1) Source range high neutron flux
(2) Intermediate range high neutron flux
(3) Power range high positive neutron flux rate
(4) Power range high negative neutron flux rate
(5) Power range high neutron flux
(6) Core overtemperature delta T (temperature difference)
(7) Core overpower delta T (temperature difference)
(8) High pressurizer pressure
(9) Low pressurizer pressure
(10) High pressurizer level
(11) Low reactor coolant flow
(12) Reactor coolant pump bus underpower
(13) Reactor coolant pump bus underfrequency
(14) Low-low steam generator water level
(15) Turbine trip
(16) Safety injection system actuation
(17) Manual

Westinghouse has modified some of the reactor trip system input signals and logics from those provided on previous designs. Reactor trips on reactor coolant pump buses undervoltage and on open reactor coolant pump breakers have been deleted and reactor trip on reactor coolant pump buses underpower will be added. Four channels will be utilized to monitor pressurizer high water level as against three channels in earlier designs and the logic circuit for reactor trip on turbine stop valve closure will be changed to "three-out-of-four" logic from the "four-out-of-four" provided in the past designs.

The sensors for the reactor coolant pump bus underpower and underfrequency trips are not in the RESAR-41 scope and Westinghouse has not specifically stipulated that these reactor trip inputs should conform to all criteria applicable to the protection systems. We have advised Westinghouse that any input to the reactor trip system, including those which are outside the nuclear steam supply system scope, should not in any way result in the degradation of the overall reactor trip system. Therefore, we will require that the reactor coolant pump bus underpower and underfrequency trip inputs, including the sensors, be designed to satisfy all requirements of IEEE Std 279-1971 without exception. We will document the resolution of the interfaces associated with this item in a subsequent report.

We have reviewed the descriptive information for the reactor trip system, including functional logic diagrams, testing provisions, bypass features, the design criteria and design bases, and the analysis provided by Westinghouse on the adequacy of these criteria and bases. We have concluded that with the satisfactory resolution of the interface design item discussed above, the design of the reactor trip system will satisfy the Commission's requirements identified in Section 7.1 of this report, and will be acceptable.

7.3 Engineered Safety Features Systems

The engineered safety features systems will be initiated and controlled by the engineered safety features actuation systems. This system will consist of an analog portion consisting of three to four redundant channels per plant parameter monitored and a digital portion consisting of two redundant logic trains which will receive inputs from the analog protection channels. The redundant logic trains will provide the needed logic to actuate the three trains of engineered safety features.

The actuation system will initiate the following functions, along with indication of their respective inputs.

(1) Safety Injection Initiation.
Low Pressurizer Pressure or Containment Pressure High, or Low Compensated Steam Line Pressure or Low-Low Compensated T cold (cold leg temperature).

(2) Reactor Trip.
Safety Injection Initiation Signal (if a trip has not already been generated by the reactor trip system).

(3) Containment Isolation Phase A.
Safety Injection Initiation Signal.

(4) Steam Line Isolation.
Low Compensated Steam Line Pressure or Containment Pressure High-High or Low-Low Compensated T cold (cold leg temperature).

(5) Emergency Boration System.
Low Compensated Steam Line Pressure or Low-Low Compensated T cold or High Containment Pressure.

(6) Containment Spray Actuation.
Containment Pressure High-High-High.

(7) Recirculation Actuation.
Safety Injection Initiation Signal and Low Refueling Tank Water Level.

(8) Auxiliary Feedwater System.
Safety Injection Initiation Signal, or Low-Low Steam Generator Level.

The trip inputs for the safety injection, emergency boration, auxiliary feedwater and containment spray systems, represent modifications from those provided on previous Westinghouse designs.

We have reviewed the design description of the engineered safety features actuation system including functional logic diagrams, testing provisions, bypass features, design criteria and design bases and the analysis provided by Westinghouse on the adequacy of these criteria and bases. We have concluded that with the satisfactory resolution of the items discussed in Sections 7.3.1, 7.3.2, and 7.3.4 below, the instrumentation and controls associated with the engineered safety features systems and the engineered safety features actuation system will satisfy the Commission's requirements identified in Section 7.1 of this report and will be acceptable.

7.3.1 Emergency Core Cooling System

Westinghouse identified nine manually controlled motor operated valves in the emergency core cooling system which should not move from their aligned position during certain phases of the loss-of-coolant accident. The valves involved are the high head safety injection hot leg isolation valves, low head safety injection hot leg isolation valves, and the accumulator isolation valves. To meet our concern with regard to spurious movement of these valves during certain phases of the loss-of-coolant system, and in lieu of design changes, Westinghouse has elected to lock power out to these valves and has documented the following criteria for the modified design.

(1) Restoration of power to these valves should be readily available to an operator within the time period required in the emergency core cooling system analysis for the operation of these valves.

(2) Plant technical specifications require proper positioning of these 9 valves and locking out power to them prior to the reactor being brought critical.

(3) Redundant position indication will be provided for all 9 of these valves.

We have concerns about the interface information related to this design change which are under review. We will report the results of this review in a subsequent report.

7.3.2 Steam Line Break

The main steam system is outside the RESAR-41 scope. In the analysis of the rupture of the main steam line, Westinghouse has assumed that for any break in any location no more than one steam generator would blow down even if one of the main steam isolation valves fails to close. To validate the assumptions Westinghouse has identified (in Table 15.4-21 of RESAR-41) equipment and circuits required in the recovery from a high energy line rupture. Most of the required equipment and circuits are outside the RESAR-41 scope.

We have concerns about the interface information related to this postulated accident which are under review. We will report the results of this review in a subsequent report.

7.3.3 Changeover from Injection to Recirculation Mode

The original design proposed by Westinghouse for changeover of the emergency core cooling system from the injection mode to the recirculation mode was dependent on total operator action. It is our position that for standard plant designs the emergency core cooling system changeover functions should be made least dependent on operator action, since there was not adequate assurance that the operator will correctly perform the required

1547 273

safety function within the required time period. We required that the instrumentation and controls provided to accomplish the changeover to the recirculation mode be designed to meet the requirements of IEEE Std 279-1971.

Westinghouse subsequently modified the design to initiate the changeover function automatically when the refueling water storage tank level reaches a level less than a low level set point in conjunction with the initiation of the safety injection signal. In the proposed design, operator action will be required to close the refueling water storage tank suction valves to the safety injection pumps once the automatic changeover is completed. We were concerned about the consequences of the refueling water storage tank suction valves not being closed by the operator after the changeover is accomplished. Westinghouse has documented that their design criteria will assure that sufficient head be provided to close the refueling water storage tank suction check valves during the changeover and that laxity on the part of the operator to close the refueling water storage tank suction isolation valves after the changeover will not impair the functioning of the safety injection pumps.

Westinghouse has provided in Amendment 15 of RESAR-41, functional logic diagrams for the refueling water storage tank Lo-Lo-1 automatic actuation signal, automatic opening of sump valves, automatic closing of safety injection pumps mini flow valves and manual control for the tank suction valves.

From our review of the descriptive design information and the above referred logic diagrams, we have concluded that the modified design for the emergency core cooling system changeover from injection to recirculation mode satisfy the Commission's requirements identified in Section 7.1 of this report and is acceptable.

At the Final Design Approval stage we will review the details of the instrumentation and controls for the changeover functions to ensure their conformance to IEEE Std 279-1971 requirements.

7.3.4 Emergency Boration System

The emergency boration system is a safety system that will be located completely inside the containment. To maintain the fluid temperature in the system within the prescribed limits, Westinghouse has required one hundred percent redundant and separate heat tracing systems for all piping, valves and flanges in the system. The heat tracing is outside the scope of RESAR-41 and will be described in applications which reference RESAR-41. The power for the redundant heat tracing systems will be supplied from the redundant engineered safety features buses. Normally, only one of the heat tracing systems will be energized and the redundant system will be manually energized by the operator on low temperature annunciation in the plant control room.

Westinghouse has been advised of our concerns with regard to terminating redundant engineered safety features power sources at single components like a common pipe or valve, since such a design might result in the compromise of the physical and electrical independence required between the plant redundant engineered safety features power sources. In addition, it is not clear from the information provided how a single temperature monitoring system will provide reliable intelligence to the operator on emergency boration system fluid temperature and how such a design will meet the single failure criterion requirements for safety systems. We will require that Westinghouse provide adequate design criteria and information to show how the proposed temperature monitoring system for the emergency boration system will meet the single failure criteria requirements for safety systems and include adequate interface information for the heat tracing system.

We will report the resolution of this item in a subsequent report.

7.3.5 Periodic Testing of Protection System

Westinghouse has documented that periodic testing of the reactor trip system and engineered safety features actuation system are in conformance with the recommendations contained in Regulatory Guide 1.22. Westinghouse has also identified eight pieces of equipment that will not be tested during reactor operation and has stated the bases for their exclusion. The bases stated are in conformance with the recommendations included in Regulatory Guide 1.22. At the Final Design Approval review stage, we will review the periodic test circuitry details for these pieces of equipment to ensure that the test capabilities will include testing of the sensor signal, actuation logic and the final output signal.

Westinghouse has committed to provide test procedures at the Final Design Approval stage, for periodic response time testing of the reactor trip system and engineered safety features actuation system and their sensors, whose adequacy has not been previously demonstrated. The scope of the program will exclude nuclear instrumentation system detectors. We will review the adequacy of the test procedures for periodic response time testing in our final design review.

We have concluded that the criteria for the periodic testing of protection systems satisfy the Commission's requirements identified in Section 7.1 of this report and are acceptable.

7.4 Systems Required for Safe Shutdown

Westinghouse has identified the following principal systems as being required for safe shutdown: the boration system, the residual heat removal system and the auxiliary feedwater system. Also, Westinghouse has included a list of instrumentation and controls for systems in the RESAR-41 scope, in addition to other design features, that are to be provided by the balance of plant designer to achieve and maintain a safe shutdown condition in the event an evacuation of the control room is required.

We have reviewed the descriptive information relating to these systems including the interface design requirements for other systems to be presented in an application referencing RESAR-41 in order to assure that the operators will be able to achieve a safe shutdown condition of the plant from outside the main control room. The review included the functional logic diagram, interface requirements, design criteria, design bases and Westinghouse's analyses of the adequacy of these criteria and bases. We have concluded that with the satisfactory resolution of the items discussed in Sections 7.4.1 and 7.4.2

1547 275

below; the instrumentation and control design of systems required for safe shutdown conforms to the Commission's requirements identified in Section 7.1 of this report and will be acceptable.

7.4.1 Residual Heat Removal System

The residual heat removal system will be located completely inside the reactor containment. Three independent parallel flow trains will be utilized to remove residual heat from the core. Each suction line will have two motor operated isolation valves in series, with one valve located inside the missile barrier and the other outside the missile barrier. The design information originally provided in RESAR-41 did not establish if the residual heat removal system satisfied the requirements of Criterion 34 of the General Design Criteria in the event of a single electrical failure in the suction line motor operated valves. It is the staff's position that consistent with satisfying the requirements of the General Design Criteria, the design of the residual heat removal system motor operated isolation valves must meet the single failure criterion both in the residual heat removal function and while preventing overpressurization of the residual heat removal system.

In Amendment 12 of RESAR-41, Westinghouse proposed a modified design for the overpressurization protection of the residual heat removal system. In the proposed design, the redundant isolation valves will be separately interlocked with independent pressure signal to prevent their being opened when the reactor coolant system pressure is greater than 425 pounds per square inch, gauge. Each valve in the same train will be powered by a separate engineered safety features bus and have individual control circuitry. With this modified design, Westinghouse's analyses of failure in the valve electric power, control and interlock circuits conclude that any single failure, even coupled with any operator error in the control room can neither result in the opening of residual heat removal system isolation valves when the reactor coolant system pressure is above the interlock set point nor prevent at least one cooling train from being placed in service when required for cooldown.

With regard to the testability of these interlock signals, Westinghouse has stated that the pressure interlock signal and logic can be tested online up to the slave relay which provides the signal to the valve control circuit without adversely affecting safety.

We have concerns about the interface information related to this design which are under review and we will report the results of our evaluation of the modified design in a subsequent report.

In Amendment 17 of RESAR-41, Westinghouse has provided a composite functional interconnection diagram identifying the electrical power independence and the pressure interlock independence provided for the isolation valves in all three residual heat removal trains. However, Appendix 7A of RESAR-41 restricts the applicability of IEEE Std 279-1971 and IEEE Std 308-1971 to the system isolation valve interlocks only. The basis for excluding the rest of the system instrumentation, control, and electrical equipment from conforming to IEEE Std 279-1971 and IEEE Std 308-1971 is not obvious from the information provided in RESAR-41. We will require that the above stated criteria be applied to the entire residual heat removal system.

In addition, Section 5.4.3 of this report identifies system inadequacies that must be corrected before the system will be acceptable. We will address any instrumentation or electrical aspects associated with system modifications undertaken to resolve the problems discussed in Section 5.4.3 in a subsequent report.

7.4.2 Instrumentation for Safe Shutdown

We have concerns about the ability of a plant utilizing RESAR-41 to meet the requirements of Criterion 19 of the General Design Criteria and to exercise effective control of the shutdown systems from outside the control room. The interface information related to this item is presently under review.

We will report the results of our review in a subsequent report.

7.5 Safety-Related Display Instrumentation

The safety-related display instrumentation will provide the operator with information readouts to enable him to perform the required appropriate manual safety functions and for post-accident and incident surveillance. The scope of our review of the safety-related display instrumentation included the monitoring of the reactor trip system, engineered safety features and post-accident and incident information. Also, it should be noted that the design of the automatic bypass indication of a protective function at the system level is outside the design scope of RESAR-41. This part of the design will be evaluated during the review of applications referencing RESAR-41.

We have reviewed the design description, design criteria, and analyses of the manner in which the design of the safety-related display instrumentation will conform to the proposed design criteria. We have concluded that the design of the safety-related display instrumentation conforms to the Commission's requirements identified in Section 7.1 of this report and is acceptable.

7.6 Other Instrumentation Systems and Requirements Required for Safety

7.6.1 Environmental and Seismic Qualification

Westinghouse originally referenced a number of topical reports in RESAR-41 with regard to the environmental and seismic qualification of instrumentation, controls and electrical equipment important to safety. We found that a number of these referenced topical reports are unacceptable.

We have not found the methods and procedures to be used to implement the environmental qualification criteria to be acceptable. We require that, prior to issuance of a Preliminary Design Approval, Westinghouse commit to a satisfactory program for demonstrating the environmental qualification of instrumentation and electrical equipment important to safety within the near future.

1547 277

Some of the specific areas of interest related to environmental and seismic qualification are discussed below.

(1) Westinghouse has committed to qualify the instrumentation, controls and electrical equipment important to safety in the RESAR-41 scope, to the requirements of IEEE Std 323-1974, "IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations." Westinghouse has also stated that they will comply with the requirements of IEEE Std 382-1972, "Trial-Use Guide for Type Test of Class IE Electric Valve Operators for Nuclear Power Generating Stations" as modified by Regulatory Guide 1.73, with the exception that stem mounted switches will be qualified separately from the valve operator.

We have concluded that upon satisfactory development of a qualification program, the commitments made by Westinghouse to comply with the requirements of IEEE Std 323-1974 and IEEE Std 382-1972 as modified by Regulatory Guide 1.73 will provide an acceptable basis for the Preliminary Design Approval.

(2) To satisfy the requirements of Section 3.11.2 of the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants" Revision 1, dated October 1972, on safety system equipment qualification, Westinghouse has provided in Amendment 17 to RESAR-41 the normal and accidental environmental qualification conditions for safety equipment located both inside and outside containment. Westinghouse has also documented that the environmental conditions stated in RESAR-41 include margin factors in accordance with IEEE Std 323-1974. The qualification program of safety equipment in RESAR-41 scope to IEEE Std 323-1974 is currently under discussion with Westinghouse.

(3) Westinghouse has stated that the residual heat removal system pump motors and emergency boration system pump motors which are located inside the containment will be qualified to the requirements of IEEE Std 334-1971, "Trial-Use Guide for Type Tests for Continuous Duty Class I Motors Installed Inside the Containment of Nuclear Power Generating Stations," though these motors are not required for continuous duty functions. Amendment 16 of RESAR-41 has also documented that the qualification of these motors is consistent with the recommendations contained in Regulatory Guide 1.40.

We have concluded that upon satisfactory development of a qualification program, the commitments made by Westinghouse to comply with the requirements of IEEE Std 334-1971 as augmented by Regulatory Guide 1.40 with regard to the qualification of the residual heat removal system and emergency boration system pump motors will provide an acceptable basis for the Preliminary Design Approval.

1547 278

(4) Westinghouse had committed to qualify reactor protection system equipment to the requirements of IEEE Std 323-1974. However, it was not clear if this qualification program would encompass the engineered safety features actuation system equipment as well. In Amendment 16 of RESAR-41, Westinghouse clarified that this qualification program will also encompass this equipment.

We have concluded that upon satisfactory development of a qualification program, the commitments made by Westinghouse to comply with the requirements of IEEE Std 323-1974 with regard to the qualification of reactor trip system and engineered safety features system equipment provides an acceptable basis for the Preliminary Design Approval.

(5) RESAR-41 Section 15.4.2.1.1 on accident analysis originally identified overpower reactor trips as one of the functions which provided necessary protection against steam pipe rupture. Amendment 14 to this section of RESAR-41 specifically deleted the overpower trips but included reactor trip in addition to the safety injection signal as required functions to provide the necessary protection against steam pipe rupture. Reference to Section 7.2.1.1.2 of RESAR-41 indicates that overpower reactor trips are primary participants in the overall reactor trip system. Westinghouse has been informed that if credit is taken for the overpower reactor trips and thereby for the functional availability of the neutron detectors, we require that these neutron detectors be qualified for the worst case environment in the containment in accordance with the requirements of IEEE Std 323-1974.

(6) With regard to the seismic qualification of Category I instrumentation and electrical equipment in the RESAR-41 scope, Westinghouse has referred to a number of topical reports. A recent addition to the list of references is WCAP-8373, "Qualification of Westinghouse Seismic Testing Procedure for Electrical Equipment Tested Prior to May 1974", which is intended to evaluate the Westinghouse seismic test program against the staff's requirements on seismic qualification. From a generic review of the above referenced topical report, we have concluded that the report in its present form does not provide an acceptable basis for seismic testing of instruments, control devices and electric equipment to assure that these safety components will meet their performance requirements during and following a safe shutdown earthquake. In addition, the results of the testing program do not satisfy all of the requirements of IEEE Std 344-1971, "IEEE Guide for Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations." We are currently holding meetings with Westinghouse to resolve this issue.

We will address the resolution of the program for seismic and environmental qualification in a subsequent report.

7.6.2 Independence and Identification of Safety Related Equipment

We have reviewed the proposed design criteria for the separation of redundant safety related equipment and their physical identification as described in Sections 7.1.2.2 and

7.1.2.3 of RESAR-41, respectively. We have concluded that these criteria meet the requirements of IEEE Std 384-1974 as augmented by Regulatory Guide 1.75 and consider the proposed design acceptable.

7.6.3 Accumulator Isolation Valves

The proposed design of the control circuits for the accumulator isolation valves includes provisions to automatically open the isolation valves on the occurrence of a safety injection signal with the reactor coolant system pressure signal above the safety injection unblock pressure, and for redundant and independent indicating systems for each valve. In addition, the design has the capability to close the isolation valve with the reactor coolant system pressurized. Westinghouse has also documented that the interlocks will be testable and meet the appropriate qualification test standards for safety equipment.

To meet the single failure criterion for electrically operated valves, Westinghouse has elected to lockout power to the accumulator isolation valves when the reactor is at power. The RESAR-41 proposed technical specifications require that for the purpose of check valve leak testing, one accumulator at a time may be isolated provided the reactor is in the hot shutdown condition. For the Final Design Application, we will require that the technical specification limit the time that an accumulator isolation valve may be isolated for no more than 8 hours.

We have concluded that the proposed design of the control circuits for the accumulator isolation valves satisfies the Commission's requirements identified in Section 7.1 of this report and is acceptable for the Preliminary Design Approval.

7.7 Control Systems Not Required for Safety

The following control systems which are not required for safety are identified in RESAR-41; reactor control, rod control, plant control system interlocks, pressurizer pressure control, pressurizer water level control, steam generator water level control, steam dump control and in-core instrumentation. Westinghouse has documented no major differences in the instrumentation and controls for the above systems and those provided in their previous designs.

With regard to an accidental withdrawal of a single rod control cluster assembly, Westinghouse has provided a brief discussion to establish that no single electrical failure could cause the accidental withdrawal of a single rod cluster control assembly. In addition, Westinghouse has concluded that more than two simultaneous component failures will be required (other than the open wire failures) to allow withdrawal of a single rod and the probability of such an occurrence is too low to have any significant consideration. During the review of the Final Design Approval application, we will review the details of the rod control system circuitries to assure that the design criteria, design bases and the failure mode and effects analysis for the rod control system have been implemented in accordance with the Commission's requirements.

We have concluded that failures in these control systems will not be expected to degrade the capabilities of the plant safety systems to any significant degree or lead to plant

1547 280

conditions more severe than those for which the safety systems are designed to protect against and that these control and instrumentation systems satisfy our requirements and are acceptable.

7.8 Anticipated Transients Without Scram

To meet the regulatory positions contained in WASH-1270, "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors" dated September 1973, Westinghouse has submitted topical report WCAP-8440 titled, "Anticipated Transient Without Trip Analysis for a Four Loop (3814 MWt) Westinghouse PWR." However, it should be emphasized that our evaluation of the adequacy of the instrumentation, controls and electrical equipment has been made completely on the information contained in RESAR-41. We will report the results of our evaluation of the above referenced topical report on this subject and the conclusions with respect to any modifications required in the instrumentation, controls and electrical equipment included in the present design in a subsequent report.

1547 281

8. ELECTRIC POWER SYSTEMS

Except for the vital instrument alternating current power supply, the design of the offsite and onsite power systems is totally outside the RESAR-41 scope and will be presented in applications referencing RESAR-41. However, Westinghouse has included some design information on the electrical loads for the RESAR-41 nuclear steam supply system that will need to be powered from the plant's power systems. Westinghouse has also stipulated that engineered safety features loads should normally be fed directly from the offsite power system, thus eliminating the dependency of these loads for power on the plant turbine-generator unit availability and preventing interruption of power to these loads on a turbine-generator trip. The alternating current onsite power interface requirements specify three redundant and independent standby power supplies and the direct current onsite power interface requirements specify four independent batteries and battery chargers. This is in conformance with the required redundancy of safety related systems and components included in RESAR-41 design.

The vital instrument alternating current power supply described in RESAR-41 includes as standard equipment four inverters and, as optional equipment the distribution bus panels including feeder breakers.

We have concerns about the interface information for electric power systems which are under review. We will report the results of this review in a subsequent report.

We have concluded from the information provided in Section 8.0 of RESAR-41, that the design requirements stated will provide an acceptable basis for developing a design for the electric power systems on any plant referencing RESAR-41, with the satisfactory completion of our review of the associated interfaces.

1547 282

9.0 AUXILIARY SYSTEMS

The proposed auxiliary systems are described in Section 9.0 of RESAR-41 and consist of the systems necessary to assure safe handling of fuel including the new and spent fuel storage racks and the fuel handling system, and the chemical and volume control system and the boron recycle system. We have reviewed these systems to determine their conformance to the applicable requirements of the General Design Criteria and Regulatory Guides. The auxiliary system designs in RESAR-41 are discussed in the following paragraphs, and include the safety related objectives of the system and the manner in which these objectives will be achieved.

9.1 Fuel Storage and Handling

9.1.1 New Fuel Storage

New fuel will be stored in racks composed of individual vertical stainless steel cells which can be fastened together in any number to form a module that can be bolted to anchors in the floor of the new fuel storage building. The new fuel storage rack design includes dry storage for one-third of a core at a center-to-center spacing of 21 inches. This spacing is sufficient to maintain the effective multiplication factor (K_{eff}) equal to or less than 0.95 even in the event the storage area were flooded with unborated water. These racks will be designed to seismic Category I requirements. The description of the new fuel storage area will be supplied in applications referencing RESAR-41. This area must provide protection for the racks from dropped objects.

We have evaluated the proposed design of the new fuel storage racks and conclude that they meet the applicable positions set forth in Regulatory Guide 1.13, and the requirements of Criterion 62 of the General Design Criteria. We conclude that the proposed design of the new fuel racks is acceptable.

9.1.2 Spent Fuel Storage

Spent fuel will be stored in racks composed of individual vertical cells which can be fastened together in any number to form a single module that can be anchored to the floor of the spent fuel pit. These spent fuel storage racks will have a capacity of storing fuel assemblies for one and one-third cores for a single reactor unit, at a center to center spacing of 21 inches. The design of the spent fuel storage rack assembly is such that it will be impossible to insert the spent fuel assemblies other than at prescribed locations. This spacing is sufficient to maintain the effective multiplication factor (K_{eff}) equal to or less than 0.95 even if unborated water is used to fill the spent fuel storage pool. The spent fuel racks will be designed to seismic Category I requirements.

The design of the spent fuel storage pool will be described in applications which reference RESAR-41.

We have evaluated the proposed design of the spent fuel storage racks and conclude that they meet the application positions set forth in Regulatory Guide 1.13 and the requirements of

Criterion 62 of the General Design Criteria. We conclude that the proposed design of the spent fuel storage racks is acceptable.

9.1.3 Fuel Handling System

Our evaluation of the fuel handling system is provided in Section 5.4.8 of this report.

9.2 Process Auxiliaries

9.2.1 Chemical and Volume Control System

The chemical and volume control system will be designed to control and maintain the reactor coolant inventory and also control the boron concentration in the reactor coolant. Purification of reactor coolant will also be accomplished by removal of corrosion and fission products from the letdown fluid and reactor coolant chemistry will be controlled through the process of chemical addition. The system will also maintain seal-water injection flow to the reactor coolant pumps and provide a means of filling, draining and pressure testing of the reactor coolant system.

In the designs proposed for RESAR-41, the chemical and volume control system will be interconnected with the residual heat removal system. Two of the three residual heat removal trains will be crossconnected to the letdown line. A low head, 450 gallons per minute reactor coolant purification pump will be provided in the letdown line. By crossconnecting this pump with the residual heat removal system during plant cooldown and cold shutdown, a high cleanup flow rate will be maintained through the mixed bed demineralizer, the cation bed demineralizer, and both reactor coolant filters. The increased cleanup flow allowed by this design will reduce the required coolant time.

Reactor coolant boron concentration will be controlled using the chemical and volume control system by one of two basic methods, (1) by adding makeup for either boration or dilution for the large reactivity changes needed during shutdown and startup, or (2) by the thermal regeneration process to compensate for the reactivity changes due to Xenon transients. The boron concentration of the reactor coolant can be continuously monitored by a boron concentration measurement system. This system measures the boron concentration of the letdown flow in the chemical and volume control system.

The thermal regeneration subsystem will control the boron concentration of reactor coolant letdown flow by varying the temperature of inline boric acid demineralizers. In this way, boric acid can be added to or removed from the reactor coolant without dilution flow. When necessary, makeup boration and dilution will be accomplished by adding either borated or pure water to the system. The use of this system will greatly reduce the volume of waste reactor coolant that must be processed by the waste processing system.

The proposed design will utilize two centrifugal and one positive displacement charging pumps. These pumps will not serve any dual functions such as for safety injection or emergency boration.

As part of the chemical and volume control system, Westinghouse includes in RESAR-41 as an option, a failed fuel detection system. This system consists of equipment designed to indicate gross fuel failure by monitoring the delayed neutron activity in the reactor coolant.

The design of the major portions of the chemical and volume control system is described in RESAR-41. Applicants referencing RESAR-41 must design the piping layout and several of the tanks in the system. In addition, adequate component cooling water at 105 degrees Fahrenheit or less must be provided to system heat exchangers and charging pumps. Provisions must also be made for maintaining a temperature of at least 65 degrees Fahrenheit or greater for all portions of the system which will normally contain a 4 percent or greater boric acid solution.

We have reviewed the design bases, system functions, components and their classifications, system operation, safety evaluation and other data included in RESAR-41 and conclude that the proposed system design is acceptable.

9.2.2 Boron Recycle System

The boron recycle system will be designed to receive and recycle reactor coolant effluent for reuse of the boric acid and makeup water. It will decontaminate the effluent by means of demineralization and gas stripping, and will use evaporation to separate and recover the boric acid and makeup water. The boron recycle system will be capable of processing the total volume of water collected during a core cycle as well as short term surges.

The basic system design and the design for many of the components is described in RESAR-41. However, applicants referencing RESAR-41 must design the piping layout and provide designs for heat tracing and certain other equipment.

The boron recycle system will be used intermittently throughout normal reactor operation and will not be required for safe plant operation or shutdown. We have reviewed Westinghouse's proposed design bases and system description for the boron recycle system and conclude that this system design is acceptable.

1547 285

10.0 STEAM AND POWER CONVERSION SYSTEM

The steam and power conversion system will convert the thermal output of the nuclear steam supply system to steam to drive the turbine-generator. This system will be designed by the balance of plant designer. The RESAR-41 design does not extend beyond the steam generator's feedwater and steam nozzles.

1547 286

11.0 RADIOACTIVE WASTE MANAGEMENT

11.1 Source Terms

Radioactive materials in liquid effluents may be released to the environment by a nuclear plant utilizing a pressurized water reactor from the liquid waste processing system, the boron recycle system, the steam generator blowdown system and the turbine building floor drain system. Of these, only the boron recycle system is within the standard scope of RESAR-41, as defined in Amendment 1 to WASH-1341. However, RESAR-41 does include concentrations of radioactive materials and flow rates in streams that (1) are input to the radioactive waste management systems, and (2) are used as the design basis for shielding and building ventilation systems for applications referencing RESAR-41.

We have reviewed the mathematical models and the parameters used to calculate primary coolant concentrations, and the input rates to the radioactive waste management systems from the components within the nuclear steam supply system. We consider that, with the exception of the fraction of the fuel that is assumed to be releasing fission products to the primary coolant, the parameters and calculations are consistent with those given in WASH-1258 and Regulatory Guide 1.42. Westinghouse has submitted operating data in topical report WCAP-8253, "Source Term Data for Westinghouse Pressurized Water Reactors" to justify a value of 0.0005 for this parameter for use in the evaluation of expected effluent releases. We have reviewed these data and have determined that a value of 0.0012 for the fraction of the fuel that is assumed to be releasing fission products is justified for use in our source term calculations. Westinghouse has agreed to this value and in Amendment 7 to RESAR-41 provided revised primary coolant concentrations based on this value that we consider acceptable for use in applications referencing RESAR-41.

11.2 Liquid Waste Systems

The boron recycle system described in Section 9.2.2 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. Westinghouse 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 wastes 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.

1547 287

The principal components that will make up the boron recycle system, along with their principal design criteria, are listed in Table 11-1 below.

TABLE 11-1

DESIGN PARAMETERS OF PRINCIPAL COMPONENTS OF BORON RECYCLE SYSTEM

Component	Number	Capacity (gallons per minute)	Quality Group
Evaporator Feed Demineralizer	2	250	C
Evaporator Condensate Demineralizer	1	120	D
Evaporator Package Polishing	1	15	C

The design capacity of the boron recycle evaporator will be 21,000 gallons per day, whereas the expected input rate will be 1800 gallons per day. The difference between the expected flows and design capacity provides adequate reserve for processing surge flows. The system design will allow wastes to be processed interchangeably between the boron recycle system and the liquid waste evaporators in the event of equipment downtime. We have concluded that the system design and capacity will be adequate for meeting the demands of the facility during anticipated operational occurrences.

1547 288

12.0 RADIATION PROTECTION

12.1 Assuring That Occupational Radiation Exposures Will Be As Low As Practicable

We reviewed the policy considerations, design considerations and operational considerations related to assuring that occupational radiation exposures will be as low as practicable for the RESAR-41 design. This included whether there is a management policy with respect to system and equipment design for as low as practicable radiation exposures. It included descriptions of how experience from past designs and operating plants may be used to develop improved radiation protection designs for the nuclear steam supply system. It included whether there is implementation of the appropriate guidance provided in Regulatory Guide 8.8, or information on proposed alternatives.

In Appendix 12.1-A of RESAR-41 Westinghouse provides radiation protection design considerations that are related to RESAR-41 equipment, and design recommendations for the balance-of-plant designer related to shielding, installation, and layout of the RESAR-41 equipment. These design considerations cover the following systems in various degrees of detail; reactor, evaporators, tanks and heat exchangers, valves, remote and/or automatic systems control operations, reactor coolant system, spent fuel pit cooling system, waste processing system, residual heat removal system, chemical and volume control system, boron recycle system, and safety injection systems.

We reviewed the RESAR-41 material for evidence that the design will be in accordance with Regulatory Guide 8.8, including incorporation of measures for reducing radiation levels and time spent where maintenance and other operations are required, reviews of specific equipment designs by competent radiation protection personnel, and instructions to designers and engineers regarding design considerations for as low as practicable radiation exposures. We also reviewed the application for evidence that Westinghouse has incorporated previously tested good design features and has used operating experience to improve on the design of the plant with regard to assuring that occupational radiation exposures will as low as practicable.

We determined that Westinghouse has shown sufficient concern and familiarity with the as low as practicable principles that we find this section acceptable in the area of design considerations.

Appendix 12.1-A of RESAR-41 includes many of the design guidance items of Regulatory Guide 8.8; however, many of these are included only as recommendations. It will therefore be necessary for applicants referencing RESAR-41 to submit a complete Section 12.0. They must show how adequate radiation protection will be provided from the RESAR-41 specified radiation sources.

1547 289

12.2 Radiation Sources

We reviewed RESAR-41 to evaluate information on radiation sources, as they relate to in-plant radiation protection. This includes the description of the sources of radiation that will be the basis for the radiation protection program, and needed for the shield design calculations by the balance of plant designer.

Our acceptance criteria require that all sources of radiation be described in the manner and to the degree needed for shielding codes used in the design process, for plans and procedures development, for assessment of occupational radiation exposure, and for equipment specification. The sources of radiation of interest are those that will necessitate shielding, special ventilation designs, traffic or access control considerations, special plans and procedures, monitoring equipment, etc.

Regulatory Guide 1.70 requires that airborne sources that can be created by leakage or release from a nuclear steam supply system, such as that described in RESAR-41, by opening normally closed containers such as tanks, pump casings or vent spaces, and the pressure vessel, be identified by location and magnitude, in a manner useful for designing appropriate ventilation systems and in specifying appropriate monitoring systems. The assumptions made in arriving at quantitative values for these various sources should be specified.

The RESAR-41 section on source terms indicates that Westinghouse provides four separate categories of neutron and gamma ray information regarding the reactor radiation source at power. In addition, this section provides radiation sources related to various systems proposed to be supplied by Westinghouse. The radioactive source terms and leakage rates necessary to complete the analysis of onsite exposure due to airborne radioactive material are given in Section 11 of RESAR-41.

Our review of the source term section examined the source term tables and the conditions given for definition of the source terms. These descriptions meet our acceptance criteria as being sufficient and appropriate for input to shielding calculations. In the cases where the total quantities of a particular source have not been provided by Westinghouse because of the limited part of the system design within its scope, the applicant referencing RESAR-41 will have to obtain and provide the added information.

12.3 Radiation Protection Design Features

The following areas of RESAR-41 relating to radiation protection design features were reviewed.

(1) The description of equipment design features to be used for assuring that occupational radiation exposures will be as low as practicable.

(2) Information concerning implementation of Regulatory Guide 8.8, Section C.3 or proposed alternatives.

1547 290

(3) The description of any special protection features that use shielding, geometric arrangment, or remote handling to assure that occupational radiation exposure will be as low as practicable.

(4) Information concerning the implementation of Regulatory Guides 1.21, 1.52, 1.69, 8.2, 8.6 and American National Standards Institute Standard N13.1-1969, or alternatives, if proposed.

RESAR-41 provided information on shielding design objectives, and in Appendix 12.1-A, describes design considerations and features which Westinghouse "recommends" regarding equipment and related systems within the scope of RESAR-41. Also described are design features of the RESAR-41, system relating to radiation protection.

Westinghouse has provided an evaluation of the reduction in radiation exposure to plant personnel that is expected from the design changes for the rapid refueling concept. Their evaluation shows a significant dose reduction as compared to refueling operations on previous Westinghouse designs. The assumptions used in arriving at these exposure values are reasonable and acceptable. Actual values will not be available until operating experience is obtained on RESAR-41 plants.

We reviewed the RESAR-41 material for evidence that Westinghouse has applied the guidance of Regulatory Guide 8.8, or that suitable alternatives have been proposed. This includes evidence that major exposure accumulating functions (maintenance, refueling, radioactive material handling, processing, inservice inspection and calibration, etc.) have been considered in equipment design and that potential radiation exposure from these activities will be kept as low as practicable by radiation protection features incorporated in the design. Acceptability of the shielding is based on factors which have not been supplied in RESAR-41, including shielding computational methods.

The RESAR-41 supplied information on equipment design features for assuring that occupational radiation exposures will be ALAP meets the guidance of Regulatory Guide 8.8 and the staff requirements. The development of a refueling concept that will significantly reduce associated occupational exposures is to be particularly noted. We determined that Westinghouse has shown sufficient concern and familiarity with the as low as practicable principals that we find this section acceptable in the area of equipment design features.

The advice and guidance provided by Westinghouse in RESAR-41 relating to ventilation systems has been reviewed against our acceptance criteria, as provided in Regulatory Guide 8.8 and in staff positions. Our review of applications referencing RESAR-41 will include a determination of whether the appropriate guidance has been applied to the final design of the plant. The Westinghouse guidance is an acceptable basis for design of these systems.

1547 291

13.0 CONDUCT OF OPERATIONS

Information relating to the conduct of operations will be provided in applications which reference RESAR-41.

We have reviewed the information in RESAR-41 related to industrial security. We found that the design of the nuclear steam supply system to meet the rigorous safety requirements set by the Commission will enhance industrial security and reduce the vulnerability of the RESAR-41 section of a plant to acts of sabotage. This will be accomplished primarily through the use and separation of redundant systems and components. Further, the location of some of the safety systems within containment, which were located external to containment in previous Westinghouse designs, will provide additional barrier protection and access control to reduce their vulnerability to sabotage.

We conclude that Westinghouse's design for the protection of the plant against acts of industrial sabotage are acceptable for the Preliminary Design Approval stage of the review process.

1547 292

14.0 INITIAL TESTS AND OPERATIONS

We have reviewed Westinghouse's test program for the RESAR-41 nuclear steam supply system as described in Section 14.1 of RESAR-41.

Westinghouse has described a proposed test program divided into two major phases; preoperational testing and startup testing. The preoperational test phase will be subdivided into individual system and/or subsystem preoperational tests, and integrated reactor coolant system heatup and pre-core loading hot functional tests. The startup test phase will be subdivided into initial core loading, postcore loading hot functional tests, initial criticality, low power physics tests, and power ascension tests.

The proposed test program includes a summary description of the test objectives, prerequisites, and interfaces for each system and/or component test as it relates to the nuclear steam supply system and to auxiliary systems that will be furnished by the balance of plant designer. The preliminary design of the facility will permit testing in accordance with the guidance and staff positions set forth in Regulatory Guides 1.41, 1.68, 1.79, and 1.80 which are the current guides that have applicability in initial test programs.

On the basis of our review, we conclude that an acceptable startup and test program can be conducted without the need for design modifications. The staff will perform a detailed review of the initial test program when the final design application is submitted.

1547 293

15.0 ACCIDENT ANALYSES

15.1 Summary

Westinghouse has performed safety analyses to evaluate the capability of the RESAR-41 nuclear steam supply system to withstand abnormal operational transients and a broad spectrum of postulated accidents without undue risk to the health and safety of the public. The events considered include all relevant types discussed in the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 1, dated October 1972. The postulated events have been classified by Westinghouse with respect to evaluation criteria as follows:

(1) Condition I-Normal Operation and Operational Transients
(2) Condition II-Faults of Moderate Frequency
(3) Condition III-Infrequent Faults
(4) Condition IV-Limiting Faults

Condition I events are those which may occur in the course of normal power operation, refueling maintenance or maneuvering of the plant. Condition I occurrences will be accommodated by sufficient design margin between any plant parameter and the value of that parameter which would require actuation of the reactor protection system. Condition I events will be handled by the reactor control systems which will automatically maintain prescribed conditions in the plant even under the most conservative set of reactivity parameters with respect to both systems stability and transient performance.

Condition II events at worst will result in a reactor trip with the plant being capable of return to operation. Condition II events will not propagate to cause a more serious Condition III or IV event and are not expected to result in fuel rod failure or reactor coolant system overpressurization.

Condition III events are very infrequent faults which will be accommodated with the failure of only a small fraction of the fuel rods although sufficient fuel damage might occur to preclude immediate resumption of operation. A Condition III event will not generate a Conditon IV fault, or result in loss of function of the reactor coolant system or containment barriers.

Condition IV events are limiting design bases which are not expected to occur, but are postulated because their consequences include a potential for the release of significant amounts of radioactive material. System design for Condition IV events will prevent a fission product release to the environment which would result in an undue risk to the health and safety of the public in excess of limits established in 10 CFR Part 100. A Condition IV event is not to cause a consequential loss of required function of systems needed to mitigate the consequences of the accident, such as the emergency core cooling system and the containment.

1547 294

Westinghouse's classification of events analyzed is itemized in Table 15-1 of this report.

TABLE 15-1

CATEGORIES OF TYPICAL TRANSIENTS AND FAULTS

Condition I

- Reactor startup
- Reactor shutdown
- Refueling operations

Condition II

- Uncontrolled control rod assembly bank withdrawal while the reactor is subcritical or at power.
- Partial loss of forced reactor coolant flow
- Startup of an inactive reactor coolant loop
- Turbine trip
- Loss of normal feedwater
- Loss of offsite power
- Uncontrolled boron dilution
- Control rod assembly misalignment
- Excessive load increase

Condition III

- Improper loading of a fuel assembly
- Complete loss of forced reactor coolant flow
- Minor secondary system pipe break
- Single control rod assembly withdrawal at full power
- Waste gas decay tank rupture
- Loss of reactor coolant from small break

Condition IV

- Control rod ejection
- Fuel handling accident
- Steam generator tube rupture
- Major secondary system pipe rupture
- Reactor coolant system rupture
- Single reactor coolant pump locked rotor

1547 295

15.2 Input Parameters and Analytical Techniques For Accident and Transient Analyses

15.2.1 Input Parameters

We reviewed the assumptions and input parameters employed by Westinghouse in the accident and transient analyses.

The departure from nucleate boiling calculations were performed using a critical heat flux multiplier of 0.90, thus providing a 10 percent design margin. Final judgment of the adequacy of this multiplier will be made after completion of the test programs discussed in Section 1.4 of this report.

Mathematical models and methods used by Westinghouse have been previously reviewed and found acceptable by the staff unless otherwise noted in this report.

Reactor protection system trip set points and the assumed trip delay times used in the analyses are tabulated in Table 15-2 of this report. These values are suitable provided that they remain conservative with respect to the set points finally implemented, fully accounting for all sensor and process delays and uncertainties.

The rod insertion time used, 2.4 seconds to reach 85 percent of the rod travel, was based on previous measurements applicable to the 12 foot 17x17 rod cluster control assemblies and included allowance for the extra travel in the 14 foot core which will be verified during testing of a prototype assembly. Instrument errors and time delays assumed for the analyses will be justified as part of the final design review of RESAR-41.

Events initiated at full power were assumed to start at a core thermal power level of 3876 megawatts, which is 1.02 times the proposed license power level to account for power measurement uncertainty in accordance with Regulatory Guide 1.49. However, Regulatory Guide 1.49 states that the possible offsite radiological consequences of postulated design basis accidents made to demonstrate acceptability of the site in accordance with 10 CFR Part 100 may be made at a higher core thermal power level not to exceed 4100 megawatts. A value of 4100 megawatts was used by Westinghouse as the initial core full power condition for analyses of the loss of normal feedwater, loss of reactor coolant from small breaks, waste gas decay tank rupture, loss-of-coolant, steam generator tube rupture, and fuel handling accidents. The small break and loss-of-coolant accident analyses will be resubmitted in response to Appendix K to 10 CFR 50 with an assumed initial thermal power level of 3876 megawatts. Although the loss of normal feedwater analysis does not involve radiological consequences, we conclude that analysis at 4100 megawatts is a conservative evaluation of system design adequacy for 3876 megawatts operation for the preliminary design review. Westinghouse has committed to perform all analyses in accordance with the guidance of Regulatory Guide 1.49 for the final design application.

1547 296

TABLE 15-2

TRIP POINTS AND TIME DELAYS TO TRIP ASSUMED IN ACCIDENT ANALYSES

Trip Function	Limiting Trip Point Assumed In Analyses	Time Delay (Seconds)
Power Range High Neutron Flux, High Setting	118 percent	0.5
Power Range High Neutron Flux, Low Setting	35 percent	0.5
Overtemperature ΔT	Variable (see Figure 15.1-1 of RESAR-41)	6.0[1]
Overpower ΔT	Variable (see Figure 15.1-1 of RESAR-41)	6.0[1]
High Pressurizer Pressure	2410 pounds per square inch, guage	2.0
Low Pressurizer Pressure	1860 pounds per square inch, guage	2.0
Low Reactor Coolant Flow (from loop flow detectors)	87 percent loop flow	1.0
Reactor Coolant Pump Underpower Trip	70 percent	1.2
Turbine Trip	Not applicable	1.0
Low-Low Steam Generator Level	Zero percent of narrow range level span	2.0
High Steam Generator Level Trip of the Feedwater Pumps, Closure of Feedwater System Valves, and Turbine Trip	75 percent of narrow range level span	2.0
Reactor Coolant Bus Underfrequency Trip[2]	58 Hertz	0.1

[1]Total time delay (including resistance temperature detector (RTD) bypass loop fluid transport delay, effect of bypass loop piping thermal capacity, RTD time response, and trip circuit channel electronics delay) from the time the temperature difference in the coolant loop exceeds the trip setpoint until the rods are free to fall

[2]Used for drop in line frequency combined with a loss of flow transient

1547 297

Core physics parameters used in the accident analyses have been reviewed and found to be suitably conservative. They were chosen to represent the most adverse conditions of core life for the event considered, with respect to reactivity coefficient, control rod worths, and local power peaking factors. Reload cores or operating configurations other than those considered must be reexamined to ascertain that they cannot result in more severe transients than have been considered.

15.2.2 Analytical Techniques

We have reviewed and approved the analytical techniques used by Westinghouse in the RESAR-41 accident and transient analyses except as noted below.

The following is a list of the computer codes which have been used in the accident and transient analyses that are under review by the staff.

Code Name	Topical Report
BLKOUT	WCAP-7898 Long Term Transient Analysis Program for Pressurized Water Reactors
LOFTRAN	WCAP-7907 LOFTRAN Code Description
FACTRAN	WCAP-7908 A FRACTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod
MARVEL	WCAP-7909 A Digital Computer Code for Transient Analysis of a Multi-Loop PWR System
THINC-IV	WCAP-7956 THINC-IV-An improved program for Thermal and Hydraulic Analysis of Rod Bundle Cores
PHOENIX	WCAP-7973 Calculation of Flow Coastdown After loss of Reactor Coolant Pump
WIT-6	WCAP-7980 WIT-6-Reactor Transient Analysis Computer Program Description.

As discussed in the remainder of Section 15 of this report, the margins predicted by these methods for the postulated accidents and transients are acceptable for the Preliminary Design Approval. We will complete our review of these codes prior to Final Design Approval of RESAR-41.

15.3 Technical Specification Limits Qualified by Accident and Transient Analyses

Results of the postulated accidents investigated are sensitive to the value of many operating parameters which define conditions at the start of the transient and govern the response of the system model to the postulated accident condition. Our review and approval of these analyses constitutes approval of the operating conditions and plant characteristics which have been found within the range that has been justified by the analyses. As a result, technical specifications must assure that operating conditions and trip setpoints are such that there is no potential for transients of more severe consequences than those predicted by the reviewed conditions.

Westinghouse has proposed limits on control rod operations and core power distribution which are consistent with limiting operating conditions qualified by the accident analyses. The proposed power distribution limits will not result in a peak linear power density in excess of 13.60 kilowatts per foot, which is the value qualified by the accident analyses. The limits are to be enforced by operating procedures and technical specification limitations on power distribution using constant axial offset procedures to assure that engineering heat flux and nuclear enthalpy rise hot channel factors do not exceed design limits. Additional procedures will require confirmation of power distribution using a movable in-core detector system at each fuel loading and periodically during power operation. A limit on core radial power asymmetry (power tilt) will be monitored and alarmed using the ex-core detector system. Axial power distribution will be controlled by control bank position and monitoring of flux difference between the top and bottom ex-core detectors.

The overtemperature ΔT trip (temperature difference) will provide protection against departure from nucleate boiling for all combinations of pressure, power, coolant temperature, and axial power distribution which are within the operating range between high and low pressure reactor trips, provided that the transient is slow with respect to piping coolant transit delays from the core to the temperature detectors (about 4 seconds) and axial peaks are below design values. The flux difference measurement will be incorporated in analog circuitry which will automatically reduce the overtemperature ΔT trip setpoint whenever flux difference limits are exceeded. Alarms on flux difference and radial power tilt will be derived from the plant process computer. The technical specifications include average temperature versus power safety limit curves with pressure as a parameter to define the trip limit with all loops operating and with one coolant loop out of service. The curves define the loci of points for which the departure from nucleate boiling ratio is greater than 1.3.

Protection against departure from nucleate boiling during loss of forced reactor coolant flow transients will be provided by the reactor coolant pump bus underpower trip or the low reactor coolant loop flow trip (87 percent of loop flow). However, the analyses submitted by Westinghouse qualify this protection only for assumed initial operating conditions within the nominal operating range of reactor pressure, steady state power level, and coolant temperature and flow conditions. Westinghouse has not proposed limits on core operating conditions which would assure that initial core coolant flow and temperature are within the range evaluated in the accident analyses. Accordingly, based on the data provided in RESAR-41, we will include the following additional core operating limits in the technical specifications.

Parameter	Limit Value*
(1) Reactor Vessel Minimum Coolant Flow	(a) 144.7 million pounds per hour* at license power level (3800 megawatts maximum)
	(b) 107 million pounds per hour* with 3 loop operation at maximum core power level of 2660 megawatts

*The limit value is the value used in the safety evaluation; technical specifications must assure that measured values are less than the tabulated value by sufficient margin to account for uncertainties.

1547 299

(2) Core Coolant Average Temperature	597.2 degrees Fahrenheit maximum
(3) Pressure in the Pressurizer	2250 $\pm$ 30 pounds per square inch, absolute

We conclude that the technical specifications proposed by Westinghouse plus the additions indicated above will be adequate to maintain core operating conditions within the limits qualified by the accident analyses, provided that final limit values and core monitoring procedures account for measurement uncertainties and power distribution uncertainties.

The latter consideration should conservatively compensate for the absence of fixed in-core instrumentation which would permit continuous monitoring of core power distribution.

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 Condition II in the evaluation and classification presented by Westinghouse.

We have compared the Condition II events of Table 15-1 to typical anticipated events normally considered for safety reviews. The event, "Complete Loss of Coolant Flow" is classified as a Condition III fault by Westinghouse but is considered as an anticipated transient by the staff and was evaluated as an anticipated transient.

We have reviewed the analyses submitted for anticipated transients to ascertain that the transients will not violate the specific criteria which follow:

(1) Pressure in the reactor coolant and main steam systems should not exceed 110 percent of design pressure (Section III of American Society of Mechanical Engineers 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. 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.

We conclude that the most limiting analysis in regard to core thermal margins is that for the uncontrolled control rod assembly bank withdrawal with the reactor at full power. For this transient the calculated minimum value of the departure from nucleate boiling ratio was approximately 1.35, which is within the limit value we find acceptable as evidence that clad integrit, will not be jeopardized.

1547 300

The most limiting transients with respect to pressure within the reactor coolant system is the loss of external electrical load transient and/or turbine trip from maximum power conditions (102 percent of power). The calculated peak primary system pressure of 2550 pounds per square inch, absolute did not result in violation of the 110 percent overpressure limit.

Various chemical and volume control system malfunctions which could lead to an unplanned boron dilution incident have been reviewed. The ones that will allow the operator the shortest time for corrective action have been analyzed starting from plant conditions of startup, power operation (automatic and manual), hot standby, cold shutdown, and refueling. The results of the analyses of these events show that the operator will have 32 minutes to take corrective action if a boron dilution incident occurs during refueling or startup. However, this time is based on the dilution flow rate not exceeding 300 gallons per minute. The basis for Westinghouse not assuming a greater dilution flow is that there will be two flow alarms set at 280 gallons per minute in series in both the dilution paths with one of them being common to both paths. The fact that the operator would have to take several independent actions in order for dilution to occur is also considered by Westinghouse.

We conclude that 32 minutes for operator action during refueling is adequate. However, in order to take credit for the assumed maximum dilution flow rate of 300 gallons per minute, we require that the three flow alarms be designed to Class IE requirements to meet the single failure criteria.

Therefore, we require that the dilution flow rate alarms be designed to Class IE requirements or that other suitable means be provided to assure that the operator will have at least 30 minutes for corrective action during a postulated boron dilution accident during refueling or startup. We will report resolution of this issue in a supplement to this report.

For power operation in the manual control mode, the fuel will be maintained within thermal limits by the overtemperature ΔT trip (temperature difference). In the manual or automatic control mode, the operator will have more than 30 minutes after receipt of the first alarm to take corrective action. We require that a minimum time of 30 minutes must be available to the operator for corrective action during power operation. Therefore, we find the consequence of a boron dilution accident acceptable for power operation.

Rod cluster control assembly (control rod) misalignment accidents including a dropped full-length control rod, dropped full-length control rod bank, and a misaligned full or part-length control rod have been analyzed by Westinghouse.

The analyses were performed using the TURTLE[1] code to determine X-Y peaking factors. We have reviewed this code and find it acceptable for reference in RESAR-41. The THINC-IV[2] code was then used to calculate the departure from nucleate boiling ratio. For the transient response to a dropped control rod or control rod bank, the LOFTRAN[3] code was used.

---

1 S. Altomare and R. F. Barry, "The TURTLE 24.0 Diffusion Depletion Code," WCAP-7758, June 1968.

2 L. E. Hochreiter, H. Chelemer, and P. T. Chu, "THINC-IV - An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores," WCAP-7956, June 1973.

3 T. W. T. Burnett, C. J. McItyre, J. C. Buker, R. P. Rose, "LOFTRAN Code Description," WCAP-7907, June 1972.

1547 301

Misaligned rods will be detectable by (1) asymmetric power distributions sensed by excore nuclear instrumentation or core exit thermocouples, (2) rod deviation alarm, and (3) rod position indicators. A deviation of a rod from its bank by 14.4 inches or twice the resolution of the rod position indicator will not cause power distributions to exceed design limits. In the event of a dropped control rod, the automatic controller may return the reactor to full power. Analysis indicates that a departure from nucleate boiling ratio of less than 1.3 will not occur during this event.

For the case of dropped control rod groups, the reactor will be tripped by the power range negative neutron flux trip and will be protected from core damage. For cases where a control rod group is inserted to its insertion limit with a single control rod in the group fully withdrawn position, analysis indicates that the departure from nucleate boiling ratio will remain greater than 1.30.

The staff concludes that anticipated transients will not lead to more serious plant conditions in the absence of other faults and that the plant design is acceptable with respect to transient response to events that might occur during the plant lifetime with the exception of the boron dilution incident during refueling operations.

15.5 Postulated Accidents

RESAR-41 presents analyses to evaluate the effects and potential consequences of postulated accidents due to single faults which have a small to extremely remote probability of occurrence. Such accidents meet the criteria of Conditions III and IV in the evaluation and classification presented by Westinghouse.

We have reviewed the accident analyses submitted by Westinghouse to assure completeness and conservatism in the analyses, and to evaluate the acceptability of the results.

15.5.1 Inadvertent Loading of a Fuel Assembly Into an Improper Position

Comparisons of calculations of the power distributions for the normal fuel loading pattern and five cases of fuel assembly and burnable poison misloadings have been presented by Westinghouse. These represent the spectrum of potential inadvertent improper loadings. With the exception of the case, "Interchange Between Region 1 and Region 2 Assemblies, (at center of core) Burnable Poison Rods Being Transferred to Region 1 Assembly," the resultant distortion of the power distribution would be detectable by the incore instrumentation (movable fission chamber detectors) provided. In the excepted case the distortion of power distribution is sufficiently small that the increase in the overall peaking factor (F_q) would be approximately the uncertainty in its measurement and hence would cause no safety problem.

A power distribution measurement with the incore instrumentation system will be required by the technical specifications to determine if misloadings exist. Thermocouples in approximately one-third of the fuel assemblies can also provide an indication of a loading mistake. In most cases, however, an improperly loaded fuel assembly will cause a quadrant power tilt that can be detected by the excore nuclear instrumentation.

In addition to these instrumentation systems which will detect misloadings, strict administrative controls will be provided to prevent such events.

We conclude that an improperly loaded fuel assembly or burnable poison cluster that would cause a significant safety problem will be detectable with the instrumentation provided.

15.5.2 Feedwater System Piping Breaks

The analysis of a major feedwater line break inside containment with loss of offsite power has been reviewed. The maximum size feedwater line break accident between the steam generator and feedwater line check valve was assumed to be the most severe case. Since the feedwater line rupture has the potential of reducing the capability of the secondary system to remove the heat generated by the core, an auxiliary feedwater system must be provided with the balance of plant to assure that adequate feedwater will be available to remove decay heat and to prevent overpressurizing of the reactor core. The analysis indicates that the assumed auxiliary feedwater capacity of 500 gallons per minute minimum at 1300 pounds per square inch, gauge will be sufficient to remove the decay heat from the core and that the relief capacity of the pressurizer safety valves will be sufficient to prevent overpressurization of the reactor coolant system.

The postulated accident was evaluated using mathematical techniques that we have not reviewed. We conclude that the results presented for a major feedwater line break are not unlike those determined for similar accidents in comparable plants and that, on this basis, they are acceptable and sufficient for the preliminary design review. We will review the methods used by Westinghouse prior to final design approval for RESAR-41.

15.5.3 Rupture of a Control Rod Drive Mechanism Housing

The mechanical failure of a control rod drive mechanism housing would result in the ejection of a rod cluster control assembly. The consequences of this 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, Westinghouse has analyzed the consequences of such an event. Methods used in the analysis are reported in WCAP-7588, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Method" Revision 1, which we have reviewed and accepted by letter to Westinghouse dated August 28, 1973. This report demonstrates that the "adiabatic" model used in the accident analysis is conservative relative to a three-dimensional kinetics calculation.

The ejected rod worths and reactivity coefficients used in the analysis have been reviewed and are reasonable. The Westinghouse criteria for gross damage of fuel are a clad temperature of 2700 degrees Fahrenheit and an energy deposition of 200 calories per gram. We find these criteria acceptable and conservative in relation to our criteria of 280 calories per gram.

1547 303

Four cases were analyzed. The beginning of cycle at 102 percent and zero power and the end of cycle at 102 percent and zero power. The worst case was the end of life 102 percent power case which resulted in a clad temperature of 2565 degrees Fahrenheit and 181 calories per gram energy deposition. As a result, gross fuel damage would not occur.

The analysis shows that less than 10 percent of the fuel goes through departure from nucleate boiling.

The assumptions and methods of analysis used by the Westinghouse are in accordance with Regulatory Guide 1.77. We conclude that the predicted consequences of a postulated rod ejection accident are acceptable.

15.5.4 Spectrum of Steam Piping Failures Inside and Outside of Containment

The analyses and effects of postulated steam line break accidents inside and outside containment during various modes of operation with and without offsite power, have been reviewed. The accident which resulted in the most severe consequences was determined and evaluated. The steam line break accident analysis in RESAR-41 assumes, as the most severe single failure, that an assumed non-return check valve in the steam line fails to close allowing backflow out the break for 10 seconds while the isolation valves shut.

Applications referencing RESAR-41 must show that the most severe single failure in their specific steam system design will not result in a greater reactivity excursion than that determined in the RESAR-41 analysis. If the excursion is more severe, then the steam line break analysis must be redone for that particular plant.

The results of the analysis of the spectrum of steam line break accidents showed no expected fuel damage and no loss of core cooling capability. The minimum departure from nucleate boiling ratio experienced by any fuel rod was greater than 1.30. The maximum pressure within the reactor coolant and main steam systems did not exceed 110 percent of the design pressures. The postulated accident was evaluated using some mathematical techniques that we have not reviewed. We conclude that the predicted consequences of the steam line break accident are acceptable for a preliminary design review. We will review the methods used by Westinghouse prior to final design approval of RESAR-41.

15.5.5 Spectrum of Piping Breaks Within the Reactor Coolant Pressure Boundary

Pursuant to the final acceptance criteria for ECCS published in the Federal Register on January 4, 1974 (Appendix K to 10 CFR Part 50), and as stated in Section 6.3.5 of this report, Westinghouse is required to submit a LOCA analysis satisfying the requirements of the new criteria. We will review this information and report our evaluation as stated in Section 6.3.5 of this report.

15.5.6 Reactor Coolant Pump Rotor Seizure

The analysis of an instantaneous seizure of a rotor of a reactor coolant pump during any allowed mode of operation has been reviewed. The parameters used as input were reviewed and found to be suitably conservative. The results of the analysis showed that the peak clad surface temperature reached was 1837 degrees Fahrenheit. This assures that the fuel

1547 304

damage will not be extensive and that there will not be a consequential loss of core cooling capability. The analysis showed that the maximum pressure within the reactor coolant and main steam systems would not exceed 110 percent of the design pressures.

We conclude that the calculated consequences of a postulated reactor coolant pump rotor seizure are acceptable for the preliminary design review. The codes used in this analysis are presently under review by the staff and will be addressed as discussed in Section 15.2.2 of this report.

15.5.7 Anticipated Transients Without Scram

A number of plant transients can be affected by a failure of the scram system to function. For a pressurized water reactor the most important include loss of feedwater, loss of load, inadvertent control rod withdrawal, and loss of alternating current power. As required by WASH-1270, "Anticipated Transients Without Scram for Water-Cooled Power Reactors," Westinghouse did submit an evaluation on the RESAR-41 docket in the form of a topical report, WCAP-8440 "Anticipated Transients Without Trip for a Four-Loop (3817 MWt) Westinghouse PWR." Our evaluation and conclusions for this analysis will be included in a supplement to this report.

15.6 Summary Conclusions

On the basis of our review of the RESAR-41 accident and transient analysis, we find the consequences of normal and anticipated transients and postulated accidents at license thermal power levels up to 3800 megawatts to be generally acceptable. However, prior to issuance of a Preliminary Design Approval, we will require that:

(1) The review of RESAR-41 anticipated transients without scram is completed and any required changes are incorporated into the design.

(2) The loss-of-coolant analysis is submitted satisfying the requirements of Appendix K to 10 CFR Part 50.

(3) Steps are taken to assure that the operator will have at least 30 minutes to respond to any boron dilution incident which can occur during refueling operations.

In addition, we have informed Westinghouse that the following items must be completed prior to the issuance of the Final Design Approval:

(1) Trip delay times and uncertainties used to establish final trip setpoints within analyses values are fully justified.

(2) Reports on the steamline break and feedwater line break accidents are submitted and reviewed by the staff.

(3) A generic review of all computer codes used in the accident analyses and identified in Section 15.2.2 of this report is completed.

(4) Rod insertion times used in the safety evaluation are verified by test results.

(5) A report describing the methods used, including the application of codes, for analyses of the loss of flow transient is submitted for generic review.

15.7 Radiological Consequences of Accidents

15.7.1 General

The accidents we analyzed in evaluating the RESAR-41 nuclear steam supply system include the hypothetical loss-of-coolant accident, leakage of the emergency core cooling system equipment following a loss-of-coolant accident, a hydrogen purge of the containment after a loss-of-coolant accident, a fuel handling accident, and a rod ejection accident. The radiological consequences of the steam generator tube failure and main steam line failure accidents will be addressed in a supplement to this report.

These evaluations have been done to show in a relative way, the magnitude of the calculated dose that will be obtained when evaluating applications referencing RESAR-41. We have made reasonable dose reduction assumptions concerning the effectiveness of various systems outside the scope of RESAR-41. Of course, for each application referencing RESAR-41, we will perform calculations using specific assumptions that are valid for the particular plant and site.

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 primary and secondary coolant system radioactivity concentrations so that potential offsite doses are a small fraction of the 10 CFR Part 100 guideline values. We will include appropriate limits on primary and secondary coolant activity concentrations in the technical specifications for those plants referencing RESAR-41.

The RESAR-41 evaluation of iodine releases resulting from tube rupture and steam line break accidents does not include the effects of iodine spiking. We believe that iodine spiking is a major factor in the iodine release and will require that it be included in the evaluation of iodine release. We will require that information currently available from operating plants be used to conservatively estimate the magnitude of the iodine spike and that this information be submitted as part of the Final Design Approval application.

RESAR-41 includes an evaluation of the iodine removal effectiveness of a containment spray system. This evaluation is presented as an example only and addresses a system which is not within the scope of RESAR-41. Therefore, we have not reviewed this evaluation or the iodine removal model used. We will review the containment spray system and its effectiveness in removing iodine for each application which references RESAR-41.

Similarly, dose models used to evaluate the environmental consequences of accidents are presented as examples only. We have not reviewed these models and will do so for each application which references RESAR-41.

1547 306

RESAR-41 includes models for secondary containment effectiveness following a loss-of-coolant accident. However, RESAR-41 does not include a containment or a secondary containment design. Any loss-of-coolant accident modeling assumptions developed for an individual plant necessarily would be based on the system design and anticipated performance of the containment and secondary containment proposed for that plant. Many of the assumptions presented are usually plant specific and some of the assumptions are not in agreement with current staff practice. We have not, therefore, reviewed these models.

15.7.2 Loss-of-Coolant Accident

We have postulated a loss-of-coolant accident for the RESAR-41 design to determine the exclusion boundary value for the relative concentration which would limit the dose consequences to the guidelines of Regulatory Guide 1.4. Although the containment and the fission product removal and control systems are not within the scope of RESAR-41, we have assumed such systems for calculational purposes. The assumed containment model includes a low leakage single containment structure surrounding the reactor and a sodium hydroxide injection system operating in conjunction with the containment spray system.

The purpose of the sodium hydroxide injection system will be to increase the iodine removal capability of the spray following a hypothetical loss-of-coolant. We have reviewed and approved spray systems having two hour thyroid dose reduction factors ranging from four to eight. We used a dose reduction factor of 5.5 for calculational purposes. Our assumptions for this accident are listed in Table 15-4, and the doses are listed in Table 15-3. We used a relative concentration of 1.0×10^{-3} for calculational purposes. This results in a two-hour thyroid dose of 360 roentgen equivalent man considerably greater than the value of 150 roentgen equivalent man needed to meet the recommendations of Regulatory Guide 1.4. The required short-term relative concentration needed to meet the guideline values of Regulatory Guide 1.4 is 4.1×10^{-4} in seconds per cubic meter. Of those sites we previously evaluated, approximately 75 percent had 0 to 2 hour atmospheric dispersion values greater than 4.1×10^{-4} seconds per cubic meter (indicating poorer dispersion conditions) at the exclusion area boundary.

As part of the loss-of-coolant accident, we have also evaluated the consequences of leakage of containment sump water containing radioactive fission products which will be circulated by the emergency core cooling system outside the containment after a postulated loss-of-coolant system. We and Westinghouse have assumed the sump water to contain a mixture of iodine fission products in agreement with Regulatory Guide 1.7. After the loss-of-coolant accident, this water will be circulated outside of the containment in an area to be designated by the balance-of-plant designer. If a source of leakage should develop, such as from a pump seal, we believe a portion of the iodine would become gaseous and would exit to the outside atmosphere. The offsite doses resulting from such a sequence of events depends upon the temperature and magnitude of the assumed leakage and the site meteorology. If the leakage occurred when the water temperature was below 212 degrees Fahrenheit, a leak rate of about 10 gallons per minute over a period of one-half hour would result in doses (without filters) which could exceed the guideline values of 10 CFR Part 100 (for a relative concentration of 1.0×10^{-3} seconds per cubic meter) from this source alone. If the leakage occurred when the fluid is near its peak temperature of 240 degrees Fahrenheit, then part of the leaking water would flash to steam, leading to additional iodine release. In this case, about two gallons per minute leakage for one-half hour (for the same relative concentration) would result in doses (without filters) which could exceed 10 CFR Part 100 guidelines, from this source alone.

If the emergency core cooling system equipment area is served by filters effective in removing iodine, the offsite doses from possible pump leakage in this area will be within the guidelines of 10 CFR Part 100, even for substantial amounts of leakage. As a result of the analysis discussed above, we will require that for those plants referencing RESAR-41, the balance-of-plant designer locate the emergency core cooling system equipment in an area served by filters which are effective in removing iodine and which conform to the requirements of engineered safety features systems.

15.7.3 Fuel Handling Accident

We have evaluated the radiological consequences of a fuel handling accident. Our assumptions for this accident are consistent with the conservative assumptions of Regulatory Guide 1.25 and are listed in Table 15-5. We assumed that the filters used to mitigate the consequences of this accident will meet the requirements of Regulatory Guide 1.52. They have, therefore, been given credit for a removal efficiency of 95 percent for all forms of iodine. We will require that those plants which reference RESAR-41 install engineered safety features filters which meet the requirements of Regulatory Guide 1.52 to mitigate the consequences of a fuel handling accident. The calculated doses are listed in Table 15-3 of this report.

Using an assumed value for a relative concentration of 1.0×10^{-3} seconds per cubic meter for calculational purposes, the resulting dose would be about 29 thyroid and 16 whole body roentgen equivalent man. Thus, the consequences of the loss-of-coolant accident are more limiting.

15.7.4 Control Rod Ejection Accident

We have evaluated the consequences of a rod ejection accident for the containment leakage mode only. The assumptions used to calculate offsite doses from a control rod ejection accident were:

(1) Power level of 4100 thermal megawatts.

(2) 10 percent fuel failed in transient.

(3) 10 percent of iodine and noble gas inventory in gap of failed fuel.

(4) Release of total gap activity in failed fuel to containment building.

(5) 50 percent plate-out of radioactive iodines.

(6) Containment building sprays are not initiated.

(7) Containment building leak rate of 0.10 percent of the containment volume per day for 24 hours and one-half of this value thereafter.

1547 308

(8) Standard ground level release meteorology and dose conversion factors.

(9) Relative concentration of 1.0×10^{-3} seconds per cubic meter.

The calculated doses are listed in Table 15-3 of this report. The two-hour thyroid dose is approximately 40 rem and the whole body dose is less than one rem.

The consequences of the leakage of fission products to the secondary system after a rod ejection accident have not been analyzed because there is insufficient information on the balance of plant outside the scope of RESAR-41. Radiological consequences of this accident may be limiting (in terms of limits for primary to secondary steam generator leakage) for certain balance of plant designs and will be evaluated on a case-by-case basis.

15.7.5 Postulated Radioactive Releases Due to Liquid Waste Tank Failures

The consequences of tank failures that could result in the release of contaminated liquids to potable water supplies is site dependent, and will be reviewed for individual license applications. We have evaluated the source terms provided in Table 11.25 of RESAR-41 for these tanks and we conclude that they are acceptable for use in calculating the radioactive releases due to liquid tank failures by applicants referencing RESAR-41.

TABLE 15-3

POTENTIAL OFFSITE DOSES DUE TO DESIGN BASIS ACCIDENTS

	Two-Hour Exclusion Boundary		Course of Accident Low Population Zone	
Accident	Thyroid (Rem)[1]	Whole Body (Rem)[1]	Thyroid (Rem)[1]	Whole Body (Rem)[1]
Loss-of-coolant	360 (150)[2]	16. (6.6)[2]	---	---
Fuel Handling	29 (12)[2]	16. (6.6)[2]	---	---
Rod Ejection	40 (16)[2]	<1 (<1)[2]	---	---

[1] REM - roentgen equivalent man

[2] For an assumed relative concentration (X/Q) of 4.1×10^{-4} seconds per cubic meter which, for the assumed parameters of our calculations, is a maximum in order to stay within the guidelines of Regulatory Guide 1.4.

1547 309

TABLE 15-4

ASSUMPTIONS USED IN THE ESTIMATE

OF DESIGN BASIS ACCIDENT DOSES

LOSS-OF-COOLANT ACCIDENT

Thermal power Level	4100 megawatts
Operating Time	3.0 years
Reactor Building Leak Rate (0-24 hours)	0.10 percent
(>24 hours)	0.05 percent
Iodine Composition	
Elemental	91 percent
Particulate	5 percent
Organic	4 percent
Relative concentration Values (seconds per cubic meter)	
0-2 hours	1.0×10^{-3}
Two-Hour Thyroid Dose Reduction Factor for Spray	5.5

TABLE 15-5

FUEL HANDLING ACCIDENT CALCULATION INPUT PARAMETERS

Shutdown Time	20 hours
Total Number of Fuel Rods in the Core	50,952
Number of Fuel Rods Involved in the Refueling Accident	264
Power Peaking Factor	1.65
Iodine Fractions Released from Pool	
Elemental	75 percent
Organic	25 percent
Effective Filter Efficiency	
Elemental	95 percent
Organic	95 percent
Relative Concentration Values, (seconds per cubic meter)	
0-2 hours	1.0×10^{-3}

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 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 RESAR-41 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 those being developed or in use for plants of a similar Westinghouse design.

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

1547 311

17.0 QUALITY ASSURANCE

17.1 General

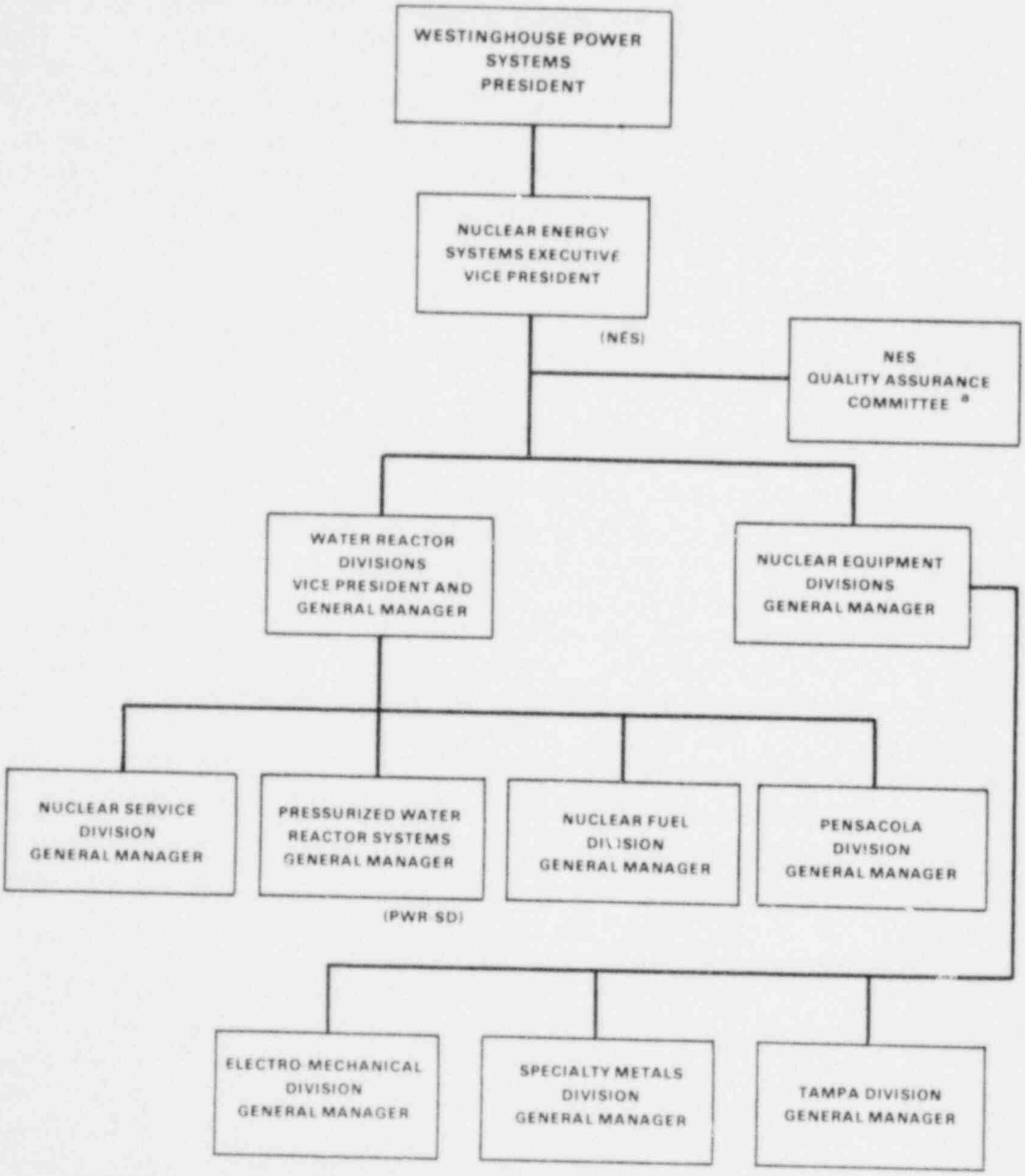
Section 17 of RESAR-41 describes, by reference to topical report WCAP-8370, "Westinghouse Nuclear Energy System Divisions Quality Assurance Plan," the quality assurance program of the Nuclear Energy Systems of Westinghouse. The program covers safety-related equipment from design through procurement, fabrication, manufacture, turnover, and, as applicable, installation, preoperational tests, and operation of a standard pressurized water reactor 3817 megawatts thermal nuclear steam supply system. Our evaluation of this quality assurance program is based on a review of the information provided and discussions and meetings with Westinghouse to determine how their quality assurance program complies with the requirements of Appendix B to 10 CFR Part 50 and the applicable Regulatory Guides.

17.2 Organization

Nuclear Energy Systems is a group of Westinghouse Divisions which provides nuclear power plant services and equipment. As shown by Figure 17-1, Nuclear Energy Systems operates under an Executive Vice-President who reports to the President, Westinghouse Power Systems. This Executive Vice-President establishes Nuclear Energy Systems quality assurance policy which each Nuclear Energy Systems Division implements. This results in uniform implementation of Appendix B to 10 CFR 50. The Pressurized Water Reactor Systems Division of Nuclear Energy Systems is the lead division with respect to design and procurement (Figure 17-2).

Each Division has an organization specifically responsible for quality assurance and for quality control which reports at a level to assure independence consistent with Criterion I of Appendix B. Quality management in each Division is free of prime responsibility for schedule or cost, has the authority to stop work pending resolution of quality matters, and has the freedom to (1) identify quality problems, (2) initiate, recommend, or provide solutions through designated channels, (3) verify implementation of solutions and (4) control further processing, delivery, or installation of nonconforming items. In each Division, persons performing Quality Assurance functions have access to higher management for arbitration of unresolved issues.

The Executive Vice President of Nuclear Energy Systems has established a Quality Assurance Committee which includes the Quality Assurance and Reliability Managers of each Division. The Manager of the Systems Division Product Assurance is Chairman of the Quality Assurance Committee. This committee is responsible for auditing activities throughout Nuclear Energy Systems to assess whether the requirements of Appendix B to 10 CFR Part 50 are effectively met. The Quality Assurance Committee has the authority to identify problems, recommend solutions, and verify effective implementation of actions and policies. The Quality Assurance Committee audits each Nuclear Energy Systems Division annually to assess the scope, implementation, and effectiveness of the division's program. The recommendations



a) The NES Quality Assurance Committee is Composed of the Quality Assurance and Reliability Managers from Each of the NES Divisions. The Committee's Chairman is the PWR-SD Product Assurance Manager. (See Figure 17-2)

Figure 17-1 ORGANIZATION OF WESTINGHOUSE NUCLEAR ENERGY SYSTEMS DIVISIONS (NES)

1547 313

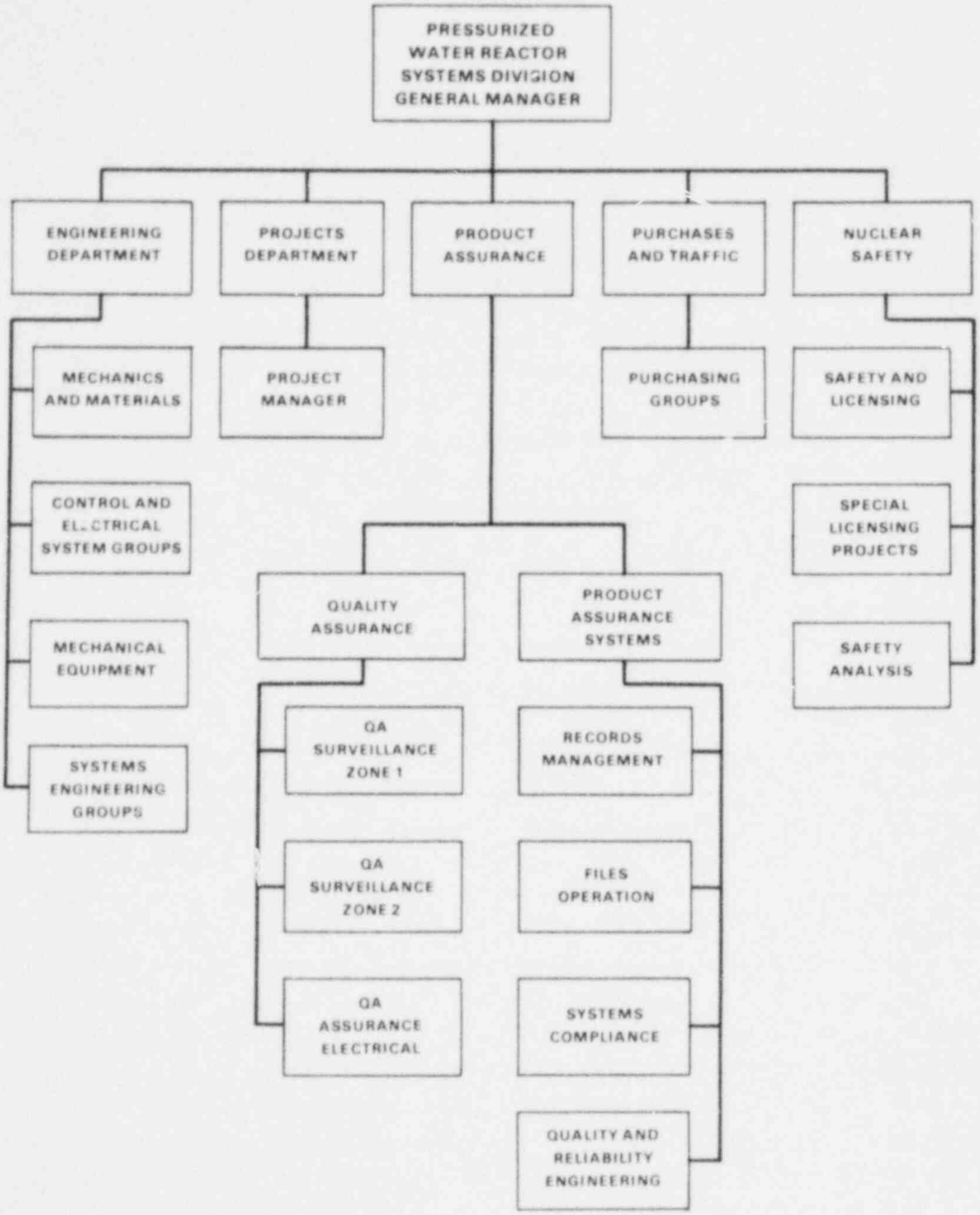


Figure 17-2 PRESSURIZED WATER REACTOR SYSTEMS DIVISION QUALITY ASSURANCE PROGRAM ORGANIZATION

1547 314

of this committee for improved and more consistent policies, when adopted, result in further policy directives authorized by the Nuclear Energy Systems Executive Vice-President.

17.3 Quality Assurance Program

The quality assurance program applies to all safety related systems and components of Westinghouse Nuclear Steam Supply Systems. The program commits Westinghouse to comply with the requirements of Appendix B to 10 CFR 50 and to follow the guidance provided by the Commission in (1) "Guidance on Quality Assurance Requirements During Design and Procurement Phase of Nuclear Power Plants - Revision 1" (WASH-1283), May 1974, and (2) "Guidance on Quality Assurance Requirements During the Construction Phase of Nuclear Power Plants" Revision 0 (WASH-1309), May 10, 1974. Westinghouse has also agreed to follow Commission guidance in "Guidance on Quality Assurance Requirements During the Operations Phase of Nuclear Power Plants" (WASH-1284), October 26, 1973, when applicable.

Since each Nuclear Energy Systems Division has a different scope of work, each Division Manager must further amplify the common quality assurance policy as necessary for local application. Each Division establishes, documents, and implements a program which assures that safety-related items meet the applicable requirements of Appendix B to 10 CFR 50. In addition, each division requires that applicable requirements of Appendix B be implemented by all sub-tier suppliers of safety-related items. The General Manager of each Division authorizes, reviews, and approves the quality assurance program for his division. A quality assurance manual, reviewed and approved by the division's quality assurance Management, defines the program. A matrix which relates the procedures of the various manuals to the applicable criteria of Appendix B to 10 CFR 50 is given.

The Nuclear Energy System quality assurance policy is communicated by means of applicable manuals and formal training and indoctrination programs. Managers in the Divisions are committed by the program to assure that their groups are familiar with the division's program and comply with applicable procedures in the quality assurance Manual.

The program includes provisions for the control of design information. Contractual requirements from an applicant for a construction permit and the contents of the Safety Analysis Report provide inputs to the design process. These inputs are reviewed as the design progresses. Analyses are accomplished in accordance with applicable codes, standards, and regulatory requirements. Knowledgeable groups within Westinghouse, including quality and reliability personnel, independently review drawings and equipment specifications prior to issuance. Cognizant Nuclear Energy Systems personnel also review supplier's detailed designs and procedures. Design changes are controlled in a manner similar to the initial design. In addition, Westinghouse performs independent design verification activities, formal in depth design reviews, and performance tests on a selective basis to confirm that equipment will perform satisfactorily. Interfaces are defined and documented.

The quality assurance program includes provisions for control of purchased items and services. Westinghouse evaluates the quality system of each prospective supplier of

1547 315

safety related items. Purchase orders are reviewed for technical and quality requirements. Quality engineers review purchase requisitions, purchase orders, and subsequent change notices. Nuclear Energy Systems reviews and retains supplier documentation which demonstrates acceptable quality. Audits and feedback of discrepancy data are used by quality engineers tc measure supplier performance.

Each Division controls nonconforming material, parts, and components to prevent their inadvertent use and provide for their identification, segregation, and disposition.

Nuclear Energy Systems requires records which show the quality of thé product. They provide a filmed copy of these records to the utility prior to plant acceptance. Prior to item installation at a plant site, a copy of the purchase order, the applicable design specification, and a quality release are also provided to the utility. The quality release identifies approved nonconformance reports.

Westinghouse executes a comprehensive audit program. This audit program provides NES management with information on the effectiveness of the quality assurance program. Westinghouse audits activities affecting quality at Westinghouse and at supplier facilities. Audit areas include all quality related procedures and operations. Trained personnel, not having direct responsibilities in the area being audited, conduct the quality assurance audits in accordance with defined procedures and checklists.

17.4 Implementation

The Office of Inspection and Enforcement has conducted inspections to examine the implementation of the quality assurance program commitments made by Westinghouse in RESAR-41 to ascertain their conformance with 10 CFR Part 50, Appendix B. The examinations encompassed the Westinghouse Nuclear Energy Systems Divisions of Pressurized Water Reactor Systems Division, Electro Mechanical Division, Specialty Metals Division, and the manufacturing divisions in Tampa and Pensacola. These examinations focused on quality assurance activities related to the design, procurement, and manufacture of systems and components for nuclear power plants; and for each organization examined, included a review of established procedures and instructions and the execution of provisions contained therein.

Based thereon, the Office of Inspection and Enforcement has determined that there are no substantive unresolved issues relating to the implementation of the quality assurance program which require further identification and followup at this time. We conclude that the implementation of the Westinghouse RESAR-41 quality assurance program commitments is consistent with the ongoing activities in the Westinghouse Nuclear Energy Systems Divisions.

Continuing acceptability will be contingent upon Westinghouse maintaining a sustained satisfactory level of program implementation which will be verified through an ongoing program of periodic inspections by the Office of Inspection and Enforcement.

17.5 Conclusions

We find that the quality assurance program described in Section 17 of RESAR-41 provides for a comprehensive system of planned and systematic controls which adequately demonstrate

Westinghouse's ability and commitment to comply with each of the eighteen criteria of Appendix B tc 10 CFR Part 50. In addition, we have determined that Westinghouse quality assurance personnel have sufficient authority, organizational freedom, and independence to perform their quality assurance functions effectively and without undue influence from those organizational elements directly responsible for cost and schedules.

We conclude that the quality assurance program described in RESAR-41 complies with the requirements of Appendix B to 10 CFR Part 50 and is acceptable.

1547 317

18.0 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The RESAR-41 application is being reviewed by the Advisory Committee on Reactor Safeguards. We intend to issue a subsequent report after the Committee's report to the Commission relative to its review is available. The subsequent report will append a copy of the Committee's report and will address 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.

1547 318

19.0 CONCLUSIONS

Based on our analysis of the proposed RESAR-41 design we have determined that upon favorable resolution of the outstanding matters discussed herein, we will be able to conclude that, in accordance with the provisions of Appendix O and Section 50.35(a) of 10 CFR Part 50:

(1) Westinghouse has described the proposed design including, but not limited to, the principal engineering criteria for the design, the interface information necessary to determine compatibility of mating designs, and has identifed 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 reasonably be left for later consideration will be supplied prior to or in the final design application;

(3) Safety features or components which require research and development have been described by Westinghouse and it has identified, and will conduct, research and development programs reasonably designed to resolve safety questions associated with such features or components;

(4) On the bases of the foregoing, there is reasonable assurance that such safety questions will be satisfactorily resolved at or before the completion of our review of a final design application for RESAR-41;

(5) Westinghouse is technically qualified to design the proposed nuclear steam supply system;

(6) A preliminary design approval of the proposed design can be granted. Modifications or additional review by the Commission's staff and modifications proposed by Westinghouse will be conducted in accordance with existing Commission policy.

1547 319

APPENDIX A
NONSTANDARD SCOPE OF RESAR-41

Amendment 1 to WASH-1341, "Programmatic Information for the Licensing of Standardized Nuclear Power Plants" defined the standard scope of standard design applications. Some of the systems included in RESAR-41 are outside this standard scope. We have reviewed the RESAR-41 systems that are not in the standard nuclear steam supply system scope and our evaluations of these systems are included in this appendix.

The section numbering system used in this appendix is based on the numbers in the main body of the text that deal with the same subject matter. This correspondence is valid in all cases to the second digit (e.g., 6.2) but does not necessarily follow to the third digit.

1547 320

6.0 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.1 Containment Pressure Response

Although the containment building is not a part of RESAR-41, Westinghouse has presented as an option, techniques for long and short-term containment pressure response evaluations within RESAR-41. We have evaluated these techniques to determine their acceptability.

For short-term pressure response analyses, i.e., the pressure response to be used in determining the adequacy of the design pressure of the containment interior compartments for a loss-of-coolant accident, Westinghouse will use the Transient Mass Distribution (TMD) code as described in WCAP-8077, "Ice Condenser Containment Pressure Transient Analysis Method" for calculation of subcompartment pressures. This code considers 100 percent liquid entrainment between compartments and solves the basic conservation equations for two-component, two-phase flow. We have reviewed this topical report and concluded by letter to Westinghouse dated December 18, 1973, that the Transient Mass Distribution code is an acceptable method for subcompartment pressure analysis provided the code options of non-augmented flow and subsonic compressibility are applied.

For the long-term loss-of-coolant containment pressure response analysis, i.e., the pressure response to be used in determining the adequacy of the containment building design pressure during loss-of-coolants accidents, Westinghouse will use the COCO computer code which is offered as an option in RESAR-41. This code provides a one-volume model of the containment. Mass and energy rates are input and containment pressure is computed as a function of time utilizing mass and energy balance calculations. The effect of active heat removal systems, such as sprays and fan-coolers, is considered as well as the action of the passive heat sink structures. Regarding use of the COCO code for containment design evaluation, we have requested that Westinghouse provide additional information related to justifying the conservatism of the assumptions made regarding the initial containment conditions, i.e., temperature, pressure, and relative humidity, and the adequacy of the modeling of the containment heat sinks.

The COCO code assumes that fluid entering the containment from the break flashes into a mixture of steam and water. The liquid falls onto the containment floor and the steam is added to the containment atmosphere and acts to pressurize the containment. The water phase is assumed to be at the saturated temperature corresponding to the total pressure of the containment. This is referred to as the pressure flash assumption. We do not agree that this assumption is sufficiently conservative for containment design purposes. A conservative approach would be to assume that the liquid phase is sufficiently dispersed to come to thermal equilibrium with the containment atmosphere before falling to the sump. This is called the temperature flash assumption. The difference in containment pressure between the two flashing models may be as much as 2.0 pounds

1547 321

per square inch. We conclude that the COCO code should be modified to provide a more conservative flashing model. Accordingly, we find the code unacceptable for use without confirmatory analysis.

In the past, containment designs have been accepted for which the containment peak pressure was calculated using the COCO code. This was done on the basis of confirmatory analyses using the CONTEMPT-LT computer code and the demonstration of adequate margin in the containment design pressure. The CONTEMPT-LT code utilizes the temperature flash assumption.

Until the above information and modification is provided, reviewed and found acceptable by us, we will continue to review the adequacy of the design pressure for containment buildings determined using the COCO code, by the use of our confirmatory analysis. An additional margin of 10 percent above our conservatively calculated peak containment pressure will then be added to establish the acceptable design pressure.

6.2.2 Combustible Gas Control Systems

Criterion 41 of the General Design Criteria requires that systems to control hydrogen which may be released into the containment shall be provided as necessary to ensure that containment integrity is maintained. We reviewed the system described in RESAR-41 for controlling containment hydrogen concentration following a loss-of-coolant accident for conformance to the requirements of Criterion 41 of the General Design Criteria and other staff requirements.

In order to mitigate the consequence of excessive hydrogen accumulation in the containment building following a loss-of-coolant accident, Westinghouse had proposed the use of two redundant electric hydrogen recombiners located inside containment and a backup purge system. These recombiners are included as options in RESAR-41.

Following a loss-of-coolant accident, hydrogen may accumulate inside the containment. The major sources of hydrogen generation include:

(1) A chemical reaction between the zirconium fuel rod cladding and steam;

(2) A chemical reaction between construction materials and water or reactive spray solutions.

(3) Radiolysis of aqueous solutions in the reactor core and in the containment sump.

Each of the two 100 percent capacity electric recombiners will be capable of processing 100 standard cubic feet per minute of containment atmosphere for post-accident hydrogen control. We have reviewed tests that have been conducted for a full-scale prototype and a production recombiner. The tests consisted of proof-of-principle tests, and functional tests for a production recombiner. These tests are described in WCAP-7709, "Electric Hydrogen Recombiner for PWR Containments" and its Supplements 1-4. The results of these tests demonstrated that the recombiner should be capable of controlling

1547 322

the hydrogen in a post-loss-of-coolant accident containment environment. The recombiner system will be designed as seismic Category I and to the Institute of Electrical and Electronics Engineers requirements for an engineered safety feature and will be designed for installation inside containment.

Westinghouse has analyzed the post-accident hydrogen concentration in a typical containment building to be used with RESAR-41. This analysis is consistent with the guidelines of Regulatory Guide 1.7. A containment volume of 2.55 million cubic feet was assumed. Using the guideline assumptions, and the typical containment building, Westinghouse has calculated that the hydrogen concentration in the containment will not reach the lower flammability limit of four volume percent. Westinghouse has concluded that the hydrogen concentration in the containment can be maintained below two volume percent by activating electric recombiners 10 days following the accident.

In our evaluation of the balance of plant design for applications proposing to use the Westinghouse recombiners we will compare the assumptions used in this analysis with those that exist for each plant (i.e., containment volume and hydrogen source terms). We will also review the provisions for atmospheric mixing within the containment.

By letter dated May 1, 1975 we accepted the Westinghouse recombiner for controlling hydrogen inside the containment following a loss-of-coolant accident as an acceptable means of satisfying the applicable requirements of Criterion 41 of the General Design Criteria.

1547 323

9.0 AUXILIARY SYSTEMS

9.1 Spent Fuel Pool Cooling and Cleanup System

Criterion 61 of the General Design Criteria requires that the fuel storage system shall be designed with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and to prevent significant reduction in fuel storage coolant inventory under accident conditions.

We reviewed the spent fuel pool cooling and cleanup system, included in RESAR-41 as an option, for conformance to the requirements of Criterion 61 of the General Design Criteria and other staff requirements.

The proposed spent fuel pool cooling and cleanup system will be designed to remove decay heat from the spent fuel pool. A secondary function will be to maintain clarity and purity of the spent fuel cooling water and the refueling water. The cooling portion of the system will consist of two 50 percent capacity trains, which will dissipate the decay heat to the component cooling water system. Both trains will normally be used to remove the decay heat load generated by storage of fuel assemblies from one-third of a core, although up to 1-1/3 cores may be stored.

The predicted performance of the system is shown below for various operating conditions.

Cores Stored	1/3	1/3	1-1/3	1-1/3
Cooling Trains operating	1	2	1	2
Pool Cooling Water Temp °F	150	120	195	150

These conditions are based upon component cooling water supplied at a maximum temperature of 105 degrees Fahrenheit at a flow rate of 1.5 million pounds per hour to each spent fuel pool heat exchanger.

Westinghouse had proposed that the spent fuel pool cooling system not be designed to seismic Category I requirements as required by Regulatory Guide 1.29. Westinghouse claimed that meeting the Category I design requirements would provide no significant benefit to the safety of the plant although they provided no justification for that position.

1547 324

Our position, stated in Regulatory Guide 1.29, is that the systems required for cooling the spent fuel pool are important to safety and accordingly must meet the requirements of Criterion 2 of the General Design Criteria to withstand the effects of earthquakes without loss of capability to perform their safety functions. Therefore, we require that the proposed design be modified to meet the requirements of Regulatory Guide 1.29. As a result, by Amendment 17, Westinghouse committed to design this system to seismic Category I requirements.

We have reviewed the design bases, system description, component description and safety evaluation included in RESAR-41 and conclude that the system will meet the applicable requirements of Criteria 2 and 61 of the General Design Criteria and Regulatory Guide 1.29. We find the proposed spent fuel pool cooling system design acceptable.

1547 325

11.0 RADIOACTIVE WASTE MANAGEMENT

11.1 Summary Description

Criterion 60 of the General Design Criteria requires that the nuclear power unit design include means to suitably control the release of radioactive materials in gaseous and liquid effluents.

We reviewed the optional liquid and gaseous radioactive waste systems in RESAR-41 for conformance with the requirements of Criterion 60 of the General Design Criteria and other staff requirements.

The radioactive waste systems proposed in RESAR-41 consist of liquid and gaseous waste systems for a standardized four-loop, single unit nuclear steam supply system for a pressurized water reactor. The design objective for these systems will be to restrict the amount of radioactive material released to the environment to levels as low as practicable.

The liquid waste system will process waste liquid streams such as equipment drains, coolant leakage, demineralizer resin flushing liquids, decontamination and laboratory waste liquids, and laundry and shower waste water. The treated liquid waste will be recycled for reuse if the plant water balance requires makeup and if the water quality is adequate. The liquid waste system will utilize evaporation, demineralization, and filtration for removal of radioactive material, chemical impurities and particulates.

Gaseous wastes will be generated during the operation of the plant from degassing primary coolant, from vents for equipment handling radioactive materials, and as a result of leakage from systems and components containing radioactive material. The gaseous waste system will treat gaseous streams for radioactive material removal by filtration, adsorption, and holdup for radioactivity decay. Small amounts of gaseous radioactive materials will be released to the environment under monitored and controlled conditions.

The liquid and gaseous waste systems will be designed to accommodate the waste produced during operation of a single unit at a maximum core thermal power level of 3800 megawatts.

We have performed an evaluation of the average quantities of radioactive materials that will be released in the liquid and gaseous plant effluents based on the system design. In our evaluation we consider waste flows, waste activities, and equipment operating performance that are consistent with normal plant operation, including anticipated operational occurrences, over the life of the plant.

The parameters used in our evaluation, along with their bases, are given in Appendix B to WASH-1258, "Proposed Rule Making Action: Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low As Practicable" For Radioactive Materials in Light-Water-Cooled Nuclear Power Reactor Effluents."

1547 326

Modified versions of the ORIGEN and STEFFEG Codes, which were the liquid and gaseous calculational models we used, are described in Appendix C to WASH-1258. Our evaluation of RESAR-41 source terms can be found in Section 11.1 of the main body of this report.

The following items are not included in the scope of RESAR-41 and will be reviewed for individual license applications which reference RESAR-41.

(1) The monitoring, sampling, and control of radioactive effluents from the building ventilation systems, the containment purge, the main condenser air ejector and turbine gland sealing system exhausts.

(2) Provisions to collect and process spills from outside storage tanks.

(3) Provisions for collecting, processing, storing, handling and decontaminating filled solid waste containers.

(4) Provisions to collect, process and monitor turbine building floor drains.

(5) Process and effluent radiological monitoring systems.

(6) The seismic design classification of the foundations and adjacent walls of structures housing radioactive waste systems.

The following items are site dependent and will be reviewed for individual license applications.

(1) The capability of overall plant design to meet "as low as practicable" dose design objectives.

(2) The consequences of component failures that could result in release of radioactive liquids to potable water supplies and the nearest surface water.

11.2 Liquid Radioactive Waste Treatment System

11.2.1 General

In our evaluation of the liquid radwaste system we have considered the liquid radwaste treatment system design, design objectives, design criteria, methods of treatment, expected releases, and principal parameters used in calculating the releases of radioactive materials in liquid effluents. Our review included the system piping and instrumentation diagrams and process flow diagrams showing methods of operation and factors that influence waste treatment, e.g., system interfaces and potential bypass routes.

The liquid radwaste treatment system will be designed to collect and process wastes based on chemical purity relative to the primary coolant. This will be determined by the origin of the waste in the plant. The system will consist of four subsystems, (1) the boron recycle system of the chemical and volume control system, (2) drain channel A, (3) drain channel B, and (4) the steam generator blowdown system.

Collection and processing of turbine building floor drain wastes are outside of the scope of RESAR-41 and will be reviewed for individual license applications. In our evaluation, we have considered that these wastes will be discharged without treatment.

The principal components making up each of the liquid radioactive waste systems, along with their principal design criteria, are listed in Table A 11-1.

Drain Channel A will process reactor grade wastes which enter the system via equipment drains, tank overflows, equipment leaks, and sample room sink drains from components outside reactor containment. Wastes will be processed through the gas stripper-evaporator package and mixed bed demineralizer.

Drain channel B will process miscellaneous low purity wastes collected in floor drains and building sumps by filtration, evaporation and demineralization. Laundry and hot shower drains will normally be filtered and released after monitoring for radioactivity. Should the radioactivity exceed a predetermined level, drain Channel B will also process these detergent wastes.

The effluent from the steam generator blowdown system will be released to the environment following dilution with condenser cooling water, when the radioactivity is below pre-determined limits. During periods of primary to secondary system leakage, the steam generator blowdown treatment system will process blowdown wastes through two cation and two mixed bed demineralizers in series.

In Amendment 7 to RESAR-41, Westinghouse calculated that the proposed liquid radwaste treatment systems will be capable of reducing the release of radioactive materials in liquid effluents to approximately 0.066 curies per year excluding tritium and dissolved gases, and approximately 62 curies per year for tritium. These calculations were based on a fraction of the fuel which is assumed to be releasing fission products to the primary coolant of 0.0012.

We have performed an evaluation of the radioactive materials that are expected to be released annually in liquid effluents during normal operation, including anticipated operational occurrences, using the parameters given in WASH-1258. We have determined that the proposed systems will be capable of reducing the release of radioactive materials in liquid effluents to less than 0.16 curies per year excluding tritium and dissolved gases, and less than 350 curies per year for tritium.

The capability of the system to meet the dose design objectives of Appendix I to 10 CFR Part 50 is site dependent, and will be reviewed for individual license applications. The cost benefit analysis required by Appendix I to 10 CFR Part 50 is site dependent and will also be performed for individual construction permit applications referencing RESAR-41.

11.2.2 Evaluation

We have evaluated the system capability to process wastes in the event of a single major equipment item failure and to accept additional wastes during operations which result in excessive liquid waste generation.

1547 328

TABLE A 11-1

DESIGN PARAMETERS OF PRINCIPAL COMPONENTS CONSIDERED IN LIQUID RADWASTE EVALUATION

Component	Number	Capacity**	Quality Group*
Boron Recycle System			
Evaporator Feed Demineralizer	2	250 gpm	C
Evaporator Condensate Demineralizer	1	120 gpm	D
Evaporator Package Polishing	1	15 gpm	C
Drain Channel A			
Waste Holdup Tank	1	10,000 gal	C
Waste Evaporator Condensate Tank	1	5,000 gal	D
Waste Evaporator Condensate Demineralizer	1	35 gpm	D
Waste Evaporator Package	1	15 gpm	C
Drain Channel B			
Chemical Drain Tank	1	600 gal	D
Laundry and Hot Shower Tank	1	10,000 gal	D
Floor Drain Tank	1	10,000 gal	D
Waste Monitor Tank	2	5,000 gal	D
Waste Monitor Tank Demineralizer	1	30 gpm	D
Steam Generator Blowdown Treatment System			
Cation Demineralizer	2	200 gpm	D
Mixed Bed Demineralizer	2	200 gpm	D

*Quality Group C components will be designed as seismic Category I and Quality Group D components will be of non-seismic design.

**gal = gallons
gpm = gallons per minute

1547 329

We have reviewed the effects of reactor operation with fission product leakage resulting from as high as one percent defective fuel. We have determined that under these conditions and for Westinghouse's assumed dilution flow of 10,000 gallons per minute, the concentration of radioactive materials in liquid effluents will be a small fraction of the limits in 10 CFR Part 20, Table 2, Column II.

The boron recycle and waste evaporators design capacities will each be 21,000 gallons per day. We have calculated the average expected waste flows from these systems and found them to be 1850 and 1340 gallons per day, respectively. The difference between the expected flows and design capacity will provide adequate reserve for processing surge flows. The design will allow wastes to be processed interchangeably between the systems in the event of equipment downtime. We have concluded that the system capacity and system design will be adequate for meeting the demands of the facility during anticipated operational occurrences.

The steam generator blowdown treatment system design offered as an option in RESAR-41 is based on phosphate chemistry control of secondary system water purity. The system is designed for a normal blowdown rate of 20 gallons per minute and a maximum blowdown rate of 50 gallons per minute. The system treatment will consist of two cation and two mixed bed demineralizers that will operate in series or separately. We have calculated the average expected blowdown rate to be approximately 29,000 gallons per day. The capacity of the system will be approximately 72,000 gallons per day. The difference between the expected flows and the design capacity will provide adequate reserve for processing surge flows.

We have concluded that the system capacity and system design will be adequate for meeting the demands of a station during anticipated operational occurrences for plants using phosphate addition to control secondary coolant chemistry. If volatile chemistry is used to maintain secondary coolant purity, the system will need to be re-evaluated.

We have evaluated the seismic design and quality group classification of liquid radwaste treatment equipment and the provisions to prevent and collect spills from indoor and outdoor storage tanks.

Liquid radwaste components will be designed to Quality Group D non-seismic Category I standards. To provide sufficient assurance of system integrity, Westinghouse has proposed to take additional measures for quality assurance which we have found to be acceptable. The seismic design of the building housing the liquid waste systems is outside the scope of RESAR-41 and will be reviewed for applications referencing RESAR-41.

All tanks that could potentially contain radioactive liquid will be provided with adequate warning of potential overflow conditions. These tanks will be equipped with level indication instrumentation, and potential overflow conditions will be annunciated in the main control room. In addition to tank level monitoring and alarming provisions

1547 330

overflows on tanks located inside the building will be collected and processed. Provisions to collect and process overflows and spillages from storage tanks located outside buildings are not within the scope of RESAR-41 and will be reviewed for applications referencing RESAR-41.

11.2.3 Conclusions

The liquid radwaste system includes the equipment and instrumentation to control the release of radioactive materials in liquid effluents. The scope of our review included (1) the system's capability to reduce the quantities of radioactive materials in liquid waste to levels which, when discharged and diluted through appropriate balance of plant systems, will be able to meet "as low as practicable" levels in accordance with 10 CFR Parts 20 and Section 50.34a of 10 CFR Part 50, considering normal operation and anticipated operational occurrences, (2) the design provisions incorporated to control releases of radioactive materials in liquid effluents in accordance with Criterion 60 of the General Design Criteria, and (3) the quality group and seismic design criteria.

We have reviewed the RESAR-41 system descriptions, process flow diagrams, piping and instrumentation diagrams and design criteria for the components of the liquid radwaste treatment system. We also performed an independent calculation of the releases of radioactive materials in liquid effluents based on the calculational methods of WASH-1258.

We find the system capable of maintaining concentrations of materials released in liquid effluents during periods of equipment downtime and design basis fuel leakage within the limits of 10 CFR Part 20, Table 2, Column II.

We find the seismic and quality group classification of the liquid waste system in accordance with our position and, therefore, acceptable for the items covered in RESAR-41.

We conclude that the liquid radwaste system described in RESAR-41 is an acceptable means of satisfying the applicable requirements of Criterion 60 of the General Design Criteria.

The basis for acceptance in our review has been conformance of Westinghouse's designs, design criteria, and design bases for the liquid radwaste system to the Commission's regulations and applicable Regulatory Guides, as referenced above, as well as staff technical positions and industry standards.

11.3 Gaseous Radioactive Waste Treatment System

11.3.1 General

In our evaluation of the gaseous radwaste systems, we have reviewed the system designs, design objectives, design criteria, methods of treatment, expected releases, and principal parameters used in calculating the releases of radioactive materials in gaseous effluents. Our review included the system piping and instrumentation diagrams, and the process flow diagrams showing methods of operation and factors that influence waste treatment, e.g., system interfaces and potential bypass routes.

The proposed gaseous radwaste treatment system will be designed to process wastes based on the origin of the wastes in the plant and their expected radioactivity levels. The gaseous waste processing system will process gases stripped from the primary coolant in the volume control tank, evaporator gas strippers, and reactor coolant drain tanks by recirculation through pressurized storage tanks and hydrogen catalytic recombiners.

Cover gas from aerated tanks and equipment will be monitored and released via the plant vent. The steam generator blowdown waste will be cooled and pumped to the main condenser from the surge tank where gases will be vented and released via the plant vent.

The gaseous radioactive waste treatment system will consist of a closed loop with two waste gas compressors, two catalytic hydrogen recombiners, and eight gas decay tanks to accumulate and to recycle the radioactive gases. The principal components in this system and their principal design criteria are listed in Table A 11-2 below.

TABLE A 11-2

DESIGN PARAMETERS OF PRINCIPAL COMPONENTS
CONSIDERED IN GASEOUS RADWASTE EVALUATION

Component	Number	Capacity (each)	Quality Group
Compressor	2	40 standard cubic feet per minute	C
Decay Tanks	8	600 cubic feet	C
Recombiners	2	50 standard cubic feet per minute	C

Westinghouse states that radioactive gases from the condenser air ejector exhaust, gaseous waste treatment system leakage, and auxiliary building ventilation exhaust should be processed through charcoal adsorbers prior to release. Actual designs of layout, monitoring, sampling, control and treatment of the ventilation exhausts are not within the scope of RESAR-41 and will be reviewed for individual license applications.

11.3.2 Evaluation

Westinghouse has estimated that the proposed gaseous radwaste treatment system will be capable of reducing the release of radioactive materials in gaseous effluents to approximately 275 curies per year for noble gases and 0.0002 curies per year for iodine-131 not including the plant ventilation exhaust releases to the environment.

We have performed an evaluation of the radioactive materials in gaseous effluents using the parameters given in Regulatory Guide 1.42 and WASH-1258 and a fraction of the fuel that is assumed to be releasing fission products to the primary coolant of 0.0012. With these parameters we have determined that the proposed gaseous radwaste treatment will be capable of reducing the release of radioactive materials in gaseous effluents to approximately 660 curies per year for noble gases and 0.038 curies per year for iodine-131 including the plant ventilation exhaust releases. The capability of the system to meet the dose design objectives of Appendix I to 10 CFR Part 50 is site dependent and will be

1547 332

reviewed for individual license applications. The cost-benefit analysis required by Appendix I to 10 CFR Part 50 is site dependent and will also be evaluated for applications referencing RESAR-41.

We have also reviewed the radiological effects of facility operation based on one percent of the operating fission product inventory in the core being released to the primary coolant. We have concluded that under these conditions the concentrations of radioactive materials in gaseous effluents will be a small fraction of the limits in 10 CFR Part 20.

We have reviewed the system design to determine its capability to process gaseous wastes during periods when major processing equipment may be down for maintenance and during periods of excessive waste generation. The volume of gases entering the pressurized storage tanks from the primary system and from tank cover gases will be relatively small due to the system to be provided for recombination of hydrogen with oxygen to form water. The bulk of the gas stored in the system will be nitrogen carrier gas. Due to the small volumetric input of reactor coolant system gases in comparison with the steady state nitrogen inventory, the holdup time provided by the system will not be significantly influenced by the letdown stripping rate from the reactor coolant system. The pressurized gas storage system design includes two storage tanks for shutdown use. These tanks contain nitrogen for use in the final stages of reactor coolant system degassing during shutdowns. Gases entering the shutdown tanks will be processed for hydrogen removal to maintain a constant gaseous inventory.

Redundant compressors and recombiners will be provided to allow operation during periods of equipment downtime. We have concluded that the system will have sufficient redundancy and the system design will be adequate for meeting the demands of the facility during anticipated operational occurrences.

We have reviewed the seismic design and quality group classification of components used in the gaseous waste treatment system and the ability of the system to withstand the effects of an explosion, if the potential for an explosive mixture exists.

The gaseous waste treatment system equipment will be designed to the Quality Group C standards of Regulatory Guide 1.26 and to non-seismic Category I criteria. The seismic design of the building housing the gaseous waste system is outside the scope of RESAR-41 and will be reviewed for applications referencing RESAR-41. The system design includes hydrogen and oxygen analyzers upstream and downstream of the recombiners that will initiate an alarm if hydrogen or oxygen concentrations vary beyond the design concentration limits. The system design will limit the hydrogen concentration downstream of the recombiner to 0.15 percent by volume and the oxygen concentration to 60 parts per million in the recombiner discharge line by automatically terminating the hydrogen and/or oxygen flow as required. In this manner the potential for explosive hydrogen-oxygen mixtures will be minimized. We have concluded that the gaseous waste system quality group and seismic design criteria and the design provisions incorporated to reduce the potential of hydrogen explosions are acceptable.

1547 333

11.3.3 Conclusions

The proposed gaseous radwaste system of RESAR-41 includes the equipment and instrumentation to control the release of radioactive materials in gaseous effluents prior to entering the plant vent. The scope of our review included (1) the system's capability to reduce the quantities and concentrations of radioactive isotopes in gaseous waste to levels which, when discharged and diluted through appropriate balance of plant systems, will be able to maintain releases of radioactive materials in gaseous effluents to "as low as practicable" levels during normal operation and anticipated operational occurrences, in accordance with 10 CFR Part 20 and Section 50.36a of 10 CFR Part 50, (2) the design provisions incorporated in accordance with Criterion 60 of the General Design Criteria, to preclude uncontrolled releases of radioactive materials in gaseous effluents, and (3) the quality group and seismic design classifications of the gaseous radwaste systems. We have reviewed the system descriptions, process flow diagrams, piping and instrumentation diagrams, and design criteria for the components of the gaseous radwaste system. We have performed an independent calculation of the expected releases of radioactive materials in gaseous effluents based on the methods of WASH-1258.

In our evaluation of Westinghouse's radioactive gaseous waste treatment system, we find that the proposed system (1) will have sufficient capacity, redundancy and flexibility to maintain releases of radioactive materials in gaseous effluents within the limits of 10 CFR Part 20, Table 2, Column 1, (2) will be designed to acceptable quality group and seismic design criteria, and (3) will have adequate protection to prevent a potential hydrogen explosion.

We conclude that the gaseous waste system described in RESAR-41 is an acceptable means of satisfying the applicable requirement of Criterion 60 of the General Design Criteria.

The basis for acceptance in our review has been conformance of Westinghouse's designs, design criteria, and design bases for the gaseous waste system to the applicable regulations and guides referenced above, and to staff technical positions and industry standards. Based on our evaluation, we find the proposed gaseous waste system acceptable.

11.4 Process and Effluent Radiological Monitoring

The process and effluent radiological monitors within the scope of RESAR-41 are shown in Table A 11-3 of this report and include those monitors required to assure that the boron recycle system and the radioactive waste systems perform their intended functions. The RESAR-41 design for process and effluent radiological monitoring will include provisions;

(1) For automatically terminating effluent releases and for sounding an alarm in the main control room in the event that the radiation levels in discharge line exceed a predetermined level.

(2) For continuous monitoring plant processes which may affect radioactivity releases to the environment.

1547 334

(3) To enable manual collection of representative samples of planned gaseous and liquid effluents prior to discharge to unrestricted areas during normal reactor operation and during anticipated operational occurrences, in order to allow laboratory measuring and recording of the quantity of each of the radionuclides present in these discharges as required by Regulatory Guide 1.21 and Section 50.36a of 10 CFR 50.

Westinghouse originally specified that releases from the waste gas processing system should be automatically terminated by a monitor located at the plant vent in the event the radioactivity exceeds a predetermined level. We required that these releases be automatically terminated by a monitor in the waste gas discharge line for more effective process and effluent control, in accordance with the guidelines of Regulatory Guide 1.21. As a result by Amendment 17, Westinghouse modified the design to control the releases from the waste gas processing system by a monitor in the waste gas discharge line.

We conclude that the design for process and effluent radiological monitoring of the systems within the scope of RESAR-41 are in accordance with the guidelines of Regulatory Guide 1.21 and other staff requirements and is acceptable.

1547 335

TABLE A 11-3

PROCESS AND EFFLUENT MONITORS

Monitor Location	Control Alarm Function
Boron Recycle Evaporator Condenser Discharge Line	Divert flow back to recycle evaporator feed demineralizer and alarm in the main control room
Liquid Radwaste Effluent Line	Terminate liquid radwaste release and alarm in the main control room
Steam Generator Blowdown Line	Terminate blowdown and alarm in the main control room
Steam Generator Blowdown Discharge Line	Terminate blowdown release and alarm in the main control room
Waste Gas Process Line	Terminate gaseous waste discharge and alarm in the main control room

1547 336

15.0 ACCIDENT ANALYSES

15.1 Hydrogen Purge Dose Analysis

Westinghouse has provided redundant recombiners for the purpose of controlling the concentration of hydrogen within containment after a design basis loss-of-coolant accident. In the event of failure of both recombiners, a backup purge system would be used. We have evaluated the additional dose an individual might receive due to purging the containment after the design basis accident. Our assumptions are listed in Table A 15-1 and the typical doses for a system without purge filters are listed in Table A 15-2.

This calculation will need to be performed on a case basis with actual site dispersion factors to assure that the loss-of-coolant accident plus the purge doses are within the exposure guidelines of 10 CFR Part 100. In some cases filtration of the hydrogen purge effluent may be required.

TABLE A 15-1

HYDROGEN PURGE DOSE INPUT PARAMETER ASSUMPTIONS

Core Thermal Power Level (megawatts)	4100
Volume of Reactor Building (cubic feet)	2.55×10^6
Purge Duration (days)	30
Holdup Time in Containment (days) Prior to Purge Initiation	29
Purge Rate (standard cubic feet per minute)	21.7
4-30 Day relative concentration (seconds per cubic meter)	5.0×10^{-6}

TABLE A 15-2

POTENTIAL OFFSITE DOSES DUE TO DESIGN BASIS ACCIDENTS

	Two-Hour Exclusion Boundary		Course of Accident Low Population Zone	
Accident	Thyroid (Rem)[1]	Whole Body (Rem)[1]	Thyroid (Rem)[1]	Whole Body (Rem)[1]
Post-LOCA Hydrogen Purge	---	---	78	<1
Gas Decay Tank Rupture	---	1.5 (<1)[2]	---	---

[1] Rem - roentgen equivalent man

[2] For an assumed relative concentration (X/Q) of 4.1×10^{-4} seconds per cubic meter which, for the assumed parameters of our calculations, is a maximum in order to stay in the guidelines of Regulatory Guide 1.4 for the loss-of-coolant accident dose.

15.2 Gas Decay Tank Rupture

We have evaluated the consequences of gas decay tank failures and determined that the doses are within the whole body dose guidelines of 10 CFR Part 100. The assumptions utilized are found in Table A 15-3.

TABLE A 15-3

GAS DECAY TANK RUPTURE

The assumptions used to calculate the offsite doses from a gas decay tank rupture are consistent with those given in Regulatory Guide 1.24 and are as follows:

(1) Gas decay tank contains one complete primary coolant loop inventory of noble gases resulting from operation with 1 percent failed fuel (29,400 curies of noble gases).

(2) Relative concentration value taken as 1.0×10^{-3} seconds per cubic meter.

1547 338

RESAR-41

APPENDIX B

CHRONOLOGY OF REVIEW OF

REFERENCE SAFETY ANALYSIS REPORT-41

November 1, 1973	Introductory meeting with Westinghouse
November 29, 1973	Application submitted for preliminary review
December 7, 1973	Letter to Westinghouse stating that application has been received and review has begun
February 13, 1974	Letter to Westinghouse stating that application is acceptable and requesting information
March 7, 1974	Letter from Westinghouse transmitting application for full review
March 11, 1974	Application docketed
March 14, 1974	Meeting with Westinghouse to discuss review schedule, interface problems, etc.
March 25, 1974	Letter to Westinghouse transmitting summary of meeting held March 14, 1974
April 5, 1974	Submittal of Amendment No. 1 consisting of partial response to letter of February 13, 1974
April 8, 1974	Letter to Westinghouse regarding staff report on anticipated transients without scram
April 10, 1974	Letter to Westinghouse transmitting review schedule for RESAR-41
May 9, 1974	Submittal of Amendment No. 2 consisting of partial response tc letter of February 13, 1974
May 22, 1974	Meeting with Westinghouse to discuss certain requests for additional information
June 4, 1974	Letter to Westinghouse requesting additional information and response to staff positions
June 20, 1974	Letter to Westinghouse requesting additional information and response to additional staff positions
June 25-26, 1974	Meeting with Westinghouse to discuss certain requests for information and interface criteria
July 26, 1974	Submittal of Amendment No. 3 consisting of partial response to requests of June 4 and June 20, 1974
August 12, 1974	Letter to Westinghouse regarding applicability of Regulatory Guides
August 19, 1974	Submittal of Amendment No. 4 consisting of partial response to requests of June 4 and June 20, 1974
September 3, 1974	Submittal of Amendment No. 5 consisting of partial response to requests of June 4 and June 20, 1974
September 12, 1974	Meeting with Westinghouse to discuss staff positions and requests for information

1547 339

September 17, 1974	Meeting with Westinghouse to discuss additional staff positions and requests for information
September 26, 1974	Letter from Westinghouse requesting extension of date of submitting information concerning anticipated transients without scram
October 1-2, 1974	Meeting with Westinghouse to discuss responses to staff positions and requests for information
October 9, 1974	Letter to Westinghouse requesting additional information and response to additional staff positions
October 15, 1974	Letter to Westinghouse concerning analyses and evaluations used in application in accordance with Regulatory Guide 1.49
October 15, 1974	Submittal of Amendment No. 6 consisting of partial response to request of June 20, 1974
October 18, 1974	Meeting with Westinghouse to discuss the definition of scope of standard nuclear steam supply system application
October 23, 1974	Letter to Westinghouse requesting additional information and response to additional staff positions
October 29, 1974	Meeting with Westinghouse, additional discussion concerning the definition of standard nuclear steam supply system scope
November 6, 1974	Letter to Westinghouse regarding WCAP-8394, "Roto-Lok Closure System Design"
November 22, 1974	Submittal of Amendment No. 7 consisting of response to request of October 9, 1974
November 22, 1974	Letter from Westinghouse transmitting information on spurious valve movement
December 3, 1974	Submittal of Amendment No. 8 consisting of partial response to request of October 23, 1974
December 5, 1974	Letter to Westinghouse granting extension of date of submittal of information requested April 8, 1974
December 16, 1974	Letter to Westinghouse requesting additional information and response to additional staff positions
December 16, 1974	Letter from Westinghouse transmitting report WCAP-8440 "Anticipated Transient Without Trip Analysis for a Four Loop (3817 MWt) Westinghouse PWR"
December 16, 1974	Submittal of Amendment No. 9 consisting of partial response to request of October 23, 1974
January 6, 1975	Submittal of Amendment No. 10 consisting of partial response to request of December 16, 1974
January 8, 1975	Letter to Westinghouse concerning Regulatory Guide 1.49, regarding maximum permissible core power levels for nuclear power plants
January 9, 1975	Letter to applicant requesting additional information

January 28, 1975	Submittal of Amendment No. 11 consisting of partial response to request of December 16, 1974
January 30, 1975	Meeting with Westinghouse to review outstanding items
January 30, 1975	Meeting with Westinghouse to discuss outstanding items
January 31, 1975	Meeting with Westinghouse to discuss RESAR-41 review schedule
February 17, 1975	Submittal of Amendment No. 12 consisting of revised and additional information
February 21, 1975	Letter to applicant requesting additional information
February 28, 1975	Submittal of Amendment No. 13 consisting of responses to requests for additional information
March 10, 1975	Submittal of Amendment No. 14 consisting of responses to requests for additional information
March 24, 1975	Letter from applicant transmitting proprietary information concerning data points generated for the construction of local kilowatt per foot limits (loss of coolant accident envelope)
March 25, 1975	Submittal of Amendment No. 15 consisting of responses to requests for additional information
April 16, 1975	Letter to applicant requesting information regarding emergency core cooling system analysis
April 18, 1975	Submittal of Amendment No. 16 consisting of responses to requests for additional information
April 29, 1975	Letter to applicant granting witholding from public disclosure the proprietary information submitted March 24, 1975
May 12, 1975	Meeting with Westinghouse to discuss requirements for interfaces
May 20, 1975	Meeting with Westinghouse to discuss criteria for identifying interfaces
June 2, 1975	Letter from Westinghouse commenting on "Programmatic Information for the Licensing of Standardized Nuclear Power Plants," WASH-1341, Amendment 1
June 9, 1975	Submittal of Amendment No. 17 consisting of interface information
June 19-20, 1975	Audit by the staff of the technical information normally transmitted by Westinghouse to its customers. Audit was for interface information and was conducted at Westinghouse offices in Pittsburgh

1547 341

APPENDIX C

BIBLIOGRAPHY

CONTAINMENT AND ENGINEERED SAFETY FEATURES SYSTEMS

Allen, A. O., "The Radiation Chemistry of Water and Aqueous Solutions," Van Nostrand Company, 1961.

American Nuclear Society, "Decay Energy Release Rates Following Shutdown of Uranium-Fuel Thermal Reactors (DRAFT)," ANS Standard ANS-5.1, Hinsdale, Illinois, October 1971.

Coward, H. F., and G. W. Jones, "Limits of Flammability of Gases and Vapors," Bureau of Mines Bulletin 503, 1952.

Moody, F. J., "Maximum Flow Rate of a Single Component, Two-Phase Mixture," Vol. 87, p. 134 Journal of Heat Transfer, February 1965.

Parsly, L. F., "Design Considerations of Reactor Containment Spray Systems Part VI," The Heating of Spray Drops in Air-Steam Atmospheres, USAEC Report ORNL-TM-2412, Oak Ridge, Tennessee, January 1970.

Rettig, W. H., G. A. Jayne, K. V. Moore, C. E. Slater, and M. L. Uptmor, "RELAP3 - A Computer Program for Reactor Blowdown Analysis," IN-1321, Idaho Nuclear Corporation, June 1970.

Richardson, L. C., L. J. Finnegan, R. J. Wagner, and J. M. Waage, "COMTEMPT - A Computer Program for Predicting the Containment Pressure-Temperature Response to a Loss-of-Coolant Accident," IDO-17220, Phillips Petroleum Company, June 1967.

Schmitt, R. C., G. E. Bingham, J. A. Norberg, "Simulated Design Basis Accident Tests of the Carolina Virginia Tube Reactor Containment," Final Report, IN-1403, Idaho Nuclear Corporation, December 1970.

Slaughterbeck, D. C., "Comparison of Analytical Techniques Used to Determine Distribution of Mass and Energy in the Liquid and Vapor Regions of a PWR Containment Following a Loss-of-Coolant Accident," Special Interim Report, Idaho Nuclear Corporation, January 1970.

Slaughterbeck, D. C., "Review of Heat Transfer Coefficients for Condensing Steam in a Containment Building Following a Loss-of-Coolant Accident," IN-1388, Idaho Nuclear Corporation, September 1970.

Tagami, T., "Interim Report on Safety Assessments and Facilities Establishment Project in Japan for Period Ending June 1965 (No. 1)," Prepared for the National Reactor Testing Station, February 28, 1966 (Unpublished work).

Uchida, H., A. Oyama, and Y. Toga, "Evaluation of Post-Incident Cooling Systems of Light-Water Power Reactors," in Proceeding of the Third International Conference on the Peaceful Uses of Atomic Energy held in Geneva, August 31 - September 9, 1964, Volume 13, Session 3.9, New York: United Nations 1965, (A/Conf. 28/P/436) (May 1964) pp. 93-104, Division of Technical Review, U. S. Nuclear Regulatory Commission, Washington, D. C.

MECHANICAL ENGINEERING

American National Standards Institute, "Power Piping," ANSI B31.1-1973, Hinsdale, Illinois, 1973.

American Society for Testing Materials, "Surveillance Tests on Structural Materials in Nuclear Reactors," Annual Book of ASTM Standards, Part 30, 1916 Race Street, Philadelphia, Pa. 19103, July 1973.

United States Atomic Energy Commission Guideline Document, "Inservice Inspection Requirements of Nuclear Power Plants Constructed with Limited Accessibility for Inservice Inspections," United States Nuclear Regulatory Commission, Washington, D. C. 20555, January 1969.

1547 342

MATERIALS ENGINEERING

American Society of Mechanical Engineers, "Protection Against Non-Ductile Failure," Section III, 1972 Summer Addenda, including Appendix G, United Engineering Center, 345 East 47th Street, New York, New York 10017.

American Society of Mechanical Engineers, "Methods and Definitions for Mechanical Testing of Steel Products," ASME Specification SA-370-71b, 1971 Edition, Summer and Winter, 1972 Addenda, Section II, Part A - Ferrous, United Engineering Center, 345 East 47th Street, New York, New York 10017.

American Society for Testing Materials, "Standard Method for Conducting Dropweight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels," ASTM-E-208-67, Annual Book of ASTM Standards, Part 31, 1916 Race Street, Philadelphia, Pa., July 1973.

American Society for Testing Materials, "Notched Bar Impact Testing of Metallic Materials," ASTM E 23-72, Annual Book of ASTM Standards, Part 31, 1916 Race Street, Philadelphia, Pa. 19103, July 1973.

American Society for Testing Materials, "Copper-Copper Sulfate-Sulfuric Acid Test for Detecting Susceptibility to Intergranular Attack in Stainless Steels," Annual Book of ASTM Standards, Part 3, 1916 Race Street, Philadelphia, Pa. 19103, 1973.

OTHER

American National Standards Institute, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," ANSI N18.2, Hinsdale, Illinois, August 1974.

1547 343

APPENDIX B

CHRONOLOGY OF REGULATORY RADIOLOGICAL REVIEW OF

SOUTH TEXAS PROJECT, UNITS 1 AND 2

November 8, 1973	Meeting with applicants regarding early review of quality assurance program
April 4, 1974	Meeting with applicants to discuss status of application to be tendered
May 14, 1974	Letter from applicants transmitting application for preliminary review
May 21, 1974	Letter to applicants acknowledging receipt of application
June 20, 1974	Letter to applicants stating application acceptable for docketing
July 5, 1974	Application docketed
July 9, 1974	Meeting with applicants regarding results of preliminary review
July 29, 1974	Submittal of Amendment No. 1, consisting of responses to letter of June 20, 1974
August 5, 1974	Letter from applicants transmitting seismic reflection sections
August 6, 1974	Letter from applicants providing schedule for submitting responses
August 12, 1974	Submittal of Amendment No. 2, consisting of corrected pages
August 22, 1974	Letter to applicants transmitting licensing review schedule
August 22, 1974	Letter from applicants transmitting information relative to an LWA
August 29, 1974	Letter to applicants requesting additional information
August 29-30, 1974	Meeting with applicants to discuss staff acceptance review concerning seismology, geology and hydrology of site
September 3, 1974	Letter to applicants concerning transportation by staff during site visit
September 16, 1974	Submittal of Amendment No. 3, consisting of information concerning soils studies

1547 344

September 16, 1974	Letter to applicants requesting additional information and response to staff positions
September 17, 1974	Letter from applicants transmitting additional information for Amendment No. 3
September 20, 1974	TWX to applicants requesting information concerning schedule
September 23, 1974	Letter from applicants in response to TWX of September 20, 1974
September 25-26, 1974	Site visit and meeting with applicants to discuss site related matters
September 27, 1974	Meeting with applicants to discuss electrical, instrumentation and control, auxiliary systems and containment
September 28, 1974	TWX from applicants in response to TWX of September 20, 1974
October 1, 1974	Submittal of Amendment No. 4, consisting of response to request dated August 29, 1974
October 2-3, 1974	Site visit and meeting with applicants to discuss site-related matters
October 10, 1974	Meeting with applicants to discuss seismic design
October 15, 1974	Submittal of Amendment No. 5, consisting of response to letter dated June 20, 1974
October 18, 1974	Letter to applicants requesting additional information
November 1, 1974	Submittal of Amendment No. 6, consisting of response to request of September 16, 1974
November 11, 1974	Letter from applicants transmitting photographs of the site
November 12-13, 1974	Site visit and meeting with applicants to discuss geological issues
November 15, 1974	Letter to applicants requesting additional information and response to additional staff positions
November 18, 1974	Submittal of Amendment No. 7, consisting of response to requests of September 16 and October 18, 1974
November 22, 1974	Letter to applicants requesting additional information

December 11, 1974	Letter from applicants providing information concerning site suitability
December 13, 1974	Submittal of Amendment No. 8, consisting of response to requests of September 16 and November 22, 1974
December 23, 1974	Submittal of Amendment No. 9, consisting of response to requests of September 16 and October 18, 1974
January 8, 1975	Submittal of Amendment No. 10, consisting of changes in the liquid waste systems, steam generator blowdown and condensate system
January 10, 1975	Meeting with applicants to discuss geophysical investigations
January 10, 1975	Letter to applicants requesting additional information and response to additional staff positions
January 17, 1975	Letter to applicants requesting additional information and response to additional staff positions
January 21, 1975	Submittal of Amendment No. 11, consisting of response to request of September 16, 1974
January 24, 1975	Meeting with applicants to discuss questions and positions related to containment safety
January 24, 1975	Letter from applicants concerning applicants' commitments to monitor subsidence and associated phenomena
January 30, 1975	Letter to applicants requesting additional information and response to additional staff positions
February 12, 1975	Letter to applicants requesting additional information
February 13, 1975	Meeting with applicants to discuss hydrology and soil structure interaction
February 14, 1975	Submittal of Amendment No. 12, consisting of response to request dated September 16, 1974
February 20, 1975	Letter to applicants requesting additional information
February 21, 1975	Letter to applicants requesting additional information
February 28, 1975	Submittal of Amendment No. 13, consisting of response to requests dated January 10 and 17, 1975

1547 346

February 28, 1975	Submittal of drawings in support of Amendment No. 13
March 7, 1975	Submittal of Amendment No. 14, consisting of response to requests dated January 10 and 17, 1975
March 7, 1975	Letter from applicants transmitting information requested in February 13, 1975 meeting
March 12, 1975	Letter to applicants requesting additional information and response to staff position on hydrogen production analysis
March 14, 1975	Submittal of Amendment No. 15, consisting of response to requests of January 10, January 30, and February 20, 1975
March 17, 1975	Letter to applicants concerning submittal of information relative to industrial security
March 19, 1975	Submittal of Amendment No. 16, consisting of response to requests dated January 10, January 30, and February 21, 1975
March 19, 1975	Letter from applicants transmitting report entitled, "Cyclic Triaxial Tests-Data Plots"
March 19, 1975	Letter from applicants transmitting report entitled, "Linear Study"
March 19, 1975	Letter from applicants transmitting report entitled, "Status of Deep Soil Boring"
March 24, 1975	Letter from applicants transmitting information concerning security plan
March 31, 1975	Submittal of Amendment No. 17, consisting of response to requests dated January 10 and January 30, 1975
April 11, 1975	Submittal of Amendment No. 18, consisting of clarification to previous requests and updated information
April 14, 1975	Letter to applicants transmitting Site Suitability Report
April 14, 1975	Submittal of Amendment No. 1 to License Application, consisting of updated general and financial information
April 17, 1975	Letter from applicant transmitting report, "Interpretation of Imagery - Observed Tonal Anomalies"

April 22, 1975	Submittal of Amendment No. 19, consisting of clarification to previous requests and updated information
April 30, 1975	Submittal of Amendment No. 20, consisting of clarification to previous requests and updated information
May 2, 1975	Meeting with applicants to discuss fire protection and other outstanding auxiliary power system items
May 7, 1975	Letter from applicants transmitting information in response to staff's requests
May 9, 1975	Submittal of Amendment No. 21, consisting of clarification to previous requests and updated information
May 13, 1975	Letter from applicants transmitting a discussion of the "Geotechnical Instrumentation Program" for the South Texas Project
May 16, 1975	Letter from applicants transmitting report entitled, "Cooling Effects of Low Pressure Carbon Dioxide in a Total Flooding System"
May 22, 1975	Submittal of Amendment No. 22, consisting of clarification to previous requests and updated information
May 22, 1975	Meeting with applicants to discuss RESAR-41 interface problems as they relate to the South Texas Project
May 23, 1975	Meeting with applicants to discuss heave and settlement and subsidence monitoring programs
May 27, 1975	Letter to applicants transmitting revised safety review schedule
June 6, 1975	Letter to applicants requesting additional information and response to positions concerning report submitted May 13, 1975
June 6, 1975	Letter from applicants transmitting information relative to May 13, 1975 submittal
June 10, 1975	Letter to applicants advising of staff positions regarding fuel surveillance and control room habitability
June 13, 1975	Submittal of Amendment No. 23, consisting of clarification to previous requests and updated information
June 13, 1975	Letter from applicants in response to letter of May 27, 1975

June 18, 1975	Meeting with applicants to discuss problems in delay of schedule and policy regarding priority of standardized plant applications
June 20, 1975	Submittal of Amendment No. 24, consisting of clarification to previous requests and updated information
June 26, 1975	Letter to applicants concerning schedule for radiological safety hearing
June 27, 1975	Submittal of Amendment No. 25, consisting of response to previous requests, information regarding interface, and other changes
June 27, 1975	Letter to applicants regarding outstanding items and requesting response to additional staff positions
June 30, 1975	Letter to applicants regarding staff position on environmental qualification of Class IE equipment
July 3, 1975	Letter to applicants concerning ECCS design and analysis
July 3, 1975	Letter to applicants concerning staff position on vibration testing of reactor internals
July 3, 1975	Submittal of Amendment No. 26, consisting of responses to requests for information and updated information
July 9, 1975	Letter from applicants in response to letter of June 27, 1975
July 9, 1975	Letter from applicants transmitting figures inadvertently omitted from May 16, 1975 submittal
July 17, 1975	Meeting with applicants to discuss outstanding items
July 18, 1975	Submittal of Amendment No. 27, consisting of information regarding various design changes
July 22, 1975	Letter to applicants requesting information concerning interface issues
July 25, 1975	Submittal of Amendment No. 28, consisting of information requested by staff and relative to applicants' commitments made at meeting held on July 17, 1975, and other updated information

1547 349

APPENDIX C

ELECTRICAL AND MECHANICAL EQUIPMENT SEISMIC QUALIFICATION PROGRAM

I. Seismic Test for Equipment Operability

1. A test program is required to confirm the functional operability of all seismic Category I electrical and mechanical equipment and instrumentation during and after an earthquake of magnitude up to and including the safe shutdown earthquake. Analysis without testing may be acceptable only if structural integrity alone can assure the design intended function. When a complete seismic testing is impracticable, a combination of test and analysis may be acceptable.
2. The characteristics of the required input motion should be specified by one of the following:
   (a) response spectrum
   (b) power spectral density function
   (c) time history
   Such characteristics, as derived from the structures or systems seismic analysis, should be representative of the input motion at the equipment mounting locations.
3. Equipment should be tested in the operational condition. Operability should be verified during and after the testing.
4. The actual input motion should be characterized in the same manner as the required input motion, and the conservatism in amplitude and frequency content should be demonstrated.
5. Seismic excitation generally has a broad frequency content. Random vibration input motion should be used. However, single frequency input, such as sine beats, may be applicable provided one of the following conditions are met:
   (a) The characteristics of the required input motion indicate that the motion is dominated by one frequency (i.e., by structural filtering effects).
   (b) The anticipated response of the equipment is adequately represented by one mode.
   (c) The input has sufficient intensity and duration to excite all modes to the required magnitude, such that the testing response spectra will envelope the corresponding response spectra of the individual modes.
6. The input motion should be applied to one vertical and one principal (or two orthogonal) horizontal axes simultaneously unless it can be demonstrated that the equipment response along the vertical direction is not sensitive to the vibratory motion along the horizontal direction, and vice versa. The time phasing of the inputs in the vertical and horizontal directions must be such that a purely rectilinear resultant input is avoided. The acceptable alternative is to have vertical and horizontal inputs in-phase, and then repeated with inputs 180 degrees out-of-phase. In addition, the test must be repeated with the equipment rotated 90 degrees horizontally.
7. The fixture design should meet the following requirements:
   (a) Simulate the actual service mounting.
   (b) Cause no dynamic coupling to the test item.
8. The in-situ application of vibratory devices to superimpose the seismic vibratory loadings on the complex active device for operability testing is acceptable when application is justifiable.

1547 350

9. The test program may be based upon selectively testing a representative number of mechanical components according to type, load level, size, etc. on a prototype basis.

II. Seismic Design Adequacy of Supports

1. Analyses or tests should be performed for all supports of electrical and mechanical equipment and instrumentation to ensure their structural capability to withstand seismic excitation.
2. The analytical results must include the following:
   (a) The required input motions to the mounted equipment should be obtained and characterized in the manner as stated in Section I.2.
   (b) The combined stresses of the support structures should be within the limits of ASME Section III, Subsection NF - "Component Support Structures" (draft version) or other comparable stress limits.
3. Supports should be tested with equipment installed. If the equipment is inoperative during the support test, the response at the equipment mounting locations should be monitored and characterized in the manner as stated in Section I.2. In such a case, equipment should be tested separately and the actual input to the equipment should be more conservative in amplitude and frequency content than the monitored response.
4. The requirements of Sections I.2, I.4, I.5, I.6 and I.7 are applicable when tests are conducted on the equipment supports.

APPENDIX D
BIBLIOGRAPHY

NOTE: Documents referenced in or used to prepare this Safety Evaluation Report, excluding those listed in the PSAR, may be obtained at the source stated in the Bibliography or, where no specified source is given, at most major public libraries. Correspondence between the Commission and the applicants 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 applicants and the Commission may also be inspected at the Matagorda County Court House, 1700 Seventh Street, Bay City, Texas. Specific documents relied upon by the Regulatory staff and referenced in this Safety Evaluation are as follows:

Meteorology

1. Cry, G. W., "Tropical Cyclones of the North Atlantic Ocean," Technical Paper No. 55, U. S. Department of Commerce, Weather Bureau, Washington, D. C., 1965.
2. Holzworth, G. C., "Mixing Heights, Wind Speeds, and Potential for Urban Air Pollution Throughout the Contiguous United States," AP-101, Environmental Protection Agency, Office of Air Programs, Research Triangle Park, North Carolina, 1972.
3. Korshover, J., "Climatology of Stagnating Anticyclones East of the Rocky Mountains, 1936-1970," NOAA Technical Memorandum ERL ARL-34, Silver Spring, Maryland, 1971.
4. Sagendorf, J., 1974: A Program for Evaluating Atmoshperic Dispersion from a Nuclear Power Station. NOAA Technical Memorandum ERL ARL-42. From USNRC, Office of Nuclear Reactor Regulation, Washington, D. C., or the Air Resources Laboratory, NOAA, Field Research Office, Idaho Falls, Idaho.
5. SELS Unit Staff, National Severe Storms Forecast Center, "Severe Local Storm Occurrences, 1955-1967," ESSA Technical Memorandum WBTM FCST 12, Office of Meteorological Operations, Silver Spring, Maryland, 1969,
6. Smith, M. E. (ed.), 1968: Recommended Guide for the Prediction of the Dispersion of Airborne Effluents. The American Society of Mechanical Engineers, New York, N. Y.
7. Thom, H. C. S., "Tornado Probabilities," Monthly Weather Review, October-December 1963, pp. 730-737.
8. Thom, H. C. S., "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.
9. U. S. Department of Commerce, Environmental Data Service: "Local Climatological Data, Annual Summary with Comparative Data - Houston, Texas." Published annually through 1972.
10. U. S. Department of Commerce, Environmental Data Service: "Local Climatological Data, Annual Summary with Comparative Data - Galveston, Texas." Published annually through 1972.
11. U. S. Department of Commerce, Environmental Data Service: "Local Climatological Data, Annual Summary with Comparative Data - Victoria, Texas - Published annually through 1972.

1547 352

12. U. S. Department of Commerce, Environmental Data Service: Storm Data, Published monthly, Asheville, N. C.

Geology and Seismology

13. Docekal, J., "Earthquakes of the Stable Interior with Emphasis on the Midcontinent," Ph.D. Dissertation, University of Nebraska, 1970.
14. Eardley, A. J., "Structural Geology of North America," Harper & Row, New York, 1962.
15. Fenneman, N. M., Physical divisions of the United States: U. S. Geological Survey, Washington, D. C. , 1969.
16. Flawn, P. T., "Basement Rocks of the Texas Coast Plain," Gulf Coast Assoc. of Geological Societies, 1964.
17. Gulf Coast Assoc. Geol. Socs. and Am. Assoc. Petroleum Geologists, 1972, Tectonic Map of the Gulf Coast Region, U.S.A.
18. Gupta, I. N. and Nuttli, O. W., "Spatial Attenuation of Intensities for Central U. S. Earthquakes," (Abstract) Transactions, American Geophysical Union, 1975.
19. Hendron, A. J., 1975, Monitoring for differential subsidence at the Allens Creek Site, Allens Creek Nuclear Generating Station Safety Evaluation Report.
20. King, P. B. and Beikman, H. M., U. S. Geol. Survey, Geologic Map of the United States: Washington, D. C., 1974.
21. King, P. B., "The Tectonics of North America - A Discussion to Accompany the Tectonic Map of North America Scale 1:5,000,000," Geological Survey Professional Paper 628, Washington, D. C., 1969.
22. Neumann, F., "Earthquake Intensity and Related Ground Motion," University Press, Seattle, 1954.
23. Nuttli, O. W., "The Mississippi Valley Earthquakes of 1811 and 1812: Intensities, Ground Motion and Magnitudes," Bulletin of the Seismological Society of America, 1973.
24. Nuttli, O. W., and Zollweg, J. E., "The Relation Between Felt Area and Magnitude for Central United States Earthquakes," Bulletin of the Seismological Society of America, 1974.
25. Trifunac, M. D., and Brady, A. G., "On the Correlation of Seismic Intensity Scales with Peaks of Recorded Strong Ground Motion," Bulletin of the Seismological Society of America, 1975.

Structural Engineering

26. "Wind Forces on Structures," Final Report of the Task Committee on Wind Forces of the Committee on Loads and Stresses of the Structural Division, Transactions of the American Society of Civil Engineers, Paper No. 3269, Vol. 126, Part II, 1961, p. 1124-1198, 345 East 47th Street, New York, N. Y., 10017.
27. Gwaltney, R. C., "Missile Generation and Protection in Light Water-Cooled Power Reactor Plants," USAEC Report ORNL-NSIC-22, Oak Ridge, Tennessee, September 1968.
28. Williamson, R. A., and Alvy, R. R., "Impact Effect of Fragments Striking Structural Elements," Holmes and Narver, Revised Edition, 1973.

Engineered Safety Features

29. Allen, A. O., "The Radiation Chemistry of Water and Aqueous Solutions," Van Nostrand Company, 1961.
30. Coward, H. F., and Jones, G. W., "Limits of Flammability of Gases and Vapors," Bureau of Mines Bulletin 503, 1952.

1547 353

31. Moody, F. J., "Maximum Flow Rate of a Single Component, Two-Phase Mixture," Vol. 87, p. 134, Journal of Heat Transfer, February 1965.

32. Parsly, L. F., "Design Considerations of Reactor Containment Spray Systems Part VI," The Heating of Spray Drops in Air-Steam Atmospheres, USAEC Report ORNL-TM-2412, Oak Ridge, Tennessee, January 1970.

33. Retting, W. H., Jayne, G. A., Moore, K. V., Slater, C. E., and Uptmer, M. L., "RELAP3 - A Computer Program for Reactor Blowdown Analysis," In-1321 Idaho Nuclear Corporation, June 1970.

34. Richardson, L. C., Finnegan, L. J., Wagner, R. J., and Waage, J. M., "CONTEMPT - A Computer Program for Predicting the Containment Pressure Temperature Response to a Loss-of-Coolant Accident," IDO-17220, Phillips Petroleum Company, June 1967.

35. Schmitt, R. C., Bingham, G. E., Norberg, J. A., "Simulated Design Basis Accident Tests of the Carolina Virginia Tube Reactor Containment," Final Report, IN-1403, Idaho Nuclear Corporation, December 1970.

36. Shepard, R. M, Massie, H. W., Mark, R. H., and Docherty, P. J., "Westinghouse Mass and Energy Release Data for Containment Design," Topical Report, WCAP 8312, Westinghouse Electric Corporation, Pittsburgh, Pa., March 1974.

37. Slaughterbeck, D. C., "Comparison of Analytical Techniques Used to Determine Distribution of Mass and Energy in the Liquid and Vapor Regions of a PWR Containment Following a Loss-of-Coolant Accident," Special Interim Report, Idaho Nuclear Corporation, January 1970.

38. Slaughterbeck, D. C., "Review of Heat Transfer Coefficients for Condensing Steam in a Containment Building Following a Loss-of-Coolant Accident," IN-1388, Idaho Nuclear Corporation, September 1970.

39. Tagami, T., "Interim Report on Safety Assessments and Facilities Establishment Project in Japan for Period Ending June 1965 (No. 1)," Prepared for the National Reactor Testing Station, February 28, 1966 (Unpublished work).

40. Uchida, H., Oyama, A., and Toga, Y., "Evaluation of Post-Incident Cooling Systems of Light-Water Power Reactors," in Proceeding of the Third International Conference on the Peaceful Uses of Atomic Energy Held in Geneva August 31 - September 9, 1964, Volume 13, Session 3.9, New York: United Nations 1965, (A/Conf. 28/P/436) (May 1964) pp. 93-104.

41. Wagner, R. J., and Wheat, L. L., "CONTEMPT-LT Users Manual," Interim Report I-214-74-12.1, Aerojet Nuclear, August 1973.

Effluent Treatment

42. "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.