Safety Evaluation Report

NUREG-0104

U. S. Nuclear Regulatory Commission

related to the preliminary design of the

Office of Nuclear Reactor Regulation

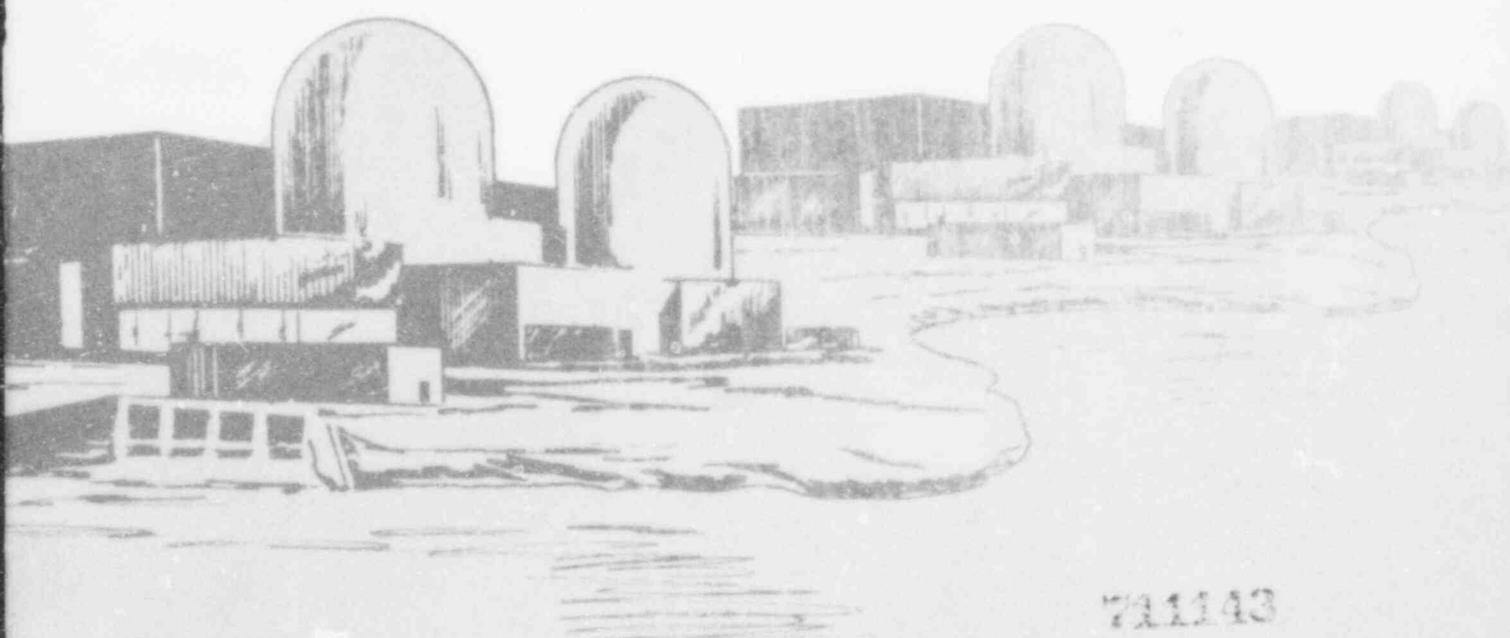
Standard Reference System RESAR - 3S

Docket No. STN 50-545

Westinghouse Electric Corporation

December 1976

7906040349



711143

POOR ORIGINAL

~~717~~ 311

Available from
National Technical Information Service
Springfield, Virginia 22161
Price: Printed Copy $7.50 ; Microfiche $3.00

~~717 312~~

NUREG-0104

December 30, 1976

SAFETY EVALUATION REPORT

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

U.S. NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

WESTINGHOUSE ELECTRIC CORPORATION

REFERENCE SAFETY ANALYSIS REPORT

RESAR-3S

DOCKET NO. STN 50-545

711145

~~717 313~~

TABLE OF CONTENTS

711146

717 314

TABLE OF CONTENTS (Continued)

711147

~~717~~ 315

TABLE OF CONTENTS (Continued)

711149 ~~717 317~~

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

PAGE

711151

~~717~~ 319

TABLE OF CONTENTS (Continued)

711152

~~717 320~~

LIST OF TABLES

711153

~~717~~ 321

LIST OF FIGURES

~~717~~ 322

1.0 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

The Westinghouse Electric Corporation (hereinafter referred to as Westinghouse) tendered on June 30, 1975, with the United States Nuclear Regulatory Commission (hereinafter referred to as the Commission) a proposed preliminary standard or reference system design, designated as the RESAR-3S design, for a nuclear steam supply system. The submittal was in the form of an application for a Preliminary Design Approval by the Commission and was in response to Option 1 of the Commission's standardization policy, WASH-1341, "Programmatic Information for the Licensing of Standardized Nuclear Plants." Option 1 allows for the review of a "reference system" that involves an entire facility design or major fraction of a facility design outside the context of a license application. The application was docketed on July 31, 1975.

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. The Commission's standardization policy, WASH-1341, was issued on August 20, 1974. Amendment 1 to WASH-1341, dealing with "options" and "overlaps," was issued on January 16, 1975. The regulations governing the submittal and review of standard designs under the "reference system" option are found in Appendix O to Part 50 and Section 2.110 of Part 2 of Title 10 of the Code of Federal Regulations (hereinafter referred to as 10 CFR).

A standard safety analysis report in the form of a Westinghouse Reference Safety Analysis Report, RESAR-3S, was submitted with the application. The information in RESAR-3S has been supplemented by Amendments 1 through 13. RESAR-3S and these amendments are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. 20555.

This safety evaluation report summarizes the results of the technical evaluation of the proposed RESAR-3S design performed by the Commission's staff, delineates the scope of the technical matters considered in evaluating the radiological safety aspects of the RESAR-3S design, addresses the comments made by the Advisory Committee on Reactor Safeguards in its report of July 14, 1976, and addresses the resolution of the outstanding issues previously identified during our review. Environmental aspects were not considered in our review of RESAR-3S, but will be addressed in each utility application for a construction permit which references RESAR-3S.

Based on our evaluation of the proposed RESAR-3S nuclear steam supply system design, we conclude that (1) a Preliminary Design Approval for the proposed design can be granted, (2) the proposed design can be incorporated by reference in construction permit and standard balance-of-plant design applications, and (3) a nuclear steam supply system utilizing the proposed design can be constructed without endangering the health and safety of the public. Our detailed conclusions are presented in Section 19 of this report.

Utility applicants referencing RESAR-3S in the future will retain architect-engineers, constructors, turbine-generator vendors, and consultants as needed. Prior to a decision for issuance of a construction permit, we will evaluate for each utility application which references RESAR-3S the technical competence of the applicant and its contractors to manage, design, construct, and operate a nuclear power plant.

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 proposed RESAR-3S nuclear steam supply system. Prior to a decision for issuance of an operating license for any application which references RESAR-3S, we will review the final design of the proposed RESAR-3S nuclear steam supply system 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.

Amendment 1 to WASH-1341 states that utility applications for construction permits tendered on or after January 1, 1976 that reference a standard design should only reference those portions of standard design safety analysis reports consistent with the standardized scope as defined in Amendment 1 to WASH-1341. The WASH-1341 document further states that a standard design applicant wishing to standardize items not within the standardized scope, may do so by submitting the necessary information for staff review under the topical report program. After January 1, 1976, non-standardized scope items may be referenced by utility applications only as part of the topical report program.

RESAR-3S originally contained descriptions of certain "optional" systems that were not within the standardized scope of Amendment 1 to WASH-1341. At our request and in accordance with the provisions of Amendment 1 to WASH-1341, Westinghouse deleted from RESAR-3S the descriptions of these systems. In accordance with the provisions of WASH-1341, Westinghouse subsequently submitted for our review the descriptions of these systems in the form of topical reports.

Westinghouse also offers several of its systems and components which are in the standardized scope of Amendment 1 to WASH-1341 as options to its customers. We have reviewed these systems, have identified them as being options, and have presented our evaluation of these systems in the appropriate sections of this report.

711156

~~717~~ 324

In the course of our safety review of the material submitted, we held numerous meetings with representatives of Westinghouse to discuss the proposed nuclear steam supply system 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-3S.

As a result of our review, a number of changes were made in the proposed nuclear steam supply system design. These changes are described in the amendments and are discussed in the appropriate sections of this report.

A chronology of the principal actions relating to the processing of the application is included as Appendix A to this report. A bibliography for this report is included as Appendix B. A copy of the Advisory Committee on Reactor Safeguards' report on RESAR-3S is included as Appendix C.

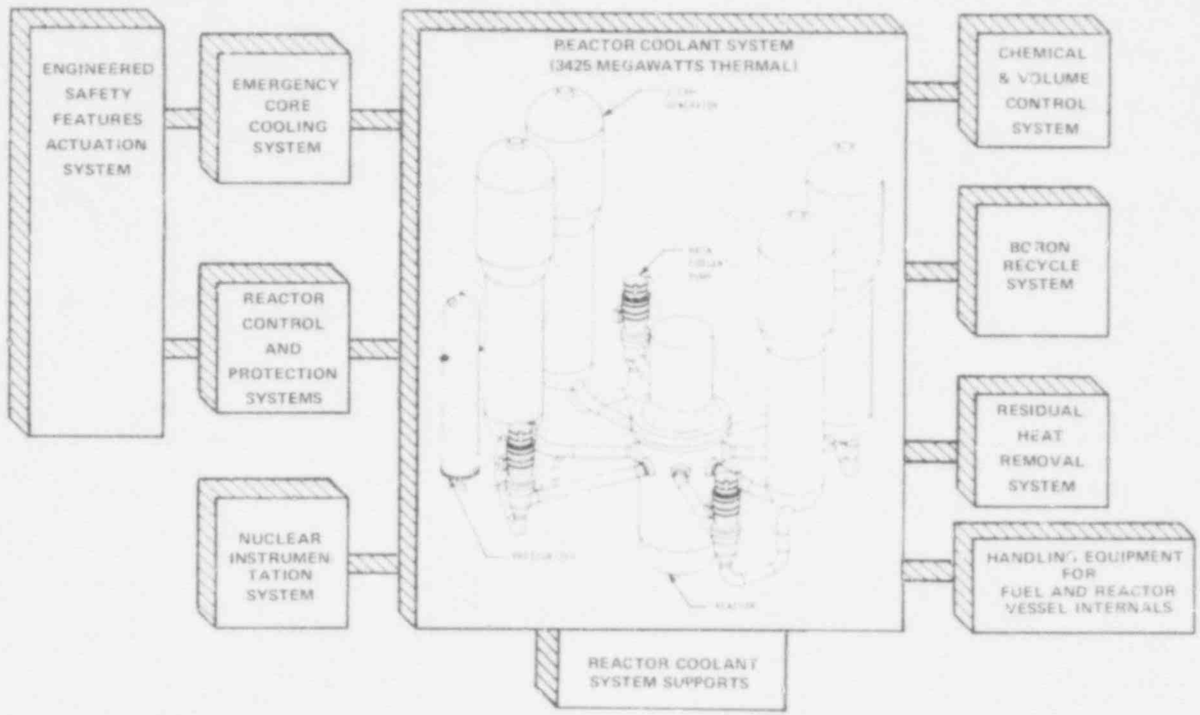
Since a standard nuclear steam supply system does not include the entire facility, it is necessary to specifically and extensively describe the safety-related interfaces between the nuclear steam supply system and the balance of plant. Interface information addresses the pertinent safety-related design requirements including the operating environment; inputs to transient and accident analyses; and the layout, structural, and performance requirements necessary to assure the compatibility of the nuclear steam supply system with its mating portion of the plant and site. This information has been included in RESAR-3S and our evaluation of the interface information in RESAR-3S is contained in Section 1.7 of this report.

1.2 General Description

The proposed RESAR-3S standard design will consist of a nuclear steam supply system with a thermal power rating of 3425 megawatts, which includes a core thermal power of 3411 megawatts plus 14 megawatts from pump heat, and a four loop reactor coolant system. The proposed reactor coolant system will be housed in a containment building which is not within the scope of RESAR-3S and which will be designed by applicants that utilize RESAR-3S. The scope of RESAR-3S will include many of those components and systems which are directly related with the normal operation and emergency shutdown of the reactor, and will include a standard system of integral supports for the reactor coolant system. Figure 1-1 graphically summarizes the design scope of RESAR-3S. A listing of the major components and systems within the scope of RESAR-3S is presented in Table 1-1 of this report. A more detailed listing is presented in Table 1.7-1 of RESAR-3S.

Not included in the RESAR-3S scope are the conventional balance-of-plant features such as the auxiliary service facilities and the general service facilities. Such facilities include plant buildings and structures, the ultimate heat sink, onsite and offsite electrical systems, the main steam system exclusive of the steam generators, and the turbine-generator and its auxiliaries. However, the RESAR-3S scope does

711157 ~~717 325~~



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

Figure 1-1 RESAR - 3S NUCLEAR STEAM SUPPLY SYSTEM

POOR ORIGINAL

711158

~~717 326~~

TABLE 1-1

COMPONENTS AND SYSTEMS WITHIN THE SCOPE OF RESAR-35

Reactor Core

Reactor Coolant System

- Reactor Vessel
- Reactor Vessel Internals
- Rod Cluster Controls
- Burnable Poison Rods
- Control Rod Guide Thimble Plugs
- Encapsulated Primary Neutron Sources
- Encapsulated Secondary Neutron Sources
- Control Rod Drive Mechanisms
- Steam Generators
- Reactor Coolant Pumps
- Pressurizer
- Pressurizer Relief Tank

Chemical and Volume Control System

- Pumps with Motors
- Heat Transfer Equipment
- Tanks and Demineralizers
- Filters and Orifices

Boron Concentration Meter

Residual Heat Removal System

- Residual Heat Removal Pumps
- Residual Heat Exchangers

Emergency Core Cooling System

- Safety Injection Pumps with Motors
- Accumulators

Boron Recycle System

- Pumps with Motors
- Recycle Evaporator
- Tanks and Demineralizers with Resin
- Filters

Nuclear Equipment Supports

- Reactor Vessel Supports
- Steam Generator Supports
- Reactor Coolant Pump Supports
- Pressurizer Support Ring
- Reactor Coolant Loop Pipe Whip Restraints

TABLE 1-1 (Continued)

Piping

- Reactor Coolant System

Valves, Including Operators

- Reactor Coolant System
- Chemical and Volume Control System
- Residual Heat Removal System
- Emergency Core Cooling System
- Boron Recycle System

Handling Equipment for Fuel and Reactor Vessel Internals

- Refueling and Fuel Handling Machines
- Fuel and Rod Cluster Control Handling Equipment
- New Fuel Storage Racks
- Spent Fuel Storage Racks

Instrumentation and Control

- Main Control Room Panel Board
- Board and Panel Mounted Instruments
- Auxiliary Control Panels
- Reactor Control and Protection System Reactor Process Control
- Reactor Protection Logic
- Engineered Safeguards Initiation and Actuation Circuits Testing System
- Rod Control System
- Rod Position Indication System
- Steam Dump Control System
- Feedwater Control System
- Nuclear Instrumentation System
- Process Instrumentation and Controls
- Containment Pressure Sensors
- Leading Edge Flowmeter Measurement System
- Incore Instrumentation

Loose Parts Monitor

Plant Electrical Equipment

- Rod Drive Power Supply System
- Pressurizer Heater Variable Power Controller
- Vital Instrument Alternating Current Power Supply for Nuclear Steam Supply System Instrumentation and Control
- Motors for Westinghouse-Supplied Equipment

711160

~~717 328~~

include the delineation of interface information pertaining to the compatibility with RESAR-3S of those balance-of-plant features that have a direct bearing on the integrity or on the functional capability of the systems within the RESAR-3S scope.

For those systems that are within the scope of RESAR-3S, the detailed piping design and layout are provided only for the reactor coolant system. For all other systems, RESAR-3S includes the necessary interface information for the balance-of-plant applicant to design the piping and system layout.

The RESAR-3S nuclear steam supply system is designed such that system and components within the nuclear steam supply system that are important to safety will not be shared between units at multi-unit stations.

1.2.1 Reactor System

The proposed pressurized water reactor system will include the reactor vessel and nozzles, integral supports, reactor vessel head assembly, the reactor core, and all internal appurtenances required to support the reactor core. This is all within the scope of RESAR-3S.

The proposed RESAR-3S 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 will consist of slightly enriched uranium-dioxide powder that will be compacted by cold pressing and then sintered to the desired density. All fuel rods will be internally pressurized with helium during the welding process. The design height of the fuel pellets within each rod is 143.7 inches, while the overall fuel rod length will be 151.6 inches.

The RESAR-3S fuel rods will be combined in a 17 x 17 array to form fuel assemblies. These fuel assemblies will have eight spacer grids and contain guide thimble channels for the neutron absorber rods, burnable poison rods, or neutron source assemblies. The RESAR-3S core will be formed of 193 fuel assemblies. All fuel rods within a given fuel assembly will have the same uranium enrichment. Fuel assemblies of three different uranium enrichments - approximately 2.10, 2.60, and 3.10 weight percent uranium-235 - are proposed to be used in the initial core loading to establish a favorable radial power distribution. Two regions consisting of the two lower uranium enrichments will be interspersed so as to form a checkerboard pattern in the central portion of the core. The third region, which will be arranged around the periphery of the core, will contain the highest uranium enrichment. The fuel reloading pattern will be typically similar to that of the initial core with depleted fuel interspersed checkerboard style in the center and new fuel on the periphery. The core will normally operate approximately one year between refuelings. Approximately one-third of the core will be replaced at each refueling.

711161 ~~717 329~~

1.2.2 Reactor Control System

The RESAR-3S reactor control system will include full and part length rod cluster control assemblies, burnable poison rods, and the capability for regulation of the boric acid concentration in the reactor coolant. The burnable poison rods will normally only be used for the initial core of any RESAR-3S plant because of the first core's higher reactivity. The mechanical control assemblies will consist of clusters of stainless steel clad, silver-indium-cadmium alloy absorber rods which will be inserted in guide tubes in the fuel assemblies. There will be two categories of full length control rod assemblies - control assemblies and shutdown assemblies. Control assemblies are designed to compensate for reactivity changes due to variations in operating conditions of the reactor, and the shutdown assemblies are designed to have the necessary negative reactivity to provide an adequate shutdown margin. The RESAR-3S control system for the full length control assemblies is designed to accommodate plant step load changes of ten percent and ramp changes of five percent per minute over the range of 15 to 100 percent of full power under normal operating conditions. The part length control assemblies are designed to control axial neutron flux shape and axial xenon oscillations. The soluble boric acid neutron absorber will be varied to control long term reactivity changes resulting from fuel depletion and fission product buildup, cold to hot zero power reactivity change, reactivity changes produced by intermediate-term fission products such as xenon and samarium, and burnable poison depletion.

1.2.3 Reactor Coolant System

The RESAR-3S reactor coolant system will include the reactor; four coolant loops, each with a steam generator and a reactor coolant pump; a pressurizer with associated relief and safety valves and a pressurizer relief tank; and two residual heat removal trains. Significant parameters for the reactor coolant system are listed below. This coolant system design does not include loop stop valves.

Normal Operating Pressure, pounds per square inch, absolute	2250
Reactor Power, megawatts, thermal	3411
Reactor Vessel Inlet Temperature, degrees Fahrenheit	558.1
Reactor Vessel Outlet Temperature, degrees Fahrenheit	618.3
Total Reactor Flow Rate, pounds per hour	140,300,000
Steam Pressure, pounds per square inch, absolute	1,000
Total Steam Flow, pounds per hour	15,140,000

The reactor coolant pumps are designed to circulate reactor coolant through the core and steam generators in order to transfer the thermal output of the core to the secondary side of the steam generators. An electrically heated pressurizer will be connected to the suction side of the reactor coolant pump in one of the coolant loops. The pressurizer is designed to establish and maintain the reactor coolant pressure. Two air operated relief valves and three self-actuated safety valves will be connected to the pressurizer vapor space and their discharge will be to the pressurizer relief tank.

~~717~~ 330

711162

The two residual heat removal trains will each consist of a heat exchanger and a circulation pump. The pumps' suctions will be connected to the hot legs of two reactor coolant loops and their discharges will be connected to the safety injection pump discharge lines. The residual heat removal system will be used to remove decay heat during normal plant cooldown and shutdown.

1.2.4 Engineered Safety Features

The engineered safety features within the scope of RESAR-3S include the major portions of the emergency core cooling system and those portions of the containment isolation system relating to the systems within the scope of RESAR-3S. This system is designed to provide core cooling and protection for the complete range of postulated primary and secondary coolant pipe break sizes, which are evaluated in Section 15.5 of this report.

Those portions of the emergency core cooling system within the scope of RESAR-3S include four accumulator tanks connected to four cold leg safety injection lines and two high pressure and one low pressure injection systems with provisions for recirculating the borated coolant at the end of the injection phase. Each of the three systems will be able to take its suction from either the refueling water storage tank or the containment sumps, which are outside the scope of RESAR-3S. The emergency core cooling system is designed to provide core cooling in the event of either large or small ruptures of the reactor coolant system.

The boric acid injection portion of the emergency core cooling system, which is part of one of the high pressure injection systems, will consist of the boron injection tank, boron injection surge tank, boron injection recirculation loop, charging pumps, and the associated valves. The boron injection tank will be connected to the reactor coolant system by means of a loop from the refueling water storage tank, through the charging pumps, to the boron injection tank inlet. The boron injection tank outlet will be connected through a common manifold pipe to pipes connected to each of the four reactor coolant cold leg lines.

The boric acid injection portion of the emergency core cooling system is designed to provide sufficient shutdown capability in the event of any single steam pipe rupture or spurious lifting of a pressure relief valve.

1.2.5 Protection Systems

The plant protection systems within the scope of RESAR-3S include the reactor trip system and the engineered safety features actuation system.

The reactor trip system will consist of sensors connected with analog circuitry consisting of two to four redundant channels, designed to monitor various plant

711163

~~717 331~~

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 full length control rod assemblies into the core and shut the reactor down.

The engineered safety features actuation system will consist of instrumentation and controls to sense accident situations and initiate operation of the necessary engineered safety features. The system will consist of (1) an analog portion employing three to four redundant channels per plant parameter being monitored and (2) a digital portion utilizing two redundant logic trains which will receive inputs from the analog protection channels and actuate the engineered safety features.

The functions initiated by the engineered safety features actuation system include the following:

(1) Reactor trip
(2) Safety injection
(3) Containment cooling
(4) Auxiliary feedwater flow
(5) Containment isolation
(6) Steam line isolation
(7) Main feedwater line isolation
(8) Emergency diesel operation
(9) Control room isolation
(10) Containment spray actuation

1.2.6 Chemical and Volume Control System

The RESAR-3S chemical and volume control system will include two centrifugal and one positive displacement charging pumps; the volume control and chemical mixing tanks; the mixed and cation bed demineralizers, the reactor coolant filters, various heat exchangers, and the reactor coolant purification pump; the boron thermal regeneration subsystem; and various other components described in Sections 1.7 and 9.3 of RESAR-3S. The system will be located outside of the containment building.

The chemical and volume control system will be connected to various systems including the reactor coolant system, the waste processing systems, and the reactor makeup water system. The system is designed to control and maintain the reactor coolant inventory and to control the boron concentration in the reactor coolant.

1.2.7 Power Sources

Westinghouse has identified in RESAR-3S as an interface requirement that a minimum of two independent emergency onsite power supplies must be provided, each of which must be able to 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-3S.

1.3 Comparison with Similar Designs

The RESAR-3S nuclear steam supply system proposed design is essentially the same as that of plants utilizing the Westinghouse RESAR-3 Consolidated Version nuclear steam supply system design such as the Standardized Nuclear Unit Power Plant System (SNUPPS) plants which include Wolf Creek (Docket Number STN 50-482), Callaway (Docket Numbers STN 50-483 and STN 50-486), Tyrone Energy Park (Docket Number STN 50-484), and Sterling (Docket Number STN 50-485). To the extent feasible and appropriate, we have made use of our previous evaluations of those features that are similar to the RESAR-3S design. Where this has been done, the appropriate sections of this report identify the other facilities involved. Our safety evaluation reports for these other facilities are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. 20555.

1.4 Requirements for Future Technical Information

Westinghouse has identified in Section 1.5 of RESAR-3S the verification test programs applicable to the RESAR-3S 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 that we have determined to be necessary to verify the RESAR-3S design and their objectives is contained in Table 1-2 of this report.

All the verification test programs listed in Table 1-2 have been completed; however, we have not completed our review of the results of the programs. These test programs are discussed further in Section 4.0 of this report.

Based on the staff's review of the verification programs, we have concluded that (1) the test programs outlined in RESAR-3S will provide the necessary information to verify the RESAR-3S nuclear steam supply system design 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.

In addition, we have listed in Table 1-3 items discussed in this report which will require the submittal of additional technical information prior to approval of the final design. Also indicated in Table 1-3 are references to the sections in this report in which each of the items are discussed. We have determined that this information is of the type that in accordance with the provisions of Section 50.35 of 10 CFR Part 50 can be left for later consideration.

1.5 Summary of Principal Review Matters

Our technical review and evaluation of the information submitted by Westinghouse included the principal review matters summarized below.

~~717 333~~

711165

TABLE 1-2

VERIFICATION TEST PROGRAMS

Test	Objective
I. Verification Tests	
Rod Cluster Control Spider Tests	Verify structural adequacy
Grid Tests	Verify structural adequacy
Departure from Nucleate Boiling	Determine effect of 17 x 17 geometry on departure from nucleate boiling heat flux
Single Rod Burst Test	Determine maximum flow blockage
Fuel Assembly Structural Test	Determine mechanical strength of assembly
Prototype Assembly Tests	Demonstrate performance of 17 x 17 fuel assembly
II. Loss-of-Coolant Accident Heat Transfer Tests	
G-Loop Tests	Simulate blowdown in fuel assembly

~~717 334~~

711166

TABLE 1-3

ITEMS REQUIRING ADDITIONAL TECHNICAL INFORMATION

Item	Section(s) in this Report in Which Discussed
. Seismic and environmental qualification of Class IE equipment	3.7 7.6.1
. Fuel surveillance	4.2.1.3 4.2.1.4 4.4.1
. Fuel rod bowing	4.2.1.3 4.4.1 6.3.4
. Hydraulic load on fuel assemblies during a postulated loss-of-coolant accident	4.2.1.3
. Dynamic analysis of reactor internals and piping loops	4.2.2
. Use of part length control rods	4.3.1
. Departure from nucleate boiling analysis	4.4.1
. Submittal and review of HYDNA computer code	4.4.1
. Verification of THINC computer code	4.4.1
. Reactor coolant pump overspeed	5.4.1.2
. Changeover from injection to recirculation mode	7.3.2
. Sensor response time testing	7.3.3
. Electrical grid decay rate verification	8.1
. Waterhammer effects	10.2
. Justification of trip delay times	15.2.1
. Verification of control rod insertion times	15.2.1
. Loss of normal feedwater analysis	15.2.1
. Completion of review of computer codes used in accident analyses	15.2.2
. Submittal and review of steamline and feedwater line break accident analyses	15.5.2 15.5.4
. Submittal and review of loss of flow transient	15.3
. Anticipated transients without scram	15.5.7
. Iodine spiking	15.7.1

711167

~~717 335~~

We evaluated the design and expected performance characteristics of the systems and components important to safety to determine whether they are in accord with the 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. Of course, the acceptability of particular sites will be determined during the course of our review of utility applications for construction permits which reference RESAR-3S.

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-3S nuclear steam supply system design is combined with an acceptable balance-of-plant design.

1.6 Resolution of Outstanding Issues

In our report to the Advisory Committee on Reactor Safeguards on RESAR-3S and at the Advisory Committee on Reactor Safeguards meetings on RESAR-3S, we had identified certain outstanding issues which required that Westinghouse provide additional information to confirm that the proposed design would meet our requirements or where our review was not yet complete. We have resolved all of these issues in a manner acceptable for issuance of a Preliminary Design Approval. These items are discussed in the applicable sections of this report.

We are presently considering on a generic basis the question of whether capability should be provided for transferring heat from the reactor to the environment from normal reactor operating conditions to cold shutdown using only safety-grade systems. If we determine that this capability should be provided, we will require that the RESAR-3S design and the designs of the balance-of-plant portions of applications referencing RESAR-3S be modified accordingly. We have determined that such modifications are technically feasible and conclude that this matter can be left for post-preliminary design approval stage consideration.

1.7 Interface Information

Interface information must be specified to assure that components and systems within a standard design will perform their intended safety functions. Interface information, therefore, is utilized to provide a basis for assuring that the safety-related aspects of the matching portions of a nuclear plant design are compatible.

711168

~~717 336~~

In various sections of this report we have discussed interface information as we determined that it was appropriate. Westinghouse has provided interface information for its design throughout the various sections of RESAR-3S and in particular Section 1.7.

We have reviewed the interface information provided by Westinghouse and have determined that this information is sufficient to determine the compatibility of the safety-related systems and components within the scope of RESAR-3S with the balance-of-plant design to be submitted in applications referencing RESAR-3S. The interface information provided in RESAR-3S is also adequate to determine the validity of the RESAR-3S accident analyses when RESAR-3S is referenced by a balance-of-plant design application. We, therefore, conclude that the RESAR-3S interface information is acceptable.

711169

~~747~~ 337

2.0 SITE CHARACTERISTICS

Appendix 0 to 10 CFR Part 50 requires that standard design applications include the site parameters postulated for the design, and an analysis and evaluation of the design in terms of these postulated site parameters.

Since RESAR-3S describes a proposed standard nuclear steam supply system, the only site-related parameter which is directly related to its design is the seismic input which is discussed in Section 3.5.1 of this report. Protection of the RESAR-3S systems and components important to safety from all design basis site-related parameters and phenomena, such as tornadoes and floods, will be provided by buildings, structures, and systems which are outside the scope of RESAR-3S. Westinghouse has identified the safety-related equipment within the scope of RESAR-3S and required as an interface that applications utilizing RESAR-3S provide adequate protection for this equipment from site-related phenomena.

We have reviewed the identification of safety-re`.ted equipment provided in RESAR-3S and the site-related phenomena for which protection must be provided and conclude that they are acceptable.

711170

~~747~~ 338

3.0 DESIGN CRITERIA FOR SYSTEMS AND COMPONENTS

3.1 Conformance with the General Design Criteria

Westinghouse states that the RESAR-3S nuclear steam supply system will be designed 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 statement, we conclude that the RESAR-3S nuclear steam supply system can be designed to meet the requirements of the General Design Criteria applicable to the nuclear steam supply system. Discussions regarding compliance with each applicable criterion are presented in Section 3.1 of RESAR-3S.

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 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 within the scope of RESAR-3S.

Westinghouse has used 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, "Quality Group Classifications and Standards." Westinghouse has applied this classification system 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 to (1) prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary, (2) shutdown the reactor and maintain it in a safe shutdown condition, and (3) contain radioactive material.

RESAR-3S fluid systems pressure-retaining components important to safety that are classified Quality Group A, B, or C will be constructed to the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (hereinafter referred to as the ASME Code) as follows:

711171

717 339

Quality Group	Component Code Section III, Division 1, 1974 Edition
A	Class 1
B	Class 2
C	Class 3

Quality Group A components will comply with Section 50.55a of 10 CFR Part 50. Quality Groups B and C components will comply with Subsection NA-1140 of e ASME Code.

Components that are classified Quality Group D will be constructed to Divisions 1 or 2 of Section VIII of the ASME Code or to American National Standards Institute Standard B31.1-1973, as appropriate. Quality Group D components such as orifices, boron meters, strainers, and gas traps will not be constructed to any code.

Seismic Category I systems and components are identified in Section 3.2.1 of RESAR-3S. The purification loop of the chemical and volume control system, the boron thermal regeneration subsystem, and the boron recycle system are non-seismic Category I portions of systems within the scope of RESAR-3S that also perform a safety function but, because of their limited radioactivity content and location within seismic Category I structures, need not be designed to seismic Category I requirements. These fluid systems and those fluid systems identified in Section 3.2.2 of this report have been classified in an acceptable manner in conformance with Regulatory Guide 1.26 in Table 3.2-1 and on system piping and instrumentation diagrams in RESAR-3S. As noted in Section 3.2.2, excluded from this review are those structures and balance-of-plant fluid systems that interface with RESAR-3S fluid systems.

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, for fluid system pressure-retaining components important to safety within the scope of RESAR-3S, the system quality group classification with regard to system and component design, fabrication, erection, and testing conforms to the quality standard requirements cited above and, therefore, is acceptable.

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

711172

~~717~~ 340

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.

With respect to these safety-related Westinghouse fluid systems and components, including their supports, within the scope of RESAR-3S, we have reviewed the seismic classification to determine that those systems and components designed to withstand, without loss of function, the effects of a safe shutdown earthquake, have been classified as seismic Category I items. These seismic Category I fluid systems and components are (1) the reactor coolant system, (2) the emergency core cooling system, (3) the residual heat removal system, (4) the chemical and volume control system, and (5) portions of the fuel handling system. Excluded from this review are structures and balance-of-plant fluid systems that interface with RESAR-3S fluid systems and those portions of the RESAR-3S systems which are within the balance-of-plant scope. These are identified in Table 1.7-1 and the appropriate sections of RESAR-3S. The safety and seismic classification of these structures, systems, and components will be reviewed for each application which references RESAR-3S.

Systems and components important to safety that will be designed to withstand the effects of a safe shutdown earthquake and remain functional have been identified in an acceptable manner and classified as seismic Category I items in conformance with Regulatory Guide 1.29, "Seismic Design Classification," in Table 3.2-1 of RESAR-3S. 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 seismic Category I systems which will not be required to perform a safety function.

We conclude that, for systems and components important to safety within the scope of RESAR-3S, the seismic classification conforms to seismic Category I requirements and, therefore, is acceptable. The basis for acceptance is the conformance of the Westinghouse designs, 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 generated both from within the containment (internally generated missiles) and externally. The responsibility for protection of safety-related systems and components is not within the scope of RESAR-3S. Therefore, our review was limited to identifying (1) the sources of internally generated missiles and (2) the systems and components to be protected from missiles as interface

~~717~~ 341

711173

requirements. RESAR-3S does not include the design analysis and criteria used for 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 potential. Similarly, we did not include tornado-generated missiles in our review of RESAR-3S. Therefore, applicants referencing RESAR-3S must consider the effects of postulated missiles and provide the necessary protection to all safety-related components.

We have reviewed the RESAR-3S 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 of equipment and systems within the scope of RESAR-3S.

Section 3.5 of RESAR-3S describes the characteristics of postulated missiles which may occur inside the containment from failure of equipment within the scope of the RESAR-3S nuclear steam supply system. These missiles include control rod drive mechanism missiles, valve bonnet missiles, piping temperature sensing element assembly missiles, reactor coolant pump temperature sensing element missiles, pressurizer instrument well missiles, and pressurizer heater missiles. The characteristics of these postulated missiles are identified as interface information to be used by balance-of-plant designers in providing adequate missile protection.

We conclude that, for systems and components important to safety within the scope of RESAR-3S, the identification of RESAR-3S equipment to be protected from missiles and the description of the postulated missiles generated from RESAR-3S equipment conform 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 be appropriately protected against the dynamic effects from the postulated rupture of piping.

We reviewed RESAR-3S to determine that the design will accommodate the effects of postulated pipe breaks and jet impingement forces from postulated piping system ruptures. Westinghouse states that the criteria to be employed for determination of the systems to be evaluated, the location 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 is 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, "Protection Against Pipe Whip Inside Containment."

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. Because the detailed design of the reactor coolant system component supports may vary in actual plants incorporating the RESAR-3S nuclear steam supply system, we will require that each applicant referencing RESAR-3S supplement the information provided in RESAR-3S 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 component support designs lie within the design envelope of WCAP-8082.

RESAR-3S covers only the pipe break criteria for the reactor coolant loop piping. Pipe break criteria for other high energy and moderate energy lines inside and outside the containment will be included in each plant application referencing RESAR-3S.

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

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

(2) The emergency core cooling system can be expected to perform its intended function.

(3) Systems and components important to safety will be appropriately protected.

Westinghouse has provided as interface information the pressures and temperatures of the fluids in systems within the RESAR-3S scope. In addition, they have identified the RESAR-3S 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 for applications which reference RESAR-3S.

On the basis of our review, we conclude that the criteria that will be used for the identification, design, and analysis of reactor coolant loop piping where postulated breaks may occur constitute an acceptable basis for meeting the applicable requirements of Criteria 1, 2, 4, 14, and 15 of the General Design Criteria. ~~717 343~~

711175

3.5 Seismic Design

Criterion 2 of the General Design Criteria requires that systems and components important to safety be designed to withstand the effects of earthquakes. We reviewed the RESAR-3S 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 seismic Category I systems and components within the scope of RESAR-3S.

The RESAR-3S nuclear steam supply system will be designed to withstand a maximum horizontal ground acceleration of 0.4 times the normal gravitational acceleration at zero period with ground response spectra as specified in Figure 3.7-1 of RESAR-3S. In addition, the building design must meet the following restrictions:

(1) The reactor building structure shall have maximum zero period acceleration at the operating deck of 2.0 times the normal gravitational acceleration.

(2) The reactor building structure shall have a maximum zero period acceleration amplification at the operating deck of 5.0 times the maximum ground acceleration at the site.

(3) Buildings other than the reactor building containing safety class equipment shall have maximum floor accelerations at zero period at the highest floor elevation at which Westinghouse equipment is located as specified in Figure 3.7-2 of RESAR-3S.

The ground response spectra and damping values specified in RESAR-3S are consistent with the recommendations of Regulatory Guides 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," and 1.61, "Damping Values for Seismic Design of Nuclear Power Plants."

We find this information adequate to determine the acceptability of the RESAR-3S seismic design. Of the sites we have previously evaluated, approximately 90 percent had a safe shutdown earthquake characterized by a maximum horizontal ground acceleration equal to or less than 0.4 times normal gravitational acceleration.

Westinghouse will provide balance-of-plant designers with response spectra for all support points of the nuclear steam supply system piping and primary equipment such as the reactor, reactor coolant pumps, steam generator, and pressurizer. We will require that these response spectra envelop the response spectra for the actual site conditions and structures for applications utilizing RESAR-3S.

711176

~~717~~ 344

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 seismic Category I systems and components. Governing response parameters will be combined by the square root of the sum of the 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 the squares of the three components for the modal response spectrum method or by algebraic combination at each time step for the time history method.

Floor response spectra inputs to be used for design and test verification of structures, systems, and components will be described in applications referencing RESAR-3S. 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.

For the case where a component or system is supported from two or more locations with relative displacements and different response spectra, it is our position that where the response spectrum method is used, the procedure involve 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 Paragraphs NB-3652 and NB-3653-1 of Section III of the ASME Code.

Westinghouse has stated that there will be no components in the RESAR-3S scope of analysis which will be connected between buildings, and that the primary components of the reactor coolant system will be supported at no more than two floor elevations. The staff position on seismic Category I piping systems supported from two or more locations is applicable only to the reactor coolant system in RESAR-3S because this is the only system in RESAR-3S where the system piping is within the RESAR-3S scope. We will apply this position on applications which reference RESAR-3S for all other seismic Category I piping systems.

7111177 ~~717~~ 345

Analysis for the case where a component or system is supported from two or more locations with relative displacements and different response spectra complies with our position and Section III of the ASME Code.

We conclude that the dynamic methods and procedures for seismic systems analyses proposed by Westinghouse provide an acceptable basis for seismic design of the reactor coolant system and all seismic Category I components within the scope of RESAR-3S.

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 be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

We reviewed the RESAR-3S criteria, testing procedures, and dynamic analyses to be employed to ensure structural and functional integrity of the reactor coolant piping system, 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 reactor coolant system piping, components, and component supports within the scope of RESAR-3S classified as 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 service as required by Paragraph NB-3622.3 of Section III of the ASME Code. The ASME Code requires 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 Paragraph NB-3622.3 of Section III of the ASME Code. The preoperational vibrational testing of Class 1 auxiliary piping and Class 2 piping will be reviewed in applications referencing RESAR-3S.

Since RESAR-3S provides a preoperational piping test program which covers only the reactor coolant loop and surge line piping, the testing program appropriate to all other piping will be provided and reviewed on each plant application referencing RESAR-3S.

We will require that the preoperational piping testing programs include development of loads similar to those experienced during reactor operation and be consistent with our positions concerning preoperational piping dynamics effects test programs. Selected locations in the reactor coolant piping system that will be subjected to visual inspection and measurement (if needed) as performed by the piping designer

711178

~~717 346~~

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. If, during the test, the reactor coolant piping system 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.

Proper functioning of safety-related mechanical equipment is essential to assure the capability of such equipment to perform protective actions in the event of a safe shutdown earthquake. The dynamic testing and analysis procedures which will be implemented 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 are adequately designed to withstand seismic disturbances are acceptable.

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 identified in RESAR-3S will continue to function during and after the 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 assuring that the design of the equipment supports will withstand the dynamic loads associated with seismic events as well as operational vibratory loading conditions without gross loss of structural integrity.

The RESAR-3S reactor internals structures are similar to those of Indian Point Unit 2 which has been established as the prototype for a four-loop plant internals verification program and was fully instrumented and tested during initial start-up. Differences between the RESAR-3S reactor and that at Indian Point Unit 2 result from the use of 17 x 17 fuel assemblies and the replacement of the annular thermal shield with neutron shielding panels in the RESAR-3S design. These internal modifications have been analyzed and will be confirmed by instrumenting the Trojan reactor which utilizes 17 x 17 fuel assemblies and neutron shielding panels. We will review the Trojan test data to substantiate the acceptability of the design modifications. In addition, applicants referencing RESAR-3S will conduct the confirmatory prefunctional and hot functional testing examination for internals integrity to fulfill the requirements of Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing."

The preoperational vibration test program outlined in RESAR-3S will be used to verify the design adequacy of the reactor internals under loading conditions that will be comparable to those experienced during operation. The proposed combination of tests, predictive analysis, and post-test inspection will provide adequate assurance that

711179

717 347

the reactor internals can be expected to withstand flow-induced vibrations without loss of structural integrity during their service lifetime. We will review the preoperational vibration test program that will be submitted with the final design application in accordance with Regulatory Guide 1.20 for assurance that it constitutes an acceptable basis for demonstrating the design adequacy of the reactor internals in satisfying the applicable requirements of Criteria 2 and 14 of the General Design Criteria.

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 the reactor coolant piping system, restraints, components, and component supports classified as ASME Class 1 within the scope of RESAR-3S, is acceptable. 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 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 recognizes the need to properly interpret all potential dynamic loads for the design 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 postulated loss-of-coolant accident conditions for the most adverse postulated loading event provides assurance 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 be designed to withstand the effects of earthquakes.

We have reviewed the RESAR-3S information concerning design transients and methods of analysis for seismic Category I components, including those designated as Class 1, 2, 3, or component supports under Section III of the ASME Code and component supports, reactor internals, and other components not covered by the ASME Code.

We reviewed the list of transients to be used in the design and fatigue analysis of all RESAR-3S ASME Code Class 1 components, component supports, and reactor internals within the reactor coolant pressure boundary. The number of events for each transient are included in RESAR-3S 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., the 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, and seismic events that are contained in the ASME 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-3S 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.

We find that the design transients, plant conditions, and loading combinations specified provide an acceptable 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 descriptions of the computer programs that will be used in dynamic and static analyses to determine the structural and functional integrity of seismic Category I ASME Code and non-ASME 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. A brief description and the extent of application of each of these computer programs are included in RESAR-3S.

As required by Appendix B of 10 CFR Part 50, we determined that the applicability and validity of the above computer programs have been shown by one of the following methods:

711181

~~717 349~~

(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.

Westinghouse employs an inelastic method of analysis to evaluate the design of safety-related ASME Code Class 1 components, component supports, reactor internals, and other non-ASME Code items for the faulted plant condition (NB-3225 and Appendix F of the ASME 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 ASME Code.

We reviewed the inelastic stress and deformation design limits specified by Westinghouse for ASME Code Class 1 components, and for component supports, reactor internals, and other non-ASME Code items, and the methods of analysis used to calculate the stresses and deformations resulting from faulted condition loadings. We find these limits and methods to be acceptable.

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 ASME Code Class 1, 2, and 3 components and component supports, and other non-ASME Code items are in conformance with established technical positions and criteria described above.

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 adequate for the design of these items. We, therefore, conclude that the analysis methods for seismic Category I components are acceptable.

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 be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

711182

717 350

We have reviewed the RESAR-3S information concerning the structural integrity and operability of pressure-retaining components, their supports, and core support structures within the scope of RESAR-3S which are designed in accordance with the rules of Section III of the ASME 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 ASME Code Class 1, 2, 3, and component support items within the scope of RESAR-3S 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 ASME Code constructed items within a system, including the categorization of the appropriate plant and component operating condition for each initiating event, such as the postulated loss-of-coolant accident and the safe shutdown earthquake used with each loading combination, are judged by comparison with the recommendations of Regulatory Guide 1.48, "Design Limits and Loading Combinations for Seismic Category I Fluid System Components," and with appropriate standards acceptable to the staff developed by professional societies and standards organizations. The corresponding stress limits applied to the design of ASME Code-constructed items are specified in the appropriate subsections of Division 1 of Section III of the ASME Code. The need for more conservative stress limits for active components and their supports are considered in the context and with the other features of the operability assurance program.

The objectives in reviewing the loading combinations and stress limits employed by Westinghouse in the design of ASME Code Class 1, 2, 3, and component support items within the scope of RESAR-3S 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 ASME 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 ASME Code Class 2 and 3 Components

All safety-related ASME Code Class 2 and 3 systems and components within the scope of RESAR-3S 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 ASME Code Class 2 and 3 pressure-retaining components in systems classified as seismic 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

711183

~~717 351~~

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-3S design load combinations and associated stress and deformation limits specified for all ASME Code Class 2 and 3 components within the scope of RESAR-3S constitute an acceptable basis for design in satisfying Criteria 1, 2, and 4 of the General Design Criteria and are consistent with our 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 within the scope of RESAR-3S during or after postulated accidents or natural events.

The objective of our review of the pump and valve operability assurance program was to determine whether the program will assure the operability of a component which is required to function to shut down the plant or mitigate the consequences of an accident.

The operability assurance program proposed by Westinghouse applies to active pumps and valves in seismic Category I systems within the scope of RESAR-3S 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 "active" functions, such as pump operation and valve opening or closure, 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 valves and ASME Code Class 2 and 3 active pumps.

3.6.3.4 Pressure Relieving Devices

The design criteria for the installation of the RESAR-3S pressure-relieving devices are not within the scope of RESAR-3S and, therefore, findings as to acceptability will be made during our review of individual applications referencing RESAR-3S.

711184 ~~717~~ 352

3.6.3.5 Component Support Design

The primary system component support design must provide adequate margins of safety under all plant operating conditions. The supports included within the scope of RESAR-3S are described in Table 1.7-1 of RESAR-3S.

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, such as the postulated loss-of-coolant accident and the safe shutdown earthquake used with each loading combination, were judged by comparison with the recommendations of Regulatory Guide 1.48 and with appropriate standards developed by professional societies and standards organizations that are acceptable to us. 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 ASME Code.

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 RESAR-3S 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 in RESAR-3S the responsibilities between Westinghouse and the balance-of-plant designer for mechanical components, systems, and testing procedures. We find this delineation consistent with what is done for custom plants and, therefore, acceptable.

Westinghouse has committed to furnish the balance-of-plant designer with necessary interface information in accordance with our positions. We find these commitments acceptable.

Westinghouse's delineation of responsibility and commitments to furnish interface information will assure that the integrated plant design is within the design envelope of RESAR-3S, thus achieving compatibility between the nuclear steam supply systems and components and the balance-of-plant design.

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 the 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 within the scope of RESAR-3S such that a test program will provide baseline preservice testing information and a periodic testing schedule.

Westinghouse has committed to provide the reference test data specified in Subsections IWP and IWV of Section XI of the ASME Code for pumps and valves within its scope of supply.

Compliance with these ASME Code requirements constitutes an acceptable basis for satisfying the applicable portions of Criteria 37, 40, 43, and 46 of the General Design Criteria.

Each utility application referencing RESAR-3S will be required to provide a program to include reference data for pumps and valves procured outside the scope of RESAR-3S, and the utility will be required to implement an inservice testing program covering periodic testing of pumps and valves for the life of the plant in accordance with Subsections IWV and IWP, respectively of Section XI of the ASME Code.

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 earthquakes without losing their capability to perform their intended safety functions.

The proper functioning of essential instrumentation and electrical equipment in the event of a safe shutdown earthquake is necessary to initiate protective actions including, for example, the operation of the engineered safety features.

Westinghouse has stated that the required seismic tests will conform to the procedures as specified in the Institute of Electrical and Electronics Engineers (IEEE) Standard 344-1975, "Guide for Seismic Qualification of Class I Electric Equipment for Nuclear

Power Generating Stations." A complete listing of the instrumentation and electrical equipment within the scope of RESAR-3S is found in Table 1.7-1 of RESAR-3S.

Regulatory Guide 1.89, "Qualification of Class IE Equipment for Nuclear Power Plants," provides that in the case of applications for which the safety evaluation report issue date is July 1, 1974, or after, the qualification of Class IE equipment take into account aging and environmental effects as specified in IEEE Standard 323-1974, "IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations." Westinghouse will conduct seismic qualification tests for all equipment after pre-qualification in accordance with IEEE Standard 323-1974, including the Nuclear Power Engineering Committee position statement of July 24, 1975, for aging and environmental effects. The seismic tests will conform to the procedures as specified in IEEE Standard 344-1975, which will account for multi-axis and multi-frequency effects of seismic excitation and fatigue effects caused by a number of operational basis earthquake events.

Westinghouse has submitted Topical Report WCAP-8587, "Environmental Qualification of Westinghouse NSSS Class IE Equipment." This report describes the Westinghouse program for demonstrating the environmental qualification of instrumentation and electrical equipment important to safety. We are currently evaluating the test methods and procedures to be adopted by Westinghouse as described in WCAP-8587 to satisfy the objective of IEEE Standard 323-1974 with regard to the environmental qualification of instrumentation, controls, and electrical equipment important to safety.

We conclude that the commitments made by Westinghouse will facilitate the development of a seismic qualification testing program which, when implemented for presently available seismic Category I instrumentation and electrical equipment, will provide adequate assurance that such equipment will function properly during the excitation from vibratory forces imposed by the safe shutdown earthquake and under the conditions of post-accident operation and are, therefore, acceptable.

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.

4.0 REACTOR

4.1 Summary

Criterion 10 of the General Design Criteria requires that the reactor core and associated systems 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-3S in support of the proposed reactor design. Our evaluation is contained in the following sections.

The RESAR-3S nuclear steam supply system is designed to operate at a thermal power rating of 3425 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 3411 megawatts. The 14 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.

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. 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 fuel assembly nozzles. Alignment and transverse spacings will be maintained by eight spacer grids spaced uniformly along the axis of the assembly.

All fuel rods will be internally prepressurized with helium during final welding to minimize cladding compressive stresses during service. The level of prepressurization is designed both to preclude any cladding tensile stresses due to a net internal pressure and to preclude clad flattening. The specific level of prepressurization 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 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 thermal performance of the reactor fuel. 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 staff and reviewed. The methods addressed include testing, mechanical analyses, thermal and hydraulic analyses, and accident analyses. The results of our review are reported in "Technical Report on Densification of Westinghouse PWR Fuel" issued 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.

The Westinghouse predictions of uranium dioxide densification are founded entirely upon empirical correlations. The data employed by Westinghouse were obtained from the examination of Westinghouse fabricated fuel irradiated in commercial power reactors. The values of the correlation parameters are both typical of the Westinghouse fuel fabrication process and independent of the fuel assembly dimensions. The Westinghouse predictions are conservative relative to their data. We independently

711189

~~717~~ 357

TABLE 4-1

FUEL MECHANICAL DESIGN COMPARISON

Design Parameter	Westinghouse RESAR-3S	Westinghouse Typical Operation Fuel
FUEL ASSEMBLY		
Rod Array	17x17	15x15
Number of Fueled Rods	264	204
Number of Spacer Grids	8	7
Number of Guide Thimbles	24	20
Inter-rod Pitch, inches	0.496	0.563
Average Thermal Output (4 loop), kilowatts per foot	5.4	7.0
FUEL PELLETS		
Density (theoretical), percent	95	94
Fuel Weight/Unit Length (per rod, not assembly), pounds per foot	0.364	0.462
FUEL CLADDING		
Outside Radius, inches	0.187	0.211
Thickness, inches	0.0225	0.0243
Radius/Thickness Ratio	8.31	8.68

711190

717 358

assessed the Westinghouse analysis of densification effects by comparing predictions to data from the Saxton experimental pressurized water reactor. We conclude that the methods to be employed by Westinghouse will consider the effects of densification in the reactor fuel assemblies in a manner which adequately describes the fuel behavior and are, therefore, acceptable.

4.2.1.3 Mechanical Performance

Although limited operating experience exists on 17x17 fuel assemblies, substantially all of the in-reactor operating experience with Westinghouse fuel rods and assemblies is applicable to the RESAR-3S 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-3S 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-3S 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.

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.

Confidence that the mechanical characteristics of the RESAR-3S fuel assemblies are predictable is enhanced by the results of out-of-reactor mechanical tests. Although most of the current results are from tests on typical 15x15 fuel assemblies, we expect the mechanical behavior of the 17x17 fuel assemblies to be similar since the 17x17 fuel assembly is only a slight mechanical extrapolation from the 15x15 fuel assembly. Topical reports describing the tests and analyses that have been performed by Westinghouse on the 17x17 fuel assemblies are listed in Table 4-3.

We have reviewed Topical Report WCAP-8278 and have determined that it provides an acceptable basis for demonstrating that the design of the 17x17 fuel assembly is adequate to withstand the effects of flow induced vibration under normal operating and transient conditions. Our final approval of this report is awaiting only the confirmatory results of the post irradiation surveillance program at Trojan. This matter is discussed further in Sections 3.6.1.1 and 4.2.1.4 of this report.

711131

~~717~~ 359

TABLE 4-2

RANGE OF DESIGN PARAMETER EXPERIENCE

Parameter	Range of Power Reactor Experience
Fuel Rod Array	14 x 14, 15 x 15, and 17 x 17
Rods per Assembly	179 to 264
Guide Thimbles per Assembly	16 to 24
Assembly Envelope, inches	7.76 to 8.43
Inter-rod Pitch, inches	0.563 to 0.463
Plenum Length, inches	3.27 to 6.69
Prepressurization, pounds per square inch, absolute	14.7 to over 400
Diametral Gap, inches	0.0065 to 0.0075
Spacer Grids/Assembly	7 to 9
Fuel Column Height, inches	120 to 144

TABLE 4-3

GENERIC DESIGN EVALUATION TOPICAL REPORTS

Tests & 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
Safety Analyses of the 17x17 Fuel Assembly for Combined Seismic and Loss-of-Coolant Accident	WCAP-8236
Fuel Rod Bowing	WCAP-8691
Revised Clad-Flattening Model	WCAP-8377

711132

717 360

The consideration of fuel rod bowing in the 17x17 design was previously analyzed by Westinghouse and documented in Topical Report WCAP-8346. The topical report described an analysis of rod bowing based upon deliberation of the potential mechanisms causing fuel rod bowing. The analyses were rigorous and compatible with the available data. The methodology of the topical report was approved with the requirement that observations of fuel rod bowing in modified fuel assemblies (rod-off-bottom) substantiate this methodology. Subsequent observations, however, indicated that the magnitude of rod bow was underpredicted.

Consequently, Westinghouse has reassessed its analysis in light of this new information and has documented its findings in Topical Report WCAP-8691. In this report, Westinghouse has documented its rod bowing experience to date which is based upon the inspection of 27 different regions of fuel (about 25,000 fuel rods) including more than 70 assemblies at burnups beyond 27,000 megawatt days per metric tonne of uranium. This experience has demonstrated the exposure (burnup) dependence of rod bowing.

We have completed our review of WCAP-8691 and have concluded that with the modifications described in our "Interim Safety Evaluation Report on Westinghouse Fuel Rod Bowing," dated April 1976, WCAP-8691 provides acceptable methods for predicting the magnitude of fuel rod bowing and for evaluating power density changes due to local changes in moderation. The effect of rod bowing on departure from nucleate boiling is discussed in Section 4.4.1 of this report.

Seismic effects and vertical loads from postulated double-ended hot and cold leg breaks during the postulated loss-of-coolant accident were analyzed in WCAP-8236. We found this analysis acceptable. However, Westinghouse subsequently postulated a new asymmetric hydraulic horizontal load caused by a postulated pipe break within the biological shield. Westinghouse has performed a preliminary analysis which indicates that the fuel assemblies will be able to accommodate this load. We conclude that this is acceptable for the preliminary design review stage. We will review this matter for individual applications referencing RESAR-3S at the final design review stage.

All of the other topical reports listed in Table 4-3 have been reviewed and approved.

4.2.1.4 Fuel Surveillance

Performance of the fuel during operation will be indirectly monitored by measurement of the activities of both the primary and secondary coolant for compliance with technical specification limits. Onsite surveillance normally includes examinations of fuel rod integrity, fuel rod and fuel assembly dimensions, alignment, and surface deposits.

For new fuel designs for which there is no operating experience, we require that a supplemental fuel surveillance program be conducted. The supplemental fuel surveillance program is directed at monitoring the behavior of the actual fuel systems as they perform in-reactor thus demonstrating the adequacy of the conclusions reached from the design evaluation. Such a program is being conducted on two of the first plants using the 17x17 design and includes the capability to perform destructive fuel rod tests.

4.2.1.5 Conclusion

We conclude that the cladding integrity of the RESAR-3S fuel will be maintained and that significant amounts of radioactivity will not be released during normal operation. Our conclusion is based on (1) operating experience with similar fuel, (2) results of out-of-reactor tests based on assemblies of similar design, (3) increased thermal margins of the 17x17 fuel, (4) technical specification requirements 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

We have reviewed the information presented in RESAR-3S 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 including all combinations of design loadings that will be accounted for in the design of the core support structure, such as 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 ASME 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.

Additional discussion of the design of the reactor pressure vessel internals can be found in Sections 3.6.2 and 3.6.3 of this report.

Westinghouse has committed to 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-3S 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 ASME 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 re internals conform to established technical procedures, positions, standa , and criteria that we find acceptable.

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 ASME Code 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.

711135

718 002

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 and compatibility, fabrication controls, and the extent of testing proposed by Westinghouse.

We have reviewed the adequacy and suitability of the materials specified for the lower core support structure including the core barrel, neutron shield pad assembly, core baffle, lower core plate and core supports; the upper core support structure including the top support plate, beam sections, upper core plate, support column, and guide tube assemblies; and the in-core instrumentation support structure.

Westinghouse has identified by specification the materials used for construction of these components. We have determined that these materials meet the requirements of Sections II and III of the ASME Code. 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 materials that will be used in the reactor vessel internals are in conformance with the requirements of Appendix I of Section III of the ASME Code.

Residual cold work in austenitic stainless steel is known to accelerate water corrosion. Expressing cold work in terms of increased yield strength, a yield strength of 90,000 pounds per square inch for Types 304 and 316 stainless steel corresponds to residual cold work greater than ten percent and less than 20 percent. We have selected a yield strength of 90,000 pounds per square inch as a conservative criterion for the use of cold worked austenitic stainless steel in light water reactor internals. This control imposed on the use of cold worked stainless steel will provide adequate protection during reactor operation from conditions which could lead to stress corrosion of the materials and loss of reactor internal structural integrity.

The only stainless steel material that will be used in the reactor vessel internals with yield strength greater than 90,000 pounds per square inch is the Type 403 stainless steel used for the core holddown spring. However, significant crack growth is considered to be impossible for this component considering the stress state and possible flaw size. The core holddown spring is acceptable based upon ASME Code Case 1337, which requires quenching or normalizing the material from 1775 to 1825 degrees Fahrenheit and a minimum tempering temperature of 1125 degrees Fahrenheit for four hours in order to minimize susceptibility to stress-corrosion cracking of the material. We conclude that this material will be compatible with the reactor coolant and is acceptable for this use.

7111 36

~~718 003~~

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 Guides 1.31, "Control of Stainless Steel Welding"; 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants"; and 1.44, "Control of the Use of Sensitized Stainless Steel."

The controls specified in Table 5.2-9 of RESAR-3S 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.

The materials selection, fabrication practices, and examination and protection procedures will be performed in accordance with the recommendations of the ASME Code and staff. 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 staff and ASME 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-3S nuclear steam supply system will be controlled by permanent devices such as the rod cluster control assemblies, temporary devices such as the burnable poison assemblies used only in the initial core, and boric acid, a soluble chemical neutron absorber. 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 53 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, such as power and temperature variations. The control and shutdown groups will provide adequate shutdown margin

711137

718 004

(1.6 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.

The manually controlled part length rods will be designed to control the axial neutron flux shape and axial xenon oscillations should they occur. Restrictions on the use of the part length rods are discussed in Section 4.3.1 of this report.

The soluble boric acid neutron absorber will be varied to control long term reactivity changes resulting from fuel depletion and fission product buildup, cold to hot zero power reactivity change, reactivity changes produced by intermediate-term fission products such as xenon and samarium, and burnable poison depletion.

For the RESAR-3S accident analyses, a conservative rod drop time of 2.1 seconds to 85 percent insertion has been used. This time is based on tests conducted at the Westinghouse Test Engineering Laboratory in the D-loop hydraulic test facility.

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 accident events. These load combinations were compared with those specified for each of the plant operation conditions as defined in Paragraph NB-3113 of the ASME Code.

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 Criteria.

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-3S, we reviewed the materials information relating to mechanical properties, the methods to control sensitization of stainless steel, welding and brazing, compatibility, testing, and cleaning and cleanliness control.

All parts of the control rod drive mechanisms that will be exposed to reactor coolant will be fabricated exclusively from austenitic stainless steels, martensitic stainless steels, nickel-chrome-iron alloy, and cobalt based alloys. All pressure containing parts will be made from Type 304 austenitic stainless steel. The 400 series martensitic stainless steel will be used only wherever magnetic flux will be carried by parts exposed to the main coolant. Cobalt based alloys will be used for the pins and latch tips, and nickel-chrome-iron alloy will be used for latch assembly springs. Hard chrome plating will provide wear surfaces on the sliding parts and will prevent galling between mating parts.

The reactivity control components metals that will be exposed to the primary coolant will be Types 304 and 308 austenitic stainless steel, nickel-based alloys, 17-4 PH stainless steel, and nickel-based braze. The reactivity control components consist of the full and part length rod cluster control assemblies, the burnable poison assemblies, the neutron source assemblies, and the thimble plug assemblies. All components of full and part length rod cluster control assemblies will be fabricated from Types 304 and 308 stainless steel except for the retainer, which will be of 17-4 PH material, and the springs, which will be Inconel-718 alloy at coolant surfaces.

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 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 ASME Code.

The controls imposed on heat treatment and fabrication of the materials of the reactivity control system are such as to minimize the probability of stress-corrosion cracking and provide assurance of satisfactory service performance. Specifically, the controls that will be imposed upon the austenitic stainless steel of the control rod drive mechanisms will limit maximum yield strength to 90,000 pounds per square inch and will conform to the recommendations of Regulatory Guides 1.31, "Control of Stainless Steel Welding"; 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants"; and 1.44, "Control of the Use of Sensitized Stainless Steel." In addition, the martensitic stainless steel will be tempered at a minimum temperature of 1100 degrees Fahrenheit.

Of the reactivity control components, the full- and part-length absorber control rods, the neutron source rods, and the burnable poison rods will use cold worked Type 304 austenitic stainless steel tubing. The tubing, which is welded and drawn, has a typical yield strength range of 80,000 to 90,000 pounds per square inch. Although Westinghouse's Materials Specification permits yield strengths up to 95,000 pounds per square inch, the Materials Specification requires the tubing to pass the American Society of Testing Materials 262, Practice E intergranular corrosion test. On this basis, we conclude that this limit is acceptable. Material cleaning and cleanliness control will be in accordance with Regulatory Guide 1.37 for all of the reactivity control components and the aging treatment for the precipitation hardenable 17-4 PH material of the rod cluster control assembly retainer will be at 1100 degrees Fahrenheit in accordance with our requirements.

Conformance with the ASME Code and the recommendations of the regulatory guides mentioned above, and with the stated limits 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 our review, we conclude that the design, fabrication, and testing of the control rods and control rod drives will be in accordance with Section III of the ASME Code and our requirements and are, therefore, acceptable.

4.3 Nuclear Design

The nuclear design of the RESAR-3S nuclear steam supply system is the same as that employed in the Westinghouse RESAR-3 Consolidated Version reference design. Our review of the nuclear design of the RESAR-3S nuclear steam supply system was based on the information provided in RESAR-3S, referenced topical reports, and discussions with Westinghouse.

The design bases presented for the nuclear design of the fuel and reactivity control systems are acceptable and comply with all applicable General Design Criteria.

Descriptions of the fuel assembly enrichments, physics of the fuel burnout process, burnable poison distributions, soluble boron concentrations, delayed neutron fraction, and neutron lifetimes have been provided. The values presented for these parameters meet the design bases and satisfy the applicable sections of the General Design Criteria.

We conclude, on the bases of our review and the similarity of the nuclear design with that of other approved nuclear steam supply systems, that the RESAR-3S nuclear design is acceptable.

711200 718 007

4.3.1 Power Distribution

Westinghouse's design bases pertaining to power distribution control are as follows:

(1) The core will not be operated with peaking factors greater than 2.32 under normal operating conditions to ensure that the initial conditions used in the postulated loss-of-coolant accident analysis are valid.

(2) During normal operation and faults of moderate frequency, the peak power in the fuel will be limited to 18 kilowatts per foot to prevent fuel melting.

(3) During normal operation and faults of moderate frequency, the power distributions will be limited to prevent the departure from nucleate boiling ratio from decreasing below 1.30.

Westinghouse has presented a detailed discussion of the power distribution control and monitoring techniques to be used in conjunction with the RESAR-3S design and the power distributions that result from employing these techniques. We have reviewed this information to ascertain that the design bases given above can be met.

The reactor will be provided with two types of neutron monitoring instrumentation systems to measure core power distributions - 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 five 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 an indication of total power, relative power in each quadrant of the core, and the relative power in the top and bottom of the core. Limits placed on the axial power offset, as measured from the relative power in the top and bottom of the core, and the radial tilt will ensure that the core peaking factor can be maintained below the design limit value and all power distributions produced will be conservative relative to the design power distribution used in the departure from nucleate boiling analyses.

The power distribution monitoring procedure proposed involves the maintenance of an essentially constant axial offset as measured by the excore detectors. 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 thus limiting the magnitude of axial xenon transient effects on the peaking factor. This will be achieved by restricting operation to a plus or minus five percent band about a target value of flux difference (upper minus lower excore detector readings) as measured under equilibrium full power conditions with essentially no rods in the core. This target value must be updated monthly. 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. Greater flexibility is allowed below 50 percent of full power.

Control of the flux difference within the target band will be accomplished using the full-length control rods and the boron control system. This is referred to as Mode A by Westinghouse. This procedure is identical to the one reviewed and approved in several recent operating license cases and currently in use at operating Westinghouse reactors. Recent operating experience from some of these operating plants indicates that these plants can be effectively operated within a plus or minus five percent offset band.

A second means of power distribution control, referred to as Mode B by Westinghouse, involves the use of the part-length control rods together with the full-length rods and the soluble boron control system. 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, only Mode A control will be allowed. All operating Westinghouse plants are now using this mode of control. Except for some limitations on load follow capability near end of core life, operation without part length rods presents no operational difficulties. Since the use of part-length control rods is not required for safe operation of the plant, and based on our evaluation of power distribution in the RESAR-3S core, we conclude that the power distribution can be controlled in an acceptable manner.

Westinghouse has selected a value of 2.32 as the design peaking factor and, to justify the use of this value, has performed extensive calculations to predict the expected power peaking that can occur during both steady-state and load follow operations when using constant axial offset control. The conditions studied included the extremes allowed by both control modes such as continuous operation at the limits of the band and operation outside the band for up to 24 hours below 90 percent power. An allowance for calculational error of five percent was applied to the expected peaking factors. This error 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 value. Thus, the design peaking factor of 2.32 is appropriate for use in the safety analysis.

The power distributions produced during normal operation with constant axial offset control can be degraded during faults of moderate frequency (defined in Section 15.1 of this report as Condition II transients) and this degradation must be demonstrated to be acceptable within the context of the design bases listed earlier. To show that fuel melting does not occur during faults of moderate frequency, Westinghouse has evaluated the effects of mispositioning full-length and part-length rods and allowing axial xenon redistribution without operator intervention for up to one quarter hour. For those events in which overpower conditions could arise, such as rod withdrawal conditions, it was assumed for analysis purposes that total power would be limited to 118 percent by the overpower trip. Otherwise, maximum total

~~718 009~~

 711202

power was assumed to be limited to 102 percent. This analysis demonstrates that peak linear power density can be limited to prevent fuel melting. Further demonstration of this is presented in the accident analyses. We find this justification sufficient for the preliminary design review stage.

Power distribution must also be limited to ensure that the departure from nucleate boiling ratio is not less than 1.30 during both normal operation and faults of moderate frequency. Westinghouse currently uses a chopped cosine axial power shape with a peak-to-average value of 1.55 and a design enthalpy rise hot channel factor as the basis for the departure from nucleate boiling related safety analyses and protection system settings.

Westinghouse has selected a value of 1.55 as the design enthalpy rise hot channel factor at full power. Studies performed by Westinghouse show that the design enthalpy rise hot channel factor is conservative assuming control rods are inserted to the power dependent insertion limits. Constant axial offset control requires that rods be positioned above these insertion limits thus providing margin to the design limits during normal operation.

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 calculated values of these coefficients and has also evaluated the accuracy of these calculations. We have reviewed the calculated values of the reactivity coefficients and have concluded that they adequately represent the full range of expected values. We have also concluded that the reactivity coefficients used in the safety analysis conservatively bound the expected values including uncertainties.

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 will measure the moderator temperature coefficient and the power coefficient during startup tests of plants referencing RESAR-3S 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 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.

711203

718 010

Control will be achieved with movable control rods and through the variation of boron concentration in the reactor coolant. Calculations made by Westinghouse show that 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 and for calculational uncertainties. In addition, the chemical and volume control system will be capable of shutting down the reactor by adding soluble boron poison, maintaining it in the cold shutdown condition at any time during the core life, and maintaining the reactor at least five percent subcritical when refueling with control rods removed. This combination of control systems satisfies the requirements of Criterion 26 of the General Design Criteria.

Core reactivity will be controlled by means of boron chemical poison dissolved in the coolant, the control rod assemblies, and burnable poison rods. The reactor will be operated at steady-state full power with most of the full-length control rods withdrawn. Limited insertion of the full-length control rods will permit compensating for fast reactivity changes, such as the effects of minor variations in moderator temperature and boron concentrations, and controlling the axial power distribution without impairing shutdown capability. Soluble boron poison will be used to compensate for slow reactivity changes including those associated with fuel burnup, change in xenon and samarium concentration, buildup of long-life fission products, burnable poison rod depletion, and the large moderator temperature change from cold shutdown to hot standby. The soluble boron poison system will provide the capability to take the reactor at least ten percent subcritical in the cold shutdown condition.

The proposed full-length control rod assemblies are divided into two groups - control and shutdown. The control groups are used during normal operation whereas the shutdown groups are always withdrawn prior to criticality and are available to provide for rapid reactor shutdown if required. 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.

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 conclude that Westinghouse's assessment of reactivity control is suitably conservative, that adequate negative reactivity worth has been provided by the control system to assure shutdown capability, and that the control rod and soluble boron worths are acceptable for use in the accident analysis.

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-3S. 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. Westinghouse has verified this stability in a startup physics test for a 193 fuel assembly core. The core is stable to axial xenon oscillations until a core exposure of 12,000 megawatt days per metric tonne of uranium is reached. 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 in RESAR-3S 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-3S. Therefore, these calculated physics characteristics are appropriate for use in the accident analysis.

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 used 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 fuel rods 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-3S 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 described below. A minimum allowable limit of 1.28 has been justified based on statistical analyses of applicable 17x17 data.

711205

718 012

(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 of this report. A comparison of these parameters with those of the RESAR-3 Consolidated Version is given in the table.

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-8536, "Critical Heat Flux Testing of 17x17 Fuel Assembly Geometry with 22-inch Spacing." We have reviewed this method and accepted it for use in applications using 17x17 fuel.

Another parameter that influences the thermal-hydraulic design of the core is rod-to-rod bowing within fuel assemblies (mechanical and nuclear aspects of fuel rod bowing are discussed in Section 4.2.1.3 of this report). Experimental data on the extent of bowing in the 17x17 fuel design is not yet available; however, acceptable methods based on data obtained with the 15x15 fuel design are available at this time. Although the 17x17 fuel design includes a departure from nucleate boiling penalty for rod bowing, recent data show this penalty to be inadequate. We have determined, however, that other design margins exist to offset the presently-indicated penalty due to rod bow.

Prior to the final RESAR-3S design data will be available such that rod-to-rod bowing will be adequately accounted for. Based on the observations that rod bowing is a time dependent process for which operational penalties can be imposed if deemed necessary, we conclude that the presently available information is adequate for issuance of a construction permit or a Preliminary Design Approval. The adequacy of the final design will be determined at the final design review stage.

In steady-state, two-phase, heated flow in parallel channels, the potential for hydrodynamic instability always exists. For years, Westinghouse has used the HYDNA code to predict the inception of hydrodynamic instability for its reactors. Although the HYDNA Code assumes that the core consists of parallel closed channels, Westinghouse has demonstrated by experiment that flow in parallel open channels which more accurately describes the flow in Westinghouse's reactors, is more stable than in parallel closed channels.

Results of HYDNA calculations for RESAR-3S show that the inception of hydrodynamic instability will occur at a power level in excess of 185 percent of rated power.

TABLE 4-4

COMPARISON OF THERMAL-HYDRAULIC DESIGN PARAMETERS
FOR THE RESAR-3S AND RESAR-3 CORES

	RESAR-3S	RESAR-3 Consolidated Version[a]
Reactor Core Heat Output (megawatts thermal)	3411	3411
System Pressure, Nominal (pounds per square inch, absolute)	2250	2250
Minimum Departure from Nucleate Boiling Ratio for Design Transients	1.30	1.30
Total Thermal Flow Rate (million pounds per hour)	140.3	142.2
Effective Flow Rate for Heat Transfer (million pounds per hour)	134.0	135.8
Average Velocity Along Fuel Rods (feet per second)	16.7	16.8
Average Mass Velocity (million pounds per hour per square foot)	2.62	2.66
Coolant Temperature (degrees Fahrenheit)		
Design Nominal Inlet	558.1	557.3
Average Rise in Core	62.7	62.3
Core Heat Transfer Surface Area (square feet)	59,700	59,900 (59,700)[b]
Average Heat Flux (British thermal units per hour per square foot)	189,800	189,400 (189,800)[b]
Maximum Heat Flux (British thermal units per hour per square foot)	440,300	454,600 (474,500)[b]
Maximum Thermal Output for Normal Operation (kilowatts per foot)	12.6	13.0 (13.6)[b]
Fuel Central Temperature at Beginning of Life, Maximum at 100 Percent Power (degrees Fahrenheit)	3250	3250 (3500)[b]

[a]Without fuel densification effects

[b]With fuel densification effects

711207 ~~718 014~~

Although the HYDNA code has not been submitted for our detailed review, we conclude that the margin for hydrodynamic stability is acceptable for the issuance of a construction permit or a Preliminary Design Approval. We will review the methods used in the HYDNA code prior to the 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] Westinghouse has calculated that at the beginning of core life at 100 percent power, with a linear heat generation rate of 12.6 kilowatts per foot, the fuel centerline temperature will be 3250 degrees Fahrenheit. The peak power density that would occur for a reactor trip at the 118 percent maximum over power 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 4150 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. These methods are general in nature and apply to all Westinghouse plants, including RESAR-3S. We conclude that the Westinghouse calculations adequately show that there will be no fuel melting.

For the reactor described in RESAR-3S and other 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. On the bases of our review of these codes to date, we do not anticipate that any significant changes to the thermal-hydraulic design of the core will be required and, therefore, conclude that the presently available information is adequate for issuance of a construction permit or a Preliminary Design Approval.

---

[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).

4.4.2 Conclusions

On the basis of our review of the thermal-hydraulic design of the proposed RESAR-3S core including the design criteria and the steady state analysis of the core thermal-hydraulic performance, we have identified two codes that must be reviewed for the final design. These result in requirements for:

(1) Verification of the THINC code on both the subchannel and core wide bases, and

(2) 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 conclude that the presently available information on the preliminary thermal-hydraulic design of the RESAR-3S reactor is acceptable for issuance of a construction permit or Preliminary Design Approval.

In the event that the analytical methods are determined not to be conservative during the final design review, appropriate restrictions on operations can be established at the operating license stage for plants employing the RESAR-3S nuclear steam supply system.

711209

~~718 016~~

5.0 REACTOR COOLANT SYSTEM

5.1 Summary

Section 50.2(v) of 10 CFR Part 50 defines the reactor coolant pressure boundary as 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. A description of the scope of the RESAR-3S reactor coolant system is contained in Section 1.2 of this report and Section 1.7 of RESAR-3S.

The residual heat removal system, emergency core cooling system, and chemical and volume control system are the principal systems connected to the reactor coolant system. The proposed RESAR-3S 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 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, a 31-inch inside diameter crossover pipe from the steam generator outlet to the reactor coolant pump inlet, and a 27.5-inch inside diameter cold leg pipe connecting the pump discharge to the reactor vessel inlet. The RESAR-3S reactor coolant system design does not include loop stop valves. 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. The reactor coolant system will include a pressurizer relief tank, together with the interconnecting piping and instrumentation necessary for operational control, and to receive, condense, and cool steam discharged from the pressurizer safety valves. The entire reactor coolant system will be located within the containment building. A simplified diagram of the reactor coolant system is provided in Figure 5-1 of this report.

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. The reactor coolant system design is essentially the same as that of RESAR-3 Consolidated Version.

5.2 Integrity of the Reactor Coolant Pressure Boundary

5.2.1 Design of Reactor Coolant Pressure Boundary Components

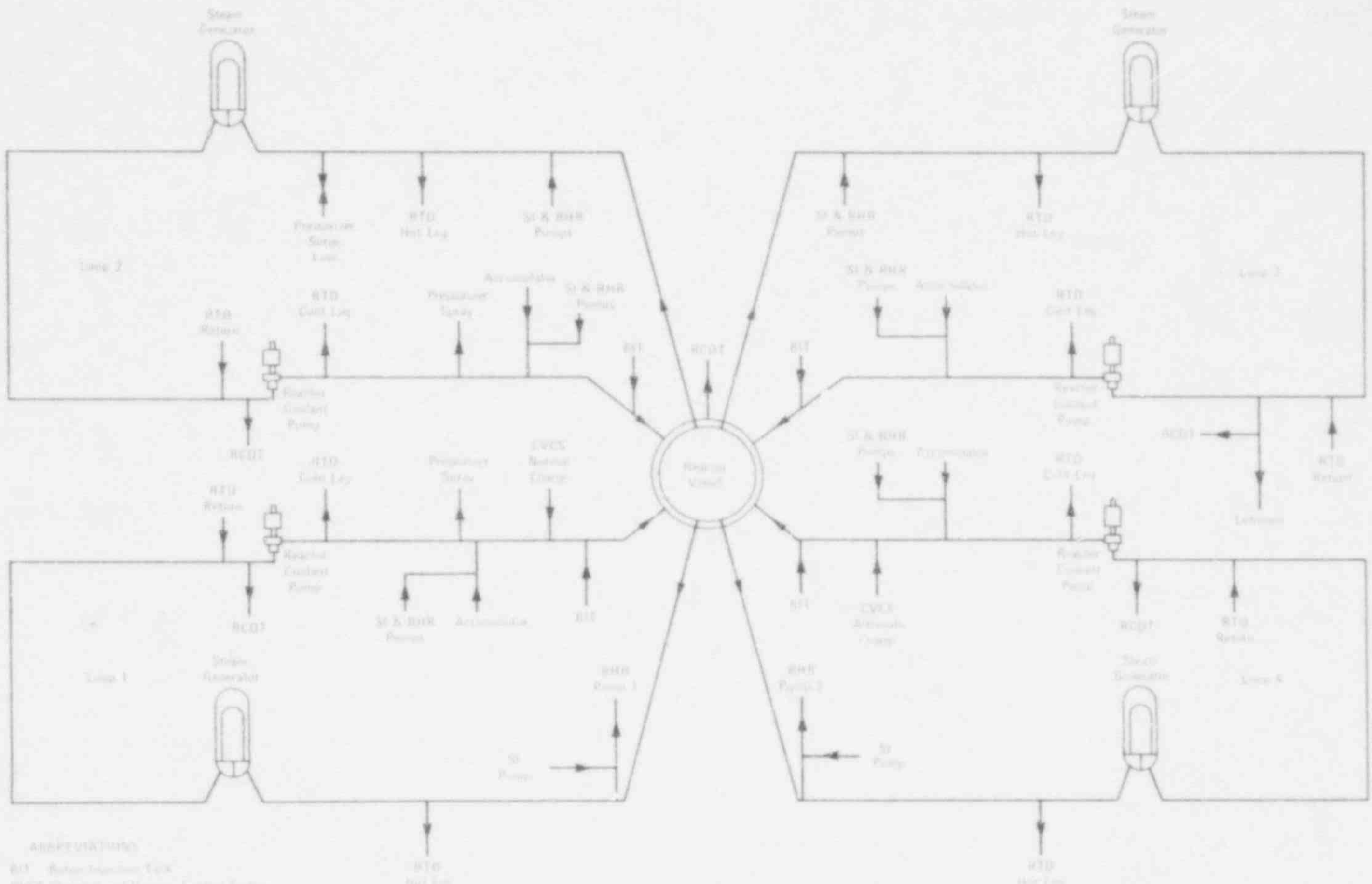
Criterion 4 of the General Design Criteria requires that structures, systems, and components important to safety be designed to accommodate the effects of normal operation, maintenance, testing, and postulated accidents. We reviewed the design of the reactor coolant pressure boundary components to determine that component quality will be commensurate with the importance of the safety function of the reactor coolant pressure boundary. Our general review of Class 1, 2, and 3 components is contained in Section 3.6.3 of this report.

We determined that the design loading combinations specified under Section III of the ASME 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, "Design Limits and Loading Combinations for Seismic Category I Fluid System Components." 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 condition 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.

We have reviewed the information provided in RESAR-3S and conclude that pressure-retaining components of the reactor coolant pressure boundary as defined in Section 50.55a of 10 CFR Part 50 have been properly identified in Table 3.2-1 of RESAR-3S and classified as ASME Section III, Code Class 1 components.

POOR ORIGINAL



ABBREVIATIONS

BIT Boron Injection Tank
CVCS Chemical and Volume Control System
RCDT Reactor Coolant Drain Tank
RHR Residual Heat Removal
RTD Resistance Temperature Detector
SI Safety Injection

Figure 5.1 REACTOR COOLANT SYSTEM

~~718 019~~

7111212

Westinghouse states that reactor coolant pressure boundary components will be constructed in accordance with the requirements of the applicable codes and addenda as specified in Section 50.55a of 10 CFR Part 50. In conformance with these requirements, the code edition and the applicable addenda for each ASME Section III, Code Class 1 component will be based on the dates related to the construction permit applications referencing RESAR-3S and identified in the applicants' preliminary safety analysis reports.

We conclude that construction of the components of the reactor coolant pressure boundary in conformance with the ASME Code and the Commission's regulations provides adequate assurance that component quality will be commensurate with the importance of the safety function of the reactor coolant pressure boundary and is acceptable.

Westinghouse has stated that no ASME 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. Westinghouse has also stated its intent to comply with Regulatory Guides 1.84, "Code Case Acceptability-ASME Section III Design and Fabrication," and 1.85, "Code Case Acceptability-ASME Section III Materials." In the event the use of new ASME Code cases are planned, staff authorization shall be obtained prior to their application in the construction of ASME Section III, Class 1 components.

We conclude that compliance with the requirements of these ASME Code cases, in conformance with the Commission's regulations, will result in a component quality level that is commensurate with the importance of the safety function of the reactor coolant pressure boundary and is acceptable.

5.2.2 Overpressurization Protection

Protection of the primary system against overpressurization will be provided by two power operated pressure relief valves and three safety valves, which are within the scope of RESAR-3S. The three safety valves in conjunction with the steam generator safety valves, which are not within the scope of RESAR-3S, 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 to 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,

711213

~~718 020~~

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 Topical Report WCAP-7769, "Overpressure Protection for Westinghouse Pressurized Water Reactors," Revision 1, along with specific revisions pertinent to RESAR-3S. The margin for overpressure protection predicted in WCAP-7769, Revision 1, is acceptable.

There have been several reported incidents of reactor vessel overpressurization in pressurized water reactors during startup and shutdown in which the limitations of Appendix G to 10 CFR Part 50 have been exceeded. We have initiated discussions with applicants, licensees, and vendors, including Westinghouse, on a generic basis relative to overpressure protection during these conditions. We will require that any corrective measures resulting from our review of this generic matter be incorporated in the RESAR-3S design and in applications referencing RESAR-3S.

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 reactor coolant pressure boundary construction materials 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, we reviewed 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 by Westinghouse, and will be procured in accordance with the requirements of Section III of the ASME Code, including Addenda and Code cases appropriate to comply with Appendix B to 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.

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 304 and 316 stainless steels will normally be used. Because these compositions are susceptible to stress-corrosion cracking when exposed to certain environmental conditions, process controls will be exercised during all stages of component manufacturing and reactor construction to avoid sensitization of the material and to minimize exposure of the stainless steel to contaminants that could lead to stress-corrosion cracking.

Nonmetallic thermal insulation used on austenitic stainless steel components will conform with the recommendations of Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

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. 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 ASME Code.

The specification for reactor coolant water chemistry is shown in Table 5.2-9 of RESAR-3S. The reactor coolant system water chemistry has been selected to minimize corrosion. Periodic analysis of the chemical composition will be performed to verify that the coolant water quality conforms to the specification. The chemical and volume control system will provide the means for adding chemicals to the coolant to scavenge oxygen and to control the hydrogen ion concentration. Hydrazine and hydrogen will be used to scavenge oxygen, and lithium hydroxide will be used for hydrogen ion concentration control.

The controls imposed on reactor coolant chemistry are in conformance with the recommendations of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," and provide reasonable assurance that the reactor coolant pressure boundary components will be adequately protected during operation from conditions that could lead to stress corrosion of the materials and loss of structural integrity of a component.

The 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 Regulatory Guide 1.44 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.

718 022

All materials must meet the acceptance standards of Article NB-2330 of Section III of the ASME Code and the requirements of Appendix G to 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 ASME Code (1974 Edition). In addition, materials for the reactor vessel will meet the acceptance criteria of Appendix G to 10 CFR Part 50.

The fracture toughness tests and procedures required by Section III of the ASME Code, as augmented by Appendix G to 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 ASME Code, and the results of fracture toughness tests performed in accordance with the ASME Code and the Commission's regulations in establishing safe operating procedures, provide adequate safety margins during operating, testing, maintenance, and postulated accident conditions. Compliance with these ASME Code provisions and the Commission's 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 ASME Code, Section III, Appendix D, Paragraph D-1200, supplemented by Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy Steel." 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.

The controls imposed during procurement and fabrication of the ferritic steel pressure-retaining components of the reactor coolant pressure boundary will also conform to the recommendations of Regulatory Guides 1.34, "Control of Electroslag Weld Properties," 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Plants," and 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components."

Compliance with the above recommendations constitutes an acceptable basis for meeting in part the requirements of Criteria 1 and 14 of the General Design Criteria.

~~718~~ 023

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 demonstrated 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. We have reviewed the proposed welding procedures and have found them to be in compliance with our requirements.

The controls that will be imposed upon components constructed of austenitic stainless steel used in the reactor coolant pressure boundary conform to the recommendations of Regulatory Guides 1.31 "Control of Stainless Steel Welding," 1.34, "Control of Electroslag Weld Properties," 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel," 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," and 1.44, "Control of the Use of Sensitized Stainless Steel."

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 sound and free from hot cracking (microfissures) and in a metallurgical condition which precludes susceptibility to stress-corrosion cracking during service. Conformance with the above regulatory guides constitutes an acceptable basis for meeting the applicable 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 Criteria 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 ASME Code, "Rules for Inservice Inspection of Nuclear Power Components."

7911217

718 024

We reviewed Westinghouse's definition of the reactor coolant pressure boundary against the inspection requirements of Section XI of the ASME Code for all Class I pressure-containing components and their supports except for those components excluded under IWB-1220 of Section XI of the ASME Code. The RESAR-3S 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, and

(3) The reactor coolant system safety and relief valves.

We reviewed the design and arrangement of the reactor coolant system components to determine conformance with the requirements of WA-1500, "Accessibility," of Section XI of the ASME Code. The design of the reactor coolant system incorporates provisions for access for inservice inspection of all ASME Class 1, 2, and 3 pressure-retaining components and systems in accordance with Section XI of the ASME Code. Tools and equipment have been designed and are included in RESAR-3S to facilitate the remote inspection of those areas of the reactor vessel not readily accessible to inspection personnel.

5.2.4.2 Conclusions

We will require that selected welds and weld heat-affected zones be inspected periodically. We conclude that the designs of all ASME Code Class 1 components of the reactor coolant pressure boundary incorporate provisions for access for inservice inspections in accordance with Section XI of the ASME Code and that methods have been developed to facilitate the remote inspection of those areas of the reactor vessel not readily accessible to inspection personnel. We will require that each RESAR-3S plant maintain the access provided by Westinghouse to ensure that the inservice inspection program for all Class 1, 2, and 3 components will be conducted in accordance with the requirements of Section XI of the ASME Code. 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 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. Compliance with the applicable portions of these inservice inspection and hydrostatic test requirements of Section XI of the ASME Code constitutes an acceptable basis for satisfying the requirements of Criterion 32 of the General Design Criteria.

~~718 025~~

711218

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 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 have reviewed Westinghouse's material specifications for the reactor vessel and closure studs. Their adequacy for use in the construction of such components was assessed on the basis of their material, mechanical, and physical properties; the effects of irradiation on these materials; their corrosion resistance; and fabricability. We reviewed the welding controls and procedures for low alloy and austenitic steel welds.

The fracture toughness of the ferritic materials to be used for the reactor vessel and the 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 the American Society for Testing and Materials (ASTM) Standard E 185-73 and Appendix H to 10 CFR Part 50.

The composition of the ferritic materials specified for the reactor vessel and the allowable limits for residual elements such as copper, sulfur, and phosphorous were reviewed.

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 its 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 its toughness properties, Westinghouse has stated that should results of surveillance tests indicate that the toughness has degraded to an unacceptable level, the reactor

711219

~~718~~ 026

vessel can be annealed to restore the toughness to an acceptable level. We will require that each RESAR-3S plant maintain the capability for an in-place anneal of the reactor vessel if required.

We reviewed the adequacy of the reactor vessel material surveillance program to monitor changes in the fracture toughness properties of the 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. We conclude that the program is in compliance with ASTM Standard E 185-73 and Appendix H to 10 CFR Part 50.

We determined that the ferritic materials in the reactor coolant pressure boundary will meet the toughness requirements of Section III of the Code. In addition, the material for the reactor vessel and reactor vessel studs will meet the acceptance criteria of Appendix G to 10 CFR Part 50.

5.3.1.2 Conclusions

The specifications of the materials to be used for the construction of the reactor vessel and its appurtenances have been identified and found to be in conformance with Section III of the ASME 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 ASME Code have been submitted by Westinghouse. Since certification has been made by Westinghouse that the materials and fabrication requirements of Section III of the ASME 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 ASME Code. Demonstrations have been made using these special techniques and have satisfied all requirements of the ASME 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 ASME Code. The controls imposed on welding preheat temperatures will be in conformance with the recommendations of Regulatory Guide 1.50, "Control of Preheat Temperature for Welding of Low-Alloy

Steel," 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 to ensure adequate delta ferrite content will be in conformance with Regulatory Guide 1.31, "Control of Stainless Steel Welding."

The fracture toughness tests required by the ASME Code and by Appendix G to 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 ASME Code as a guide in establishing safe operating procedures, and use of the results of the fracture toughness tests performed in accordance with the ASME Code and Commission's regulations, will provide adequate safety margins during operating, testing, maintenance, and postulated accident conditions. Compliance with these ASME Code provisions and Commission's 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 testing 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 test

(3) Pressure-temperature limits for heatup and cooldown operations

(4) Pressure-temperature limits for core operation

Appendices G and H to 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 Appendix G to Section III of the ASME Code, "Protection Against Nonductile Failure," during heatup, cooldown, and test conditions. Appendix G to 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 operating limit curves, with

7112210 718 028

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 the pressure-temperature operational and test limitations for acceptability by performing check calculations using the methods referenced in the ASME Code and in the Welding Research Council Bulletin 175, "PVRC Recommendations on Fracture Toughness."

We conclude that upon the satisfactory resolution of the generic matter of overpressurization, as discussed in Section 5.2.2 of this report, the reactor will be capable of being operated in a manner which will meet the requirements of Appendix G to Section III of the ASME Code and Appendix G to 10 CFR Part 50.

The use of Appendix G of the ASME Code as a guide in establishing safe operating limitations, using results of the fracture toughness tests performed in accordance with the ASME Code and the Commission's regulations, will assure adequate safety margins during operation, testing, maintenance, and postulated accident conditions. Compliance with these ASME Code provisions and the Commission's regulations constitutes an acceptable basis for satisfying the requirements of Criterion 31 of the General Design Criteria.

5.3.3 Reactor Vessel Integrity

All portions of RESAR-3S 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 the factors contributing to the structural integrity of the reactor vessel and conclude that the design, materials, fabrication, inspection, and quality assurance requirements will conform to applicable Commission regulations and regulatory guides and to the rules of Section III of the ASME Code. The stringent fracture toughness requirements of the Commission's regulations and Section III of the ASME Code 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 ASME Code and Appendix G to 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 Code and any pertinent ASME Code cases;

(2) Will be made from materials of controlled and demonstrated high quality;

711222

~~718 029~~

(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 wil 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 1.30 under normal and transient operating conditions. The estimated design loop flow will be 35.08 million pounds per hour.

Sufficient pump rotational inertia (95,000 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 1.30 in the event of loss-of-pump power.

The reactor coolant pumps will be vertical, single stage, centrifugal, shaft seal pumps. Suction will be from the bottom and discharge will be horizontal. The pumps will be composed of three regions - the hydraulics, 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.

7912230 ~~718 030~~

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 us. 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. We will be following the development and performance of this program as well as other industry analytical and experimental programs on a generic basis.

We have determined that additional protective measures, such as prevention of excessive pump overspeed or limitation of potential consequences to safety-related equipment, are technically feasible. If the results of the generic investigations of this matter indicate that additional protective measures are necessary to assure that an acceptable level of safety is maintained, we will require that they be implemented.

Information in RESAR-35 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 to be used, were reviewed to establish that the flywheel materials will exhibit adequate fracture toughness at normal operating temperature.

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 Code.

The flywheel design, including allowable stresses, design overspeed considerations, shaft and bearing adequacy, and the consequences of pump seizure is being reviewed on a generic basis. If the results of this review indicate that modifications are warranted, we will require them to be made. The loss of flow incident is evaluated in Section 15.4 of this report.

The probability of 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 access for an inservice inspection program for reactor coolant pump flywheels have been reviewed and found acceptable on the basis of conformance with Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity," and established industry codes and standards. We will require that each RESAR-3S plant maintain access to the pump flywheel to ensure that the inservice inspection program will be conducted in accordance with the recommendations of Regulatory Guide 1.14.

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 several 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-3S steam generators will be similar to those used in the RESAR-3 Consolidated Version design.

The secondary side overpressure protection system is not within the scope of the RESAR-3S design and is, therefore, not described. However, we require that the following measures, which are assumed in the transient and accident analyses and are, therefore, interface requirements, be incorporated in the balance-of-plant design. We have determined that these measures are technically feasible. 711225

~~718~~ 032

(1) Safety relief valves must have a total capacity of 105 percent of the design steam flow at 110 percent of the steam generator design pressure.

(2) Power operated relief valves must pass at the no-load pressure a steam flow equal to 15 percent of the steam flow used for plant design. There must be at least one valve per main steam line located upstream of the main steam line isolation valve.

(3) The maximum capacity of any single safety or power operated relief valve must be less than or equal to 1.05 million pounds per hour.

(4) The main steam line isolation valves located outside the containment must shut off flow from either the forward or reverse direction within five seconds.

(5) The safety and relief valves must be at full operation when the accumulated pressure is 103 percent of the valve setting.

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 generators form important parts of the boundary.

We have reviewed the selection of steam generator materials and the controls which will be exercised during the fabrication of these components. The steam generators will be fabricated as ASME Code Class I components. The mechanical properties for the materials selected for the steam generators will meet the ASME Code requirements as stated in Appendix I of Section III and Parts A, B, and C of Section II of the ASME Code. Welding procedures and fabrication processes will be qualified in accordance with the requirements of Sections III and IX of the ASME Code. Fracture toughness of ferritic materials used in the steam generator construction will meet the requirements of Article NB-2300 and Appendix G, Paragraph G-2000 of Section III of the ASME Code.

The procedures for weld-depositing corrosion-resistant cladding on the tube sheet will be qualified according to the requirements of Article QN-214 of Section IX of the ASME Code. The Inconel 600 tubes will be expanded for the full depth of the tube sheet to avoid the presence of a deep crevice between the tube and tube sheet pursuant to the recommendations of Materials Engineering Branch Technical Position MTEB 5-3, "Monitoring of Secondary Side Water Chemistry in PWR Steam Generators." which is contained in Section 5.4.2.1 of the Standard Review Plan. The welds between the tubes and tube sheet will meet the requirements of Sections III and IX of the ASME Code.

Onsite cleaning and cleanliness control will be in accordance with the recommendations of Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," and as stated in American National Standards Institute Standard N45.2.1-1973, "Cleaning of Fluid Systems and Associated Components for Nuclear Power Plants."

Conformance with the above stated applicable codes, standards, positions, and criteria constitutes an acceptable basis for meeting the applicable requirements of Criteria 14, 15, and 31 of the General Design Criteria.

The staff has under consideration appropriate monitoring of secondary water chemistry and inservice inspection programs to further enhance steam generator tube integrity. Upon completion of our review, we will consider appropriate recommendations or requirements for use in connection with the RESAR-3S design.

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-3S 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 ASME Code Class 1 and 2 components including individual tubes as recommended in Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes," and Section XI of the ASME Code. We will require that each RESAR-3S plant provide access to the steam generators to ensure that the inservice inspection program will be conducted in accordance with the recommendations of Regulatory Guide 1.83 and the requirements of Section XI of the ASME Code. Conformance with Regulatory Guide 1.83 and Section XI of the ASME 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 provides the means for filling and draining the refueling cavity. The system will consist of two 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 two of the reactor coolant system loops and the return lines will be connected to the cold legs of the four reactor coolant system loops.

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 two pumps and two 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 sixteen hours after being placed into operation. If one of the two 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.

Use of the residual heat removal system for normal plant cooldown will not compromise its use as part of the emergency core cooling 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 system recirculation cooling should the need arise.

Residual heat removal system isolation has been provided through the use of interlocks which prevent opening of the isolation valves when the reactor coolant system pressure is greater than 425 pounds per square inch, guage, and automatically close the isolation valves when the reactor coolant system pressure reaches 750 pounds per square inch, guage. This is in compliance with our position and is, therefore, acceptable. These interlocks are discussed further in Section 7.6.4 of this report.

We determined that the RESAR-3S residual heat removal system design did not originally meet the single failure criterion in that one of the two isolation valves in the suction lines of each of the two trains were powered from the same source, the failure of which would have prevented the operation of both residual heat removal system trains. Westinghouse proposed utilizing the auxiliary feedwater system along with the steam generator power-operated relief valves as backup to the residual heat removal system.

It is our position that the system provided to remove residual heat be capable of reducing the reactor coolant temperature to a cold shutdown value (200 degrees Fahrenheit or less) within a reasonable period of time (on the order of a day) with either only onsite or only offsite power available and assuming the most limiting single failure. The auxiliary feedwater system along with the steam generator power-operated relief valves, however, would not be capable of reducing the reactor coolant temperature to a cold shutdown value within a reasonable period of time.

Consequently, Westinghouse specified as an interface requirement in RESAR-3S that the balance-of-plant design include provisions for supplying Class IE electrical power to the residual heat removal system suction isolation valves in such a manner that the single failure criterion is satisfied for both system operation and isolation. We

 711228 ~~718 035~~

find this to be acceptable. In addition, however, Westinghouse has described in RESAR-3S a temporary power supply arrangement as a means of accomplishing this interface requirement. We have not reviewed Westinghouse's proposed temporary power supply arrangement since the specific design will be provided by the balance-of-plant designer and, therefore, will be reviewed in applications referencing RESAR-3S.

We have determined that with Westinghouse's suction line isolation valve power interface requirement, the RESAR-3S residual heat removal system design complies with the requirements of Criteria 19 and 34 of the General Design Criteria and, therefore, conclude that the RESAR-3S residual heat removal system design is acceptable.

The residual heat removal system will be inspected periodically during normal plant operation by applicants referencing RESAR-3S. Recalibration of the instrumentation channels, should it be necessary, will be done during each refueling operation.

We are presently considering on a generic basis the question of whether capability should be provided for transferring heat from the reactor to the environment from normal reactor operating conditions to cold shutdown using only safety-grade systems, with only offsite or onsite power available, and assuming the most limiting single failure. If we determine that this capability should be provided, we will require that the RESAR-3S design and the designs of the balance-of-plant portions of applications referencing RESAR-3S be modified accordingly. We have determined that such modifications are technically feasible and conclude that this matter can be left for post-preliminary design approval stage consideration.

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 ten 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 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 ten 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 decrease and the electric heaters will act to restore the normal operating pressure. During insurges, as the 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. Two 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 is within the scope of RESAR-3S and is designed to condense and cool the discharge from the pressurizer safety and relief valves. The tank will normally contain water under a predominantly nitrogen atmosphere. However, connections will be provided to permit the gas in the tank to be periodically analyzed to monitor the concentration of hydrogen and/or oxygen. The sample lines and gas monitors are not within the scope of RESAR-3S.

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 the pressurizer discharge during a ten percent step load decrease. The pressurizer response was evaluated for a discharge of pressurizer 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.

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 110 percent of the steam volume discharged from the pressurizer. The tank and rupture disc 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-3S, for any anticipated transient the pressurizer relief tank pressure will not exceed the design pressure of rupture discs. Therefore, there is no anticipated transient for which reactor coolant would be released to the containment.

Based on the margin provided in the design bases for the pressurizer relief tank and the determination that reactor coolant will not be released to containment for any anticipated transients, we conclude that the design of the pressurizer relief tank is acceptable.

5.4.6 Safety and Relief Valves

The pressurizer safety valves are within the scope of RESAR-3S and 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, absolute, in compliance with Section III of the ASME Code. This objective will be met without reactor trip or any operator action.

The relief valves are within the scope of RESAR-3S and 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.

These valves are similar to those used in previous Westinghouse designs and are acceptable.

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, 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-3S includes, as an option, a loose parts monitoring system which we have found acceptable for use in previous Westinghouse plants. We will impose the requirement for installation of an acceptable loose parts monitoring system on each applicant utilizing RESAR-3S.

5.4.8 Reactor Vessel Supports

On May 7, 1975, we were informed by a licensee of a pressurized water reactor, Virginia Electric and Power Company, that the asymmetric loading resulting from a postulated pipe rupture in the reactor coolant system had not been taken into account in the original design of the reactor pressure vessel support system for the North Anna Units 1 and 2 (Docket Nos. 50-338 and 339).

This loading results from the forces induced on the internals within the reactor vessel caused by differential pressure conditions within the vessel immediately following a postulated loss-of-coolant accident. In addition, the asymmetric loading from transient differential pressures that would exist around the exterior of the reactor vessel from the same postulated pipe rupture was not included in the original design analysis. However, the symmetric loadings from such a postulated pipe rupture were included in the original analysis of the reactor pressure vessel supports.

It is our opinion that these factors related to the design of the reactor pressure vessel supports are generic in nature and may apply to the RESAR-3S design. Accordingly, we are taking steps to review this problem on a generic basis to determine the extent of the problem.

We have informed Westinghouse of the nature of this problem and have requested Westinghouse to verify that the design procedures for the reactor pressure vessel support system will properly include the asymmetric forces described above in the final design of the supports. Westinghouse has provided verification that the final design will include the asymmetric forces.

Based on our review of this generic problem to date, we have determined that the methodology necessary to model the complete reactor coolant system in sufficient detail to determine analytically the magnitudes and phase relationships of the vessel support system loads from the transient pressure differentials has been developed by Westinghouse. The calculational techniques have been refined so that it is practical to evaluate the actual dynamic system response to all the known transient loads. Furthermore, Westinghouse has informed us that structural analyses based on the loads developed by the worst case loading demonstrate that Westinghouse reactor coolant support systems now being designed can sustain these loads and remain within conservative design basis stress limits comparable to those stress limits specified in Appendix F of Section III of the ASME Code.

On the basis of our review of this problem to date, we have concluded that Westinghouse can properly account for these forces during the final design of the reactor vessel support system.

5.5 Conclusion

Our review of the RESAR-3S reactor coolant system included review of the integrity of the reactor coolant pressure boundary, the reactor vessel and its appurtenances, system component and subsystem designs, and the residual heat removal system. We have determined 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 conclude that the design is acceptable.

711232 718 039

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-3S 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. A description of the scope of the RESAR-3S engineered safety features is contained in Section 1.2 of this report and Section 1.7 of RESAR-3S.

The engineered safety features provided within the scope of RESAR-3S and evaluated herein are those portions of the containment isolation system relating to the systems within the scope of RESAR-3S and the emergency core cooling system. The piping design and layout and certain components are outside the scope of RESAR-3S. A detailed description of the scope of RESAR-3S can be found in Section 1.7 of RESAR-3S.

We have reviewed the proposed RESAR-3S 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 Chapter 15 of this report. They will be designed 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-3S describes a nuclear steam supply system utilizing a four-loop reactor coolant system, a 3411 megawatt thermal pressurized water reactor, and associated auxiliary systems. The containment systems for a nuclear generating station utilizing the RESAR-3S design will include a reactor containment structure, containment heat removal systems, containment isolation systems, and containment combustible gas control systems. However, RESAR-3S includes only the portions of the containment isolation system for the systems within the scope of RESAR-3S. The remainder of the containment systems will be included in applications which reference RESAR-3S.

741233

Westinghouse has provided mass and energy release information for postulated loss-of-coolant accidents for use in establishing the containment design conditions and designing the containment subcompartments and component supports. These calculations are designed to maximize mass and energy release so as to conservatively maximize containment pressure for purposes of establishing containment design. The containment pressure calculations which minimize containment pressure for the emergency core cooling system analysis are discussed in Section 6.3 of this report.

Applicants which reference RESAR-3S will provide the pressure evaluations for containment design, the design pressure evaluation for subcompartment designs, and the containment response to ruptures in the secondary system. The containment type, such as dry, subatmospheric, or pressure suppression, is not specified in RESAR-3S. For any containment type, the mass and energy information provided by Westinghouse in RESAR-3S 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 from a postulated loss-of-coolant accident. Westinghouse assumed a containment pressure of 61.7 pounds per square inch, absolute, following the initial blowdown to calculate the mass and energy release data.

The containment pressure calculations for the emergency core cooling system evaluation are only applicable to dry containment types. Therefore, applications which reference RESAR-3S and utilize containment types other than dry will be required to provide additional analyses and/or justification.

6.2.1 Containment Functional Design

The containment will provide a low leakage barrier that encloses the nuclear steam supply system including the reactor, steam generators, reactor coolant pumps, and pressurizer, as well as certain components of the engineered safety feature systems. RESAR-3S contains no specific information on containment design. However, the effects of operation and accident conditions of the RESAR-3S systems on the containment design must be accounted for.

Westinghouse has calculated the mass and energy release rates resulting from postulated loss-of-coolant accidents for use in containment design calculations and for use in containment subcompartment analysis. These calculations are described in Westinghouse Topical Report WCAP-8264, "Westinghouse Mass and Energy Release Data for Containment Design," Revision 1. Westinghouse has also provided long-term mass and energy release data in Tables 6.2-1 through 6.2-4 of RESAR-3S.

We have reviewed the methods and assumptions described in WCAP-8264 and have concluded that these methods will conservatively maximize mass and energy release to the containment and are, therefore, acceptable. We will require that applicants which reference RESAR-3S demonstrate that the mass and energy release data calculated by Westinghouse is applicable to their specific containment design.

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-3S.

6.2.2 Containment Isolation System

The containment isolation system is designed to isolate the containment atmosphere from the outside environment under accident conditions. Only those containment isolation valves for RESAR-3S systems are within the scope of RESAR-3S and are evaluated herein. The detailed descr ption of isolation provisions for the balance of plant will be supplied in applications utilizing RESAR-3S. A complete listing of the containment isolation valves within the scope of RESAR-3S is provided in Table 6.2-7 of RESAR-3S.

Double barrier protection, in the form of closed systems and isolation valves will be provided so that no single valve or piping failure n result in the loss of containment integrity. The reactor building isolation si nal will be activated by high reactor building pressure. Certain containment is lation valves will also isolate following low steam line pressure or low primary ystem pressure. Following receipt of a containment isolation signal, all fluid pen ations within the scope of RESAR-3S 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.

The two containment building sump recirculation lines, each of which will supply suction to two high pressure and one low pressure injection systems and the containment spray system, will each be provided with a single, motor-operated gate valve outside the containment building. The valve will be enclosed in a leak-tight compartment. Also, the piping from the sump to the valve compartment will be enclosed in a concentric guard pipe. The valve compartment and the guard pipe will not open to the containment building atmosphere.

We have reviewed the design of the containment sump recirculation line isolation provisions and conclude that system reliability will be greater with only one valve in the line. In addition, we have determined that the recirculation system is closed outside the containment and that a sin e failure of an active component can be accommodated with only one isolation val e in the line. As an interface requirement, the closed system outside containment will be missile protected, seismic Category I, and Safety Class 2 design, and will have a design temperature and pressure rating at least equivalent to that for the containment. On this basi, we conclude that the proposed containment isolation provisions for the containment recirculation lines are in conformance with Criterion 56 of the General Design Criteria and are, therefore, accept le.

The residual heat removal system suction lines will each contain two motor-operated valves inside containment. These valves (1) will be equipped with diverse "prevent open" and "autoclose" interlocks to prevent overpressurization of the residual heat removal system, (2) will be closed during reactor operation, and (3) will remain closed in the event of an accident. In addition, the outermost valve in each line will also be a containment isolation valve. The residual heat removal system suction lines will be connected to the same closed, engineered safety features system outside containment as the containment sump recirculation lines.

It is our position that a single containment isolation valve in each residual heat removal system suction line inside containment in conjunction with the closed engineered safety features system outside containment satisfies our double barrier protection containment isolation requirement. On this basis, we conclude that the proposed containment isolation provisions for the residual heat removal system suction lines are in conformance with Criterion 55 of the General Design Criteria and are, therefore, acceptable.

Containment isolation valves in engineered safety features lines which are required to perform a safety function following an accident are operated remotely from the control room. Automatic closure of these valves would defeat their design purpose since following an accident they must be opened or remain open in order for the engineered safety features to operate. On this basis, we conclude that the remote operation of these valves is in conformance with Criteria 55 and 56 of the General Design Criteria and is, therefore, acceptable.

Our review of the containment isolation system within the scope of RESAR-3S has included schematic drawings and descriptive information for the isolation provisions for fluid systems within the scope of RESAR-3S which penetrate the containment boundary. The review has also included Westinghouse's proposed design bases for the containment isolation provisions and analyses of the functional capability of the containment isolation system.

Based on our review, we conclude that the containment isolation provisions within the scope of RESAR-3S are in conformance with Criteria 54, 55, 56, and 57 of the General Design Criteria and are, therefore, acceptable.

6.2.3 Combustible Gas Control in Containment

Following a postulated loss-of-coolant accident, hydrogen may accumulate inside the containment. The major sources of hydrogen generation within the scope of RESAR-3S include chemical reaction between the zirconium fuel rod cladding and steam, and radiolysis of aqueous solutions in the reactor core and in the containment sump.

Westinghouse has analyzed the post-loss-of-coolant accident hydrogen generation from the steam supply system described in RESAR-3S with respect to sources of hydrogen generation described above. This analysis is consistent with the guidelines of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," and is, therefore, acceptable.

In our evaluation of applications referencing RESAR-3S, we will consider any additional sources of hydrogen generation and assure that the assumptions used in the RESAR-3S

711236

~~718 043~~

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.

Hydrogen recombiners, which are used to limit the concentration of hydrogen in the containment following a postulated loss-of-coolant accident, are not within the scope of RESAR-3S. A hydrogen sampling system, hydrogen recombiners, and a backup purge system design will be described in applications referencing RESAR-3S.

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 be provided which can perform its safety function assuming a single failure.

The RESAR-3S 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. The emergency core cooling system will also be designed to protect against steam line break consequences. The RESAR-3S emergency core cooling system will be similar in design, size, and capacity to those of the Commanche Peak and Trojan plants which are also designed for core outputs of 3411 megawatts thermal.

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. These requirements will be met even with minimum engineered safety features available.

The emergency core cooling system will have the required number, diversity, reliability, and redundancy of components such that no single failure of active emergency core cooling system equipment during the short term or no single failure of active or passive equipment 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).

The boric acid injection portion of the emergency core cooling system will be designed to control the reactivity insertion accompanying the rapid cooldown following any single steam line rupture or spurious relief valve lifting. Control of the reactivity insertion will be accomplished by injection of high concentration boric acid solution into the reactor coolant system. The range of steam line ruptures protected against is up to and including the double-ended circumferential rupture of the largest pipe in the steam system.

6.3.2 System Design

In the event of a postulated 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, the primary coolant, and the metal and core stored energy will be released to the containment. 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.

Following a postulated loss-of-coolant accident, the emergency core cooling system will operate initially in the passive accumulator mode and the active high head injection mode, then in the active low head injection mode, and finally in the recirculation mode.

The emergency core cooling system will consist of four accumulator tanks, two high pressure and one low pressure injection systems, with provisions for recirculation of the borated coolant after the end of the injection phase. Various combinations of these systems will assure core cooling for the complete range of postulated break sizes.

Each of the four accumulators will have a total volume of 1350 cubic feet with a minimum volume of borated water of 850 cubic feet and a maximum volume of nitrogen gas of 500 cubic feet at a minimum pressure of 600 pounds per square inch, gauge. The minimum boric acid concentration will be 1900 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 of this report, these valves will be provided with appropriate interlocks to assure that the valves will be open during power operation when availability of the accumulators is required.

Upon actuation of a safety injection signal, the high pressure injection mode of operation will consist of the operation of two centrifugal charging pumps, rated at 150 gallons per minute each at a design head of 5800 feet, which provide high pressure injection of boric acid solution by means of the boron injection tank whose contents are maintained at a nominal 21,000 parts per million boron concentration, into the reactor coolant system. Also designed to operate during the high pressure injection mode are two safety injection pumps, rated at 425 gallons per minute each at a design head of 2680 feet, which will take their suction from the refueling water storage tank which contains a boron concentration of 2000 parts per million.

Low pressure injection will be provided by two residual heat removal pumps, rated at 3000 gallons per minute each at a design head of 375 feet, which will take their suction from the refueling water storage tank.

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 following manual realignment of several valves to complete the change-over from the injection mode to the recirculation mode, the emergency core cooling system will provide the long-term cooling requirements by recirculating the reactor coolant, which will have spilled from the ruptured pipe and collected in the sump, back to the reactor vessel. The return of the sump water will be through the reactor coolant cold legs for 17.5 hours after the accident as discussed in Section 6.3.4 and simultaneously through both the hot and cold legs thereafter to control boron precipitation in the core during long-term post-accident cooling.

The boric acid injection portion of the emergency core cooling system will consist of the boron injection tank, boron injection surge tank, boron injection recirculation loop, charging pumps, and the associated valves. The system piping, layout, and heat tracing are not within the scope of RESAR-3S. The boron injection tank will contain 900 gallons of 21,000 parts per million boric acid solution and will be connected to the reactor coolant system by means of a loop from the refueling water storage tank, through the charging pumps, to the boron injection tank inlet. The boron injection tank outlet is connected through a common manifold pipe to pipes connected to each of the four reactor coolant cold legs.

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 two recirculation pumps 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.

As an interface requirement, Westinghouse has stated that redundant and separate heat tracing must be provided in applications referencing RESAR-3S. This heat tracing will be installed on all piping, valves, flanges, instrumentation lines, and pump casings carrying the 21,000 parts per million boric acid 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.

6.3.3 Design Evaluation

We reviewed the proposed emergency core cooling system 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's ability to withstand a single active failure during the short term or a single active or passive failure during the long term following a postulated loss-of-coolant accident.

The two safety injection pumps will receive water from the refueling water storage tank through a common header. We determined that the initial proposed design did not meet the single failure criterion in that the header contained a single valve, the closure of which would prevent the safety injection pumps from receiving water. Westinghouse subsequently modified its design by adding a second valve in parallel with the original single valve. We have determined that this arrangement meets the single failure criterion and conclude, therefore, that it is acceptable.

The miniflow bypass line from the two safety injection pumps converge into a common header. We determined that the original proposed design did not meet the single failure criterion in that the header contained a single valve, the closure of which, when pumping against the pumps' shutoff head, could severely damage both safety injection pumps. Westinghouse subsequently modified its design by replacing the single valve in the header with two valves, one in each miniflow bypass line. We have determined that this arrangement meets the single failure criterion and conclude, therefore, that it is acceptable.

Westinghouse has identified six 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 as follows:

(1) Safety injection pump discharge (hot leg injection), one valve in each of two injection trains, which must remain closed.

711240

~~718 047~~

(2) Safety injection pump discharge (cold leg injection), one valve to a common header in the cold leg injection train, which must remain open.

(3) Residual heat removal system pump discharge (hot leg injection), one valve a common header in the hot leg injection train, which must remain closed.

(4) Residual heat removal system pump discharge (cold leg injection), one valve in each of the two injection trains, which must remain open.

We determined that the emergency core cooling system design did not originally meet the single failure criterion in that electrical malfunctions could result in spurious valve movements to undesirable positions and thereby result in loss of capability of the system to perform its intended safety function. Westinghouse, therefore, elected to lockout power to these valves with power to be restored by manual action at the motor control centers.

It is our position, however, that when lockout of power to valves that are required to open or close in various safety system operational sequences is elected in lieu of design changes in order to meet the single failure criterion, that (1) capability be provided to lockout and restore motive power to the valves from the main control room and (2) redundant position indication be provided in the main control room. This position is documented in Electrical, Instrumentation and Controls Systems Branch Technical Position EICSB 18, "Application of the Single Failure Criterion to Manually-Controlled Electrically-Operated Valves," which is contained in Appendix 7-A of the Standard Review Plan. Consequently, Westinghouse included the capability to lockout and restore motive power to the six affected valves from the main control room and to provide in the main control room redundant indication of the positions of the valves. We have determined that with this modification the design meets the single failure criterion and conclude, therefore, that it is acceptable.

6.3.4 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.

Appendix K to 10 CFR Part 50 of the Commission's regulations requires that the combination of emergency core cooling system subsystems to be assumed operative shall be those available after the most damaging single failure of emergency core cooling system equipment has occurred. The worst single failure was identified by Westinghouse as the loss of one residual heat removal pump, which along with the assumptions of maximum containment cooling and reduced emergency core cooling flow, results in the maximum calculated peak clad temperature.

Westinghouse had originally provided in RESAR-3S the results of its small and large break loss-of-coolant accident analysis assuming that the fluid temperature in the

 711241 ~~718 048~~

upper head region of the reactor vessel was equal to that of the cold leg. On August 9, 1976, however, Westinghouse informed the staff that the fluid temperature in the upper head region of the reactor vessel may be higher than that assumed in the loss-of-coolant accident analysis. A thermocouple reading at Connecticut Yankee (Docket No. 50-213) confirmed that the fluid temperature in the upper head is higher than that originally assumed.

Consequently, Westinghouse reanalyzed the loss-of-coolant accident conservatively assuming that the fluid temperature in the upper head region of the reactor vessel is equal to that of the hot leg. The large break loss-of-coolant accident analysis was limited to a spectrum of four double-ended guillotine breaks with discharge coefficients of 0.4, 0.6, 0.8, and 1.0. To supplement the analysis, Westinghouse submitted Topical Reports WCAP-8566, "Westinghouse ECCS Four-Loop Plant (17 x 17) Sensitivity Studies," and WCAP-8865, "Westinghouse ECCS Four-Loop Plant (17 x 17) Sensitivity Studies," which cover other break sizes, types, and locations and demonstrate that the guillotine breaks are the worst cases for this type plant.

The analyses submitted by Westinghouse identified the worst break as the double-ended cold leg guillotine break with a Moody multiplier of 1.0. The calculated peak clad temperature was 2148 degrees Fahrenheit which is within the acceptable limit of 2200 degrees Fahrenheit as specified in Section 50.46(b) of 10 CFR Part 50. In addition, the maximum local metal/water reaction of 6.7 percent, and total core wide metal/water reaction of less than 0.3 percent were well below the allowable limits of 17 percent and one percent, respectively. The analyses were performed based on an assumed total peaking factor of 2.32, 102 percent of the rated core power level of 3411 megawatts thermal, and 102 percent of a peak linear power density of 12.6 kilowatts per foot.

The postulated small break loss-of-coolant accident analysis included a three-break spectrum specific to RESAR-3S and referenced Westinghouse Topical Report WCAP-8356, "Westinghouse Emergency Core Cooling System - Plant Sensitivity Studies." The four-inch diameter pipe break was identified as the limiting small break with a calculated peak clad temperature of 1673 degrees Fahrenheit. This clearly indicates that the postulated small break loss-of-coolant accident is not the limiting case.

The effect of rod bow on fuel rod behavior has not been included in the emergency core cooling system analysis for RESAR-3S in an explicit manner. We expect that prior to startup of the first RESAR-3S plant, information on rod bow for Westinghouse 17 x 17 fuel will be available and will be used to assess the effect of rod bow in emergency core cooling system performance. The operating technical specification limits established during the operating license review stage of plants referencing RESAR-3S will include a consideration of rod bow. Restrictions on operations can be imposed if the results of rod bow studies indicate the need. This matter is discussed further in Section 4.2.1.3 of this report. The presently available information is adequate for issuance of a construction permit or a Preliminary Design Approval. The adequacy of the final design will be determined at the final design review stage.

711242

718 049

Appendix K to 10 CFR Part 50 of the Commission's regulations requires that the effect on the containment pressure of operation of all the installed pressure reducing systems and processes be included in the emergency core cooling system evaluation. For this evaluation, it is conservative to minimize the containment pressure since this will increase the resistance to steam flow in the reactor coolant loops and reduce the reflood rate in the core. Following a postulated loss-of-coolant accident, the pressure in the containment building will be increased by the addition of steam and water from the primary reactor system into the containment atmosphere. After initial blowdown, heat transfer from the core, primary metal structures, and steam generators to the emergency core cooling system water will produce additional steam. This steam, together with any emergency core cooling system water, is released from the primary system postulated break to the containment during both the blowdown and the reflood and post-reflood phases.

Energy removal within the containment occurs by several means. Steam condensation on the containment walls and internal structures serves as a passive energy heat sink that becomes effective early in the blowdown transient. Subsequently, the operation of the containment heat removal systems, such as containment sprays and fan coolers, will remove energy from the containment atmosphere. When the energy removal rate exceeds the rate of energy addition from the primary system, the containment pressure will decrease.

The emergency core cooling system containment pressure calculations for RESAR-3S were made with the Westinghouse emergency core cooling system evaluation model. We concluded that Westinghouse's containment pressure model is acceptable for the emergency core cooling system evaluation. We require, however, that justification of the plant-dependent containment input parameters used in the analysis be submitted for our review of each plant utilizing RESAR-3S.

We have reviewed the containment input data postulated by Westinghouse relating to containment net-free volume, passive heat sinks, and containment heat removal systems and find that the data for the passive heat sinks are conservative in comparison with our recommendations contained in the staff technical paper, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation." The passive heat sink data are based on measurements within the containment of similar nuclear plants.

Each application utilizing the RESAR-3S emergency core cooling system evaluation must show that the significant containment parameters for the balance of plant are conservative when compared with those used in RESAR-3S.

On the basis of Westinghouse's use of our recommendations contained in the staff technical paper described above, we conclude that the plant-dependent information used for the analysis to determine the minimum emergency core cooling system containment pressure following a postulated loss-of-coolant accident for RESAR-3S is conservative. Therefore, the analysis is acceptable for use in the evaluation of emergency core cooling system performance.

Appendix K to 10 CFR Part 50 of the Commission's regulations also requires that the combination of emergency core cooling subsystems to be assumed operative shall be available assuming the most severe single failure. This worst single failure was identified by Westinghouse as the loss of one residual hea removal pump, which provided, within a consistent set of assumptions, (1) the maximum containment cooling a reduction in emergency core cooling flow and (2) the maximum calculated peak clad temperature.

A review of the RESAR-3S piping and instrumentation diagrams has indicated that spurious actuation of specific motor-operated valves was considered in the selection of the worst single failure. We have concluded that the emergency core cooling system performance will be adequate in the event of any postulated failure of a single active component.

We have also reviewed the proposed procedures and the system design for preventing excessive boric acid buildup in the reactor vessel during the post loss-of-coolant accident long-term cooling period and have concluded that switchover time from cold to simultaneous hot and cold leg injection must be changed from 24 hours, as proposed by Westinghouse, to 17.5 hours after a loss-of-coolant accident. This change is required to assure that, in the event of a cold leg break, the concentration of the boric acid in the core region does not exceed the solubility limits. We also require that in the case where only two subsystems are available, they should be aligned in such a manner that one subsystem injects into the hot leg and the other into the cold leg. This arrangement would assure that even in the case of a hot leg break, sufficient flow through the core is provided. These changes are administrative and will be verified during our review of the final design and incorporated in the technical specifications.

We will require that in applications referencing RESAR-3S, the applicants use actual values for the emergency core cooling system piping flow resistances, emergency core cooling system and reactor coolant system volumes, and residual heat removal system piping flow resistances. In addition, the effects of rod bowing will be considered in the development of the technical specifications for the nuclear peaking factors for plants utilizing RESAR-3S. To prevent water hammer, we will also require that venting provisions be described in the final design application for the emergency core cooling fill system. The emergency operating procedures will also be reviewed during the operating license stage of review.

On the basis of our review of the information submitted by Westinghouse, we conclude that (1) the loss-of-coolant analyses that were performed conservatively represent the RESAR-3S design and are in conformance with the requirements of Appendix K to 10 CFR Part 50, (2) the emergency core cooling system performance conforms to the peak clad temperature and maximum oxidation and hydrogen generation criteria of 10 CFR Part 50.46, (3) the emergency core cooling system performance will be adequate in the event of any postulated failure of a single component and (4) adequate systems are available to provide long term core cooling. Therefore, we conclude that the design of the RESAR-3S emergency core cooling system is acceptable.

The boric acid injection portion of the emergency core cooling system has been designed to deliver concentrated boric acid solution to the reactor coolant system to control the reactivity insertion following a postulated steam line break. While the concentrated boric acid solution is being injected into the reactor coolant system, the shrinkage caused by the cooldown following a steam line break will be made up by the water taken from the refueling water storage tank.

The postulated steam line break analysis, which is provided in Section 15.4 of RESAR-3S, indicates that although limited fuel cladding damage is permissible for a Condition IV accident (defined as a limiting fault in Section 15.1 of this report), the minimum departure from nucleate boiling ratio does not go below 1.30. We conclude that fuel damage will not occur from the main stram line break accident.

The boric acid injection portion of the emergency core cooling system will include the valves, pumps, tanks, and recirculation equipment needed to provide reactivity control in the event of a steam line break. We have reviewed the drawings, component descriptions, performance analysis, design criteria and interface information and have concluded that the boric acid portion of the emergency core cooling 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 failure of an active component and that no fuel damage will occur. On this basis, we conclude that the proposed design of the boric acid portion of the emergency core cooling system is acceptable.

6.3.5 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 applicants referencing RESAR-3S 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.

 711215 ~~718 052~~

(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, applicants referencing RESAR-3S 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, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps," 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors," and 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors." The emergency core cooling system components will be designed and fabricated to permit inspection and inservice tests in accordance with Section XI of the ASME Code. We find this acceptable.

6.3.6 Conclusion

The emergency core cooling system will include the valves, pumps, motors, and instrumentation needed to provide protection for the loss-of-coolant accident. We have reviewed the drawings, component descriptions, design criteria, performance analyses, and interface information and have determined that the RESAR-3S emergency core cooling system design conforms to the Commission's requirements as set forth in the General Design Criteria, regulatory guides, and staff technical positions as cited above. We, therefore, conclude that the proposed design of the emergency core cooling system is acceptable.

6.4 Engineered Safety Features Materials

We have reviewed the mechanical properties of materials selected for the emergency core cooling system and find that they will satisfy Appendix I of Section III and Parts A, B, and C of Section II of the ASME Code and our 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 system satisfy the recommendations of Regulatory Guides 1.31, "Control of Stainless Steel Welding," 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel," 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," and 1.44, "Control of the Use of Sensitized Stainless Steel." 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-3S must show that the controls on the hydrogen ion concentration 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-3S, in accordance with the recommendations of Regulatory Guide 1.7.

We have reviewed the selection of materials proposed for the emergency core cooling system 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 conclude 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 ASME Code, the recommendations of the regulatory guides mentioned above, and with our 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.

711247

~~718~~ 054

7.0 INSTRUMENTATION AND CONTROLS

7.1 General

The RESAR-3S instrumentation and control systems have been reviewed using the Commission's General Design Criteria, applicable standards of the Institute of Electrical and Electronics Engineers (IEEE), applicable regulatory guides, and staff technical positions as bases for evaluating their adequacy. The documents used in the review are listed in Table 7-1.

7.1.1 Interface Information

We have reviewed the interface information provided in RESAR-3S for the instrumentation and controls associated with the proposed design. We have found that the interface information and criteria contained in RESAR-3S, as supplemented by the additional interface requirements included in this report, provide reasonable assurance that the balance-of-plant design can be accomplished in a manner that will validate the assumptions in Section 15 of RESAR-3S. Based on the above, we conclude that the instrumentation and control systems specified in RESAR-3S can be implemented in an acceptable manner.

We have identified in the sections that follow interface information in addition to that provided in RESAR-3S. Our interface acceptance criteria for specific RESAR-3S systems are listed in Table 7-2 of this report. We will review the implementation of each interface requirement specified in RESAR-3S as supplemented by the additional interface requirements included in this report during our review of applications referencing RESAR-3S to ascertain that these requirements are satisfied.

7.2 Reactor Trip System

The RESAR-3S 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 throughout the input cabinet. The logic circuits can be connected to produce various logic combinations such as "two-out-of-four" and "one-out-of-two." 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

7112418

718 055

TABLE 7-1

DOCUMENTS USED IN THE
REVIEW OF INSTRUMENTATION AND
CONTROLS SYSTEMS

1. Westinghouse Reference Safety Analysis Report, RESAR-3S

2. 10 CFR Part 50 and Appendix A to 10 CFR Part 50, "General Design Criteria for Nuclear Power Plants"

3. Regulatory Guides, Division 1, Power Reactors

4. Electrical, Instrumentation and Control Systems Branch technical positions

5. Institute of Electrical and Electronics Engineers (IEEE) standards

   a. IEEE Standard 279-1971 - "Criteria for Protection Systems for Nuclear Power Generating Stations"
   b. IEEE Standard 308-1971 - "Criteria for Class IE Electric Systems for Nuclear Power Generating Stations"
   c. IEEE Standard 323-1974 - "IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations"
   d. IEEE Standard 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 Standard 336-1971 - "Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations"
   f. IEEE Standard 338-1971 - "Trial Use Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems"
   g. IEEE Standard 379-1972 - "Trial Use Guide for the Application of the Single Failure Criterion to Nuclear Power Generating Station Protection Systems"
   h. IEEE Standard 38?-1972 - "Trial Use Guide for Type Test of Class I Electric Valve Operators for Nuclear Power Generating Stations"
   i. IEEE Standard 384-1974 - "Trial Use Standard: Criteria for Separation of Class IE Equipment and Circuits"

711249

~~718~~ 056

TABLE 7-2

INTERFACE ACCEPTANCE CRITERIA FOR INSTRUMENTATION AND CONTROL SYSTEMS

CRITERIA	TITLE	RTS: REACTOR TRIP	RTS: ALL OTHER TRIPS	ESF: ECCS	ESF: [illegible]	ESF: ESFAS	ESF: CONTAINMENT SYSTEMS (BOP)	SYSTEMS FOR SAFE SHUTDOWN: [illegible]	SYSTEMS FOR SAFE SHUTDOWN: AUXILIARY FEEDWATER SYSTEM (BOP)	POST-ACCIDENT MONITORING SYSTEM PAMS	OTHER SYSTEMS REQUIRED FOR SAFETY: ECCS SWITCHOVER	OTHER SYSTEMS REQUIRED FOR SAFETY: [illegible]	OTHER SYSTEMS REQUIRED FOR SAFETY: RHR ISOLATION VALVES	OTHER SYSTEMS REQUIRED FOR SAFETY: INSTRUMENTATION AND CONTROL POWER SUPPLY		REACTOR CONTROL, AND CONTROL PRESSURIZER PRESSURE & WATER LEVEL CONTROLS, STEAM GENERATOR WATER LEVEL CONTROL, STEAM DUMP CONTROL AND IN-CORE INSTRUMENTATION
10 CFR PART 50	CONTENTS OF APPLICATION TECHNICAL INFORMATION	X	X	X	X	X	X	X	X	X	X	X	X	X		X
	TECHNICAL SPECIFICATIONS	X	X	X	X	X	X	X	X	X	X	X	X	X		X
	CODES AND STANDARDS	X	X	X	X	X	X	X	X	X	X	X	X	X		X
GENERAL DESIGN CRITERIA (GDC) APPENDIX A TO 10 CFR PART 50	(SEE STANDARD REVIEW PLAN TABLE 7-1 FOR SPECIFIC GDC & TITLE)	X	X	X	X	X	X	X	X	X	X	X	X	X		X
IEEE STANDARDS: IEEE STD 279-1971 (ANSI N42.7-1972)		X	X	X	X	X	X	X	X	X	X	X	X	X		
[illegible]		[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]		
REGULATORY GUIDES (RG): [illegible]		[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]		[illegible]
BRANCH TECHNICAL POSITIONS (BTP): [illegible]		[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]		[illegible]

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 reactor trips are 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 undervoltage
(13) Reactor coolant pump bus underfrequency
(14) Low-low steam generator water level
(15) Turbine trip
(16) Safety injection system actuation
(17) Low feedwater flow
(18) Manual

We have reviewed the descriptive information for the reactor trip system, including functional logic diagrams, testing provisions, bypass features, interface information, design criteria, design bases, and the analysis provided by Westinghouse on the adequacy of these criteria, bases, and interface information. On the basis of our review, we conclude that the design of the reactor trip system satisfies our requirements identified in Section 7.1 of this report and is acceptable.

The sensors for the reactor coolant pump bus undervoltage and underfrequency trips are not in the RESAR-3S scope. It is our position 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 undervoltage and underfrequency trip inputs, including the sensors, be designed to satisfy all the requirements of IEEE Standard 279-1971. Specifically, we will require that the balance-of-plant design for the reactor coolant pump undervoltage and underfrequency trip inputs and other reactor trip system interfaces satisfy the following requirements:

(1) The reactor coolant pump undervoltage and underfrequency relays and their associated instrument transformers and related connections shall be qualified for Class IE service and be installed in a seismic Category I structure.

711251

718 058

(2) The reactor coolant pump undervoltage and underfrequency trip inputs shall satisfy all other criteria identified in Table 7-2 of this report.

(3) The other reactor trip system interfaces in the balance-of-plant scope shall satisfy all the criteria identified in Table 7-2 of this report.

7.3 Engineered Safety Features Systems

The engineered safety features systems will be initiated and controlled by the engineered safety features actuation system which is within the scope of RESAR-3S. 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 portion. Each of the digital logic trains will be capable of actuating the required redundant engineered safety features systems.

The engineered safety features actuation system will initiate the following functions:

(1) A reactor trip, provided one has not already been generated by the reactor trip system.

(2) Cold leg injection isolation valves which are opened for injection of borated water by safety injection pumps into the cold legs of the reactor coolant system.

(3) Charging pumps, safety injection pumps, residual heat removal pumps, and associated valving which provide emergency makeup water to the cold legs of the reactor coolant system following a loss-of-coolant accident.

(4) Containment air recirculation fans and filtration system which serve to cool the containment and limit the potential for release of fission products from the containment by reducing the pressure following an accident.

(5) Those pumps outside the scope of RESAR-3S which serve as part of the heat sink for containment cooling, such as service water and/or component cooling water pumps.

(6) Motor driven auxiliary feedwater pumps.

(7) Phase A containment isolation, whose function is to prevent fission product release.

(8) Steam line isolation to prevent the continuous, uncontrolled blowdown of more than one steam generator and thereby uncontrolled reactor coolant system cooldown.

711252

718 059

(9) Main feedwater line isolation as required to prevent or mitigate the effect of continued feedwater flow.

(10) Start the emergency diesels to assure backup supply of power to emergency and supporting systems components.

(11) Isolate the control room intake ducts to meet control room occupancy requirements following a loss-of-coolant accident.

(12) Containment spray actuation which performs the following functions:

(a) Initiates containment spray to reduce containment pressure and temperature following a loss-of-coolant or steamline break accident inside of containment.

(b) Initiates Phase B containment isolation which isolates the containment following a loss of reactor coolant accident or a steam or feedwater line break within containment to limit radioactive releases. (Phase B isolation together with Phase A isolation results in isolation of all but the safety injection and spray lines penetrating the containment.)

We have reviewed the design description of the engineered safety features actuation system including functional logic diagrams, testing provisions, bypass features, design criteria, design bases, and the analysis provided by Westinghouse on the adequacy of these criteria and bases and conclude that they are acceptable.

The containment isolation system, steam line isolation system, containment spray system, and the auxiliary feedwater system are outside the RESAR-3S scope. We will require that the balance-of-plant design satisfy the specific interface requirements for the electrical instrumentation and controls associated with the engineered safety features systems as identified in Table 7-2.

7.3.1 Emergency Core Cooling System

Westinghouse has identified six 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. We determined that these valves did not originally meet the single failure criterion in that electrical malfunctions could result in spurious valve movements to undesirable positions and thereby result in loss of capability of the emergency core cooling system to perform its intended safety function. In lieu of design changes, Westinghouse subsequently elected to lock out power to these valves with power to be restored by manual action at the motor control centers.

It is our position that when lockout of power to valves that are required to open or close in various safety system operational sequences is elected in lieu of design changes in order to meet the single failure criterion, (1) capability be provided

711253

718 060

to lockout and restore motive power to the valves from the main control room and (2) redundant position indication be provided in the main control room. This position is documented in Electrical, Instrumentation and Controls Systems Branch Technical Position EICSB 18, "Application of the Single Failure Criterion to Manually-Controlled Electrically-Operated Valves," which is contained in Appendix 7-A of the Standard Review Plan.

Consequently, Westinghouse included the capability to lockout and restore motive power to the six affected valves from the main control room and to provide in the main control room redundant indication of the positions of these valves. We have determined that with this modification the design meets the single failure criterion and conclude, therefore, that it is acceptable.

7.3.2 Changeover from Injection to Recirculation Mode

The proposed design of the RESAR-35 emergency core cooling system incorporates automatic initiation of switchover from the injection mode to the recirculation mode whereby the containment sump isolation valves will be automatically opened on receipt of a refueling water storage tank low level signal in conjunction with a safety injection signal. Operator action will be required to realign the charging and safety injection pumps for the recirculation mode and to close the suction line valves from the refueling water storage tank.

We are concerned that when emergency core cooling system changeover functions are dependent on operator action, the operator might not correctly perform the safety function within the required time period. Further, we require that the instrumentation and controls provided to accomplish the changeover to the recirculation mode be designed to meet IEEE Standard 279-1971.

We were particularly 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 will be provided to close the refueling water storage tank suction check valves during the changeover and that failure 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.

The question of completely automatic switchover is a generic con[illegible]h is under discussion between Westinghouse and us. We have determined that com[illegible]ely automatic changeover is technically feasible. Westinghouse has committed to incorporate in the emergency core cooling system design any changes which may result consequent to the resolution of this generic concern.

On the basis of this commitment and our review of the design information and logic diagrams, we conclude that the design for the emergency core cooling system changeover from injection to recirculation mode satisfies our requirements referred to in Section 7.1 of this report and is acceptable.

We will require that those portions of the emergency core cooling system related to changeover to recirculation that are outside the scope of RESAR-3S meet the design criteria specified in Table 7-2 of this report.

7.3.3 Boric Acid Injection Portion of the Emergency Core Cooling System

It is our position that redundant heat tracing and redundant safety grade temperature monitoring systems be provided on all systems containing high concentration boric acid solution when those systems are relied upon to mitigate the consequences of an accident. Also included are components through which the solution is recirculated in order to prevent precipitation.

The temperature monitoring system to be provided must indicate and alarm in the control room any deviation from the temperature control band at selected locations in the system. This ensures that the temperature of the solution is above the precipitation temperature for the concentration present in the system. The monitoring system should satisfy the requirements of IEEE Standard 279-1971.

The heat tracing is outside the scope of RESAR-3S, however, Westinghouse has provided interface criteria requiring one hundred percent redundant heat tracing for all piping, valves, and flanges. Westinghouse is providing the required temperature monitoring described above for those parts of the system that are within the scope of RESAR-3S.

We have reviewed the design description of the boric acid injection portion of the emergency core cooling system including functional logic diagrams, testing provisions, bypass features, interface information, design criteria, and design bases and the analyses provided by Westinghouse on the adequacy of these criteria and bases. Based on this review we conclude that the electrical instrumentation and controls within the scope of RESAR-3S associated with the boric acid injection portion of the emergency core cooling system conform to our requirements and are, therefore, acceptable.

In addition to meeting the interface requirements specified in RESAR-3S, for applications which reference RESAR-3S, we will require that the heat tracing satisfy the following requirements:

(1) The temperature monitoring system shall be consistent with all the safety criteria implemented in the boric acid injection portion of the emergency core cooling system itself.

(2) Should the heat tracing be designed using redundant emergency power sources, such a design should not compromise the physical and electrical independence requirements between the plant redundant engineered safety features power sources and should satisfy the recommendations of Regulatory Guide 1.75, "Physical Independence of Electr[illegible] systems."

711255 ~~718 062~~

In addition, we will require that those parts of the boric acid injection portion of the emergency core cooling system which are outside the scope of RESAR-3S meet the design criteria specified in Table 7-2 of this report.

7.3.4 Periodic Testing of Reactor Trip System and Engineered Safety Features Actuation 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, "Periodic Testing of Protection System Actuation Functions." We have reviewed the documentation on periodic testing of the engineered safety features actuation system up to the final actuated devices, such as pumps, valves, and breakers, and conclude that the design conforms to Regulatory Guide 1.22. The review to determine compliance with Regulatory Guide 1.22 for the testing of these final actuated devices will be covered in our review of applications which reference RESAR-3S.

Westinghouse is developing a program for response time testing of sensors for the reactor protection system and engineered safety features excluding the nuclear detectors. We will review the program during our review of the final design. Technical specifications are provided in RESAR-3S requiring response time testing of the reactor trip system and engineered safety features.

We conclude that the criteria for the periodic testing of protection systems satisfy the requirements identified in Section 7.1 of this report and are acceptable.

7.3.5 Main Steam System

The Westinghouse analysis of the postulated main steam line break accident assumes a blowdown of no more than one steam generator. The interface information provided in Section 10.1 of RESAR-3S stipulates that a failure of any main steam line or malfunction of a valve installed therein must not cause uncontrolled flow from more than one steam generator. To validate these assumptions, Westinghouse has identified (in Table 15.4-7 of RESAR-3S) equipment and circuits required in the recovery from a high energy line rupture. Most of the required equipment and circuits are outside the RESAR-3S scope.

In specific project applications referencing RESAR-3S, we will require that the design for the main steam system satisfy the following requirements as identified in RESAR-3S.

(1) The electrical instrumentation and controls for the power operated relief valves must be independent and designed such that no single failure can cause opening of more than one power operated relief valve.

~~718 063~~

711256

(2) Any single failure in the electrical instrumentation and controls for the main steam isolation valves should not cause a failure of valves downstream of the main steam isolation valves.

(3) Failure in any single valve in either the upstream or downstream side of the main steam isolation valves should not result in steam flow in excess of the amount established in RESAR-3S accident analysis.

We will review applications referencing RESAR-3S to determine that the electrical instrumentation and controls meet the interface requirements identified in RESAR-3S and to assure that no single instrumentation or controls failure will result in the blowdown of more than one steam generator.

7.4 Systems Required for Safe Shutdown

Westinghouse has identified the following capabilities as being required for safe shutdown: boration, residual heat removal, and auxiliary feedwater, the latter of which is not in the scope of RESAR-3S. Also, Westinghouse has included a list of instrumentation and controls for systems in the RESAR-3S scope, in addition to other design features, that will 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 described in applications utilizing RESAR-3S 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. Areas of particular interest are discussed below.

7.4.1 Residual Heat Removal System

We determined that the RESAR-3S residual heat removal system design did not originally meet the single failure criterion in that one of the two isolation valves in the suction lines of each of the two trains were powered from the same source, the failure of which would have prevented the operation of both residual heat removal system trains. Westinghouse proposed utilizing the auxiliary feedwater system along with the steam generator power-operated relief valves as backup to the residual heat removal system.

It is our position that the system provided to remove residual heat be capable of reducing the reactor coolant temperature to a cold shutdown value (200 degrees Fahrenheit or less) within a reasonable period of time (on the order of a day) with either only onsite power available and assuming the most limiting single failure. The auxiliary feedwater system along with the steam generator power-operated relief valves, however, would not be capable of reducing the reactor coolant temperature to a cold shutdown value within a reasonable period of time.

Consequently, Westinghouse specified as an interface requirement in RESAR-3S that the balance-of-plant design include provisions for supplying Class IE electrical power to the residual heat removal system suction isolation valves in such a manner that the single failure criterion is satisfied for both system operation and isolation. We find this to be acceptable. In addition, however, Westinghouse has described in RESAR-3S a temporary power supply arrangement as a means of accomplishing this interface requirement. We have not reviewed Westinghouse's proposed temporary power supply arrangement since the specific design will be provided by the balance-of-plant designer and, therefore, will be reviewed in applications referencing RESAR-3S.

We have determined that with Westinghouse's suction line isolation valve power interface requirement, the RESAR-3S residual heat removal system design complies with the requirements of Criteria 19 and 34 of the General Design Criteria.

7.4.2 Instrumentation for Safe Shutdown

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, Westinghouse has identified instrumentation for monitoring the steam generator pressure, steam generator level, pressurizer pressure, pressurizer level, and direct monitoring of reactor coolant system temperature and interface requirements for local instrumentation for the auxiliary feedwater system, componenet cooling water system, and service water system as necessary instrumentation.

We have reviewed the design description of the instrumentation required for safe shutdown including interface information, design criteria, and design bases. Based on this review we conclude that the electrical equipment, instrumentation, and controls within the scope of RESAR-3S associated with the RESAR-3S instrumentation required for safe shutdown comply with our requirements and are, therefore, acceptable.

In addition to meeting the interface requirements specified in RESAR-3S, we will require that those portions of the systems required for safe shutdown outside the scope of RESAR-3S meet the design criteria specified in Table 7-2 of this report.

7.4.3 Conclusion

We have reviewed the design description of the systems required for safe shutdown including functional logic diagrams, testing provisions, bypass features, interface information, design criteria, and design bases and the analysis provided by Westinghouse on the adequacy of these criteria and bases. Based on this review we conclude that with the exception of the residual heat removal system, the electrical equipment, instrumentation, and controls within the scope of RESAR-3S associated with the RESAR-3S systems required for safe shutdown conform to our requirements and are therefore acceptable.

~~718 065~~

711258

In addition to meeting the interface requirements specified in RESAR-3S, we will require that those portions of the systems required for safe shutdown outside the scope of RESAR-3S meet the design criteria specified in Table 7-2 of this report.

7.5 Safety-Related Display Instrumentation

The safety-related display instrumentation will provide information to enable the operator to perform the required manual safety actions and to determine the effect of those actions after a reactor trip. The readouts will include those necessary for post-accident surveillance and those required to maintain the plant in a hot shutdown condition or to proceed to cold shutdown.

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. The design of the automatic bypass indication of a protective function at the system level is outside the design scope of RESAR-3S. We require that applications which reference RESAR-3S provide a system for automatic bypass indication of safety-related functions consistent with the recommendations of Regulatory Guide 1.47, "Bypassed and Inoperable Status Indications for Nuclear Power Plant Safety Systems."

The Westinghouse design will provide, for safety-related RESAR-3S scope equipment, those devices necessary for bypass and status indication and the design will interface with the balance-of-plant design such that the overall design is consistent with the recommendations of Regulatory Guide 1.47.

We have reviewed the design description, design criteria, interface information, and analyses of the manner in which the design of the safety-related display instrumentation will conform to the proposed design criteria. We conclude that the design of the safety-related display instrumentation conforms to our requirements referred to in Section 7.1 of this report and is, therefore, acceptable.

In addition to meeting the interface requirements specified in RESAR-3S, we will require that those portions of the safety-related display instrumentation outside the scope of RESAR-3S meet the design criteria specified in Table 7-2 of this report as well as the additional requirements stated above.

7.6 Other Instrumentation Systems Required for Safety and Other Safety-Related Matters

7.6.1 Environmental and Seismic Qualification

Westinghouse originally referenced a number of topical reports in RESAR-3S with regard to the environmental and seismic qualification of instrumentation, controls, and electrical equipment important to safety. We found a number of these referenced topical reports to be unacceptable. Therefore, we required that Westinghouse commit to a satisfactory program for demonstrating the environmental qualification of instrumentation and electrical equipment important to safety.

In response to this requirement, Westinghouse has committed to qualify the instrumentation, controls, and electrical equipment important to safety in the RESAR-3S scope, to the requirements of IEEE Standard 323-1974, "IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations" including the Nuclear Power Engineering Committee's position statement of July 24, 1975 by an appropriate combination, acceptable to us, of any or all of the following: type testing, operating experience, qualification by analysis, and ongoing qualification. In addition, Westinghouse has submitted Topical Report WCAP-8587, "Environmental Qualification of Westinghouse NSSS Class IE Equipment." This report describes the Westinghouse program for demonstrating the environmental qualification of instrumentation and electrical equipment important to safety. We are currently evaluating the test methods and procedures to be adopted by Westinghouse as described in WCAP-8587 to satisfy the objective of IEEE Standard 323-1974 with regard to the environmental qualification of instrumentation, controls, and electrical equipment important to safety to assure the operability of essential systems following such events as a loss-of-coolant accident or a main steam line break accident inside containment.

With regard to the seismic qualification of seismic Category I instrumentation and electrical equipment in the RESAR-3S scope, Westinghouse has referred to a number of topical reports. A recent addition to the list of references is Topical Report 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 our requirements for 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. Westinghouse has, however, committed to make the required seismic tests conform to the procedures specified in IEEE Standard 344-1975, "IEEE Guide for Seismic Qualification of Class I Electrical Equipment for Nuclear Power Generating Stations." These test procedures account for multi-axis and multi-frequency effects of seismic excitation and fatigue effects caused by a number of operating basis earthquake events. We find this commitment to be acceptable.

We conclude that the above commitments for qualification of Class IE equipment in the RESAR-3S scope of supply will facilitate the development of a seismic qualification program consistent with the objectives established in IEEE Standard 323-1974 and that they provide an acceptable basis for the preliminary design approval of the Class IE equipment seismic qualification.

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-3S, respectively. We conclude that these criteria meet the requirements of IEEE Standard 384-1974 as augmented by Regulatory Guide 1.75 and consider the proposed design acceptable.

For each application which utilizes RESAR-3S, we require that the balance-of-plant design criteria for the separation of redundant safety-related equipment and their physical identification also satisfy the above stated requirements.

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 above the safety injection unblock pressure and for redundant and independent indicating systems for each valve. The interlocks will be testable and will meet the applicable qualification test standards for safety equipment.

To meet the single failu . terion for electrically-operated valves, Westinghouse has elected to lockout power the accumulator isolation valves when the reactor is at power. For the purpose of check valve leak testing, one accumulator at a time may be isolated provided the reactor is held in a subcritical condition. The technical specifications will require the valve to be reopened before a return to criticality is permitted.

We conclude that the proposed design of the control circuits for the mulator isolation valves satisfies our requirements referred to in Section 7.1 of this report and meets the positions of Electrical, Instrumentation and Control Systems Branch Technical Position EICSB 4, "Requirements on Motor-Operated Valves in the ECCS Accumulator Lines," which is contained in Appendix 7-A of the Standard Review Plan, and is, therefore, acceptable.

In addition to meeting the interface requirements specified in RESAR-3S, for applications which ref nce RESAR-3S, we will require that the balance-of-plant instrumentation and controls for these isolation valves satisfy the criteria for the emergency core cooling system identified in Table 7-2 of this report.

7.6.4 Residual Heat Removal System Overpressure Protection Interlocks

Because of the potentially severe consequences of overpressurization of the residual heat removal systems, we require a high degree of assurance that the low pressure residual heat removal system be isolated from the high pressure in the reactor coolant system.

It is our position, therefore, that, in addition to satisfying the requirements of Criterion 34 of the General Design Criteria in the residual heat removal function, the overpressurization protection of the residual heat removal system from the reactor coolant system be assured by providing two motor-operated isolation valves in series on each inlet line between the high pressure reactor coolant system and the low pressure residual heat removal system. The initial design proposed in RESAR-3S did not conform with our position for a high pressure to low pressure interface with regard to providing interlocks on these isolation valves.

In the proposed initial design, the redundant isolation valves were to have been separately interlocked with independent pressure signals to prevent their being opened when the reactor coolant system pressure is greater than 425 pounds per square inch, guage, and automatically closed when a predetermined pressure is exceeded. Each valve in the same train was to have been powered by a separate engineered safety feature bus and have individual control circuitry. Westinghouse maintained that this protection interlock design in conjunction with the independence between the power and control circuitries for the redundant isolation valves provided adequate protection. With regard to the testability of these interlock signals, Westinghouse stated that the pressure interlock signal and logic could be tested online up to the slave relay which provides the signal to the valve control circuit without adversely affecting safety. We determined that this design did not comply with our requirement for diversity of pressure interlocks.

Consequently, we required that the pressure interlocks provided for the redundant isolation valves be derived from pressure signals using diverse principles. Westinghouse subsequently modified their design to provide diverse "prevent open" and "autoclose" interlocks. We find this to be acceptable and conclude that the electrical instrumentation and controls associated with overpressure protection of the residual heat removal system are acceptable.

In applications which reference RESAR-3S, we will require that the balance-of-plant design satisfy the following interface requirements:

(1) To maintain the electrical power independence and pressure interlock independence for residual heat removal system isolation valves, the power assignment for the redundant trains of the residual heat removal system shall satisfy the interface requirements provided in Table 8.1-2 of RESAR-3S.

(2) The balance-of-plant interfaces for the residual heat removal system at the nuclear steam supply system boundary shall satisfy all the criteria identified in Table 7-2 of this report.

7.6.5 Conclusion

We have reviewed the design description of all other instrumentation systems required for safety including functional logic diagrams, testing provisions, bypass features, interface information, design criteria, and design bases, and the analysis provided by Westinghouse on the adequacy of these criteria and bases. Based on this review we conclude that the electrical equipment, instrumentation, and controls within the scope of RESAR-3S associated with the RESAR-3S instrumentation required for safety conform to our requirements and are, therefore, acceptable.

In addition, we will require that those portions of the instrumentation systems required for safety outside the scope of RESAR-3S meet the design criteria specified in Table 7-2 of this report.

718 069

7.7 Control Systems Not Required for Safety

The following control systems which are not required for safety are identified in RESAR-3S: reactor control, rod control, plant con ol system interlocks, pressurizer pressure control, pressurizer water level control, eam generator water level control, steam dump control, and incore instrumentation. Westinghouse has documented no major differen s in the instrumentation and controls for the above systems and those provided in its previous designs.

Based on our review of the preliminary design of the RESAR-3S control systems, we conclude 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 conditions more severe than those for which the safety systems are designed to protect against and that these control and instrumentation systems satisfy r requirements and are acceptable.

In addition to meeting the interface requirements specified in RESAR-3S, we will require that those portions of the RESAR-3S control systems outside the scope of RESAR-3S meet the design criteria specified in Table 7-2 of this report.

7.8 Anticipated Transients Without Scram

Our review of anticipated transients without scram is contained in Section 15.5.7 of this report.

711263

~~718 070~~

8.0 ELECTRIC POWER SYSTEMS

Except for the specification of certain interface requirements, the electric power systems are totally outside the RESAR-3S scope and will be described in applications which reference RESAR-3S.

8.1 Interface Requirements

We have identified in Table 8-1 of this report our interface acceptance criteria for electric power systems. These criteria will form the basis for our review of applications which reference RESAR-3S to determine overall design conformance with the staff's requirements.

Westinghouse has specified in Section 8 of RESAR-3S the alternating and direct current loads, load groupings, safety load sequencing, minimum number of independent power sources, and other electric power system requirements of the RESAR-3S design. It is our position that these interface requirements be satisfied in the balance-of-plant electric power system design to validate the assumptions made in the RESAR-3S accident analysis and to provide an acceptable basis for our conclusion that the RESAR-3S design will satisfy the staff's requirements.

The following additional interface requirements shall form the basis of our review of each application which references RESAR-3S:

(1) Westinghouse states in Section 8.0 of RESAR-3S that the nuclear steam supply system is designed such that no fuel damage will occur if the plant sustains a credible grid decay rate of up to five Hertz per second based on an underfrequency reactor trip setpoint of 57 Hertz as specified in the technical specifications. Westinghouse further states that a lower underfrequency reactor trip setpoint will prevent fuel damage for credible grid decay rates of less than five Hertz per second.

We are currently evaluating Westinghouse Topical Report WCAP-8424, "An Evaluation of Loss of Flow Accidents Caused by Power System Frequency Transients in Westinghouse PWR's," which provides the bases for Westinghouse's statements. We find this is acceptable for the preliminary design stage of review. We will determine the acceptable grid decay rate limit, and, hence, the acceptable underfrequency reactor trip setpoint during the final design review stage.

For applications which reference RESAR-3S where credit is taken for the reactor coolant pump coastdown and the pumps must be disconnected on grid frequency excursions beyond the acceptable limits, we require that the balance-of-plant

718 071

721264

TABLE 8-1

INTERFACE ACCEPTANCE CRITERIA FOR ELECTRIC POWER SYSTEMS

CR. ERIA	TITLE	OFFSITE POWER SYSTEM	ONSITE A.C. POWER SYSTEM	ONSITE D.C. POWER SYSTEM
	CONTENTS OF APPLICATION			
10 CFR PART 50	TECH..CAL INFORMATION	X	X	X
	TECHNICAL SPECIFICATIONS	X	X	X
	CODES AND STANDARDS	X	X	X
GENERAL DESIGN CRITERIA (GDC) APPENDIX A TO 10 CFR PART 50	(SEE STANDARD REVIEW PLAN TABLE 8-1 FOR SPECIFIC GDC & TITLE)	X	X	X
IEEE STANDARDS				
IEEE STD 279-1971			X	X
IEEE STD 308-1971		X	X	X
IEEE STD 317-1972			X	X
IEEE STD 323-1974			X	X
IEEE STD 334-1974			X	
IEEE STD 336-1974		X	X	X
IEEE STD 338-1971			X	X
IEEE STD 344-1975			X	X
IEEE STD 379-1972			X	X
IEEE STD 382-1972			X	X
IEEE STD 383-1974			X	X
IEEE STD 384-1974			X	X
IEEE STD 387-1972			X	
IEEE STD 450-1972				X
REGULATORY GUIDES (RGS)				
RG 1.6			X	X
RG 1.9			X	
RG 1.22		X	X	X
RG 1.29			X	X
RG 1.30		X	X	X
RG 1.32		X		X
RG 1.40			X	
RG 1.41		X	X	X
RG 1.47		X	X	X
RG 1.53			X	X
RG 1.62			X	
RG 1.63		X	X	X
RG 1.68		X	X	X
RG 1.70		X	X	X
RG 1.73			X	X
RG 1.75			X	X
RG 1.81			X	X
RG 1.89			X	X
RG 1.93		X	X	X
BRANCH TECHNICAL POSITIONS (BTPs)				
BTP EICSB 1			X	X
BTP EICSB 2			X	
BTP EICSB 6				X
BTP EICSB 7			X	X
BTP EICSB 8			X	
BTP EICSB 10			X	X
BTP EICSB 11		X		
BTP EICSB 17			X	
BTP EICSB 21		X	X	X
BTP EICSB 27			X	X

711265

~~718 072~~

design be such that the reactor coolant pump breakers and the associated instrumentation and controls be designed and qualified in accordance with the requirements of IEEE Standards 279-1971 and 308-1971 including that the breakers be located in a seismic Category I structure.

(2) We are concerned about the spurious operation of thermal overload devices provided for safety-related motor operated valves which could negate the completion of the required safety functions of a system during an accident condition. Since RESAR-3S does not include the thermal overload protection criteria for safety-related motor operated valves and does not provide the criteria for the application of these devices, we require that the balance-of-plant design for thermal overload protection of safety-related motor operated valves in RESAR-3S satisfy either of the following requirements:

(a) Thermal overload protection of safety-related system motor operated valves shall have the trip setpoints set at a value high enough to prevent spurious trips due to design inaccuracies, trip set point drift, or variations in the ambient temperature at the installed location. The trip setpoint chosen shall be consistent with that of any branch circuit protective device used. Periodic tests are required and shall be performed on each of the thermal overload devices to verify the accuracy and reliability of the overload trip setpoint.

(b) Thermal overload protection may be bypassed under accident conditions and the bypass circuitry shall be designed to IEEE Standard 279-1971 criteria as appropriate for the rest of the safety-related system.

8.2 Conclusion

With the balance-of-plant design satisfying the interface criteria identified in Table 8-1 of this report, the requirements of RESAR-3S, and the additional requirements described above, we conclude that the electric power systems will be compatible with RESAR-3S safety systems functional requirements and will validate the assumptions made in the RESAR-3S accident analysis. A detailed review of the electric power systems will be performed on individual applications which reference RESAR-3S.

711266

718 073

9.0 AUXILIARY SYSTEMS

Auxiliary systems include such systems as the uel storage and handling facilities and systems; water systems; process auxiliaries, and air conditioning, heating, cooling, and ventilation systems. The auxiliary systems included in the RESAR-3S scope include the new and spent fuel storage racks, fuel handling system, chemical and volume control system, and the boron recycle system. In addition, Westinghouse has provided certain interface requirements for the station service water system; ultimate heat sink; component cooling water system; and the air conditioning, heating, cooling, and ventilation systems.

9.1 Fuel Storage and Handling

9...1 New and Spent Fuel Storage

New and spent fuel will be stored in racks, which are included in the RESAR-3S scope. Each fuel storage rack will be composed of individual vertical cells which can be fastened together in any number to form a module that can be firmly bolted to anchors in the floor of the fuel storage area. The new fuel racks will have a storage capacity of one-third of a core and the spent fuel racks will have a storage capacity of one and one-third cores. The new and spent fuel racks will be designed to seismic Category I requirements.

Unlike previously approved Westinghouse plants, the new and spent fuel racks will be located in a single underwater storage facility. Also, the minimum center-to-center spacing has been reduced from 21 inches to 14.2 inches. This spacing is sufficient to maintain the effective multiplication factor at a value of 0.95 or less for fuel having the highest anticipated enrichment and arranged in unborated water in the optimum moderation configuration.

We have reviewed the proposed design bases and interface requirements of the new and spent fuel storage racks and have determined that they meet the applicable positions set forth in Regulatory Guide 1.13, "Fuel Storage Facility Design Basis," and the requirements of Criterion 62 of the General Design Criteria. Therefore, we conclude that the proposed design of the new and spent fuel racks is acceptable.

The design of the fuel storage area will be described in applications referencing RESAR-3S.

9.1.2 Fuel Handling System

711267

The fuel handling system will consist of the equipment needed for the transfer of new and spent fuel between the fuel storage area and the reactor vessel. The fuel

718 074

handling system also provides for the safe disassembly, handling, and reassembly of the reactor vessel head during refueling operations.

The equipment within the RESAR-3S scope includes the refueling machine, fuel handling machine, new fuel elevator, fuel transfer system, and handling equipment. Other components of the fuel handling system will be described in applications referencing RESAR-3S. The fuel handling system will be designed such that in the event of a safe shutdown earthquake, the equipment will not fail in a manner that could affect safety-related equipment.

At our request, Westinghouse has provided the results of their analyses of the consequences of dropping the reactor vessel head assembly to show that core cooling capability will be maintained and core damage will be precluded. We have determined, however, that the analyses are incomplete in that they do not consider possible damage resulting from dropping the vessel head while it is enroute to the head laydown area.

Westinghouse has advised us that they are analyzing this accident. However, until we are able to determine that the consequences of this accident are acceptable, we have imposed an interface requirement that applicants referencing RESAR-3S provide an overhead reactor vessel head assembly handling system that is designed so that the connected load would not fall in the event of a single failure or malfunction. This handling system, or single failure proof crane shall be designed as a safety system and shall be designed, fabricated, installed, inspected, tested, and operated in accordance with the Auxiliary and Power Conversion Systems Branch Technical Position APCSB 9-1, "Overhead Handling Systems for Nuclear Power Plants," which is contained in Standard Review Plan 9.1.4. If Westinghouse provides additional information which demonstrates acceptable consequences from the postulated dropping of the reactor vessel head while it is enroute to the head laydown area, a single failure proof crane will not be required.

9.2 Water Systems

9.2.1 Service Water System and Ultimate Heat Sink

Westinghouse has provided, as interface requirements, the nuclear steam supply system heat loads that must be transferred from the component cooling water system to the service water system and to the ultimate heat sink. We have reviewed these heat loads for completeness and conclude they are acceptable and that they enable applicants which reference RESAR-3S to design an acceptable service water system and ultimate heat sink. These system designs will be evaluated in applications which reference RESAR-3S.

9.2.2 Component Cooling Water System

The component cooling water system will provide cooling water to dissipate heat from various components for both normal and post-accident operation. The component cooling water system design will be described by applicants utilizing RESAR-3S.

7712 68 ~~218 075~~

The RESAR-3S equipment that is to be cooled by component cooling water includes the four reactor coolant pumps, two residual heat removal heat exchangers, two residual heat removal pumps, the letdown heat exchanger, the excess letdown heat exchanger, the seal water heat exchanger, two centrifugal charging pumps, two safety injection pumps, the reciprocating charging pump, the letdown chiller, and the recycle evaporator.

Westinghouse has identified interface requirements for all components within the scope of RESAR-3S that require component cooling water. The interfaces are identified by pressure, temperature, heat transfer rate, flow rate, water chemistry, and the quality group and seismic classification for the RESAR-3S equipment at each physical interface connection.

The seals and bearings of the reactor coolant pumps require continuous cooling by the component cooling water system during all modes of operation. Termination of coolant flow to one or more pumps may lead to fuel damage due to a locke rotor. Therefore, we require that the design of that portion of the component cooling water system that supplies cooling water to the reactor coolant pumps satisfy the following criteria:

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

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

Therefore, in addition to meeting the component cooling water system interface requirements specified in RESAR-3S, we require that balance-of-plant applicants referencing RESAR-3S provide a component cooling water system that meets the following:

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

(2) The component cooling water supply to the reactor coolant pumps shall be capable of withstanding a single active failure or a moderate energy line crack as defined in Auxiliary and Power Conversion Systems Branch Technical Position APCSB 3-1 and be designed to seismic Category I, Quality Group C and ASME Code Section III, Class 3 requirements.

However, if Westinghouse can demonstrate that the reactor coolant pumps are capable of operating without cooling for longer than 30 minutes without loss of function and the need for operator protective action and if safety-grade instrumentation is provided to detect the loss of component cooling water to the reactor coolant pumps and to alarm in the control room, that portion of the component cooling water system that supplies cooling water to the reactor coolant pumps can be desi- ed to non-seismic Category I requirements and Quality Group D. The entire instrumentation system, including audible and visual statu indicators for loss of component cooling water should meet the requirements of IEEE Standard 279-1971.

The com onent cooling water system design will be reviewed in applications referencing RESAR-3S.

9.3 Process Auxiliaries

9.3.1 Chemical and Volume Control System

The chemical and volume control system will be designed to control and maintain the reactor coolant inventory and to control the boron concentration in the reactor coolant. In addition, purification of reactor coolant will also be accomplished by the process of demineralization and reactor coolant chemistry will be controlled th ugh 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 the reactor coolant system. The centrifugal charging pumps also serve as high head safety injection pumps in the emergency core cooling ystem. This function of the charging pumps is described in Section 6.3.2 of this repc .

Reactor coolant boron concentration will be controlled using the chemical and volume control system by adding makeup for either boration or dilution for the large reactivity changes needed during shutdown and startup, or by thermal regeneration 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 which 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 reduce the volume of waste reactor coolant that must be processed by the waste processing system.

The charging pumps w ll be available to supply borated water from the refueling water storage tank or from the boric acid tanks to maintain refueling boron concentration. The injection of the borated water will be through the normal charging path or, as a

711270 718 077

backup, through the seal injection path. All portions of the chemical and volume control system necessary for safe shutdown are designed to seismic Category I requirements.

Applicants utilizing RESAR-3S must provide adequate component cooling water at 105 degrees Fahrenheit or less to the letdown heat exchanger, excess letdown heat exchanger, and seal water heat exchanger. In addition, a reactor makeup water storage tank, resin fill tank, boric acid tanks, and reactor makeup water pumps must be provided as well as provisions for maintaining a temperature of 65 degrees Fahrenheit or greater for all portions of the system which will normally contain a nominal four percent boric acid solution.

We reviewed the adequacy of Westinghouse's proposed design criteria and design bases for performing the necessary functions of the chemical and volume control system during normal, abnormal, and accident conditions. We conclude that the design criteria and design bases are in conformance with the General Design Criteria and are, therefore, acceptable.

9.3.2 Boron Recycle System

The boron recycle system will be designed to collect and process the excess 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 are described in RESAR-3S. However, applicants utilizing RESAR-3S must provide designs for heat tracing and certain other equipment identified in Table 1.7 of RESAR-3S.

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 criteria and interface requirements for the boron recycle system. We determined that failures of the system will not affect safe plant operation or shutdown. We conclude that the boron recycle system design is acceptable.

9.4 Air Conditioning, Heating, and Ventilation Systems

The air conditioning, heating, and ventilation systems required for essential and nonessential areas outside containment will be provided by the balance-of-plant designer. However, at our request, Westinghouse provided temperature and humidity design interface requirements for those areas that will house RESAR-3S safety-related equipment.

7112271~~8 078~~

We have reviewed Westinghouse's range of temperature and humidity design interface requirements for areas housing safety-related equipment furnished as a part of RESAR-3S and conclude the information is adequate for an applicant referencing RESAR-3S to design the necessary air conditioning, heating, and ventilation systems. These systems will be evaluated in applications referencing RESAR-3S.

9.5 Fire Protection System

The fire protection system is outside the scope of RESAR-3S and, therefore, will be addressed in applications referencing RESAR-3S. The only aspects of fire protection within the scope of RESAR-3S are the specification of fire protection criteria for RESAR-3S scope equipment and interface requirements.

As a result of investigations and evaluations being conducted by the staff on nuclear power plant fire protection systems, we will request that Westinghouse re-evaluate its fire protection criteria and interface requirements in conformance with Auxiliary and Power Conversion Systems Branch Technical Position APCSB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants," which is contained in Standard Review Plan 9.5.1. If, as a result of our evaluation of this information, we determine that additional requirements are indicated to further improve the fire protection capability of the RESAR-3S design, we will require that they be implemented.

711272

718 079

10.0 STEAM AND POWER CONVERSION SYSTEM

10.1 General

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 is outside the RESAR-3S scope and will be designed by the balance-of-plant designer. The RESAR-3S design does not extend beyond the steam generator's feedwater and steam nozzles.

10.2 Interface Information

The auxiliary feedwater system is an engineered safety feature that will be designed by the balance-of-plant designers for applications utilizing RESAR-3S. This system is required to fulfill the requirements of Criterion 34 of the General Design Criteria when primary plant parameters are above the design capacity of the residual heat removal system. Westinghouse has provided the required system flow rates, pressure, temperature, initiation time, and other necessary interface information regarding the auxiliary feedwater system.

We are currently evaluating design and operating conditions that could result in damage to feedwater system piping as a consequence of feedwater flow instability occurrences such as occurred at Indian Point 2 on November 13, 1973. The results of this investigation may result in further requirements being imposed upon RESAR-3S and/or applications referencing RESAR-3S so that unacceptable damage will not result from water hammer effects. We will implement the results of our generic investigation at the final design review of RESAR-3S.

We have reviewed the interface information provided by Westinghouse and have determined that adequate information is provided to enable balance-of-plant designers to design a steam and power conversion system, including the auxiliary feedwater system, that will support the RESAR-3S nuclear steam supply system. On this basis, we conclude that the interface information provided by Westinghouse relative to the steam and power conversion system is acceptable. The steam and power conversion system design, including the auxiliary feedwater system design, will be evaluated in applications which reference RESAR-3S.

~~718 080~~

711273

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-3S, as defined in Amendment 1 to WASH-1341. However, RESAR-3S does provide as interface information 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-3S.

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 found that the parameters and calculations used to obtain primary coolant concentrations are consistent with those given in NUREG-0017, "Calculation of Releases of Radioactive Materials in Liquid and Gaseous Effluents from Pressurized Water Reactors." Westinghouse has also provided the primary coolant concentrations to be used for design of the radioactive waste management systems based on a value of one percent of the fuel having cladding defects. This value is acceptable.

11.2 Waste Management Systems

The radioactive waste management systems will be designed by the balance-of-plant designers for applications utilizing RESAR-3S. We have reviewed the interface information provided by Westinghouse and have determined that adequate information is provided to enable balance-of-plant designers to design radioactive waste management systems that will support the RESAR-3S nuclear steam supply system. On this basis, we conclude that the interface information provided by Westinghou · relative to the waste management systems is acceptable. The designs of the waste management systems will be evaluated in applications which reference RESAR-3S.

711274

718 081

12.0 RADIATION PROTECTION

12.1 General

Since the RESAR-3S design is limited to the nuclear steam supply system, only certain aspects of radiation protection are within the scope of RESAR-3S. These aspects include the source terms for RESAR-3S equipment, including the reactor coolant system, and those design aspects of the RESAR-3S equipment and interface requirements related to radiation protection. Our review of these areas is discussed below.

12.2 Assuring that Occupational Radiation Exposures Will Be as Low as Is Reasonably Achievable

We reviewed the policy, design, and operational considerations related to assuring that occupational radiation exposures will be as low as is reasonably achievable for the RESAR-3S design. The review considered descriptions of how experience from past designs and operating plants have been used to develop improved radiation protection designs for the nuclear steam supply system. It also included an evaluation of the implementation of the appropriate guidance provided in Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposure as Low as is Reasonably Achievable (Nuclear Power Reactors)," and information on proposed alternatives.

In Appendix 12A of RESAR-3S, Westinghouse provides radiation protection design considerations that are related to RESAR-3S equipment, and design recommendations for the balance-of-plant designer related to shielding, installation, and layout of the RESAR-3S equipment. These design considerations cover the following in various degrees of detail: the reactor; evaporators; pumps, filters, and demineralizers; tanks and heat exchangers; remote and/or automatic systems control operations; and the reactor coolant system.

We reviewed the RESAR-3S 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, and instructions to designers and engineers regarding design considerations for achieving this. 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 be as low as is reasonably achievable.

We have determined that Westinghouse has shown sufficient concern and familiarity with the as low as is reasonably achievable principles in the area of design considerations to conclude that this aspect of radiation protection is acceptable.

Appendix 12A of RESAR-3S 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 applications which reference RESAR-3S to include a complete Section 12 in order to demonstrate that adequate radiation protection will be provided from the radiation sources specified in RESAR-3S.

12.3 Radiation Sources

We reviewed RESAR-3S to evaluate information on radiation sources for RESAR-3S equipment, as they relate to in-plant radiation protection. This includes the description of the sources of radiation that will form the basis for the radiation protection program and be used in 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 include those that will necessitate shielding, special ventilation designs, traffic or access control considerations, special plans and procedures, and monitoring equipment.

Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," provides that airborne sources that can be created by leakage or release from a nuclear steam supply system, such as that described in RESAR-3S, 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 also be specified.

In Section 12.1.3 of RESAR-3S, Westinghouse provides four 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 which will 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 specified in Section 11 of RESAR-3S.

In our review of the source term section of RESAR-3S, we 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, applicants utilizing RESAR-3S will be required to obtain and provide the added information.

711276 ~~718~~ 083

12.4 Radiation Protection Design Features

The following areas of RESAR-3S relating to radiation protection design features were reviewed:

(1) The description of RESAR-3S equipment design features to be used for assuring that occupational radiation exposures will be as low as is reasonably achievable.

(2) Information concerning the implementation of Regulatory Guide 8.8 or proposed alternatives.

(3) The description of any special protection features within the scope of RESAR-3S that use shielding, geometric arrangement, or remote handling to assure that occupational radiation exposure will be as low as is reasonably achievable.

RESAR-3S provides information on shielding design objectives, and in Appendix 12A describes design considerations and features which Westinghouse recommends regarding equipment and related systems within the scope of RESAR-3S. Also described are design features of the RESAR-3S systems relating to radiation protection.

We reviewed the RESAR-3S 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 such as maintenance, refueling, radioactive material handling, processing, inservice inspection, and calibration have been considered in equipment design and that potential radiation exposure from these activities will be kept as low as is reasonably achievable by radiation protection features incorporated in the design. Acceptability of the shielding is based on factors which have not been supplied in RESAR-3S including shielding computational methods. Shielding will be reviewed for each application utilizing RESAR-3S.

The RESAR-3S supplied information on equipment design features for assuring that occupational radiation exposures will be as low as is reasonably achievable meets the guidance of Regulatory Guide 8.8 and the staff requirements. We have determined that Westinghouse has shown sufficient concern and familiarity with the as low as is reasonably achievable principles in the area of equipment design features to conclude that this aspect of radiation protection is acceptable.

The advice and guidance provided by Westinghouse in RESAR-3S 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 which reference RESAR-3S will include a determination of whether the appropriate guidance has been applied to the final design of the plant. We conclude that the Westinghouse guidance constitutes an acceptable basis for the design of these systems.

13.0 CONDUCT OF OPERATIONS

13.1 General

Since the RESAR-3S design is limited to the nuclear steam supply system, the only aspect of the conduct of operations considered are those features of the design that enhance industrial security and reduce the vulnerability of the plant to deliberate acts which may adversely affect the plant and public safety. The other aspects of the conduct of operations such as the organizational structure of the applicant, training program, emergency planning, review and audit, plant procedures, plant records, and the balance of industrial security considerations will be addressed in applications which reference RESAR-3S.

13.2 Industrial Security

We have reviewed the information in RESAR-3S related to those features of the design that afford protection to vital equipment against acts of industrial sabotage. We found that the design features of those systems and components important to safety, including the use of redundancy, automation, independence, diversity, and protection against common mode failures, also provide protection against acts of industrial sabotage. On this basis, we conclude that Westinghouse's design for the protection of the plant against acts of industrial sabotage are acceptable for the preliminary design stage of the review process. We will review the protection against sabotage provided by the balance-of-plant design in our review of applications which reference RESAR-3S.

14.0 INITIAL TESTS AND OPERATIONS

Section 14 of RESAR-3S describes Westinghouse's test program for the proposed RESAR-3S nuclear steam supply system. 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, post-core 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 provided in the regulatory guides applicable to initial test programs which are listed below.

(1) Regulatory Guide 1.41, "Preoperational Testing of Redundant On-Site Electric Power Systems to Verify Proper Load Group Assignments."

(2) Regulatory Guide 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors."

(3) Regulatory Guide 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors."

(4) Regulatory Guide 1.80, "Preoperational Testing of Instrument Air Systems."

On the basis of our review, we conclude that an acceptable startup and test program can be conducted without the need for design modifications. Because of changes in design that may result during refinement of the preliminary design and because of possible different interactions between the nuclear steam supply system and balance-of-plant portions of a plant, we will conduct a detailed review of the initial test program during the operating license review stage of each application which references RESAR-3S.

711279

~~748~~ 086

15.0 ACCIDENT ANALYSES

15.1 Summary

Westinghouse has performed safety analyses to evaluate the capability of the RESAR-3S nuclear steam supply system to withstand normal and abnormal operational transients and a broad spectrum of postulated accidents without undue risk to the health and safety of the public. The events considered include all relevant types discussed in Regulatory Guide 1.70, "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 are expected to 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 system which will automatically maintain prescribed conditions in the plant even under the most conservative set of reactivity parameters with respect to both system 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. For infrequent incidents, the plant should be designed to limit the release of radioactive material to assure that doses to persons offsite are limited to values which are a small fraction of 10 CFR Part 100 guideline values. A Condition III event will not generate a Condition IV fault or result in loss of function of the reactor coolant system or containment barriers.

711280

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

~~718 087~~

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.

Westinghouse's classification of events analyzed is itemized in Table 15-1 of this report.

15.2 Input Parameters and Analytical Techniques for Accident and Transient Analyses

15.2.1 Input Parameters

As part of our review of the RESAR-3S accident and transient analysis, we reviewed the assumptions and input parameters employed by Westinghouse in its analyses. A discussion of the more significant assumptions and input parameters follows in this section. Unless otherwise noted in this report, mathematical models and methods used by Westinghouse have been previously reviewed and found acceptable by the staff in conjunction with approved plants using a Westinghouse nuclear steam supply system.

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, 2.1 seconds to reach 85 percent of the rod travel, was based on previous measurements applicable to the 12 foot 17x17 rod cluster control assemblies. Instrument errors and time delays assumed for the analyses will be justified as part of the final design review of RESAR-3S.

Events initiated at full power were assumed to start at a core thermal power level of 3479 megawatts, which is 1.02 times the proposed core thermal power level in accordance with Regulatory Guide 1.49, "Power Levels of Nuclear Power Plants." 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 thermal. A value of 3579 megawatts thermal 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, loss of coolant, steam generator tube rupture, and fuel handling accidents. Although the loss of normal feedwater analysis does not involve radiological consequences, we conclude that analysis at 3579 megawatts thermal is a conservative evaluation of system design adequacy for operation at 3479 megawatts thermal and is acceptable for the preliminary design review. However, we require that the loss of normal feedwater analysis be performed at 3479 megawatts thermal in accordance with the guidance of Regulatory Guide 1.49 for the final design application.

As noted in Section 4.3 of this report, the values of the 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

TABLE 15-1

CATEGORIES OF TYPICAL TRANSIENTS AND FAULTS

Condition I - Normal Operation and Operational Transients

- Reactor startup
- Reactor shutdown
- Refueling operations
- Power operation

Condition II - Faults of Moderate Frequency

- 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
- Accidental depressurization of reactor coolant system
- Accidental depressurization of main steam system
- Inadvertent operation of emergency core cooling system during power operation

Condition III - Infrequent Faults

- 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 - Limiting Faults

- Control rod ejection
- Fuel handling accident
- Steam generator tube rupture
- Major secondary system pipe rupture
- Loss-of-coolant accident
- Single reactor coolant pump locked rotor

7112S2

~~718 089~~

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 delta T (temperature difference)	Variable (see Figure 15.1-1 of RESAR-3S)	6.0[1]
Overpower delta T (temperature difference)	Variable (see Figure 15.1-1 of RESAR-3S)	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 Undervoltage Trip	70 percent	1.5
Turbine Trip	Not applicable	1.0
Low-Low Steam Generator Level	33 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]	57 Hertz	0.1

[1]Total time delay, including resistance temperature detector bypass loop fluid transport delay, effect of bypass loop piping thermal capacity, resistance temperature detector 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

711283

718 090

considered with respect to reactivity coefficients, 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.

The departure from nucleate boiling calculations were performed using a critical heat flux multiplier of 0.86, thus providing a 14 percent design margin. Final judgment of the adequacy of this multiplier will be made after the completion of our review of the test programs discussed in Section 1.4 of this report for the final design review. This matter is discussed further in Section 4.4.1 of this report.

15.2.2 Analytical Techniques

We have reviewed and approved most of the analytical techniques used by Westinghouse in the RESAR-3S accident and transient analyses. Those for which we have not completed our review are described in the following topical reports:

WCAP-7898	Long Term Transient Analysis Program for Pressurized Water Reactors - BLKOUT
WCAP-7907	LOFTRAN Code Description
WCAP-7908	A FACTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod
WCAP-7909	A Digital Computer Code for Transient Analysis of a Multi-Loop PWR System - MARVEL
WCAP-7956	THINC-IV - An Improved Program for Thermal and Hydraulic Analysis of Rod Bundle Cores
WCAP-7973	Calculation of Flow Coastdown After Loss of Reactor Coolant Pump - PHOENIX
WCAP-7980	WIT-6 - Reactor Transient Analysis Computer Program Description

All of the analytical techniques used by Westinghouse in the RESAR-3S accident and transient analyses, including those for which we have not completed our review, have been used in applications utilizing Westinghouse reactors which we have approved. As discussed in the remainder of Section 15 of this report, we have determined that the margins predicted by these methods are acceptable for the issuance of a construction permit or a Preliminary Design Approval. We will complete our review of these methods prior to approval of the final design.

15.3 Technical Specification Limits Qualified by Accident and Transient Analyses

Results of the postulated accidents investigated are sensitive to the values 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 to be 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 12.6 kilowatts per foot, which is the value qualified by all of 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 incore 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 excore detector system. Axial power distribution will be controlled by control bank position and monitoring of flux difference between the top and bottom excore detectors.

The overtemperature and overpower delta T (temperature difference) trips and the high flux trip 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 four seconds) and axial peaks are below design values. The flux difference measurement will be incorporated in analog circuitry which will automatically reduce the overtemperature delta T (temperature difference) 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 undervoltage and underfrequency trips and 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-3S, we will include the following additional core operating limits in the technical specifications at the final design review stage: 711285

~~718 092~~

Parameter		Limit Value*
(1) Reactor Vessel Mimimum Coolant Flow	(a)	140.3 million pounds per hour at maximum core power level of 3411 megawatts thermal
	(b)	100.9 million pounds per hour with three loop operation at maximum core power level of 2396 megawatts thermal
(2) Core Coolant Average Temperature		589.4 degrees Fahrenheit maximum
(3) Pressure in the Pressurizer		2250 plus or minus 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.

15.4 Anticipated Transients

A number of plant transients can be expected to occur with moderate frequency as a result of equipment malfunction 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 listed in Table 15-1 of this report 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 we consider this as an anticipated transient and we evaluated it as such.

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 the ASME 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.)

*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.

711286

718 093

(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 integrity will not be jeopardized.

The most limiting transient 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). In the analysis of the loss-of-load transient, no credit has been taken for steam dump or operation of the secondary side power operated relief valves. Steam release has been assumed through the safety relief valves, whose sizing is discussed in Section 5.4.2.1 of this report. The calculated peak primary system pressure of 2550 pounds per square inch, absolute did not result in violation of the 110 percent pressure overpressure limit.

Reactivity can be added to the core by adding primary grade water to the reactor coolant system via the reactor makeup portion of the chemical and volume control system. Various chemical and volume control system malfunctions which could lead to an unplanned boron dilution incident have been reviewed. Westinghouse has analyzed boron dilution transients starting from plant conditions of startup, power operation (automatic and manual), hot standby, cold shutdown, and refueling.

We require that a minimum of 30 minutes be available from receipt of an alarm for operator action during a boron dilution accident during refueling or startup. As a result of our requirements, Westinghouse proposed certain changes. During refueling, Westinghouse will require that valves FCV-110B, FCV-111B, 8439, 8441, and 8453 of the chemical and volume control system be locked closed. This has been included in the RESAR-35 technical specifications.

This procedure will eliminate all possible direct paths for addition of unborated water to the reactor coolant system. The only remaining path is via the reactor water storage tank. The technical specifications will require sampling of the boron concentration before addition of this water. As an additional precaution, the source range instrumentation neutron high count rate will be alarmed in both the containment and the control room and a high source range flux level will be alarmed in the control room. Typically, the source range high flux alarm will be activated one decade above the count rate setting being used. Thus, not only will addition of unborated water be prevented, but an increase in the subcritical multiplication factor will be alarmed.

We conclude that the changes proposed by Westinghouse will provide adequate protection against a boron dilution accident during refueling or startup and are, therefore, acceptable.

711287

~~718~~ 094

For power operation in the manual control mode, the fuel will be maintained within thermal limits by the overtemperature delta T (temperature difference) trip. 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 15 minutes 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-3S. The THINC-IV[2] code was used to calculate the departure from nucleate boiling ratio. For the transient response to a droppped control rod or control rod bank, the LOFTRAN[3] code was used.

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 1.30 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.

We conclude that anticipated transients will not lead to more serious plant conditions and that the plant design is acceptable with respect to transient response to events that might occur during the plant lifetime.

15.5 Postulated Accidents

RESAR-3S 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.

[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. McIntyre, J. C. Buker, and R. P. Rose, "LOFTRAN Code Description," WCAP-7907, June 1972.

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 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.

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, to prevent overpressurizing of the reactor coolant system, and to prevent uncovering of the reactor core. The analysis indicated that the assumed auxiliary feedwater capacity of 470 gallons per minute minimum at 1300 pounds per square inch, guage 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.

We determined that the results presented for a major feedwater line break are not unlike those determined for comparable plants and, on this basis, conclude that they are acceptable and sufficient for the preliminary design review. We will review the methods used by Westinghouse prior to final design approval of RESAR-3S.

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 Methods," 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 as described in Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors."

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 from the standpoint of energy deposition was the end of cycle at 102 percent power case which resulted in an energy deposition of 150 calories per gram. The end of cycle zero power case produced the highest clad temperature which was 2650 degrees Fahrenheit. The analysis also shows that less than ten percent of the fuel goes through departure from nucleate boiling. As a result, gross fuel damage would not occur.

We have determined that the assumptions and methods of analysis used by Westinghouse are in accordance with Regulatory Guide 1.77. On this basis, 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 conditions which resulted in the most severe consequences were determined and evaluated.

~~718 097~~

711290

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 their respective design pressures. We determined that the results presented for the steam line break accident are not unlike those determined for comparable plants and, on this basis, conclude that they are acceptable and sufficient for the preliminary design review stage. We will review the methods used by Westinghouse prior to final design approval of RESAR-3S.

15.5.5 Spectrum of Piping Breaks Within the Reactor Coolant Pressure Boundary

Westinghouse has performed analyses of the performance of the emergency core cooling system in accordance with the requirements of Section 50.46 of 10 CFR Part 50. The analyses considered a spectrum of postulated break sizes and locations and were performed with the evaluation model described in the Appendix K to 10 CFR Part 50. We have reviewed this information and our evaluation is contained in Section 6.3.4 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. Westinghouse has classified this accident as a Condition IV event. We consider it to be in the infrequent incident category which requires that the plant be designed to limit the release of radioactive material to assure that doses to persons offsite are limited to values which are a small fraction of 10 CFR Part 100 guideline values. 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 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 their respective design pressures.

We conclude that the calculated consequences of a postulated reactor coolant pump rotor seizure are acceptable for the preliminary design review stage.

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 transients affected include loss of normal feedwater, loss of electrical load, inadvertent control rod withdrawal, and loss of normal electrical power.

In September 1973, we issued WASH-1270, "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors" establishing acceptance criteria for anticipated transients without scram. In conformance with the requirements of Appendix A to WASH-1270, Westinghouse submitted an evaluation of anticipated transients without scram in Topical Report WCAP-8330, "Westinghouse Anticipated Transients Without Trip Analysis."

On December 9, 1975, we issued our "Status Report on Anticipated Transients Without Scram for Westinghouse Reactors" in which we noted that certain additional analyses and justification of the Westinghouse analysis model are needed and that certain changes in typical Westinghouse plant designs are indicated. These design changes include the following:

(1) Diverse means of actuating turbine trip

(2) Diverse means of initiating the auxiliary feedwater system (or the analyses must be revised to exclude the effects of automatic initiation)

(3) Diverse means of isolating the containment (or the analyses must be revised to assume manual initiation ten minutes after the anticipated transient without scram event)

(4) Diverse means of interrupting power to the rod drive mechanisms on reactor scram

We have not completed our review of Westinghouse's anticipated transients without scram model. However, we have discussed with Westinghouse additional sensitivity studies which can be used to assess the uncertainty with regard to those portions of the model which we have not accepted. Westinghouse has advised us that the results of these latest sensitivity studies indicate that the anticipated transients without scram acceptance criteria limiting reactor coolant pressure boundary stresses to "Emergency Conditions," as defined by the ASME Code, can be met without any change in the design of the reactor coolant system. Therefore, while our review of the Westinghouse model is incomplete, we believe that provision of diverse means to trip scram breakers, actuate auxiliary feedwater systems, and trip the turbine following an anticipated transient without scram event would result in satisfying our acceptance criteria. In addition, applicants referencing the generic Westinghouse analyses would be required to provide plant specific analyses to demonstrate that (1) automatic containment isolation is not necessary to meet the limits, (2) the auxiliary feedwater valves will be sufficiently open to permit the required auxiliary feedwater, and (3) the effects of anticipated transients without scram events on piping between the pressurizer and pressurizer relief tank and on the tank itself would not result in more serious consequences than analyzed.

We have determined that these design changes are technically feasible and can be incorporated into the design of a RESAR-3S plant when our review of this generic matter is completed.

711292 ~~718 099~~

We intend to continue our review of anticipated transients without scram on a generic basis and will require that any changes indicated to be needed in the RESAR-3S design by the result of approved analyses be incorporated into the design in a timely manner. We will issue a Preliminary Design Approval for RESAR-3S on this basis.

15.6 Summary Conclusions

On the basis of our review of the RESAR-3S accident and transient analysis, we find the consequences of normal and anticipated transients and postulated accidents at core thermal power levels up to 3411 megawatts thermal to be acceptable. However, the following items must be completed prior to final design approval:

(1) Trip delay times and uncertainties used to establish final trip setpoints within analyses values must be fully justified

(2) Reports on the steamline break and feedwater line break accidents must be submitted and reviewed

(3) A generic review of all computer codes used in the accident analyses and identified in Section 15.2.2 of this report must be completed

(4) Rod insertion times used in the safety evaluation must be verified by test results

(5) A report describing the methods used, including the application of codes, for analyses of the loss of flow transient must be submitted for generic review

15.7 Radiological Consequences of Accidents

15.7.1 General

As part of our review of RESAR-3S, we have analyzed the radiological consequences of several postulated accidents to show, in a relative way, the magnitude of the calculated dose to be expected when evaluating applications utilizing RESAR-3S. We have made reasonable dose reduction assumptions concerning the effectiveness of various systems outside the scope of RESAR-3S. For each application utilizing RESAR-3S, we will perform calculations using specific assumptions that are valid for the particular plant and site.

The accidents analyzed in evaluating the RESAR-3S nuclear steam supply system include the hypothetical loss-of-coolant accident including 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, a rod ejection accident, and the steam generator tube failure and main steam line failure accidents.

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

711293

7 8 100

design, we conclude 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. The RESAR-3S evaluation of iodine releases resulting from these accidents does not include the effects of iodine spiking. We believe that iodine spiking is a major factor in the iodine release and will, therefore, 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 application. Until we have evaluated any new information from Westinghouse, we will continue to use our values for iodine spiking in evaluating the balance-of-plant design and in assessing site characteristics. We will include appropriate limits on primary and secondary coolant activity concentrations in the technical specifications for those plants utilizing RESAR-3S to maintain the doses within 10 CFR Part 100 guideline values even with iodine spiking.

15.7.2 Loss-of-Coolant Accident

We have postulated a loss-of-coolant accident for the RESAR-3S 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, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors." Although the containment and the fission product removal and control systems are not within the scope of RESAR-3S, we have assumed such systems for calculational purposes. The assumed containment model includes a low leakage single containment structure surrounding the reactor and a spray additive injection system operating in conjunction with the containment spray system.

The purpose of the spray additive injection system will be to increase the iodine removal capability of the spray following a hypothetical loss-of-coolant accident. 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 15-3 and the doses are listed in Table 15-4. The required short-term relative concentration needed to meet the guideline values of Regulatory Guide 1.4 is 4.6×10^{-4} seconds per cubic meter. Of those sites we previously evaluated, approximately 65 percent had zero to two hour atmospheric dispersion values greater than 4.6×10^{-4} seconds per cubic meter (indicating poorer dispersion conditions) at the exclusion area boundary. We will require that applications utilizing RESAR-3S meet the appropriate guidelines of Regulatory Guide 1.4. We believe these guidelines can be met either by suitable selection of a site or by increasing the effectiveness of those engineered safety features designed to mitigate the dose consequences of an accident. These options involve site selection considerations and engineering features of the balance-of-plant design which are within the current state of the art.

7112318 101

TABLE 15-3

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

Thermal Power Level		3579 megawatts
Operating Time		3.0 years
Reactor Building Leak Rate (zero to 24 hours)		0.10 percent
	(greater than 24 hours)	0.05 percent
Iodine Composition		
Elemental		91 percent
Particulate		5 percent
Organic		4 percent
Two-Hour Thyroid Dose Reduction Factor for Spray		5.5

TABLE 15-4

RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS ACCIDENTS[1]

	Two-hour Dose to Thyroid (rem)	Two-hour Dose to Whole Body (rem)
Steam Generator Tube Failure	8.6	0.57
Steam Generator Tube Failure with Coincident Iodine Spike	34	0.57
Loss of Offsite Power Incident	0.62	less than 0.1
Loss of Offsite Power with Coincident Iodine Spike	0.80	less than 0.1
Steam Line Failure	1.8	less than 0.1
Steam Line Failure with 5 percent Fuel Clad Failure	25	0.18
Rod Ejection - Case 1[2]	16	less than 1
Case 2[3]	35	less than 0.1
Loss of Coolant	150	6.6
Fuel Handling	12	6.6

[1]For an assumed relative concentration of 4.6×10^{-4} seconds per cubic meter.

[2]Case 1 assumes all releases through the containment.

[3]Case 2 assumes all releases through the secondary system.

As part of the assessment of a postulated loss-of-coolant accident, we and Westinghouse 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 accident. We and Westinghouse have assumed the sump water to contain a mixture of iodine fission products in agreement with Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident." After a postulated 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. Based on the assumption that ten percent of the iodine in the water becomes gaseous, we calculate that a leak rate of about one gallon per minute over a period of one-half hour would result in doses (without filters) which could exceed 150 rem (roentgen equivalent man) for a relative concentration of 4.6×10^{-4} seconds per cubic meter 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-3S, 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 system.

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, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," 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, "Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants." They have, therefore, been assumed to have a removal efficiency of 95 percent for all forms of iodine. We will require that those plants which reference RESAR-3S 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 15-4 of this report.

Using an assumed value for a relative concentration of 4.6×10^{-4} seconds per cubic meter for calculational purposes, the resulting dose would be about 12 rem thyroid and 6.6 rem whole body. Thus, the consequences of the postulated loss-of-coolant accident are more limiting.

7911296

718 103

TABLE 15-5

FUEL HANDLING ACCIDENT CALCULATION INPUT PARAMETERS

Shutdown Time	100 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 Poo.	
Elemental	75 percent
Organic	25 percent
Effective Filter Efficiency	
Elemental	95 percent
Organic	95 percent

711297

~~718 104~~

15.7.4 Control Rod Ejection Accident

We have evaluated the consequences of a control rod assembly ejection accident assuming that the releases are through the containment (Case 1) and the secondary system (Case 2) respectively. The assumptions for Case 1 are listed in Table 15-6 and for Case 2 in Tables 15-. and 15-8. The resulting doses with a relative concentration of 4.6×10^{-4} seconds per cubic meter are listed in Table 15-4.

15.7.5 Steam Generator Tube Failure and Steam Line Failure

The assumptions used in the analysis of the steam generator tube failure and steam line failure accidents are listed in Tables 15-8, 15-9, and 15-10. The resulting doses for these accidents are listed in Table 15-4 for a relative concentration of 4.6×10^{-4} seconds per cubic meter. These doses were calculated based on the maximum activity concentrations in the primary and secondary coolants specified in the technical specifications for recent Westinghouse plants.

15.7.6 Hydrogen Purge Dose Analysis

Balance-of-plant applications will provide redundant recombiners for the purpose of controlling the concentration of hydrogen after a postulated design basis loss-of-coolant accident. In the event of failure of both recombiners, a backup purging mode will be used. We will evaluate the radiological consequences of purging during the course of our review of each application which references RESAR-3S to assure that the loss-of-coolant accident plus purge doses are within the exposure guidelines of 10 CFR Part 100. In some cases, filtration of the hydrogen purge effluent may be required.

15.7.7 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 RESAR-3S 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-3S.

15.7.8 Radioactive Waste Gas Decay Tank Failure Accident

Since the gaseous radioactive waste system is outside the scope of RESAR-3S, the consequences of the radioactive waste gas decay tank failure accident will be described in applications which reference RESAR-3S. Appropriate technical specifications will be placed on the maximum activity that can be stored in one tank at any time such that single failure of active components, such as lifting or sticking of a relief valve, will not result in radiological consequences that exceed small fractions of 10 CFR Part 100 guideline doses.

711238

~~718 105~~

TABLE 15-6

ASSUMPTIONS USED IN ANALYSIS
OF CONTROL ROD ASSEMBLY EJECTION ACCIDENT (CASE 1)*

1. Thermal power level of 3579 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 per day for 24 hours and one-half of this value thereafter.

8. Standard ground level release meteorology and dose conversion factors.

*Assumes all releases through containment.

TABLE 15-7

ASSUMPTIONS USED IN ANALYSIS
OF CONTROL ROD ASSEMBLY EJECTION ACCIDENT (CASE 2)*

1. 10 percent fuel with clad failures after accident.

2. 0.25 percent fuel melted after accident.

3. 100 percent of noble gases and 25 percent of iodines contained in melted fuel instantaneously released to reactor coolant system.

4. Pressure equalization between primary and secondary systems reached in 40 minutes.

*Assumes all releases through the secondary system.

711299

~~718~~ ~~106~~

TABLE 15-8

GENERAL ASSUMPTIONS USED IN ANALYSIS OF CONTROL ROD ASSEMBLY EJECTION ACCIDENT (CASE 2)* AND STEAM GENERATOR TUBE FAILURE AND STEAM LINE FAILURE

1. Thermal reactor power = 3579 megawatts.

2. Steam generators operating pressure = 1100 pounds per square inch absolute (maximum).

3. Maximum set pressure for lowest set safety valves = 1350 pounds per square inch, absolute.

4. Enough water storage is available to provide plant cooldown on blackout conditions when auxiliary feedwater pumps are operated.

5. Cooldown rate following accidents of 50 degrees Fahrenheit per hour with no offsite power (maximum).

6. Auxiliary feedwater pumps capable of pumping feedwater into the steam generators when the safety valves are discharging.

7. Automatic startup of auxiliary feedwater pumps following blackout and capability to deliver full flow within 60 seconds.

8. Maximum auxiliary feedwater enthalpy = 80 British thermal units per pound.

9. Safety injection water enthalpy = 80 British thermal units per pound (maximum).

10. Minimum auxiliary feedwater flow = 500 gallons per minute per steam generator.

11. Secondary system piping design capable of isolating flow to any secondary system pipe break.

12. Primary coolant volume = 12,963 cubic feet.

13. Steam generator secondary side volume = 5725 cubic feet.

14. Primary system operating conditions = [illegible]2 degrees Fahrenheit and 2235 pounds per square inch, gauge.

15. Iodine decontamination factor of 10 between water and steam.

16. Primary and secondary coolant equilibrium concentrations as limited by standard technical specifications (one microcurie per gram I-131 equivalent and 100/Ē microcuries per gram for all isotopes with half lives greater than 15 minutes, and 0.1 microcuries per gram I-131 equivalent for secondary coolant).

17. Primary to secondary leak rate as limited by standard technical specifications (one gallon per minute).

18. Iodine source spike factor of 500 after accidents.

19. All releases through the secondary system (except Rod Ejection Accident, Case 1).

20. For accidents assumed to occur in coincidence with an iodine spike. the primary coolant concentration is as limited by the standard technical specifications for 48-hour periods (60 microcuries per gram I-131 equivalent at 100 percent power).

21. Ten percent of iodine and noble gases fuel activity in [illegible]ps.

---

*Assumes all releases through the secondary system.

~~718 107~~

711300

TABLE 15-9

ASSUMPTIONS USED IN ANALYSIS OF STEAM GENERATOR TUBE FAILURE ACCIDENT

1. Isolation of failed steam generators within 30 minutes of accident.
2. Steam generators controlled at safety valve settings.
3. No more than 125,000 pounds of primary coolant are transferred to the secondary side of the faulty steam generator fr[illegible] the accident.
4. Pressure equalization between defective steam generator and primary system reached within 30 minutes.

TABLE 15-10

ASSUMPTIONS USED IN ANALYSIS OF STEAM LINE FAILURE ACCIDENT

1. Steam line isolation valves fully close within ten seconds of break.
2. Only one steam generator blows down even if one of the isolation valves fails to close.
3. Contents of one steam generator (minimum 91,000 pounds of water and maximum 9,300 pounds of steam) released to environment within 30 seconds.
4. Primary and secondary system pressures equalized after 30 minutes.
5. Primary system pressure remains at 2235 pounds per square inch, gauge for first two hours following failure.

711301

~~718~~ ~~108~~

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 Paragraph 3 of Appendix O to 10 CFR Part 50, an application for a Preliminary Design Approval of a standard design is required to include preliminary technical specifications. The regulations require an identification and justification for the selection of those variables, conditions, or other items which are determined as a result of the preliminary safety analysis and evaluation to be probable subjects of technical specifications, with special attention given for those items which may significantly influence the final design.

We have reviewed the proposed technical specifications presented in Section 16 of RESAR-3S with the objective of identifying those items that would require special attention at the preliminary design review stage, to preclude the necessity for any significant change in design to support the final technical specifications. The proposed technical specifications are similar to the standard technical specifications being developed for plants using nuclear steam supply systems designed by Westinghouse.

On this basis, we conclude that Westinghouse's proposed technical specifications are acceptable for the preliminary design review stage.

~~718~~ 109 711302

17.0 QUALITY ASSURANCE

17.1 General

Section 17 of RESAR-3S describes Westinghouse's Nuclear Energy Systems' Quality Assurance program by reference to Topical Report WCAP-8370, "Westinghouse Nuclear Energy System Divisions Quality Assurance Plan." The program covers safety-related equipment from design through procurement, fabrication, manufacture, turnover, and, as applicable, installation, preoperational tests, and operation of a standard 3425 megawatts thermal pressurized water reactor 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, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," and the applicable regulatory guides, which are listed in Table 17-1.

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 Part 50. The Pressurized Water Reactor Systems Division of Nuclear Energy Systems is the lead division with respect to design and procurement as shown by Figure 17-2.

Each division has an organization specifically responsible for quality assurance and for quality control which reports at a high enough level to assure independence consistent with Criterion I of Appendix B to 10 CFR Part 50. 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 Pressurized Water Reactor Systems Division's Product Assurance Manager is Chairman of the Quality Assurance Committee. This committee is

~~718 110~~ 711303

TABLE 17-1

REGULATORY GUIDES APPLICABLE TO
QUALITY ASSURANCE PROGRAMS

1. Regulatory Guide 1.28, "Quality Assurance Program Requirements (Design and Construction)."

2. Regulatory Guide 1.30, "Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment."

3. Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."

4. Regulatory Guide 1.38, "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants."

5. Regulatory Guide 1.39, "Housekeeping Requirements for Water-Cooled Nuclear Power Plants."

6. Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants."

7. Regulatory Guide 1.58, "Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel."

8. Regulatory Guide 1.64, "Quality Assurance Requirements for the Design of Nuclear Power Plants."

9. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."

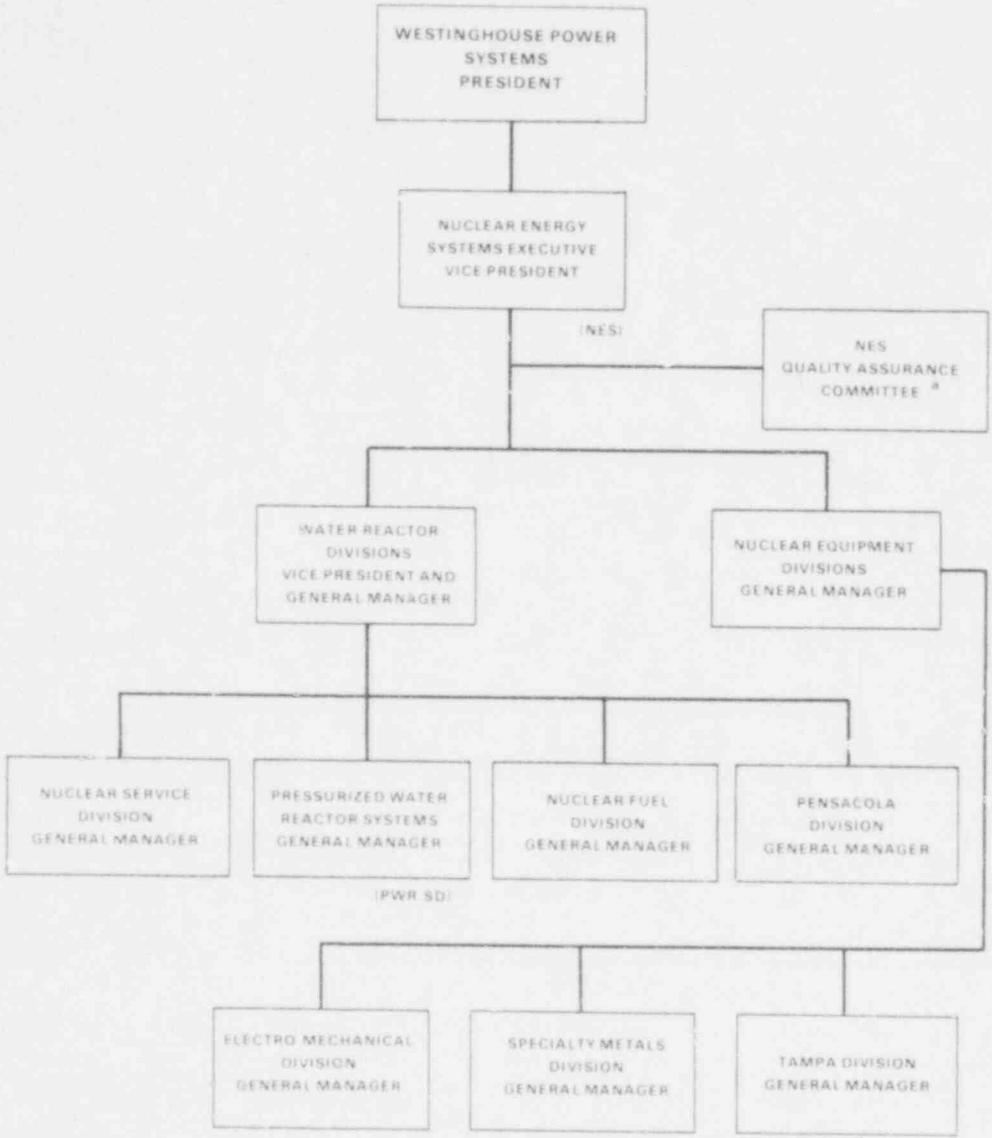
10. Regulatory Guide 1.74, "Quality Assurance Terms and Definitions."

11. Regulatory Guide 1.88, "Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records."

12. Regulatory Guide 1.94, "Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants."

711304

718 111



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)

POOR ORIGINAL

~~718 112~~

711305

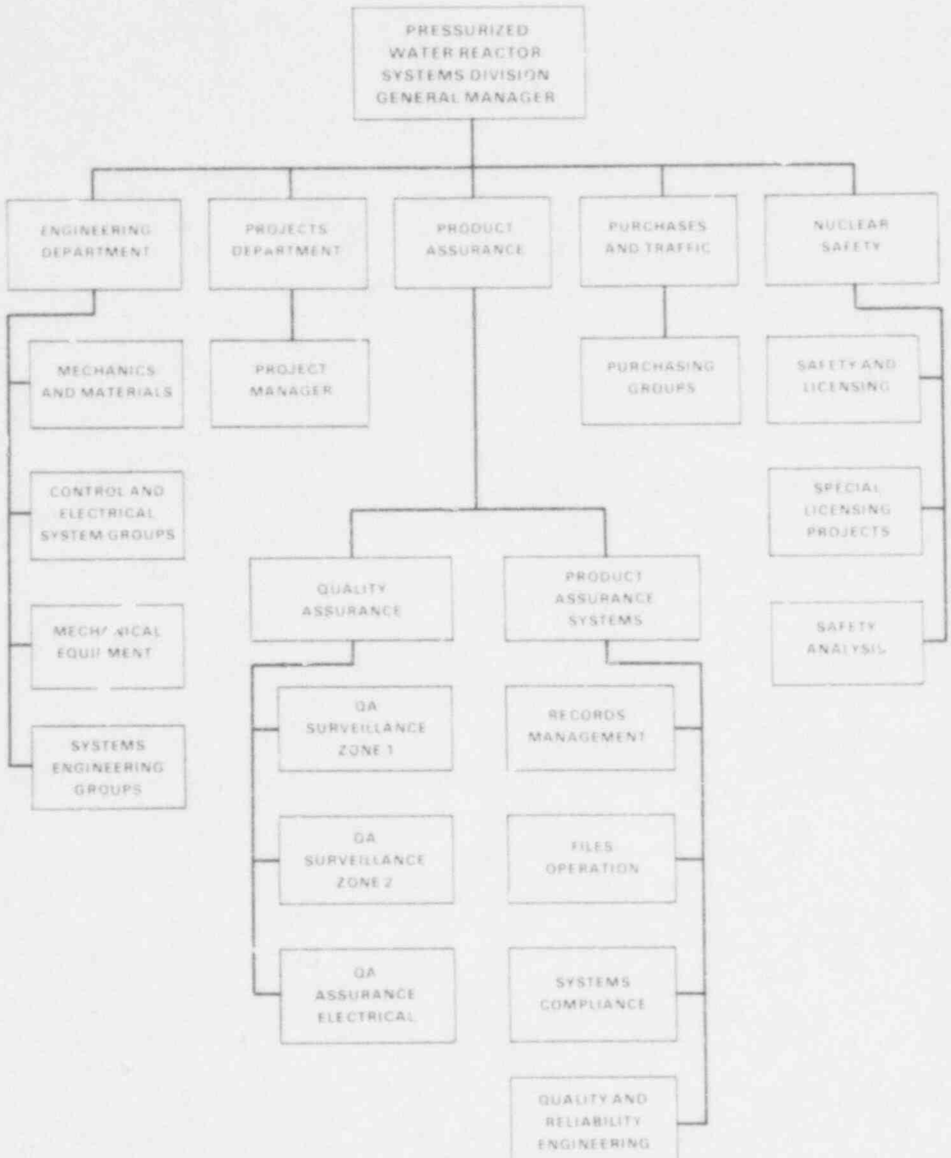


Figure 17-2 PRESSURIZED WATER REACTOR SYSTEMS DIVISION QUALITY ASSURANCE PROGRAM ORGANIZATION

POOR ORIGINAL

711306

~~218~~ 113

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 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 Part 50 and to follow the guidance provided by the Commission in WASH-1283, "Guidance on Quality Assurance Requirements During Design and Procurement Phase of Nuclear Power Plants," dated May 1974, and WASH-1309, "Guidance on Quality Assurance Requirements During the Construction Phase of Nuclear Power Plants," dated May 10, 1974. Westinghouse has also agreed to follow Commission guidance in WASH-1284, "Guidance on Quality Assurance Requirements During the Operations Phase of Nuclear Power Plants," dated 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 Part 50. In addition, each division requires that applicable requirements of Appendix B to 10 CFR Part 50 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 Part 50 is given.

The Nuclear Energy Systems 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

711307 ~~718 114~~

within Westinghouse, including quality and reliability personnel, independently review drawings and equipment specifications prior to issuance. Cognizant Nuclear Energy Systems personnel also review suppliers' 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 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 to measure supplier performance.

Each division controls nonconforming material, parts, and components to prevent inadvertent use and provide for their identification, segregation, and disposition.

Nuclear Energy Systems requires records which show the quality of the 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 Nuclear Energy Systems 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 Commission's Office of Inspection and Enforcement has conducted inspections to examine the implementation of the quality assurance program commitments made by Westinghouse in RESAR-3S to ascertain their conformance with Appendix B to 10 CFR Part 50. The examinations encompassed the Westinghouse Nuclear Energy Systems 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.

7113C8

~~718~~ 115

Based on these inspections, 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, therefore, conclude that the implementation of the Westinghouse RESAR-3S 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-3S provides for a comprehensive system of planned and systematic controls which adequately demonstrate Westinghouse's ability and commitment to comply with each of the 18 criteria of Appendix B to 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-3S complies with the requirements of Appendix B to 10 CFR Part 50 and is acceptable.

~~718 116~~ 711309

18.0 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

At its 195th meeting on July 8-10, 1976, the Advisory Committee on Reactor Safeguards completed its review of Westinghouse's application for a Preliminary Design Approval for its proposed RESAR-3S standard nuclear steam supply system. A copy of the Committee's report on RESAR-3S, dated July 14, 1976, which contains certain comments and recommendations, is included as Appendix C to this report. The actions we have taken or plan to take in response to the Committee's comments and recommendations are described in the following paragraphs.

(1) The Committee recommended that during design, procurement, construction, and startup, timely and appropriate interdisciplinary system analyses be carried out to assure complete functional compatibility across each interface for the entire spectrum of anticipated operations and postulated design basis accident conditions.

We have transmitted the Committee's recommendations to Westinghouse for its consideration in proceeding with the RESAR-3S design. We have recognized the importance of defining the safety-related interface information required to establish compatibility of RESAR-3S with the balance of plant. As discussed in Section 1.7 of this report, we have concluded that this interface information is acceptable for the Preliminary Design Approval. However, as part of our long range effort to improve the implementation of the Commission's standardization policy, we have initiated a dialogue with the nuclear industry in an effort to develop improved procedures for defining interface requirements for standard plant designs. Through this effort and additional experience that will be gained in evaluating standard plant designs during the Final Design Approval stage, we will be able to assure functional compatibility between RESAR-3S and the balance-of-plant design.

(2) The Committee stated that an issue to be resolved prior to preliminary design approval for RESAR-3S involves the possibility of a single failure leading to the loss of the residual heat removal system. The Committee recommended that this matter be resolved in a manner satisfactory to the staff and wished to be kept informed.

Our evaluation of the residual heat removal system is discussed in Section 5.4.3 of this report. This matter has been resolved in a manner satisfactory to the staff. By issuance of this report, the Committee is being informed of the results of our evaluation.

711310

(3) The Committee recommended that Westinghouse emphasize analytical and experimental programs to substantiate the conservatisms in the current Westinghouse emergency

~~718~~ 117

core cooling system evaluation model and to establish the accuracy and uncertainties in their best-estimate calculations. Timely progress reports should be provided on the performance of the 17x17 fuel, the control systems, improvements in the analytical models, test verification of analytical methods, and reliability studies undertaken to establish meaningful improvements in components, systems, and arrangements for emergency core cooling systems and the dependent auxiliaries necessary to sustain the heat transport systems. The Committee recommended that if emergency core cooling system improvements, such as obtainable from higher reflooding rates, can be achieved, consideration should be given to incorporating them into RESAR-3S.

We have transmitted the Committee's recommendations to Westinghouse for its consideration in proceeding with the RESAR-3S design.

(4) The Committee stated that further review should be made on the adequacy of the RESAR-3S provisions for the maintenance, inspection, and operational needs of the plant throughout its service life and for eventual decommissioning. In particular, the Committee stated its belief that the staff and Westinghouse should review methods and procedures for minimizing, and, if necessary, for removing accumulations of radioactive contamination so that maintenance and inspection programs can be more effectively and safely carried out.

During the past year, we have been reviewing the activation product problem, including data on occupational radiation exposures related to activation products and methods and procedures for preventing or reducing and removing accumulations of radioactive contamination in the primary coolant system of light water reactors. Information gathered at conferences on decontamination and decommissioning and inputs from specific technical sources industry have resulted in our examining this issue in more detail in the review of nuclear power plant applications. We are presently in the process of developing a research need which will address specifically the following subjects:

(a) Study of cost and effectiveness of precursor elimination.

(b) Evaluation of effectiveness of decontamination methods.

(c) Evaluation of systems and methods for corrosion rate reduction and corrosion product retention control.

The draft working paper for Regulatory Guide 8.8, Revision 2, "Information Relevant to Assuring that Occupational Radiation Exposure at Nuclear Power Stations Will Be as Low as Reasonably Achievable," incorporated all of our findings to date on methods and procedures that are effective in reducing radioactive exposure related to activation products, and was reviewed by industry in April 1975. Two areas we have identified where information is lacking or insufficient are (1) the costs associated with various measures, and (2) the quantitative

711311

718 118

benefits in reduction of occupational radiation exposure associated with the measures. Because we do not have this information, we do not require additional design features for exposure reduction in plants presently under review. We are presently undertaking a research program to answer these questions.

We are continuing our study of the problems associated with decommissioning but as yet we do not require specific design provisions for decommissioning. A few reactors have been decommissioned and we know from this experience that the resultant exposures can be kept within acceptable bounds. Because the experience in this area has been acceptable to date, we plan to continue our investigations further into this matter before we require that any special features be incorporated during the design and construction of a plant.

(5) The Committee stated its belief that Westinghouse and the staff should continue to review RESAR-3S for design changes that will further improve protection against sabotage.

The Office of Nuclear Regulatory Research has funded studies concerning possible modes of sabotage at nuclear power plants. Any recommendations resulting from these studies regarding additional design features to protect against acts of industrial sabotage will be considered by the staff for incorporation in the RESAR-3S design.

(6) The Committee pointed out that generic problems relating to large water reactors are discussed in the Committee's report dated April 16, 1976. The Committee stated its belief that procedures should be developed to incorporate approved resolution of these items into RESAR-3S.

These generic problems are being worked on by the staff, various reactor vendors, and other industrial organizations and will be the subject of continuing attention by the staff.

711312

718 119

19.0 CONCLUSIONS

Based on our evaluation of the proposed RESAR-3S nuclear steam supply system design, we conclude that:

(1) Westinghouse has described, analyzed, and evaluated the proposed design including, but not limited to, the principal engineering criteria for the design; the interface information necessary to assure compatibility between the submitted design and the balance of nuclear power plant; the envelope of site parameters postulated for the design; the quality assurance program to be applied to the design, procurement, and fabrication of safety-related features of the nuclear steam supply system; the design features that affect plans for coping with emergencies in the operation of the reactor or major portion thereof; and has identified the major features and components incorporated therein for the protection of the health and safety of the public;

(2) Such further technical or design information as may be required to complete the safety analysis 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, if any, which require research and development have been identified by Westinghouse and it has described, and will conduct, research and development programs reasonably designed to resolve safety questions associated with such features or components;

(4) On the basis of the foregoing, there is reasonable assurance that: (i) such safety questions will be satisfactorily resolved at or before the issuance of the operating license for the first nuclear power plant utilizing the RESAR-3S nuclear steam supply system; and (ii) taking into consideration the site criteria contained in 10 CFR Part 100, a facility can be constructed and operated without undue risk to the health and safety of the public, provided the site characteristics conform to the distance requirements of 10 CFR Part 100, and provided further that the balance of plant is properly designed and constructed in conformity with the interface requirements specified in RESAR-3S and in this report, as discussed above; and

(5) Westinghouse is technically qualified to design the nuclear steam supply system described in the RESAR-3S document.

711313

718 120

APPENDIX A

CHRONOLOGY OF REVIEW OF
REFERENCE SAFETY ANALYSIS REPORT RESAR-3S

June 27, 1975	Application submitted for acceptance review
July 3, 1975	Letter to Westinghouse advising of receipt of application
July 11, 1975	Meeting with Westinghouse to discuss content of application
July 31, 1975	Letter to Westinghouse stating that application acceptable for docketing and transmitting acceptance review questions
July 31, 1975	Application docketed
September 4, 1975	Submittal of Amendment No. 1, consisting of a response to letter dated July 31, 1975 including copies of piping and instrumentation drawings
October 30, 1975	Letter to Westinghouse requesting additional information and response to staff positions
October 31, 1975	Letter to Westinghouse transmitting review schedule
November 11, 1975	Letter to Westinghouse requesting additional information and response to additional staff positions
November 14, 1975	Letter to Westinghouse requesting additional information and response to additional staff positions
November 18, 1975	Letter from Westinghouse concerning review schedule
December 9, 1975	Letter to Westinghouse requesting additional information and response to additional staff positions
December 10, 1975	Meeting with Westinghouse to discuss review procedures and schedule, optional systems, and outstanding information
December 18, 1975	Submittal of Amendment No. 2. consisting of response to letters dated October 30, November 11, and November 14, 1975

711314

~~718 121~~

December 31, 1975	Submittal of Amendment No. 3, consisting of response to letter dated December 9, 1975
January 13, 1976	Meeting with Westinghouse to discuss general review matters and outstanding issues
January 19, 1976	Letter from Westinghouse providing schedule for implementation of division of scope dictated by WASH-1341, Amendment 1
January 26, 1976	Submittal of Amendment No. 4, consisting of additional response to letters dated November 11 and December 9, 1975
January 30, 1976	Letter to Westinghouse requesting additional information and response to additional staff positions
February 9, 1976	Letter from Westinghouse in response to request dated January 30, 1976
February 18, 1976	Meeting with Westinghouse to discuss outstanding issues
February 19, 1976	Submittal of Amendment No. 5, consisting of additional response to letter dated December 10, 1975 and response to letter dated January 30, 1976
February 24, 1976	Letter to Westinghouse concerning resubmittal under the topical report program the descriptions of those systems identified as "options" according to WASH-1341
March 8, 1976	Submittal of Amendment No. 6, consisting of additional response to letters dated December 10, 1975 and January 30, 1976
March 9, 1976	Meeting with Westinghouse to discuss outstanding issues
March 19, 1976	Submittal of Amendment No. 7, consisting of additional response to letter dated January 30, 1976
April 1, 1976	Letter from Westinghouse concerning staff positions on residual heat removal and residual heat removal system design
April 7, 1976	Letter to Westinghouse requesting additional information concerning anticipated transients without scram
April 7, 1976	Submittal of Amendment No. 8, consisting of additional response to letter dated January 20, 1976

711315

177

April 15, 1976	Meeting with Westinghouse to discuss review status
April 26, 1976	Submittal of Amendment No. 9, consisting of response to request for additional information
May 3, 1976	Letter to Westinghouse concerning Browns Ferry fire, transmitting revision to Standard Review Plan
May 5, 1976	Meeting with Westinghouse to discuss outstanding issues related to residual heat removal system design
May 6, 1976	Letter to Westinghouse concerning new filing procedures
May 14, 1976	Submittal of Amendment No. 10, consisting of response to request for additional information
May 14, 1976	Letter from Westinghouse regarding the May 5, 1976 meeting related to residual heat removal system design
May 25, 1976	Issuance of Report to the ACRS
June 2, 1976	Letter to Westinghouse transmitting the Report to the ACRS
June 16, 1976	ACRS subcommittee meeting with staff and Westinghouse
June 28, 1976	Submittal of Amendment No. 11, consisting of revised information concerning lockout of power to certain emergency core cooling system valves
July 8, 1976	ACRS meeting with staff and Westinghouse
July 14, 1976	Report by the ACRS issued
July 15, 1976	Letter to Westinghouse transmitting report by the ACRS
August 13, 1976	Submittal of Amendment No. 12, consisting of additional information concerning the residual heat removal system
August 20, 1976	Meeting with Westinghouse to discuss proposed residual heat removal system design modification
August 30, 1976	Letter to Westinghouse concerning new filing procedures
August 31, 1976	Letter to Westinghouse requesting additional information concerning emergency core cooling system performance

711316 ~~718 123~~

September 22, 1976	Meeting with Westinghouse to discuss capability to achieve cold shutdown using only safety-grade systems
October 11, 1976	Submittal of Amendment No. 13, consisting of response to letter dated August 31, 1976

APPENDIX B

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, "CONTEMPT - 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 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, Philadephia, 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.

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.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

July 14, 1976

Honorable Marcus A. Rowden
Chairman
US Nuclear Regulatory Commission
Washington, DC 20555

Subject: REPORT ON WESTINGHOUSE ELECTRIC CORPORATION REFERENCE SAFETY ANALYSIS REPORT, RESAR-3S

Dear Mr. Rowden:

At its 195th meeting, July 8-10, 1976, the Advisory Committee on Reactor Safeguards completed its review of the Westinghouse Electric Corporation's application for a Preliminary Design Approval (PDA) for a standardized nuclear steam supply system consisting of a pressurized water reactor as described in its Reference Safety Analysis Report, RESAR-3S. A subcommittee meeting was held with representatives of the Applicant and the Nuclear Regulatory Commission (NRC) Staff in Washington, DC, on June 16, 1976. The Committee had the benefit of discussions with representatives of the NRC Staff and the Westinghouse Electric Corporation. The Committee also had the benefit of the documents listed below.

RESAR-3S is a Westinghouse standardized four-loop, single-unit nuclear steam supply system with a core thermal power of 3411 MWt. Systems within the scope of RESAR-3S include the reactor core, reactor coolant system and supports, chemical and volume control system, emergency core cooling system, boron recycle system, residual heat removal system, fuel handling system, and associated instrumentation and controls for these systems. RESAR-3S is similar to the nuclear steam supply system of the SNUPPS projects, reviewed in ACRS reports of September 17, October 16, and December 11, 1975. The ACRS report of September 18, 1975 reviewed the Westinghouse nuclear steam supply system RESAR-41. Significant features, other than those associated with the higher power level, which were incorporated in RESAR-41 but are not in RESAR-3S, include longer fuel assemblies, a rapid refueling system, an emergency boration system, and the use of three independent injection trains in the emergency core cooling system (ECCS).

711320

~~718~~ 127

RESAR-3S has been designed for application to an envelope of plant sites which includes provision for a Safe Shutdown Earthquake with a maximum horizontal ground acceleration of 0.4g.

RESAR-3S provides for those safety-related interface requirements that are essential to designing the balance of plant to be consistent with the assumptions used in the accident analyses. Since the utility-applicant is responsible for instituting the quality assurance programs necessary to assure that all safety-related design requirements have been met, these matters will be reviewed in more detail with the utility-applicants on a case-by-case basis. The Committee recommends that during design, procurement, construction, and startup, timely and appropriate interdisciplinary syutem analyses be carried out to assure complete functional compatibility across each interface for the entire spectrum of anticipated operations and postulated design basis accident conditions. For multiple reactor units at a single station, the Committee anticipates that safety-related items in RESAR-3S would be separately provided for each reactor unit.

An issue to be resolved prior to preliminary design approval for RESAR-3S involves the possibility of a single failure leading to the loss of the residual heat removal system. The Committee recommends that this matter be resolved in a manner satisfactory to the NRC Staff and wishes to be kept informed.

The Committee recommends that Westinghouse emphasize analytical and experimental programs to substantiate the conservatisms in the current Westinghouse ECCS evaluation model and to establish the accuracy and uncertainties in their best-estimate calculations. Timely progress reports should be provided on the performance of the 17x17 fuel, the control systems, improvements in the best estimate analyses, test verification of analytical methods, and reliability studies undertaken to establish meaningful improvements in components, systems, and arrangements for ECC systems and the dependent auxiliaries necessary to sustain the heat transport systems. The Committee recommends that if studies establish that ECCS improvements, such as obtainable from higher reflooding rates, can be achieved, consideration should be given to incorporating them into RESAR-3S.

Further review should be made on the adequacy of the RESAR-3S provisions for the maintenance, inspection, and operational needs of the plant throughout its service life and for eventual decommissioning. In particular, the Committee believes that the NRC Staff and the Westinghouse Electric Corporation should review methods and procedures for minimizing, and, if necessary, for removing accumulations of radioactive contamination so that maintenance and inspection programs can be more effectively and safely carried out.

The Committee believes that Westinghouse and the NRC Staff should continue to review RESAR-3S for design changes that will further improve protection against sabotage.

Generic problems relating to large water reactors are discussed in the Committee's report dated April 16, 1976. The Committee believes that procedures should be developed to incorporate approved resolution of these items into RESAR-3S.

The Committee believes that, subject to the above comments, RESAR-3S can be successfully engineered to serve as a reference system.

Sincerely yours,

Dade W. Moeller

Dade W. Moeller
Chairman

REFERENCES

1. Westinghouse Electric Corporation, "Reference Safety Analysis-3S (RESAR-3S)", Volumes 1-8, July, 1975.
2. Amendments 1-10 to RESAR-3S.
3. USNRC, "Report to the Advisory Committee on Reactor Safeguards in the Matter of Westinghouse Electric Corporation Reference Safety Analysis Report, RESAR-3S," May 25, 1976.