NUREG-1413

Safety Evaluation Report
related to the preliminary design of the Standard Nuclear Steam Supply Reference System, RESAR SP/90

Docket No. STN 50-601
Westinghouse Electric Corporation, Incorporated

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

April 1991

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ABSTRACT

On October 24, 1983, the Westinghouse Electric Corporation tendered its application for a preliminary design approval of the advanced pressurized-water reactor design for the SP/90 reactor. The Westinghouse Reference Safety Analysis Report (RESAR SP/90, Docket No. STN 50-601), describing the design of the facility, was submitted from October 24, 1983 through March 9, 1987.

Staff of the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, has prepared this safety evaluation report of the RESAR SP/90 on the basis of its review.

Because of the stage of the design, there are open issues that have not been resolved. These issues are discussed in detail throughout this report, and a summary is provided in Section 1.6 of this report. The applicant will be required to address these and any additional such concerns that may be raised during the course of the staff's review of advanced light-water reactors in support of a final design approval application.

This report shall not constitute a commitment to issue a permit or license or in any way affect the authority of the Commission, its adjudicatory boards, and other presiding officers in any proceeding under Subpart G of Title 10 of the Code of Federal Regulations, Part 2.

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1 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

On October 24. 1983, the Westinghouse Electric Corporation (hereinafter referred to as Westinghouse or the applicant) tendered its application for a preliminary design approval (PDA) of the Westinghouse advanced pressurized-water reactor (WAPWR) design, RESAR SP/90, with the United States Nuclear Regulatory Commission (hereinafter referred to as the NRC, the Commission, or the staff). The submittal was made in accordance with Appendix O, "Standardization of Design: Staff Review of Standard Designs," of Part 50 of Title 10 of the Code of Federal Regulations (10 CFR 50). The application was docketed on November 30, 1983 (Docket No. STN 50-601).

A standard safety analysis report, Westinghouse Reference Safety Analysis Report (RESAR SP/90), describing the design of the facility was submitted in modular form during the period October 24, 1983 to March 9, 1987. In addition, the information in RESAR SP/90 has been supplemented by amendments to these modules. Tables 1.1 and 1.2 provide submittal dates of the modules and the amendments to the modules, respectively. These documents are available for public inspection at the NRC Public Document Room, 2120 L Street, N.W., Washington, D.C. 20555.

This safety evaluation report (SER) summarizes the results of the staff's radiological safety review of the RESAR SP/90 design and delineates the scope of the technical details considered in evaluating the proposed design. Environmental aspects were not considered in the staff's review of RESAR SP/90, but will be addressed in each utility's plant-specific application that references this design for a construction permit. The NRC Licensing Project Manager for RESAR SP/90 is Loren F. Donatell. He may be contacted by writing to: Division of Reactor Projects - III, IV, V and Special Projects, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

Before the promulgation of 10 CFR 52 in May of 1989, the review of RESAR SP/90 had been carried out by the staff pursuant to Appendix O to 10 CFR 50, using a similar procedure to that used for custom plant reviews for which guidance to staff reviewers is provided in the "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, July 1981. The initial phases of the NRC staff evaluation of the RESAR SP/90 design, i.e., acceptance review, requests for additional information, and responses thereto, are analogous to the construction permit stages of a licensing review for a specific facility. However, the conclusion of the review does not result in the granting of a construction permit. Instead, a preliminary design approval can be issued by the NRC staff following satisfactory completion of the reviews performed by the staff and the Advisory Committee on Reactor Safeguards. Since Appendix O to 10 CFR 52 did not change the requirements for the PDA stage of review, a PDA can be issued pursuant to 10 CFR 52 on the basis of the results of this report.

The PDA is an approval issued by the NRC deeming a standard preliminary design of a nuclear power plant acceptable for incorporation by reference in individual facility license applications and provides that the approved design be

Table 1.1 Submittal dates of RESAR SP/90 modules

Module no.	Title	Submittal date
1	Primary Side Safeguards System	October 24, 1983
2	Regulatory Conformance	November 30, 1983
3	Introduction and Site	May 16, 1984
4	Reactor Coolant System	July 11, 1984
5	Reactor System	July 31, 1984
6 and 8	Secondary Side Safeguards System/Steam and Power Conversion Systems	November 9,1984
7	Structural/Equipment Design	January 31, 1985
9	Instrumentation and Control and Electric Power	February 27, 1985
10	Containment Systems	March 9, 1987
11	Radiation Protection	September 12, 1985
12	Waste Management	September 20, 1985
13	Auxiliary Systems	October 29, 1985
14	Initial Test Program	*
15	Control Room/Human Factors Engineering	October 27, 1986
16	Probabilistic Safety Study	June 28, 1985 September 13, 1985

*Withheld until the final design approval application is to be submitted.

Table 1.2 Amendments to RESAR SP/90 modules

Module no.	Amendment no.	Submittal date
1	1	June 9, 1986
	1a	May 13, 1988
	2	November 23, 1987
	3	May 13, 1988
	4	September 19, 1988
2	1	October 8, 1987
	2	March 23, 1988
	3	October 3, 1989
3	1	January 30, 1985
	1a	May 13, 1988
	2	October 26, 1987
	3	August 25, 1988
4	Errata	July 23, 1984
	1	May 13, 1988
	2	October 26, 1988

Table 1.2 (continued)

Module no.	Amendment no.	Submittal date
5	1	November 30, 1984
	2	April 16, 1985
	3	June 4, 1986
	4	November 8, 1988
6 and 8	1	November 18, 1986
7	1	May 30, 1986
	1a	February 14, 1989
	2	February 14, 1989
9	1	July 14, 1986
	2	August 29, 1986
	3	December 17, 1987
10	-	
11	1	March 14, 1986
	2	July 27, 1988
12	1	March 22, 1988
	2	July 27, 1988
13	1	November 23, 1987
	2	March 24, 1988
	3	May 13, 1988
	4	August 5, 1988
14	-	*
15	-	
16	1	August 25, 1986
	2	October 30, 1986
	3	October 8, 1987
Addendum	1	January 7, 1988
	2	January 7, 1988
	3	May 13, 1988
	4	May 13, 1988
	5	May 13, 1988
	6	July 6, 1988
	7	May 23, 1988
	8	May 27, 1988
	9	June 14, 1988
	10	June 23, 1988

*Withheld until the final design approval application is to be submitted.

used and relied on by the staff and the Advisory Committee on Reactor Safeguards (ACRS) in their reviews of any such applications.

This report presents the staff's evaluation on the basis of its review to date. It represents only the first stages of the staff's review of the design, construction, and operating features of the RESAR SP/90 design. Before a decision is made to issue an operating license for any application that references RESAR SP/90, the staff will review the final design of the RESAR SP/90 to determine that all of the Commission's safety requirements have been met. Such a 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 staff. This report may, however, be used in support of issuance of a PDA allowing for referenceability in an application for a construction permit.

During meetings, the applicant has indicated that there is no intent to apply for final design approval (FDA) and design certification after the PDA is issued by the staff unless there is a proven interest on the part of a utility. However, in the event Westinghouse should decide to pursue an FDA and go through design certification proceedings, the staff will review the final design under the requirements of 10 CFR 52 as would then be described in the final safety analysis report and supporting documentation. For the purposes of this report, references to a FDA should not be interpreted to mean a FDA as required in subpart B of 10 CFR 52 but rather as a traditional FDA that could be issued in support of an application submitted pursuant to 10 CFR 50.

During the course of its safety review of the material submitted, the staff held numerous meetings with Westinghouse representatives to discuss the proposed system design and performance and requested Westinghouse to provide additional information that was needed for the staff's evaluation. This information was provided in amendments to RESAR SP/90 modules. As a result of the staff's review, a number of changes were made in the proposed design. These changes are described in the amendments and are discussed in the appropriate sections of this report.

Section 20 of this report provides an analysis detailing where the staff proposed departure from current regulations or where the staff was substantially supplementing or revising interpretive guidance applied to currently-licensed LWRs. For those issues discussed in this section, the Commission reviewed the basis for the approach that the staff was proposing, and accordingly, approved in part and disapproved in part the staff's recommendations as detailed in Section 20 of this report. These issues are considered fundamental to agency decisions on the acceptability of the evolutionary ALWR designs. Note that during the course of the Commission's review of advanced LWRs, additional concerns falling in the same category may arise. The applicant will be required to address these concerns in its application for an FDA, a construction permit, or an operating license.

A chronology of the principal actions related to the processing of this application is given in Appendix A to this SER. Appendix B provides the references*

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*NRC documents (e.g., NUREGs and regulatory guides are not included in Appendix B because they may be retrieved as indicated in the "Availability Notice" on the inside front cover of this report.

for this report, and Appendix C addresses unresolved safety issues. Appendix D provides a list of the principal technical reviewers who evaluated RESAR SP/90, and Appendix E lists the abbreviations used throughout this report.

Several references to Westinghouse topical reports are made throughout this report. Some of these topical reports contain information that has been authorized by the Commission to be exempt from public disclosure, as provided by 10 CFR 2.790. For each such topical report containing proprietary information, a nonproprietary version, similar in content except for the omission of the proprietary information, is provided. Reference to Westinghouse topical reports throughout this report is made to the proprietary version only.

Plant-specific applicants referencing RESAR SP/90 in the future will retain architect-engineers, constructors, and consultants as needed. Before a decision is made to issue a construction permit, the staff will evaluate, for each plant-specific application that references RESAR SP/90, the technical competence of the applicant and its contractors to manage, design, construct, and operate a nuclear power plant.

Since RESAR SP/90 does not contain the design of the entire facility, it is necessary to describe the safety-related interfaces between the Westinghouse-supplied systems and the remainder of the plant to be provided by the plant-specific applicant. Interface information addresses the pertinent safety-related design requirements necessary to ensure the compatibility of the referenced system with the plant-specific portion of the facility. This information has been included in RESAR SP/90, and the staff's evaluation of that information is discussed in Section 1.7 as well as throughout the remainder of this report, as appropriate.

Specific responsibilities of Westinghouse are delineated in RESAR SP/90. These responsibilities include such things as special studies, qualification testing of equipment, and design of equipment within the scope of the design. A plant-specific applicant referencing this design is expected to adopt all designs, tests, operating limitations, and inspections identified throughout this SER and the RESAR SP/90. This means that any commitments made by Westinghouse in RESAR SP/90 will be automatically included in the portion of the plant-specific applicant's preliminary safety analysis report that references this design, unless exceptions are requested by the plant-specific applicant. Consideration and approval of such requests will be handled on a case-by-case basis.

Plant-specific applicants referencing this design also will be required to satisfy the requirements and concerns resulting from the staff's review of this design. These requirements and concerns for plant-specific applicants are given throughout this report and are listed in Section 1.6 of this report.

1.2 General Design Description

The RESAR SP/90 is a single-unit nuclear power block (NPB) design for a 3800-MWt (megawatt thermal), four-loop pressurized-water reactor which emcompasses buildings, structures, systems, and components of the nuclear power plant.

Specifically excluded from the NPB scope are the turbine building, the waste disposal building, the service building, the administration building, the service water/cooling water structure, and the ultimate heat sink. These features will be the design responsibility of a plant-specific applicant proposing to build a facility referencing the RESAR SP/90 design.

The key areas that are included in the NPB scope are the containment building, the fuel-handling facilities, the mechanical safeguards equipment area, the auxiliary systems area, the instrumentation and controls area, the control room, the electrical power distribution equipment area, the emergency diesel generator area, and the technical support center. Figure 1.1 gives a hypothetical site plan for the facility showing the following major buildings: the containment building, the reactor external building, the turbine building, the waste disposal building, the service building, and the administration building. The containment building and the reactor external building essentially contain all the buildings, structures, systems, and components that are included in the RESAR SP/90 NPB scope. All the other major buildings listed above and shown on Figure 1.1 are excluded from the NPB scope with a few exceptions. For example, the NPB-scope waste processing system will be located in the waste disposal building and the technical support center will be located in the service building.

Table 1.3 provides a detailed listing of the buildings and structures, systems and components that are included in the NPB scope or that interface with the NPB. For those systems that are not in the NPB scope, the type of criteria that will be provided in RESAR SP/90 is indicated. Both the design criteria and interface criteria that have been provided for the areas not in the NPB scope are considered pertinent safety-related requirements.

The containment building and reactor external building will consist of a spherical steel containment vessel (SSCV), a spherical reinforced-concrete shield building, and a wraparound reactor external building on a common basement. The SSCV will contain the 3800-MWt, four-loop reactor coolant system (RCS), which will consist of the reactor vessel, four reactor coolant pumps, four steam generators, the pressurizer, and the pressurizer relief tank.

Several major components of the engineered safety systems also will be located in the SSCV, such as four containment recirculation units, four accumulators, four core-reflood tanks, four residual-heat-removal heat exchangers, and the emergency water storage tank (EWST). The capacity of the EWST is dictated by the water volume required to fill the refueling canal associated with the RESAR SP/90 design of the RCS. The EWST is a stainless steel lined tank that will be located below the nominal containment floor level. A 2500-ppm (parts per million) boron concentration will be maintained in this tank during normal plant operation. In the event of an accident, the four low-head and four high-head pumps of the integrated safeguards system (ISS) would take direct suction from the EWST and provide the required flow to the RCS and the containment spray headers. Only after all the lower elevations within the containment were flooded would water return to the EWST via spillways in the containment.

The reactor external building will essentially contain all the NPB-scope systems and components not located inside the SSCV. The reactor external building will extend 360 degrees around the secondary containment (shield building). The equipment located in the reactor external building will be arranged to separate

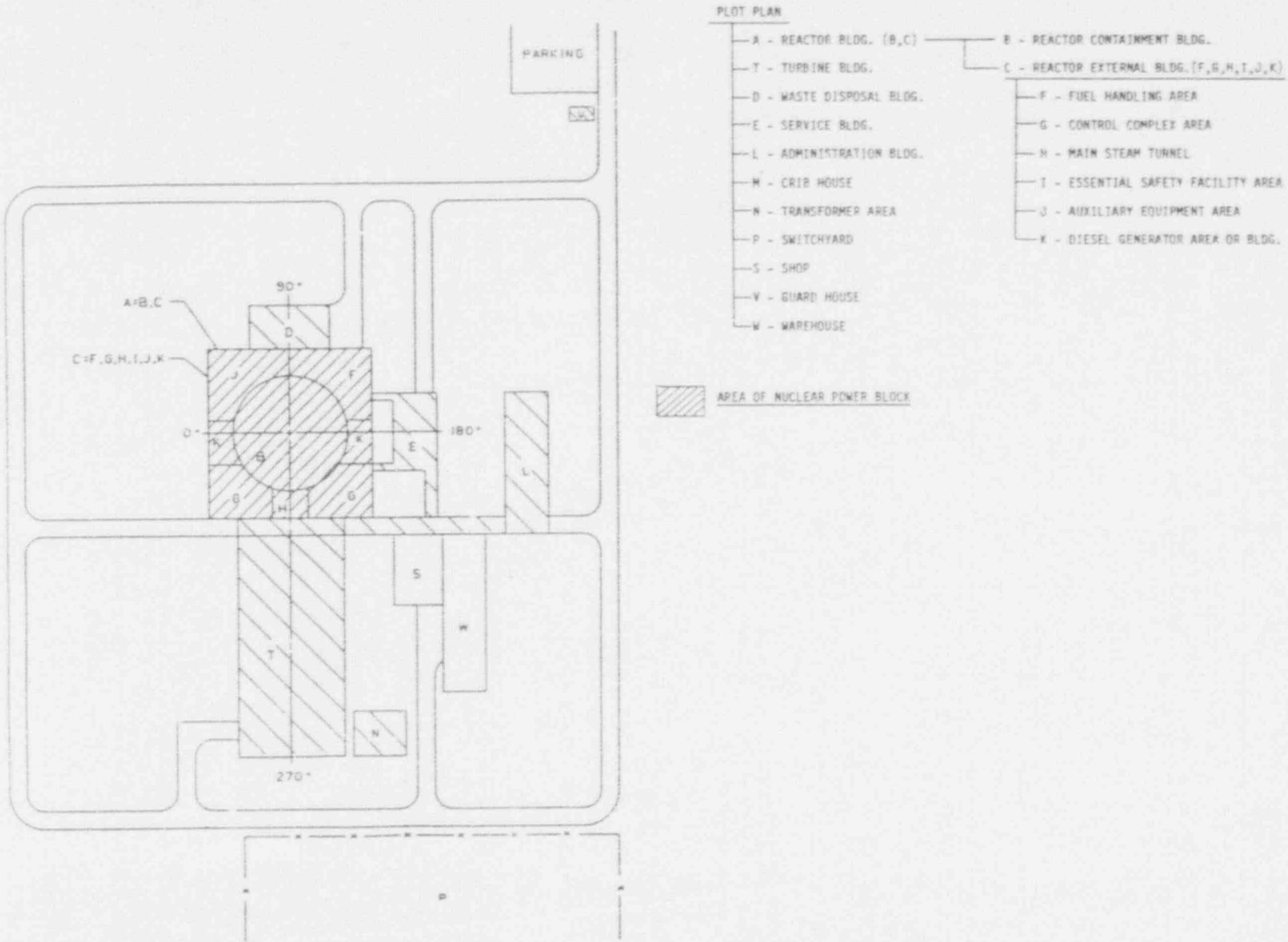

Figure 1.1 Site plan

Table 1.3 RESAR SP/90 nuclear power block scope/interface

Area/Item	NPB[1]	DC[2]	IC[3]
Mechanical:			
Reactor coolant system	x		
Integrated safeguards system	x		
- residual heat removal	x		
- emergency core cooling	x		
- containment spray	x		
Emergency feedwater system	x		
Chemical and volume control system	x		
Service water system		x	
Component cooling water system	x		
Spent fuel pool cooling and cleanup system	x		
Chilled water system	x		
Heating ventilating, air conditioning and filtration systems (for the systems, equipment and buildings in the NPB)	x		
Ultimate heat sink			x
Turbine generator			x
Other power conversion systems			x
Main steam system	x	x	
Main feedwater system	x	x	
Startup feedwater system		x	
Steam dump system		x	
Combustible gas control systems	x		
Containment leak testing systems	x		
Nuclear sampling system	x		
Post-accident sampling system	x		
Fire protection systems	x		x
Diesel generator auxiliary systems	x		
Steam generator blowdown processing system	x		
Steam generator wet layup system	x		
Reactor makeup water system	x		
Boron recycle system	x		
Equipment and floor drain system	x		
Liquid waste processing system	x		
Gaseous waste processing system	x		
Instrument compressed air systems	x		
Service gas systems			x
Piping and supports	x		
Fuel-handling, storage, and refueling equipment	x		
Materials and equipment handling, cranes and hoists	x		
Integrated reactor vessel head package	x		
Electrical:			
Electrical power systems serving NPB equipment including:			
- Class 1E electrical power systems	x		

See notes at the end of the table.

Table 1.3 (Continued)

Area/Item	NPB[1]	DC[2]	IC[3]
Electrical: (continued)			
- emergency diesel generators	x		
- rod control and power system	x		
- essential instrumentation ac power system including inverters	x		
- dc power system and batteries	x		
- lighting systems for the NPB	x		
- wiring, cabling, cable trays and supports	x		
- onsite auxiliary power system			x
- offsite power system			x
Instrumentation and controls:			
Process instrumentation and control systems related to the NPB	x		
Rod position indication	x		
Integrated control and integrated protection systems	x		
Station data processing systems	x		
Main control room and technical support center	x	x	x
Process effluent and radiation monitoring system	x		
Fire detection system	x		
Remote shutdown panel	x		
Communication systems	x		
Environmental monitoring system			x
Turbine generator and power conversion auxilary systems I&C	x		
Post-accident monitoring systems	x		
Buildings and Structures:			
Containment building	x		
Containment shield building	x		
Reactor external building (or the auxiliary building(s) housing the same complement equipment as needed to make up the NPB) For example:	x		
- essential mechanical equipment	x		
- diesels and auxiliaries	x		
- essential electrical equipment	x		
- control room	x		
- fuel-handling and storage areas			
Waste disposal building			x
Necessary equipment supports and restraints	x		
Turbine building			x
Service water structure and dams			x

[1] NPB = Nuclear power block scope of RESAR SP/90.
[2] DC = Design criteria.
[3] IC = Interface criteria.

equipment ... is not safety related from safety-related equipment, for example, train A c... nts from the train B components and radioactive (dirty) components from ... nonradioactive (clean) components.

The majority of component areas that are not safety related will be located in control areas classified as radioactive and those component areas that are safety related will be located in control areas not classified radioactive. The only safety-related component areas that are classified as radioactive areas are the four ISS safeguard component areas (SCA) that will be located in the shadow area beneath the sphere.

The reactor external building boundary includes the building volume commonly referred to as the shadow area beneath the sphere. The building volume between the primary containment (SSCV) and the secondary containment (shield building) is subdivided into seven dedicated and totally separated zones. One of these seven zones is dedicated to the chemical and volume control system (CVCS) pumps, valves, and piping that are not safety related. Two of the zones are dedicated to the two subsystems of the emergency feedwater system (EFWS) and the remaining four zones serve as the four ISS SCA.

Several key areas of the reactor external building are (1) the main control room (MCR); (2) the train A and B diesel generator rooms, which will be located in separate wings; (3) the train A and B Class 1E switchgear rooms, located in separate wings; (4) the fuel-handling area, located in the north wing; (5) the main steam tunnel, located in the south wing; (6) the electrical penetration areas, located in the southeast and southwest quadrants; (7) the emergency feedwater storage tanks, located in the south wing; (8) the component cooling water (CCW) heat exchangers, located in the south wing; (9) the CCW pumps, located directly below the CCW heat exchangers; and (10) the majority of the heating, ventilation, and air conditioning equipment.

The space between the primary containment building and the secondary containment building above the shadow area is not considered part of the reactor external building; this space is designated the annulus area.

1.2.1 Reactor System

The reactor system for the RESAR SP/90 design includes the reactor vessel, integrated head, reactor internals, control rod drive mechanisms, displacer rod drive mechanism, and the reactor core, which includes fuel assemblies, water-displacer rod assemblies, gray rod assemblies, and rod cluster control assemblies. Unique features of the design include (1) a low power density, (2) a moderator control system, and (3) a radial neutron reflector.

Westinghouse considers the RESAR SP/90 to have a significantly reduced power density as compared to other contemporary core designs for pressurized-water reactors. The core is larger in diameter, contains more fuel rods (19x19 fuel array), and has more fuel. The additional fuel loading will result in significant reductions in specific power (kW/kg), average linear power, and average rod heat flux (Btu/hr-ft^2). The lower average linear power will significantly reduce peak cladding temperature in a large-break loss-of-coolant accident (LOCA). The lower average rod heat flux will provide additional DNB (departure from nucleate boiling) margin.

For a given burnup, the increase in fuel loading will reduce the fraction of the total core loading that must be replaced at the end of the fuel cycle. The result for the same energy extraction is a reduction in the required feed enrichment. The low power density will result in a lower cycle burnup (megawatt day/ per metric ton of uranium [MWD/MTU]) because of the additional fuel loading, which will increase the number of zones or reduce the fraction of the core replaced. This will result in a lower core average burnup at end of cycle, which will reduce the required feed region enrichment.

The RESAR SP/90 moderator control system is designed to control excess reactivity by varying the amount of moderator in the core instead of using control poisons for neutron absorption. This control of reactivity will be achieved by displacing water volume in the fuel lattice during the first part of the fuel cycle and returning it later in the cycle as needed. With less water in the lattice, less neutron moderation will occur and neutrons will remain at resonant energies for a longer period of time, thus increasing neutron absorption in the fertile material, U-238, and producing more plutonium. When additional reactivity is required later in the cycle, displacer rods will be removed, thereby increasing the water content of the fuel lattice, increasing neutron moderation, and reducing the probability of fertile capture, which results in the depletion of the plutonium produced earlier in the cycle. The end result will be that the amount of fissile uranium and plutonium remaining at end of life will be about the same as in a poison-controlled core. However, the initial core enrichment will be much lower, which will result in an additional savings in ore and enrichment (separative work) requirements.

Physically, the core water content will be varied by inserting or withdrawing banks of Zircaloy-clad rods, called water-displacer rods, which contain pellets made of zirconium oxide (ZrO_2). The primary effect of these rods on core reactivity is the displacement of water, as they have a very low neutron absorption probability.

The radial neutron reflector will consist of a closely packed array of stainless-steel rods assembled in cans and located on the core periphery. It will replace the current baffel-former structure located between the barrel and fuel. Its benefit will be a reduction in net neutron leakage, which increases core reactivity and reduces feed enrichment requirements. The result will be a substantial savings in ore use with a potential for increased benefit from a low-leakage-fuel-management scheme. The reflector design also will help to reduce reactor vessel fluence levels.

The advanced reactor core will use enriched U-238 fuel. Fuel rods will consist of stacked ceramic UO_2 pellets clad in Zircaloy tubing arranged in a 19x19 array.

The RESAR SP/90 will use a control element (either a rod control cluster, gray rod cluster, or water-displacer rod cluster) in 185 of the 193 fuel assemblies. A rod drive mechanism is required to move each of those 185 control elements. The control rod cluster and gray rod clusters, which will be used to control reactor power and to shut down the reactor, will be positioned using the conventional, magnetic-jack-type drive mechanism, which provides a stepwise movement of the control rods. The 88 water-displacer rod clusters will be positioned either fully inserted or fully withdrawn from the core by means of a hydraulic mechanism called a displacer rod drive mechanism (DRDM). The DRDM will be composed of a pressure housing, a hydraulic cyclinder, the mechanical latching device, and a vent system.

The similarity of RESAR SP/90 and RESAR 414 is illustrated in Figure 1.2, which shows the equivalent inlet nozzle, downcomer, lower plenum, and upper plenum or calandria regions. The two designs are similar except for changes in the region from the upper core plate to the outlet nozzle. Because of the increased number of control elements that must be moved in the rod travel space, a new calandria structure will be provided in the RESAR SP/90 design, above the rod guide region, to turn the flow to the outlet nozzles. This approach provides for axial flow in the rod guide region thereby minimizing the potential for flow-induced vibration. The addition of a calandria at the outlet nozzle elevation results in a longer reactor vessel. The upper core plate will be much thicker to accept axial loading so that support columns will not be needed.

1.2.2 Reactor Coolant System

The RESAR SP/90 design for the RCS consists of four closed heat-transfer loops connected in parallel to the reactor vessel. Each loop contains a steam generator and a reactor coolant pump. In addition, the system includes a pressurizer, a pressurizer relief tank, and the valves and instrumentation necessary for operational control and safeguards actuation. Also included in the RCS is a reactor vessel head vent and a reactor vessel level instrumentation system (RVLIS) as well as a vent system for the DRDM, which will be used in connection with operation of the DRDMs. All system equipment will be located in the reactor containment, except certain containment isolation and process-actuated valves located in the lines connected to the pressurizer relief tank. A simplified flow diagram of the system is shown in Figure 1.3.

1.3 Comparison With Similar Facility Designs

Many features in RESAR SP/90 represent new Westinghouse designs. However, many design aspects of the facility are similar to facilities that the staff has approved previously. To the extent feasible and appropriate, the staff has relied on these earlier reviews for those RESAR SP/90 design features that were shown to be substantially the same as those previously considered. Where this has been done, the appropriate sections of this report identify the other facilities involved. The NRC safety evaluation reports for these other facilities have been published and are available for public inspection at the NRC Public Document Room at 2120 L Street, N.W., Washington, D.C. 20555.

1.4 Identification of Agents and Contractors

The Westinghouse Electric Corporation (Westinghouse) is the principal designer and contractor for the RESAR SP/90 design. However, portions of the RESAR SP/90 design (principally non-nuclear steam supply systems) were the result of a cooperative development effort between Westinghouse and Mitsubishi Heavy Industries (MHI).

1.5 Summary of Principal Review Matters

The staff's technical review and evaluation of the information submitted by the applicant considered the principal matters summarized below.

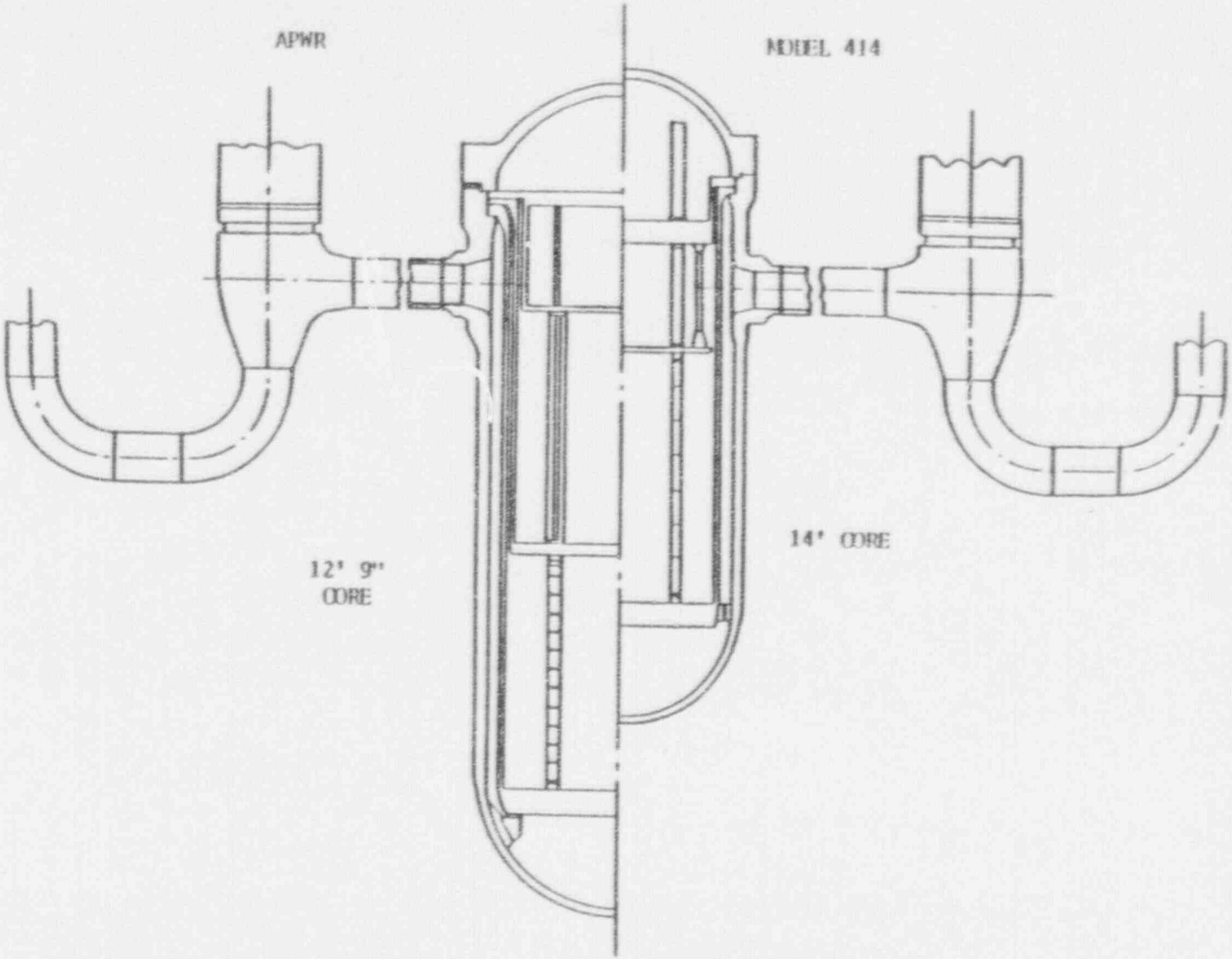

Figure 1.2 Comparison of RESAR SP/90 and RESAR 414

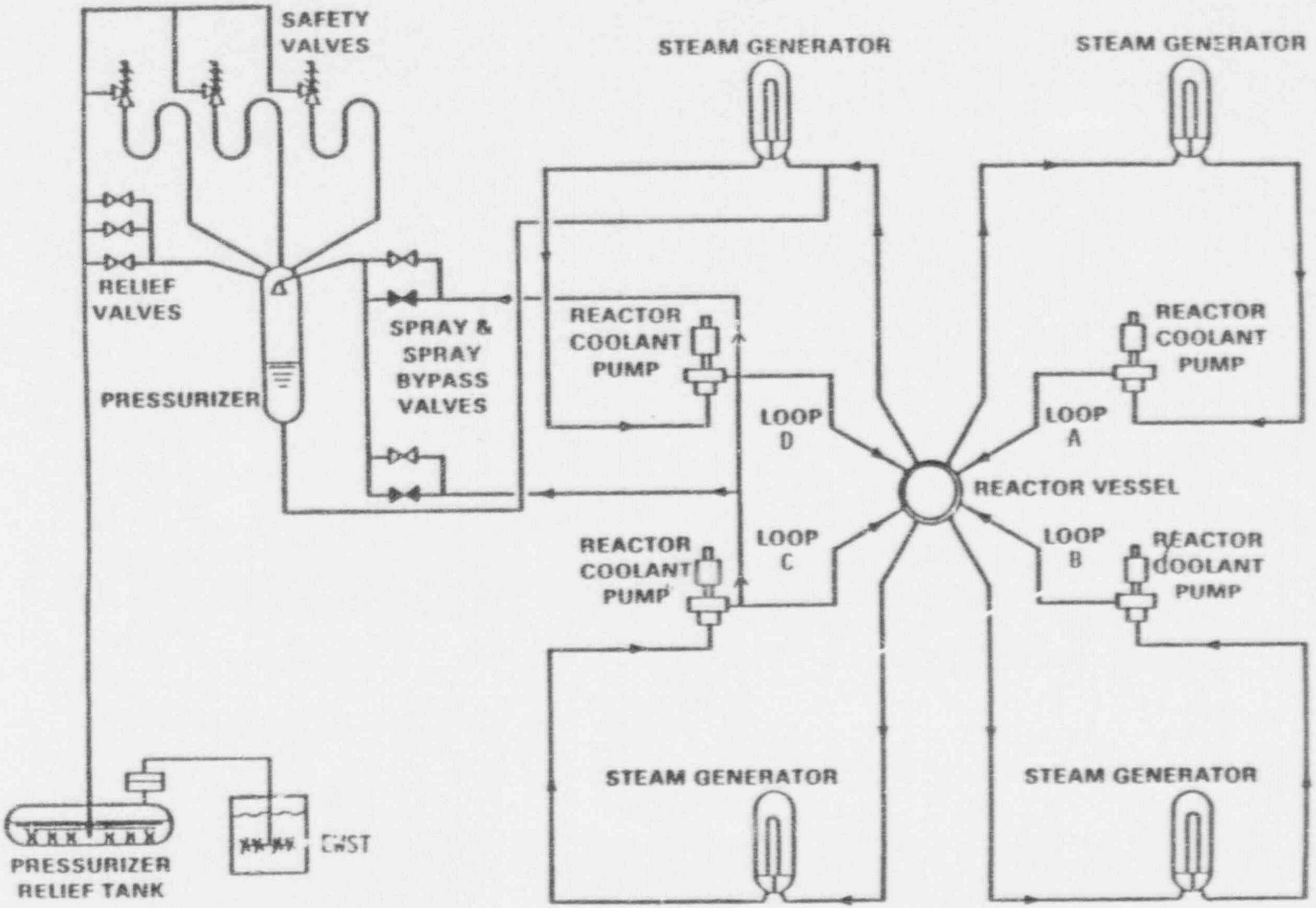

Figure 1.3 Reactor coolant system

(1) The design, fabrication, and testing criteria, and expected performance characteristics of the system and components important to safety were reviewed to determine that they are in accord with the Commission's general design criteria (GDC), quality assurance criteria (QAC), regulatory guides, and other appropriate rules, codes, and standards, and that any departures from these criteria, codes, and standards, have been identified and justified.

(2) The expected responses of RESAR SP/90 to various anticipated operating transients and to a broad spectrum of postulated accidents were evaluated, and the staff determined that the potential consequences of a few highly unlikely postulated accidents (design-basis accidents [DBAs]) would exceed those of all other accidents considered. The staff performed conservative analyses of these DBAs to determine that the calculated potential offsite radiation doses that might result, in the very unlikely event of their occurrence, would not exceed the Commission's guidelines for site acceptability given in 10 CFR 100 for the RESAR SP/90 site envelope.

(3) In addition to the above review matters, the Commission's Severe Accident Policy Statement identified two additional criteria to be addressed by the applicant for certification of a new reactor design: The applicant must:

- demonstrate technical resolution of all applicable unresolved safety issues and medium- and high-priority generic safety issues

- complete a probabilistic risk assessment and consider the severe accident vulnerabilities it exposes along with the insights that it may add to the assurance of no undue risk to public health and safety

The staff did not review all the applicable unresolved safety issues and generic safety issues in detail except where noted in this report. The applicant's final resolution of these issues will be reviewed during the FDA stage.

The staff's review of the probabilistic risk assessment is discussed in Chapter 19 of this report.

1.6 Outstanding Issues

As a result of the NRC review of the safety aspects of the RESAR SP/90 application, and due to the incomplete design, a number of items remain outstanding at the time of this report. Because it has not completed its review and reached final positions in these areas, the staff considers these issues to be open. These issues require resolution during either the review of a plant-specific application referencing RESAR SP/90 or during the review of the FDA application. The open items, with appropriate references to sections of this report given in parentheses, are listed below.

(1) scope of design (1.5, 1.7)
(2) meteorology (2.3)
(3) waves and wave runup (2.4.1)

(4) design basis roof loading/probable maximum precipitation (2.4.1)
(5) site drainage (2.4.1)
(6) ultimate heat sink (2.4.2, 9.2)
(7) seismic and geologic siting criteria (2.5, 3.7.1, 20.2.4)
(8) seismic classifications (3.2.1)
(9) system quality group classification (3.2.2)
(10) instrument sensing line supports (3.2.2)
(11) balance-of-plant Category I structure flood protection (3.4.1)
(12) site-dependent flood protection design procedures (3.4.2)
(13) effect of turbine missiles (3.5.1.3)
(14) missiles generated by natural phenomena (3.5.1.4)
(15) ultimate heat sink missile protection (3.5.2)
(16) balance of plant protection from internally generated missiles (3.5.4)
(17) plant design for protection against postulated piping failures in fluid systems outside containment (3.6.1)
(18) postulated pipe break criteria (3.6.2)
(19) leak-before-break evaluations (3.6.2)
(20) power spectral density (3.7.1)
(21) seismic instrumentation ISI program (3.7.3)
(22) containment design audit (3.8.1)
(23) effect of plant-specific Category I structures on proposed standard design (3.8.3)
(24) FATCON and WESAN computer code verification procedures (3.9.1)
(25) ASCE analysis standard for time-history solutions and response spectrum analysis (3.9.2.2)
(26) combination of closely spaced modes in seismic response analysis (3.9.2.2)
(27) independent support method analysis (3.9.2.2)
(28) preoperational flow-induced vibration test program requirements (3.9.2.3)
(29) structural design adequacy of reactor internals important to safety (3.9.2.-)
(30) stress limits for Class 2 and 3 valves (3.9.3.1)
(31) design criteria for HVAC ductwork and supports (3.9.3.1)
(32) pipe support base plate and anchor bolt design - IEB 79-02 (3.9.3.1)
(33) thermal stratification in lines attached to the reactor coolant system (3.9.3.1)
(34) analysis of limiting ATWS events (3.9.3.2, 20.2.2)
(35) inservice testing of pumps and valves (3.9.6, 20.2.4)
(36) seismic and dynamic qualification (3.10)
(37) environmental qualification of mechanical and electrical equipment (3.11, 20.2.3)
(38) approval of WCAP-9500 for use with 19x19 fuel assembly (4.2.1)
(39) fuel system design analysis and testing results (4.2.2)
(40) uncertainty analysis for four-segment ex-core detectors (4.3.1)
(41) reactor coolant system flow uncertainty (4.4.3)
(42) inadequate core cooling instrumentation (4.4.4)
(43) compliance with 10 CFR 50.55a - table identifying ASME component code, code edition, applicable addenda (5.2.1.1)
(44) identification of all code cases used in the construction of all SP/90 components of ASME Classes 1, 2 and 3 (5.2.1.2)
(45) LTOP design criteria (5.2.2.2)
(46) LOCA in modes 3 and 4 (5.2.2.2)

(47) ASME Section III Appendix G - fracture toughness (5.2.3, 5.3.1)
(48) preservice inspection program (5.2.4.1)
(49) inservice inspection program (5.2.4.1, 5.2.4.2)
(50) reactor vessel surveillance capsules - 10 CFR 50, Appendix G (5.3.1)
(51) reactor vessel pressure and temperature limits - 10 CFR 50, Appendices G and H (5.3.1, 5.3.2)
(52) Regulatory Guide 1.99 to calculate increase in RT_{NDT} (5.3.2)
(53) steam generator inservice inspection program (5.4.2)
(54) preoperational and initial startup test program (5.4.3.5)
(55) boron mixing and natural circulation test (5.4.3.5)
(56) operating procedures and administrative controls for lowered RCS inventory operation to prevent loss of RHRS (5.4.3.5, 20.2.2)
(57) protective coating failure (6.1.2)
(58) post-accident emergency cooling water chemistry technical specifications (6.1.3)
(59) subcompartment analysis (6.2.1.2)
(60) maximum pressure differential analysis for the containment fan coolers and ductwork (6.2.2)
(61) reduction of containment pressure following a design-basis accident (6.2.2, 20.2.3)
(62) evaluation of high-energy line breaks on secondary containment (6.2.3)
(63) containment purge system for hydrogen control (6.2.5, 20.2.3)
(64) hydrogen igniter power supply (6.2.5)
(65) containment leakage testing (6.2.6)
(66) minimum EWST water level analysis (6.3.1)
(67) technical specifications covering IST and ISI for the ECCS (6.3.4, 20.2.4)
(68) large break LOCA evaluation model with BASH (6.3.5)
(69) assumptions and calculations for most limiting LBLOCA, SBLOCA and ECCS flow requirements (6.3.5)
(70) control room dose analysis for a postulated LOCA in accordance with GDC 19 (6.4)
(71) control room toxic gas calculation (6.4)
(72) evaluation of the radiological consequences from postulated accidents (6.5.3)
(73) preservice and inservice inspection program for ASME class 2 and 3 components (6.6)
(74) interface requirement implementation (7.1.1)
(75) defense-in-depth analysis of the integrated protection system in accordance with NUREG-0493, Section 3.3 (7.1.2.5)
(76) testing program for the integrated protection system (7.1.2.5)
(77) verification and validation in accordance with ANSI/IEEE Std. 7-4.3.2-1982, "Application Criteria for Programmable Digital Computer Systems In Safety Systems of Nuclear Power Generating Stations" (7.1.2.5)
(78) safety related instrumentation deviations from Regulatory Guide 1.97, Revision 3 (7.5.1)
(79) signal selector adequacy and reliability (7.7.10)
(80) electrical power transmission network (8.2.1, 20.2.2)
(81) grid stability analysis (8.2.3, 20.2.2)
(82) containment electrical penetrations (8.4.1)
(83) thermal overload protection bypass (8.4.2)
(84) power lockout to motor-operated valves and redundant valve status indication (8.4.3)

(85) implementation verification of physical identification and independence of redundant safety-related electrical systems (8.4.4)
(86) load sequencing reliability (8.4.7)
(87) station blackout (8.4.8, 20.2.2)
(88) heat removal rate analysis for spent fuel pool heat exchangers (9.1.3)
(89) load drop analysis for damage to spent fuel (9.1.4)
(90) detailed design of the essential service water system (9.2.1)
(91) component cooling water system leak detection and makeup following a design-basis accident (9.2.2)
(92) detailed design of the domestic water system (9.2.4)
(93) detailed design of the ultimate heat sink system (9.2.5)
(94) detailed design of the condensate storage and transfer system (9.2.6)
(95) detailed design of the compressed air system (9.3.1)
(96) reactor external building ventilation system (9.4.2)
(97) post-accident sampling system (9.3.2.2)
(98) radwaste building ventilation system (9.4.3)
(99) turbine building ventilation system (9.4.4)
(100) annulus air cleanup and fuel building emergency exhaust system flow rates (9.4.5)
(101) essential service water pump structure ventilation system (9.4.8)
(102) plant heating system (9.4.9)
(103) fire hazards analysis (9.5.1.1, 9.5.1.3.3, 9.5.1.3.4, 9.5.1.4.8, 20.2.2)
(104) spatial separation of safety-related systems (9.5.1.2.1, 20.2.2)
(105) main control room automatically actuated fire extinguishing system (9.5.1.2.1, 20.2.2)
(106) operator action for main control room fire protection (9.5.1.2.1, 20.2.2)
(107) pressurizer relief system separation of redundant safety related components inside containment (9.5.1.2.1, 20.2.2)
(108) administrative controls (9.5.1.5, 20.2.2)
(109) failure mode and effects analysis for emergency and essential lighting systems (9.5.3.3)
(110) emergency diesel auxiliary system design (9.5.4)
(111) detailed turbine generator design (10.2)
(112) turbine disk integrity (10.2.1)
(113) main steam supply system (downstream of the MSIVs) detailed design (10.3.2)
(114) steam and feedwater systems materials (10.3
(115) secondary water chemistry (10.3.4)
(116) main condenser (10.4.1)
(117) main condenser evacuation system (10.4.2)
(118) turbine gland sealing system (10.4.3)
(119) turbine bypass system (10.4.4)
(120) circulating water system (10.4.5)
(121) condensate cleanup system (10.4.6)
(122) condensate and feedwater system (10.4.7)
(123) steam generator design (10.4.7)
(124) steam generator blowdown processing system (10.4.8)
(125) startup feedwater system (10.4.10)
(126) secondary liquid waste system (10.4.11)
(127) design of the radwaste building ventilation system (11.3)
(128) design of the turbine building ventilation system (11.3)
(129) offsite dose calculations (11.3)
(130) single failure analysis in the waste gas system (11.3)

(131) discrepancies in gaseous waste management systems (11.3)
(132 design of the solid waste processing system (11.4)
(133) process and effluent radiological monitoring and sampling systems (11.5.1, 11.5.2)
(134) ALARA policy and operational considerations (12.1.1, 12.1.3)
(135) reactor radiation source values for the at-power condition (12.2.1)
(136) tabulated airborne concentrations and expected airborne radioactivity levels for normal operation and anticipated occurrences (12.2.1)
(137) ventilation systems design (12.2.1, 12.3.3)
(138) plant-specific shielding (12.3.2)
(139) criticality accident alarm system for the spent fuel storage area (12.3.4.1)
(140) airborne radioactivity monitoring instrumentation (12.3.4.2)
(141) operational radiation protection program (12.5)
(142) conduct of operations (13.0)
(143) physical security (13.0)
(144) testing and maintenance programs (14.0, 20.2.4)
(145) technical specification setpoints for Category II events (15.1)
(146) 10 CFR 100 criteria for Category III events (15.1)
(147) criteria regarding limiting transients for Category IV events (15.1)
(148) loss of offsite power with a rod ejection (15.1)
(149) DNBR curve for steam generator relief valve or steam generator safety valve opening (15.2.1)
(150) DNBR/nuclear power transient curves for steam system piping failures (15.2.1)
(151) applicability of Westinghouse Topical Report WCAP-9226 regarding steamline rupture cases (15.2.1)
(152) stuck open PORV with fuel failure impact on radiological consequence calculations (15.2.3)
(153) time delay of reactor trip during a SGTR (15.2.5)
(154) analysis of a SGTR with a failed open PORV (15.2.5)
(155) ECCS evaluation model for LBLOCA (15.2.5)
(156) ECCS evaluation model for SBLOCA (15.2.5)
(157) dropped rod or bank analysis (15.3.3)
(158) ejected rod analysis for service level C (15.3.5)
(159) radiological consequences of design-basis accidents (15.4)
(160) site-specific control room habitability system (15.4.1)
(161) steam generator tube rupture (15.4.2)
(162) radiological consequences of the failure of a small primary coolant line outside containment (15.4.3)
(163) radiological consequences of a rod ejection accident (15.4.4)
(164) radiological consequences of a fuel handling accident (15.4.5)
(165) Technical Specifications (16)
(166) application of 10 CFR 50, Appendix B, quality assurance criteria (17.3)
(167) control room design (18)
(168) probabilistic risk assessment (19)
(169) safety issues requiring departure from current regulatory rule, guidance, or policy (20)
(170) NRC unresolved safety issues (Appendix C of this report)

1.7 Interface Information

As discussed in Section 1.1 of this report, the RESAR SP/90 does not describe an entire facility, as a result, the design description also defines interface requirements that must be imposed on the plant-specific applicant referencing RESAR SP/90 so that the remainder of the plant will provide compatible design features that will ensure the applicability, functional performance, and safe operation of the RESAR SP/90 systems.

The interface requirements range in type from general design provisions to detailed provisions. In some cases, the interfaces are in the form of detailed design analyses that are needed by the designer of the remainder of the plant for the analyses of interconnected systems and structures. The staff's review and the evaluation reflected in this report address the interface requirements either from the standpoint of general design provisions (i.e., qualitative); specific design provisions (i.e., quantitative), by incorporation of or reference to the interface requirements in the RESAR SP/90 Safety Analysis Report (SAR), or by a description of the interface mechanism between RESAR SP/90 and the remainder of the plant.

During the course of the review, the staff reviewed the detailed interface information that will be supplied to the plant-specific applicant referencing the RESAR SP/90 and that are discussed in various sections of this report. This interface information has always been a part of the contractual arrangements between the NSSS designer and balance-of-plant (BOP) designer (architect-engineer). However, for the purpose of a standard nuclear island design, the safety-related interface requirements are significant for future referencing purposes. The staff expects to see less interface information in a FDA/DC application submitted pursuant to 10 CFR 52 since the scope of the application will be for an essentially complete design.

1.8 Unresolved Safety Issues

The staff continuously evaluates the safety requirements used in its review against new information as it becomes available. In some cases, the staff takes immediate action or interim measures to ensure safety. In most cases, however, the initial staff assessment indicates that immediate licensing actions or changes in licensing criteria are not necessary. In any event, further study may be deemed appropriate to make judgments as to whether existing staff requirements should be modified. These issues being studied are sometimes called generic safety issues because they are related to a particular class or type of nuclear facility and have potentially significant public safety implications. One of the criterion identified in the Commission's Severe Accident Policy Statement requires the applicant to demonstrate technical resolution of all applicable unresolved safety issues and medium- and high-priority generic safety issues. A discussion of these matters is provided in Appendix C to this report.

2 SITE CHARACTERISTICS (RESAR SP/90 Module 3, Section 2.0)

Site characteristics pertinent to the staff's review include site boundary distances, population distributions, and nearby industrial, transportation, and military facilities as well as meteorology, hydrology, and geology. These characteristics are site specific and generally beyond the scope of RESAR SP/90. However, on the basis of information provided in RESAR SP/90, the staff's evaluation of these matters is discussed below.

2.1 Geography and Demography (RESAR SP/90 Module 3, Section 2.1)

2.2 Nearby Industrial, Transportation, and Military Facilities (RESAR SP/90 Module 3, Section 2.2, and Appendix 2A, Section 2A.2)

2.3 Meteorology (RESAR SP/90 Module 3, Section 2.3)

The staff will review site-specific safety issues on a case-by-case basis with submittal of a plant-specific application in accordance with Sections 2.1.1, 2.1.2, 2.1.3, 2.2.1, 2.2.2, 2.2.3, 3.5.1.5 , and 3.5.1.6 of the Standard Review Plan (SRP) (NUREG-0800). Where site conditions are determined to impact safety, the staff will require appropriate plant designs or site modifications, and the review findings will be discussed in the plant-specific safety evaluation report (SER).

2.4 Hydrologic Engineering (RESAR SP/90 Module 3, Section 2.4)

2.4.1 Floods

RESAR SP/90 proposes an entrance floor level that is 0.3 meter (11.8 inches) above plant grade. The site interface requirement is that the design flood level, except local site runoff, be below grade, thus protecting the powerblock structures as a result of the external design-basis flood stillwater level. Although RESAR SP/90 has inferred in Chapter 3 of Module 7 that waves and wave runup are either included in the design-basis flood level or will be controlled by hardened protection (e.g., curbing, levies), the staff will review waves, wave runup, and associated effects in site-specific applications that use this design. RESAR SP/90 references Regulatory Guides (RGs) 1.59, "Design Basis Flood for Nuclear Power Plants," and 1.102, "Flood Protection for Nuclear Power Plants." The staff concludes that these bases are acceptable criteria and notes that the wave and wave runup portion of external flooding will be reviewed with the individual plant-specific license applications.

RESAR SP/90 states that roofs of safety-related structures will be designed for a snow load of 80 lb/ft^2, which is acceptable for most contiguous U.S. locations. However, snow loads for some extreme northern contiguous U.S. locations and Alaska may exceed the design value of 80 lb/ft^2. The applicant states in Amendment 2 to RESAR SP/90 Module 3 that the probable maximum precipitation (PMP) in a 5-minute period is equal to or less than 6 inches. This is the maximum value from Hydrometeorological Report No. 52* and is conservative, although there may be a few locations, especially mountainous regions, where this value could be exceeded. Additionally, natural phenomena criteria are subject to change with time and the accumulation of better data. Therefore, it is the staff's position that the design-basis roof load for snow and PMP will be reviewed with the plant-specific application referencing this design.

The criteria utilized in RESAR SP/90 is generally conservative and acceptable for a standard plant design. Specific site drainage will be reviewed with individual plant-specific licensing applications to ensure that ponding above plant grade adjacent to safety-related buildings has been precluded. The staff notes that the 12-inch difference in elevation between plant grade and floor entrance level is generally more than adequate.

2.4.2 Ultimate Heat Sink

This is a subject that is not included in the RESAR SP/90 design and will be reviewed during plant-specific licensing applications. Interface requirements that need to be met by the ultimate heat sink will be discussed in Section 9.2.5 of this SER.

2.4.3 Groundwater

The RESAR SP/90 design includes provisions for groundwater levels up to plant grade in the design of the facility. The proposed flood and groundwater criteria are acceptable to the staff.

2.5 Geology, Seismology, and Geotechnical Engineering (RESAR SP/90 Module 3, Section 2.5 and Appendix 2A, Section 2A.5)

Because RE SP/90 is a standard plant design, the material usually provided under this tion in accordance with the Standard Review Plan (SRP, NUREG-0800) is not inclu d. However, Appendix 2A to Chapter 2 of Module 3 contains information with regard to the seismic design parameters used in the nuclear power block design.

Although this and information in Section 3.7 of Module 7 are of a general nature, the staff is concerned that unconditional acceptance of these statements may lead to design problems in the future when specific sites are chosen for plant construction. In Module 3, Appendix 2A, Section 2A.5.2, the applicant commits to a seismic design that includes a safe-shutdown earthquake (SSE) anchored to a 0.3g zero period acceleration (ZPA) response spectrum and an operating-basis earthquake (OBE) anchored to a 0.1g ZPA response spectrum in accordance with Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants." In discussing the merits of these seismic design criteria, the applicant indicates that the selection of these seismic design criteria are adequate for the Central United States and the Eastern United States.

*U.S. Department of Commerce, August 1982.

Detailed studies* of seismic hazards in the Central and Eastern United States indicate that the seismic hazards in these zones vary significantly and are dependent on the interpretations of tectonic frameworks, attenuation formulae, and other geophysical characteristics. In some cases, the seismic design parameters are dominated by local site characteristics (e.g., NUREG-1057, Safety Evaluation Report for Beaver Valley Unit 2, October 1985, and NUREG-0853, SER for Clinton Units 1 and 2, February 1982, and Supplement 3, May 1984). In evaluating the adequacy of the seismic design anchored to an SSE of 0.3g, the staff agrees that this value could, in general terms, be considered conservative for many sites in the Central and Eastern United States. However, localized exceedances of this value cannot categorically be ruled out.

In the past the staff has considered requests for an OBE design basis in which the OBE was less than one-half of the SSE on a plant-specific basis. In these cases the specified OBE was evaluated on the basis of the geology and seismology of the site under consideration. However, RESAR SP/90 relates to a standard design, which by its nature is not site specific. Therefore, its design should agree with general guidelines set forth by Criterion 2 of 10 CFR 50, Appendix A, and Paragraph V of 10 CFR 100, Appendix A, which require, in part, that the OBE shall be no less than one-half of the SSE for seismic design considerations.

The staff concludes that the statements in the RESAR SP/90 application indicating that the 0.3g SSE and 0.1g OBE are adequate and conservative seismic design parameters for the entire Central and Eastern United States cannot be supported because of the uncertainties still existing in the state-of-the-art of seismic hazard evaluations. Therefore, seismic and geologic siting criteria will be reviewed during the individual plant-specific licensing review process, an integral part of which is the determination of the SSE for the site considered. In addition, present guidelines require the OBE to be no less than one-half of the SSE.

*NUREG/CR-3756, "Seismic Hazard Characterization of the Eastern United States," NRC, April 1984; EPRI NP-4726, "Seismic Hazard Methodology for the Central and Eastern United States," Electric Power Research Institute, July 1986; and NUREG/CR-5250, "Seismic Hazard Characterization of 69 Nuclear Plant Sites East of the Rocky Mountains," January 1989.

3 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS (RESAR SP/90 Module 7, Section 3.0)

3.1 General (RESAR SP/90 Module 7, Section 3.1)

In Section 3.1 of Module 7 of the RESAR SP/90, the applicant discusses conformance of structures, components, equipment, and systems to the general design criteria (GDC) in 10 CFR 50, Appendix A. Using this information, the staff reviewed the design criteria to verify that RESAR SP/90 will be designed to meet the GDC.

The staff relies heavily on the application of industry codes and standards that have been used as accepted industry practice in its review of structures, components, equipment, and systems. The codes and standards cited in this report have been previously reviewed by the staff, found acceptable, and incorporated into the Standard Review Plan (SRP, NUREG-0800).

3.2 Classification of Structures, Components, Equipment, and Systems (RESAR SP/90 Module 7, Section 3.2)

3.2.1 Seismic Classification (RESAR SP/90 Module 7, Section 3.2.1)

GDC 2, "Design Bases for Protection Against Natural Phenomena," in part, requires that nuclear power plant structures, systems, and components important to safety be designed to withstand the effects of earthquakes without losing the capability to perform their safety function. Certain features that are designed to remain functional are necessary to ensure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (3) the capability to prevent or mitigate the consequence of accidents that could result in offsite exposures that have the potential to exceed the guidelines of 10 CFR 100. The earthquake for which these safety-related plant features are designed is defined as the safe shutdown earthquake (SSE) in 10 CFR 100, Appendix A. The SSE is defined on the basis of an evaluation of the maximum earthquake potential at the site location and is that earthquake that produces the maximum vibratory ground motion for which structures, systems, and components are designed to remain functional. Those plant features that are designed to remain functional if an SSE occurs are designated seismic Category I in Regulatory Guide 1.29, "Seismic Design Classification." The staff reviewed the design of RESAR SP/90 in accordance with SRP Section 3.2.1, which references Regulatory Guide 1.29.

The structures, components, equipment, and systems of RESAR SP/90 that are required to be designed to withstand the effects of an SSE and remain functional are identified in Table 3.2-1 of Module 7 of the RESAR SP/90 application. This table, in part, identifies major components in fluid systems, mechanical systems, and associated structures designated as seismic Category I. In addition, flow diagrams (piping and instrumentation diagrams) provided in the RESAR SP/90 application identify the interconnecting piping and valves and the boundary limits of each system classified as seismic Category I.

Westinghouse has stated in RESAR SP/90 that seismic Category I structures, components, and systems are designed to withstand the appropriate seismic loads and other applicable loads without loss of function. Seismic Category I structures are sufficiently isolated from non-Category I structures, or they are analyzed to ensure that their structural integrity is maintained during the postulated SSE. Non-seismic Category I systems, equipment, and components installed in seismic Category I structures whose failure could result in loss of required safety function of seismic Category I structures, equipment, systems, or components are either separated by distance or barrier from the affected structure, system, equipment, or component or designed together with their anchorages to maintain their structural integrity during the SSE. This commitment satisfies Regulatory Guide 1.29 relative to non-seismic Category I components.

The information regarding seismic classification in Table 3.2-1 of Module 7 is generally acceptable. However, the seismic classification of a few structures, systems, or components may be affected by the resolution of the system quality group classification issue that is discussed in Section 3.2.2 of this SER. Therefore, the staff cannot draw conclusions from its evaluation of seismic classification until the issue discussed in Section 3.2.2 has been resolved.

3.2.2 System Quality Group Classification (RESAR SP/90 Module 7, Section 3.2.2)

GDC 1, "Quality Standards and Records," requires that nuclear power plant 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. The pressure-retaining components are part of the reactor coolant pressure boundary (RCPB) and fluid systems important to safety and are necessary (1) to prevent or mitigate the consequences of accidents and malfunctions originating within the RCPB, (2) to shut down the reactor and maintain it in a safe shutdown condition, and (3) to retain radioactive material. The staff used Regulatory Guide 1.26, "Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," in its review for identifying on a functional basis the components of those systems important to safety as NRC Quality Groups A, B, C, or D. 10 CFR 50.55a specifies that components that are part of the RCPB must meet the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section III, Class 1 components.

Conformance of these RCPB components with 10 CFR 50.55a is discussed in Section 5.2.1.1 of this SER. These RCPB components are designated in Regulatory Guide 1.26 as Quality Group A. Certain other RCPB components that meet the exclusion requirements of 10 CFR 50.55a(c)(2) are classified Quality Group B in accordance with Regulatory Guide 1.26. The staff reviewed RESAR SP/90 in accordance with SRP Section 3.2.2, which references Regulatory Guide 1.26.

The applicant used American Nuclear Society (ANS) Safety Classes 1, 2, 3, and non-nuclear safety (NNS) as defined in American National Standards Institute ANSI/ANS 51.1-1983, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," in its classification of system components as an alternative method of meeting the guidance of Regulatory Guide 1.26. Safety Classes 1, 2, 3, and NNS correspond respectively to the Commission's Quality Groups A, B, C, and D in Regulatory Guide 1.26. However, the staff has not endorsed ANSI/ANS 51.1-1983 and cannot use that document in determining the acceptability of the quality group classification of structures, systems, and components. Therefore, the information in Table 3.2-1 of Module 7 is not completely acceptable. The staff has agreed that use of ANSI/ANS 51.1-1983 is acceptable for safety classification for pressure-retaining systems and components. However, ANSI/ANS 51.1-1983 may not be used for classifying all non-pressure-retaining components. One such example is the classification of the new and spent fuel storage racks. It is the staff's position that these components are important to safety and that the quality assurance criteria for new and spent fuel storage racks should be in conformance with 10 CFR 50, Appendix B, in addition to being classified as seismic Category I.

The staff has the following concerns related to the quality group classifications given to portions of the chemical and volume control system (CVCS): (1) For those portions of the CVCS that constitute the charging system and that provide seal injection to the reactor coolant pumps, including the water supply system, the applicant is requested to provide the basis for classifying these components as "non-safety related" and for not conforming to 10 CFR 50, Appendix B. (2) In its response to open items dated June 1989, Westinghouse stated that the shell side of the CVCS letdown and excess letdown heat exchangers is NNS but that it is required to mechanically support the tube bundles that are Safety Class 3. The staff agrees that the shell side functionally can be NNS. However, if the structural integrity of the tube side is partly dependent upon the structural integrity of the shell side, then the shell side should be classified as "important to safety."

In the same transmittal, Westinghouse stated that instead of a seal water heat exchanger to cool the flow returning from the reactor coolant pump (RCP) seals, the RESAR SP/90 design uses a seal injection heat exchanger to cool the water flowing into the seals. Table 3.2-1 of RESAR SP/90 classifies the tube side of this heat exchanger as Safety Class 3, seismic Category I but it is not on the Q-list for the CVCS. The shell side is NNS but seismic Category I. Westinghouse also described the bases for these classifications and presented examples of tests and analysis on RCP seals experiencing full temperature operation with a loss of seal injection. In all cases the pump/seal combinations achieved stable operation within the normal limits of pump seal operations. Results of the tests showed that the Model 100 RCP has weeks or months of endurance with no seal injection. The staff will further review the full RCP seal test results during the FDA stage.

The applicant's position regarding the safety class of portions of safety-related instrument sensing lines does not meet all of the guidelines in Regulatory Guide 1.151, "Instrument Sensing Lines," and, therefore, is not completely acceptable. Regulatory Positions C.2.b and C.3 of Regulatory Guide 1.151 state that instrument sensing lines that are connected to ASME Class 2 and Class 3 process piping, respectively, and that are used to actuate or monitor safety-related systems should not be less than ASME Class 2 (Safety Class 2) seismic Category 1 and ASME Class 3 (Safety Class 3) seismic Category 1, respectively, from their connection to the process piping or vessel to the sensing instrumentation. Westinghouse has stated that "safety-related instrument sensing lines will be designed in accordance with ASME III except for supports which will be seismic Category I, but will not be designed to ASME NF criteria. Installation will be in accordance with a QA Category I, non-ASME program that meets 10 CFR 50, Appendix B, requirements."

The staff takes the position that all safety-related instrument sensing lines and supports should conform to the guidelines of Regulatory Guide 1.151, or an acceptable alternate design must be provided. Westinghouse has not provided alternate design criteria for the sensing line supports to justify that the supports will not be designed to ASME NF criteria. This position is unacceptable to the staff and will be reviewed again during the FDA stage.

Westinghouse has stated that the reactor internals are classified Safety Class 3 in accordance with ANSI/ANS 51.1. Information in Subsection 3.9.5 of RESAR SP/90 Module 5, "Reactor System," clearly states that the core support structure will be constructed to ASME III, Subsection NG, "Core Support Structures." For the remainder of the reactor internals, Table 3.2-1 in both Module 5 and Module 7 state that the principal construction code is ASME III, Subsection NG. The staff can accept Safety Class 3, however, Section 3.9.5.3 of RESAR SP/90 Module 5 states that "internal structures are analyzed to meet the intent of the ASME Code in accordance with ASME III, Subsection NG, subarticle NG-3311(c). This is not acceptable because subarticle NG-3311(c) states that the requirements of Article NG-3000, "Core Support Structure Design," apply to internal structures only as specifically stipulated by the certificate holder that manufacturers the core supports.

Subarticle NG-3311(c) further states that the certificate holder shall certify that the design used for the internal structures shall not adversely affect the integrity of the core support structures. The certificate holder may be Westinghouse, however even if this is the case, in order to meet the guidelines in SRP Section 3.9.5 II.C, the staff requires a specific commitment from Westinghouse as the applicant, not the certificate holder, that the design criteria, loading conditions and analyses that provide the basis for the design of reactor internals, other than the core support structures, meet the guidelines of ASME III, Subsection NG, Article NG-3000.

It is the staff's intent to perform a complete review of system quality group classifications at the FDA stage.

3.3 Wind and Tornado Loadings (RESAR SP/90 Module 7, Section 3.3)

3.3.1 Wind Design Criteria (RESAR SP/90 Module 7, Section 3.3.1)

All safety-related structures exposed to wind forces will be designed to withstand the effects of the design wind. The specified design wind has a velocity of 110 mph with an annual probability of occurrence of 0.02 acting at 10 meters above the ground in open terrain. An importance factor of 1.11 is used in the design criteria to consider higher winds with a probability of occurrence of 0.01. This is in accordance with ANSI Standard A58.1-1982, "Minimum Design Loads in Buildings and Other Structures."

The procedures used to transform the wind velocity into pressure loadings on structures and the corresponding shape coefficients are in accordance with ANSI Standard A58.1. Shape coefficients for the reactor exterior building are calculated using American Society of Civil Engineers (ASCE) Paper No. 3269.

The staff concludes that the plant design criteria for wind loading are acceptable and meet the requirements of GDC 2. The staff bases its conclusion on the following:

The applicant has met the requirements of GDC 2 with regard to the capability of the structures to withstand design wind loadings by incorporating the following into the design:

- appropriate considerations for the most severe wind recorded for the site with an appropriate margin

- appropriate combinations of the effects of normal and accident conditions with the effect of natural phenomena

- the importance of the safety function to be performed

The applicant has met these requirements by using ANSI Standard A58.1 and ASCE Paper No. 3269, which the staff has reviewed and found acceptable, to transform the wind velocity into an effective pressure on structure and to select pressure coefficients corresponding to the structural geometry and physical configuration.

3.3.2 Tornado Design Criteria (RESAR SP/90 Module 7, Section 3.3.2)

The parameters chosen for the design-basis tornado are based on ANSI Standard 2.3-1983, "Standard for Estimating Tornado and Extreme Wind Characteristics at Nuclear Power Sites," which has not been completely accepted by the staff. All safety-related structures exposed to tornado forces are designed to resist a tornado with a maximum wind speed of 320 mph, a translational speed of 70 mph maximum and 5 mph minimum, at a radius of 540 ft. The atmospheric pressure drop is assumed to be 1.96 psi. These tornado design criteria are not in agreement with Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants," which was issued in April 1974. The applicant argued that the ANSI Standard 2.3-1983 is based on more updated tornado occurrence data and statistics and that it has selected the tornadic windspeed corresponding to a probability of 10E-7 per year at the worst location in the United States as the design basis for the RESAR SP/90.

However, in re-evaluating and recommending revision of Regulatory Guide 1.76, the staff used a considerable quantity of tornado data and proposed an interim position on design-basis tornado characteristics. The maximum wind speed in Region I, based on a 10E-7 probability of occurrence and estimated from the upper end of the 90 percent strike probability confidence level, is 330 mph. Although this value has been reduced from 360 mph as stated in Regulatory Guide 1.76, it is still higher than the 320 mph proposed by the applicant. The details of design-basis tornado characteristics for the Region I tornado zone are: rotational speed 260 mph, maximum translational speed 70 mph, radius of maximum rotational speed 150 feet, pressure drop 2.4 psi, and rate of pressure drop 1.7 psi/sec.

Westinghouse has agreed to utilize this interim position in lieu of the recommendations of ANSI Standard 2.3 1983. In its response to open items dated June 1989, Westinghouse stated:

> The maximum tornado wind speed of 330 mph will be utilized in establishing the velocities of the postulated tornado missiles. The plant will be designed for the Spectrum II missiles identified in Standard Review Plan 3.5.1.4, Rev. 2. The missile velocities corresponding to the design maximum w[illegible] speed of 330 mph will be taken as

the average of the velocities given for Region I (maximum wind speed = 360 mph) and Region II (maximum wind speed = 300 mph).

This response is acceptable for the PDA stage of review. The effects of tornadoes on the plant-specific structures will need to be reviewed for each specific application of a completed design using criteria that are applicable to that application.

3.4 Water Level (Flood) Design (RESAR SP/90 Module 7, Section 3.4)

3.4.1 Flood Protection (RESAR SP/90 Module 7, Section 3.4.1)

To ensure conformance with the requirements of GDC 2 and 10 CFR 100, Appendix A, with regard to protection against flooding, RESAR-SP/90 addresses the flood protection measures provided for the nuclear power block (NPB) portions of the structures, systems, and components whose failure as a result of flooding could prevent safe shutdown of the plant or result in the uncontrolled releases of significant radioactivity. These structures, systems and components are protected from the effects of the design-basis flood levels or flood conditions including wave and wind effects by the following methods:

(1) designed to withstand effects of the design-basis flood level or flood condition

(2) positioned to preclude effects of the design-basis flood level or flood condition

(3) housed within structures that satisfy method 1 or 2 above.

The criteria for the design-basis flood conform to the guidelines of Regulatory Guides 1.59 and 1.102 for incorporated barriers. Systems located below grade are protected by a combination of a waterproofing system for structures and the location of safety-related systems in watertight compartments. Below-grade penetrations are provided with waterproof seals.

Safety-related equipment within plant structures is protected against flooding from component failures by equipment location and drainage. Equipment location provisions include physical compartment separation, watertight compartments with watertight access doors, and sealed penetrations. Instrumentation includes alarms in the control room for "open" access doors and "high" water levels.

The staff concludes that the NPB portions of the design of structures, systems, and components for flood protection conform to the requirements of GDC 2 and 10 CFR 100, Appendix A with respect to protection against natural phenomena and conform to the guidelines of Regulatory Guides 1.59 and 1.102 concerning flood protection. However, the staff will review the balance-of-plant seismic Category I structures, systems and components during the plant-specific licensing process for the SP/90 design for flood protection against the guidance of SRP Section 3.4.1, "Flood Protection."

3.4.2 Water Level (Flood) Design Procedures

Site-dependent design procedures are not provided in RESAR SP/90. This includes procedures to determine the design flood level resulting from the most unfavorable condition or combination of conditions that produce the maximum water level at the site and to determine the loadings on safety-related structures induced by the design flood. However, the applicant has committed that all safety-related structures are designed to protect the safety-related systems, equipment, and components from the probable maximum flood and the highest ground-water level and to have the capability to withstand the effects of the flood or highest ground-water level in accordance with the requirements of GDC 2. Therefore, this issue will be reviewed on a plant-specific basis for each application using criteria that are applicable to that application.

3.5 Missile Protection (RESAR SP/90 Module 7, Section 3.5)

3.5.1 Missile Selection and Description (RESAR SP/90 Module 7, Section 3.5.1)

3.5.1.1 Internally Generated Missiles (Outside Containment) (RESAR SP/90 Module 7, Section 3.5.1.1)

In RESAR-SP/90, Westinghouse indicates that protection against postulated internally generated missiles outside containment such as missiles generated by rotating or pressurized equipment, described in GDC 4, is provided by any one or a combination of compartmentalization, barriers, separation, and equipment design. The primary means used in RESAR SP/90 design to provide protection to safety-related systems and components from damage resulting from internally generated missiles is to ensure design adequacy against generation of missiles, and, in those cases where missile formation can occur, to design plant features to contain their effects. Safety-related systems are physically separated from systems that are not safety related and redundant components of safety-related systems are physically separated so that the potential missiles could not damage both trains of a safety-related system.

RESAR SP/90 SAR provides an evaluation of potential missile sources from rotating equipment and pressurized components and the associated protection. This evaluation includes typical internal missile sources such as valve stems, valve bonnets, instrument wells, and pump impellers. On the basis of the design of these components, Westinghouse states that none of these are credible missiles and that the likelihood of bonnets in valves rated at 600 psig and below becoming missiles is remote because of the low probability of simultaneous failure of the bonnet-to-body bolts. Westinghouse further indicates that nuts, bolts, nut-and-bolt combinations, and nut-and-stud combinations are not considered potential missiles because they have only a small amount of stored energy. In addition, in its response (by a letter dated June 14, 1988) to the staff's RAI, Westinghouse indicated that the evaluation and position in RESAR-SP/90 with regard to missiles generated by the above cited parts are consistent with that provided in other applications (e.g., South Texas FSAR) that were approved by the staff.

The staff concludes that Westinghouse's design to maintain the capability for a safe plant shutdown and to prevent unacceptable radiological release in the event of internally generated missiles outside containment is in conformance with the requirements of GDC 4.

3.5.1.2 Internally Generated Missiles (Inside Containment) (RESAR SP/90 Module 7, Section 3.5.1.2)

Protection against postulated, internally generated missiles inside containment, such as missiles generated by rotating or pressurized equipment, described in the requirements of GDC 4, is provided by any one or a combination of barriers, separation, and equipment design. The primary means of providing protection to safety-related equipment from damage resulting from internally generated missiles is provided by shield walls and separation within the containment.

RESAR SP/90 SAR provides an evaluation of all potential sources of missiles inside containment. For systems located inside the containment, Westinghouse's approach is to ensure design adequacy against generation of missiles, rather than allow missile formation, and to contain their effects. The credible missiles postulated and the protection or the reasons for no concern follow:

- Top plug of the control rod drive mechanism (CRDM) - CRDM missiles and any secondary missiles generated by their impact can be contained by the integrated head missile shield.

- Valves located on the top of the pressurizer postulated to generate vertical missiles and any secondary missiles generated by their impact - Protection is provided by the concrete roof slab, which prevents damage to the containment, engineered safeguards components, and components outside the pressurizer compartment.

- Temperature and pressure sensor assemblies connected to the reactor coolant system - The missile characteristics of these assemblies are not of concern from a containment penetration standpoint.

- Pressurizer heaters - Inasmuch as they would be ejected in a downward direction, no damage to safety-related structures, systems, and components inside the containment would occur.

In a response (by a letter dated June 14, 1988) to the staff's RAI, Westinghouse indicated that the evaluation and position in RESAR SP/90 SAR with regard to missiles generated by the above cited parts and/or components are consistent with that provided in other applications (e.g., South Texas FSAR) that were approved by the staff.

The staff concludes that through the use of barriers, separation, and equipment design, the design is in full conformance with the requirements of GDC 4.

3.5.1.3 Turbine Missiles (RESAR SP/90 Module 7, Section 3.5.1.3)

GDC 4 requires that nuclear power plant structures, systems, and components important to safety shall be protected against dynamic effects, including the effects of turbine missiles.

Since the turbine-generator is not part of the nuclear power block for RESAR SP/90 design, the applicant did not evaluate the effect of turbine missiles on the facility's structures, systems, and components important to safety. The plant-specific applicant will be required to address this issue during the plant-specific licensing process referencing the RESAR SP/90 design.

3.5.1.4 Missiles Generated by Natural Phenomena (RESAR SP/90 Module 7, Section 3.5.1.4)

GDC 2 requires that structures, systems, and components essential to safety be designed to withstand the effects of natural phenomena. GDC 4 requires that these same plant features be protected against missiles. The missiles generated by natural phenomena of concern are those resulting from tornadoes. Section 3.3.2 of this SER addresses tornado design criteria and the design-basis tornado.

Since the identification of the design-basis tornado and the selection of the spectrum of missiles are site-specific, the staff will review the design of the facility for protection against hazards resulting from missiles generated by the design-basis tornado in accordance with SRP Section 3.1.5.4, "Missiles Generated by Natural Phenomena," during the plant-specific licensing process referencing the RESAR SP/90 design.

3.5.2 Structures, Systems, and Components (SSC) To Be Protected from Externally Generated Missiles (RESAR SP/90 Module 7, Section 3.5.2)

Safety-related structures, systems, and components were reviewed with respect to their capability to perform functions required for attaining and maintaining a safe shutdown condition during normal or accident conditions, mitigating the consequence of an accident, or preventing the occurrence of an accident assuming impact from externally generated missiles. With the exception of the ultimate heat sink and its associated facility, systems and components, Westinghouse has identified all SSC, including SSC associated with the spent fuel storage facility, to be protected from externally generated missiles. This identification is in accordance with GDC 4, the guidelines of Regulatory Guide 1.117, "Tornado Design Classification," and Regulatory Guide 1.13, "Spent Fuel Storage Facility," and is, therefore, acceptable.

Westinghouse states that the evaluation of the integrity of the facility ultimate heat sink against externally generated missiles is not part of the SP/90 nuclear power block.

The staff concludes that Westinghouse's list of safety-related SSCs to be protected from externally generated missiles and the provisions in the plant design providing this protection are in accordance with the requirements of GDC 4 with respect to missile effects and guidelines of Regulatory Guides 1.13 and 1.117 concerning protection of spent fuel and other safety-related plant features (excluding the ultimate heat sink) from tornado missiles, and is, therefore, acceptable. The design of the missile protection features for the ultimate heat sink and its associated systems and components is the responsibility of the plant-specific applicant and will be reviewed by the staff during the plant-specific licensing process referencing the RESAR SP/90 design.

3.5.3 Barrier Design Procedures (RESAR SP/90 Module 7, Section 3.5.3)

Missile barriers and protection structures are designed to withstand and absorb missile impact loads to prevent damage to safety-related components. Information has been provided indicating that procedures are to be used in the design of structures, shields, and barriers to resist the effect of missiles. Structural members designed to resist missile impact will be designed for flexural, shear, and buckling effects using the equivalent static load obtained from the evaluation of structural response. Stress and strain limits for the equivalent static load will comply with the requirements of applicable codes or specifications as defined in the Standard Review Plan and applicable regulatory guides except for the area local to the missile impact. In that area, the stress and strain may exceed the allowables provided there will be no loss of function of any safety-related system. The National Defense Research Committee's (NDRC's) modified formula is used for missile penetration calculation in the design of concrete barriers. The Stanford formula for missile penetration is used for design of steel barriers. These formulas have been accepted by the staff (see SRP Section 3.5.3) and their use is acceptable.

In RESAR SP/90 Module 7, the applicant did not address the minimum acceptable concrete barrier thickness requirements to meet local damage prediction pertaining to tornado-generated missiles as stated in SRP Section 3.5.3. The permissible ductility ratio for reinforced concrete and steel structural elements subjected to impactive and impulsive loads pertaining to overall damage prediction was not discussed. Since Westinghouse-supplied equipment generally is not designed to withstand the impact of postulated missiles, the plant-specific designer will have to consider the effect of postulated missiles and provide design criteria for safety-related structures, shields, and barriers. Consequently, the acceptability of the barrier design for the total plant will be reviewed on a plant-specific case-by-case basis.

In the "Responses to NRC DSER Open Issues 1-41," dated September 1989, Westinghouse committed to comply with the NRC position on local and overall damage predictions as stated in SRP Section 3.5.3, paragraph II.1 and 2. Minimum acceptable concrete barrier thickness requirements to meet local damage predictions pertaining to tornado-generated missiles were adequately addressed and permissible ductility ratio for reinforced concrete and steel structural elements was discussed. Westinghouse also committed to comply with the additional NRC positions on ductility of concrete structures contained in Regulatory Guide 1.142, "Safety-Related Structures for Nuclear Power Plants." Steel structures are designed to specifications of the American Institute of Steel Construction (AISC). Maximum allowable ductility ratios for steel structures have been added. The staff considered these commitments and discussions are consistent with NRC staff positions and are therefore acceptable.

3.5.4 Missile Protection Interface Requirements (RESAR SP/90 Module 7, Section 3.5.4)

The balance-of-plant (BOP) designer considers the effects of the postulated missiles identified by the nuclear power block applicant, as well as those to be identified by the BOP applicant, and provides the necessary protection to safety-related components as determined by the missile selection bases provided in the RESAR SP/90 SAR.

The staff will review the BOP design of the facility for protection from internally generated missiles in accordance with SRP Section 3.5.1.1, "Internally Generated Missiles (Outside Containment)," during the plant-specific licensing process referencing the RESAR SP/90 design.

3.6 Protection Against Dynamic Effects Associated With the Postulated Rupture of Piping (RESAR SP/90 Module 7, Section 3.6)

Westinghouse states that in the event of the high- or moderate-energy pipe failure within the plant, adequate protection is provided to ensure that essential structures, systems, or components are not affected by the effects of postulated piping failure.

3.6.1 Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Inside and Outside Containment (RESAR SP/90 Module 7, Section 3.6.1)

GDC 4 requires that structures, systems, and components important to safety be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures and from events and conditions outside the containment. The plant design for protection against postulated piping failures in fluid systems inside containment is addressed in Section 6.2.1 of this SER.

With respect to plant design for protection against postulated piping failures in fluid systems outside containment, Westinghouse has briefly provided the design bases and committed to provide more detailed information during the final design approval (FDA) stage of review. The staff will review the information in accordance with SRP Section 3.6.1 and its associated Branch Technical Position (BTP) ASB 3-1.

3.6.2 Determination of Rupture Locations and Dynamic Effects Associated With the Postulated Rupture of Piping (RESAR SP/90 Module 7, Section 3.6.2)

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

The staff's review, in accordance with SRP Section 3.6.2, pertains to the methodology used for protecting safety-related structures, systems, and components against the effects of postulated pipe breaks both inside and outside containment. The staff used the review procedures identified in SRP Section 3.6.2 to evaluate the effect that breaks in high-energy fluid systems would have on adjacent safety-related structures, systems, or components with respect to jet impingement and pipe whip.

Pipe whip need be considered only in those high-energy piping systems having fluid reservoirs with sufficient capacity to develop a jet stream. The criteria for determining high- and moderate-energy lines are in BTP ASB 3-1 in SRP Section 3.6.1. RESAR SP/90 Module 7, Table 3.6-1, contains a list of all high-energy systems.

In response to staff Question 210.45, the applicant revised Section 3.6.2.2.1, "Forcing Functions for Jet Thrust," of Module 7 to the RESAR SP/90 application.

In this revision, the ANSI/ANS 58.2-1980 Standard, "Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture," was referenced as the basis for analytical methods used to calculate jet thrust loads that would result from a postulated pipe break. The staff has concluded that this proposed methodology is consistent with applicable guidelines in SRP Section 3.6.2 and is acceptable. The analytical methods proposed in Section 3.6.2.3.1 of Module 7 for determining jet impingement effects resulting from postulated breaks also is consistent with SRP Section 3.6.2 and is acceptable.

In response to staff Question 210.39 and in their response to open items dated June 1989, Westinghouse revised parts of Section 3.6.2 of RESAR SP/90 Module 7 and provided additional clarification and commitment to include postulated pipe break criteria that is consistent with SRP Section 3.6.2. The following are staff positions relative to these responses:

(1) RESAR SP/90 Section 3.6.2.1.1.A.2.b(1) contains unacceptable criteria for postulating breaks in ASME Class 1 piping. To be acceptable, breaks should be postulated where the maximum stress range as calculated by ASME Section III, Subsection NB, Subarticle NB-3653, Equation 10, alone exceeds 2.4 S_m, where S_m is the design stress intensity as defined in ASME Section III.

In its response, Westinghouse requested that the staff evaluate the effect of the 1987 revision (Rev. 1) to BTP MEB 3-1 for pipe of less than 6 inches in diameter. At issue is the change that deleted ASME III, Subsection NB-3653, equations 12 and 13 from the threshold stress criteria for postulating breaks. This left only equation 10 as the guideline; that is, if equation 10 is greater than 2.4 S_m one has to postulate a break. The previous version required either equation 12 or 13 to be greater than 2.4 S_m in addition to equation 10 being greater than 2.4 S_m before a break had to be postulated. The change to BTP MEB 3-1 was based on:

- lower stresses in equation 10 after the ΔT term was deleted

- arbitrary intermediate break deletion reduced the number of postulated breaks

This position was discussed in the Federal Register Volume 52, No. 118, page 23377 on June 19, 1987.

(2) In RESAR Sections 3.6.2.1.1.C.1.b and 3.6.2.1.2.3.A, alternate criteria to that of SRP Section 3.6.2 is proposed for postulating breaks and cracks, respectively, in "seismically analyzed" piping that is not designated as ASME class category, but that is designed in accordance with ANSI/ASME Standard B31.1-1986, "Power Piping." The staff has concluded that the proposed criteria is as conservative as SRP Section 3.6.2 guidelines and is acceptable. The design basis for non-seismic piping in the vicinity of seismic piping is discussed in RESAR SP/90 Module 7, Section 3.7.3.13. The non-seismic piping is analyzed for the SSE condition. This is also acceptable.

(3) BTP MEB 3-1, Section B.1.c(4), in SRP Section 3.6.2 states that structures that separate high-energy lines from essential components should be designed to withstand the most severe break in the line even if other SRP Section 3.6.2 guidelines do not require a break in that line to be postulated.

In their response to open items dated June 1989 Westinghouse stated: "a structure that separates a high-energy line outside containment, which has not been demonstrated to meet mechanistic pipe break requirements, from an essential component is designed to withstand the consequences of the pipe break in the high-energy line which produces the greatest effect at the structure, irrespective of the fact that the criteria of Sections 3.6.2.1 B and C might not require such a break to be postulated." This response is not completely acceptable because the commitment should

- not be limited to outside containment
- reference RESAR SP/90 Module 7, Section 3.6.2.1.1.A, in addition to Sections 3.6.2.1.1 B and C
- not be included in RESAR SP/90 Module 7, Section 3.6.2.1.1.E, which is leak-before-break (LBB) and overrides Sections 3.6.2.1.1.A, B, and C criteria if LBB is accepted

(4) Branch Technical Position MEB 3-1, Section B.1.b(6)(C), in SRP Section 3.6.2 states that guard pipe assemblies should be subjected to a single pressure test at a pressure not less than their design pressure, where design pressure is defined as the maximum operating pressure of the enclosed pipe.

In its response to open items dated June 1989, Westinghouse stated: "Guard pipe assemblies are designed in accordance with ANSI/ANS 58.2-1988 except that the pressure test is performed at a pressure not less than their design pressure. The design pressure is the maximum operating pressure of the enclosed process pipe." This response is acceptable.

SRP Section 3.6.2 sets forth certain criteria for the analysis and subsequent augmented inservice inspection requirements for high-energy piping within the containment penetration area where breaks are not postulated. Breaks need not be postulated in those portions of piping within the containment penetration region that meet the requirements of ASME Section III, Subsection NE, Subarticle NE-1120, and the additional requirements outlined in BTP MEB 3-1 to SRP Section 3.6.2. Augmented inservice inspection is required for those portions of piping within the break exclusion region. The information in RESAR Section 3.6.2.1.1.D.3, "High Energy Piping in Containment Penetration Areas," Module 7, Amendment 2, provides sufficient clarification of the Westinghouse position to demonstrate that its criteria is just as conservative as SRP Section 3.6.2. Westinghouse does not take advantage of that part of SRP Section 3.6.2 and BTP MEB 3-1, Section B.1.b(1)(c), which allows stresses higher than 2.25 S_m as long as a plastic hinge is not formed. The five-way restraints near the turbine building wall ensure main steam isolation valve (MSIV) operability by resisting torsional and bending moments produced by a postulated pipe break either upstream or downstream of the containment isolation boundary. This is accomplished by locating these restraints near the valves.

In its response to open items dated June 1989, Westinghouse provided the following acceptable clarification related to the design of pipe rupture restraints:

Where supports are attached to restraint structures the restraint is designed to the criteria described in Section 3.9.3.4 of RESAR

SP/90 PDA Module 7, "Structural/Equipment Design," for ASME Code, Section III supports. The design of pipe whip restraint structures includes a dynamic load factor. This factor accounts for the inertia load in the restraint structure and its value depends on the dynamic characteristics of the structural system.

The staff considers leak-before-break evaluations to be plant specific because parameters such as potential piping degradation mechanisms, piping geometry, materials, fabrication procedures, loads, and leakage detection systems are plant specific. Thus, although leak before break can be considered, where justified, the staff will review leak-before-break analyses on a plant-specific basis using the current evaluation methodology.

Pending satisfactory resolution of the issues discussed above, the staff concludes that the criteria for postulating pipe rupture locations and the methodology for evaluating the subsequent dynamic effects of these ruptures are in accordance with SRP Section 3.6.2, meet GDC 4, and are acceptable. Thus, the provision for protection against dynamic effects associated with pipe ruptures of the reactor coolant pressure boundary inside containment and the resulting discharging fluid will provide adequate assurance that design-basis loss-of-coolant accidents will not be aggravated by the sequential failures of safety-related piping and emergency core cooling system performance will not be degraded by these dynamic effects. In addition, the proposed piping and restraint arrangement and applicable design considerations for high- and moderate-energy fluid systems inside and outside containment, including the reactor coolant pressure boundary, will provide adequate assurance that the structures, systems, and components important to safety that are in close proximity to the postulated pipe ruptures will be protected. The design is of a nature to mitigate the consequences of pipe ruptures so that the reactor can be safely shut down and maintained in a safe shutdown condition in the event of a postulated rupture of a high- or moderate-energy piping system inside or outside containment.

3.7 Seismic Design (RESAR SP/90 Module 7, Section 3.7)

3.7.1 Seismic Input (RESAR SP/90 Module 7, Section 3.7.1)

Westinghouse stated that the establishment of zero period accelerations (ZPA) for an operating-basis earthquake (OBE) and a safe-shutdown earthquake (SSE) in Subsection 2A.5.2 of Appendix 2A to Module 3 of RESAR SP/90 is beyond the review scope of Module 7. A ZPA of 0.1 g for an OBE and 0.3 g for an SSE as free field input motions are established as the baseline seismic condition for the nuclear power plant design application. It should be noted that the applicant's proposed ratio of the OBE to SSE is not currently acceptable according to 10 CFR 100, Appendix A. However, by letter dated November 22, 1988, the staff stated that the OBE should not control the design of safety systems. The staff will consider this issue again at the FDA stage of review.

The free field design response spectra shown in Module 7 of RESAR SP/90 for the horizontal and the vertical components of SSE and OBE are in conformance with Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," and are therefore acceptable only with respect to the spectral shape and not with respect to the ZPA values.

Synthesized time histories of 10-second total duration are generated for each of the thr(components, two horizontal and one vertical, of the SSE seismic design response spectra. By letter dated September 22, 1987, the staff requested detailed descriptions of these time histories including time durations of rise, strong motion, and decay stages; justification for using a total duration of 10 seconds; and the power spectral density of the design time histories. This information was incorporate in Amendment 2 (January 1989) to RESAR SP/90 Module 7, "Strucutural/Equipment Design." The power spectral density (PSD) for the time histories was not included, but Westinghouse has committed to develop the PSD early in the next design phase and will submit it as part of the FDA application. The staff finds this acceptable for the PDA stage of review.

The specific percentage of critical damping values used in typical safety-related structures and components are in conformance with Regulatory Guide 1.61, "Damping Valves for Seismic Analysis for Nuclear Power Plants." Acceptability of damping values for some equipment and systems, such as the primary coolant loop systems, fuel assemblies, and control rod drive mechanism, will not be addressed here. Westinghouse has provided sufficient information on methods of calculating magnitudes for soil springs and dampers. The magnitudes of these springs and dampers were based on the standard equations enumerated in Tables 3300-1, 3300-2 and Figure 3300-3 of ASCE Standard 4-86. This is acceptable.

3.7.2 Seismic System and Subsystem Analyses (RESAR SP/90 Module 7, Sections 3.7.2, 3.7.3)

The scope of review of the seismic system and subsystem analyses for the plant includes the seismic analysis methods for all safety-related structures, systems, and components. It includes an evaluation of the procedures for modeling seismic soil-structure interaction, development of floor response spectra, inclusion of torsional effects, determination of the potential overturning of safety-related structures, and determination of composite damping.

The seismic system analyses of the building structure of the NPB are performed by the time-history method using a direct integration approach. The floor response spectra thus generated are indicative of the frequency content of the soil-structure system. Seismic systems are modeled by appropriately accounting for the effects of soil-structure interaction to simulate the overall behavior of the seismic systems. The stick model represents the mass and stiffness properties of the reactor external building, the concrete shield building, interior concrete shield, and the steel spherical containment fastened on the common basemat. The discrete masses are lumped at the nodes located at the floor levels and the locations of major discontinuity of the building systems.

In the soil-structure interaction analysis, the building stick model is coupled with the discrete soil dynamic properties through a common nodal point of the mass center of the basemat. The soil mass is modeled by both the half-space impedance method and the finite boundary method. The equivalent stiffness and damping coefficients of a soil medium are characterized by three types of soil at a shear wave velocity of 1000 ft/sec, 2500 ft/sec, and infinite to envelop the potential site-specific variability of soils. Amplification resulting from local site conditions shall be consistent with the staff acceptance for Module 3.

The responses of structures and systems as well as floor response spectra are obtained by means of the generic time-history analysis. The frequency intervals used for computing the floor response spectra are consistent with those of

Regulatory Guide 1.122, "Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components." Torsional effects and stability against overturning are considered.

Both the time-history solution and the response spectrum analysis technique are used for analyzing the subsystems of the NPB. The generic floor response spectra serve as design input for the subsystems with consideration of the three components of earthquake motion and the combination of modal responses that are consistent with the recommendations of Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis."

The staff finds the seismic system and subsystem analysis as described in the RESAR SP/90 SAR to be acceptable.

3.7.3 Seismic Instrumentation (RESAR SP/90 Module 7, Section 3.7.4)

The applicant is required to meet the requirements of 10 CFR 100, Appendix A, by providing the instrumentation that is capable of measuring the effects of an earthquake that meets the requirements of GDC 2. The installation of the specified seismic instrumentation in the reactor containment structure and at other safety-related structures, systems, and components constitutes an acceptable program to record data on seismic ground motion as well as data on the frequency and amplitude relationship of the seismic response of major structures and systems.

A prompt readout of pertinent data at the control room can be expected to yield sufficient information to guide the operator on a timely basis for the purpose of evaluating the seismic response in the event of an earthquake. Data obtained from such installed seismic instrumentation will be sufficient to determine that the seismic analysis assumptions and the analytical model used for the design of the plant are adequate and that allowable stresses are not exceeded under conditions where continuity of operation is intended. The applicant's commitment to provide such seismic instrumentation in its design complies with Regulatory Guide 1.12, "Instrumentation for Earthquakes."

The applicant has not provided the inservice inspection program that will verify operability by performing channel checks, calibrations, and functional tests at acceptable intervals required by 10 CFR 50.55a. The applicant has commited to develop a proposed inservice inspection program. The staff will review the proposed program during the FDA stage of review.

3.8 <u>Design of Seismic Category I Structures</u> (RESAR SP/90 Module 7, Section 3.8)

3.8.1 Steel Containment (RESAR SP/90 Module 7, Section 3.8.2)

The containment vessel is designed as a free-standing, spherical, welded steel shell, 60-meter inside diameter, and 42 mm thick. The lower portion of the shell, below an elevation of 92.2 meters, is encased between the building foundation concrete and the interior structure base concrete without any structural connection between the steel and concrete.

The steel containment is designed, fabricated, and tested in accordance with the provisions of ASME Code, Section III, Subsection NE. All structural steel nonpressure parts, such as ladders and walkways, are designed in accordance with the AISC, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings." While these codes have been accepted generally for steel containments, they may not be adequate or applicable to the design of stiffened or unstiffened spherical shells. In response to this concern, Westinghouse addressed (Amendment 2 to RESAR SP/90 Module 7) the use of ASME Code Case N-284, which provides in great detail the stability (buckling) criteria for determining the structural adequacy against buckling of containment shells with complex shell geometries and loading conditions. The staff considers this to be adequate and applicable to the design of the spherical containment vessel.

10 CFR 50, Appendix B, requires that design control measures be established to provide for verifying or checking the adequacy of design, such as by design reviews, alternate calculational methods, or a suitable testing program. The staff cannot accept a standard design of containment without performing a design audit or reviewing a structural integrity test. The staff will perform a detailed design audit when the vessel designer has been selected and the design calculations have been prepared during the FDA stage of review.

3.8.2 Concrete and Steel Internal Structures (RESAR SP/90 Module 7, Section 3.8.3)

The internal structures are those concrete or steel structures inside of the containment pressure boundary that support the reactor coolant system components and related piping systems and equipment. The concrete structures also will provide radiation shielding. The design of the internal structures consists of the primary shield wall, various compartments walls, refueling canal walls, operating floor, and intermediate slabs and platforms.

The applicant's design, fabrication, and testing of the internal structures will be in accordance with AISC, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings;" American Concrete Institute, ACI-349, "Code Requirements for Nuclear Safety-Related Structures;" and ASME Section III, Division 1, Subsection NF. These codes and standards have been accepted by the staff (SRP Section 3.8.3).

The criteria used in the design analysis and construction of the containment internal structures to account for anticipated loadings and postulated conditions that may be imposed upon the structures during their service lifetime are in conformance with established criteria and with codes, standards, and specifications acceptable to the regulatory staff. The use of these criteria provides reasonable assurance that, in the event of earthquakes and various postulated accidents occurring within the containment, the interior structures will withstand the specific design conditions without impairment of structural integrity or the performance of required safety function.

3.8.3 Other Seismic Category I Structures (RESAR SP/90 Module 7, Section 3.8.4)

The major seismic Category I structure covered by this section is the reactor external building. This includes the shield building, auxiliary equipment area, the fuel handling are.. the control complex area, the diesel generator building, the main steam tunnel, and the essential safety facility area.

The applicant's design, fabrication, and erection of other seismic Category I structures will be in accordance with AISC specification and ACI-349, which have been accepted by the staff (SRP Section 3.8.4).

The criteria used in the analysis, design, and construction of all the plant Category I structures to account for anticipated loadings and postulated conditions that may be imposed upon each structure during its service lifetime are in conformance with established criteria, codes, standards, and specifications acceptable to the regulatory staff. The use of these criteria provides reasonable assurance that, in the event of winds, tornadoes, earthquakes, and various postulated accidents occurring within the structures, the structures will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions. Other Category I structures for the plant-specific structures and their effect on the proposed standard design will be reviewed on a case-by-case basis during the plant-specific licensing process referencing the RESAR SP/90 design.

3.8.4 Foundations (RESAR SP/90 Module 7, Section 3.8.5)

The reactor external building foundation design is a reinforced concrete mat supported directly on firm soil or sound rock. It is separated from the foundation mats of other adjacent structures to eliminate any structural interaction.

Reinforced concrete foundations are designed as seismic Category I structures. The criteria used in the analysis, design, and construction of all the plant seismic Category I foundations to account for anticipated loadings and postulated conditions that may be imposed upon each foundation during its service lifetime are in conformance with established criteria, codes, standards, and specifications acceptable to the regulatory staff. The use of these criteria provides reasonable assurance that, in the event of winds, tornadoes, earthquakes, and various postulated events, seismic Category I foundations will withstand the specified design conditions without impairment of structural integrity and stability or the performance of required safety functions.

3.9 Mechanical Systems and Components (RESAR SP/90 Module 7, Section 3.9)

The staff reviewed the structural integrity and functional capability of various safety-related mechanical components in the plant design in accordance with SRP Sections 3.9.1 through 3.9.6. This review was not limited to ASME Code components and supports, but was extended to other components such as control rod drive mechanisms, certain reactor internals, and any safety-related piping designed to industry standards other than the ASME Code. In addition, load combinations, allowable stresses, methods of analysis, summary of results, and preoperational testing were reviewed. A mechanical component must be demonstrated to perform its safety-related function under all postulated combinations of normal operating conditions, system operating transients, postulated breaks, and seismic events.

3.9.1 Special Topics for Mechanical Components (RESAR SP/90 Module 7, Section 3.9.1)

The staff reviewed the information in Section 3.9.1 of Module 7 to the RESAR SP/90 application with regard to design transients and methods of analysis used for all seismic Category I components, component supports, core support structures, and reactor internals designated as Classes 1 and CS under ASME Code Section III as well as those not covered by the Code. The assumptions and procedures used for the inclusion of transients in the fatigue evaluation of ASME Code Classes 1 and CS were reviewed. The staff also reviewed the computer programs used in the design and analysis of seismic Category I components and their supports and experimental and inelastic analytical techniques.

In Section 3.9, Table 3.9-1, of Module 7, the applicant provided the design transients for five plant operating conditions and the number of cycles for each of the design transients that will be used in the fatigue evaluation of reactor coolant system components. The ASME operating conditions and their definitions follow.

- ASME Service Level A - normal conditions
- ASME Service Level B - upset conditions, incidents of moderate frequency
- ASME Service Level C - emergency conditions, infrequent incidents
- ASME Service Level D - faulted conditions, low-probability postulated events
- testing conditions

The applicant used computer codes to analyze mechanical components. A list showing all computer programs used by the applicant for static and dynamic analyses to determine the structural integrity and functional integrity of seismic Category I Code and non-Code items and the analyses to determine stresses, along with a description of the program is included in the RESAR SP/90 application. Design control measures to verify the adequacy of the design of safety-related components are required by 10 CFR 50, Appendix B.

The staff has only partially reviewed the WECAN program, which is listed in RESAR SP/90 Section 3.9.1.2.1. The information in WCAP-8929, "Benchmark Problem Solutions Employed for Verification of the WECAN Computer Program," that is applicable to the dynamic analysis of linear and nonlinear elastic beam-type structures is the only portion of WECAN that has been accepted by the staff. However, WECAN contains many other features and extensive capabilities such as plate and shell structures, elastic-plastic and creep deformation, and heat transfer analyses. None of these features have been independently verified, although their theoretical bases are consistent with present state-of-the-art. Staff Question 210.56 requested the applicant to identify and discuss any of these additional WECAN features that will be used in the design of the RESAR SP/90. The applicant's response to Question 210.56 referenced its response to Question 220.7 in which the applicant provided (in RESAR SP/90) Addendum 2, dated January 7, 1988) detailed descriptions, including verification procedures for the WECAN, DEBLIN2, and ASHD computer programs. Additionally, Westinghouse provided detailed information for the FATCON and WESAN in its response to open items dated June 1989 but failed to include verification procedures in the submittal.

Pending satisfactory resolution of the open item discussed above, the staff concludes that the design transients and resulting load combinations with appropriate specific design and service limits for mechanical components and supports are acceptable and meet the applicable portions of GDC 1, 2, 14, and 15; 10 CFR 50, Appendix B; and 10 CFR 50, Appendix A.

The applicant has met GDC 14 and 15 by demonstrating that the design transients and resulting loads and load combinations with appropriate specific design and service limits that the applicant has used for designing ASME Code Class 1 and CS components and supports and reactor internals provide a complete basis for design of the reactor coolant pressure boundary for all conditions and events expected over the service lifetime of the plant.

The applicant has met 10 CFR 50, Appendix B, and GDC 1 by submitting information that demonstrates the applicability and validity of the design methods and computer programs used for the design and analysis of seismic Category I ASME Code Classes 1, 2, 3, and CS structures and non-Code structures within the present state-of-the-art limits and by design control measures that are acceptable to ensure the quality of the computer programs.

3.9.2 Dynamic Testing and Analysis of Systems, Components, and Equipment (RESAR SP/90 Module 7, Section 3.9.2)

The staff has reviewed the methodology, testing procedures, and dynamic analyses employed by the applicant to ensure the structural integrity and functionality of piping systems, mechanical equipment, and their supports under vibratory loadings.

The staff's review included (1) the piping vibration, thermal expansion, and dynamic effect testing; (2) the seismic system analysis methods; (3) the dynamic responses of structural components within the reactor caused by steady-state and operational flow transient conditions for prototype reactors; (4) flow-induced vibration testing of reactor internals to be conducted during the preoperational and startup test program; and (5) the dynamic analysis methods used to confirm the structural design adequacy and functional capability of the reactor internals and piping attached to the reactor vessel when subjected to loads from a loss-of-coolant accident (LOCA) in combination with a safe shutdown earthquake.

3.9.2.1 Piping Preoperational Vibration and Dynamic Effects Testing (RESAR SP/90 Module 7, Section 3.9.2.1)

Piping vibration, thermal expansion, and dynamic effects testing will be conducted during a preoperational testing program. The purpose of these tests is to ensure that the piping vibrations are within acceptable limits and that the piping system can expand thermally in a manner consistent with the design intent. During the plant's preoperational and startup testing program, the applicant will test various piping systems for abnormal, steady-state, or transient vibration and for restraint of thermal growth. Systems to be monitored will include (1) ASME Code Class 1, 2, and 3 piping systems; (2) high-energy piping systems inside seismic Category I structures; (3) high-energy portions of systems whose failure could reduce the functioning of seismic Category I plant features to an unacceptable safety level; and (4) seismic Category I portions of moderate-energy piping systems located outside containment. Steady-state vibration, whether flow induced or caused by nearby vibrating machinery, could cause 10^8 or 10^9 cycles of stress in the pipe during the 40-year life of the plant. For this reason, the staff requires that the stresses associated with steady-state vibration be minimized and limited to acceptable levels. The test program will consist of a mixture of instrumented measurements and visual observations by qualified personnel.

The information in Section 3.9.2.1 of Module 7 to the RESAR SP/90 application provides a general discussion of the proposed piping preoperational test program for the RESAR SP/90. The staff's current position on this issue is that for PDA, a commitment is required to develop a test program to support the FDA application, which will use testing procedures and acceptance criteria in the draft ANSI/ASME OM3-1987, "Requirements for Preoperational and Initial Start-Up Vibration Testing of Nuclear Power Plant Systems." The staff is participating in the development of this standard. For the past several years, the staff has been using the acceptance criteria from the OM3 draft dated May 1985 as a guide in its review of piping preoperational test programs for near-term operating license plants.

In Section 3.9.2.1 of Module 7 the applicant has provided a commitment to implement a preoperational test program based on draft ANSI/ASME OM3-1987. The staff has reviewed the draft OM3-1987 standard criteria and finds that these will provide an acceptable level of safety for piping vibration during the plant's 40-year life.

The staff concludes that the applicant will meet GDC 14 and 15 with respect to the design and testing of the reactor coolant pressure boundary. This provides reasonable assurance that rapidly propagating failure and gross rupture will not occur as a result of vibratory loadings. In addition, an acceptable vibration, thermal expansion, and dynamic effects test program that will be conducted during startup and initial operation of specified high- and moderate-energy piping, including all associated restraints and supports, ensures that design conditions will not be exceeded during normal operation including anticipated operational occurrences. The tests provide adequate assurance that the piping and piping supports will be designed to withstand vibrational dynamic effects as a result of valve closures, pump trips, and other operating modes associated with the design-basis flow conditions. In addition, the tests provide assurance that adequate clearances and free movement of snubbers will exist for unrestrained thermal movement of piping and supports during normal system heatup and cooldown operations. The planned tests will develop loads similar to those experienced during transient and normal reactor operations.

3.9.2.2 Seismic Subsystem Analysis (RESAR SP/90 Module 7, Section 3.7.3)

The staff used the guidelines of SRP Section 3.9.2 to perform its review of seismic subsystem analysis contained in Section 3.7.3 of Module 7 to the RESAR SP/90 application. Areas reviewed were seismic analyses methods, determination of the number of earthquake cycles, basis for selection of frequencies, the combination of modal responses and spatial components of an earthquake, criteria used for damping, torsional effects of eccentric masses, interaction of other piping with seismic Category I piping, and Category I buried piping systems.

The scope of the staff's review included the seismic analysis methods for all seismic Category I piping systems and components. The staff reviewed the manner in which the dynamic system analysis is performed, the method of selection of significant modes, whether the number of masses or degrees of freedom is adequate, and how consideration is given to maximum relative displacements. The review included design methodologies and procedures used for the evaluation of the interaction of non-seismic Category I piping with seismic Category I piping and the seismic methods that consider the effect of movement at support points and penetrations. In Section 3.7.3.12 of Module 7, the applicant stated that there are no buried seismic category piping systems in the RESAR SP/90 design.

In addition, the staff reviewed seismic analysis procedures for reactor internals. The system and subsystem analyses are performed by the applicant on an elastic basis. Modal response spectrum, multidegree of freedom, and time-history methods form the basis for the analyses of all major seismic Category I systems and components.

RESAR Section 3.7.3.1 of Module 7 references the American Society of Civil Engineers (ASCE) seismic analysis standard, "Seismic Analysis of Safety-Related Nuclear Structures," May 1984 draft, for methodology used in both time-history solutions and response spectrum analysis of subsystems. In Question 210.47, the staff requested that the applicant provide justification for any differences between the ASCE 1984 standard draft methodology and applicable parts of SRP Sections 3.7.2 and 3.9.2. The applicant stated that the September 1986 Edition of the ASCE standard contains requirements that are just as conservative and in some cases more conservative than SRP Sections 3.7.2.II.1.a and 3.9.2.II.2.a(1). The staff has not been provided with sufficient information to accept the applicant's response. To be acceptable, the staff will require more detailed information. In addition, the response to Question 210.47 does not appear to be consistent with the information in RESAR Module 7, Section 3.7.3.6, "Three Components of Earthquake Motion," which the staff finds acceptable.

As stated in Regulatory Guide 1.92, t current staff position allows three options in combining closely spaced modes. In the first paragraph of RESAR Section 3.7.3.7 of Module 7, the applicant has proposed a fourth optional method of algebraic combination of modes with closely spaced frequencies, which is not one of the methods stated in Regulatory Guide 1.92. In response to staff Question 210.50, the applicant provided justification for the use of this optional method. The justification is based on Roseblueth and Elorduy's Double Sum Method (Reference 6 in the applicant's response to Question 201.50) for a combination of mode-by-mode responses from lower frequency (<33 Hz) modes, and algebraic sum for combination of mode-by-mode responses for higher frequency (>33 Hz) modes. However, the Rosenblueth and Elorduy's Double Sum Method also retains algebraic signs that the staff does not find acceptable. In addition, there are some differences between the proposed method and the current staff position as stated in Regulatory Guide 1.92. The staff does not find the justification for these differences to be acceptable.

The proposed use of algebraic sum for combination of mode-by-mode responses for higher frequencies (≥33 Hz) is also not acceptable because the sense of sign gets lost with the response spectrum method. Either an absolute sum or square-root-of-the-sum-of-the-squares (SRSS) combination should be used at these frequencies. In its response to open items dated June 1989, Westinghouse provided the following clarification: "The total co-directional seismic response is obtained by combining the individual modal responses utilizing the SRSS method. For systems having modes with closely spaced frequencies, this method is modified to include the possible effect of these modes." Additionally, equation 2 of Section 3.7.3.7 was revised to define R_i as the absolute value of mode $_i$ and it was further stated that the methodology for combination of modes with closely spaced frequencies the absolute valve of the modal responses is used.

This response does not provide information that is consistent with Regulatory Guide 1.92. The modified equation (2) in Section 3.7.3.7 of RESAR SP/90 Module 7 does not agree with equation (4) in position 1.2.1 of Regulatory Guide 1.92.

In response to staff Question 210.51, the applicant submitted Reference 10 from Section 3.7.5 of Module 7 to the RESAR SP/90 application as an optional method in Section 3.7.3.7.B of Module 7 to account for high frequency (>33 Hz) modes. In this method, the dynamic response of modes up to 33 Hz is calculated by the standard response spectrum analysis method and the responses beyond 33 Hz are combined by algebraic sum. These responses of modes below 33 Hz and above 33 Hz are then combined by the SRSS method. This response, combined with a clarification provided by Westinghouse in its response to open items dated June 1989, provides sufficient information for the staff to find this method acceptable.

Also in response to Question 210.51, the applicant also submitted References 11, 12, and 13 from Section 3.7.5 of Module 7 as a basis for its multiple response spectrum analyses discussed in Sections 3.7.3.7.C, 3.7.3.7.D, and 3.7.3.9.A.2 of Module 7. In these methods, the representative maximum modal response is obtained by either algebraic or SRSS combinations of individual support point inputs, depending on the type of non-uniform input excitation. In its response to open items dated June 1989, Westinghouse offered the following:

> Proportional Input - For proportional input, the support motion at a given point can be obtained through multiplication of a reference excitation by a real number. This type of input is applicable in in case of support locations in the same supporting structure and is analogous to the use of the proper phase characteristic of the building motion for the calculation of response due to seismic anchor motions (Subsection 3.7.3.9.B). For this type of input, the representative maximum modal response is obtained by algebraic combination of contributions of the individual support point inputs. This combination is applicable when the response of the supporting structure is dominated by modes with in-phase displacement, such as rigid body modes and low frequency modes.
>
> Non-proportional Input - For non-proportional input, the support motions at each location cannot be obtained by multiplication of a reference excitation by a real number. This type of input is applicable in the case of support locations in different supporting structures. The representative maximum modal response is obtained by square root sum of the square combination of the contributions of the individual support point inputs. This type of input is applicable when the conditions described above for proportional input are not satisfied.

The definition of support groups under the category of proportional input in the response is not acceptable. Algebraic summation of within-group responses is acceptable only when a support group is defined by supports that have the same time-history input. This usually means all supports are located on the same floor or portions of a floor in a specific structure (flexible structure).

In response to Question 210.52, the applicant has stated that the methods used to calculate system response to seismic support displacement comply with SRP Section 3.9.II.2.g, with one exception. In the case of support locations in the same supporting structure, the proper representation is frequently found to be in-phase motion of the support location. This occurs when the seismic displacements of the supporting structure are dominated by modes with in-phase displacements, such as rigid body modes and low frequency modes. Examples of such supporting structures are those large concrete supporting structures found in the containment shell and the containment interior buildings. The staff agrees with the applicant that the assumption of in-phase seismic support movement is valid under the conditions stated above. For such conditions, the applicant's representation provides the same level of conservatism as required by SRP Section 3.9.II.2.g. Therefore, the staff finds it acceptable.

In its response to open items dated June 1989, Westinghouse submitted the following proposed revision to Section 3.7.3.9.C of RESAR SP/90 Module 7.

Section 3.7.3.9.C of RESAR SP/90 Module 7, "Structural/Equipment Design," has been revised so that inertial responses calculated by the envelope response spectra method are combined with seismic anchor motion analysis by the absolute sum method. It states:

> The results of modal envelope seismic spectra analysis in Item A.1 above and the results of seismic anchor motion analysis in Item B above are combined by the absolute sum method when required by consideration of the ASME classification of stress. The results of the modal non-uniform seismic spectra results in Item A.2 above and the results of seismic anchor motion analysis in Item B above are combined by the SRSS method.

This revision is not completely acceptable. The last sentence in the revision states that the results of the modal non-uniform seismic spectra analyses and seismic anchor motion analysis are combined by the SRSS. The staff's position on this issue, as stated in SRP Section 3.9.2, is that inertia effects and seismic anchor movement should be combined by absolute sum.

In response to staff Question 210.55, the applicant stated that for RESAR SP/90 the application of ASME Code Case N-411 damping values is consistent with all of the conditions in Regulatory Guide 1.84, "Design and Fabrication Code Case Acceptability -- ASME Section III, Division 1," Revision 24, except for one, composite damping for piping system models with flexible in-line building-mounted equipment. Additional clarification was provided in the Westinghouse response to open items dated June 1989. In that response Westinghouse stated:

> The damping values for auxiliary piping systems are shown in Table 3.9-6. Piping systems with different nominal diameters and different damping characteristics are evaluated using the methods of Subsection 3.7.3.15. Alternatively, the damping values in Figure 3.7-8 may be used. When Figure 3.7-8 is used, energy absorbing supports are not used for auxiliary piping systems. Figure 3.7-8 is applicable to piping systems which have flexible in-line components and flexible building mounted components, such as valves and tanks, respectively.

This response references RESAR SP/90 Module 7, Section 3.7.3.15, which in turn references ASCE-486. ASCE-486 has not yet been endorsed by the staff for piping systems. If the composite damping value or nonproportional damping models proposed in ASCE-486 are to be used in piping system analyses, the staff requests more details relative to the bases for this criteria. Based on the responses to Question 210.55 and open item #30, it is the staff's conclusion that when the damping values from Table 3.7-8 of Module 7 are used, the RESAR SP/90 application meets the guidelines of Regulatory Guide 1.84.

For the dynamic analysis of seismic Category I piping, each piping system was idealized as a mathematical model consisting of lumped masses connected by elastic members. The stiffness matrix for the piping system was determined using the elastic properties of the pipe. This included the effects of torsional, bending, shear, and axial deformations as well as change in stiffness resulting from curved members. The mode shapes and the undamped natural frequencies were then obtained. The dynamic response of the system was calculated by using the response spectrum method of analysis. For a piping system that was supported at points with different dynamic excitations, the response analysis was performed using either a modal envelope response spectra analysis or a non-uniform seismic response spectra analysis method.

Pending satisfactory resolution of the issues discussed above, the staff concludes that the applicant will meet GDC 2 with regard to demonstrating the design adequacy of all seismic Category I piping systems, components, and their supports to withstand earthquakes by meeting the regulatory positions of Regulatory Guides 1.61 and 1.92 or their equivalent and by providing acceptable seismic analysis procedures and criteria. The scope of review of the seismic subsystem analysis included the seismic analysis methods of all Category I piping systems, components, and their supports. It included review of procedures for modeling and inclusion of torsional effects, seismic analysis of multiply supported equipment and components with distinct criteria and procedures for evaluation of the interaction of non-Category I piping. The review also included criteria and seismic analysis procedures for reactor internals.

3.9.2.3 Preoperational Flow-Induced Vibration Testing of Reactor Internals (RESAR SP/90 Module 5, Section 3.9.2.4)

The staff requires that flow-induced vibration testing of reactor internals be conducted at all plants in accordance with applicable guidelines of Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing." The purpose of this test is to demonstrate that flow-induced vibrations similar to those expected during operation will not cause unanticipated structural damage.

In response to staff Question 210.31, the applicant revised RESAR Subsection 3.9.2.4 of Amendment 4 to Module 5 to state that the first RESAR SP/90 plant to become operational will be classified as a "prototype" and tested in accordance with the guidelines for prototype plants in Regulatory Guide 1.20. In addition to this acceptable commitment, the staff also required an explicit statement that every plant referencing the RESAR SP/90 to be built will undergo flow-induced vibration testing in accordance with applicable guidelines of Regulatory Guide 1.20 for nonprototype plants. In its response to open items dated June 1989, Westinghouse committed to perform hot functional testing for each SP/90 plant in accordance with the applicable guidelines of Regulatory Guide 1.20 for nonprototype plants. The staff finds this acceptable.

However, the SP/90 response to Generic Issue C-12, "Primary System Vibration Assessment," in Module 2, Amendment 3, is not completely consistent with the above commitments in Module 5 and requires clarification. The response in Module 2 states that reactor internals will undergo flow-induced vibration testing in accordance with applicable guidelines of Regulatory Guide 1.20 for nonprototype plants. This response should be revised to be consistent with the commitment in Module 5 relative to prototype testing of the first RESAR SP/90 plant. Westinghouse also stated in the same submittal that a scoping analysis for the preoperational flow-induced vibration test program for the first SP/90 reactor internals has been performed. This study has indicated that it may not be necessary to have a dummy core in place; however, given the preliminary nature of this study no firm decision on the presence of a dummy core has been made at this time.

As part of the detailed design phase, Westinghouse will perform a more extensive analysis to determine the requirements (test conditions, instrumentation, etc.) for the preoperational flow-induced vibration test program in general and the need for a dummy core in particular. If the results of this study show that a dummy core is not required, the justification for that decision will be included in the RESAR SP/90 FDA application.

Pending satisfactory resolution of the open items discussed above, the staff concludes that the applicant has met GDC 1 and 4 with regard to the reactor internals being designed and tested to quality standards commensurate with the importance of the safety functions being performed and being appropriately protected against dynamic effects (1) by meeting Regulatory Guide 1.20 for the conduct of preoperational vibration tests and (2) by having a preoperational vibration program planned for the reactor internals that provides an acceptable basis for verifying the design adequacy of these internals under test loading conditions comparable to those that will be experienced during operation. The combination of tests, predictive analysis, and post-test inspection provides adequate assurance that the reactor internals will, during their service life, withstand flow-induced vibrations of the reactor without loss of structural integrity. The integrity of the reactor internals in service is essential to ensure the proper positioning of reactor fuel assemblies and the incore instrumentation system to permit safe reactor operation and shutdown.

3.9.2.4 Dynamic System Analysis of Reactor Internals Under Faulted Conditions

The staff reviewed the following sections of the RESAR SP/90 application:

- 3.9.2.5, Module 5, "Dynamic System Analysis of the Reactor Internals Under Faulted Conditions"
- 3.9.5, Module 5, "Reactor Internals"
- 3.6.2.1.1.E, Module 7, "Piping Within Mechanistic Pipe Break Criteria"
- 3.7.3.14, Module 7, "Seismic Analyses for Reactor Internals (core, core supports, mechanisms)
- 3.9.5, Module 7, "Reactor Pressure Vessel Internals"

One of the staff's positions on this issue is that all reactor internals important to safety should be designed to withstand either (1) the combined loads resulting from a simultaneous LOCA and SSE or (2) the combined loads resulting from simultaneous applicable pipe break, if any, and SSE, where the application of mechanistic pipe break criteria (leak-before-break) has been justified for the reactor coolant loop.

In its response to open items dated June 1989, Westinghouse provided the following commitment:

> A digital computer program, which was developed for the purpose of calculating local fluid pressure, flow, and density transients that occur in pressurized water reactor coolant systems during a LOCA; or other evaluation methods will be used to evaluate the structural effects on the reactor internals of the pipe breaks postulated for the SP/90.
>
> The reactor internals important to safety were evaluated to withstand the combined loads resulting from a simultaneous loss-of-coolant accident (LOCA) and safe shutdown earthquake (SSE). This combination was performed by the SRSS method, square root sum of the squares.
>
> The results of these evaluations will be provided in the RESAR SP/90 FDA document.

The staff finds this commitment to be acceptable.

Pending satisfactorily resolution of the above open item, the staff concludes that the applicant will meet applicable portions of GDC 2 and 4 by performing a dynamic system analysis that 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 a postulated LOCA (or applicable pipe rupture) and SSE. The analysis provides adequate assurance (1) 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 (2) that the resulting deflections or displacements at any structural element of the reactor internals will not distort the reactor internals geometry to the extent that core cooling may be impaired. The methods used for component analysis have been found to be compatible with those used for the system analysis. Therefore, the combination of component and system analyses is acceptable.

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures (RESAR SP/90 Module 7, Section 3.9.3)

The staff reviewed loading combinations and their respective stress limits, the design and installation of pressure-relief devices, and the design and structural integrity of components and component supports of ASME Code Classes 1, 2, and 3.

This review was conducted in accordance with SRP Section 3.9.3 to determine the structural integrity and functional capability of pressure-retaining components, their supports, and core support structures that are designed in accordance with ASME Code Section III or earlier industrial standards.

3.9.3.1 Loading Combinations, Design Transients, and Stress Limits (RESAR SP/90 Module 7, Section 3.9.3.1)

The staff reviewed the methodology used for load combinations and the selected values of allowable stress limits. The applicant has evaluated all ASME Code Class 1, 2, and 3 components, component supports, core support components, control rod drive components, and other reactor internals using the load combinations and stress limits provided in RESAR Section 3.9.3 of Module 7.

The ASME Code requires that a design specification be prepared for Class 1, 2, and 3 components such as pumps, valves, and piping systems. The design specification is intended to become a principal document governing design and construction of these components and should include specification of loading combinations and other design data. The Code also requires a design report for ASME Code Class 1, 2, and 3 piping and components. During its review of the RESAR SP/90 FDA application, the staff will audit and review design documents for selected pumps, valves, and piping systems to determine that the selected design specifications and design reports are in compliance with ASME Code requirements and are acceptable.

In response to staff Question 210.61, the applicant revised Table 3.9-3 of Module 7 to the RESAR SP/90 application to include valve discs as part of the pressure-retaining boundary. The allowable stress limits as given in RESAR Table 3.9-3 are acceptable. However, RESAR Table 3.9-5 for Class 2 and 3 valves does not contain similar criteria. In its response to open items dated June 1989, Westinghouse committed to revise Table 3.9-5 to include valve discs along with valves for the various service levels. The staff finds this acceptable.

In response to staff Question 210.62, the applicant provided design criteria for safety-related heating, ventilation, and air conditioning (HVAC) ductwork and supports in Section 3.9.3.5 of Module 7, Amendment 2 to the RESAR SP/90 application. The following standards will be used in the design:

- High Pressure Duct Construction Standards, Sheet Metal and Air Conditioning Contractors' National Association (SMACNA, 1975)
- Specification for the Design of Cold Formed Steel Members, American Iron and Steel Institute (AISI)
- Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Power Plants, American Institute of Steel Construction (AISC-N690, 1984)
- Seismic Analysis of Safety Related Nuclear Structures, American Society of Civil Engineers (ASCE, 1986)

The staff has not yet endorsed any of the above standards; therefore, it requests the applicant to provide a more detailed response to identify the specific sections of these standards that will be used in the design of HVAC ductwork and supports. In addition, the applicant has proposed allowable stresses for ductwork under SSE conditions of 1.6 times the normal allowables of AISI. The staff finds this unacceptable unless justification is provided on the basis of test data to demonstrate that this will not lead to loss of leak tightness and functionability during a seismic event. In its response to open items dated June 1989, Westinghouse changed the AISC code used for qualification of HVAC supports from AISC-N690 to AISC-S326, which is acceptable for these supports. However, the items noted above will remain as open items to be addressed at the FDA stage of review.

Staff Question 210.26 requested that the applicant revise RESAR Subsection 6.5.1.2 of Module 2 to provide sufficient information to respond to IE Bulletin

79-02, "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts." The applicant referenced Appendix B to American Concrete Institute (ACI) Standard 349 (no date), "Code Requirements for Nuclear Safety-Related Structures," in response to IE Bulletin 79-02. According to Regulatory Guide 1.142, ACI-349 is approved except for Appendix B. This appendix requires a margin of 3 on concrete expansion anchor bolts whereas IE Bulletin 79-02 requires 4 or 5. Appendix B also does not adequately address base plate flexibility. In addition, the applicant's response to staff Question 210.66 with regard to stress limits for the design of bolts embedded in concrete and expansion anchor bolts in safety-related component supports cites ACI-349, Appendix B, as the basis for its proposed allowable stress limits. The use of ACI-349, Appendix B, is unacceptable for reasons discussed above. This will remain an open item until the inconsistencies between the applicant's and the staff's position (as delineated in IE Bulletin 79-02) have been resolved.

As a result of recent information, the staff has received with regard to thermal stratification in several operating plants, the applicant is requested to provide assurance that systems connected to the reactor coolant system (RCS) have been reviewed in sufficient detail so that unisolable sections of piping connected to the RCS will not be subjected to combined cyclic and static thermal stresses as a result of temperature stratification or oscillations that could cause high stresses or fatigue failure. Such stratifications or oscillations could be caused by leakage across valves, design deficiencies, or faulty operational procedures. The applicant should provide assurance that all unisolable sections that are vulnerable to such stratification or oscillations have been thoroughly evaluated in the design analysis of the piping and appropriate pressure and temperature monitoring programs will be implemented to preclude the possibility of subjecting the unisolable sections of all piping connected to the RCS to stresses that could cause fatigue failure. The applicant must also address compliance with IE Bulletins 88-08 and 88-11 when this information is provided.

Pending satisfactory resolution of the open items, the staff finds that the applicant has met 10 CFR 50.55a and GDC 1, 2, and 4 with regard to the design and service load combinations and associated stress and deformation limits specified for ASME Code Class 1, 2, and 3 components by ensuring (1) that systems and components important to safety are designed to quality standards commensurate with their importance to safety and (2) that these systems can accommodate the effects of normal operation as well as postulated events such as LOCAs and the dynamic effects resulting from earthquakes. The specified design and service combinations of loading as applied to ASME Code Class 1, 2, and 3 pressure-retaining components in systems designed to meet seismic Category I standards provide assurance that, in the event of an earthquake affecting the site or other service loading caused by postulated events or system operating transients, the resulting combined stresses imposed on system components will not exceed allowable stress and strain limits for the materials of construction. Limiting stresses under such loading combinations provides a conservative basis for the design of system components to withstand the most adverse combination of loading events without loss of structural integrity.

3.9.3.2 Design and Installation of Pressure-Relief Devices (RESAR SP/90 Module 7, Section 3.9.3.3; Module 2, Sections 3.1.15, 4.9)

The staff reviewed RESAR Section 3.9.3.3 of Module 7 and Sections 3.1.15 and 4.9 of Module 2 with respect to the design, installation, and testing criteria applicable to the mounting of pressure-relief devices used for the overpressure protection of ASME Code Class 1, 2, and 3 components. This review, conducted in accordance with SRP Section 3.9.3, includes evaluation of the applicable loading combinations and stress criteria. The design review extends to consideration of the means provided to accommodate the rapidly applied reaction force when a safety valve or relief valve opens and the transient fluid-induced loads applied to the piping downstream of a safety or relief valve in a closed discharge piping system.

In accordance with Item II.D.1 of NUREG-0737, "Clarification of TMI [Three Mile Island] Action Plan Requirements," pressurized-water reactor and boiling-water reactor licensees and applicants are required to conduct testing to qualify the RCS relief and safety valves, block valves, and associated piping and supports under expected operating conditions for design-basis transients and accidents.

In mid-1980, the NRC staff initiated a program for Commission approval of a course of action that would lead to the establishment of TMI-2 related requirements for pending construction permit applications. This program led to the issuance of NUREG-0718, Revision 2, "Licensing Requirements for Pending Applications for Construction Permits and Manufacturing Licenses," which specifies those NRC action plan items (NUREG-0660) that are required to be implemented or committed to by a pending applicant before receiving a construction permit or a license to manufacture. In addition, the NRC has issued a revision to 10 CFR 50.34, "Contents of Applications; Technical Information," that essentially incorporates the post-TMI requirements of NUREG-0718 into its regulations.

NUREG-0737 contains those post-TMI requirements that have been Commission approved for implementation by operating plant licensees and applicants. In many cases, the specific requirements of NUREG-0737 are identical to those of NUREG-0718 and 10 CFR 50.34(f) and are discussed in Section 3.1 of Module 2 to the RESAR SP/90 application. In Section 3.1.15 of Module 2, the applicant discusses RCS valve testing requirements of 10 CFR 50.34(f)(2)(r) that potentially impact the RESAR SP/90 design. The purpose of this regulation is to require qualification of relief, safety, and block valves to the RCS under expected operating conditions (including solid-water and two-phase flow conditions) and anticipated-transients-without-scram (ATWS) conditions.

Generic RCS valve testing (sponsored by the Electric Power Research Institute [EPRI]) has been conducted in support of operating plant licensees and applicants. The EPRI program included representative fluid conditions including solid-water and two-phase flow conditions. The EPRI program did not, however, include specific consideration of ATWS conditions.

Operating plant licensees and applicants have submitted documentation to demonstrate applicability of the generic EPRI test results to their plant-specific RCS valves, expected fluid conditions, and piping and support configurations. The applicant expects that the generic EPRI test results will be directly applicable to the RESAR SP/90 plant design. The basis for this assumption is that the latest Westinghouse pressurizer power-operated relief valves and safety valves were included in the test program. However, 10 CFR 50.34(f)(2)(x) requires that consideration of ATWS conditions shall be included in the test program. In RESAR Section 4.9 of Module 2, the applicant provided a response to NRC Generic Issue A-9, "Anticipated Transients Without Scram." Generic Issue A-9 was superseded by 10 CFR 50.62, "Requirements for Reduction of Risk from ATWS Events for Light Water Cooled Nuclear Power Plants." In this response, the applicant stated that the RESAR SP/90 will be designed with an ATWS mitigating system that generates a turbine trip and emergency feedwater start signal independent of the integrated protection system. The system will be designed to be safety grade to the maximum extent feasible from a cost-benefit viewpoint and will be highly reliable. In addition, the RESAR SP/90 design will contain new features that provide additional mitigation capability for ATWS events and ATWS considerations will be factored into the sizing and number of pressurizer power-operated relief valves. A detailed analysis of the limiting ATWS events will be performed during the FDA stage of review to demonstrate that ATWS acceptance criteria are met.

In RESAR Section 3.1.15 of Module 2, in response to 10 CFR 50.34(f)(2)(x), the applicant stated that it will document the applicability of the generic EPRI test results to the RESAR SP/90 design (including valve designs, piping and support designs, and fluid conditions) during the licensing process for the RESAR SP/90 design. If the generic EPRI test results do not envelope the specific RESAR SP/90 design, Westinghouse will either (1) perform additional testing or (2) demonstrate justification for not performing additional testing possibly through additional analyses and/or evaluations.

The staff concludes that the applicant's response to Generic Issue A-9, which the staff is assuming is implicitly a response to 10 CFR 50.62 combined with the response to 10 CFR 50.34(f)(2)(x), provides an acceptable approach to use for responding to this TMI item at the PDA stage of review. During its review of the FDA, the staff will perform a detailed review of this issue. Part of that review will be an evaluation of the applicant's analysis of the limiting ATWS events to determine if the EPRI test results envelop the RESAR SP/90 design.

The staff concludes that the applicant has met 10 CFR 50.55a and GDC 1, 2, and 3 with regard to the criteria to be used for design and installation of ASME Code Class 1, 2, and 3 overpressure-relief devices by ensuring that safety and relief valves and their installations will be designed to standards that are commensurate with their safety functions and that they will accommodate the effects of discharge caused by normal operation as well as postulated events such as LOCAs and the dynamic effects resulting from the SSE. The applicant also has met GDC 14 and 15 with regard to ensuring that the reactor coolant pressure boundary design limits for normal operation, including anticipated operational occurrences, will not be exceeded. The criteria used by the applicant in the design and installation of ASME Code Class 1, 2, and 3 safety and relief valves provide adequate assurance that, under discharging conditions, the resulting stresses will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under the loading combinations associated with the actuation of these pressure-relief devices provides a conservative basis for the design and installation of the devices to withstand these loads without loss of structural integrity or impairment of the overpressure protection function.

3.9.3.3 Component Supports (RESAR SP/90 Module 7, Section 3.9.3.4)

The staff reviewed RESAR Section 3.9.3.4 of Module 7 with regard to the methodology used in the design of ASME Code Class 1, 2, and 3 component supports. The review includes assessment of design and structural integrity of the supports and addresses three types of supports: plate and shell, linear, and component standard types. In response to the staff's question 210.65, Westinghouse stated that all ASME Code Class 1, 2, and 3 component supports for RESAR SP/90 will be constructed in accordance with the rules of ASME Code, Section III, Subsection NF, "Component Supports," 1986 Edition or later. To agree with the staff's position relative to the jurisdictional boundary between ASME NF supports and non-NF supports, this commitment should be changed to Subsection NF, "Component Supports," 1987 Addenda to the 1986 Edition or later. Loading combinations for component supports and Class CS component evaluation findings are discussed in Sections 3.9.3.1 and 3.9.5, respectively, of this SER.

The staff concludes that the applicant has met 10 CFR 50.55a and GDC 1, 2, and 4 with respect to the design and service load combinations and associated stress and deformation limits specified for ASME Code Class 1, 2, and 3 component supports by ensuring (1) that component supports important to safety will be designed to quality standards commensurate with their importance to safety and (2) that these supports will accommodate the effects of normal operation as well as postulated events such as LOCAs and the dynamic effects resulting from the SSE. The combination of loadings (including system operating transients) considered for each component support within a system, including the designation of the appropriate service stress limit for each loading combination, will meet Regulatory Guides 1.124, "Design Limits and Loading Combinations for Class 1 Linear-Type Component Supports" and 1.130, "Design Limits and Loading Combinations for Class 1 Plate- and Shell-Type Component Supports" and will be in accordance with NUREG-0484, Revision 1. The specified design and service loading combinations used for the design of ASME Code Class 1, 2, and 3 component supports in systems classified as seismic Category I provide assurance that, in the event of an earthquake or other service loadings caused by postulated events or system operating transients, the resulting combined stresses imposed on system components and component supports will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative design basis to ensure that support components will withstand the most adverse combination of loading events without loss of structural integrity.

3.9.4 Control Rod Drive Systems (RESAR SP/90 Module 7, Section 3.9.4)

The staff reviewed the design of the control rod drive system up to its interface with the control rods in accordance with SRP Section 3.9.4. The rods and control rod drive mechanism (CRDM) must be demonstrated to be capable of reliably controlling reactivity changes either under conditions of anticipated normal plant operational occurrences or under postulated accident conditions. The primary function of the CRDMs are to insert and withdraw the rod cluster control assemblies (RCCAs) in discrete steps or to hold the RCCA stationary within the core to control reactivity. The staff reviewed the information in Section 3.9.4 of Modules 5 and 7, Section 1.5.1.3.1 of Module 5 and Section 3.2 of Module 5 with regard to the analyses and tests performed to ensure the structural integrity and functional capability of this system during normal operation and under accident conditions.

The CRDM pressure housings are ASME Code Class 1 components, and the CRDMs are ASME Code Class 2. Both are designed to applicable requirements of ASME Code, Section III. Loading combinations for CRDMs are discussed in Section 3.9.3.1 of this SER.

A full-scale CRDM test program was performed at the Westinghouse D-Loop test facility. D-Loop is a high-temperature, high-flow-rate facility that can test full-size components under simulated conditions of chemistry, temperature, pressure, and flow. For the CRDM tests, the D-Loop contained a complete full-scale reactivity control cluster drive line that included the reactivity control cluster, drive rod and couplings, and the CRDM. Associated equipment in the test facility included the pressure vessel, simulated lower internals with a prototype fuel assembly and simulated upper internals with a prototype reactivity control cluster rod guide and an upper calandria mock-up. The CRDM relies on forced air cooling to maintain its magnetic coils at a safe operating temperature. Therefore, thermocouples were installed to permit calculation of total heat rejection.

The CRDM was exercised for 7.8 million steps in a pattern that included simulated baseload operation, load follow, frequency control, and rod drops. Throughout the test period of approximately 6 months, the CRDM operated without problems. The measured heat rejection was similar to that found on previous CRDM tests. Subsequent to a post-test inspection, Westinghouse estimated that, with regard to wear, the latch arms could probably have operated up to approximately 10 million steps. During previous tests of current Westinghouse CRDM designs, the maximum stepping duty did not exceed 4 million steps, primarily because load-follow operation and frequency control were not considered.

The staff concludes that the design of the control rod drive system is acceptable for the PDA and will meet GDC 1, 2, 14, 26, 27, and 29 and 10 CFR 50.55a. The applicant meets GDC 1 and 10 CFR 50.55a with regard to designing components important to safety to quality standards commensurate with the importance of the safety functions to be performed. The design procedures and criteria used for control rod drive systems are in conformance with appropriate ANSI standards and ASME codes. The applicant meets GDC 2, 14, and 26 with regard to designing the control rod drive system to withstand the effects of earthquakes and anticipated normal operation occurrences with adequate margins to ensure its structural integrity and functional capability and with extremely low probability of leakage or gross rupture of the reactor coolant pressure boundary. The specified design transients, design and service loadings, combinations of loads, and limiting the stresses and deformations under such loading combinations are in conformance with the appropriate ANSI standards and ASME codes and acceptable regulatory positions specified in SRP Section 3.9.3. The applicant meets GDC 27 and 29 with regard to designing the control rod drive system to ensure its capability to control reactivity and cool the reactor core with appropriate margin in conjunction with either the emergency core cooling system or the reactor protection system. The operability assurance program is acceptable with respect to meeting system design requirements in observed performance as to wear, functioning times, latching, and overcoming a stuck rod.

3.9.5 Reactor Pressure Vessel Internals (RESAR SP/90 Module 7, Section 3.9.5)

SRP Section 3.9.5 applies to load combinations, allowable stress limits, and other criteria used in the design of the reactor internals. However, the staff limited its review to the design and analysis of the reactor internals and the deformation limits specified for those components. Quality group classification and loading combinations for reactor internals are discussed in Sections 3.2.2 and 3.9.3.1, respectively, of this SER.

RESAR Section 3.9.5.1.3.4 of Module 5 discusses the bottom-mounted instrumentation (BMI) flux thimbles that penetrate through the bottom of the reactor pressure vessel. A problem of unacceptable accelerated wear of Westinghouse-designed BMI thimbles was identified in a European plant in 1985. Since that time, thimble tube thinning has been detected in many U.S. pressurized-water reactors (PWRs) that contain Westinghouse designed thimbles. As a result, on July 26, 1988, the staff issued NRC Bulletin 88-09, "Thimble Tube Thinning in Westinghouse Reactors." This bulletin requests all licensees for Westinghouse PWRs that use BMI thimbles to establish and implement an inspection program to periodically confirm the BMI thimble tube integrity. In response to staff Question 210.33 about this issue, the applicant committed to review the RESAR SP/90 BMI design to determine if any design changes are warranted before it submits the FDA application. The staff concludes that this is an acceptable commitment for the PDA and will address this issue during the FDA review.

Pending satisfactory resolution of the quality group classification, the staff concludes that design of reactor internals is acceptable and meets GDC 1, 2, 4, and 10 and 10 CFR 50.55a. The applicant meets GDC 1 and 10 CFR 50.55a with regard to designing the reactor internals to quality standards commensurate with the importance of the safety functions to be performed. The design procedures and criteria used for the reactor internals are in conformance with the requirements of Subsection NG of ASME Code Section III. The applicant meets GDC 2, 4, and 10 with regard to designing components important to safety to withstand the effects of earthquake and the effects of normal operation, maintenance, testing, and postulated LOCAs with sufficient margin to ensure that their capability to perform their safety functions is maintained and the specified acceptance fuel design limits are not exceeded.

The specified design transients, design and service loadings, and combinations of loading as applied to the design of the reactor internals structures and components provide reasonable assurance that, in the event of an earthquake or of a system transient during normal plant operation, the resulting deflections and associated stresses imposed on these structures and components will not exceed allowable stresses and deformations under such loading combinations. This provides an acceptable basis for the design of these structures and components to withstand the most adverse loading events that have been postulated to occur during service lifetime without loss of structural integrity or impairment of function.

3.9.6 Inservice Testing of Pumps and Valves (RESAR SP/90 Module 7, Section 3.9.6)

The staff reviewed the inservice testing of certain safety-related pumps and valves typically designated as ASME Code Classes 1, 2, or 3 in accordance with SRP Section 3.9.6. Other pumps and valves not categorized as ASME Code Class 1, 2, or 3 may be included in this review if the staff considers them to be safety related. In Section 3.9.3 of this SER, the staff discusses the design of safety-related pumps and valves in the RESAR SP/90. The load combinations and stress limits to be used in the design of pumps and valves ensure that the component pressure boundary integrity will be maintained. In addition, the applicant will periodically test and perform periodic measurements of all safety-related pumps and valves. These tests and measurements will be performed in accordance with Section XI of the ASME Code. The tests will verify that these pumps and valves operate successfully when called on. Periodic measurements of various parameters will be compared to baseline measurements to detect long-term degradation of the pump or valve performance. In RESAR Section 3.9.6 of Amendment 2 to Module 7, the applicant has stated that an inservice test program will be prepared before a specific plant operating license referencing the RESAR SP/90 is issued. This program will include all safety-related pumps and valves even if they are not classified as ASME Class 1, 2, or 3. The program will conform to the applicable requirements of ASME Section XI and 10 CFR 50.55a(g).

The content and scope of an acceptable test program for safety-related pumps and valves in evolutionary LWR designs has been under review by the staff. The results of this effort will be included in the staff's review during the FDA stage.

In its response to open items dated June 1989, Westinghouse provided clarification related to testing difficulties for inservice pump and valve testing. Westinghouse stated that generically for the SP/90 design all remote safety-related valves cannot be exercised/tested to ASME Section XI requirements during full-power operation. Valves that should not be tested include: (1) valves providing the RCS pressure boundary such as the RHR suction isolation valves; (2) the pressurizer power-operated relief valves (PORVs) and emergency letdown isolation valves; (3) the steam generator PORVs; (4) valves whose operation would result in plant transients such as the main steam isolation valves, main feedwater isolation valves, letdown isolation valves of the chemical volume control system, steam generator blowdown isolation valves, and reactor coolant pump seal injection isolation valves. However, it should be noted that all remote safety-related valves for the safety injection and emergency feedwater system functions can be exercised/tested in accordance with ASME Section XI requirements at full power.

Additionally, during residual heat removal (RHR) cooldown, the RHR flowpath includes the four RHR heat exchanger and reactor vessel injection paths. Following cooldown during the refueling shutdown, all four high-head safety injection pumps would be tested at full flow taking suction from the emergency water storage tank (EWST) and returning water to the EWST via the full-flow test line downstream of the RHR heat exchangers.

Several safety systems connected to the reactor coolant pressure boundary have design pressure below the rated RCS pressure. In addition, some systems that are rated at full reactor pressure on the discharge side of pumps have pump suction below RCS pressure. To protect these systems from RCS pressure, two or more isolation valves will be placed in series to form the interface between the high-pressure RCS and the low-pressure system. The leaktight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low-pressure systems.

In response to staff Question 210.71 with regard to periodic leak testing of pressure isolation valves, the applicant referenced its response to Question 210.38. In that response a commitment is made to perform periodic leak testing of all pressure isolation valves in accordance with the requirements of the Westinghouse Standard Technical Specifications. The staff has concluded that this is an acceptable commitment for the RESAR SP/90 PDA stage. The staff will perform a detailed review of this issue during its review of the FDA application.

In addition, as discussed in Section 14 of this SER, the applicant will be required to address inservice inspection and testing as part of a program to ensure design reliability during the FDA stage of review.

Pending resolution of the open items identified above, the staff concludes that the applicant's commitment to a pumps and valves program is acceptable pending meeting the requirements of 10 CFR 50, Appendix A, GDC 37, 40, 43, 46, 54, and §50.55a(g). This conclusion is based on the applicant's commitments to provide a test program to ensure that safety-related pumps and valves will be in a state of operational readiness to perform necessary safety functions throughout the life of the plant.

3.10 Seismic and Dynamic Qualification of Safety-Related Mechanical and Electrical Equipment (RESAR SP/90 Module 7, Section 3.10)

A general discussion of seismic and dynamic qualification is provided in RESAR SP/90 Module 7, Sections 3.9.2.2 and 3.9.3.2 for mechanical equipment and Section 3.10 for electrical equipment. All of these sections contain numerous references to Institute of Electrical and Electronics Engineers (IEEE) Standard 344-1975 and Regulatory Guide 1.100, "Seismic Qualification of Electrical Equipment for Nuclear Power Plants." The staff's position for current applications, such as RESAR SP/90, is that IEEE Standard 344-1987 and Regulatory Guide 1.100, Revision 2, should be used for seismic qualification of electrical and mechanical equipment. Westinghouse has not yet established a position on IEEE Standard 344-1987 and Regulatory Guide 1.100, Revision 2. Westinghouse will provide its position during the FDA stage of review.

The staff reviewed the remaining information in the above Module 7 Sections, such as deciding factors for performing tests, analyses, or combinations of both; the considerations in defining the seismic and other relevant dynamic load input motions; and the proposed demonstration of the adequacy of the qualification program. This information if sufficient for the PDA stage of review to meet the intent of SRP Section 3.10. During its review of the RESAR SP/90 FDA application, the staff will perform a detailed evaluation of the seismic and dynamic qualification program. Part of this review will be an audit of the first RESAR SP/90 equipment qualification files in accordance with SRP Section 3.10.

A general discussion of the pump and valve operability assurance program is provided in Subsection 3.9.3.2 of Module 7 to the RESAR application. The staff's comments above about the references to IEEE Standard 344 and Regulatory Guide 1.100 also apply to this operability program. The remaining information in Subsection 3.9.3.2 with regard to tests and analyses that will be conducted to ensure structural integrity and operability of safety-related pumps, valves, and applicable supports in the event of an SSE, after a number of postulated occurrences of an OBE and in combination with other relevant dynamic and static loads, is sufficient for the PDA stage of review to meet the intent of SRP Section 3.10.

Pending satisfactory resolution of the issue discussed above, the staff concludes that appropriate programs for seismic and dynamic qualification and pump and valve operability assurance have been outlined in the RESAR SP/90 PDA application. After implementation, these programs must meet applicable portions of GDC 1, 2, 4, 14, and 30 of 10 CFR 50, Appendices A and B, and 10 CFR 100, Appendix A.

During its review of the RESAR SP/90 FDA application, the staff will perform a detailed evaluation of both of the above programs, including the implementation of each program. Part of this review will be audits of the first RESAR SP/90 files to review the results of tests and analyses that were performed to ensure the proper implementation of criteria established in the PDA review, to ensure that adequate qualification has been demonstrated for all equipment and supports, and to verify that all applicable loads have been properly defined and accounted for in the testing/analyses performed.

3.11 Environmental Design of Mechanical and Electrical Equipment (RESAR SP/90 Module 7, Section 3.11)

Equipment that is used to perform a necessary safety function must be demonstrated capable of maintaining functional operability under all service conditions postulated to occur during its installed life for the time it is required to operate. This requirement is embodied in GDC 1 and 4 and Sections III, XI, and XVII of 10 CFR 50, Appendix B. More detailed requirements and guidance relating to the methods and procedures for demonstrating this capability for electrical equipment have been set forth in 10 CFR 50.49, and NUREG-0588, "Interim Staff Position of Environmental Qualification of Safety-Related Electrical Equipment," which supplements IEEE Standard 323.

Information related to environmental qualification of safety-related equipment is provided partially in Section 3.11 of the RESAR SP/90 application and, by reference, in the following documents:

- "Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety-Related Electrical Equipment," WCAP-8587, Revision 6, November 1983

- "Equipment Qualification Data Packages," WCAP-8587, Supplement 1, latest revision

- "Westinghouse Water Reactor Divisions Quality Assurance Plan," WCAP-8730, Revision 9a, Amendment 1, February 1981

In Section 3.11 of RESAR SP/90 Westinghouse committed to meet IEEE Standard 323-1974, including IEEE Standard 323a-1975 (the Nuclear Power Engineering Comittee Position Statement of July 24, 1974), by either type test, operating experience, analysis, or an appropriate combination of these methods, by employing the methodology described in the final NRC-approved version of WCAP-8587 above.

It is the staff's position that the commitment to IEEE-323 for environmental qualification of electrical equipment is not sufficient and that commitments to the guidance provided in Regulatory Guide 1.89, "Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants," and NUREG-0588 be made in RESAR SP/90.

Although there are no detailed requirements for mechanical equipment, GDC 1, GDC 4, and Appendix B to 10 CFR 50 contain requirements appropriate for environmental qualification of safety-related mechanical equipment.

The applicant committed to meet the requirements of GDC 1 and 4 in RESAR SP/90 Section 3.1. However, for the general methods of implementing the requirements of 10 CFR 50, Appendix B, the applicant refers to WCAP-8730 above.

It is the staff's position that, although it is appropriate for the staff to assess the implementation of the environmental qualification program through the review of topical reports such as those referenced in RESAR SP/90, Westinghouse did not explicitly commit to methods for satisfying the requirements of 10 CFR 50, Appendix B, Sections III, XI and XVII for environmental qualification of safety-related mechanical equipment in its RESAR SP/90 application. In a submittal dated June 28, 1989, Westinghouse stated that Section 3.11 of RESAR SP/90 will be modified during the FDA stage to specifically reference the latest codes and standards used in the RESAR SP/90 design. This will enable staff review regarding compliance with the requirements of 10 CFR 50, Appendices A and B, and 10 CFR 100, Appendix A. Therefore, the staff will review Westinghouse commitments with regard to the environmental qualification of safety-related equipment during the FDA stage of review.

4 REACTOR (RESAR SP/90 Module 5, Section 4.0)

4.1 Summary Description of Reactor (RESAR SP/90 Module 5, Section 4.1)

The RESAR SP/90 is designed to operate at a core thermal power of 3800 megawatts thermal (MWt). Sufficient margin exists to ensure that fuel damage will not occur during steady-state operation or anticipated operational occurrences.

The nuclear steam supply system (NSSS) is a four-loop design that will have a primary coolant flow rate of 150×10^6 lb/hr. The reactor coolant and moderator will be light water at a nominal system pressure of 2250 psia. The reactor core will consist of 193 fuel assemblies. Each assembly will contain a 19x19 array that includes 296 fuel rods. The center position in the assembly will be reserved for use by in-core instrumentation.

Two new features of the RESAR SP/90 design are the water displacer rod assembly, which will allow moderator control by varying the fuel assembly water content, and the gray rod assembly, which will assist in load-follow maneuvers to adjust power output.

The nuclear design analyses and evaluations establish physical locations for control rods, withdrawal sequence of the water displacer and gray rods, and physical parameters such as fuel enrichments and neutron absorber concentration in selected fuel rods and the coolant.

The thermal hydraulic design analyses and evaluation establish coolant flow parameters to ensure that adequate heat transfer is provided between the fuel cladding and the reactor coolant. The thermal hydraulic design takes into account local variations in dimensions, power generation, flow distribution, and mixing.

The staff review addressed in Section 4 of this report was performed in accordance with the applicable portions of the Standard Review Plan (SRP) (NUREG-0800).

4.2 Fuel System Mechanical Design (RESAR SP/90 Module 5, Section 4.2)

The fuel assembly for the RESAR SP/90 reactor will consist of 296 fuel rods, 1 instrumentation tube, and 16 guide thimbles in a 19x19 array. The guide thimbles will house the control rod assembly, water displacer rod assembly, or gray rod assembly, depending on location. The water displacer rods will be similar to water rods in a BWR in that they will control peaking by varying the fuel assembly water content. The gray rods will be absorber rods with less neutron absorption capability than rod cluster control assemblies (RCCA). These gray rods will be used for load-follow maneuvers.

The fuel rods will be supported by eight intermediate Zircaloy grids and top and bottom Inconel grids. Some selected fuel assemblies may have a number of fuel rods with the integral fuel burnable absorber, which is a thin boride coating on the uranium dioxide (UO_2) fuel pellets. The RCCA absorber will consist of boron carbide (B_4C) pellets with silver/indium/cadmium (Ag/In/Cd alloy on end tips. One of the distinct features of this fuel design is that the fuel assemblies are reconstitutable by removing either the top or the bottom grid.

4.2.1 Design Criteria (RESAR SP/90 Module 5, Section 4.2.1)

The fuel design criteria used for RESAR SP/90 are based on those criteria approved for WCAP-9500 for the 17x17 optimized fuel assembly (OFA) design. This approach is acceptable. However, as communicated by R. L. Tedesco (NRC) to T.N. Anderson (Westinghouse) by letter dated May 22, 1981, there are certain conditions attached to the approval of WCAP-9500 and these conditions must be addressed during the FDA stage of review.

4.2.2 Design Analyses and Testing (RESAR SP/90 Module 5, Section 4.2.1.7)

Westinghouse supplied only limited analytical results for the PDA review. A number of fuel system components require tests for confirmation of adequate performance. In Module 5 of the PDA application, Westinghouse states that these tests have not yet been performed. The applicant must provide complete design analyses and testing results during the FDA stage of review.

4.3 Nuclear Design (RESAR SP/90 Module 5, Section 4.3)

The staff conducted the review of the nuclear design of the RESAR SP/90 in accordance with the guidelines provided by SRP Section 4.3, on the basis of information contained in Module 5 of the RESAR SP/90 application, referenced topical reports, Westinghouse's responses to staff questions, and by comparison with previously reviewed characteristics of other Westinghouse reactors.

4.3.1 Differences From Standard Westinghouse Reactors

The nuclear-related physical and parametric design characteristics and methodology of analysis of RESAR SP/90 are in many aspects very similar to those of more typical Westinghouse reactors that have been approved by the staff and are already in operation or near operation or that have been extensively reviewed by the staff. However, there are many differences in detail between these reactors and the RESAR SP/90. These differences could be significant to the staff's evaluation of the RESAR SP/90 nuclear design. In Module 5 of the PDA application, Westinghouse compares characteristics of RESAR SP/90 to RESAR 414, for which the staff had completed a PDA review of the reactor core. However, the staff believes it would be useful for this review to consider a wider range of comparisons such as the typical Westinghouse four-loop 3411 MWt reactors (e.g., the Callaway plant); the same reactor with optimized fuel (e.g., Byron); an otherwise typical reactor, but with 3800 MWt power and a 14-foot-long core (i.e., South Texas Project), the RESAR 414, and other Westinghouse reactors with individual features related to the RESAR SP/90 design. The comparison will indicate where distinctive features of the RESAR SP/90 design have been bounded by past approved designs, where they are unique, and their possible safety significance.

The primary differences between these reactors and the RESAR SP/90 fall in seven categories: (1) core size and power density, (2) fuel and control rod geometry and material content, (3) moderator to fuel volume fraction variation and control via water displacement rods, (4) the use of gray rods in control, (5) a radial reflector, (6) four-segment ex-core neutron detectors and N16 detectors, and (7) core physics parameters.

Core Size and Power Density

The core volume of RESAR SP/90 will be nearly 50 percent larger than a typical Westinghouse reactor core (e.g., Callaway) because the effective diameter of the RESAR SP/90 core will be 2 feet larger. The core height will be between that of Callaway and South Texas. The rated power is the same as South Texas and RESAR 414 at 3800 MWt compared to 3411 MWt for Callaway. The power level and volume will result in an average core power density of 78 kW/liter, which is 25 percent lower than that for a typical Westinghouse plant, and a kW/kgU ratio, which is 83 percent lower than a typical Westinghouse plant and 24 percent lower than a typical plant with optimized fuel. This large reduction is a significant factor in the core safety review. The number of fuel pins (and length) will be increased so that the average linear power density is 5.06 kW/ft, which is 93 percent of typical cores with and without optimized fuel. The design total peaking factor, F_Q, however, will be 2.60 compared to 2.32 for a typical plant, and the peak design linear power density will be 13.2 kW/ft compared to 12.6 kW/ft for the typical reactor (South Texas and RESAR 414 have a peak linear power density of 13.0 and 14.0, respectively. The higher F_Q should be operationally straightforward to maintain, without the more restrictive required procedures of the standard offset control (providing an F_Q of 2.32) and without a significant increase above the standard peak linear power density (and within the range considered for RESAR 414).

Neutron flux levels will be about the same in RESAR SP/90 as a typical core, and axial xenon stability is expected to be similar and controllable, with no new problems anticipated. However, the larger diameter core may provide less stability for x-y (azimuthal) xenon oscillations (typical cores are stable) although Westinghouse calculations indicate the RESAR SP/90 core should be stable. It is staff policy to require x-y stability tests of a new reactor class; in response to questions, Westinghouse has indicated that such tests will be carried out for the first reactor. The staff finds this commitment acceptable.

The staff concludes that the increased volume/power ratio of the RESAR SP/90 will provide an improved average power density state with a design peak that is not significantly higher than that of typical Westinghouse cores and that provides no expected concomitant deleterious effects and a net improvement in margins to potential problem areas. Therefore, the staff finds this acceptable.

Fuel and Control Rods

The number (193) of fuel assemblies for RESAR SP/90 will be the same as for a typical Westinghouse design. However, the RESAR SP/90 fuel assemblies will be larger, contain more fuel pins (296 vs. 264), and have a different fuel-thimble geometry. The fuel pin array is 19x19. The fuel pin diameters (both pellet and cladding) will be larger than those of a typical 17x17 core, but will be less than those of a 15x15 core. The control rod thimbles will be larger than those of a typical reactor and will replace four fuel pins, rather than the usual one. The center thimble for instrumentation replaces one fuel pin. The thimble pattern in the lattice is different from that of a typical reactor, but is dispersed throughout the assembly. The burnable poison will not be in separate elements in the thimbles, but will be used in some fuel pins as an integral fuel burnable absorber (IFBA) material. All the control thimbles in all assemblies (except for a few near the core edge) will be associated with control elements, either control rods, gray rods, or water displacement rods (WDRs).

In spite of the differences between RESAR SP/90 and the Westinghouse reactors, geometric parameters are mostly bounded by previously analyzed geometries and physics characteristics are, for the most part, not changed significantly beyond those previously analyzed. Neutronics calculation methodology remains the same as for typical designs.

Fuel pin diameters are bounded by the 17x17 and 15x15 fuel-pin-array sizes, thus presenting no new lattice calculation problem. The larger water holes produced by the larger control thimbles will be comparable to those for Combustion Engineering reactor fuel for which Westinghouse has done reload analyses and has approved methodology for calculating power distributions (the primary problem associated with the larger holes). The fuel-to-water ratio (U/H) for end-of-cycle maximized moderator states (with the gray rods and WDRs out of the core) will be comparable to that for optimized fuel in a 17x17 array. The beginning of cycle U/H will be somewhat smaller than for any typical Westinghouse reactor, but this presents little problem for the methodology for calculating reactivity coefficients or power distributions of significance to safety. The particular form of IFBA has not been generally used in typical reactors, but the geometry and density of the absorber will be such that it does not present a difficult physics calculational problem. Standard methodologies for IFBA calculations have been checked against higher order methods and reactor tests have shown favorable comparability to these calculations. Furthermore, the IFBA will have been used in the Westinghouse VANTAGE 5 fuel before operation of RESAR SP/90 cores.

Thus, the RESAR SP/90 geometry and burnable poison material for the RESAR SP/90 design, while different in some details from typical Westinghouse cores, is suitably bounded by previously analyzed geometries or is sufficiently similar and presents no apparent analytical concerns, nor is it expected to present safety-related analysis problems. Therefore, the staff finds it acceptable.

Water Displacement Rods

The primary new PWR design concept for the RESAR SP/90 core is the moderator volume fraction variability over the operating cycle. This will be accomplished by inserting WDRs in over half of the control thimbles at beginning of cycle and removing them during the cycle. The WDRs will be nearly inert to neutrons and will act only to remove moderator from the core. This system will provide a "harder" neutron spectrum at beginning of cycle to produce more plutonium (via captures in the ^{238}U resonance energy region) and a "softer" spectrum at end of cycle to burn the plutonium, thus improving the cycle efficiency via the "spectral shift." It also will lower the amount of boron required in the moderator at beginning of cycle (given all other parameters are the same, e.g., burnable poison). Thus it will provide a more negative moderator temperature coefficient at beginning of cycle (averaged over the cycle, an improvement for problems with anticipated transients without scram [ATWS]) as well as lowering the neutronically inefficient parasitic capture by the boron. The WDRs will be totally removable (in a series of groups) part way through the cycle, or in smaller (symmetrical) groups at various times during the cycle.

There will be 88 WDR control rods (with typically 24 rodlets each) with a reactivity worth that will vary depending on reactor conditions (primarily the moderator boron content). The maximum worth of all WDRs will be about 2.7 percent Δk, which is considerably less than the total control rod worth (typically greater than 7 percent). Thus, there will be no unusual reactivity problem involved in withdrawal. The anticipated power distribution problems involved with withdrawal also are not novel and should present no problem. It is anticipated that withdrawal will occur at partial power levels, at least until experience with the system has been developed.

The nuclear design for the WDR provides improved cycle neutronic efficiencies, improved moderator coefficients, no new problems related to reactivity effects or power distribution, and is therefore acceptable.

Gray Rods

To improve power level change control (e.g., maintain power distribution within limits during rapid changes) late in cycle, the RESAR SP/90 design includes gray rods in addition to the usual control (and safety) rods and soluble boron in the moderator. The control rods will have eight normal B_4C Ag/In/Cd rodlets. The eight gray rodlets are similar to the WDR rodlets except they use a more absorbing cladding. There are 28 gray rods compared to 69 control (and safety) rods. Total reactivity worth of the gray rods will be about 0.5 percent Δk. These rods will be used (in a full-in or full-out state only) late in the cycle (when boron concentration is low) in place of adjustment to the boron concentration. As with recent Westinghouse reactors, coolant temperature variation from the normal (power level) program also will be used to provide flexibility to the load-change maneuvers. While the staff has not reviewed detailed procedures for the use of the gray rods, their use would appear to be feasible and is not expected to present unusual problems. Their reactivity strength is such that the safety analyses for postulated events involving rods are bounded by those for the control rods. Therefore, the staff finds their use acceptable.

Radial Reflector

The RESAR SP/90 is designed with a radial neutron reflector made of close packed steel rods in place of the standard baffle-form structure. The neutron reflector will reflect fast neutrons back into the core, which increases core reactivity by a small amount and thus improves the economics of the fuel cycle, slightly improves the radial power distribution, and decreases the fast neutron flux at the pressure vessel. The reflector and other changes in reactor geometry will result in about a 50 percent reduction in fast (above 1 million electronvolts [MeV]) neutron fluence at the pressure vessel compared to a typical reactor. Flux levels for other parts of the spectrum will remain about the same. The reduction may produce some lowering of flux levels at the ex-core neutron instrumentation locations, but it does not appear that it will affect their operability. Beyond this concern, there are no significant safety concerns with regard to the reflector; therefore, the staff finds it acceptable.

Neutron Detectors and N16 Detectors

The core power distribution monitoring and protection systems designed for the RESAR SP/90 are different than those for typical Westinghouse reactors. The RESAR SP/90 design uses the N16 power level and the four-segment ex-core neutron detector systems, which replace the delta coolant temperature power level system and the two-segment ex-core detectors used in many Westinghouse reactors. The staff is familiar with these systems because they were a part of the RESAR 414 design and were at least partially reviewed for that PDA. The staff reviewed and approved a form of the N16 system as part of the overpower and departure-from-nucleate-boiling-ratio (DNBR) protection system at Comanche Peak. The four-segment ex-core neutron detector system is a part of the Shearon Harris design. The staff reviewed and partially accepted this system in the RESAR 414 only as a monitoring system for axial power distribution because there was not sufficient information for an uncertainty analysis review of the accuracy of axial distribution monitoring. The N16 system is an improved substitute for the delta coolant temperature power level system and involves no significant change in operation. However, the four-segment ex-core detector system has the potential to provide a distinct improvement over the two-segment system with its ability to monitor the axial power distribution and to remove many of the operating restrictions entailed when using the constant axial offset control (CAOC) mode to maintain power distribution within peaking factor limits. RESAR SP/90 will use a form of CAOC with relaxed offset limits and no penalties for exceeding limits. The ex-core system will determine axial power distributions and warn an operator of operation near axially dependent peaking limits. Thus, it appears that it will not be difficult to maintain the design 2.60 total peaking factor limit with this system. The staff will review details of the uncertainty assignments during the FDA stage of review.

Core Physics

The reactor core nuclear parameters differ in detail from typical Westinghouse reactors because of the physical differences in the RESAR SP/90 design as discussed in this report. However, the effect of these parameter differences is, for the most part, not significant in the analyses of postulated events or provides more favorable results.

The average power density will be lower than typical Westinghouse cores, but the design total peaking factor will be larger, so the peak linear power density (kW/ft) assumed for most event analyses will be slightly larger (5 percent) than typical facilities. However, the peak linear power density for RESAR SP/90 will be smaller than that assumed for RESAR 414. The design radial peaking factor ($F_{\Delta H}$) is about the same as a typical Westinghouse core and, combined with the lower power density and longer core length, will result in about the same heat production in the peak pin as in a typical plant. Note that the heat production for RESAR SP/90 is less than that for the South Texas facility.

Control rod reactivity worths (e.g., total worth for scram calculations, bank worth for withdrawal events, and individual worth for rod ejection events and stuck rod events) are all within a similar range (varying with cycle and time in cycle) as those for typical Westinghouse reactors. Thus, events involving control rods will not be significantly different from those for typical reactors. The reactivity worths for the gray rods and WDRs will be less than for the control rods and thus will not alter this condition. The total control rod worths and requirements will be similar to those of typical reactors and shutdown margins will be similar or slightly larger than for other Westinghouse reactors.

The Doppler coefficient is affected by the fuel pin size, lattice spacing, and fuel/moderator ratio as affected by the gray rod and WDR positions. The latter causes a larger-than-normal coefficient range over the cycle. However, RESAR SP/90 coefficients generally fall within the range of typical Westinghouse reactors, considering the effects of standard and optimized 17x17 and 15x15 fuel geometries. Thus, Doppler dependent transient events are not significantly changed.

The moderator temperature coefficients differ a little from typical Westinghouse reactors because the RESAR SP/90 design incorporates WDRs. At beginning of cycle, the potentially lower moderator boron content will produce a more negative coefficient. The degree of additional safety provided by this feature will depend on the amount of burnable poison used in reload designs. End-of-cycle coefficients will be similar to, or only slightly more negative than, typical reactors because of increased plutonium production. The end-of-cycle fuel-to-moderator ratio is about the same as optimized fuel, and plutonium burnup results in plutonium discharge not dissimilar from normal. This will have only a small effect on the steam line break analysis and required shutdown margin for RESAR SP/90.

The design fuel discharge burnup (39,500 megawatt days/metric ton of uranium [MWD/MTU]) will be larger than for most present Westinghouse designs (about 33,000), but such burnup has become more standard in recent reload requests and presents no new physics analysis problem. The first and reload cycle for RESAR SP/90 are expected to be for 18 months with a fuel discharge burnup of approximately 13,000 MWD/MTU. This is not unusual compared to other reactors that have been reviewed. The kinetic parameters, i.e., beta and generation time, used in transient analyses are not different from normal. The methodology used in determining parameters and doing transient analyses are not different from those used for typical Westinghouse reactors.

The neutronic parameters and analyses methodology are either improved or are not significantly different from those approved for typical Westinghouse reactors. Therefore, the staff finds the parameters presented and methodology used for RESAR SP/90 core physical analyses acceptable.

With the exception of the differences that have been described above, the neutronic features and resulting review of the RESAR SP/90 design are essentially the same as for a typical Westinghouse reactor.

4.3.2 Design Bases (RESAR SP/90 Module 5, Section 4.3.2)

Design bases presented in Module 5, Section 4.3, of RESAR SP/90 comply with the applicable general design criteria. Acceptable fuel design limits are specified per the requirements of GDC 10, a negative prompt feedback coefficient is specified per the requirements of GDC 11, and the tendency toward divergent operation (power oscillation) is not permitted per the requirements of GDC 12. A control and monitoring system is provided per the requirements of GDC 13. This system will automatically initiate a rapid negative reactivity insertion to prevent exceeding fuel design limits in normal operation or anticipated transients per the requirements of GDC 20. The control system is designed so that a single malfunction or single operator error will cause no violation of fuel design limits per the requirements of GDC 25. A reactor coolant boration system is provided that is capable of bringing the reactor to cold shutdown conditions per the requirements of GDC 26, and a control system is provided to control reactivity changes during accident conditions when combined with the engineered safety features per the requirements of GDC 27. Reactivity insertion accident conditions will be limited so that no damage to the reactor coolant system pressure boundary occurs per the requirements of GDC 28.

The staff concludes that the design bases are acceptable.

4.3.3 Design Description (RESAR SP/90 Module 5, Section 4.3.2)

In RESAR SP/90, Westinghouse describes the first-cycle fuel loading, which will consist of three different enrichments and will have a first-cycle length of approximately one and a half years. The enrichment distribution, burnable poison distribution, soluble poison concentration and higher isotope (actinide) content as a function of core exposure are given. The staff concludes that the values for the delayed neutron fraction and prompt neutron lifetime at beginning and end of cycle are consistent with those normally used and are acceptable.

Power Distribution

The applicant's design bases affecting power distribution are given below:

- The design total peaking factor (F_Q) in the core will not be greater than 2.60 during normal operation at full power in order to meet the initial conditions assumed in accident analyses.
- Under normal conditions (including maximum overpower) the peak fuel power will not produce fuel centerline melting.
- The core will not operate during normal operation or anticipated operational occurrences with a power distribution that will cause the DNBR to fall below 1.17, using the WRB-2 correlation (design minimum nominal DNBR is 1.42 using "Improved Thermal Design Procedure," and a minimum of 1.58 is assumed in safety analyses).

The 2.60 F_Q will be determined and maintained via direct surveillance using an ex-core four-segment neutron detector system for axial power distributions and periodically measured radial power distributions and radial peaking factors (F_{xy} and $F_{\Delta H}$, respectively). These factors also provide conservative initial conditions for events described in Section 15, which ensure that the peak power does not cause centerline fuel melting or result in DNB during anticipated operational occurrences.

The applicant has described the manner in which the core will be operated and power distribution monitored so as to ensure that these limits are met.

Reactivity Coefficients

The reactivity coefficients are expressions of the effect on core reactivity of changes in core conditions such as power, fuel and moderator temperature, moderator density, and boron concentration. These coefficients vary with fuel burnup and power level. The applicant has presented values of the coefficients in RESAR SP/90 and has evaluated the uncertainties of these values. The staff reviewed the calculated values of reactivity coefficients and concludes that they adequately represent the full range of expected values.

The staff reviewed the reactivity coefficients used in the transient and accident analyses and concludes that they conservatively bound the expected values, including uncertainties. Further, moderator and power Doppler coefficients along with boron worth are measured as part of the startup physics testing to ensure that actual values are within those used in these analyses.

Control

A significant amount of excess reactivity will be built into the core to allow for changes in reactivity that result from reactor heatup, load following, and fuel burnup with consequent fission product buildup. The excess reactivity will be controlled by a combination of full-length control rods, gray rods, water displacement rods (WDRs), and soluble boron. Soluble boron is used to control changes caused by

- moderator density and temperature changes from ambient to operating temperatures
- equilibrium xenon and samarium buildup
- fuel depletion and fission product buildup (that portion not controlled by lumped burnable poison)
- transient xenon resulting from load following

Gray rods also will be used for these purposes late in cycle when the boron level is low.

Control rods will be used to control reactivity changes caused by

- moderator reactivity charges from hot zero to full power
- fuel temperature changes (Doppler reactivity changes)

WDRs will be used early in cycle to reduce the requirement for boron in the moderator and to improve cycle neutronic efficiency.

Burnable poison rods will be placed in some fuel assemblies to be used for radial flux shaping and to control part of the reactivity change resulting from fuel depletion and fission product buildup.

The applicant has provided data to show that adequate control exists to satisfy the above requirements with enough additional control rod worth to provide a hot shutdown effective multiplication factor less than the design-basis value during initial and equilibrium fuel cycles with the most reactive control rod stuck out the core. In addition, the chemical and volume control system will be capable of shutting down the reactor by adding soluble boron and maintaining it shut down in the cold, xenon free condition at any time in core life. These two systems satisfy the requirements of GDC 26.

Comparisons have been made between calculated and measured control rod bank worth in operating reactors and in critical experiments. These comparisons lead to the conclusion that bank worths may be calculated to within approximately 10 percent. In addition, bank worth measurements will be performed as part of the startup test program to ensure that conservative values have been used in safety analyses.

The staff concludes that the applicant has made suitably conservative assessments of reactivity control requirements and that adequate control rod worths have been provided to ensure shutdown capability.

Control Rod Patterns and Reactivity Worths

The control rods are divided into two categories: shutdown rods and regulating rods. The shutdown rods always will be completely out of the core when the reactor is at operating conditions. Core power changes will be made with regulating rods that are nearly out of the core when it is operating at full power. Regulating rod insertion will be controlled by power-dependent insertion limits required by technical specifications to ensure that

- There is sufficient negative reactivity available to permit rapid shutdown of the reactor with adequate margin.

- The worth of a control rod that might be ejected is not greater than that which has been shown to have acceptable consequences in the safety analyses. (This also applies to gray and WDRs.)

On the basis of its review, the staff concludes that rapid shutdown capability exists at all times in core life assuming the most reactive control rod assembly is stuck out of the core.

Stability

The stability of the core to xenon-induced spatial oscillations is discussed in the RESAR SP/90 application. The overall negative reactivity (power) coefficient provides assurance that the reactor will be stable against total power oscillation.

The applicant concluded that sustained radial or azimuthal xenon oscillations are not possible. This conclusion is based on calculations and measurements on operating reactors. The applicant has committed to perform tests on the first plant that uses the RESAR SP/90 design to confirm these conclusions. This core is predicted to be unstable with respect to axial xenon oscillations. However, the applicant has acceptably shown that axial oscillations may be controlled by the regulating rods to prevent reaching any fuel damage limits. The staff has reviewed Westinghouse's analyses and commitments in this matter and finds them acceptable.

Criticality of Fuel Assemblies

Criticality of fuel assemblies outside the reactor is precluded by adequate design of fuel transfer and storage facilities. The applicant presented information on calculational techniques and assumptions that it used to ensure that criticality is avoided. The staff reviewed this information and the criteria to be employed and finds them to be acceptable.

Vessel Irradiation

The applicant presented values for the neutron flux in various energy ranges at mid-height of the pressure vessel inner boundary. The applicant calculated core flux shapes by standard design methods used with a transport theory calculation (Sn) that results in a neutron flux of about 1.0 x 10^{10} neutrons per square centimeter per second having energy greater than 10^6 electronvolts at the inner vessel boundary. This results in a fluence of about 1.0 x 10^{19} neutrons per square centimeter for a 40-year vessel life with an 80-percent use factor. The methods used for these calculations are state of the art, and the staff concludes that acceptable analytical procedures have been used to calculate the vessel fluence.

4.3.4 Analytical Methods (RESAR SP/90 Module 5, Section 4.3.3)

The applicant has described the computer programs and calculational techniques used to obtain the nuclear characteristics of the reactor design. The calculations consist of three distinct types, which are performed in sequence: determination of effective fuel temperatures, generation of macroscopic few-group parameters, and space-dependent few-group diffusion calculations. The programs used (e.g., LASER, TWINKLE, LEOPARD, TURTLE, and PANDA) have been applied as part of the applications for other Westinghouse-designed nuclear plant facilities and the predicted results have been compared with measured characteristics obtained during startup tests for first cycle and reload cores. These results have validated the ability of these methods to predict experimental results. Therefore, the staff concludes that these methods are acceptable for use in calculating the nuclear characteristics of the RESAR SP/90 design.

4.3.5 Conclusions

The staff performed all areas of review and followed review procedures in accordance with SRP Section 4.3 either for the RESAR SP/90 reactor design, for previous comparable reactors, or for topical report reviews.

The applicant has described the computer programs and calculation techniques used to predict the nuclear characteristics of the reactor design and has provided examples to demonstrate the ability of the analyses to predict reactivity and physics characteristics.

To allow for changes of reactivity as a result of reactor heatup, changes in operating conditions, fuel burnup, and fission product buildup, a significant amount of excess reactivity is designed into the core. The applicant has provided substantial information relating to core reactivity balances for the first cycle and has shown that means have been incorporated into the design to control excess reactivity at all times. The applicant has shown that sufficient control rod worth is available to make the reactor subcritical with an effective multiplication factor within design limits in the hot condition at any time during the cycle with the most reactive control rod stuck in the fully withdrawn position. The staff concludes that the applicant's assessment of reactivity control requirements over the first core cycle is suitably conservative and that adequate negative worth has been provided by the control system to ensure shutdown capability. Reactivity control requirements will be reviewed for additional cycles as this information becomes available. The staff also concludes that nuclear design bases, features, and limits have been established in conformance with the requirements of GDC 10, 11, 12, 13, 20, 25, 26, 27, and 28.

This conclusion is based on the following:

(1) The applicant has met the requirements of GDC 11 with regard to prompt inherent nuclear feedback characteristics in the power operating range by (a) calculating a negative Doppler coefficient of reactivity and (b) using calculational methods that have been found acceptable. The staff has reviewed the Doppler reactivity coefficients in this case and found them to be suitably conservative.

(2) The applicant has met the requirements of GDC 12 with regard to power oscillations that could result in conditions exceeding specified acceptable fuel design limits by (a) showing that such power oscillations are not possible and/or can be easily detected and thereby remedied and (b) using calculational methods that have been found acceptable.

(3) The applicant has met the requirements of GDC 13 with regard to provisions of instrumentation and controls to monitor variables and systems that can affect the fission process by providing (a) instrumentation and systems to monitor the core power distribution, control rod positions and patterns, and other process variables such as temperature and pressure, and (b) suitable alarms and/or control room indications for these monitored variables.

(4) The applicant has met the requirements of GDC 26 with regard to provision for two independent reactivity control systems of different designs by having a system (a) that can reliably control anticipated operational occurrences, (b) that can hold the core subcritical under cold conditions, and (c) that can control planned, normal power changes.

(5) The applicant has met the requirements of GDC 27 with regard to reactivity control systems that have a combined capability in conjunction with poison addition by the emergency core cooling system of reliably controlling reactivity changes under postulated accident conditions by (a) providing a movable control rod system and a liquid poison system, and (b) performing calculations to demonstrate that the core has sufficient shutdown margin with the most reactive control rod stuck in the fully withdrawn position.

(6) The applicant has met the requirements of GDC 28 with regard to postulated reactivity accidents by (a) meeting the regulatory position in RG 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," (b) meeting the criteria on the capability to cool the core, and (c) using calculational methods that have been found acceptable for reactivity insertion accidents.

(7) The applicant has met the requirements of GDC 10, 20, and 25 with regard to specified acceptable fuel design limits by providing analyses demonstrating (a) that normal operation, including the effects of anticipated operational occurrences, have met fuel design criteria; (b) that the automatic initiation of the reactivity control system ensures that fuel design criteria are not exceeded as a result of anticipated operational occurrences and ensures the automatic operation of systems and components important to

safety under accident conditions; and (c) that no single malfunction of the reactivity control system causes violation of the fuel design limits.

4.4 Thermal-Hydraulic Design (RESAR SP/90 Module 5, Section 4.4)

The scope of the staff's review of the thermal-hydraulic design of the core for RESAR SP/90 included the design basis and the steady-state analysis of the core thermal-hydraulic performance. The acceptance criteria used as the basis of the staff's evaluation are set forth in SRP Section 4.4, "Thermal and Hydraulic Design." The review concentrated on the difference between the proposed design and those designs that have been previously reviewed and found acceptable by the staff.

4.4.1 Thermal-Hydraulic Design Bases (RESAR SP/90 Module 5, Section 4.4.1)

The principal thermal-hydraulic design basis for the RESAR SP/90 is the avoidance of thermal-hydraulic-induced fuel damage during normal steady state operation and anticipated operational transients. To satisfy the design basis, design analysis is performed and design limits are established based on the criteria addressed below.

Departure From Nucleate Boiling

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

In RESAR SP/90 Module 5, Section 4.4.1.1, Westinghouse presented the thermal-hydraulic design basis for the DNB as follows:

> There will be at least a 95 percent probability that departure from nucleate boiling (DNB) will not occur on the limiting fuel rods during normal operation, operational transients, or during transient conditions arising from faults of moderate frequency (ANS [American Nuclear Society] Condition I and II events), at a 95 percent confidence level. Historically this has been conservatively met by adhering to the following thermal design basis: there must be at least a 95 percent probability that the minimum departure from nucleate boiling ratio (DNBR) of the limiting power rod during ANS Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used. The DNBR limit for the correlation is established based on the variance of the correlation such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the calculated DNBR is at the DNBR limit.

Fuel Temperature

The fuel temperature basis is given in RESAR SP/90 Module 5, Section 4.4.1.2 as follows:

During modes of operation associated with ANS Condition I and ANS Condition II events, there is at least a 95 percent probability at the 95 percent confidence level that the peak kW/ft fuel rods will not exceed the UO_2 melting temperature. The melting temperature of unirradiated UO_2 is taken as 5080°F, and decreasing 58°F per 10,000 MWD/MTU. By precluding UO_2 melting, the fuel geometry is preserved and possible adverse effects of molten UO_2 on the cladding are eliminated. To preclude center melting and as a basis for overpower protection system setpoints, a calculated centerline fuel temperature of 4700°F has been selected as the overpower limit. This provides sufficient margin for uncertainties in the thermal evaluations as described in RESAR SP/90 Module 5, Section 4.4.2.9.1.

Core Flow Design Basis

A minimum of 93.5 percent of the thermal flow rate passes through the fuel rod region of the core and is effective for fuel rod cooling. Coolant flow through the thimble tubes as well as the leakage from the core barrel-baffle region into the core are not considered effective for heat removal.

Core cooling evaluations are based on the thermal flow rate (minimum flow) entering the reactor vessel. A maximum of 6.5 percent of this value is allowed as bypass flow. This includes rod cluster control guide thimble cooling flow, head cooling flow, baffle leakage, and leakage to the vessel outlet nozzle.

Hydrodynamic Stability

The hydrodynamic stability design basis for RESAR SP/90 is as follows:

"Modes of operation associated with ANS Condition I and II events shall not lead to hydrodynamic instability."

4.4.2 Thermal-Hydraulic Design Methodology (RESAR SP/90 Module 5, Section 4.4.2)

Thermal-Hydraulic Comparison

The thermal-hydraulic design of RESAR SP/90 is not similar to any operating units. It is similar to the RESAR 414 design, which has been previously reviewed and approved by the staff but not used in any actual reactor units. Table 4.4-1 of the RESAR provides a comparison of the design parameters between RESAR SP/90 and RESAR 414.

The fundamental differences in core geometry between RESAR SP/90 and RESAR 414 are the size and number of thimbles and fuel rods. These changes result in a net increase in heat transfer surface area. In addition, a typical Inconel R-grid design is changed to a Zircaloy design having similar mixing characteristics and the enthalpy hot channel factor is increased in the RESAR SP/90 design. The combined effects of the above changes on DNBR result in similar limiting DNBRs at nominal operating conditions for the same average core exit temperature.

Departure From Nucleate Boiling

DNBRs are calculated using the WRB-2 critical heat flux (CHF) correlation. The coupled THINC-IV/THINC I computer code is used to determine the flow distribution in the core and the local conditions in the hot channel for use in the DNB correlation. The applicant has provided data of its CHF tests which model the RESAR SP/90 fuel assembly that demonstrated the CHF characteristics of the RESAR SP/90 fuel assembly can be adequately described by the WRB-2 correlation with a DNBR design criterion of 1.17. The staff agrees with the applicant's assessment.

4.4.3 Design Abnormalities

Fuel Rod Bowing

The phenomenon of fuel rod bowing, as described in WCAP-8691, Revision 1 (which has been reviewed and approved by the staff), must be accounted for in the DNBR safety analysis of Condition I and Condition II events for each plant application. Applicable generic credits for margin resulting from the evaluation of DNBR and/or margin obtained from measured plant operating parameters (such as $F^N_{\Delta H}$ or core flow) that are less limiting than those required by the plant safety analysis, are used to offset the effect of rod bow.

The safety analysis for the RESAR SP/90 core maintained sufficient margin between the safety analysis limit DNBR (1.58 for thimble and typical cells) and the design limit DNBR (1.42 for thimble and typical cells) to accommodate full-flow and low-flow DNBR penalties identified in WCAP-8691 for a 17x17 fuel assembly. The amount of fuel rod bow and its associated DNBR penalties is predicted to be less for RESAR SP/90 fuel assemblies than that for Westinghouse 17x17 fuel assemblies because RESAR SP/90 fuel has a larger fuel rod diameter, thicker cladding, and smaller spacing between grids. This evaluation is based on the application of rod bow scaling factors given in Appendix C and D of the NRC-approved Westinghouse rod bow topical report (WCAP-8691).

The maximum rod bow penalties accounted for in the design safety analysis are based on a region average burnup of 33,000 MWD/MTU. At burnups greater than 33,000 MWD/MTU, credit is taken for the effect of $F^N_{\Delta H}$ burndown, as a result of the decrease in fissionable isotopes and the buildup of fission product inventory, and no additional rod bow penalty is required.

The staff concludes that rod bow penalties have been properly offset by the DNBR margins calculated by Westinghouse.

Crud Deposition and Flow Uncertainty

Crud deposition in the core and the associated change in core pressure drop will be detected by flow measurement as described in the RESAR SP/90 application. Therefore, the technical specifications for a plant-specific RESAR SP/90 facility must require that the reactor coolant system flow be monitored at least every 24 hours.

The thermal design flow for RESAR SP/90 is defined as 96.8 percent of the best-estimate flow. The procedure for verifying this value must be included in plant-specific technical specifications and tests must be made of the primary system before initial criticality to verify that a conservative primary system coolant flow rate has been used in the design and analyses of the plant.

Hydrodynamic Stability

In steady-state, two-phase, heated flow in parallel channels, the potential for hydrodynamic instability exists. The applicant provided the following information to support the contention that the RESAR SP/90 core is thermohydraulically stable.

Boiling flows may be susceptible to thermohydrodynamic instabilities. These instabilities are undesirable in reactors because they may cause a change in thermohydraulic conditions that may lead to a reduction in the DNB heat flux relative to that observed during a steady flow condition or to undesired forced vibrations of core components. Therefore, a thermohydraulic design criterion was developed to state that modes of operation under Condition I and II events shall not lead to thermohydrodynamic instabilities.

Two specific types of flow instabilities are considered for Westinghouse PWR operation. These are the Ledinegg type, or flow-excursion type, of static instability and the density-wave type of dynamic instability.

A Ledinegg instability involves a sudden change in flow rate from one steady state to another. This instability occurs when the slope of the reactor coolant system pressure drop-flow rate curve ($\partial\Delta P/\partial G_{internal}$) becomes algebraically smaller than the loop supply (pump head) pressure drop-flow rate curve ($\partial\Delta P/\partial G_{external}$). The criterion for stability is thus $\partial\Delta P/\partial G_{internal} > \partial\Delta P/\partial G_{external}$. The Westinghouse pump head curve has a negative slope ($\partial\Delta P/\partial G_{external} < 0$) whereas the reactor coolant system pressure drop-flow curve has a positive slope ($\delta\Delta P/\partial G_{internal} > 0$) over the Condition I and Condition II operational ranges. Thus, the Ledinegg instability will not occur.

The mechanism of density wave oscillations in a heated channel has been described by Lahey and Moody (ANS, 1977). Briefly, an inlet flow fluctuation produces an enthalpy perturbation. This perturbs the length and the pressure drop of the single-phase region and causes quality or void perturbations in the two-phase regions that travel up the channel with the flow. The quality and length perturbations in the two-phase regions create two-phase pressure drop perturbations. However, since the total pressure drop across the core is maintained by the characteristics of the fluid system external to the core, then the two-phase pressure drop perturbation feeds back to the single-phase region. These resulting perturbations can be either attenuated or self-sustained.

A simple method has been developed by Ishii et al. (November 1976) for parallel closed channel systems to evaluate whether a given condition is stable with respect to the density-wave type of dynamic instability. This method had been used to assess the stability of typical Westinghouse reactor designs under Condition I and II operation. The results indicate that a large margin to density-wave instability exists; e.g., increases on the order of 200 percent of rated reactor power would be required for the predicted inception of this type of instability.

The application of the method of Ishii to Westinghouse reactor designs is conservative as a result of the parallel open channel feature of Westinghouse PWR cores. For such cores, there is little resistance to lateral flow leaving the flow channels of high-power density. There also is energy transfer from channels of high power density to lower power density channels. This coupling with cooler channels has led to the opinion that an open channel configuration is more stable than the above closed channel analysis under the same boundary conditions. Flow stability tests have been conducted where the closed channel systems were shown to be less stable than when the same channels were cross connected at several locations. The cross connections were such that the resistance to channel-to-channel cross flow and enthalpy perturbations would be greater than that which would exist in a PWR core that has a relatively low resistance to cross flow.

Flow instabilities that have been observed have occurred almost exclusively in closed channel systems operating at low pressure relative to the Westinghouse PWR operating pressures. Kao, Morgan, and Parker (ANS, 1973) analyzed parallel closed channel stability experiments simulating a reactor core flow. These experiments were conducted at pressures up to 2200 psia. The results showed that for flow and power levels typical of power reactor conditions, no flow oscillations could be induced above 1200 psia.

The rod bundle DNB tests provided additional evidence that flow instabilities do not adversely affect thermal margin. Many Westinghouse rod bundles have been tested over wide ranges of operating conditions with no evidence of premature DNB or of inconsistent data that might be indicative of flow instabilities in the rod bundle.

Westinghouse concluded that thermohydrodynamic instabilities will not occur under Condition I and II modes of operation for Westinghouse PWR designs. A large power margin, greater than doubling rated power, exists to predicted inception of such instabilities. Analysis shows that minor plant-to-plant differences in Westinghouse reactor designs such as fuel assembly arrays, core power to flow ratios, and fuel assembly length, will not result in cross deterioration of the above power margins.

The staff concludes that past operating experience, flow stability experiments, and the inherent thermohydraulic characteristics of Westinghouse PWRs provide a basis for accepting the stability evaluation for RESAR SP/90 in support of the PDA.

4.4.4 Operating Abnormalities

Loose Parts Monitoring Systems

In the RESAR SP/90 application, the applicant provided documentation of its loose-parts monitoring system, which utilizes the Westinghouse digital metal impact monitoring system (DMIMS) and is the same system previously reviewed and approved by the staff for the Virgil Summer and Shearon Harris plants. Although the DMIMS is not a Class 1E system, equipment inside containment is designed to remain functional during an operating-basis earthquake and anticipated radiation exposures. Data base channel checks and functional tests are incorporated in the DMIMS designs.

Operators will be trained in the operation and maintenance of the DMIMS before a plant-specific startup. In the RESAR SP/90 application, Westinghouse states that the overall design of its DMIMS is in conformance with RG 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," Revision 1. This is acceptable to the staff.

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The RESAR SP/90 application, describes the inadequate core cooling instrumentation, which is based on the generically approved Westinghouse-designed reactor vessel level instrumentation system (RVLIS). The RESAR does not give the item-by-item documentation required by NUREG-0737, Item II.F.2. This documentation must be provided on a plant-specific basis by an applicant referencing the RESAR SP/90 design.

4.4.5 Conclusion

The staff concludes that the thermohydraulic design of the initial core of RESAR SP/90 is contingent on satisfactory resolution of the items discussed above.

4.5 Reactor Materials (RESAR SP/90 Module 5, Section 4.5)

4.5.1 Control Rod Drive Structural Materials (RESAR SP/90 Module 5, Section 4.5.1)

The staff evaluated the structural materials selected for the control rod drive mechanism against the requirements of General Design Criteria (GDC) 1, 14, and 26 and 10 CFR 50.55a. The applicant has demonstrated (1) that the properties of materials selected for the control rod drive mechanism components that will be exposed to the reactor coolant satisfy Parts A, B, and C of Section II and Appendix I of Section III of the ASME Code and Regulatory Guide 1.85, "Code Case Acceptability ASME Section III Materials," and (2) that the yield strength of cold-worked austenitic stainless steels does not exceed 90,000 psi. Conformance with the recommendations of Regulatory Guide 1.85 is addressed in Section 5.2.1.2 of Module 4 of the RESAR SP/90 application.

The controls imposed upon the ferrite content of austenitic stainless-steel filler materials satisfy most of the recommendations of Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal." The applicant's alternative approach of using chemical analysis of the weld metal deposit to determine ferrite content is acceptable to the staff.

The controls imposed on austenitic stainless steels to reduce sensitization satisfy, to the extent practical, the recommendations of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel." The staff finds the applicant's justification for waiving testing to show nonsensitization of fittings that do not have inaccessible cavities or chambers to preclude rapid cooling when water is quenched or sprayed acceptable. The applicant confirmed that the tempering and aging temperatures of heat treatable materials in the control rod drive mechanism are specified to eliminate the susceptibility to stress corrosion cracking in the reactor coolant. The fabrication and heat treatment practices that will be performed provide assurance that stress corrosion cracking will not occur during the design life of the components. The compatibility of all materials used in the control rod system in contact with the reactor coolant satisfies the criteria of Articles NB-2160 and NB-3120 of Section III of the ASME Code. Cleaning and cleanliness controls are in accordance, to the extent practical, with ANSI Standard N 45.2.1-1973, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants," and Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."

The staff concludes that the structural materials selected for the control rod drive mechanism are generally acceptable and meet the requirements of GDC 1, 14, and 26 and 10 CFR 50.55a.

4.5.2 Reactor Internals and Core Support Materials (RESAR SP/90 Module 5, Section 4.5.2)

The design, fabrication, and testing of the materials to be used in the reactor internals and core support structure will meet GDC 1 and 10 CFR 50.55a and, therefore, will be adequate for structural integrity. The controls imposed on components to be constructed of austenitic stainless steel satisfy, to the extent practical, Regulatory Guides 1.31 and 1.44. Where the recommendations of those regulatory guides were not followed, the alternative approaches have been reviewed by the staff and are acceptable (see Section 4.5.1 of this SER).

The materials to be used for the construction of components of the reactor internals and core support structure will conform with the applicable ASME Code. Conformance with Regulatory Guide 1.85 is discussed in Section 4.5.1. As proven by extensive tests and satisfactory performance, the specified materials are compatible with the expected environment and corrosion is expected to be negligible. The controls imposed on the reactor coolant chemistry will provide reasonable assurance that the reactor internals and core support structure will be protected adequately during operation from conditions that could lead to stress corrosion of the materials and loss of component structural integrity.

The material selection, fabrication practices, examination and testing procedures, and control practices performed in accordance with these recommendations provide reasonable assurance that the materials used for the reactor internals and core support structure are in a metallurgical condition to preclude inservice deterioration. Conformance with the ASME Code and the regulatory guides constitutes an acceptable basis for meeting, in part, GDC 1 and 10 CFR 50.55a.

The staff concludes that the materials to be used for the construction of the reactor internals and core support structure are acceptable and meet GDC 1 and 10 CFR 50.55a.

4.6 Functional Design of Reactivity Control Systems (RESAR SP/90 Module 5, Section 4.6)

The staff reviewed the reactivity control systems in accordance with SRP Section 4.6 and performed an audit review of each of the areas listed in the "Areas of Review" portion of the SRP section according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for the staff's evaluation of the reactivity control systems with regard to the applicable regulations of 10 CFR 50.

The control rod drive system (CRDS), the safety injection system (SIS), and the chemical and volume control system (CVCS) constitute the reactivity control systems.

The CRDS will consist of control rod drive mechanisms to which the rod cluster control assemblies will be attached. The control rod drive mechanisms are discussed and evaluated in Section 3.9.4 of this SER. The rod cluster control assemblies are divided into two categories: control and gray rod assemblies.

The control category of rod cluster control assemblies are designed to be automatically inserted or withdrawn to compensate for changes in reactivity associated with power-level changes and power distribution, variations in moderator temperature, or changes in boron concentration. The gray rod cluster control assemblies, which will be fully withdrawn during power operations, will be used solely to insert large amounts of negative reactivity to shut down the reactor. Refer to Section 4.3 of this SER for further discussion on these features.

The rod cluster control assemblies will be the primary shutdown mechanism for normal operation, accidents, and transients. They will insert automatically upon receipt of a reactor trip signal, thereby complying with the requirements of GDC 29.

Failure of electrical power to a rod cluster control assembly will result in the insertion of that assembly, as will shearing of the connection between the rod cluster control assembly and control rod drive mechanism. Single failure of a rod cluster control assembly is considered in the transient and accident analyses and include analysis of the most reactive rod cluster assembly stuck out of the core. The Westinghouse analysis of accidental withdrawal of a rod cluster control assembly resulted in acceptable consequences. This conforms to the requirements of GDC 23 and 25. Refer to Sections 4.3 and 15 of this SER for further evaluation.

The SIS is automatically actuated to inject borated water into the reactor coolant system when a safety injection signal is received. The SIS pumps take suction from the emergency water storage tank. The SIS is discussed further in Section 6.3 of this SER.

The CVCS is primarily designed to accommodate slow or long-term reactivity changes such as those caused by fuel burnup or by variation in the xenon concentration resulting from changes in power level. The CVCS will be used to control reactivity by adjusting the dissolved boron concentration in the reactor coolant system. The boron concentration will be controlled (1) to obtain optimum rod cluster control assembly positioning; (2) to compensate for reactivity changes associated with variations in coolant temperature, core burnup, and xenon concentration; and (3) to provide shutdown margin for maintenance and refueling operations. No credit is taken for the boration capabilities of the CVCS in the analyses of transients (evaluated in Section 15 of this SER). Module 13, "Auxiliary Systems," of RESAR SP/90 provides information on the capabilities and system description of the CVCS. RESAR SP/90 Module 5, Section 4.6.4 states that "prior proper operation of the CVCS, has been presumed as an initial condition to evaluate transients and appropriate technical specifications will be prepared to ensure current operation or remedial actions." On the basis of this statement, the staff finds this portion of the reactivity control system acceptable.

Soluble poison concentration will be used to control slow operating reactivity changes. If necessary, a rod cluster control assembly movement also can be used to accommodate such changes, but assembly insertion will be used mainly to control anticipated operational occurrences even with a single malfunction, such as a stuck rod. In either case, fuel design limits will not be exceeded. The soluble poison control will be capable of maintaining the core subcritical under cold shutdown conditions. This conforms to the requirements of GDC 26.

The reactivity control systems, including the addition of concentrated boric acid solution by the SIS, will be capable of controlling all anticipated operational changes, transients, and accidents. All accidents are calculated with the assumption that the most reactive rod cluster control assembly is stuck out and cannot be inserted, which complies with the requirements of GDC 27. Compliance with the requirements of GDC 28 is discussed in Sections 4.3 and 15 of this SER.

The staff concludes that the reactivity control system functional design meets the requirements of GDC 23, 25, 26, 27, and 29 with regard to its fail-safe design, malfunction protection design, redundancy and capability, combined systems capability, reactivity limits, and protection against anticipated operational occurrences; therefore, it is acceptable. The control rod drive system meets the acceptance criteria of SRP Section 4.6 and is acceptable.

5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS (RESAR SP/90 Module 4, Section 5.0)

5.1 Summary Description (RESAR SP/90 Module 4, Section 5.1)

The reactor coolant system (RCS) design consists of four similar heat transfer loops connected in parallel to the reactor vessel. Each loop contains a reactor coolant pump, steam generator, and associated piping and valves. In addition, the system includes a pressurizer, pressurizer relief and safety valves, interconnecting piping, and the instrumentation necessary for operational control. All the above components will be located in the containment building.

During operation, the RCS 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 RCS at a flow rate and temperature consistent with achieving the reactor core thermohydraulic performance. The water also will act as a neutron moderator and reflector and as a solvent for the neutron absorber used for chemical shim control.

The RCS pressure boundary provides a barrier against the release of radioactivity generated within the reactor and is designed to ensure a high degree of integrity throughout the life of the plant.

The RCS pressure will be controlled by the use of the pressurizer; water and steam will be maintained at saturation conditions by electrical heaters and water sprays. Steam can be formed (by the heaters) or condensed (by the pressurizer spray) to minimize pressure variations as a result of contraction and expansion of the reactor coolant volume. Spring-loaded safety valves and power-operated relief valves connected to the pressurizer will provide for steam discharge from the RCS. Discharged steam will be piped to the pressurizer relief tank (pressurizer relief discharge system) where the steam will be condensed and cooled by mixing with water.

5.2 Integrity of Reactor Coolant Pressure Boundary (RESAR SP/90 Module 4, Section 5.2)

5.2.1 Compliance With Codes and Code Cases (RESAR SP/90 Module 4, Section 5.2.1)

5.2.1.1 Compliance With 10 CFR 50.55a (RESAR SP/90 Module 4, Section 5.2.1.1)

10 CFR 50.55a, "Codes and Standards," requires the following for components important to safety:

- Components in the reactor coolant pressure boundary (RCPB) must meet the requirements for Class 1 (Quality Group A) components in the ASME Code, Section III, except for those components which meet the exclusion requirements of 10 CFR 50.55a(c)(2).

- Components classified as Quality Group B and C must meet the requirements for Class 2 and 3 components, respectively, in the ASME Code, Section III.

The pressure-retaining components of the RCPB as defined by 10 CFR 50.55a, "Codes and Standards," have been properly classified in RESAR SP/90 Table 3.2-1 as ASME Code, Section III, Class 1 components. These Class 1 components are designated Quality Group A in conformance with Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radio-Waste-Containing Components of Nuclear Power Plants." The Quality Group A RCPB components were reviewed in accordance with SRP Section 5.2.1.1. Other pressure-retaining components, such as those constructed to ASME Code, Section III, Class 2 and Class 3, are reviewed in Section 3.2.2 of this SER.

SRP Section 5.2.1.1 recommends that safety analysis reports for both construction permits and operating licenses contain a table identifying the ASME component code, code edition, applicable addenda and, when required, the component order date of all ASME Code, Section III, Class 1 and 2 pressure vessel components, piping, pumps, and valves in the RCPB. The staff reviews this table for compliance with 10 CFR 50.55a. In Subsection 5.2.1.1 of Module 4 to the RESAR SP/90 application, the applicant states that this information "will be determined in connection with the specific plant application." The staff has concluded that this is an acceptable commitment for the RESAR SP/90 application because the staff finds it more desirable for an applicant referencing this design to use the latest ASME Code edition and addenda that would be applicable at the time of application for a construction permit. This information will have to be provided on a plant-specific basis at the time of application for a construction permit that references the RESAR SP/90 design. Receipt of acceptable information during the construction permit phase of the staff's review will constitute an acceptable basis for satisfying GDC 1 on a plant-specific basis.

5.2.1.2 Applicable Code Cases (RESAR SP/90 Module 4, Section 5.2.1.2)

In accordance with the guidelines in SRP Section 5.2.1.2, the staff's review of this section of the RESAR SP/90 application involves a determination of the acceptability of ASME Code and ANSI code case interpretations. These code cases contain requirements or special rules that may be used for application in the construction of ASME Code Class 1, 2, and 3 components for light-water-cooled nuclear power plants. To be acceptable, all of the code cases that will be used in the construction of RESAR SP/90 must be approved in one of the following regulatory guides:

(1) Regulatory Guide 1.84, "Code Case Acceptability in ASME Section III - Design and Fabrication." This guide lists those Section III ASME code cases oriented to design and fabrication that are acceptable to the staff for implementation in the licensing of nuclear power plants.

(2) Regulatory Guide 1.85, "Code Case Acceptability in ASME Section III - Materials." This guide lists those Section III ASME code cases oriented to materials and testing which are acceptable to the staff for implementation in the licensing of nuclear power plants.

(3) Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1." This guide lists those Section XI ASME code cases that are acceptable to the staff for use in the inservice inspection of light-water-cooled nuclear power plants.

Compliance with one or more of the above regulatory guides meets the requirements of GDC 1 and applicable requirements of 10 CFR 50.55a.

In Section 5.2.1.2 of Module 4 to the RESAR SP/90 application, the applicant has provided the following commitments:

(1) Westinghouse controls its suppliers to

- Limit the use of code cases to those listed in Regulatory Position C.1 of the revisions of Regulatory Guides 1.84 and 1.85 in effect at the time the equipment is ordered, except as allowed in item 2 below.
- Identify and request permission for use of any code cases not listed in Regulatory Position C.1 of the revisions of Regulatory Guides 1.84 and 1.85 in effect at the time the equipment is ordered, where use of such code cases is needed by the supplier.
- Permit continued use of a code case considered acceptable at the time of equipment order, where such code case was subsequently annulled or amended.

(2) Westinghouse will seek NRC permission for the use of code cases needed by suppliers and not yet endorsed in Regulatory Position C.1 of the revisions of Regulatory Guides 1.84 and 1.85 in effect at the time the equipment is ordered. Suppliers may use these code cases only if NRC permission is obtained or is otherwise ensured (e.g., a later version of the regulatory guide which includes endorsement of the code case).

Additionally Westinghouse, in their response to open items dated May 1989, has provided the following clarification:

> "When conditionally approved code cases are used in the construction of ASME Section III Class 1, 2, 3, MC, NF, and NG components, the additional conditions identified in Regulatory Guides 1.84 (Design and Fabrication), 1.85 (Materials) and 1.147 (Inservice Inspection) as applicable to these conditionally approved code cases will be complied with."

The staff concludes that the above commitments at the RESAR SP/90 PDA stage of review meet the guidelines of SRP Section 5.2.1.2, which includes satisfying the applicable requirements of GDC 1 and 10 CFR 50.55a, and are acceptable. During the FDA review of RESAR SP/90, the staff will require that the applicant identify all code cases that will be used in the construction of all RESAR SP/90 components of ASME Classes 1, 2, and 3.

5.2.2 Overpressure Protection (RESAR SP/90 Module 4, Section 5.2.2)

The staff has reviewed the design for overpressure protection for RESAR SP/90 in accordance with SRP Section 5.2.2. Overpressure protection for the RCPB is provided by pressurizer safety valves, the steam generator safety valves, and the liquid relief valves of the residual heat removal system (RHRS), in combination with the action of the reactor protection system and the guidance in the plant operating procedures. The combination of these features provides overpressure protection as required by GDC 15, the ASME Code, Section III, and 10 CFR 50, Appendix G. These requirements ensure RCPB overpressure protection for power operation and low-temperature operation (startup and shutdown). Both modes of overpressure protection are discussed below.

5.2.2.1 Overpressure Protection During Power Operation (RESAR SP/90 Module 4, Sections 5.2.2.1 - 5.2.2.9)

For power operation, the pressurizer power-operated relief valves (PORVs) will be sized to limit the pressurizer pressure to a value below the high-pressure trip set point for all design load-reduction transients up to and including a full-load rejection with steam dump actuation and automatic reactor control. The PORVs also will limit challenges to the pressurizer safety valves. However, in analyzing operational transients and faulted conditions with regard to the concern of RCS overpressurization, credit is taken only for pressurizer safety valves.

There are three pressurizer safety valves provided in the RESAR SP/90 design. Each safety valve will be spring loaded and have a relieving capacity of 500,000 pounds per hour of saturated steam at 2485 psig. The combined capacity of the pressurizer safety valves can accommodate the maximum pressurizer surge and prevent the RCS pressure at the point of the highest pressure in the system from exceeding the limit of 110 percent of design pressure as stipulated in ASME Code, Section III. The maximum pressurizer surge rate is based on the worst RCS pressure transient, identified to be a complete loss of steam flow to the turbine (complete loss of load) with the reactor operating at 102 percent of engineered safeguards design power. In this analysis, feedwater flow also is assumed to be lost and no credit is taken for operation of the pressurizer PORVs, pressurizer level control system, rod control system, condenser steam dump system, or main steam line atmospheric steam dump valves. The following was provided by Westinghouse in their response to open items dated May 1989 to clarify the above transient: "The reactor is maintained at full power, with no credit being taken for either the reactor trip on turbine trip or the first safety grade reactor trip. Credit is taken for steam relief through the steam generator safety valves."

The pressurizer safety valves will discharge steam to the pressurizer relief tank through a common manifold. Each pressurizer safety valve discharge line will incorporate a control board temperature indicator and alarm to notify the operator of steam discharge as a result of either leakage or actual valve operation.

Overpressure protection for the main steam system will be provided by steam generator safety valves. The main steam system safety valve capacity is based on providing enough relief to remove the engineered safeguards design steam flow while limiting the maximum main steam system pressure to less than 110 percent of the steam generator shell side design pressure.

The safety valves will be designed in accordance with ASME Code, Section III, and periodic testing and inspection will be performed in accordance with ASME Code, Section XI, to the extent practical and comply with all applicable portions of 10 CFR 50.55a(g).

The staff concludes that the design criteria of the overpressure protection system for RESAR SP/90 at-power operating conditions comply with the guidelines of SRP Section 5.2.2 and the requirements of GDC 15.

5.2.2.2 Overpressure Protection During Low-Temperature Operation (RESAR SP/90 Module 4, Section 5.2.2.10)

SRP Section 5.2.2 requires that the system for overpressure protection during low-temperature operation of the plant shall be designed in accordance with the requirements of BTP RSB 5-2.

The low-temperature overpressure protection (LTOP) system for the RESAR SP/90 design is provided by the relief valves in the RHRS. An RHRS relief valve is provided in each of the four RHR pump suction lines. During plant cooldown, all four RHR subsystems normally will be aligned when the RCS pressure and temperature are below 400 psig and 350°F, respectively. Once aligned, the RHR pump suction isolation valves will be deenergized in their open position. The results of the Westinghouse analyses, based on the maximum mass addition and heat input transients, have shown that two RHR relief valves are sufficient to prevent the limits, provided in 10 CFR 50, Appendix G, from being exceeded as the result of the initiation of all credible mass or heat input transients during low-temperature operations.

RESAR SP/90 Section 5.2.2.10 does not provide design criteria to ensure that the RHR relief valves will be made available for overpressure protection during plant startup. The staff requires that Westinghouse provide design criteria to ensure that the necessary overpressure protection of the RCS will be provided during all low-temperature operating conditions when overpressurization is a concern. These design criteria should include the transients that were considered for the design of LTOP systems. The staff will review the criteria when submitted for the next licensing stage.

In RESAR SP/90 Section 5.2.2.10.1, Westinghouse stated that throughout refueling/shutdown, at least two of four RHR subsystems will be aligned at all times. This will enable pump maintenance and testing, as required on the remaining two of four subsystems. However, under this provision, the RESAR SP/90 design may not have two RHR relief valves available for overpressure protection during a single failure of an RHR isolation valve to close. In response to this concern, Westinghouse provided clarification in its response to open items dated May 1989. Westinghouse stated that the commitment to remove power from the open RHR suction isolation valves ensures that no single failure can cause any of these valves to inadvertently close and result in an RHR relief valve being unavailable for overpressure protection. Also an alarm will be provided to alert the operator when LTOP is required to ensure that at least two RHR subsystems are aligned and that all four valves in at least two RHR subsystems remain open. Furthermore, plant procedures will state that maintenance during plant shutdown should be limited to one of four RHR subsystems when overpressure protection is required.

Although the LTOP system is designed to maintain RCS pressure within allowable limits, system interlocks and administrative controls will be provided to minimize the potential for any transient that could actuate the low-temperature overpressure initiation system. The integrated safeguards system will be interlocked so that, before a subsystem is aligned for the RHR function, the high-head safety injection pump discharge isolation valve must be closed. The normal plant operating procedures will maximize the use of a pressurizer cushion (steam bubble) during periods of low-pressure, low-temperature operation. This cushion will dampen the plant's response to potential transient generating inputs, thereby providing easier pressure control with the slower response rates. This provides reasonable assurance that most potential transients can be terminated by operator action before the RHR relief valves actuate.

With regard to emergency core cooling system (ECCS) alignment, the RESAR SP/90 application provides the following administrative controls to prevent low-temperature overpressurization transients:

> To preclude inadvertent ECCS actuation during heatup and cooldown, blocking of the safety injection signal actuation logic below 1975 psia will be required.
>
> During RCS cooldown, closure and power lockout of the accumulator isolation valves, as well as power lockout to all the high-head safety injection pumps and to nonoperating charging pumps, will be required.

These actions are to be taken when RCS pressure is approximately 1000 psig, shortly before RHR operation begins.

In response to the staff request for additional information regarding the consequences of a potential LOCA during shutdown mode without ECCS flow, the applicant stated in a letter dated May 13, 1988, that the initiation of a LOCA in modes 3 and 4 and blockage of the safety injection signal actuation logic to prevent inadvertent ECCS initiation is currently being investigated generically for Westinghouse-designed plants. Upon completion of this generic investigation, the applicability of the generic conclusions relative to a LOCA in modes 3 and 4 will be applied to the RESAR SP/90 design during the FDA stage. The staff finds this Westinghouse commitment acceptable. The staff will evaluate the resolution of this issue during review of the FDA application.

With the exception of the unresolved items identified above, the staff concludes that the low-temperature overpressure protection system of RESAR SP/90 design meets GDC 15 and 31 and 10 CFR 50 Appendix G because it has implemented the guidelines of BTP RSB 5-2. Therefore, pending satisfactory resolution of the above issues, the staff finds the LTOP system acceptable.

5.2.3 Reactor Coolant Pressure Boundary Materials (RESAR SP/90 Module 4, Section 5.2.3)

The staff evaluated the materials specified in the RESAR SP/90 application for the reactor coolant pressure boundary (RCPB) against the requirements of GDC 1, 4, 14, 30, and 31 of Appendix A and Appendices B and G to 10 CFR 50 and the requirements of 10 CFR 50.55a.

The materials to be used for construction of components of the RCPB have been identified by specification and found to be in conformance with the requirements of Section III of the ASME Code. Compliance with the Code provision for materials specifications satisfies the quality standards requirements of GDC 1 and 30 and 10 CFR 55.55a.

The materials to be used for construction of the RCPB that is exposed to the reactor coolant have been identified and all of the materials are compatible with the primary coolant water, which will be chemically controlled in accordance with appropriate technical specifications. This compatibility has been proven by extensive testing with satisfactory performance. Conforming to the recommendations of Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless

Steel," or the alternative approaches taken by the applicant are acceptable to the staff (see Section 4.5.1 of this SER).

General corrosion of all materials in contact with reactor coolant is determined to be negligible and, accordingly, general corrosion is not a concern. Compatibility of the materials with the coolant and compliance with the Code provisions satisfy the requirements of GDC 4 with regard to compatibility of components with environmental conditions.

The applicant has indicated that the materials of construction for the RCPB will be compatible with the thermal insulation used in these areas. The thermal insulation used on the RCPB will be either the reflective stainless-steel type or will be made of nonmetallic compounded materials. Nonmetallic component thermal insulation should be in conformance with the recommendations of Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steels." Conformance with these recommendations will satisfy the requirements of GDC 14 and 31 with regard to prevention of failure of the RCPB.

The ferritic steel tubular products and the tubular products fabricated from austenitic stainless steel have been found to be acceptable by nondestructive examinations in accordance with provisions of the ASME Code, Section III. Compliance with these Code requirements satisfies the quality standards requirements of GDC 1 and 30 and 10 CFR 50.55a.

The fracture toughness tests required by the ASME Code, augmented by 10 CFR 50, Appendix G, are particular to individual plants and this aspect will be addressed on a plant-specific basis by an applicant referencing the RESAR SP/90 design. The use of Appendix G to ASME Code Section III and the results of fracture toughness tests performed in accordance with the Code and NRC regulations for establishing safe operating procedures, provide adequate safety margins during operating, testing, maintenance, and postulated-accident conditions. Compliance with the Code provisions and NRC regulations on an individual plant-specific basis will satisfy the requirements of GDC 31 and 10 CFR 50.55a regarding prevention of fracture of the RCPB.

The applicant's approach to following the recommendations of Regulatory Guides 1.44, 1.37, and 1.31 are discussed in Section 4.5.1 of this SER.

The controls to be followed during material selection, fabrication, examination, and protection from sensitization and contamination should provide reasonable assurance that the RCPB components of austenitic stainless steels are in a metallurgical condition that minimizes susceptibility to stress corrosion cracking during service.

The controls to be imposed on welding ferritic and austenitic steels under conditions of limited accessibility satisfy, to the extent practical, the recommendations of Regulatory Guide 1.71, "Welder Qualification for Areas of Limited Accessibility." The applicant's contractor will maintain close supervisory control of the welders to ensure that the most skilled welders are used in areas of limited accessibility. The staff concludes that as such welds are inspected, qualification of the welders making acceptable welds occurs automatically under the Code because the welders will have demonstrated their capability to make the welds under limited access conditions. These controls satisfy the quality standards requirements of GDC 1 and 50 and 10 CFR 50.55a.

The controls to be imposed on weld cladding of low-alloy steel components with austenitic stainless steel are in accordance with Regulatory Guide 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components." These controls will provide reasonable assurance that the practices that could result in cracking of the undercladding will be restricted and that cracking of components made from low-alloy steels will not occur during fabrication. If cracking does occur, the required Code examinations should detect such flaws. The controls also satisfy the quality standards requirements of GDC 1 and 30 and 10 CFR 50.55a.

These controls will provide reasonable assurance that welded components of austenitic stainless steel will not develop microfissures during welding. These controls meet the quality standards requirements of GDC 1, GDC 30, and 10 CFR 50.55a and they satisfy the requirements of GDC 14 with regard to prevention of leakage and failure of the RCPB.

The RESAR SP/90 design will take alternative approaches to the recommendations of Regulatory Guide 1.50, "Control of Preheat Temperature for Welding Low Alloy Steels." The alternative preheat practices are documented in WCAP-8577, "The Application of Preheat Temperature After Welding of Pressure Vessel Steels," which has been accepted by the staff. The alternative approaches taken by the applicant are that welding procedures are qualified within the preheat temperature ranges required by ASME Code Section IX rather than at the minimum preheat temperature, and preheat temperatures are maintained for an extended period of time rather than maintained until the start of post-weld heat treatment. The staff concludes that these alternative approaches are adequate to prevent hydrogen cracking (the concern of Regulatory Guide 1.50) and will not cause other problems. Accordingly, the staff accepts these alternative approaches.

The applicant has stated that the RESAR SP/90 design will comply with Regulatory Guide 1.34, "Control of Electroslag Weld Properties."

The specified materials of construction to be exposed to the reactor coolant, secondary coolant, and containment sprays are compatible with the expected environment, as proven by extensive testing and satisfactory performance and conform to ASME Code, Sections II, III, and IX. General corrosion of all materials is expected to be negligible except for carbon steel. When these materials are observed as part of the inservice visual and/or nondestructive inspection program, material integrity will be ensured for subsequent service.

The external nonmetallic insulation to be used on austenitic-stainless-steel components conforms with the recommendations of Regulatory Guide 1.36. The onsite cleaning and cleanliness controls to be used during fabrication and erection conform, to the extent practical, to the regulatory positions of Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants," and Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel." The use of materials of proven performance in service and the conformance with the recommendations of the stated regulatory guides and Codes constitute an acceptable basis for satisfying the requirements of GDC 4, "Environmental Design," with regard to the compatibility of materials and components with the environmental conditions associated with normal operation, maintenance, testing, and postulated-accident conditions.

The selection and use of these materials further satisfies the requirements of GDC 14, "Reactor Coolant Pressure Boundary," as it relates to the design having an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

The staff concludes that the materials to be used for fabrication of the reactor coolant pressure boundary are acceptable, and meet the requirements of GDC 1, 4, 14, 30, and 31 of Appendix A and Appendices B and G to 10 CFR 50 and the requirements of 10 CFR 50.55a.

5.2.4 Reactor Coolant Pressure Boundary Inservice Inspection and Testing (RESAR SP/90 Module 4, Section 5.2.4)

5.2.4.1 Compliance With the Standard Review Plan

Although Westinghouse has provided information regarding the preservice inspection (PSI) program, the plant-specific applicant must submit a PSI program plan during review of its application in order for the staff to evaluate compliance with the regulations and applicable ASME Code Section XI requirements. The staff review to date was conducted in accordance with SRP Section 5.2.4, "Reactor Coolant Pressure Boundary Inservice and Testing," except as discussed below.

Compliance with SRP Section 5.2.4, paragraph II.3, "Acceptance Criteria, Examination Categories and Methods," cannot be evaluated without submittal of a complete PSI program plan describing the individual component and piping system welds along with the examination requirements.

Compliance with SRP Section 5.2.4, paragraph II.4, "Acceptance Criteria, Inspection Intervals," has not been evaluated because this area applies only to inservice inspections (ISI), not to PSI. This subject will be addressed during review of the ISI program when submitted during the FDA stage of review or during review of a plant-specific license application referencing the RESAR SP/90 design.

Compliance with SRP Section 5.2.4, paragraph II.5, "Acceptance Criteria, Evaluation of Examination Results," has been evaluated. The applicant committed in the RESAR SP/90 PDA application to incorporate ASME Code Article IWB-3000, "Acceptance Standards for Flaw Indications," into the PSI program. However, ongoing NRC generic activities and research projects indicate that the presently specified ASME Code procedures may not always be capable of detecting the acceptable size flaws specified in the IWB-3000 acceptance standards. For example, ASME Code procedures specified for volumetric examination of the reactor vessel, bolts and studs, and piping have not proven to be capable of detecting the acceptable size flaws in all cases.

The staff will continue to evaluate the development of new or improved procedures and will require that these improved procedures be made a part of the inservice examination requirements. The applicant's repair procedures based on ASME Code Article IWB-4000, "Repair Procedures," have not been reviewed. Repairs are not generally necessary in the PSI program. This subject will be addressed during the staff review of the ISI program.

Compliance with SRP Section 5.2.4, paragraph II.7, "Acceptance Criteria, Code Exemptions," will be evaluated when a complete PSI program is submitted by the plant-specific applicant. Although exemptions from Code examinations are permitted, the PSI program must list the exemptions and criteria taken in accordance with the Code.

Compliance with SRP Section 5.2.4, paragraph II.8, "Acceptance Criteria, Relief Requests," has not been completely evaluated because the applicant has not identified limitations to examination. Specific areas where ASME Code examination requirements cannot be met are normally identified during the performance of the PSI. The plant-specific applicant will be required to identify all plant-specific areas where ASME Code, Section XI, requirements cannot be met and provide a supporting technical justification for relief.

5.2.4.2 Examination Requirements

GDC 32, "Inspection of Reactor Coolant Pressure Boundary," Appendix A to 10 CFR 50, requires that those components that are part of the RCPB be designed to permit periodic examination and testing of important areas and features to assess their structural and leaktight integrity. To ensure that no deleterious defects develop during service, selected welds and weld heat-affected zones are to be examined periodically.

The design of the ASME Code Class 1 and 2 components of the RCPB incorporates provisions for access for inservice examinations, as required by Subarticle IWA-1500 to ASME Code, Section XI. 10 CFR 50.55a(g) defines the detailed requirements for the preservice and inservice programs for light-water-cooled nuclear power facility components. On the basis of the construction permits issued by the NRC on or after July 1, 1974, this section of the regulations requires that a PSI program be developed and implemented using at least the edition and addenda to ASME Code, Section XI, applied to the construction of the particular components. However, components (including supports) may meet requirements set forth in subsequent editions and addenda of Section XI that are incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein. The initial ISI program must comply with the requirements of the latest edition and addenda to ASME Code, Section XI, in effect 12 months before the date of issuance of the operating license, subject to the limitations and modifications listed in 10 CFR 50.55a(b).

5.2.4.3 Evaluation of Compliance With 10 CFR 50.55a(g)

The applicant has committed to comply with the applicable provisions of 10 CFR 50.55a and has stated that the PSI program will provide details of areas subject to examination, as well as the method and extent of the preservice examinations. Based on review of the above information, additional information cited in Section 5.2.4.4 below should be addressed during the plant-specific review.

5.2.4.4 Issues To Be Addressed During Plant-Specific Review

The specific areas where the applicable Code requirements cannot be met can only be identified after the examinations are performed. The plant-specific applicant should commit to identify all plant-specific areas where the Code requirements cannot be met and provide a supporting technical justification for relief. The information required for the staff to complete its review of this matter is listed below.

- The plant-specific applicant will be required to submit a PSI program plan. The PSI program should include reference to the ASME Code Section XI edition and addenda that will be used for the selection of components for examinations, lists of the components subject to examination, a description of the components exempt from examination by the applicable Code, and the examination isometric drawings.

  Plans for preservice examination of the reactor pressure vessel welds should address the degree of compliance with Regulatory Guide 1.150, "Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations," Revision 1, February 1983. The applicant should discuss the near-surface examination and resolution with regard to detecting service-induced flaws and the use of electronic gating as related to the volume of material near the surface that is not being examined.

- The plant-specific applicant must submit all relief requests with a supporting technical justification.

The staff's review of the PSI program will be completed when the applicant submits its PSI program.

Because it is a plant-specific item, the initial ISI program has not been submitted. This program will be evaluated after the applicable ASME Code edition and addenda can be determined based on 10 CFR 50.55a(b) but before ISI commences during the first refueling outage.

However, as discussed in Section 14 of this SER, the applicant will be required to address preservice/inservice inspection and testing as part of a program to ensure design reliability during the FDA stage of review.

5.2.4.5 Conclusions

The conduct of periodic examinations and hydrostatic testing of pressure-retaining components of the RCPB, in accordance with the requirements of ASME Code, Section XI, and 10 CFR 50, will provide reasonable assurance that structural degradation or loss of leak-tight integrity occurring during service will be detected in time to permit corrective action before the safety functions of a component are compromised. Compliance with the preservice and inservice examinations required by the Code and 10 CFR 50 will constitute an acceptable basis for satisfying the inspection requirements of GDC 32. The PSI and ISI programs will be reviewed and addressed during review of the FDA application and on a plant-specific basis.

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection (RESAR SP/90 Module 4, Section 5.2.5)

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

C.1 of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems."

Sources, disposition, and indication of identified leakage are described below.

- Residual Heat Removal System (Suction Side)

  The RHRS, which is part of the integrated safeguards system (ISS), will be isolated from the reactor coolant system (RCS) on the suction side by motor-operated valves. Significant leakage past these valves (two in series) will be detected by lifting of relief valves, accompanied by increasing pressurizer relief tank level, pressure, and temperature indications and alarms on the main control board.

- Safety Injection Accumulators

  The accumulators will be isolated from the RCS by check valves. Leakage past these valves (two in series) and into the accumulator subsystem will be detected by redundant control room accumulator pressure and level indications and alarms.

- Core Reflood Tanks

  The core reflood tanks will be isolated from the RCS by check valves. Leakage past these valves (two in series) and into the core reflood tank subsystem will be detected by redundant core reflood tank pressure and level indications and alarms.

- High-Head Safety Injection (HHSI)/RHR Discharge Subsystem

  The HHSI/RHR pump discharge portion of the ISS will be isolated from the RCS by check valves. Leakage past these valves will eventually pressurize this portion of the system and result in lifting of relief valves, accompanied by alarms of increasing boron recycle holdup tank levels.

- Reactor Vessel Head Gasket Monitoring Connections

  Reactor vessel flange seal leakage will be detected by two leakoff connections, one between the inner and outer O-ring, and one outside the outer O-ring. Leakage will be indicated and alarmed at the main control board by a surface-mounted resistance thermocouple that will monitor the leakage before it is collected in the reactor coolant drain tank.

- Component Cooling Water System

  Leakage from the RCS to the component cooling water system (CCWS), which services all RCPB associated components that require cooling, will be detected by the CCWS radiation monitoring equipment and/or increasing surge tank level. Components serviced by this system will include reactor coolant pump thermal barriers, RHR heat exchangers, letdown and excess letdown heat exchangers, and the reactor coolant pump seal injection heat exchangers.

* Steam Generator Tube Leakage

  Steam generator tube leakage from the RCS to the secondary system will be detected by condenser air ejector radiation monitors and by liquid sampling of the steam generators. Samples from each steam generator will indicate reduced pH from the presence of boric acid having leaked from the RCS to the secondary system.

* Power Operated Relief Valves (PORVs) or Pressurizer Safety Valves

  Leakage from PORVs or pressurizer safety valves is directed to the pressurizer relief tank. This leakage is monitored by the temperature instrumentation in the valve discharge piping and by tank pressure, temperature, and level instrumentation. Leakage collected in the pressurizer relief tank is directed to the reactor coolant drain tank for subsequent treatment and discharge.

Indication of unidentified leakage from the RCPB into the containment is provided by four main sources: containment airborne particulate radiation monitors, containment gaseous radiation monitors, containment fan cooler condensate flow monitors, and containment sump level monitors. The particulate and gaseous monitors will operate continuously to detect radiation in the containment atmosphere. Indication and alarms will be provided in the main control room. The radiation monitors will be seismic Category I and located in flood- and tornado-protected structures, thus meeting the requirment of GDC 2 and the guidelines of Regulatory Guide 1.45. The containment fan cooler condensate monitoring system will measure the liquid runoff from the containment fan cooler units to regulate containment humidity. Indication and alarms for this system will be provided in the control room. If a break were to occur in the primary system, the resulting coolant flow would pass to the containment atmosphere, causing airborne contamination and an increase in humidity, or it would condense and fall to the floor. Unidentified leakage to the containment or reactor cavity sumps would be detected by a measurable increase in the sump. The increase of sump level will be indicated and alarmed at the control room. In addition, since containment and reactor cavity sump pumps normally will operate very infrequently, unusually high frequency of pump operation would be another indication of gross leakage. The sump-flow measuring system is testable and can be calibrated as required. Additional sources of indication of unidentified leakage include containment humidity monitors, charging pump operation, and liquid inventory indications.

The staff concludes that the design of the RCPB leakage detection systems are diverse and provide reasonable assurance that primary system leakage (both identified and unidentified) will be detected and meet the requirements of GDC 2 and 30 with respect to protection against natural phenomena and provisions for RCPB leak detection and identification and the guidelines of Regulatory Guides 1.29 and 1.45 with respect to seismic classification and RCPB leakage detection system design and are acceptable.

5.3 Reactor Vessel (RESAR SP/90 Module 4, Section 5.3)

5.3.1 Reactor Vessel Materials (Materials and Fabrication) (RESAR SP/90 Module 4, Section 5.3.1)

The staff evaluated the specified reactor vessel materials against the requirements of GDC 1, 4, 14, 30, 31, and 32 to Appendix A of 10 CFR 50; the material testing and monitoring requirements of Appendices B, G, and H to 10 CFR 50; and the requirements of 10 CFR 50.55a. The applicant's conformance to the recommendations in Regulatory Guides 1.34, 1.43, 1.50, and 1.65 are discussed in Section 4.5.2 of this SER.

The materials that will be used for construction of the reactor vessel and its appurtenances have been identified by specification and found to be in conformance with ASME Code, Section III. Special requirements of the applicant with regard to control of residual elements have been identified and are considered acceptable. Compliance with the Code provisions for material specifications satisfies the quality standards requirements of GDC 1 and 30 and 10 CFR 50.55a.

Ordinary processes will be used for the manufacture, fabrication, welding, and nondestructive examinations of the reactor vessel and its appurtenances. Nondestructive examinations will be performed in addition to Code requirements. Since the applicant has certified that the requirements of ASME Code, Section III, will be complied with, the processes and examinations to be used are considered acceptable. Compliance with these Code provisions meets the quality standards requirements of GDC 1 and 30 and 10 CFR 50.55a.

When welding components of austenitic steels, Code controls will be supplemented by conforming with the recommendations of regulatory guides. The controls that will be imposed on delta ferrite in austenitic-stainless-steel welds satisfy most of the recommendations of Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal." The alternate approaches that will be taken by the applicant have been reviewed by the staff and are acceptable (see Section 4.5.1 of this SER).

The controls (during all stages of welding) specified to avoid contamination and sensitization that could cause stress corrosion cracking in austenitic stainless steels conform with the recommendations of regulatory guides. The controls to avoid contamination and excessive sensitization of austentic stainless steel satisfy, to the extent practical, the recommendations of Regulatory Guide 1.44. The alternative approaches that will be taken by the applicant are acceptable (see Section 4.5.1 of this SER). The controls that will be used will provide reasonable assurance that welded components are not contaminated or excessively sensitized before and during the welding process. These controls satisfy the quality standards requirement of GDC 1 and the material compatibility requirement of GDC 4. The onsite cleaning and cleanliness controls of austenitic stainless steel are in conformance with the recommendations of Regulatory Guide 1.37 or the applicant's alternative approaches are acceptable to the staff as discussed in Section 4.5.1 of this SER. These controls will provide reasonable assurance that austenitic-stainless-steel components will be properly cleaned on site, thus satisfying the requirements of 10 CFR 50, Appendix B, for the cleaning and preservation of material and equipment.

The staff has reviewed the fracture toughness of ferritic reactor vessel and reactor coolant pressure boundary (RCPB) materials and the materials surveillance program for the reactor vessel beltline. The acceptance criteria and references that are the basis for this evaluation are set forth in paragraphs II.5, II.6, and II.7 (10 CFR 50, Appendices G and H) as specified in SRP Section 5.3.1. This review is discussed below.

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

The edition and addenda of the ASME Code that are applicable to the design and fabrication of the reactor vessel and RCPB components are specified in 10 CFR 50.55(a). The specific ASME Code edition and addenda will depend on the date the construction permit is issued. Since no construction permits for the RESAR SP/90 design have been issued, the required Code edition and addenda must be provided by the plant-specific applicant referencing the RESAR SP/90 design.

RESAR SP/90 indicates that the reactor vessel will comply with the fracture toughness requirements of 10 CFR 50, Appendix G. However, to confirm this, the plant-specific applicants referencing this design must provide fracture toughness data from the limiting reactor vessel materials.

Appendix G, "Protection Against Non-Ductile Failures," Section III of the ASME Code, will be used with the fracture toughness test results required by 10 CFR 50, Appendices G and H, to calculate the pressure-temperature limitations for the RESAR SP/90 reactor vessels. These tests provide reasonable assurance that adequate safety margins against the possibility of non-ductile behavior or rapidly propagating fracture can be established for all pressure-retaining components of the reactor coolant boundary. The use of ASME Appendix G as a guide in establishing safe operating procedures and the use of the results of the fracture toughness tests performed in accordance with the ASME Code and NRC regulations, will provide adequate safety margins during operating, testing, maintenance, and anticipated transient conditions. Compliance with these Code provisions and NRC regulations constitutes an acceptable basis for satisfying the requirements for GDC 31.

The materials surveillance programs will be used to monitor changes (resulting from exposure to neutron irradiation and the thermal environment) in the fracture toughness properties of ferritic materials in the reactor vessel beltline region, as required by GDC 32, "Inspection of Reactor Coolant Pressure Boundary." The description of the surveillance programs in RESAR SP/90, which must be in compliance with 10 CFR 50, Appendix H, and American Society for Testing and Materials (ASTM) E-185-82, "Standard Recommended Practices for Surveillance Tests for Nuclear Reactor Vessels," requires fracture toughness data be obtained from material specimens that are representative of the limiting base, weld, and heat-affected zone materials in the beltline region. These data will permit the determination of the conditions under which the vessel can be operated with adequate margins of safety against fracture throughout its service life.

In the RESAR SP/90 application, Westinghouse indicates that the reactor vessel materials surveillance programs will comply with the requirements of 10 CFR 50, Appendix H, and ASTM E 185-82. However, to confirm this conclusion, plant-specific applicants must identify the specific materials in each surveillance capsule, the capsule lead factors, the surveillance capsule withdrawal schedule, the neutron fluence to be received by each capsule at the time of its withdrawal, and the vessel end-of-life peak neutron fluence.

The materials surveillance program, required by 10 CFR 50, Appendix H, will provide information on the effects of irradiation on material properties so that changes in the fracture toughness of the material in a RESAR SP/90 reactor vessel beltline can be properly assessed and adequate safety margins against the possibility of vessel failure can be provided.

Compliance with 10 CFR 50, Appendix H, and ASTM E-185 ensures that the plant-specific surveillance program will be capable of monitoring radiation-induced changes in the fracture toughness of the reactor vessel material and satisfies the requirements of GDC 32.

There is reasonable assurance that the surveillance program will monitor the change in the beltline region material properties to the extent required for establishing pressure-temperature limits and to preserve the integrity of the vessel. The surveillance program will generate sufficient information to permit the determination of conditions under which the reactor vessel will be operated with an adequate margin against rapidly propagating fracture throughout its service lifetime. These items will be reviewed during review of the plant-specific application.

The staff concludes that the reactor vessel materials to be used in the RESAR SP/90 design are generally acceptable, and, to the extent practicable, meet the requirements of GDC 1, 4, 14, 30, 31, and 32 of 10 CFR 50 Appendix A; the material testing and monitoring requirements of 10 CFR 50, Appendices B, G, and H; and the requirements of 10 CFR 5.55a.

The plant-specific applicant referencing the RESAR SP/90 design will be required to comply with 10 CFR 50.55a and 10 CFR 50 Appendices G and H as discussed above.

5.3.2 Pressure-Temperature Limits (RESAR SP/90 Module 4, Section 5.3.2)

The review of the applicant's pressure-temperature limits for operation of the reactor vessels cannot be completed until the plant-specific applicant submits the required pressure-temperature limit curves for staff evaluation. The acceptance criteria and list of references that are the basis for this evaluation are set forth in SRP Section 5.3.2. This review is discussed below.

10 CFR 50, Appendices G and H, describe the conditions that require pressure-temperature limits and provide the general bases for these limits. These appendices specifically require that pressure-temperature limits must provide safety margins at least as great as those recommended in the ASME Code, Section III, Appendix G. 10 CFR 50, Appendix G, requires additional safety margins whenever the reactor core is critical (except for low-level physics tests) for the material in the closure flange and beltline regions.

The staff reviewed the pressure-temperature limits (listed below) that are imposed on the RCPB during operation and testing to ensure that they provide adequate safety margins against non-ductile behavior or rapidly propagating failure of ferritic components, as required by GDC 31.

- pressure hydrostatic tests
- inservice leak and hydrostatic tests
- heatup and cooldown operations
- core operation

10 CFR 50, Appendices G and H, require the applicant to predict the amount of increase in the nil-ductility reference temperature, RT_{NDT}, resulting from neutron irradiation. Any further increase in RT_{NDT} is then added to the initial RT_{NDT} to establish the adjusted reference temperature. The staff's recommended method for calculating the increase in RT_{NDT} resulting from neutron irradiation is contained in Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Material."

Revision 2 of the guide provides a graph, tables, and formulae for calculating the increase in RT_{NDT}. The relationships contained in the guide were derived by statistical analysis of 216 material data points that were reported from testing of irradiated materials. These materials were contained in surveillance capsules that had been irradiated inside U.S. commercially operated nuclear reactor vessels. As more surveillance data becomes available, this guide may need additional revision because the relationship between the increase in RT_{NDT} and neutron fluence is empirically derived from analysis of material surveillance data from U.S. commercially operated nuclear reactor vessels.

In the RESAR SP/90 application, Westinghouse indicates that the increase in RT_{NDT} resulting from neutron irradiation will be determined using the radiation damage curves developed by Westinghouse instead of those presented in Regulatory Guide 1.99. The bases for this proposal is contained in a letter from T. N. Anderson to the Secretary of the Commission dated June 23, 1978. In the letter, the author indicates that as a result of 10 data points from the Point Beach-1 and Connecticut Yankee surveillance program, the embrittlement appears to reach a limiting or steady-state value well below that predicted by Regulatory Guide 1.99. The staff's statistical analysis of all available surveillance data did not conclude that embrittlement reaches a limiting steady-state value. Hence, the proposal is unacceptable and the applicant must determine, for the RESAR SP/90 reactor vessel, the increase in RT_{NDT} resulting from neutron irradiation using the latest published revision of Regulatory Guide 1.99.

The applicant indicates that pressure-temperature operating curves will comply with Appendix G to 10 CFR 50 and Appendix G to ASME Code, Section III. The applicant also indicates that these operating limits will be included in the technical specifications of the plant-specific applicant. Hence, the staff will confirm that the plant-specific applicant's pressure-temperature limits comply with the requirements of Appendix G to 10 CFR 50 and Appendix G to ASME Code Section III when the plant-specific applicant referencing the RESAR SP/90 design submits its technical specifications.

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

5.3.3 Reactor Vessel Integrity (RESAR SP/90 Module 4, Section 5.3.3)

Although most areas are reviewed separately in accordance with the staff's review plans, reactor vessel integrity is of such importance that a special summary review of all factors relating to reactor vessel integrity is warranted. In this section, the staff has reviewed the fracture toughness of ferritic materials used for reactor vessel and reactor coolant pressure boundary, the pressure-temperature limits for operation of the reactor vessel, and the materials surveillance program for the reactor vessel beltline. The acceptance criteria and reference that are the basis for this evaluation are set forth in SRP Section 5.3.3, paragraphs II.2, II.6 and II.7 (10 CFR 50, Appendices G and H). This review is discussed below.

The staff has reviewed the information in each area to ensure that it is complete and that no inconsistencies exist that would reduce the certainty of vessel integrity. The areas reviewed and parenthetical reference to the sections of this report in which they are discussed are:

- design (Section 5.3.1)
- materials of construction (Section 5.3.1)
- fabrication methods (Section 5.3.1)
- operating conditions (Section 5.3.2)

The staff concludes from its review of the above factors, which contribute to the structural integrity of the reactor vessel, that plant-specific applicants referencing the RESAR SP/90 design can comply with 10 CFR 50, Appendices G and H.

5.4 Component and Subsystem Design (RESAR SP/90 Module 4, Section 5.4)

5.4.1 Reactor Coolant Pump Flywheel (RESAR SP/90 Module 4, Section 5.4.1)

The safety objective of this review is to ensure that the integrity of the primary reactor coolant pump flywheel is maintained to prevent failure at normal operating speeds and speeds that might be reached under accident conditions. The basis for review is outlined in SRP Section 5.4.1.1 and Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity," which describes a method acceptable to the NRC staff in implementing GDC 4, "Environmental and Missile Design Bases," of Appendix A to 10 CFR 50 with regard to minimizing the potential for failure of reactor coolant pump flywheels.

The flywheel will consist of two thick plates bolted together. Each plate is generally fabricated from SA-533, Grade B, Class 1 steel. The flywheel material is produced by a process that minimizes flaws in the materials. The finished flywheels are subjected to 100 percent volumetric ultrasonic inspection using procedures and acceptance standards specified in ASME Code Section III.

The nil-ductility transition temperature (NDTT) of the flywheel material will be no higher than 10°F. The Charpy V-notch energy level will be at least 50 foot-pounds with a 35-mil lateral expansion at 70°F in both the parallel and normal orientation with respect to the rolling direction of the flywheel material. Hence, an RT_{NDT} of 10°F can be assumed.

The calculated stresses at operating speed that will result from centrifugal forces and the interference fit on the shaft will be within the Regulatory Guide

1.14 limits. The pump will run about 1190 rpm and may operate briefly at over-speed up to 110 percent during a loss of offsite power. The design speed is 125 percent of the operating speed; hence, the flywheels also will be tested at 125 percent of the maximum synchronous speed of the motor. The combined stresses at design over-speed resulting from interference fit and centrifugal forces are within the limits of Regulatory Guide 1.14.

The flywheels will be inspected by removing the cover and any crack that may have developed will be noticed. The inservice inspection program requires a 100-percent volumetric ultrasonic inspection and follows the requirements of ASME Code, Section XI, and the recommendations of Regulatory Guide 1.14.

The staff has reviewed the proposed materials, fabrication, design, and inspections applicable to the pump flywheels and concludes that they comply with Regulatory Guide 1.14.

5.4.2 Steam Generator (RESAR SP/90 Module 4, Section 5.4.2)

The staff evaluated the materials specified for the steam generator against the requirements of GDC 1, 14, 15, and 31 of 10 CFR 50, Appendix A.

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

The requirements of GDC 14 and 15 will be met to ensure that the RCPB and associated auxiliary systems will be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture during normal operation and anticipated operational occurrences. The selection and use of the materials will satisfy the requirements of GDC 14, as they relate to a design having an extremely low probability of abnormal leakage, of rapidly propapating failure, and of gross rupture. The crevice between the tubesheet and the inserted tube will be minimized because the tubes will be hydraulically expanded to the full depth of insertion of the tube in the tubesheet. The tube expansion and subsequent positive contact pressure between the tube and the tubesheet will preclude a buildup of impurities from forming in the crevice region and reduce the probability of crevice boiling. Thermally treated Inconel-690 tubing should be effective in resisting general corrosion and intergranular stress corrosion cracking in the secondary coolant environment, as indicated by testing and limited service experience.

The requirements of GDC 31 with regard to fracture toughness of the ferritic materials will be met because the pressure boundary materials of ASME Class 1 components of the steam generator will be required by the applicant to comply with the fracture toughness and tests requirements of Article NB-2300 to ASME Code, Section III. The materials of the ASME Class 2 components of the steam generator also will be required to comply with the fracture toughness requirements of Article NC-2300.

Code cases used in material selection will conform to Regulatory Guides 1.84 and 1.85.

The onsite cleaning and cleanliness controls during fabrication and erection will conform to the recommendations of ANSI Standard N45.2.1-1973 and Regulatory Guide 1.37.

Additional measures are incorporated in the design to manage sludge. The center of the tube sheet contains an untubed region to collect sludge and provide a blowdown path. Provisions are made for sludge collection and sludge lancing.

The materials of construction exposed to the reactor coolant, secondary coolant, and containment sprays will be compatible with the expected environment, as proven by extensive testing and satisfactory performance, and conform to ASME Code, Sections II, III, and IX. (For example, alloy 800 will be used for the quatrefoil tube supports.) General corrosion of all materials is expected to be negligible except for carbon and low alloy steels. Where these materials will be exposed to the secondary coolant, general corrosion is expected to be negligible; where these materials might be exposed to leaking primary coolant, their behavior will be readily observed as part of the inservice visual and/or nondestructive inspection program that will be performed to ensure their integrity for subsequent service. It is recommended that copper-free materials of construction be used in contact with the secondary coolant.

Westinghouse discussed the possibility of degradation of steam generator tubes resulting from either mechanical or flow-induced vibration. Westinghouse analyzed the three vibration mechanisms of vortex shedding, turbulence, and fluid elastic excitation, and have been analyzed it was concluded that fatigue degradation as a result of flow-induced vibration is not anticipated.

Since the date of this analysis (June 1984) preceded the steam generator tube rupture that occurred at North Anna Unit 1 on July 15, 1987, the steam generator vibration analysis should be re-examined. This was the subject of NRC IE Bulletin 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes." Because of this, and unfavorable experience in other operating steam generators, the design of the antivibration bars should also be re-examined.

The external nonmetallic insulation to be used on austenitic-stainless-steel components conforms with the recommendations of Regulatory Guide 1.36. The onsite cleaning and cleanliness controls during fabrication and erection will conform to the positions of Regulatory Guides 1.37 and 1.44. The use of materials of proven performance in service and the conformance with the recommendations of the stated regulatory guides and codes constitute an acceptable basis for satisfying the requirements of GDC 4 with regard to the compatibility of materials and components with the environmental conditions associated with normal operation, maintenance, testing and postulated accident.

The selection and use of the materials further satisfies the requirements of GDC 14 as it relates to the design having an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

The staff concludes that the materials to be used in the above systems are acceptable from a corrosion point of view, and that they meet the requirements of GDC 1, 14, 15, and 31 of 10 CFR 50, Appendix A.

GDC 32, "Inspection of Reactor Coolant Pressure Boundary," requires in part that components that are part of the RCPB be designed to permit periodic inspection and testing of important areas and features to assess their structural and leaktight integrity. The steam generators will be designed to meet the ASME Code requirements for Class 1 and 2 components. Provisions also will be made to permit inservice inspection of the Class 1 and 2 components, including individual steam generator tubes. The design aspects provide access for inspection, including six access ports, and the proposed inspection program will follow the recommendations of Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes," Revision 1, and the requirements of plant-specific technical specifications. It will comply with the applicable parts of ASME Code, Section XI, with respect to the inspection methods to be used, provisions for a baseline inspection, selection and sampling of tubes, inspection intervals, and actions to be taken in the event that defects are identified. The staff finds this acceptable for the PDA stage of review. The steam generator inservice inspection program will be evaluated on a plant-specific basis.

5.4.3 Residual Heat Removal System

The staff reviewed the RHRS for the RESAR SP/90 design in accordance with SRP Section 5.4.7. The RHRS is designed as a subsystem of the integrated safeguards system (ISS) and as a safety-related system that will integrate the functions of the residual heat removal, emergency core cooling, and containment spray. Those components within the ISS that perform an RHR function are the four low-head pumps, four RHR heat exchangers, and associated valves, piping, and instrumentation.

The RHRS is designed to remove heat from the RCS during a reactor shutdown after the RCS temperature and pressure have been reduced to approximately 350°F and 400 psig, respectively. The RHRS is capable of reducing the RCS temperature to the cold shutdown condition and maintaining this temperature until the plant is started again.

The RHRS will operate in the following modes:

- Cold Shutdown: The primary function of the RHRS is to recirculate reactor coolant through the core and through the RHR heat exchangers during plant cooldown and maintain the RCS in cold shutdown conditions.

- Containment Spray: In the event of a large break LOCA or a main steam line break accident, the low-head pumps will function as containment spray pumps and take suction from the emergency water storage tank (EWST) and deliver to the containment spray ring headers.

- Emergency Core Cooling System, Recirculation Mode: The low-head pumps will be aligned to perform a long-term emergency core cooling recirculation function.

- Refueling: The low-head pumps will be used to transfer refueling water from the EWST to the refueling canal at the beginning of refueling operations. Refueling water will be returned to the EWST from the refueling cavity at the end of refueling operations by a gravity drain.

The RESAR SP/90 design is required to meet BTP RSB 5-1, "Design Requirements of the Residual Heat Removal System." Westinghouse's approach to meet these requirements is discussed below.

5.4.3.1 Functional Requirements

SRP Section 5.4.7 requires the RHRS for RESAR SP/90 design to meet GDC 1 through 5. Compliance with GDC 1 through 4 is addressed in Section 3 of this SER. The staff has concluded that GDC 5 is satisfied because components of the RHRS will not be shared between units.

During normal plant shutdown when non-safety-related equipment and offsite power are available, decay heat will be removed by the main feedwater system, the condenser steam dump system, and circulating water system. During plant emergency shutdown, when non-safety-related equipment and offsite power are not available, decay heat will be transferred from the core by RCS natural circulation with the steam generators as the heat sink. This will be achieved by using the safety-related steam generator safety valves and power-operated atmospheric steam dump valves to vent vaporized secondary coolant. Secondary coolant makeup will be provided via the emergency feedwater pumps from the safety grade emergency feedwater storage tanks. When the steam generators are being used as the reactor heat sink during the first phase of plant cooldown before the point when the RHRS can be utilized, a single active failure of any component does not render all steam generators ineffective as a heat sink. Either of the two emergency feedwater trains (each train includes one turbine-driven pump and one motor-driven pump) will have sufficient capacity to provide for all steam generator makeup requirements.

Position G of BTP RSB 5-1 states that the seismic Category I water supply for the emergency feedwater system for a PWR shall have sufficient inventory to permit operation at hot shutdown for at least 4 hours, followed by cooldown to the conditions permitting operation of the RHRS. The inventory needed for cooldown shall be based on the longest cooldown time needed with either only onsite or only offsite power available with an assumed single failure. The design criteria for the emergency feedwater storage tank specified in RESAR SP/90 Section 10.4.9.2.1.2 has been amended by Westinghouse in response to open items dated May 1989. The Westinghouse response states:

> Two equally sized emergency feedwater storage tanks are provided, one in each subsystem. The tanks are safety grade, seismically qualified and protected from missiles, fire, etc. The tanks contain a sufficient quantity of condensate quality water to permit operation at hot standby conditions for eight hours, followed by cooldown to the conditions permitting operation of the residual heat removal subsystem. The required emergency feedwater inventory in the tanks is based on both of the following:
>
> 1. An extended time (6 hours) to cool down to RHR cut-in conditions is assumed, based on a natural circulation cooldown associated with only onsite power being available. This conservative assumption maximizes the decay heat input to the RCS during the cooldown.
>
> 2. One reactor coolant pump is assumed to be operating during the cooldown, continuously adding heat to the RCS. This is consistent with only onsite power being available, and is an additional conservatism which maximizes the total heat input to the RCS during the cooldown.

In addition, the combined water inventory of both tanks is sufficient to allow indefinite hot standby operation. An indefinite time is satisfied by:

a. having one day's supply of emergency feedwater stored in the EFWST's and taking credit for availability of the alternate water supplies in that time, or

b. use of primary side feed and bleed operations.

Both the cooldown and indefinite hot standby operations can be conducted using only safety grade equipment and with the most limiting single failure assumed.

The tank design provides additional volumes of water to allow for inaccuracies in the tank level indicators, overflow and pumpout margins and spill allowance.

The response satisfactorily demonstrates compliance of the emergency feedwater storage tank with Position G of BTP RSB 5-1 and is acceptable.

The combination of RCS contraction as a result of cooldown or the opening of one of the three safety-related pressurizer PORVs will depressurize the RCS. The discharge of the pressurizer PORVs will be directed to the pressurizer relief tank where it will be condensed and cooled. By using safety-related equipment, as stated above, the RCS will be cooled and depressurized to approximately 350°F and 400 psig so that the RHRS can be operated.

The second phase of the plant cooldown is from the RCS temperature of 350°F to cold shutdown temperature of 200°F. During this stage, the RHRS will be brought into operation. Low-head pumps will circulate the reactor coolant, and the heat exchangers in the RHRS will remove heat from the RCS. The RHR heat exchangers will transfer the residual heat to the component cooling water system, which ultimately will transfer the heat to the service water system and the ultimate heat sink.

Adding borated water to the RCS during the cooldown will control core reactivity. The safety-related high-head safety injection pumps and their associated piping and valves from the emergency water storage tank (EWST) to the RCS, safety-related emergency letdown lines from RCS to EWST (including their isolation valves), and the safety-related pressurizer PORVs and their associated block valves will control boration and RCS inventory. All systems required for cold shutdown will be capable of being operated from the control room with either only onsite or only offsite power available.

Four separate trains for each unit will provide redundancy in the RHR system. Each RHR train will be powered by an emergency diesel generator that also powers one other RHR train. Leak detection for the RHR system is discussed in Section 5.2.5 of this SER. Isolation valve and power supply redundancy are discussed separately in this section.

The sizing of the decay heat removal equipment that will not be affected by LOCA analysis results were designed using decay heat generation rates from ANS Standard 5.1 (October 1979). The decay heat generation rates in ANS Standard 5.1 do not contain the conservatisms that historically have been employed by the NRC as presented in BTP RSB 9-2. The approach taken for the RESAR SP/90 decay heat values is consistent with the approach taken in the EPRI requirements for advanced light-water reactors. It is intended that the sizing of decay heat removal equipment which is not affected by Appendix K to 10 CFR 50 is designed using decay heat generation rates from ANS Standard 5.1-1979. Evaluations have shown ANS 5.1-1979 to be conservative compared to the most realistic evaluations of decay heat generation and is an acceptable basis for design of the RESAR SP/90 decay heat removal equipment

5.4.3.2 RHRS Isolation

The RHRS valving arrangement is designed to provide adequate protection to the RHRS from overpressurization when the reactor coolant system is operating at high pressure. There will be two redundant motor-operated isolation valves between each RHR pump suction and the RCS hot legs. Each of these valves will be separately and independently interlocked to prevent opening if RCS pressure is greater than a specified value and to automatically close if RCS pressure exceeds a higher specified value. The RHRS will be isolated from the RCS on the discharge side by the check valves in each return line and a normally closed MOV located outside containment. Operator action to open the RHR flowpath will be accomplished normally from the main control room as required by BTP RSB 5-1.

5.4.3.3 RHR Pressure Relief

Each of the four RHRS suction lines will be equipped with a pressure relief valve designed to relieve the combined flow of the charging pumps at the relief valve set pressure. These relief valves also will protect the RCS from overpressurization during low-temperature operating conditions, as discussed in Section 5.2.2.2 of this report.

Each of the four RHRS discharge lines will be equipped with a pressure relief valve designed to relieve the possible backleakage through the valves isolating the RHRS from the RCS. These relief valves will be adequate to protect the RHR from overpressurization.

To respond to a staff concern about thermal relief protection in the RHRS piping between isolation valves for the RESAR SP/90 design, Westinghouse provided the following acceptable clarification in its response to open items dated May 1989:

> Thermal relief protection is provided between pairs of normally closed RHRS isolation valves, as follows:
>
> 1. In the RHR pump suction lines from the RCS - Between 8 inch motor operated valves 9000A and 9001A, for example, by means of 3/4 inch check valve 9019A, which will relieve pressure from between valves 9000A and 9001A to the RCS side of 9000A.
>
> 2. In the ISS pump full flow test lines to the EWST - Between 6 inch motor operated valves 8813A and 8814A, for example, by means of 3/4 inch check valve 8815A. This will relieve pressure from between valves 8813A and 8814A to the upstream side of 8814A, which in turn is protected against overpressure by relief valve 9020A.

3. In the connections between the RHR/CS pump discharges and the containment spray headers - Between 6 inch motor operated valves 9009A and 9011A, for example, by means of 3/4 inch check valve 9010A, which will relieve pressure from between valves 9009A and 9011A to the upstream side of 9009A, which in turn is protected against overpressure by relief valve 9022A. Note that 6 inch valve 9009A is normally open (aligned for containment spray); it is closed for RHR operation.

5.4.3.4 RHR Pump Protection

Each RHR pump is protected from low flow resulting from high discharge line pressure or discharge line isolation by its miniflow path because the RHR pump miniflow lines will contain no valves; therefore, they cannot be isolated by operator action.

Individual flow and pump inlet/outlet temperature instruments will monitor the condition of each of the four RHR pumping trains. A readout for each of these instruments will be in the main control room (MCR). In addition, the RHR flow will have a low-flow alarm and the RHR inlet temperature will have a high-temperature alarm; both alarms will annunciate in the MCR. The RHR inlet temperature will provide measurement of the RCS temperature during RHR operation.

Westinghouse has provided the following response to address the staff concerns regarding lowered RCS inventory operation as discussed in NRC Generic Letter 87-12; the staff considers this commitment acceptable.

> During certain shutdown periods, it may be necessary to perform inspection and/or maintenance operations on the steam generators and reactor coolant pumps. Toward the end of the associated cooldown the reactor coolant inventory is reduced sufficiently to drain the steam generator channel heads and install steam generator isolation devices (nozzle dams). The RCS water level is lowered while RHR operation is continued; this is termed "mid-loop" operation.
>
> Following nozzle dam installation, the RCS water level is raised to the appropriate level for continuation of the inspection/maintenance work (just below the vessel flange) or for refueling (top of refueling canal), unless the reactor coolant pump shaft is to be removed. Pump shaft removal requires that mid-loop operation be continued.
>
> To ensure its continued availability to perform the residual heat removal function during mid-loop operation, the following features are incorporated in the design of the reactor coolant system (RCS) and the residual heat removal (RHR) portion of the integrated safeguards system (ISS):
>
> 1. The layout of the RCS hot leg piping and the steam generator channel head is such that installation of the nozzle dams can be performed with an 80% level in the hot leg piping; this is 9.3 inches above the actual mid-plane elevation.
>
> 2. With the conventional Westinghouse arrangement of a residual heat removal piping connection at 45° from horizontal, it

has been calculated that onset of vortexing with attendant air ingestion would occur at a level 3.0 inches below mid-plane elevation. Therefore, during "mid-loop" operation, a margin in excess of 12 inches would exist between normal operating level and the critical level at which RHR pump operation may be impaired due to high levels of air entrainment. While this is a significant improvement relative to current plants, Westinghouse commits to install, in addition, a vortex breaker in each RHR suction nozzle. This vortex breaker consists of a 24 inch long section of 14 inch Schedule 140 piping connected in a vertical direction to the bottom of the hot leg piping; the 8 inch RHR suction line is connected to the bottom of this vortex breaker. With a vortex breaker, air ingestion commences at about the same water elevation as with a conventional RHR suction nozzle; however, the amount of air entrainment will remain below 10% unless the hot leg is completely drained. Therefore, the potential for RHR pump damage has essentially been eliminated.

3. The RHR pump suction line is "self-venting," i.e., it slopes continuously upward from the pump to its connection to the hot leg (vortex breaker). If the pump should stop during mid-loop operation (due to interruption of electric power, for example) any air bubbles present in the pump or suction piping will be vented back up through the suction line to the water surface in the hot leg. This feature provides for re-starting the pump under conditions which automatically assure a flooded suction.

4. Separate narrow range level transmitters, calibrated for low temperature conditions, indicate the RCS water level between the bottoms of two hot legs and the tops of the steam generator inlet elbows in the same loops during the approach to and conduct of mid-loop operation. Indication in the main control room and low level alarms are provided.

5. The range of the wide range pressurizer level instrumentation used during "cold" operations, has been expanded to the bottom of the hot legs. This provides a continuous level indication in the main control room, transitioning to the range of the two, more accurate, narrow range loop level instruments.

6. The RHR pumps will be designed to operate without undergoing cavitation or other adverse effects under conditions of no subcooling in the hot legs. Specifically, definition of design values for "NPSH available," [net positive suction head] "NPSH required" (by the pump) and the required layout characteristics (elevation difference, pipe routing, etc.) will be coordinated to assure that the RHR pumps can be started and run at their full RHR flowrate even if boiling in the reactor vessel is occurring. This assures that the normal RHR function can be readily used to recover from a temporary loss of cooling.

7. A locally mounted flow transmitter in each RHR return header (downstream of the RHR heat exchanger), with readout in the main control room, indicates RHR return flow to the reactor vessel. A low alarm will alert the operator to a decrease in RHR flow in the associated subsystem.

8. The drain down of the RCS to mid-loop operation level and RCS inventory control during mid-loop operation is performed by the operator in the main control room, using the RHR to CVCS letdown flowpath and normal CVCS functions. This will eliminate the need to coordinate local actions in the containment with the control room operators to control RCS drain down rate and level.

9. Procedures will require that one of the four HHSI pump subsystems always will be available for use during mid-loop operations. This will ensure that a backup source of water for restoring RCS inventory is readily available and can be actuated from the main control room.

10. At least two incore thermocouples will be available to directly measure the core exit temperature during mid-loop RHR operation. Each of these thermocouples will be on separate instrument electrical channels. Also, since the SP/90 incore thermocouples are independent of the RV head, their availability can be maximized; however, these thermocouples will be retracted from the core region during the actual movement/replacement of the fuel. It should be noted that when fuel is being moved the refueling cavity would be flooded.

Note that these design features provide the operator in the main control room with all required instrumentation, alarms, and operation controls necessary to adjust, maintain, and take any necessary recovery actions for both RCS inventory control and heat removal.

Additionally, during the Final Design Application (FDA) phase, Westinghouse will perform evaluations to examine potential design criteria to establish procedures and administration controls that will reasonably ensure that containment closure will be achieved prior to the time at which a core uncovery could result from a loss of RHR coupled with an inability to initiate alternate cooling or addition of water to the RCS inventory.

In addition to these design features, appropriate operating and emergency procedures will be defined to guide and direct the operator in the proper conduct of mid-loop operation, and to aid in detection and correction of off-normal conditions which might occur during such operations.

5.4.3.5 Tests, Operational Procedures, and Supporting Systems

Westinghouse has committed to providing the preoperational and initial startup test program and ensuring its conformance with Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants," in the FDA application.

The proposed tests will be reviewed by the staff to ensure that all necessary tests for demonstrating the operation of the RHRS are incorporated during the FDA stage of review.

To meet the requirements of BTP RSB 5-1, the staff requires that a boron mixing and natural circulation test be performed in the first plant using the RESAR SP/90 design. In their response to open items dated May 1989, Westinghouse committed to perform a combined natural circulation and boron mixing test as part of the pre-operational and initial startup test program. The test will demonstrate the capability of the RESAR SP/90 design to provide natural circulation cooling of the RCS, and to mix injected borated water in the RCS under natural circulation conditions.

The results of these tests, together with appropriate analysis, will be used to demonstrate the capability of the RESAR SP/90 design to achieve a cold shutdown condition using only safety grade equipment.

The staff will review the test results during the initial startup test program.

The staff has reviewed the design of the component cooling water system to ensure that sufficient cooling capability will be available to the RHR heat exchangers. The acceptability of this cooling capacity and its conformance to GDC 44, 45, and 46 are discussed in Section 9.2 of this SER.

The RHRS is designed to meet seismic Category I requirements and with no motor-operated valves that are subject to flooding. Provisions to protect equipment from flooding are discussed in Section 3.4 of this SER. The RHRS capability to withstand pipe whip inside and outside containment, as required by GDC 4, is discussed in Section 3.6 of this SER.

All RHR lines will have containment isolation features. Satisfaction of the requirements of GDC 56 and 57 and Regulatory Guide 1.11 is discussed in Section 6.2.4 of this report.

The outline of the operating procedures to bring the RESAR SP/90 to cold shutdown has been provided in Section 5.4.7.2.6 of the RESAR SP/90 application. Westinghouse has committed to provide operating procedures and administrative controls necessary for lowered RCS inventory operation to prevent loss of RHRS in the FDA application. The staff finds this commitment acceptable.

5.4.3.6 Conclusions

The RHR function is accomplished in two phases for the RESAR SP/90 design, the initial cooldown phase and the RHRS operation phase. In the event of loss of offsite power, the initial phase of cooldown will be accomplished by using the auxiliary feedwater system and the atmospheric steam dump valves to reduce the RCS temperature and pressure to values that permit operation of the RHRS. The RHRS removes core decay heat and provides long-term core cooling after the initial phase of reactor cooldown. The scope of review of the RHRS for the PDA application included piping and instrumentation diagrams, failure modes and effects analysis, and design performance specifications for essential components. The review included the Westinghouse-proposed design criteria and design bases for the RHRS and its analyses of the adequacy of those criteria and bases and the conformance of the design to those criteria and bases.

Except for the unresolved issues identified above, the staff concludes that the design of RHRS is acceptable and meets the requirements of GDC 2, 5, 19, and 34.

5.4.4 Pressurizer Relief Tank

The pressurizer relief discharge system will collect, cool, and direct the steam and water discharged from various safety and relief valves in the containment for processing. The system will consist of the pressurizer relief tank (PRT), the discharge piping from the pressurizer relief and safety valves, the relief tank internal sparger, spray nozzles with associated header and piping, the tank nitrogen supply, the drain to the liquid waste processing system, the relief tank rupture discs, and the rupture disc discharge piping. The system is not safety-related (Quality Group D, nonseismic Category I) and is not part of the RCPB because all of its components will be downstream of the RCS safety and relief valves. Therefore, its failure would not affect the integrity of the RCPB.

The PRT will be sized to absorb the energy content of 110 percent of the full-power pressurizer steam volume through the primary relief and safety valves. Other relief valve discharge to the pressurizer relief tank will be from the RHR system and from the CVCS. Releases from these sources will be less than the design-basis release from the pressurizer. The internal spray and bottom drain on the PRT will be used to cool the water within the tank. The contents may also be cooled by recirculation through a heat exchanger of the liquid waste processing system. A nitrogen blanket will be provided in the tank to permit expansion of entering steam and to control the tank internal atmosphere. If a discharge exceeding the design bases should occur, the rupture discs on the tank would pass the discharge through piping to the emergency water storage tank. The contents of the PRT will be normally drained to the waste holdup tank in the waste processing system or the recycle holdup tank in the boron recycle system via the reactor coolant drain tank pumps. The rupture discs on the pressurizer relief tank have a total capacity equal to the combined capacity of the pressurizer safety and relief valves. The tank and the rupture disc holders are designed for full vacuum to prevent collapse if the contents cool following a discharge without nitrogen being added. The PRT will be provided with instrumentation in the control room to indicate pressure and temperature and alarms for high or low level, high pressure, and temperature.

The PRT will be located at grade elevation inside the containment, which provides protection against natural phenomena. The tank will be separated from safety-related equipment so that its failure does not compromise the capability to safely shut down the plant, and possible ruptured disc fragments do not present a missile hazard when the disc ruptures. Thus, the requirements of GDC 2 and 4, and the guidelines of Regulatory Guide 1.29, Position C.2 are satisfied.

The staff concludes that the pressurizer relief discharge system meets the requirements of GDC 2 and 4 with respect to the need for protection against natural phenomena and internal missile protection because its failure does not affect safety system functions, it meets the guidelines of Regulatory Guide 1.29 concerning seismic classification, and is acceptable.

6 ENGINFERED SAFETY FEATURES (RESAR SP/90 Module 1, Section 6.0)

6.1 Engineered Safety Feature Materials (RESAR SP/90 Module 1, Section 6.1)

6.1.1 Metallic Materials (RESAR SP/90 Module 1, Section 6.1.1)

The staff evaluated the engineered safety feature (ESF) materials against the requirements of GDC 1, 14, and 31 and 10 CFR 50.55a with regard to ensuring an extremely low probability of leakage, of rapidly propagating failure, and of gross rupture. The materials selected for ESF satisfy Appendix I of Section III and Parts A, B, and C of Section II of the ASME Code and the staff position that the yield strength of cold-worked stainless steels shall be less than 90,000 psi. Fracture toughness of the ferritic materials meets the requirements of the Code.

The controls on the use and fabrication of the austenitic stainless steel of the systems will follow most of the recommendations of Regulatory Guides 1.31 and 1.44. The alternative approaches taken by the applicant have been reviewed and are acceptable to the staff. Fabrication and heat treatment practices that will be performed provide assurance that the probability of stress corrosion cracking will be reduced during the postulated accident time interval.

The staff evaluated the reactor coolant boundary and associated auxiliary systems against the requirements of GDC 1, 14, and 31 and 10 CFR 50, Appendix B, with regard to ensuring an extremely low probability of leakage, of rapidly propagating failures, and of gross rupture. The controls to be placed on concentrations of leachable impurities in nonmetallic thermal insulation used on components of the ESF follow the recommendations of Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel." Conformance with this guide provides a basis for meeting the requirements of GDC 1, 14, and 31 and 10 CFR 50, Appendix B.

The requirements of GDC 4, 35, 41 and 10 CFR 50, Appendix B, have been met with regard to compatibility of ESF components with environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents.

The controls of the pH and chemistry of the reactor containment sprays and the emergency core cooling water following a loss-of-coolant or design-basis accident will be adequate to reduce the probability of stress corrosion cracking of austenitic stainless steel components and welds of the ESF systems in containment throughout the duration of the postulated accident to completion of cleanup.

lso, the controls of the pH of the sprays and cooling water, in conjunction with controls on selection of containment materials, will be in accordance with Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," and will provide assurance that the sprays and cooling water will not give rise to excessive hydrogen gas evolution resulting from corrosion of containment metal or cause serious deterioration of the materials in containment.

The controls that will be placed on component and system cleanup are in accordance with recommendations of Regulatory Guide 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants." These controls will provide assurance that the components and systems have been protected against damage or deterioration as required by 10 CFR 50, Appendix B.

The staff concludes that the engineered safety features materials specified in the RESAR SP/90 design are acceptable and meet the requirements of GDC 1, 4, 14, 31, 35, and 41 and 10 CFR 50, Appendices A and B to 10 CFR 50, and 10 CFR 50.55a.

6.1.2 Organic Materials (RESAR SP/90 Module 1, Section 6.1.2)

The staff verified that protective coatings applied to equipment inside the containment building will meet the testing requirements of ANSI Standard N101.2, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities," American National Standards Institute (1972), and the quality assurance guidelines of Regulatory Guide 1.54, "Quality Assurance Requirement for Protective Coatings Applied to Water-Cooled Nuclear Power Plants." Compliance with these requirements and guidelines provides assurance that the protective coatings will not fail under design-basis accident conditions and will not generate significant quantities of solid debris or combustible gas that could adversely affect the ESF. The staff reviewed the protective coating systems inside the containment building in accordance with SRP Section 6.1.2.

The applicant stated that the flow paths of the emergency water storage tank are specifically designed to accommodate virtually unlimited quantities of failed protective coatings and other debris without significantly blocking the containment sump screen or ingesting into the integrated safeguards systems. The applicant considers the protective coating systems inside the containment building to be Nuclear Service Level II. The applicant further states that quality assurance and documentation are not mandatory and proposes to use them only to the extent required by the project specification. The applicant has not provided information to verify that debris from failed protective coatings will not significantly block the containment sump screen or ingest into the integrated safeguards system. Nevertheless, the staff concludes that protective coating systems not meeting the quality assurance guidelines of Regulatory Guide 1.54 or test standards of ANSI Standard N101.2 will meet other regulatory criteria or requirements, including the acceptance criteria in SRP Sections 6.1.1, 6.2.2, 6.2.5, and 6.5.2 and the regulatory positions in Regulatory Guide 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

The staff determined that plant-specific applicants referencing RESAR SP/90 should either comply with the regulatory positions of Regulatory Guide 1.54 and the standards of ANSI Standard N101.2 or provide an analysis to show that debris from all failed protective coatings inside the containment building will not significantly affect the post-accident operation of the recirculation fluid systems. The applicant also should provide information on the total amounts of protective coatings that meet and do not meet the standards of ANSI Standard N101.2, and, if analysis of debris formation is made, other pertinent information for staff independent evaluation.

The staff concludes that plant-specific applicants referencing the RESAR SP/90 design should either use protective coating systems that meet the acceptance criteria of SRP Section 6.1.2 or provide justifications, with pertinent information and detailed analysis, for deviations from the acceptance criteria. To ensure the integrity of containment coating throughout the life of the plant, plant-specific surveillance and maintenance programs should be provided.

6.1.3 Post-Accident Emergency Cooling Water Chemistry (RESAR SP/90 Module 1, Section 6.5.2)

This review relates to providing and maintaining the proper pH of the containment sump water and recirculating water following a design-basis accident to reduce the likelihood of stress corrosion cracking of austenitic stainless steel. The staff reviewed the post-accident emergency cooling water chemistry in accordance with SRP Section 6.1.1 and BTP MTEB 6-1.

The initial core cooling and containment spray solution will be borated water with a nominal boron concentration, as boric acid, of 2500 parts per million. Baskets of trisodium phosphate will be placed in the containment compartment housing the reactor coolant drain tank and the reactor coolant drain tank pump. The trisodium phosphate will dissolve and mix with the containment spray water that drains into the reactor coolant drain tank compartment and then flows to the emergency water storage tank (containment sump). The final pH of the recirculating solution for emergency reactor core cooling and containment spray will be at least 7.0 when approximately 18,000 pounds of trisodium phosphate are placed in the baskets.

The interface requirement of a minimum pH of 7.0 for the post-accident emergency cooling water will be met by specifying the minimum amount of trisodium phosphate to be placed in the baskets above the containment sump by individual applicants referencing RESAR SP/90 and establishing a surveillance program that will ensure availability of the chemical additive for dissolution when needed. The staff determined that the interface criteria meet acceptance criterion II.B.a of SRP Section 6.1.1 and BTP MTEB 6-1.

Provided that the interface requirement will be met by plant technical specifications, the staff concludes that the post-accident emergency cooling water chemistry is acceptable.

6.2 Containment Systems (RESAR SP/90 Module 10, Section 6.2)

6.2.1 Containment Functional Design (RESAR SP/90 Module 10, Section 6.2.1)

6.2.1.1 Containment Structure (RESAR SP/90 Module 10, Section 6.2.1.1)

The containment is a free-standing, spherical, welded steel shell. The lower portion of the shell is encased between the building foundation concrete and the interior structure-based concrete, without any structural connection between the steel and the concrete. The containment completely encloses the entire reactor, steam generators, reactor coolant loops, select major components of the engineered safety features, and the emergency water storage tank.

RESAR SP/90 provides containment analyses on a spectrum of reactor coolant system and secondary system pipe ruptures to establish the containment functional design pressure and temperature and to establish the pressure and temperature conditions for environmental qualification of safety-related equipment located inside containment. The containment pressure and temperature analyses for primary and secondary breaks were performed using the Westinghouse computer code COCO, with the assumption of the most limiting single active failure and the availability or nonavailability of offsite power, depending on which assumption results in the highest containment temperatures and pressures.

The spectrum of breaks in the reactor coolant system included a doubled-ended guillotine break in the hot leg, double-ended guillotine breaks in the cold leg and the reactor coolant pump suction, and a 0.6-ft^2 double-ended and 3-ft^2 split break in the pump suction line. For the double-ended guillotine pump suction break, both minimum and maximum emergency core cooling system (ECCS) flow were considered. The design-basis break was determined to be the double-ended hot leg rupture with operation of maximum ECCS equipment and failure of one containment spray pump. This accident resulted in the highest containment pressure of 36.4 psig, versus the containment design pressure of 46.9 psig, providing a design margin of 28.8 percent.

The spectrum of secondary system breaks included various sizes of double-ended and split breaks of the main steam line at five different power levels from 0 to 102 percent. Main feedwater line breaks were not provided as the results of these are considered to be bounded by those of steam line breaks. The most limiting single active failure was determined to be, and assumed in the analysis, the loss of one emergency diesel, which results in the loss of one train each of the containment heat removal systems. The maximum containment pressure (33.22 psig) resulted from the postulated double-ended rupture of a steam line at hot zero power. The maximum containment temperature (315°F) resulted from the postulated 1.1-ft^2 split rupture of a steam line at 102-percent power.

The staff concludes that Westinghouse has satisfactorily demonstrated the adequacy of the RESAR SP/90 containment functional design for a LOCA or a main steam line break.

The containment is designed for an external pressure of 2.0 psig. Inadvertent operation of the spray system would cause a reduction in the containment pressure during normal plant operation. In a response to the staff's request for additional information (RAI), Westinghouse indicated that no analysis had been performed. However, based on comparison with similar facilities and because the refueling water storage tank will be located inside the containment, the temperature difference between the containment atmosphere and spray water is less than that for a conventional plant. Therefore, it is expected that this pressure would be less than -2.0 psig. On the basis of its review, the staff agrees with Westinghouse's rationale and finds the containment design external pressure of 2.0 psig acceptable.

6.2.1.2 Subcompartment Analysis (RESAR SP/90 Module 10, Section 6.2.1.2)

Subcompartment analyses are required to determine the acceptability of the design differential pressure loadings on containment internal structures from high-energy line ruptures. The effects of these breaks in containment subcompartments will be analyzed in the FDA application to establish criteria for the structural design of the compartment walls. The staff will review these subcompartment analyses against the guidance of SRP Section 6.2.1.2, "Subcompartment Analysis," during the FDA review.

6.2.1.3 Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (RESAR SP/90 Module 10, Section 6.2.1.3)

Westinghouse provided the mass and energy release rate data for a spectrum of reactor coolant pipe break sizes and locations that are the basis for the containment functional analyses. The break locations include the largest cold and hot leg breaks and a range of pump suction breaks.

The effect of single failures of various ECCS components on the mass and energy releases is included in these data. Two analyses bound this effect for the pump suction double-ended rupture. No single failure is assumed in determining the mass and energy releases for the maximum safeguards case. For the minimum safeguards case, the single failure assumed is the loss of one emergency diesel generator. This failure results in the loss of one safety injection train. The analysis of both maximum and minimum safeguards cases ensures that the effect of all credible single failures is bounded. A single failure analysis is not performed for the hot leg double-ended rupture since the ECCS has no effect on the maximum containment pressure, which occurs at the end of blowdown. The staff has reviewed the spectrum of breaks and the single failures considered in the RESAR SP/90 application and finds them acceptable.

The methods used to compute the mass and energy release rates from postulated reactor coolant pipe breaks for containment functional analyses are documented in Westinghouse's Topical Report WCAP-8312A, "Westinghouse Mass and Energy Release Data for Containment Design," which has been approved by the NRC. Therefore, the mass and energy release rate data for postulated reactor coolant pipe breaks are acceptable for use in the containment function analyses.

6.2.1.4 Mass and Energy Release Analysis for Postulated Secondary Pipe Ruptures Inside Containment (RESAR SP/90 Module 10, Section 6.2.1.4)

Westinghouse computed the mass and energy release rates for a spectrum of postulated main steam line breaks (MSLBs). The spectrum includes both double-ended and split ruptures at power levels of 102 percent, 75 percent, 50 percent, 25 percent and at hot shutdown conditions. The analyses included the effect of main and auxiliary feedwater additions, reactor coolant system metal heat storage, steam generator reverse heat transfer, and uninsulated feedwater line volumes. In addition to assuming a loss of offsite power, the analyses considered three single active safety system failures; loss of one emergency diesel generator, failure of one main steam isolation valve, and failure of one main feedwater isolation valve. The loss of one emergency diesel generator is the worst single active failure.

Mass and energy release rates for the spectrum of MSLBs were calculated using the method as documented in Westinghouse's Topical Report WCAP-8822, "Mass and Energy Release Following a Steam Line Rupture," which has been approved by the NRC.

The staff concludes that the MSLB mass and energy release rate data listed in the RESAR SP/90 application are acceptable for the containment functional analysis.

6.2.1.5 Minimum Containment Pressure Analysis for Performance Capability Studies on the ECCS

10 CFR 50, Appendix K, "ECCS Evaluation Models," states in part that the containment pressure used for evaluating cooling effectiveness during reflood and spray cooling shall not exceed a pressure calculated conservatively for this purpose. It further requires that the calculation includes the effects of operation of all installed pressure reducing systems and processes. BTP CSB 6-1, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation," provides additional guidance for the performance of a minimum containment pressure analysis and should be used when the analysis is performed.

In the response to the staff's RAI (by a letter dated June 14, 1988), Westinghouse provided an analysis performed in accordance with 10 CFR 50, Appendix K, to conservatively calculate the minimum containment pressure. The results of the analysis indicated a peak containment pressure of approximately 38.5 psia with decay to approximately 25.5 psia at 284 seconds. Westinghouse stated that, based on this analysis, it used a conservatively low, constant containment pressure of 24.7 psia throughout the entire RESAR SP/90 ECCS analyses.

The assumptions for initial containment temperature, relative humidity, and pressure used in the analysis were 90°F, 100 percent, and 14.7 psia, respectively. A conservatively low value of -2.2°F was used for the external containment temperature. The conservative containment free volume used for the analysis was 3.17×10^6 cubic feet. The values for the heat capacities and conductivities used for the various structural heat sinks inside containment are in good agreement with those used in other minimum containment pressure analyses.

The pressure reducing equipment used for the RESAR SP/90 minimum containment pressure analysis consisted of containment spray pumps and fan coolers. All four containment spray pumps were assumed to operate at a conservative runout flow rate for the pressure analysis, and all four containment fan coolers were assumed to operate.

The staff concludes that the above input parameters used in the minimum containment pressure analysis are acceptably conservative and in conformance with BTP CSB 6-1 and 10 CFR 50, Appendix K. Therefore, the above minimum containment pressure of 24.7 psia is acceptable for use in the RESAR SP/90 ECCS analyses.

6.2.2 Containment Heat Removal Systems (RESAR SP/90 Module 10, Section 6.2.2)

The function of the containment heat removal systems is to remove heat from the containment atmosphere to limit, reduce, and maintain at acceptably low levels the containment pressure and temperature following a LOCA or secondary system pipe rupture. In addition to the passive means of heat removal (heat transferred by/or stored in walls, structures, and equipment), the design includes two separate, active containment heat removal systems. These are the containment fan cooler system and the containment spray system.

The containment fan cooler system will consist of multiple recirculation cooling units each connected to an associated recirculation fan. The containment atmosphere will be drawn through the cooling units by the recirculation fans, heat will be transferred to the component cooling water system (CCWS) by air-to-water heat exchange coils, and the cool air will be discharged to the lower levels of the containment. Two coolers will be cooled by CCWS train A and two coolers will be cooled by train B. During normal operation, three of the four coolers will operate at high speed to maintain air temperature below the design value. Upon receipt of a safety injection signal, one of the train A and one of the train B coolers will start low speed operation, the rest of the coolers will be tripped, and the CCWS water flow to the operating coolers will be increased to accommodate the higher heat transfer. However, Westinghouse has not provided an analysis to show the calculated pressure differential across the ductwork and housing of the coolers during accident conditions to verify that they have been designed to withstand the maximum pressure differential. In response to the staff's RAI of December 17, 1987, Westinghouse indicated that the necessary analyses will be performed during the FDA review. The staff will review the analyses, when available, and report the findings in the SER for the RESAR SP/90 FDA.

The containment spray function will be performed by the integrated safeguards system (ISS). Those components within the ISS that will perform a containment spray function are the four low head pumps, the EWST, and the associated valves, piping, and instrumentation. Two redundant sets of spray ring headers will be used to spray the containment atmosphere. In the event of a high containment pressure signal, the four low-head pumps will receive an automatic signal to start and the containment spray header isolation valves will open. The low-head pumps will draw water from the EWST and deliver it to the containment spray headers located at the top of the containment.

In the RESAR SP/90 application, Westinghouse indicates that the net positive suction head (NPSH) calculations for the low-head pumps were performed in accordance with the recommendations of Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps," November 1970. In addition, in a response to the staff's RAI, Westinghouse provided for staff review a summary of the analysis performed to verify that adequate NPSH is available for the containment spray system pumps following a LOCA or MSLB. The staff finds the available NPSH for the containment spray pumps acceptable.

The staff concludes that the containment heat removal systems, in conjunction with the passive containment heat sinks, are capable of removing sufficient heat from the containment atmosphere following a LOCA or a MSLB to maintain the containment pressure below the design value. However, the two systems should be capable of providing a rapid cooldown of the containment atmosphere following a design-basis accident (DBA) to satisfy requirements of GDC 38. It has been the staff's position that the containment pressure be reduced to less than 50 percent of the calculated pressure for the DBA within 24 hours after the postulated accident. RESAR SP/90 application indicates that, 24 hours after the postulated accident, the containment pressure is significantly higher than 50 percent of the calculated peak pressure. In a response (by a letter dated June 14, 1988) to the staff's RAI, Westinghouse indicated that there is a significant amount of margin available between the calculated peak containment pressure and the containment design pressure (36.4 psig compared to 46.9 psig, respectively). Therefore, Westinghouse considers that the guideline should be reduction of the containment pressure to 50 percent of the design pressure within 24 hours, rather than to 50 percent of the calculated peak pressure. Westinghouse's approach is consistent with that in the Electric Power Research Institute Requirements Document (EPRI-RD). The approach, which represents a deviation from the guideline of the SRP, is under consideration by the staff in its review of EPRI-RD. The staff's resolution of this matter will apply to RESAR SP/90.

The containment heat removal systems satisfy the provisions of Regulatory Guides 1.26 and 1.29 and meet or invoke the design, quality assurance, redundancy, power source and instrumentation and control requirements for engineered safety features. The RESAR SP/90 application provides single-failure analyses and other information demonstrating the ability of the containment heat removal systems to function following postulated single-active failures.

Regulatory Guide 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems," provides guidelines to be met by reactor building sumps that are designed to be sources of water for the ECCS and the containment spray system following a LOCA. There will be four independent and physically separated sump pits provided in the in-containment emergency water storage tank (EWST). Each of these will be assigned to a separate ISS subsystem. The suction piping for one high-head pump and one low-head pump will be connected to its individual EWST pit. Large vertical trash racks and fine screens will be provided for each of the four EWST sump pits to collect debris. A minimum water level will be maintained in the EWST if a LOCA should occur to ensure adequate NPSH, low screen approach velocities, and no vortexing. The sump design meets the intent of the guidelines of Regulatory Guide 1.82.

In the RESAR SP/90 application, Westinghouse states that the containment heat removal systems are designed to permit periodic inspection and testing in accordance with GDC 39 and 40.

Pending the successful resolution of the above matters, the staff concludes that the containment heat removal systems for the RESAR SP/90 design are acceptable.

6.2.3 Secondary Containment Functional Design (RESAR SP/90 Module 10, Section 6.2.3)

The function of the secondary containment is to collect any fission products that could leak from the primary containment structure into the annular secondary containment air volume (annulus) and contiguous areas following a design-basis LOCA and is designed to provide radiation shielding as well as missile protection for the steel containment and other safety-related features. The system will consist of a reinforced concrete shield building and a wrap-around reactor external building surrounding the containment, adjacent vaults, and penetration areas. An annulus air cleanup system will maintain a pressure lower than ambient in the annulus to prevent the uncontrolled release of radioactivity into the environment.

In meeting the requirements of GDC 4 to protect structures, systems, and components important to safety against dynamic effects resulting from high-energy line ruptures, as part of the design bases, Westinghouse states in the RESAR SP/90 application that the secondary containment is designed to withstand the transient pressure and temperature conditions as a result of either a LOCA within the primary containment or a high-energy line rupture within the secondary containment. However, the evaluation for such an event will be conducted at the FDA stage. The staff will review this evaluation during the FDA stage to ensure Westinghouse's commitment of meeting the requirements of GDC 4.

The annulus air cleanup system is designed in accordance with the guidelines 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," and is able to maintain the design negative pressure differential of 0.50 inch wg (water gauge). All openings, such as personnel doors and equipment hatches, are under administrative control with position indicators and alarms having readout and alarm capability. Therefore, the requirements of GDC 16 are satisfied.

The fraction of primary containment leakage bypassing the secondary containment and escaping directly to the environment is conservatively assumed to be all the leakage attributed to penetrations and isolation valves requiring Type B and C tests per 10 CFR 50, Appendix J. This leakage is limited to 0.60 of the total allowable integrated containment leakage. This is a conservative approach to meeting the guidelines of BTP CSB 6-3. Therefore, the requirements of GDC 43 and 10 CFR 50, Appendix J, are satisfied.

The staff concludes that the secondary containment functional design, with the exception noted above, meets the requirements of GDC 16 and 43 and 10 CFR 50, Appendix J, and is acceptable.

6.2.4 Containment Isolation System (RESAR SP/90 Module 10, Section 6.2.4)

The function of the containment isolation system is to allow the normal and emergency passage of fluids through the containment boundary while preserving the ability of the boundary to prevent or limit the escape of fission products that may result from accidents. The containment isolation system includes the portions of all fluid systems penetrating the containment that perform the isolation function. Each penetration is designed so that in the event of a single failure, the containment integrity will be maintained.

The containment isolation system will be part of the leak-tight barrier against uncontrolled release of radioactivity to the environment, thus satisfying the requirements of GDC 16.

The containment isolation system (1) will be protected from the effects of earthquakes, tornadoes, hurricanes, floods, and external missiles; (2) is designed to remain functional after a safe shutdown earthquake (SSE) and to perform its intended function following the postulated hazards of internal missiles and pipe breaks; and (3) is designed and will be fabricated to codes consistent with the quality group classification of Regulatory Guide 1.26 (Quality Group B), and the seismic category assigned by Regulatory Guide 1.29 (seismic Category 1). The requirements, therefore, of GDC 1, 2, and 4 are satisfied.

According to the design bases, each line that will penetrate the containment and be either part of the reactor coolant pressure boundary (RCPB) or connect directly to the containment atmosphere will be provided with containment isolation valves in accordance with GDC 55 and 56. The isolation valves outside the containment will be located as close to the containment as practical, and upon loss of actuating power, air-operated automatic isolation valves will fail closed.

ECCS and containment spray system lines will be provided with remote-manual valves and will not be automatically operated on a containment isolation signal. Provisions will be made to detect possible leakage from these lines outside containment.

"Closed" systems (except instrument sensing lines) will be provided with a containment isolation valve in accordance with GDC 57. "Closed" systems are those that are not part of the RCPB or not connected directly to the containment atmosphere, but that meet the additional design requirements of protection against missiles and high-energy line breaks, seismic Category I standards; ASME Code, Section III, Safety Class 2 classification; and design temperature and pressure equal to that of the containment.

Instrument lines will be provided with isolation valves in accordance with GDC 55 and 56 and guidelines provided in Regulatory Guides 1.11, "Instrument Lines Penetrating Primary Reactor Containment," and 1.141, "Containment Isolation Provisions for Fluid Systems." Instrument line systems, closed inside and outside containment, are designed to withstand the containment temperature and pressure conditions following a LOCA and to withstand dynamic effects.

The staff finds the design provisions of the containment isolation system in accordance with the requirements of GDC 55, 56, and 57.

The staff reviewed the initiation signals of the containment isolation system to determine conformance to the requirement of GDC 54 and 10 CFR 50.34(f)(XIV) with regard to diversity.

The containment isolation system is designed as a phased isolation system. The Phase A signal, which will isolate the majority of containment fluid line penetrations, will be initiated by any signal initiating a safety injection, containment high radiation, high-1 containment pressure, or manual actuation. The Phase B containment isolation signal, which will isolate the component cooling water lines from the reactor coolant pump motors and thermal barriers, will be actuated by a high-3 containment pressure signal or manually. The containment ventilation isolation signal will be used to isolate the containment purge valves and will be initiated by a Phase A signal. The purge valves also will close on high radiation level in the containment. The steam line isolation signal will be used to isolate the steam generators to prevent excessive cooldown of the reactor coolant system or overpressurization of the containment if a steam line failure occurs. The signal will be initiated by a high-3 containment pressure signal or manually. If the steam line break results in high steam line pressure rate or high-2 containment pressure, only the steam generators will be isolated. If the steam line break causes a low steam line pressure, a containment isolation signal will be generated as well as the steam line isolation signal. The staff concludes that there is adequate diversity of parameters sensed to initiate containment isolation, thus satisfying the GDC 54 aspect of reliable isolation capability.

In addition, the following aspects of the containment isolation system design, established in the RESAR SP/90 application, are in conformance with requirements of GDC 54, with respect to isolation and containment capability:

- Automatic isolation valves are provided in those lines that must be isolated immediately following an accident.

- For those valves for which automatic closure is not desired remote-manual operation is available from the control room.

- Each power-operated isolation valve may also be controlled independently by positioning hand switches in the control room.

- The use of motor-operated valves which fail as is upon loss of actuating power is based on the consideration of that valve position that ensures the greatest plant safety.

- Containment isolation valves required to be operated after a DBA are powered by the Class IE electric power system.

- Deliberate manual actuation is required to change the position of containment isolation valves in addition to resetting the original actuation signal. The design does not allow ganged reopening of containment isolation valves.

- Piping and valves associated with containment penetrations are designed and located to permit preservice and inservice inspection in accordance with the ASME Code, Section XI. Each line penetrating containment is also provided with testing features to allow containment leak tests in accordance with 10 CFR 50 Appendix J.

- The design includes provisions to allow the operator in the control room to know when to isolate fluid systems that are equipped with remote-manual isolation valves. The provisions include instruments to measure sump water level, system flow, temperature, and pressure.

RESAR SP/90 states that the closure time of all containment isolation valves required to close upon receipt of an engineered safeguards actuation signal is 10 seconds or less. In addition, the main containment purge system (42-inch) isolation valves are open only during cold shutdown conditions and, as such, are not required to shut against a post-LOCA containment pressure. Thus the containment isolation valves closure times are acceptable and the main purge system satisfies the guidelines of BTP CSB 6-4.

The staff concludes that the containment isolation system meets the requirements of GDC 1, 2, 4, 16, 54, 55, 56, and 57 and 10 CFR 50.34(f) (XIV) and satisfies provisions of Regulatory Guide 1.141 and conforms to staff regulatory positions. The design of the system is therefore acceptable.

6.2.5 Combustible Gas Control System (RESAR SP/90 Module 10, Section 6.2.5)

Following a LOCA, hydrogen may accumulate within containment as a result of (1) metal-water reaction between the Zircaloy fuel cladding and the reactor coolant, (2) radiolytic decomposition of the water in the reactor core and the sump, and (3) corrosion of aluminum and zinc by emergency core cooling and containment spray solutions. To monitor and control the buildup of hydrogen within containment, RESAR SP/90 design provides the following systems and mechanisms:

- hydrogen recombiner system
- hydrogen igniter system
- containment hydrogen monitoring system
- containment hydrogen mixing
- containment hydrogen purge system

The hydrogen recombiner system will consist of two redundant electric hydrogen recombiners located inside containment. Each recombiner will be powered from a separate safeguards bus and will be provided with separate power and control panels. The recombiners will be operated and controlled manually from the control building. The system is classified seismic Category I.

The hydrogen monitoring system will consist of two identical, independent units powered from independent Class 1E power sources. The system is classified seismic Category I, Class 1E, and is designed to maintain its integrity and operability under all conditions following a design-basis accident, degraded core accident, or core melt accident. The system piping is designed in accordance with the criteria of Regulatory Guide 1.26, Quality Group B. The hydrogen monitoring system will have a range of 0 to 20 percent volume concentration over the pressure range of -2 to 60 psig. The output signal of the hydrogen monitoring system will be indicated locally and will record and alarm in the control room.

The hydrogen igniter system will control hydrogen concentration during and following degraded core and core melt accidents when rapid release of hydrogen cannot be controlled by the hydrogen recombiners. It is designed to safely accommodate hydrogen generated by the equivalent of 100-percent metal-water reaction and limit uniformly distributed hydrogen concentration in the containment to 10 percent during and following an accident. The original design provided that the igniters will be powered from Class 1E panels that will have normal and alternate ac power supplies from offsite sources, however, current commitments for severe accident purposes call for d.c. powered igniters. The new design will be reviewed during the plant specific licensing process. The system is classified seismic Category I.

The containment purge system for normal operation also is designed to aid in hydrogen control, if necessary, during and following a design-basis accident. The system is not designed seismic Category I, except for portions of the system that constitute part of the containment boundary. However, Westinghouse has not described how the containment hydrogen purge system has been designed in accordance with the guidelines of Regulatory Guide 1.7. Therefore, the staff is not able to conclude that the containment purge system for hydrogen control is acceptable.

All combustible gas control systems will be subject to periodic operational tests and inspections in accordance with plant-specific procedures and technical specifications.

Westinghouse analyzed the production and accumulation of hydrogen within containment using the guidelines provided by Regulatory Guide 1.7 and has shown that a single recombiner is sufficient to limit the hydrogen concentration in the containment to below the Regulatory Guide 1.7 lower flammability limit of 4.0 volume percent.

Hydrogen mixing to prevent excessive stratification and to eliminate areas of potential stagnation will be provided by the containment fan coolers and by natural convection processes. Mixing also will be assisted by the containment spray system when it is operating. During post-LOCA operation at least two of the four fan coolers will be discharging air to the lower elevations of the containment. The return air will be drawn from the upper part of the containment for recirculation, creating an upward flow within the containment. The discharge velocity of the air and natural circulation will provide additional means of atmospheric mixing. The staff concludes that the active and natural means that will be provided for hydrogen mixing can ensure essentially uniform concentrations of hydrogen and limit the potential of local hydrogen pocketing.

Pending satisfactory resolution of the staff's concern with regard to the hydrogen purge system, the staff concludes that the combustible gas control systems for the RESAR SP/90 design comply with the requirements of 10 CFR 50.44 and the provisions of Regulatory Guide 1.7 and GDC 41, 42, and 43, and is therefore acceptable.

6.2.6 Containment Leakage Testing (RESAR SP/90 Module 10, Section 6.2.6)

The applicant states that the reactor containment, containment penetrations, and containment penetration barriers will be designed to permit periodic leakage rate testings required by GDC 52, 53, and 54. However, compliance with the requirements of 10 CFR 50, Appendix J, to perform Type A, B, and C testing will be shown in the FDA application. The staff will evaluate the containment leakage testing program for the RESAR SP/90 design during the FDA stage of review and report its findings in the SER for the FDA.

6.2.7 Fracture Prevention of Containment Pressure Boundary (RESAR SP/90 Module 7, Section 3.8)

GDC 51 requires the reactor containment boundary to be designed with sufficient margin to ensure that under operating, maintenance, testing, and postulated accident conditions (1) the ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly progagating fracture is minimized. The ferritic materials of the containment pressure boundary that are evaluated by the staff are those that are applied during the fabrication of the containment vessel, equipment hatch, personnel access locks, penetrations, and fluid system components, including the valves required to isolate the system. Some of the components reviewed are parts of the containment system that are not backed by concrete but must sustain loads during the performance of the containment function under the conditions cited by GDC 51.

The fracture toughness requirements for containment pressure boundary ferritic materials are those identified for Class 2 components in the Summer 1977 Addenda of Section III or later editions and addenda of Section III of the ASME Code. The applicant has stated that "the materials used in the design of all RESAR SP/90 containment pressure boundary components meet the requirements of Class 2 components of the Summary 1977 Addenda of ASME Code, Section III, or will meet the applicable fracture toughness criteria that may supersede the 1977 Addenda prior to the RESAR SP/90 FDA application." On the basis of this statement, the staff concludes that compliance with Code requirements will provide reasonable assurance that the containment pressure boundary materials will behave in a nonbrittle manner, that the probability of rapidly propagating fracture will be minimized, and that the requirements of GDC 51 will be satisfied.

6.3 Emergency Core Cooling System (RESAR SP/90 Module 1, Section 6.3)

The staff has reviewed the emergency core cooling system (ECCS), which is part of the integrated safeguard system (ISS) for the RESAR SP/90 design, in accordance with SRP Section 6.3. The staff performed a review of each of the areas listed in paragraph I, "Areas of Review," of SRP Section 6.3 in accordance with the guidelines provided in the SRP review procedures. The principal bases for the staff's review of this system are conformance to 10 CFR 50.46, Appendix K to 10 CFR 50, and GDC 2, 5, 17, 19, 20, 35, 36, and 37.

6.3.1 System Design and Functional Requirements

The ISS is designed as a safety-related system that integrates the functions of residual heat removal, emergency core cooling, and containment spray. It will consist of high- and low-head pumps, residual heat removal heat exchangers, an emergency water storage tank, accumulators, core reflood tanks, and associated piping, valving, and instrumentation. The ISS emergency core cooling will cool the reactor core as well as provide additional shutdown capability following a LOCA. The subsystem functional parameters were selected so that, when integrated, the requirements of Appendix K to 10 CFR 50 and 10 CFR 50.46 are met for a LOCA with a single failure in the ECCS. The reliability of the system has been considered in the design of the ECCS, and redundant and diverse subsystems are provided to enhance the overall reliability of the ECCS.

The four injection trains will be independent of each other except that each train will be powered by an emergency diesel generator that will be common to one other train. Each train will contain one high-head pump, one low-head pump, and a residual heat removal heat exchanger in the common discharge line that will lead to a reactor vessel injection nozzle. The suction lines of the high-head pump and the low-head pump will both be connected to the emergency water storage tank inside the containment. The high-head pumps are the primary ECCS pumps. The low-head pumps are normally used as the residual heat removal and containment spray pumps. During a LOCA when the containment spray function is not required, the low-head pumps will be used to inject borated water from the emergency water storage tank to the reactor vessel to perform their ECCS function. The low-head pumps also are used for core reflooding and long-term core cooling during the post-LOCA recirculation mode.

As discussed above, one high-head pump and one low-head pump in each ECCS subsystem will share a common discharge header that will lead to a reactor vessel injection nozzle. The low-head pump discharge flow path will normally be lined up with the containment spray header. There will be a normally closed motor-operated isolation valve in the low-head pump discharge line upstream of the common header. The high-head pump will start on a safety injection signal to serve its safety injection function and the low-head pump will start on the containment isolation B signal to serve the containment spray function. During a LOCA, when the containment spray function will not be required, the low-head pumps can be used to inject borated water from the EWST to the reactor vessel to perform their ECCS function. When both the high-head pump and low-head pump are operated in parallel for ECCS flow injection, the potential exists for the low-head pump to operate "dead-headed" because of a high discharge pressure in the common discharge header. To preclude "dead-heading" of the low-head pump, the RESAR SP/90 design provides each low-head pump with a non-isolable miniflow path, including a miniflow heat exchanger, which will be continuously supplied with component cooling water and will be sized for this mode of operation.

Since both the onsite electric power supply system and the offsite electrical power supply system are provided to permit functioning of the ECCS, the ECCS design meets the requirements of GDC 17.

The EWST will be located at the lower elevation inside the containment building. In the event of a LOCA, the reactor coolant will "spill" to the EWST, thus establishing a continuous source of coolant for the high- or low-head pumps. This design eliminates the need that exists in conventional designs to switch over from the refueling water storage tank to the containment sump for long-term recirculation. Analyses will be performed to determine the minimum water level in the EWST during recirculation during the FDA stage of design. These analyses will consider the amount of water trapped in lower containment compartments and the delay time for water to return to the EWST. The staff will review the results of these analyses during the FDA review.

There will be four accumulators provided for rapidly refilling the reactor vessel following an intermediate or large break LOCA. The accumulators will provide passive injection of water to the RCS via the four RCS cold legs. Each of the four accumulators will be connected to individual cold legs of the RCS. The accumulators are designed to inject water to the RCS at a pressure similar to the setting for a plant with a conventional design. Four core reflood tanks will be provided with the discharge line of each tank connected to one of the four reactor vessel injection nozzles. Similar to the accumulators, the core reflood tanks will provide passive water injection to the reactor vessel, except that the reflood tanks will have slightly less capacity than the accumulators and are designed to inject water to the reactor at a low RCS pressure.

One RHR heat exchanger will be installed in each of the four ECCS pumping trains at the discharge header for high- and low-head pumps. These heat exchangers will be used to remove heat from the reactor coolant during operation in the post-LOCA recirculation mode.

The low-head pumps and the accumulators discussed above will be used as the backup emergency core cooling features. No credit is assumed for these systems in the analyses that are provided to satisfy the requirements of Appendix K to 10 CFR 50 and 10 CFR 50.46.

The four high-head pumps will be sized such that one high-head pump will provide sufficient injection flow to prevent core uncovery for small LOCAs up to at least a 6-inch-diameter break size. The four core reflood tanks and the four high-head pumps will provide eight separate means for injecting water directly into the reactor vessel. Any combination of five of these eight entities are sufficient to meet the large break LOCA functional requirements.

As specified in SRP 6.3, paragraph II, the ECCS system will be initiated either manually or automatically on (1) low pressurizer pressure, (2) high containment pressure, or (3) low steam line pressure. This meets the requirements of GDC 20.

The ECCS will have the capability to be manually actuated, monitored, and controlled from the control room, as required by GDC 19. The ECCS will be supplemented by instrumentation that will enable the operator to monitor and control the ECCS equipment following a LOCA so that adequate core cooling may be maintained. The acceptability of the proposed ECCS instrumentation and controls is addressed further in Section 7.3 of this report.

RESAR SP/90 analyses to determine the net positive suction head (NPSH) available from the EWST to the high- and low-head ECCS pumps was performed with sufficient margin and meets the regulatory position stated in Regulatory Guide 1.1.

As required in SRP Section 6.3, paragraph III.11, the valve arrangement on the ECCS discharge lines has been reviewed with respect to adequate isolation between the RCS and the low-pressure ECCS. This isolation will be provided by two check valves in series with a motor-operated isolation valve in each ECCS discharge line. The isolation valves will be open during normal plant operation. This arrangement is acceptable because periodic leak testing across each check valve will be performed during plant operation per the requirements of the ASME Code, Section XI, for inservice inspection.

Containment isolation features for all ECCS lines, including instrument lines, and the requirements of GDC 56 and Regulatory Guide 1.11 are discussed in Section 6.2.4 of this SER.

During normal plant operation, the ECCS lines will be maintained in a filled condition. Maintaining the lines in a filled condition will minimize the likelihood of water hammer during system initiation.

As specified in SRP Section 6.3, paragraph II.B, no ECCS components will be shared between units, which meets the requirements of GDC 5.

6.3.2 Evaluation of Single Failures (RESAR SP/90 Module 1, Section 6.3.2)

As specified in SRP Section 6.3, paragraph II, the staff has reviewed the system description and piping and instrumentation diagrams to verify that sufficient core cooling will be provided following a LOCA with and without offsite power, assuming a single failure.

The worst single failure to affect ECCS performance is the failure of one diesel generator. Following this postulated single failure, two high-head pumps, two low-head pumps, four accumulators, and four core reflood tanks will be available for operation. This is more than sufficient ECCS flow required during a post-LOCA operation, as discussed in Section 6.3.1 of this SER.

The staff reviewed the plant's capability for hot leg injection during the long-term-recirculation phase to preclude excessive buildup of boron concentration in the reactor vessel. The staff concludes that there is sufficient redundancy in the design of the injection lines and pumps to ensure adequate hot leg injection. This meets the requirements of SRP Section 6.3, paragraph III.6.

The staff concludes that the ECCS of the RESAR SP/90 design complies with the single-failure criterion of GDC 35.

6.3.3 Design Qualification of Emergency Core Cooling System (RESAR SP/90 Module 1, Section 6.3.2)

The ECCS is designed to seismic Category I requirements in compliance with Regulatory Guide 1.29. The structures in which the system is housed are designed to withstand a safe-shutdown earthquake (SSE) and other natural phenomena, as discussed in Section 3 of this SER. The ECCS protection against missiles and against pipe whip inside and outside containment also are discussed in Section 3 of this SER.

The ECCS design to permit periodic inspection in accordance with ASME Code, Section XI, which constitutes compliance with GDC 36, is discussed in Section 6.6 of this SER.

6.3.4 Testing (RESAR SP/90 Module 1, Sections 6.3.4.1 and 6.3.4.2)

Westinghouse has committed to demonstrate the operability of the ECCS by subjecting all components to preoperational and periodic testing, as required by Regulatory Guides 1.68 and 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors," and GDC 37.

Preoperational testing of the ECCS will be performed with the system cold and the reactor vessel head removed following system flushing and hydrostatic testing. Subsequent system performance testing will be conducted during refueling with each subsystem aligned to take suction from the EWST and to discharge to the EWST via the system test line. Each pump also can inject into the reactor vessel with the overflow from the reactor vessel spilling into the refueling canal. The ECCS will be tested in conjunction with the emergency diesels to verify the performance of the safety injection signal generation and transmission, ECCS pump starting times, and pump flow rates. ECCS pump head and flow characteristics will be verified for both the reactor vessel injection and hot leg injection modes of operation. The preoperational test also includes the discharge of accumulator and core reflood tanks into the reactor vessel with its head off.

Periodic testing of ECCS components and their supporting systems will be performed, and ECCS pumps will be tested using miniflow lines. Valves necessary for post-LOCA operation will be exercised through a complete cycle. The staff evaluation of the miniflow capacity is addressed in Section 6.3.1 of this SER.

Following completion of any required maintenance, a full-flow system test line will be used to verify the ECCS functional flow requirements with the reactor at power.

The plant-specific technical specifications referencing the RESAR SP/90 design will specify requirements for the frequency of the periodic testing and the acceptability of testing. A description of the inservice inspection program will be included in the plant-specific technical specifications to meet the requirements of ASME Code, Section XI, for inservice inspection.

6.3.5 ECCS Performance Evaluation

The ECCS is designed to deliver fluid to the RCS to limit the fuel cladding temperature following LOCAs that require ECCS actuation. The ECCS also is designed to remove the decay and sensible heat during the recirculation mode.

The staff reviewed conformance of the ECCS design to criteria of 10 CFR 50.46. However, the Westinghouse LOCA evaluation model approved by the staff may not be applicable to the RESAR SP/90 design with regard to plant-specific configurations in node arrangement and control systems. In response to the staff request, Westinghouse stated that the Westinghouse 1981 evaluation model with BASH noding was used in the RESAR SP/90 ECCS large break analyses. Additional nodes were required in the upper plenum to predict better transient behavior and to account for the presence of the core reflood tanks. The RESAR SP/90 large break LOCA evaluation model with BASH will be submitted to the NRC for review and approval concurrent with the RESAR SP/90 FDA application. The staff finds this commitment acceptable and will review the model during the FDA stage.

RESAR SP/90 Module 1 Section 6.3.2 stated that the four core reflood tanks and the four high-head pumps will provide eight separate means for injecting water directly into the reactor vessel. Any combination of five of these eight entities are sufficient to meet the large break LOCA functional requirements. However, in RESAR SP/90 Module 1 Sections 6.3.3 and 15.6.4, the ECCS flow assumption used in the LOCA analysis includes all ECCS pumps, accumulators, and reflood tanks. The staff requested clarification of this discrepancy and also requested a list of the minimum required ECCS flow to the reactor vessel for LOCAs of various break sizes and confirmation that the ECCS design for RESAR SP/90 meets the requirements of GDC 35 and 10 CFR 50.46. The Westinghouse response was transmitted by letter dated June 28, 1989 (Letter No. NS-NRC-89-3446) and covered several RESAR SP/90 open items.

The Westinghouse response stated the following:

> In general, Westinghouse does not establish minimum flows for LOCAs for a large number of break sizes because it would require an excessive amount of computer analysis.
>
> Instead, the following procedure is normally used in the case of a new design such as the SP/90:
>
> (1) single point high head and low head flow requirements are established based on what have historically been the most limiting small and large LOCA's.
>
> (2) safety injection system pumps and tanks are sized using the above flow requirements.
>
> (3) LOCA sensitivity analyses for a wide spectrum of breaks are performed to demonstrate that applicable criteria are met.
>
> (4) if applicable criteria are not met, sizes of safety injection pumps and tanks are adjusted and LOCA analysis are repeated as necessary.

In the case of the RESAR SP/90 design, the most limiting small break was judged to be the rupture of one of the direct vessel injection lines because it would leave only a single high-head safety injection (HHSI) pump available for delivery to the reactor coolant system. In addition, Westinghouse established an internal criteria that there should be no core uncovery for this particular small break. This combination of sizing criteria led to a flow requirement of 65 lbm/sec at 1000 psia, which in turn resulted in the pump described in RESAR SP/90 Module 1, "Primary Side Safeguards System," Table 6.3-2 (Sheet 1 of 6) and Figure 6.3-4. The capability of this pump to meet the above requirements is demonstrated in RESAR SP/90 Module 1, Figure 15.6.4-36, which shows no core uncovery for a 4.313-inch diameter break (this is the size of the direct vessel injection nozzle diameter), even assuming that the accumulators do not deliver.

The most limiting large break was judged to be the cold leg double-ended LOCA. The flow requirement for this event was established as being able to support a

1-inch-per-second flouding rate plus margin to account for uncertainties and entrainment. In the case of the RESAR SP/90, this led to a value of 643 lbm/sec at 60 psia. This total flow had to be provided by the combination of two HHSI pumps and four core reflood tanks (CRT), except that an additional component failure was assumed to ensure additional margin in the design to provide the integrated safeguards system with the capability to sustain an additional single failure without a significant degradation in safety performance. The above flow requirement led directly to the CRT sizing shown in Table 6.3-2 (Sheet 3 of 6) of RESAR SP/90 Module 1. The accumulators were scaled up from existing plants, except that also in this case additional conservatism was introduced by requiring that, with only two out of four accumulators delivering, there would be no significant degradation in safety performance. This resulted in the accumulators also described in Table 6.3-2 (Sheet 3 of 6) of RESAR SP/90 Module 1.

Using the equipment sizes thus determined, the LOCA analyses for various break sizes as reported in Subsection 15.6 of RESAR SP/90 Module 1 were performed.

Westinghouse also noted that there are large margins to the 10 CFR 50.46 acceptance criteria and, therefore, concluded that the requirements of GDC 35 and 10 CFR 50.46 are fully met.

While the staff concludes that this response is acceptable for the PDA stage, the FDA stage will require a thorough review of the assumptions and calculations that led to the selection of the most limiting large break LOCA, small break LOCA, and the resultant ECCS flow requirements.

6.3.6 Conclusions

The ECCS includes the piping, valves, pumps, heat exchangers, instrumentation, and controls used to transport heat from the reactor core after a LOCA, including a pipe break or a spurious relief or safety valve opening in the RCS that could result in a discharge larger than that which could be replaced by the normal makeup system. The scope of review of the ECCS included piping and instrumentation diagrams, failure modes and effects analyses, and design specifications for essential components. The staff review included the Westinghouse design criteria and design bases for the ECCS and the manner in which the design conforms to these criteria and bases.

Except for the unresolved items identified above, the staff concludes that the design criteria of the ECCS are acceptable and meet the requirements of 10 CFR 50.46, Appendix K to 10 CFR 50, and GDC 2, 5, 17, 19, 20, 35, 36, and 37.

6.4 Control Room Habitability (RESAR SP/90 Module 13, Section 6.4)

The control room habitability system is designed to maintain the control room in a safe, habitable condition following a postulated accident. The control room ventilation system is designed as an engineered safety feature to protect plant operators against the effects of accidental releases of toxic and radioactive gases. The fresh air intake is monitored for radioactivity in the form of particulates, iodine, and noble gases by redundant monitoring equipment. Redundant chlorine detectors and redundant smoke detectors will sample the fresh air intake for toxic fumes. If toxic or radioactive gases are detected, the inlet and exhaust isolation valves will close, and operation of the emergency circulation system will be automatically actuated. In the event of a LOCA, the control room habitability zone will be automatically isolated, and the emergency circulation system will be actuated.

The control room ventilation system is designed to use habitability zone isolation in conjunction with low-leakage building construction to achieve protection of personnel from toxic or radioactive gases. Filtered air recirculation also will be used for protection against airborne radioactive contaminants.

The staff's review of the control equipment area ventilation system was performed in accordance with SRP Section 9.4.1, and the staff concludes that the design of the system equipment is in conformance with the requirements of GDC 4.

The applicant has protected the control room operators against radiation by using shielding and a filtration system in its RESAR SP/90 design to remove airborne contaminants. The review of the engineered-safety-feature atmospheric cleanup system was performed in accordance with SRP Section 6.5.1.

Westinghouse has stated that the guidance of Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," and 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," was followed with regard to toxic gas analysis and protective provisions from chlorine, specifically.

Westinghouse has not provided a dose analysis to demonstrate that the control room personnel will not receive radiation exposures in excess of 5 rems whole body, or its equivalent to any part of the body, for the duration of a postulated LOCA, in accordance with GDC 19. Since this dose analysis is site-specific, the staff will review this dose analysis during the plant-specific licensing process of an applicant referencing the RESAR SP/90 design.

Westinghouse has not provided a toxic gas calculation to demonstrate that the control room personnel will not be adversely affected by toxic gases. Since the nature of onsite chemical and offsite hazardous chemicals and meteorological dispersion characteristics are site-specific, the staff will review these aspects during the plant-specific licensing process of an applicant referencing the RESAR SP/90 design.

Pending acceptable toxic gas and radiation dose calculations, the staff will determine whether the control room habitability systems are adequate to provide safe, habitable conditions within the control room under both normal and accident conditions in accordance with GDC 19.

6.5 Fission Product Removal and Control Systems (RESAR SP/90 Module 1, Section 6.5)

6.5.1 Engineered-Safety-Feature (ESF) Atmospheric Cleanup Systems

The ESF atmospheric cleanup systems will consist of process equipment and instrumentation necessary to control the release of radioactive iodine and particulate material following a design-basis accident. The two filtration systems designed for this purpose are the control room emergency circulation filter system and the annulus air cleanup and fuel building exhaust system.

The control room emergency circulation filter system is designed to protect the control room personnel against the effects of an atmospheric release of radioactive materials. It will consist of redundant circulation filter trains with high-efficiency particulate air (HEPA) filters and carbon adsorbers, and is automatically actuated upon a loss-of-coolant accident or detection of radiologic contaminants in the fresh air intake of the control room ventilation system. This system is designed to limit doses to control room personnel to within GDC 19 criteria as they relate to systems designed for habitability of the control room under accident and LOCA conditions.

The annulus air cleanup and fuel building emergency exhaust system is designed as a dual purpose ESF system. It will provide post-LOCA evacuation and filtration of the annular secondary containment air volume and filtration of the fuel handling area atmosphere following a fuel handling accident. The system will consist of redundant filter trains, fans, dampers, ductwork, and controls to achieve the required safety functions.

The dual-purpose design of the annulus air cleanup and fuel building emergency exhaust system was reviewed for initiation and isolation provisions under the two modes of operation. The annulus air cleanup mode will be initiated by a safety injection signal indicating a loss-of-coolant accident. The fuel handling building emergency exhaust mode will be initiated by high radiation levels in the fuel area indicating a fuel handling accident. In addition to the automatic mode of operation, either mode will have the capability to be manually initiated from the control room. Sufficient component redundancy will be provided to ensure system capacity for performing either safety function. Redundant dampers will be provided to isolate parallel ducting required for the dual-purpose function. This will prevent inadvertent damper operation from breaching the pressure boundary of either building when the system is in operation. It is concluded that the dual-purpose function of the system does not inhibit the safe operation of either mode.

Acceptance criteria for the ESF atmosphere cleanup systems are based in part on meeting the relevant requirements of GDC 41, 42, and 61. An acceptable way to meet these requirements is to use the regulatory positions of Regulatory Guide 1.52 as it relates to the design, testing, and maintenance of ESF atmosphere cleanup system air filtration and adsorption units. The staff reviewed the RESAR SP/90 ESF air cleanup systems with regard to Regulatory Guide 1.52, Revision 2. The results of the review are described below.

Both systems are designed so that they can operate after a design-basis accident. They have provisions to prefilter air, remove moisture, and meet Regulatory Guide 1.52 guidelines for charcoal adsorption. The systems provide redundancy, are designed to seismic Category I requirements, and are able to actuate automatically. Redundant trains in the systems are powered from independent Class 1E power sources. HEPA filters, carbon adsorbers, ductwork, and other components of the cleanup trains conform to acceptable codes and standards, such as ANSI N509 and ANSI N510 with respect to design, testing, and maintenance of the system components. In addition, the ESF air cleanup systems, which will be equipped with radioactivity monitors, meet the requirements of GDC 64 with regard to monitoring radioactive releases from these systems.

The staff concludes that the design of the ESF atmosphere cleanup systems meets the requirements of GDC 41, 42 and 43 by adhering to the guidelines of Regulatory

Guide 1.52 and other acceptable industry standards. The design meets the requirements of GDC 61 as it relates to the design of systems for radioactivity control under normal and postulated accident conditions. The design also meets the requirements of GDC 64 with respect to monitoring radioactive releases. Compliance with GDC 19 is described in Section 6.4 of the SER; otherwise, the staff finds the ESF atmosphere cleanup systems to be acceptable.

6.5.2 Containment Spray System (Fission Product Removal) (RESAR SP/90 Module 10 Section 6.5.2)

The staff reviewed the effectiveness of the containment spray and spray additive system for removing fission products following a design-basis accident in accordance with SRP Section 6.5.2.

The containment spray system design includes four independent trains, two of which together are capable of providing adequate spray coverage of the containment atmosphere. The spray system will be actuated automatically by high containment pressure and will draw a boric acid solution from the emergency water storage tank located within the containment building. The two trains will deliver the spray soluion to SPRAYCO-1713A nozzles at a flow rate of 6000 gallons per minute. The applicant estimated that approximately 80 percent of the net free volume of the containment building will be swept by any two of the four trains. The remaining volume, which will be located below the operating deck, will be mixed with the sprayed volume by the air flow of the containment fan coolers. The spray solution will dissolve the trisodium phosphate in the baskets to raise the pH of the solution to at least 7.0. The containment spray system will operate for 2 hours or longer. The containment spray system is designed to permit preoperational testing, periodic operational testing, and inspection for proper system operability.

The staff determined that similar containment spray systems have been tested for effectiveness in removing soluble volatile and particulate materials from air. With two of the four trains in operation, the staff estimated that the post-accident mean residence time of soluble volatile and particulate materials in the containment atmosphere will be 1 hour or less. The staff determined that the system meets GDC 41, "Containment Atmosphere Cleanup." The staff also determined that the pre-operational testing, periodic operational testing, and inspection of the containment spray system meet GDC 42, "Inspection of Containment Atmosphere Cleanup Systems," and GDC 43, "Testing of Containment Atmosphere Cleanup Systems." The staff concludes that the containment spray and spray additive system is acceptable.

6.5.3 Fission Product Control Systems

The fission product control systems and structures are designed to limit the release of radioactive materials after accidents so that the resulting offsite doses are less than the guideline values of 10 CFR 100. The evaluation of the radiological consequences from postulated accidents is not within the scope of the RESAR SP/90 because part of the mathematical computation model is site specific. However, key parameters of systems and structures that may be relied on to limit releases will be used in the mathematical model. These are the primary containment, containment isolation system, containment spray system, secondary containment, and annulus air cleanup system. Parameters of the containment minipurge system also enter the model if it is assumed that the system is in operation when the postulated accident occurs. The acceptability of the design of these systems is discussed in detail in Section 6.2 of this SER.

The verification that values of key parameters are within acceptable limits and confirmation of the applicability of modeling assumptions will be reviewed during the review of the plant-specific licensing process of an applicant referencing RESAR SP/90 design.

6.6 Inservice Inspection of Class 2 and 3 Components

GDC 36, 39, 42, and 45 require in part that Class 2 and 3 components be designed to permit appropriate periodic inspection of important component parts to ensure system integrity and capability. Section 50.55a(g) of 10 CFR 50 defines the requirements for the preservice and inservice inspection programs for light-water-cooled nuclear power facility components.

Westinghouse has not provided a preservice or inservice inspection program for Class 2 and 3 components. The plant-specific applicant will be required to address this issue during the licensing process of its application referencing the RESAR SP/90 design. However, as discussed in Section 14 of this SER, Westinghouse will be required to address preservice/inservice inspection and testing as part of a program to ensure design reliability during the FDA stage of review.

7 INSTRUMENTATION AND CONTROLS (RESAR SP/90 Module 9, Section 7.0)

7.1 General

The staff reviewed the design of the RESAR SP/90 instrumentation and control systems against applicable general design criteria, standards of the Institute of Electrical and Electronics Engineers (IEEE), regulatory guides, and staff technical positions. The documents used in the staff's review are listed in Section 7, Table 7-1, "Acceptance Criteria for Instrumentation and Control Systems Important to Safety," of the Standard Review Plan.

7.1.1 Interface Information (RESAR SP/90 Module 9, Section 7A)

The staff reviewed the interface information provided in RESAR SP/90 for the instrumentation and controls associated with the proposed design. Table 7.1 of this SER shows the interface acceptance criteria for the instrumentation and control systems. During review of plant-specific licensing applications referencing the RESAR SP/90 design, the staff will review the implementation of each interface requirement specified in the RESAR SP/90 application as supplemented by the interface requirements included in this SER to ensure that these criteria are met.

The staff concludes that the interface information and criteria contained in RESAR SP/90 application, 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 SP/90.

7.1.2 Safety System Description (RESAR SP/90 Module 9, Section 7.1)

7.1.2.1 Introduction

Westinghouse has proposed a new safety system for RESAR SP/90 to implement the protective functions of reactor trip and engineered safety features system actuation. Unlike previous Westinghouse designs, RESAR SP/90 does not have a separate reactor trip system or engineered safety features actuation system. The RESAR SP/90 approach is similar to that proposed in the RESAR 414 and proposes a composite safety system, similar to that identified the Institute of Electronics, in "Trial-Use Standard Criteria for Safety Systems for Nuclear Power Generating Stations" (IEEE Std 603-1977). The RESAR SP/90 safety system will consist of the aggregate of electrical and mechanical equipment necessary to mitigate the consequences of design-basis accidents. The elements of the safety system are shown in Figure 7.1 of this report, and the subsystems of the safety system are discussed below.

The integrated protection system will sense plant conditions and generate signals to initiate reactor trip and engineered safety features system actuation. The

Table 7.1 Interface acceptance criteria for instrumentation and control systems*

	SER sections and RESAR SP/90 systemms**											
	7.1, 7.2, 7.3	7.4			7.5	7.6	7.7***					8
Criteria	Safety System	RHRS	EFWS	Boration System	Information Systems Important To Safety	Interlocks Required for Safety	APCS	RC	SLC	PC	BCS	Power Supply for I&C
10 CFR Part 50												
10 CFR §50.34	X	X	X	X	X	X	X	X	X	X	X	X
10 CFR §50.36	X	X	X	X	X	X						X
10 CFR §50.55a	X	X	X	X	X	X	X	X	X	X	X	X
GDC, Appendix A, 10 CFR 50												
GDC 1	X	[illegible]	X	X	X	X	X	X	X	X	X	X
GDC 2	X	X	X	[illegible]	X	X						X
GDC 3	X	X	X	X	X	X						X
GDC 4	X	X	X	X	X	X						X
GDC 5	X	X	X	X	X	X						X
GDC 10	X	X	X	X	X	X						X
GDC 13	X	X	X	X	X	X	X	X	X	X	X	X
GDC 15	X	X	X	X	X	X	X	X	X	X	X	X
GDC 19	X	X	X	X	X	X	X	X	X	X	X	X

See footnotes at end of table.

Table 7.1 (Continued)

	SER sections and RESAR SP/90 systems**											
	7.1, 7.2, 7.3	7.4			7.5	7.6	7.7***					8
Criteria	Safety System	RHRS	EFWS	Boration System	Information Systems Important To Safety	Interlocks Required for Safety	APCS	RC	SLC	PC	BCS	Power Supply for I&C
GDC 20	X	X	X	X	X	X						X
GDC 21	X	X	X	X	X	X						X
GDC 22	X	X	X	X	X	X						X
GDC 23	X	X	X	X	X	X						X
GDC 24	X	X	X	X	X	X	X	X	X	X	X	X
GDC 25	X			X	X			X			X	
GDC 26	X	X			X		X	X	X	X	X	
GDC 27	X	X			X		X	X	X	X	X	
GDC 28	X					X	X	X	X	X	X	
GDC 29	X	X	X	X	X	X	X	X	X	X	X	
GDC 33		X			X	X						
GDC 34	X	X			X	X						
GDC 35					X	X						
GDC 37	X				X	X						
GDC 38	X				X	X						

See footnote at end of table.

Table 7.1 (Continued)

	SER sections and RESAR SP/90 systemms**											
	7.1, 7.2, 7.3	7.4			7.5	7.6	7.7***					8
Criteria	Safety System	RHRS	EFWS	Bora-tion System	Information Systems Important To Safety	Interlocks Required for Safety	APCS	RC	SLC	PC	BCS	Power Supply for I&C
GDC 40	X				X	X						
GDC 41	X				X	X						
GDC 43	X				X	X						
GDC 44	X				X	X						
GDC 46	X				X	X						
GDC 50	X				X	X						
GDC 54	X				X	X						
GDC 55	X				X	X						
GDC 56	X				X	X						
GDC 57	X				X	X						
IEEE Standards												
IEEE Std 279 (ANSI N42.7)	X	X	X	X	X	X	X	X	X	X	X	X
IEEE Std 308			X		X	X						
IEEE Std 323	X	X	X	X	X	X						X

See footnotes at end of table.

Table 7.1 (Continued)

	SER sections and RESAR SP/90 systemms**											
	7.1, 7.2, 7.3	7.4			7.5	7.6	7.7***					8
Criteria	Safety System	RHRS	EFWS	Bora-tion System	Information Systems Important To Safety	Interlocks Required for Safety	APCS	RC	SLC	PC	BCS	Power Supply for I&C
IEEE Std 317	X	X		X	X	X	X	X	X	X	X	X
IEEE Std 336 (ANSI N45.2.4)	X	X	X	X	X	X	X	X	X	X	X	X
IEEE Std 338	X	X	X	X	X	X						X
IEEE Std 344 (ANSI N41.7)	X	X	X	X	X	X						X
IEEE Std 379 (ANSI N41.2)	X	X	X	X	X	X	X	X	X	X	X	X
IEEE Std 384 (ANSI N41.14)	X	X	X	X	X	X						X
RGs †												
RG 1.6	X	X	X	X	X	X						X
RG 1.11	X	X		X	X	X						
RG 1.12	X					X						
RG 1.22	X	X	X	X	X	X						X
RG 1.29	X		X		X	X						
RG 1.30	X	X	X	X	X	X	X	X	X	X	X	X

See footnotes at end of table.

Table 7.1 (Continued)

	SER sections and RESAR SP/90 systemms**											
	7.1, 7.2, 7.3	7.4			7.5	7.6	7.7***					8
Criteria	Safety System	RHRS	EFWS	Boration System	Information Systems Important To Safety	Interlocks Required for Safety	APCS	RC	SLC	PC	BCS	Power Supply for I&C
RG 1.32	X	X	X	X	X	X						X
RG 1.47	X	X	X	X	X	X						X
RG 1.53	X	X	X	X	X	X						X
RG 1.62	X	X	X	X	X	X						X
RG 1.63	X	X		X	X	X						X
RG 1.67	X					X						
RG 1.68	X				X							X
RG 1.70	X	X	X	X	X	X	X	X	X	X	X	X
RG 1.75	X	X	X	X	X	X						X
RG 1.80	X	X	X	X		X						
RG 1.89	X	X	X	X	X	X						X
RG 1.95	X					X						
RG 1.97					X							
RG 1.100	X	X	X	X	X	X						X
RG 1.105	X	X	X	X	X	X						X

See footnotes at end of table.

Table 7.1 (Continued)

	SER sections and RESAR SP/90 systemms**											
	7.1, 7.2, 7.3	7.4			7.5	7.6	7.7***					8
Criteria	Safety System	RHRS	EFWS	Bora-tion System	Information Systems Important To Safety	Interlocks Required for Safety	APCS	RC	SLC	PC	BCS	Power Supply for I&C
RG 1.118	X	X	X	X	X	X						X
RG 1.120	X	X	X	X	X	X	X	X	X	X	X	X
RG 1.151	X	X	X	X	X							
BTPS of the ICSB ††												
BTP ICSB 3		X	X	X		X						X
BTP ICSB 4		X	X	X		X						X
BTP ICSB 12	X	X										
BTP ICSB 13	X		X			X						
BTP ICSB 14	X							X				
BTP ICSB 15	X											
BTP ICSB 20	X	X	X			X						
BTP ICSB 21	X	X	X	X	X	X						X
BTP ICSB 22	X	X	X	X	X	X						X
BTP ICSB 26	X											

See footnotes at end of table.

Table 7.1 (Continued)

*Seismic design was based on operating-basis earthquake rather than safe-shutdown earthquake.

**Abbreviations:

ANSI	American National Standards Institute
APCS	automatic power control system
BCS	boron control system
BTP	branch technical position
CFR	Code of Federal Regulations
EFWS	emergency feedwater system
I&C	instrumentation and control
ICSB	Instrumentation and Control Systems Branch
IEEE	Institute of Electrical and Electronics Engineers
PC	pressurizer controls (both pressure and water level)
RC	rod control (system)
RG	regulatory guide
RHRS	residual heat removal system
SER	safety evaluation report
SLC	secondary loop controls (steam generator water level and steam dump)

***The control systems not required for safety are acceptable if failure of control system components or total systems do not significantly affect the ability of plant safety systems to function as required, or cause plant conditions more severe than those for which the plant safety systems are designed. Section 7.7 of this table lists those criteria, standards, and guides used as references when making such determinations.

†Regulatory guides included that are not referenced in this SER before this section:

RG 1.22, "Periodic Testing of Protection System Actuation Functions."
RG 1.30, "Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment."
RG 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Proection Systems."
RG 1.62, "Manual Initiation of Protection Action."
RG 1.67, "Installation of O
RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants."
RG 1.80, "Preoperational Testing of Instrument Air Systems."
RG 1.105,"Instrument Spans and Setpoints."

††Branch technical positions are located in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Appendix 7-A.

integrated protection system will use digital processing techniques to perform safety functions. Although the staff has completed safety reviews of other protection system designs that use digital processing techniques (i.e., programmable equipment), the types of digital computers and their application in the integrated protection system design will be different from those of the previously reviewed systems.

The protective action system will carry out the protective functions (i.e., reactor trip and engineered safety features system actuation) on demand from the integrated protection system. Safety-related display instrumentation will provide safety system operating status and information to enable the operator to perform required safety functions. The essential auxiliary supporting systems will provide services required for the safety system to perform its protective functions.

The RESAR SP/90 safety system has been modified from previous Westinghouse designs for reactor trip and engineered safety features actuation systems. The major changes include the following:

- addition of reactor trip functions for departure from nucleate boiling ratio and linear heat generation rate
- modification of the reactor coolant flow and reactor coolant pump under-frequency reactor trip functions
- a hybrid system design combining both analog and digital computer modules to perform trip functions and logic manipulations
- use of multiplexed digital data communications and fiber optic cables for transmitting data within channels, among channels, and between safety and non-safety systems
- modification of the reactor trip breaker logic

7.1.2.2 Integrated Protection System (RESAR SP/90 Module 9, Sections 7.1.1.2, 7.1.1.3)

The integrated protection system, as shown in Figure 7.1, is the subsystem of the RESAR SP/90 safety system in which the majority of the new design features have been implemented. The integrated protection system is designed to perform the processing and command initiation tasks of the safety system--that is, it will monitor process parameters, perform calculations of key reactor parameters, and initiate reactor trip and engineered safety features actuation.

The integrated protection system will consist of four redundant channels, contained in integrated protection cabinets (IPCs), each with its own set of process inputs. Each channel will read the process measurements and perform any required processing and calculations on the measurements. The results of the processing (trip or no trip) from the channel will be combined with similar results received

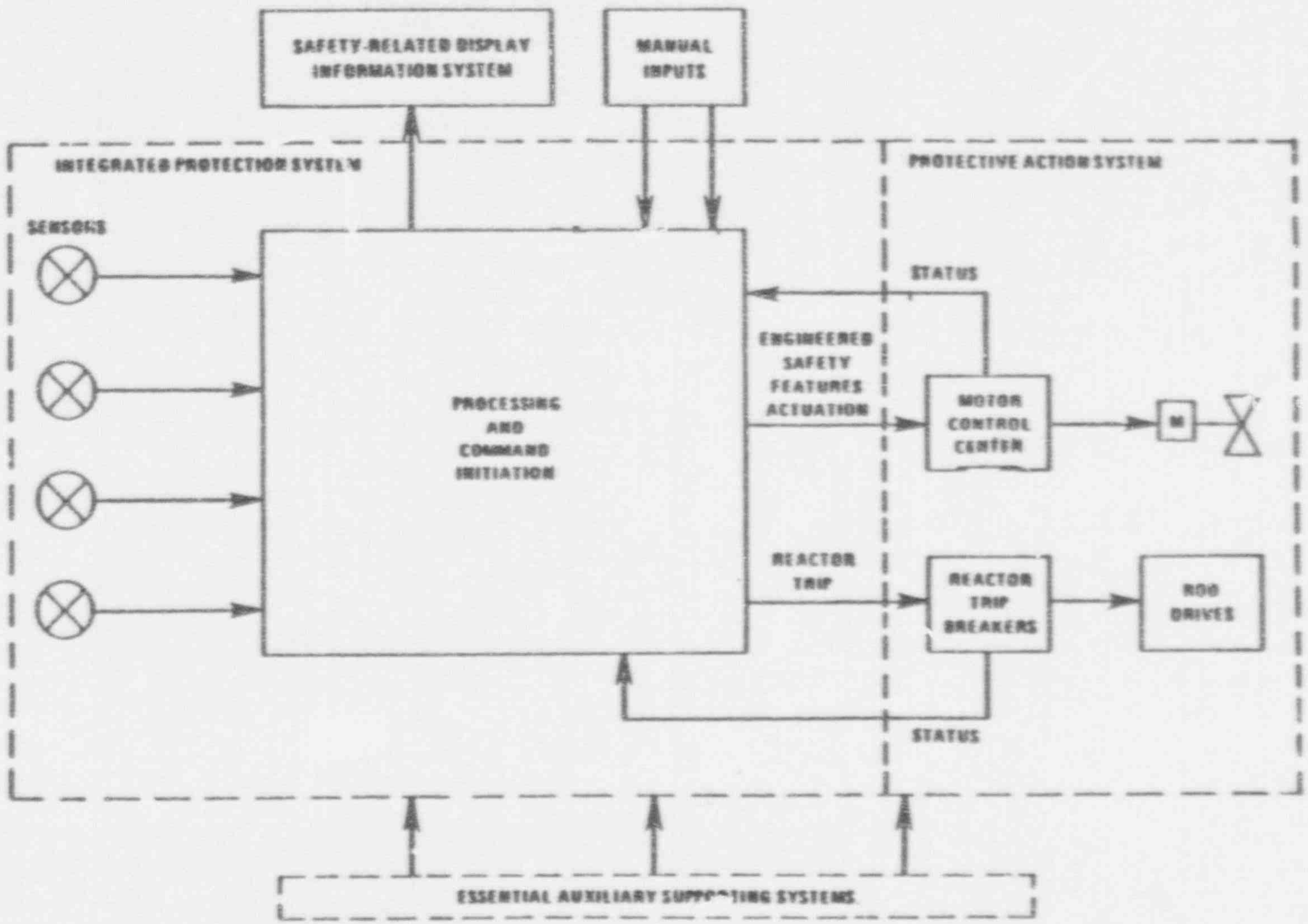

Figure 7.1 Resar SP/90 safety system elements

as inputs from the remaining three channels in a trip logic computer. The trip logic computer will generate the reactor trip functions when required. The results of the channel's processing will be transmitted to the other three channels where a similar logic manipulation will be performed. Based on the results of the logic manipulation, each channel will provide a reactor trip signal to one of the four reactor trip actuation trains.

The integrated protection system also will contain two engineered safety features actuation cabinets (ESFACs). Each ESFAC will receive inputs from the four channels regarding the status of the protective functions and perform the voting logic manipulations for the engineered safety systems actuation. Based on the results of the logic manipulations, each ESFAC will provide signals to actuate the appropriate engineered safety features in one of the two redundant engineered safety features system trains.

Each channel of the integrated protection system will include a manually initiated automatic testing system. Each channel will transmit information on various parameters and system status to the safety-related display information system and provides data to the plant control system for use in plant operational control.

A block diagram of one channel of the integrated protection system is shown in Figure 7.2 of this SER. The major modules are summarized as follows:

- Process sensors will monitor key variables and convert them to analog or digital signals for transmission to the IPCs. In addition to the sensors used in previous Westinghouse systems, the integrated protection system will sensor inputs from reactor coolant pump speed and the nitrogen-16 power monitoring detectors.

- IPCs will provide signal conditioning and digitizing for analog sensor inputs. These cabinets will provide the coincidence logic to generate trip outputs for the reactor trip breakers. They also will provide bistable trip outputs for actuation of the engineered safety features system.

- ESFACs will provide the coincidence logic for the bistable trip inputs from the IPCs to generate system level actuation signals for the engineered safety features system.

- Integrated logic cabinets will provide the component level actuation signals developed from the system level actuation signals to actuate the engineered safety feature system, final actuation devices, and actuated equipment.

- The reactor trip functions performed by the integrated protection system and the engineered safety features system actuation functions performed are identified, respectively, in Sections 7.2 and 7.3 of this SER.

7.1.2.3 Protective Action System (RESAR SP/90 Module 9, Section 7.1.1.4)

The protective action system is the subsystem of the safety system that executes the protective functions for reactor trip and engineered safety features on demand from the integrated protection system.

The protective action system will consist of four reactor trip actuation trains and two engineered safety features actuation trains. The protective action system also will furnish status information to the integrated protection system

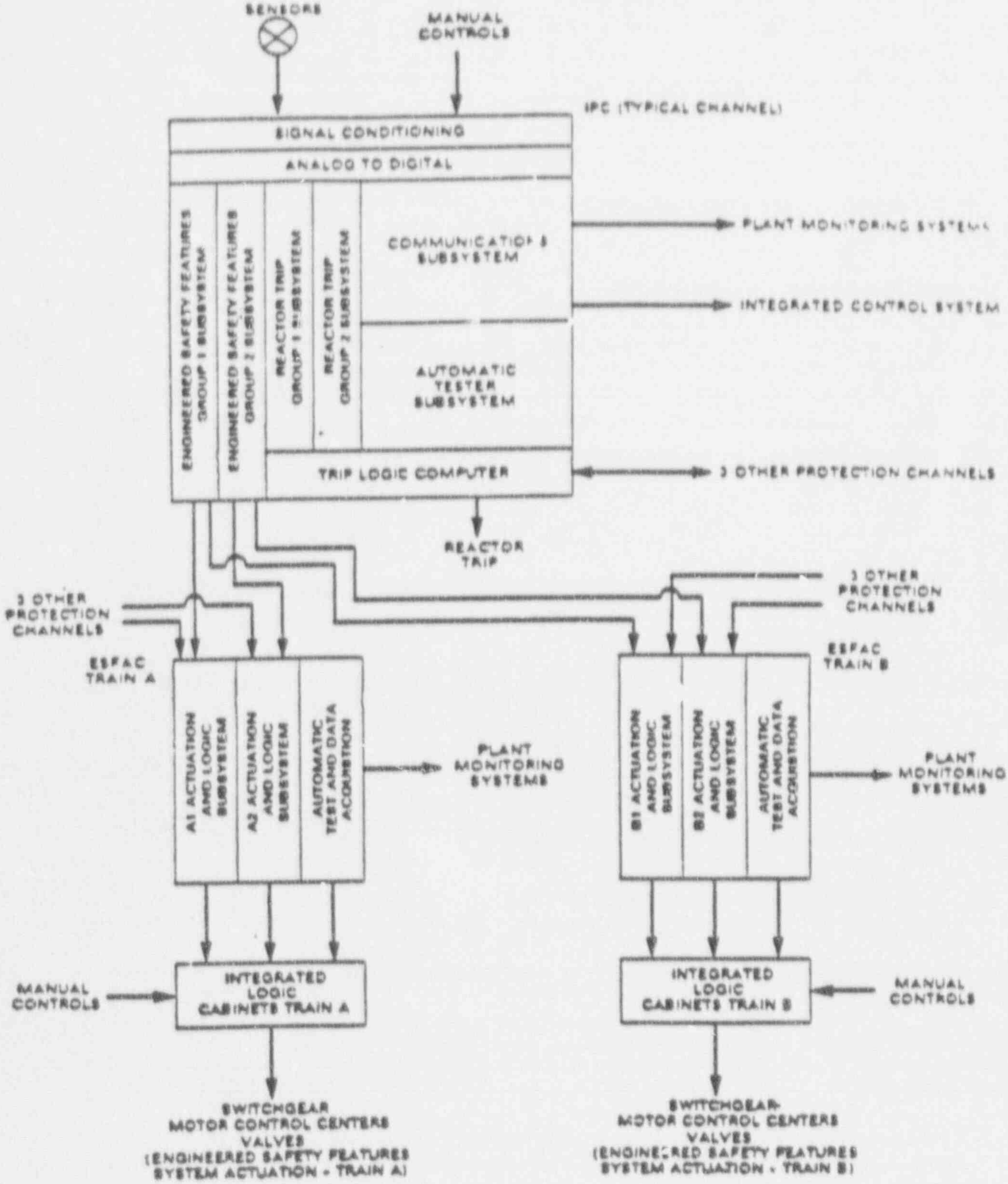

Figure 7.2 Integrated protection system block diagram (typical for one channel)

for interlocks and permissives and for transmission to the safety-related display information system.

The reactor trip actuation trains will consist of eight circuit breakers (two in each train) interconnected in a two-out-of-four logic configuration. Each reactor trip actuation train will receive a trip signal from the associated channel of the integrated protection system. When a trip signal is received from a channel, the circuit breakers in the associated reactor trip actuation will open. If the circuit breakers in two or more trains are opened, power to the control rod drive mechanism will stop, allowing the control rods to fall by gravity into the core.

The engineered safety features actuation trains will consist of redundant and independent fluid systems and associated actuation devices and equipment necessary to accomplish the engineered safety features protective functions. Each engineered safety features actuation train will receive signals from the associated integrated logic cabinets of the integrated protection system. The signals from the integrated logic cabinets will actuate equipment in the associated engineered safety features system in each train.

The reactor trip functions that result in the protective action system executing a reactor trip and the engineered safety features system actuation functions that will result in the protective action system actuating engineered safety features equipment are identified, respectively, in Sections 7.2 and 7.3 of this SER.

7.1.2.4 Design Criteria and Design Bases (RESAR SP/90 Module 9, Section 7.1.2)

Design Criteria

Westinghouse has identified the design criteria that are applicable to the safety system in Table 7.1.1 of RESAR SP/90. The staff has reviewed this table for conformance with the design criteria for safety instrumentation and control systems, identified in Section 7.1 of this SER and has concluded that it is acceptable. Table 7.1 of this SER also shows the interface acceptance criteria for the safety system that must be satisfied by the balance-of-plant designs referencing RESAR SP/90.

Design Bases (RESAR SP/90 Module 9, Section 7.1.2.1)

The design bases for RESAR SP/90 establish

(1) the events that require a protective function to ensure that the station remains within the limits for safe operation

(2) the classification of these events according to their frequency of occurrence

(3) the limiting safety consequences for each classification

(4) the performance requirements for the safety system functions and equipment that are required to operate to ensure that the consequences identified in item 3 are not exceeded

The results of the staff's review and evaluation of the RESAR SP/90 design bases with respect to items 1, 2, and 3 above will be included in Section 15 of this SER.

The staff's review of the design bases for the RESAR SP/90 safety system included the performance requirements in item 4 above and the additional design-bases information required by Section 3 of IEEE Standard 279-1971 to ensure that the design-bases information provides reasonable assurance that the protective functions will be satisfactorily implemented in the safety system. RESAR SP/90 documents detailed design-bases information regarding protective function performance requirements, bypasses, interlocks and permissives, equipment qualification, and interaction between the safety system and the control system.

7.1.2.5 Defense-In-Depth Evaluation

During its review of the RESAR 414 PDA, the staff paid special attention to the potential for adverse interaction between the control, reactor trip, and engineered safety features functions and testing procedures and safety problems associated with manufacturing and maintenance errors. Results of the staff's review efforts in these areas were presented in the SER for RESAR 414, NUREG-0491, "Reference Safety Analysis Report (RESAR-414 Nuclear Steam Supply System Standard Design) Preliminary Design Approval (PDA), November 1978, and in NUREG-0493, "A Defence - In-Depth and Diversity Assessment of the RESAR - 414 Integrated Protection System," February 1979.

The staff concluded that the Westinghouse design principles and the architecture of the integrated protection system were consistent with the defense-in-depth guidelines and identified the analyses and tests which Westinghouse would be required to include as part of its design verification program (NUREG-0491). These analyses and tests were to provide information necessary to demonstrate that the defense-in-depth principle had been implemented in the final design of the integrated protection system.

Because of the similarity between the integrated protection system of the RESAR 414 and RESAR SP/90, the staff will require the same type of analyses and tests, per the guidance of NUREG-0493, to be provided for the RESAR SP/90 during its FDA review. Specifically:

(1) Defense-In-Depth Analyses

The staff will require Westinghouse to perform a design analysis of the integrated protection system in accordance with the defense-in-depth guidelines given in Section 3.3 of NUREG-0493. The staff also will require that a suitable testing program be conducted to demonstrate that the integrated protection system performs as analyzed.

(2) Testing

Periodic Testing - The detailed design will be subject to confirmation in the verification program. The staff will require that a positive means be provided to ensure the operator that, when a test sequence in one channel has been completed, the channel has been restored to operating condition, including re-initialization.

On-Line Validity Checking - The staff will require that continuous validity checking included in the software be limited to necessary functions such as watch-dog timers and block check sums and that Westinghouse establish the failure basis for each on-line test.

(3) Verification and Validation

Westinghouse has committed to provide a verification and validation program for RESAR SP/90 that will closely conform to ANSI/IEEE Standard 7-4.3.2-1982, "Application Criteria for Programmable Digital Computer Systems In Safety Systems of Nuclear Power Generating Stations." The staff will require Westinghouse to identify and justify all deviations from this standard.

7.1.2.6 Conclusions

The staff concludes that the design bases, including the design bases requirements for the integrated protection system, and design criteria will provide reasonable assurance that the protective functions can be satisfactorily implemented in the safety system and are acceptable for preliminary design approval.

Additionally, the staff conditions its approval on satisfactory results of the defense-in-depth evaluation per Section 7.1.2.5 above to be performed during the FDA phase of the RESAR SP/90.

7.2 Reactor Trip Functions (RESAR SP/90 Module 9, Section 7.2)

The RESAR SP/90 reactor trip functions are performed by the safety system. The staff's review of this system is described in Section 7.1.2 of this SER. The protective functions for reactor trip and the conditions for which they are designed to give protection are summarized below.

- Nuclear Power Overpower Conditions

  source range high neutron flux trip
  intermediate range high neutron flux trip
  power range high neutron flux trips (high and low set points)
  high positive flux rate trip
  high negative flux rate trip

- Adverse Core Conditions

  low departure from nucleate boiling ratio trip
  high linear heat generation rate (kilowatts per foot) trip

- Pressurizer Conditions

  pressurizer low pressure trip
  pressurizer high pressure trip
  pressurizer high water level trip

- Reactor Coolant Low Flow Conditions

  low reactor coolant flow trip
  reactor coolant pump underspeed trip

- Steam Generator Water Level Conditions

  any steam generator low water level trip
  any steam generator high water level trip

- Reactor Trip on Turbine Trip (for plants without full-load rejection capability)

- Reactor Trip on Safety Injection

- Manual Reactor Trip

The staff's evaluation of the analyses and performance of these reactor protective trip functions with regard to the transients and accidents for which they are designed to provide protection is discussed in Section 15 of this SER.

The staff concludes that there is reasonable assurance that the reactor trip functions can meet the staff's requirements and are acceptable for preliminary design approval.

7.3 Engineered Safety Features Systems Actuation Functions (RESAR SP/90 Module 9, Section 7.3)

The RESAR SP/90 engineered safety features systems are actuated by the safety system, is described in Section 7.1.2 of this SER. The engineered safety features systems actuation functions and the engineered safety features systems that these functions are designed to actuate include those listed below.

- Safety Injection System

  manual safety injection
  high (Hi-1) containment pressure
  pressurizer low pressure
  low compensated steam line pressure
  low-3 compensated cold leg temperature

- Containment Spray System

  manual containment spray
  high (Hi-3) containment pressure

- Containment Isolation (Phase-A) System (Phase-A isolates nonessential process lines penetrating containment.)

  safety injection signal (automatic and manual)
  manual Phase-A isolation

- Containment Isolation (Phase B) System (Phase-B isolates remaining process lines other than safety injection and spray lines.)

  high (Hi-3) containment pressure
  manual containment spray

- Containment Ventilation Isolation System

  safety injection (automatic or manual)
  manual Phase-A isolation
  manual containment spray

- Steam Line Isolation System

  high steam line negative pressure rate
  high (Hi-2) containment pressure
  low compensated steam line pressure
  low-3 compensated cold leg temperature
  manual steam line isolation

- Feedwater Line Isolation (closure of isolation valves, modulating valves, bypass valves, and trip of main feedwater pumps)

  steam generator high water level
  safety injection (automatic or manual)
  low-2 cold leg temperature
  low pressurizer level coincident with reactor trip (closure of modulating valves only)

- Emergency Feedwater System

  safety injection (automatic or manual)
  low-steam generator water level coincident with loss of startup feedwater system
  low-2 steam generator water level coincident with reactor trip

- Block Boron Dilution System

  high source range flux doubling

The staff's evaluation of the analyses and performance of the engineered safety features systems protective functions with regard to the accidents for which they are designed to provide protection are discussed in Section 15 of this SER.

The staff concludes there is reasonable assurance that the engineered safety features system actuation functions can meet the staff's requirements and are acceptable for preliminary design approval.

7.4 Systems, Instrumentation and Controls Required for Safe Shutdown

7.4.1 Systems Required for Safe Shutdown (RESAR SP/90 Module 9, Section 7.4.1)

RESAR SP/90 identifies the minimum systems required to achieve and maintain safety grade shutdown with loss of offsite power. The systems and components included are emergency Class 1E electrical power, emergency feedwater system, residual heat removal system, borated water supply with the high head safety injection pumps, reactor coolant system pressure relief system, steam generator power-operated relief valves (PORVs) and bypasses, emergency letdown system, reactor protection system, and component cooling and service water systems. The RESAR SP/90 application also includes a list of instrumentation and controls required to achieve and maintain a safe shutdown from inside as well as outside the main control room.

7.4.2 Instrumentation and Controls for Safe Shutdown (RESAR SP/90 Module 9, Section 7.4.1)

To meet the requirements of GDC 19 and to effect control of the safe shutdown systems from inside or outside the control room, the RESAR SP/90 design provides instrumentation for steam generator pressure and level, pressurizer pressure and level, reactor coolant temperatures, source range neutron flux, and emergency feedwater flow and supply tank level. In addition, controls, both in the control room and at local control panels, will be provided to trip the turbine, to trip the reactor, and to manually control the systems and components required to achieve and maintain safe shutdown.

7.4.3 Conclusion

The staff concludes that the systems, instrumentation, and controls required for safe shutdown from inside or outside of the main control room with loss of offsite power provided in the RESAR SP/90 design, when implemented in accordance with the requirements in Table 7.1, satisfy the staff's requirements and are acceptable for preliminary design approval.

7.5 Information Systems Important to Safety

Indicators, annunciators, recorders, lights, and displays provide sufficient information to enable the operator to perform the required action during normal operating conditions as well as during and following an accident.

7.5.1 Safety-Related Display Instrumentation (RESAR SP/90 Module 9, Section 7.5)

Westinghouse performed an analysis for RESAR SP/90 to identify the appropriate set of variables and to establish the appropriate design bases and qualification criteria for instrumentation so the operator will be able to monitor conditions in the reactor coolant system, the secondary heat removal system, the containment, the engineered safety features systems, and the safe-shutdown systems. The instrumentation provides information for all operating conditions, including anticipated operational occurrences, accidents, and post-accident conditions.

Table 7.5-1 in the RESAR SP/90 application identifies the safety-related display instrumentation and includes the following information for each variable:

- instrument range or status
- type and category (according to the definition in Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Revision 3)
- environmental qualification
- seismic qualification
- number of channels
- power supply

The staff concludes that the safety-related display instrumentation to be provided is acceptable except for those items listed below, which will be addressed during the FDA phase of the RESAR SP/90 review.

(1) Range has not been specified or is inadequate for

wide range steam generator water level
pressurizer level
containment water level
reactor vessel water level
reactor coolant pump status
ac, dc, vital instrument voltage
quench tank pressure
quench tank level
quench tank temperature
volume control tank level
high level radiation waste level
radiation gas holdup tank pressure
plant vent air flow rate
condenser air ejector flow rate
steam generator safety/relief valve radiation level
steam generator safety/relief valve flow rate
environs radiation level
other potential source flow rate
residual heat removal (RHR) heat exchanger outlet temperature
core exit temperature
steamline pressure
meteorological parameters (atmospheric stability)

(2) Environmental qualification is unacceptable for

plant vent radiation level
volume control tank level
component cooling water header temperature
ac, dc, vital instrument voltage
plant vent air flow rate
condenser air ejector radiation level
condenser air ejector flow rate
steam generator safety/relief valve flow rate

(3) Instrumentation has not been addressed for

particulates and halogens
plant and environs radiation
analysis of primary coolant gamma spectrum
flow in high-pressure and low-pressure injection systems

(4) Direct measures of the required variable need to be provided for

pressurizer heater power availability
containment heat removal
containment atmosphere temperature

(5) Instrumentation is unacceptable for the steamline radiation monitor because it does not meet the redundancy requirement.

The staff has identified those items listed below as acceptable deviations from Regulatory Guide 1.97 on the basis of the justification given.

- Accumulator tank level is acceptable as a Category 3 variable in lieu of Category 2 because the accumulator tank pressure will be provided as a Category 2 variable and will be used to determine if the accumulator has discharged.

- Containment sump water temperature is acceptable as a Category 3 variable in lieu of Category 2 because the RHR heat exchanger inlet temperature will be provided as a Category 2 variable and can be used to determine containment sump temperature when the sump is the water source for the RHR system.

- Boric acid charging flow is acceptable as a Category 3 variable in lieu of Category 2 because the refueling water storage tank level, high-head safety injection flow, low-head safety injection flow, containment water level, and emergency core cooling system (ECCS) valve status will be provided for monitoring the performance of the ECCS system and the ECCS will not take suction from the boric acid tank.

- Containment isolation valve status is acceptable as a Category 2 variable in lieu of Category 1 based on the justification provided in Note H to Table 7.5-1 in the RESAR SP/90 application.

7.5.2 Bypassed and Inoperable Status Indication (RESAR SP/90 Module 9, Section 7.5.4)

The design of the RESAR SP/90 includes an automatic bypass and inoperable status indication system that will provide indication in the control room when the protection system or systems that are actuated or controlled by the protection system are intentionally rendered inoperable or bypassed. Inoperability of protection systems that rely on support systems will be automatically indicated when a related support system is rendered inoperable or bypassed. The design of the RESAR SP/90 meets the intent of Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems."

7.5.3 Conclusion

As noted in Section 7.1.2 of this report, the staff finds acceptable the design criteria and design bases for the safety system that provide the safety-related display instrumentation and the bypassed and inoperable status indication. The staff concludes that there is reasonable assurance that the safety-related display instrumentation (except as noted above in Section 7.5.1) and the bypassed and inoperable status indication meet the staff's requirements and are acceptable for preliminary design approval. However, those items discussed in Section 7.5.1 must be addressed satisfactorily during the FDA phase of the RESAR SP/90 review.

7.6 Interlock Systems Important to Safety (RESAR SP/90 Module 9, Section 7.6)

Interlock systems important to safety are designed to reduce the probability of occurrence of specific events or to maintain safety systems in a state to ensure their availability when required.

7.6.1 Residual Heat Removal Isolation Valve Interlocks (RESAR SP/90 Module 9, Section 7.6.2)

The residual heat removal system will have two independent and redundant subsystems, each consisting of a pump, heat exchanger, and vaiving. The inlet to each subsystem will include two motor-operated valves in series to isolate the high-pressure reactor coolant system (RCS) from the lower pressure residual heat removal system. The isolation valves in each subsystem are provided with both "prevent-open" and "auto-closure" pressure interlocks that are independent and diverse. Each isolation valve within each subsystem will be powered by a separate power supply and interlocked with a separate RCS pressure transmitter.

7.6.2 Accumulator and Core Reflood Tank Discharge Isolation Valve Interlocks (RESAR SP/90 Module 9, Section 7.6.3)

The accumulators and the core reflood tank are pressure vessels partially filled with borated water and pressurized with nitrogen gas. During normal operation each accumulator and the reflood tank will be isolated from the RCS by two check valves in series. Should the RCS pressure fall below the accumulator pressure, the check valves will open and borated water will be forced into the RCS. To prevent injection of borated water at low-pressure operation during shutdown and startup, each accumulator and the core reflood tank will be provided with a motor-operated isolation valve in series with the check valves.

The operator will close the valve shortly after the RCS is depressurized below the safety injection unblock set point.

The motor-operated isolation valves will be controlled by switches on the main control board and will be interlocked as follows:

- These valves will open automatically on receipt of a safety injection signal ("S").
- These valves will open automatically whenever the RCS pressure is above the safety injection unblock pressure (P-11 interlock).

- These valves will not be able to be closed as long as an S-signal is present.

After the RCS pressure is decreased during shutdown and the motor-operated isolation valves are closed, power to the valves will be disconnected to prevent accidental operation. Lights will be actuated by the valve cam-operated switches and will indicate valve position in the control room. Alarms will be operated by the valve stem-mounted limit switches and motor-operator limit switches and will be activated when a valve is not fully open and the pressure in the system is above the safety injection unblock level.

7.6.3 RCS Overpressure Protection During Low-Temperature Operation

The RCS overpressure protection during low-temperature operation will be dependent on the semiautomatic opening of two pressurizer power-operated relief valves (PORVs). The actuation logic for the PORVs continuously monitors RCS temperature and pressure conditions.

Auctioneered RCS temperature signals will be processed to generate a reference pressure limit. When the actual system pressure approaches the calculated limit, an alarm will be generated to alert the operator to manually arm the system. An actuation signal to open the PORVs will be generated when the system pressure exceeds the reference pressure and the system is armed.

7.6.4 Conclusion

As noted in Section 7.1.2 of this report, the staff finds acceptable the design criteria and design bases for the interlock systems important to safety as described above. The staff concludes that there is reasonable assurance that the interlock systems can meet the staff's requirements and are acceptable for preliminary design approval.

7.7 Control and Instrumentation Systems Not Required for Safety (RESAR SP/90 Module 9, Section 7.7)

The general design objectives of the plant control and instrumentation systems are to:

- establish and maintain power equilibrium of the primary and secondary systems during steady-state unit operation

- constrain operational transients so as to preclude unit tripping and to re-establish steady-state operation

- provide the reactor operator with monitoring instrumentation that indicates all required input and output control parameters of the systems and provides the capability of assuming manual control of the system

The RESAR SP/90 control and instrumentation systems that are not required for safety described in are the automatic-power control system, pressurizer pressure control system, pressurizer water level control system, steam generator water level control system, steam dump control system, boron control system, rod control system, and the plant control signals for monitoring and indication.

The staff reviewed the control and instrumentation systems with the following objectives:

- to establish that these systems are not safety related
- to verify that no credit is taken for the operability of these systems in the accident analyses in Section 15 of RESAR SP/90
- to ensure that failures of these control systems would not impair the capability of the protection system in any significant manner or cause conditions more severe than those for which the safety-related systems are designed
- to establish that control system designs meet applicable requirements of the general design criteria and national standards with regard to independence between control and protection functions

7.7.1 Automatic-Power Control System (RESAR SP/90 Module 9, Section 7.7.1.1)

The automatic-power control system is an integrated control system that will provide automatic load following and provide remote dispatching capability within the load range of 15 to 100 percent.

Inputs to the automatic-power control system include the following:

- current power level
- core axial power distribution (axial offset)
- cold leg temperature
- boron concentration
- control rod position
- gray rod position
- water displacer rod position

Operating constraints include control rod insertion and withdrawal limits, range of allowable operating temperatures, and range of allowable power distribution. The automatic-power control system provides an integrated approach to reactivity control with the capability for independent operation of individual control systems, if required.

7.7.2 Pressurizer Pressure Control System (RESAR SP/90 Module 9, Section 7.7.1.8)

The pressurizer pressure control system will maintain or restore RCS pressure during normal operation and following normal operational transients. The control elements to achieve the pressure changes will be the pressurizer heaters and sprays. A portion of the pressurizer heaters will be portionally controlled with integral action to correct small pressure variations. The remaining (backup) heaters activated when required to control large pressure decreases. Large pressure increases will be controlled by activating pressurizer spray.

Because of the different dynamic characteristics, separate control algorithms will be provided for the heaters and the spray. Inputs for the heater control will be pressurizer water level, N-16 power, average cold-leg temperature and average pressurizer pressure. Inputs for the spray control will be average pressurizer pressure signals.

7.7.3 Pressurizer Water Level Control System (RESAR SP/90 Module 9, Section 7.7.1.9)

During normal plant operation from hot zero load to 100-percent power, water inventory in the RCS will be automatically maintained by varying charging flow to produce the flow demanded by the pressurizer water level control system. Reference water level for the pressurizer will be programmed as a function of measured cold-leg temperature and N-16 derived reactor power. The water level error signal, established from comparison of measured and programmed water level, will be processed by the controller to regulate charging flow.

7.7.4 Steam Generator Water Level Control System (RESAR SP/90 Module 9, Section 7.7.1.10)

Continued delivery of feedwater to the steam generators is required to maintain a heat sink for the reactor coolant system. Each steam generator will be equipped with a feedwater flow controller to maintain a programmed water level as a function of turbine load. Water level control will encompass initially feeding the steam generators by using the startup feedwater system as well as the main feedwater system by way of the main and bypass feedwater valves. Continuous feedwater control will be provided between all modes of operation.

During high-power operation, control of the main feedwater control valves will be provided by the control system using inputs of steam generator level, feedwater flow, and steam flow. Compensation for shrink and swell will be provided using inputs of steam header pressure and N-16 power.

For low-power operation, control of the bypass feedwater valves will be provided by the control system using steam generator level as input. Steam dump in the pressure control mode may be used in conjunction with the low-power feedwater control to minimize the impact of any operating disturbance.

During startup, control of the startup feedwater valves provided by the control system using steam generator level as input.

7.7.5 Steam Dump Control System (RESAR SP/90 Module 9, Section 7.7.1.11)

The steam dump control system is designed to automatically accommodate abnormal load rejections and to regulate heat removal during startup, cooldown, and hot standby condition.

The control system accommodates for load rejections by bypassing main steam around the turbine through steam dump valves. The trip open logic to these valves will be provided by a power mismatch control system that is designed to detect substantial primary/secondary load imbalances based on inputs of turbine first-stage pressure and nuclear (N-16) power.

For a controlled cooldown and a hot, low-power condition before turbine loading, the control system will regulate heat removal by controlling steam release from the steam header using a steam pressure controller.

7.7.6 Boron Control System (RESAR SP/90 Module 9, Section 7.7.1.6)

The boron control system, as directed by the automatic power control system, under normal operation, will provide for demanded changes in boron concentration.

In conjunction with the boron recycle system, the boron control system must be capable of changing boron concentration as required to perform load following when reactor coolant system boron concentration is sufficiently high.

7.7.7 Rod Control System (RESAR SP/90 Module 9, Section 7.7.1.2)

The rod control system is designed to provide automatic control of the reactor coolant system temperature over the entire range of power operation by generating direction and speed demand signals for the control rods. At low-power levels, the rod control system will provide direct control of nuclear power and, in conjunction with automatic steam pressure control, will ensure RCS temperature is maintained at a proper value based on the difference between demanded nuclear power (final value and rate) and measured nuclear power. Automatic low-power rod control will be prevented if automatic steam pressure control is unavailable. At high-power levels (15-100 percent), the rod control system will maintain the reactor coolant temperature within a specified range by measurement of the cold leg temperature and inputs of nuclear power. Manual control of the control rods also will be provided.

The control rod banks will be divided into two groups to obtain smaller incremental reactivity changes. All rod control cluster assemblies in a group will move simultaneously and individual position indication will be provided for each rod control cluster.

In addition to the full-length, high-worth control rods and the boron control system, the reactor design for reactivity control includes low-worth gray control rods that will be either fully inserted or fully withdrawn. Automatic control of the gray rods will be provided by the gray rod control system, which will respond to insertion or withdrawal demand signals from the automatic-power control system. Insertion or withdrawal of gray rods, when required, will be in a preselected sequence, with the sequence for withdrawal opposite that for insertion.

7.7.8 Plant Control System Interlocks (RESAR SP/90 Module 9, Section 7.7.1.5)

Control rod stops will be provided to prevent abnormal power conditions that could result from excessive control rod withdrawal initiated by a control system malfunction or from an operator violation of administrative procedures.

Automatic turbine load runback, initiated by an approach to departure from nucleate boiling on high kilowatt-per-foot conditions, will be provided to prevent high-power operation, which, if reached, would create an undesirable condition or reactor trip.

A turbine loading stop will be provided to limit turbine loading during ramp load increases if the reactor coolant cold-leg temperature is below a programmed reference level.

7.7.9 Plant Control Signals for Monitoring and Indication (RESAR SP/90 Module 9, Sections 7.7.1.3, 7.7.1.4)

Control rod position monitoring will be provided by two separate systems. A digital rod position indication system measure the actual position of each control rod using discrete coils concentrically mounted to the rod drive pressure housing to magnetically sense the presence of the rod drive shaft. A demand position system will count the pulses generated in the rod drive control system to provide a digital readout of the demanded bank position.

A rod deviation alarm will be generated by the plant computer whenever an individual rod position deviates from the position of other rods in the same bank by a preset limit. Alarms also provided if any shutdown rod has left its fully withdrawn position, or if any control rod is in its bottom position except as part of the normal insertion sequence.

Control rod insertion and withdrawal limits will be provided by the N-16 power monitors. Two alarms will be provided to warn the operator about excessive rod insertion. A low alarm will alert the operator to an approach to rod insertion limits that will require the operator to add boron by following normal procedures for the chemical and volume control system or by inserting gray rods. A low-low alarm will alert the operator to take immediate action to add boron by any one of several alternate methods or to insert gray rods.

7.7.10 Signal Selector (RESAR SP/90 Module 9, Section 7.7.1.12)

The integrated control system design includes a signal selector that will isolate the integrated protection system from the control system and will evaluate information from the integrated protection system and transmit data to the control system. Since the integrated protection system will use bypass logic during test and maintenance wherein the 2-out-of-4 logic reverts to 2-out-of-3 logic, control and protection system interaction may be a potential problem. To ensure that independence between the protection system and control systems is maintained and control/protection system interaction is avoided, the signal selector must be designed and utilized such that the requirements of Section 4.7 of IEEE Standard 279-1971 are met. The signal selector is designed to accomplish this by eliminating any invalid signal (such as from a failed sensor within the protection system) from use in the control system.

In response to a staff concern, Westinghouse stated that the signal selector will be similar to that discussed in WCAP-8899, "Westinghouse Model 414 Control System Signal Selection Device," previously evaluated in the RESAR-414 PDA SER, NUREG-0491. In that SER, the staff concluded that the proposed design of the signal selector was acceptable for preliminary design approval subject to demonstration of the following items by the final design of the signal selector:

(1) Demonstrate the adequacy of the signal selector for limiting situations (such as one signal failed and one bypassed), for transient operation of the plant, and for spatial asymmetries in process variables.

(2) Demonstrate adequate reliability of the signal selector function, and define and evaluate credible failure modes of the signal selector and the effects of such modes, including the possibility of systematic, nonrandom

concurrent failures of the two signal selector modules. Modification in the design will be required if the analyses show that failure of the signal selector can result in plant conditions more severe than the design basis of the protection system.

The staff finds the signal selector design acceptable for the PDA stage of review subject to demonstration of the above items during the FDA stage.

7.7.11 Conclusion

The staff concludes that there is reasonable assurance that these systems not required for safety, as discussed above, are acceptable for preliminary design approval.

8 ELECTRIC POWER SYSTEMS (RESAR SP/90 Module 9, Section 8.0)

8.1 General

The primary bases for evaluating the adequacy of the emergency power systems for RESAR SP/90 were the requirements in GDC 1, 5, 17, 18, 50, and the review guidance in Regulatory Guide 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems"; Regulatory Guide 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies"; Regulatory Guide 1.75, "Physical Independence of Electric Systems"; Regulatory Guide 1.63, "Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants," Revision 3, February 1987; Regulatory Guide 1.32, "Criteria for Safety Related Electric Power Systems for Nuclear Power Plants"; and IEEE Standard 308-1974, "Criteria for Class 1E Power Systems for Nuclear Power Generating Stations."

8.2 Offsite Power System (RESAR SP/90 Module 9, Section 8.2)

8.2.1 General Description (RESAR SP/90 Module 9, Section 8.2.1)

The offsite power system is the preferred source of power for this plant design. This system includes the grid, transmission lines, transformers, switchyard components, and associated control systems provided to supply electric power to safety-related and other equipment. The electrical grid will be the source of energy for the offsite power system. The safety function of the offsite power system (assuming that the onsite power systems are not available) is to provide sufficient capacity and capability to ensure that the specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary (RCPB) are not exceeded and to ensure that core cooling, containment integrity, and other vital functions are maintained in the event of accidents. The objectives of the staff review are to determine that the offsite power system (1) satisfies the criteria set forth in Section 8.1 of this report and (2) can reliably perform its design functions during normal plant operation, anticipated operational occurrences, and accident conditions.

Power will be supplied from the main generator to the switchyard through the main transformer. The transmission network will supply the offsite power for startup, normal operation, and safe shutdown. The details of the transmission network will be supplied by the plant-specific applicant. The main transformer will be connected to the main generator via a generator circuit breaker. The supply for the primary winding of the auxiliary transformer will be tapped in between the generator circuit breaker and the low-voltage bushings of the main step-up transformer. In the event of unit trip (except for faults in the generator/transformer zone), the circuit breaker will isolate the generator from the systems. The 13.8-kV system will remain energized by backfeeding through the main and auxiliary transformer, thereby providing an immediate access to the offsite power system. The main generator circuit breaker will be subjected to a testing program to demonstrate its capability to perform its intended function during steady-state, transient, and accident conditions. During all modes of operation, the supply from the standby transformer will serve as a backup.

The RESAR SP/90 is designed to have two physically independent and separate power sources. The auxiliary transformer and the standby transformer have the capacity to supply both non-Class 1E and both Class 1E load groups simultaneously. The two circuits will be independent so that a failure of one circuit will not affect the other and result in loss of both circuits.

The 13.8-kV buses will be provided with a fast transfer scheme. In the event of a fault in the zone, including the generator, generator bus, main step-up transformer, and unit transformer, the 13.8-kV buses will be transferred from the unit auxiliary transformer to the standby transformer. The fast transfer will be supervised by synchronizing check relaying to ensure that voltage and phasing are acceptable before initiating the switchover.

In addition, a third engineered safety feature (ESF) transformer will be provided to supply either one of the redundant 4.16-kV Class 1E loads during plant operation. This configuration isolates the redundant buses from faults in a common supply as well as isolating each bus from a fault on the redundant bus. The switchyard power circuit breakers are designed with duplicate and redundant systems, that is, two independent battery systems, two trip coils per breaker, and two independent protective relay schemes.

The two physically independent transmission circuits provide RESAR SP/90 with two immediate access circuits that exceed the minimum requirements of GDC 17 and are, therefore, acceptable.

8.2.2 Testability

Provisions are provided, in accordance with GDC 18, for periodically testing the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system. This includes provision for testing the fast transfer feature. The applicant has complied with the requirements of GDC 18 and has satisfactorily addressed testability of the electrical power system's transfer capabilities.

8.2.3 Grid Stability Analysis

The design of the offsite power system is the responsibility of the plant-specific applicant. The results of the applicant's grid stability analysis should indicate that the loss of the nuclear unit or the most critical unit on the grid or the most critical transmission line shall not adversely affect the capability of the system to furnish shutdown power to the Class 1E buses. The studies also should consider single- and double-line three-phase faults on the high-voltage transmission lines adjacent to and remote from the switchyard.

The staff will review the results of the plant-specific applicant's stability study to establish that the offsite power will be stable under all fault conditions and other conditions noted above. Acceptable demonstration of this capability will satisfy the applicable portions of GDC 17.

8.2.4 Adequacy of Electric Distribution System Voltages

Events at the Millstone station have shown that adverse effects on the Class 1E loads can be caused by sustained low-grid-voltage conditions when the Class 1E buses are connected to offsite power. These low-voltage conditions will not be detected by the loss-of-voltage relays (loss of offsite power) whose low-voltage pickup setting is generally in the range of 0.7 per unit voltage or less.

The Millstone events also demonstrated that improper voltage protection logic can itself cause adverse effects on the Class 1E systems and equipment (such as spurious load shedding of Class 1E loads from the standby diesel generators and spurious separation of Class 1E systems from offsite power as a result of normal motor-starting transients).

An event at Arkansas Nuclear One station and the subsequent analysis performed disclosed the possibility of degraded voltage conditions existing on the Class 1E buses even with normal grid voltages because of deficiencies in equipment between the grid and the Class 1E buses or because of the starting transients experienced during certain accident events not originally considered in the sizing of these circuits.

The staff evaluation of the RESAR SP/90 design for conformance with the Branch Technical Position PSB-1, "Adequacy of Station Electric Distribution System Voltages," is discussed below.

RESAR SP/90 is designed to have two redundant and independent emergency buses. Two voltage sensing schemes will be employed on each 4.16-kV Class 1E bus to initiate the required logic signal. One scheme will recognize a loss of voltage, and the other recognizes a degraded voltage. Four potential transformers on each bus will provide the necessary input voltages to the protective devices necessary to achieve the above protection.

Four instantaneous undervoltage relays will be used to recognize a loss of voltage. The output contacts of these relays will be directed to logic circuits that process the four undervoltage input circuits into the 2-out-of-4 logic circuit. This scheme will be used on each bus. The loss-of-voltage logic signal will be set below the minimum bus voltage encountered during diesel generator sequential loading. A brief time delay will be employed to prevent false trips arising from transient undervoltage (spike) conditions.

A diverse protection scheme will be used to recognize a degraded voltage. Each of the above four potential transformers will provide an analog output signal of 0-120 volts. This signal is directed to logic circuits and processors that convert the analog signals into a 2-out-of-4 logic signal, whenever the signal drops below a preset value. This scheme serves only to trip the incoming offsite power circuit breakers when that power source has been determined to be degraded. This design cannot adversely affect the sequential loading of the diesel generators.

The degraded voltage logic signal will be set at the minimum permissible continuous bus voltage. A time delay will be provided to prevent damage to or spurious tripping of the permanently connected Class 1E loads by limiting the amount of time they are exposed to a degraded voltage. The final voltage and time set points will be determined based on an analysis of the auxiliary power distribution system, including the Class 1E buses at all voltage levels. The use of a safety injection signal contact in series with the degraded voltage logic circuit output contact will ensure that the Class 1E buses will be immediately separated from the offsite power system whenever an accident occurs and the offsite power system is not able to accept the loads continuously. An alarm also will be provided to alert the operator to a degraded voltage condition.

If preferred power is available to the 4.16-kV Class 1E buses following a loss-of-coolant accident (LOCA), the Class 1E loads will be started in programmed time increments by the load sequencers. The emergency standby diesel generators will be automatically started, but not connected to their bus. However, in the event that preferred power is lost following a LOCA, the load sequencers will function to shed selected loads and automatically connect the associated standby diesel generator to its bus. (Connection of the standby diesel generator to the 4.16-kV Class 1E bus will be performed by the diesel generator control circuitry.) Load sequencers will then function to start the required Class 1E loads in programmed time increments.

The voltage levels at safety-related buses will be optimized for the expected load conditions throughout the anticipated range of voltages of the offsite system by adjustment of transformer taps. This analysis will be verified by testing. This is in accordance with the BTP PSB-1 and is, therefore, acceptable.

8.2.5 Conclusion

The staff concludes that the offsite power system for RESAR SP/90 meets the requirements of GDC 17 and that the testability of the fast transfer scheme meets the requirements of GDC 18.

8.3 <u>Onsite Emergency Power Systems</u> (RESAR SP/90 Module 9, Section 8.3)

8.3.1 AC Power System (RESAR SP/90 Module 9, Section 8.3.1)

The ac onsite power system is a Class 1E system that serves as a standby to the offsite power system. The safety function of the ac onsite emergency power system (assuming the offsite power system is not functioning) is to provide sufficient capacity and capability to ensure that the structures, systems, and components important to safety perform as intended. The staff reviewed the ac onsite emergency power system to determine if it has the required independence from the offsite system, meets the single-failure criterion, is testable, and has the capacity, capability, and reliability to supply power to all required safety loads in accordance with the requirements of GDC 5, 17, and 18.

The design of the onsite ac power system consists of various auxiliary electrical systems designed to provide electric power to Class 1E and non-Class 1E station loads. The standby emergency ac power system will be an independent, onsite system designed to automatically start and provide adequate power for Class 1E loads to ensure safe plant shutdown when preferred and alternate power sources are not available.

The Class 1E portion of the onsite power system consists of two redundant and independent 4160-V distribution systems with their 480-V load centers and motor control centers, 120-V ac power system, 125-V dc system, and the standby power supplies (diesel generator units).

Onsite emergency power for each safety-related load group will be supplied by its diesel generator. Each diesel generator will be automatically started by either a safety injection signal or loss of voltage to the respective 4.16-kV Class 1E bus to which each generator will be connected. As each generator reaches rated voltage and frequency, the generator breaker connecting it to the corresponding 4.16-kV bus will close. With the safety injection signal, connection of the diesel generator to the 4.16-kV bus will not be made unless the preferred source of power is lost and the incoming breaker is open. The diesel generator will be able to accept loads within 10 seconds after receipt of a starting signal, and all automatically sequenced loads will be connected to the Class 1E bus. A fast responding exciter and voltage regulator will ensure voltage recovery of the diesel generator after each load step. Momentary voltage and frequency dips will not exceed a maximum of 25 percent below nominal rating (4.16-kV) for voltage and 5 percent for frequency. The diesel generators will be tested before plant startup to demonstrate their capability to satisfy design requirements. The suitability and qualification testing program of each diesel generator unit of the standby power system is confirmed in accordance with IEEE Standard 387-1977 and Regulatory Guide 1.9, Revision 2, and is, therefore, acceptable.

The applicant states that if the diesel generator sets are of a type or size not previously used as standby emergency power sources in nuclear power plant service, reliability qualification testing for the diesel generator sets will be performed in accordance with IEEE Standard 387-1977. The staff finds this acceptable.

Manual starting of each diesel generator from the control room is incorporated into the design to permit periodic testing. During testing, the diesel generator will be manually synchronized to its bus after reaching rated voltage and frequency. Automatic synchronizing will not be used. An accident signal occurring during periodic testing of a diesel generator will automatically override the test mode and place the diesel generator in the emergency mode. With the accident signal present, connection of the diesel generator to the 4.16-kV bus will not be made unless the preferred source of power is lost. Relays at the diesel generator will detect generator rated voltage and frequency conditions and provide a permissive interlock for the closing of the respective generator circuit breaker. If the preferred source of power is lost without a LOCA, the load sequencer system will initiate the starting of the diesel generators and shed all loads except the load center.

Position C.7 of Regulatory Guide 1.9, Revision 2 (which endorses IEEE Std 387-1977), requires that diesel generator protective trips, except diesel engine overspeed and generator differential, be bypassed when the diesel generator is required for a design-basis event. All protective trips will be required during periodic testing. Any other trips retained during a design-basis event must utilize coincident logic in order to avoid spurious trips. With the exception of engine overspeed and generator differential, all trips will be bypassed under an accident signal. This is in conformance with Regulatory Guide 1.9 and is, therefore, acceptable.

The diesel generators will be monitored from the control room, and each protective device, when actuated, will initiate an annunciator in the control room. These functions also will be provided with alarms in the diesel generator room. The alarms will be set so that they provide a warning of impending trouble before the diesels trip. Diesel generator status will be indicated and alarmed in the control room. If running, automatic shutdown of a diesel generator also will be annunciated. The local diesel generator starting mode selector switch normally will be in the automatic mode position. The only time it will be in the manual position is for maintenance. The diesel generator starting mode selector switch will be located in the control room and normally will be in the automatic mode (emergency standby) position. The only time it will be in the manual (off-auto) position is during maintenance or routine periodic testing. When the switch is in the manual position, an alarm in the control room will persist until the switch is returned to the automatic mode. Other manual controls will alarm if failure to return them to the start position inhibits automatic operation of the diesel generators. The staff concludes that each condition that renders a diesel generator unit incapable of responding to an automatic emergency start signal will be alarmed in the control room. The alarm system display of "by-passed and inoperable status" of the diesel generators fulfills the requirements of Regulatory Guide 1.47 and is, therefore, acceptable.

Four independent Class 1E 120-V vital instrument ac power supplies will be provided to supply the four channels of the protection system and reactor control systems. Each vital instrument ac power supply will consist of one inverter, one distribution bus, and one manual transfer switch. Normally, the inverter, which will be fed by a separate Class 1E battery system, will supply power to the vital ac bus. If an inverter is inoperable or is to be removed from service, the vital ac bus will be supplied from the spare inverter or alternate source via 480/120-V regulating transformer. A key interlock will be provided to ensure that only a single power source is connected to the vital bus at one time.

The nonvital 120-V instrument ac power supply is designed to furnish reliable power supply to all non-safety-related plant instruments. The nonvital instrument ac system will be divided into two panelboard sections. Each section will be supplied by a common single-phase isolation transformer connected to a Class 1E motor control center. In the event of the loss of normal auxiliary power, the transformers will be automatically energized by the emergency diesel generators.

The staff finds the four uninterruptible power supplies for the reactor protection systems are designed to be independent.

The applicant has applied the following design criteria to the Class 1E equipment:

- Motor Size

  For all motors rated above 480-V, the horsepower is equal to or greater than the maximum horsepower required by the driven load normal running or runout conditions.

- Minimum Motor Accelerating Voltage

  All Class 1E motors are specified with accelerating capability with voltage at 75 percent of the motor nameplate rating. To prevent valve damage from the oversizing of motors, all motor-operated valve actuators are specified with accelerating capability with voltage at 80 percent of the nameplate rating. The electrical system is designed so that the total starting voltage drop on the Class 1E motor circuits is less than that required to accelerate those motors.

- Motor Starting Torque

  The motor starting torque is capable of starting and accelerating the connected load to normal speed within sufficient time to perform its safety function for all expected operating conditions.

- Minimum Motor Torque Margin Over Pump Torque Through Accelerating Period

  The minimum torque margin (accelerating torque) is such that the pump-motor assembly reaches nominal speed within sufficient time to perform its safety function at design minimum terminal voltage.

- Interrupting Capacities

  The interrupting capacities of the protective equipment are determined as follows:

  - Switchgear

    Switchgear interrupting capacities are greater than the maximum short-circuit current available at the point of application. The magnitude of the short-circuit currents in the medium-voltage systems is determined in accordance with ANSI Standard C37.010-1972. The offsite power system, a single operating diesel generator, and running motor contributions are considered in determining the fault level. All motors connected to the bus are considered to be running when the short circuit is postulated. High-voltage circuit breaker interrupting capacity ratings are selected in accordance with ANSI Standard C37.06-1979.

  - Load Centers, Motor Control Centers, and Distribution Panels

    Load centers, motor control centers, and distribution panel circuit breakers have a symmetrical rated interrupting current as great as the determined total available symmetrical current at the point of application. Symmetrical current is determined in accordance with the procedures of ANSI C37.16-1973 for low-voltage circuit breakers other than molded-case breakers and of National Electric Manufacturers Association (NEMA) Standards Publication AB 1 for molded-case circuit breakers.

- Grounding Requirements

  Equipment and system grounding will be designed using IEEE Standard 80-1971, "Guide for Safety in AC Substation Grounding," and IEEE Standard 142-1972, "Recommended Practice for Grounding of Industrial and Commercial Power Systems," as a guide.

The above criteria are in conformance with the criteria cited in Section 8.1 of this report and are acceptable.

The staff finds that there are no automatic transfers of loads or sources between redundant emergency buses, which is in accordance with Regulatory Guide 1.6. There is no sharing of emergency power sources between units, which is in accordance with Regulatory Guide 1.81, "Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants." The two divisions of the emergency power and distribution system are independent, the overall system meets the single-failure criterion and has the capability and capacity as required by GDC 17. The design is in conformance with IEEE Standard 308-1974, as endorsed by Regulatory Guide 1.32. The electric power systems are designed to permit inspection and testing of all Class 1E systems. Periodic testing will be performed on a scheduled basis to demonstrate the operability and continuity of all safety-related systems and components, in accordance with GDC 18. Therefore, the staff finds the emergency onsite ac power system for the RESAR SP/90 design to be acceptable.

8.3.2 DC Power Systems (RESAR SP/90 Module 9, Section 8.3.2)

The station dc systems are designed to supply power to the plant instrumentation and control under all modes of plant operation. In addition, upon loss of ac power, the dc systems will provide power for emergency lighting and turbine-generator auxiliary motors.

The dc power system will consist of four independent Class 1E 125-V dc subsystems, one non-Class 1E 125-V dc system, and one non-Class 1E 250-V dc system. The dc power system is designed to provide reliable and continuous power for controls, instrumentation, inverters, and dc emergency auxiliaries.

Each Class 1E dc power subsystem will consist of one 125-V battery, one battery charger, one inverter and distribution switchboards. The battery chargers for dc subsystems A and C will be supplied power from different Class 1E buses of load group 1 (Train A). Similarly, the battery chargers for dc subsystem B and D will be supplied 480-V ac power from different Class 1E buses of load group 2 (Train B). The inverters will provide four independent 120-V ac vital instrumentation and control power supplies for the channels of reactor protection and engineered safety features. Redundancy and independence of components will preclude the loss of both trains as a result of a single failure. There are no bus ties or sharing of power supplies between redundant trains.

The Class 1E batteries will be sized to have capacity that is greater than 50 percent above the required capacity based on initial design. This margin is greater than that required by IEEE Standard 450-1975, which requires 80-percent capacity battery replacement. Batteries will be sized for 2 hours of operation without the support of battery chargers. The capacity of each Class 1E battery charger will be based on the largest combined demand of all the steady-state loads and the charging capacity to restore the battery from the design minimum charge state to the fully charged state within 12 hours regardless of the status of the plant at the time at which these demands occur. The Class 1E batteries, chargers, and dc switchgear of each separation group will be located in separate rooms of the seismic Category 1 building and are designed to maintain their functional capability during and after a safe shutdown earthquake. Electrical separation also will be maintained to ensure that a single failure in one train does not cause failure of the redundant train. There will be no sharing between redundant Class 1E trains of equipment such as batteries, battery chargers, or distribution panels. The batteries will be located in separate rooms where the ventilation system is designed to preclude hydrogen accumulation. Battery room temperature also will be controlled or the batteries will be derated accordingly.

One spare battery charger and one spare inverter will be provided for the Class 1E dc power block. These items will be physically located central to all of the Class 1E dc systems. However, they will not be electrically connected. In the event of the failure of these devices, the spare can be connected to the affected subsystem. Therefore, the malfunctioning inverter or charger can be repaired without imposing long-term disruption of the system.

The specific requirements for monitoring the dc power systems derive from the generic requirements in Sections 5.3.2(4), 5.3.3(5), and 5.3.4(5) of IEEE Standard 308-1974. In summary, these general requirements state that the dc system composed of batteries, distribution systems, and chargers shall be monitored to the extent that it can be shown to be ready to perform its intended function.

Accordingly, the guidelines used in the staff's review of the dc power system designs are that the following indications and alarms of the Class 1E dc power system should be provided in the control room:

- battery current (ammeter-charge/discharge)
- battery charger output current (ammeter)
- dc bus voltage (voltmeter)
- battery charger output voltage (voltmeter)
- battery discharge alarm
- dc bus undervoltage and overvoltage alarm
- dc bus ground alarm (for ungrounded systems)
- battery breaker(s) or fuse(s) open alarm
- battery charger output breaker(s) or fuse(s) open alarm
- battery charger trouble alarm (one alarm for a number of abnormal conditions which are usually indicated locally)

The staff concludes that the above-cited monitoring, augmented by the periodic test and surveillance requirements that will be included in the technical specifications, provide reasonable assurance that the Class 1E dc power system is designed to perform its intended safety function.

Compliance of the 125-V dc power system for RESAR SP/90 to the above cited position was discussed with the applicant. By letter dated July 14, 1986, the applicant provided additional information in response to the staff's concerns. In regard to the 125-V dc system, the applicant indicated that control room instrumentation or annunciation shall be provided to monitor the status of each Class 1E dc system or as follows:

- battery current (also local)
- battery charger output current (also local)
- dc voltage indication (also local at dc bus and charger)
- battery discharge alarm
- dc bus undervoltage and overvoltage alarm
- dc bus ground alarm (also at dc bus)
- battery breaker trip or open
- battery charger malfunction alarm

8.3.3 Conclusion

The staff concludes that two fully redundant Class 1E systems will be provided. The systems will be testable, independent, and conform to the requirements of Regulatory Guides 1.6 and 1.32. These systems meet the requirements of GDC 17 and 18 and are, therefore, acceptable.

8.4 Other Electrical Features and Requirements for Safety

The staff reviewed certain electrical subsystems or aspects of the RESAR SP/90 station design to determine that these electrical features and requirements are implemented in accordance with all applicable acceptance criteria set forth in Section 8.1 of this SER.

8.4.1 Containment Electrical Penetrations

To meet the requirements set forth in IEEE Standard 317-1972, "Electrical Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations," as augmented by the recommendations of Regulatory Guide 1.63 the containment electrical penetration assemblies for RESAR SP/90 should be designed to withstand, without loss of mechanical integrity, the maximum available fault current for a period of time sufficiently long to allow backup circuit protection to operate assuming a failure of the primary protective device. The circuit overload protection systems for electrical penetration assemblies should meet the single-failure criterion.

In response to the staff's Draft Safety Evaluation Report, issued June 1988, Westinghouse has supplied the following additional information and commitment by way of "Westinghouse Response to Draft Safety Evaluation Report Open Issues 42-81," dated May 1989:

The staff accepts the requirements of IEEE Standards 317-1983 and 741-1986 in Regulatory Guide 1.63, Revision 3 (February 1987). The wording of the revision below is based on the current requirements. Details of penetration backup protection is in part dependent on specific characteristics and details of vendor equipment installed in the plant. The revision is as follows:

> Both safety-related and nonsafety-related electrical penetrations are protected against short-circuit damage. Protective devices are selected and coordinated to ensure that rated short-circuit current and rated short-circuit thermal capacity is not exceeded. When a penetration assembly cannot indefinitely withstand the maximum current available due to a fault inside containment backup protection is provided. Backup protection consists of dual primary protection operating separate interrupting devices, or primary and backup protection operating separate interrupting devices. The details of the containment electrical penetration protection (Regulatory Guide 1.63) are described in the following.
>
> 1. The only medium voltage power circuits passing through the electrical penetrations are the reactor coolant pump motor power feeders. Backup protection will be provided by one of two methods. One method provides backup protection through coordination of pump feeder and bus supply breakers. A plant-specific coordination study is required to verify that this method is adequate. A second method provides backup protection via a second breaker in series. If this second method is utilized, the trip circuits will be supplied from separate battery systems.
>
> 2. For 480-volt loads fed from load centers, primary protection is provided by the feeder breaker for the individual load. Backup protection is provided by series fuses or by coordinated protection with the load center bus supply breaker. Where a backup circuit breaker is used, the trip circuits will be supplied from separate battery systems.

3. For 480-volt loads fed from motor control centers, primary protection is provided by the individual circuit breaker. Coordinated protection with the motor control center supply breaker is the preferred method of backup protection. Protection for each circuit will be reviewed. Where coordinated protection cannot be achieved, backup protection will be provided by an independent current limiting device installed with the supply breaker or by a separate backup breaker connected in series.

4. 125-Vdc control circuits are protected by fuses and the system is ungrounded. Backup protection is provided by distribution supply breakers.

5. 120-Vac control circuits are low energy circuits which are protected by one fuse. The available short circuit current for faults in the containment is generally sufficiently low that backup protection is not required. Control circuits will be analyzed and backup devices will be provided where required.

6. Instrumentation circuits are low energy circuits. The available short circuit current for faults in the containment is generally sufficiently low that backup protection is not required. Instrumentation circuits will be analyzed and backup devices will be provided where required.

During the detailed design phase, the exact method of overcurrent protection for each containment penetration conductor will be defined. Containment penetration overcurrent protective devices including backup devices which are required to be operable will be identified in the FDA submittal.

The staff finds these criteria acceptable with the exception of the medium voltage power circuits feeding the reactor coolant pumps. To protect the medium voltage circuit containment penetration from a single failure of dc control power, which would prevent the primary and backup feeder circuit breaker from clearing an electrical fault in the RCP motor power feeder circuit, independent sources of dc control power should be provided for the RCP motor feeder breaker and bus supply breaker. Therefore, if the first method is to be used, independent sources of dc control power are to be provided for the RCP motor feeder breaker and the bus supply breaker. This is not in accordance with IEEE Standard 317 as augmented by Regulatory Guide 1.63 and is, therefore, not acceptable. The staff will review the method of overcurrent protection for containment penetration protection during the FDA stage or during review of a plant-specific application referencing the RESAR SP/90 design.

8.4.2 Thermal Overload Protection Bypass

Motor-operated valves with thermal overload protection devices for the valve motors will be used in safety systems and their auxiliary supporting systems. Operating experience has shown that indiscriminate application of thermal overload protection devices to the motors associated with these valves could result in needless hindrance to successful completion of safety functions. Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor-Operated Valves," Revision 1, recommends (in Position C.1) bypassing thermal overload devices during accident conditions or (in Position C.2) selecting the set points for the thermal overloads in a manner that precludes spurious trips. In the RESAR SP/90 design, before core loading, the thermal overload relay trip contacts for all Class 1E valves are permanently bypassed. Position C.1 of Regulatory Guide 1.106 recommends (1) the thermal overloads to be continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing or (2) those thermal overloads that are normally in force during plant operation to be bypassed under accident conditions.

While the RESAR SP/90 design satisfies the concern for inadvertent tripping of safety equipment by thermal overloads during accident conditions, it also unnecessarily subjects the motors to overloads that may occur during testing and normal operation resulting in potential motor damage. It is the staff's judgment that one of the positions in Regulatory Guide 1.106 should be used as described to provide protection against motor failure for more frequent use such as during testing. Westinghouse should address this concern for the staff's review during the FDA stage of review.

8.4.3 Power Lockout to Motor-Operated Valves

BTP ICSB 18 (PSB) specifies that all valves requiring power lockout meet the single-failure criterion for fluid systems and their required position be listed in the technical specifications. The position indications for these valves also must meet the single-failure criterion and redundant valve status indication must be provided to the main control room operator.

Motor-operated valves in the SP/90 design that require power lockout to meet the single-failure criteria are

- accumulator isolation valves 8949 A, B, C, D
- core reflood tank isolation valves 9097 A, B, C, D

Westinghouse has committed to incorporating this information into the technical specifications to be submitted during the RESAR SP/90 FDA stage of review.

The staff will review the inclusion of these valves and valve indications in the technical specifications during the FDA stage or during the licensing process of a plant-specific application referencing the RESAR SP/90 design. Additionally, the staff will review the position indications against the single-failure criterion and the inclusion of redundant valve status indication at the same time.

8.4.4 Physical Identification and Independence of Redundant Safety-Related Electrical Systems

The applicant has provided the criteria for physical identification and separation of electrical equipment to preserve the independence of redundant equipment. Physical identification of safety-related electrical systems will be accomplished as discussed below.

Each cable and raceway will be given a unique alphanumeric identification and will be color coded to indicate its separation group. This identification will provide a means of distinguishing a cable or raceway associated with a particular separation group.

Exposed raceways containing Class 1E cables will be marked by color codes in a distinct permanent manner at intervals not to exceed 15 feet and at points of entry to and exit from enclosed areas. In general, all Class 1E cables and associated cables will have color-coded jackets throughout their entire length. Field cables will be color coded at intervals not to exceed 5 feet. Equipment, raceways, and cables in raceways associated with all normal plant equipment that is not Class 1E will be uniquely identified and separately routed from safety-related cables and raceways.

The minimum separation between redundant cable trays will be 3 feet between trays separated horizontally and 5 feet between trays separated vertically for plant areas that are free from potential hazards such as missiles, external fires, and pipe whip. In the cable spreading areas and the control room, the minimum separation between redundant cable trays will be 1 foot between trays separated horizontally and 3 feet between trays separated vertically. Where termination arrangements preclude maintaining the minimum separation distance, the redundant circuits will run in solidly enclosed raceways or other barriers provided between redundant circuits in accordance with Regulatory Guide 1.75.

In cases where redundant trays cross over each other in areas where only electrical equipment is located, there will be a minimum vertical separation of 15 inches (free air space). A barrier will be provided to extend 1 foot from each side of the trays and 3 feet along each tray from the crossover.

Each diesel generator, including its associated auxiliaries, will be located in a separate room of a seismic Category 1 structure. Power and control cables for the diesel generators and associated switchgear will be routed to maintain physical separation.

Four Class 1E battery supplies will be located in the seismic Category 1 building. Each battery will be located in a separate room. The battery charger, inverter, and dc buses associated with each of the four subsystems will be in separate rooms outside the battery rooms.

The staff finds the applicant's design criteria regarding physical identification, separation, and independence of redundant safety-related electrical systems to be in accordance with Regulatory Guide 1.75, however, due to fire protection policy for advanced plants placing more restrictive design criteria on electrical separation it is expected that this design will change significantly. The physical independence of redundant safety-related electrical systems will be reviewed during the plant specific licensing process.

The implementation of these criteria is the responsibility of the plant-specific applicant. The staff will verify the implementation of these design criteria during the licensing process for plant-specific applications referencing the RESAR SP/90 design.

8.4.5 Nonsafety Loads on Emergency Sources

Present regulatory practice for operating license applications allows the connection of nonsafety loads--in addition to the required safety loads--to Class

1E (emergency) power sources if it can be shown that the connection of nonsafety loads will not result in degradation of the Class 1E system. The RESAR SP/90 design provides for the connection of both safety and essential non-Class 1E loads to the Class 1E power systems.

The design provides a safety feature sequencer that will automatically disconnect all nonsafety loads connected to 4.16-kV and 480-V circuit breakers upon receipt of a safety injection signal. After the breakers have been tripped and the safety injection signal has been manually reset, the operator can close these under administrative control to re-energize the selected nonsafety loads, should their operation be desired

The staff concludes that the design is in accordance with Regulatory Guide 1.75 and acceptable. Use of isolation devices meets the requirements of isolation devices per IEEE Standard 384-1974 and Regulatory Guide 1.75, Revision 2, and is acceptable.

8.4.6 Submerged Electrical Equipment as a Result of a Loss-of-Coolant Accident

The staff is concerned that fluid from the reactor coolant system and from operation of the emergency core cooling system (ECCS), following a LOCA, may collect in the primary containment and reach a level that may cause certain electrical equipment located inside the containment to become submerged and thereby rendered inoperable. The failure of both safety-related and non-safety-related electrical equipment may cause electrical faults that could compromise the operability of redundant emergency power sources or the integrity of containment electrical penetrations. In addition, the safety-related electrical equipment that may be submerged would be required to mitigate the consequences of an accident for the short-term and long-term ECCS functions and for containment isolation.

The applicant has stated that all electrical equipment will be located above the maximum flood level. The spillways for the emergency water storage tank will be extended above the containment floor, precluding flooding of essential electrical equipment. The staff finds this acceptable.

8.4.7 Use of a Load Sequencer With Offsite Power

The RESAR SP/90 design includes load sequencing for the connection of loads for engineered safety features to the emergency buses when power is being supplied either from offsite sources or from the diesel generators.

The load sequencing will be accomplished by the integrated protection system, in particular by the redundant trains of the engineered safety features actuation (ESFAC) subsystem. If safety limits for plant parameters were being approached, the ESFAC subsystem would transmit system-level actuation signals to the integrated logic cabinets (ILCs) to actuate the required protective components. These ESFAC signals will include the necessary sequencing information where applicable. The status of the Class 1E electrical buses, and therefore the sequencing information, will be transmitted to the ILCs depending on the availability of offsite power.

Westinghouse has committed to demonstrating the reliability of the sequencing function as part of the integrated protection system verification and validation efforts for the FDA stage of review.

The staff finds this acceptable for the PDA stage and will verify implementation at the FDA stage of review.

8.4.8 Station Blackout (Unresolved Safety Issue (USI) A-44)

Although station black was addressed in Module 2 of the RESAR SP/90 application, the applicant will be required to update its response to satisfactorily address station blackout during the FDA stage of review.

8.4.9 Conclusion

The staff concludes that the design of offsite power system, including its protection schemes, permits periodic testing and is in accordance with GDC-18 and is acceptable. However, the staff will review the design of the RCP motor electrical penetrations further during the FDA stage. The staff will review the additional information required with regard to motor-operated valves that require power lockout when it is supplied.

The staff will verify the reliability of the sequencing function during the FDA stage of review.

In addition, the applicant will be required to revise its station blackout response to satisfactorily address station blackout during the FDA stage.

8.5 Relevant Criteria Status Information

The applicant has committed to conform to Branch Technical Positions ICSB-2, "Diesel Generator Reliability Qualification Testing," and ICSB-17, "Diesel Generator Protective Trip Circuit Bypass." These positions are contained in IEEE Standard 387-1977 and Regulatory Guide 1.9, Revision 2, which endorses the standard. The applicant also has committed to conform to IEEE Standard 387-1977 and Regulatory Guide 1.9, Revision 2. This second commitment is discussed below.

The staff is presently revising Regulatory Guide 1.9 to cover station blackout and diesel generator reliability maintenance, to endorse IEEE Standard 387-1984 and to incorporate the periodic testing guidelines contained in Regulatory Guide 1.108, which will then be withdrawn. It is not expected that this revision will substantively alter existing guidelines except in regard to station blackout. Therefore, Regulatory Guide 1.9, Revision 3, will be applied during future licensing processes of plant-specific applications referencing the RESAR SP/90 design.

9 AUXILIARY SYSTEMS (RESAR SP/90 Module 13, Section 9.0)

9.1 Fuel Storage and Handling (RESAR SP/90 Module 13, Section 9.1)

9.1.1 New Fuel Storage (RESAR SP/90 Module 13, Section 9.1.1)

The new-fuel storage facility (NFSF) will be located in the fuel building. It is designed to provide onsite dry storage for 80 new fuel elements, representing approximately 40 percent of a full core load. It will include the new fuel storage racks and the new fuel storage area, which will be totally enclosed with walls and a steel plate top.

The fuel building is designed to seismic Category I criteria, as are the storage racks and the storage area. The building is also designed against flooding and tornado missiles. Thus, the requirements of GDC 2 and the guidelines of Regulatory Guide 1.29 are satisfied.

The NFSF is designed to store unirradiated, low-emission fuel assemblies. Accidental damage to the fuel would release relatively minor amounts of radioactivity that would be accommodated by the fuel handling ventilation system. The facility will be accessible to plant personnel for inspection. Thus, the requirements of GDC 61 with regard to fuel storage and handling and radioactivity control are satisfied.

The new fuel racks are designed to store the fuel assemblies in an array with minimum center-to-center spacing of 13.29 inches, which is sufficient to maintain the array in a subcritical condition even when fully loaded. For a flooded condition, assuming new fuel of the highest anticipated enrichment, the effective multiplication factor will not exceed 0.95. The effective multiplication factor will not exceed 0.98 with fuel of the highest anticipated enrichment and optimum moderation (aqueous foam or mist). A cover will be provided over each fuel storage position. Covers and racks are designed to preclude inadvertent placement of a fuel assembly in other than its prescribed position. Thus, the requirements of GDC 62 with regard to prevention of criticality of fuel storage and handling are satisfied.

The staff concludes that the NFSF is in conformance with the requirements of GDC 2, 61, and 62 as they relate to protection against natural phenomena, radiation protection, and prevention of criticality and that it is acceptable.

9.1.2 Spent Fuel Storage (RESAR SP/90 Module 13, Section 9.1.2)

The spent-fuel storage facility (SFSF) will be located in the fuel building. It is designed to provide underwater storage for approximately 1000 fuel assemblies. The facility will include the spent fuel racks, the lined spent fuel pit containing the storage racks and the adjacent areas of the fuel transfer canal, and the spent fuel shipping cask loading pit. The spent fuel pit is separated from the adjacent areas by leaktight gates.

The fuel building is designed to seismic Category I criteria, as are the storage racks and pit, the fuel transfer canal area, and the shipping cask loading area.

The leaktight gates are seismically designed to preclude their failure during an SSE and falling onto the spent fuel racks. The stainless steel spent fuel pit liner plate also is designed to stay in place in an SSE, thus preventing potential mechanical damage to the spent fuel or blocking the flow of cooling water around the fuel. The fuel building also is designed against flooding and tornado missiles. This satisfies the requirements of GDC 2 and the guidelines of Regulatory Guides 1.13, 1.29, and 1.117.

The SFSF is designed to maintain structural integrity following postulated hazards such as internal missiles and pipe breaks according to the design bases of the system. This satisfies the requirements of GDC 4 and the guidelines of Regulatory Guide 1.13.

The design provides the capability to perform periodic inspections and testing, and the capability of providing suitable shielding for radiation protection according to the system's design bases. This satisfies the requirements of GDC 61.

The seismic Category I storage rack arrangement provides a fuel array adequate to maintain the effective multiplication factor below 0.95 for both normal storage and in a case of accidental dropping of a fuel assembly. The design of the storage racks is such as to preclude placement of a fuel assembly in a position other than the prescribed location. The racks are designed to withstand the impact of a dropped fuel assembly without unacceptable damage to the fuel and can withstand the maximum uplift force exerted by the fuel-handling machine. This satisfies the requirements of GDC 62.

The spent fuel pit cooling and cleanup system is designed with instrumentation to detect conditions that could result in the loss of decay heat removal capabilities. In addition, the liner seam welds will have a collection system to detect leaks in the spent fuel pit liner although the SFSF is designed to maintain leaktight integrity to prevent the loss of cooling water from the pit. In the event of a loss of integrity of the watertight gate, while one of the adjacent smaller pits on the fuel transfer canal is drained, a minimum of 10 feet of water is maintained above the top of the fuel. In addition, all piping penetrations into the pit are designed to preclude draining the pit down to an unacceptable level. This satisfies the requirements of GDC 63.

The staff concludes that the design of the spent fuel storage facility is in conformance with the requirements of GDC 2, 4, 61, 62, and 63 as they relate to protection against natural phenomena, missiles, pipe break effects, radiation protection, prevention of criticality, and monitoring provisions, as well as the guidelines of Regulatory Guides 1.13, 1.29, and 1.117 and is acceptable.

9.1.3 Spent Fuel Pit Cooling and Cleanup System (RESAR SP/90 Module 13, Section 9.1.3)

The spent fuel pit cooling and cleanup system (SFPCS) will consist of a cooling portion and a purification portion. The cooling portion of the SFPCS is designed to remove the decay heat generated by the stored fuel assemblies. The purification portion is designed to remove impurities and to limit the concentrations of radioactive materials in the water used in the reactor cavity, the spent fuel pit, and the cask loading pit, and the refueling water stored in the emergency water storage tank. This system, in conjunction with the chemical and volume control system and the reactor makeup water system, will be capable of adjusting the boric acid concentration and water inventory of the emergency water storage tank and the spent fuel pit.

The cooling portion of the SFPCS will consist of two 100-percent capacity independent cooling trains, each with a pump and a heat exchanger. The purification portion will consist of demineralizers, filters, and pumps as well as a skimmer system to keep the surface of the water in the spent fuel pit and the reactor refueling cavity clean.

The cooling portion of the SFPCS and the nonisolatable portions associated with the emergency water storage tank and containment penetrations are designed as safety related. The safety-related portions of the system are housed in the auxiliary equipment area of the reactor external building, which is a seismic Category I, flood- and tornado-protected structure. The system itself, with the exception of the purification portion, is designed to Quality Group C and seismic Category I requirements. Failure of the portion that is not safety related will not affect the operation of the cooling train because it will be provided with an isolation capability. The design meets the requirements of GDC 2 and the guidelines of Regulatory Guides 1.13, 1.26, and 1.29 with regard to seismic and quality group classification of the spent fuel pit cooling system.

The SFPCS is designed to remain functional and perform its intended function after hazards such as internal missiles and pipe breaks, thus satisfying GDC 4.

Each of the cooling trains of the SFPCS will limit the spent fuel pit water temperature to a maximum of 120°F during normal refueling. The system will limit the spent fuel water temperature to less than or equal to 150°F following an emergency core removal, immediately following a normal refueling, with both trains in operation. Finally, the system is designed to maintain the spent fuel pit water temperature less than 185°F, taking into consideration maximum anticipated decay heat generation rate resulting from emergency removal of a full core immediately following a normal refueling with the maximum anticipated burnup and decay heat, with the fuel placed in the pit at the minimum anticipated time, with the maximum cooling water temperature, and with the worst single failure. These temperature limits fall within the staff's acceptance criteria. However, the RESAR-SP/90 application does not address the analysis conducted and the parameters used to determine the heat removal rate for the spent fuel pit heat exchangers. The applicant should provide this analysis during the FDA stage of review so that the staff can verify the required heat removal rate against the guidances described in the BTP ASB 9-2, "Residual Decay Energy for Light-Water Reactors for Long-Term Cooling."

All connections to the spent fuel pit will be near the normal water level or provided with antisiphon holes to preclude possible siphon draining of the pit water. The safety-related component cooling water system will provide cooling water to the fuel pit heat exchanger and transfer its heat to the ultimate heat sink via the essential service water system. Capability will be provided to isolate components so that safety functions are not compromised. The necessary redundancy also will be provided so that safety functions can be performed assuming a single active component failure coincident with loss of offsite power. This satisfies the requirements of GDC 44.

The active components of the SFPCS will be able to be tested during plant operation and provisions will be made to allow for in-service inspection of components at appropriate times specified in ASME Code, Section XI. This satisfies the requirements of GDC 45 and 46.

Normal makeup to the spent fuel pit to replace losses and maintain proper water level for shielding will be provided by the reactor makeup water system, providing a seismic Category I source. Backup makeup water will be provided by the emergency water storage tank, which also is a seismic Category I source. This satisfies the requirements of GDC 61 and the guidelines of Regulatory Guide 1.13.

The design includes provisions to detect and alarm loss of flow and high temperature conditions that could result in a possible loss of decay heat removal capabilities. The liner plant welds also will be equipped with a leak-chase system to detect and collect leakage through the liner. This satisfies the requirements of GDC 63.

The purification portion of the SFPCS will provide the capability to remove radioactive materials, corrosion products, and impurities from the spent fuel pit water, thus meeting the requirements of GDC 61 with regard to the appropriate filtering systems for fuel storage. It also will be capable of reducing occupational exposure of radiation by removing radioactive products, thus meeting the requirements of 10 CFR 20.1(c) with regard to maintaining radiation exposures as low as reasonably achievable. The SFPCS purification portion will confine radioactive materials in the pit water in the demineralizer and filters, thus meeting Regulatory Position C.2.f(2) of Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable," with regard to reducing the spread of contaminants from the source. It will remove suspended particles from the pit water by filters, thus meeting Regulatory Position C.2.f(3) of Regulatory Guide 8.8 with regard to removing crud through physical action.

Subject to verification of the heat removal rate at the final design stage, the staff concludes that the design of the cooling portion of the spent fuel pit cooling system is in conformance with the requirements of GDC 2, 4, 44, 45, 46, 61, and 63 as they relate to protection against natural phenomena, missiles and environmental effects, cooling water capability, in-service inspection, functional testing, fuel cooling and radiation protection, and monitoring provisions and with the guidelines of Regulatory Guides 1.13, 1.26, and 1.29 relating to the system's design, seismic and quality group classifications. It is acceptable.

The staff also concludes that the purification portion of the system meets GDC 61, 10 CFR 20.1(c), and the appropriate sections of Regulatory Guide 8.8 and is acceptable.

9.1.4 Light Load Handling System (Related to Refueling) (RESAR SP/90 Module 13, Section 9.1.4)

The light load handling system (LLHS), in conjunction with the fuel storage area, will provide the means of transporting, handling, and storing new and spent fuel. The system will have the equipment necessary to conduct refueling operations, such as the refueling machine, fuel handling machine, new fuel elevator, and fuel transfer system, and associated handling tools and devices. The handling of fuel during refueling is controlled by a series of interlocks to ensure that fuel handling procedures are maintained.

The refueling machine design is a rectilinear bridge and trolley system with a vertical mast extending down into the refueling water. The machine will be used to handle new and spent fuel within the reactor vessel and refueling cavity inside the containment. Electrical interlocks and limit switches on the bridge and trolley drives will prevent damage to fuel assemblies. The refueling machine will be restricted so that it only raises a fuel assembly to a height at which the water provides a safe radiation shield.

The fuel handling machine design is similar to the refueling machine design without a telescoping mast. It will be used exclusively for handling fuel assemblies within the fuel handling area. A number of safety features, such as interlocks, overload protection devices, and restraining bars, will be provided to prevent damage to fuel assemblies.

The new fuel elevator will be used to lower new fuel assemblies to the bottom of the spent fuel pit where they can be transported by the fuel handling machine to the storage racks.

The fuel transfer system will include an underwater motor-driven transfer car that runs on tracks extending from the refueling cavity, through the transfer tube, and into the refueling canal. Fuel assemblies will be placed on the car within the refueling cavity by the refueling machine and removed in the fuel handling area by the fuel handling machine after passing through the transfer tube. Interlocks and limit switches will be provided to prevent unsafe operation.

The system will be housed within the fuel handling area of the reactor external building, which is a seismic Category I, flood- and tornado-protected structure. In addition, in the event of an SSE, the handling equipment will not fail in such a manner as to damage seismic Category I equipment. This satisfies the requirements of GDC 2.

The RESAR-SP/90 application indicates that the maximum kinetic energy developed by some loads exceeds that developed by a fuel assembly and its associated tool, if dropped. However, an analysis showing that such an event would not lead to unacceptable damage to the spent fuel has not been provided. The staff will require this analysis during the FDA stage of review to verify that the effects of such an event will not exceed the effects of postulated fuel handling accidents.

RESAR-SP/90 states that the system conforms to GDC 61 and 62 in that it meets the guidelines of ANS 57.1, "Design Requirements for LWR Fuel Handling System." The design provides for inspection and testing of safety-related components and suitable shielding for radiation protection.

Pending resolution of the above items, the staff concludes that the design of the LLHS is in conformance with the requirements of GDC 2, 61, and 62 with regard to protection against natural phenomena and to safe fuel handling, including prevention of criticality, and is acceptable.

9.1.5 Overhead Heavy-Load Handling System (RESAR SP/90 Module 13, Section 9.1.5)

The overhead heavy load handling system (OHLHS) will have the equipment necessary for the safe handling of the spent fuel cask and for safe disassembly, handling, and reassembly of the reactor vessel head and internals during refueling operations. The containment polar crane will be used for handling heavy loads in containment and the cask handling crane will be used for handling heavy loads in the fuel handling area.

The system will be housed within the fuel handling area of the reactor external building and the containment. Both buildings will be seismic Category I, flood- and tornado-protected structures. The containment polar crane and the cask handling crane are designed to seismic Category I requirements. The design satisfies the requirements of GDC 2 and the guidelines of Regulatory Guides 1.13 and 1.29.

The OHLHS is designed in accordance with the guidance of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," Section 5.1, and NUREG-0554, "Single-Failure-Proof Cranes for Nuclear Power Plants." The design satisfies the requirements of GDC 4 and 61.

The staff concludes that the OHLHS is in conformance with the requirements of GDC 2, 4, and 61 with regard to protection against natural phenomena and internal missiles and to safe handling of the spent fuel cask, and the guidelines of NUREG-0612 and Regulatory Guides 1.13 and 1.29 and is acceptable.

9.2 Water Systems (RESAR SP/90 Module 13, Section 9.2)

9.2.1 Essential Service Water System (RESAR SP/90 Module 13, Section 9.2.1)

The essential service water system (ESWS) is designed as a safety-related system to provide water to selected plant components and transfer heat from these components to the plant ultimate heat sink. The system will provide cooling to the component cooling heat exchangers, emergency diesel generator coolers, and essential chilled water chiller units. The system will operate under normal and emergency conditions.

The RESAR-SP/90 application indicates that the detailed design of the ESWS is not within the scope of the Westinghouse advanced pressurized water reactor (WAPR) power block since the ultimate heat sink is site dependent. However, Westinghouse has established the design criteria and preliminary design to ensure that the ESWS will be compatible with the systems in the WAPR power block scope.

The system will be protected against the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, and external missiles. Details of the structures that will house the system will be provided in a site dependent addendum. In addition, the system itself will be designed to the Quality Group Classification assigned by Regulatory Guide 1.26 and the seismic category assigned by Regulatory Guide 1.29. Therefore, the requirements of GDC 2 will be satisfied.

The system will be designed to remain functional after an SSE and to perform its intended function following the postulated hazards of internal missiles and pipe breaks. Therefore, the requirements of GDC 4 will be satisfied.

The system will have two totally independent subsystems, each of which will consist of two essential service water (ESW) pumps, two strainers, and associated piping valves, and instrumentation. Following a design-basis event, such as a LOCA, the ESW pumps will automatically start. A single ESW pump in either one of the two ESWS subsystems will provide sufficient water to safeguard components to ensure post-accident design-basis heat removal. Each of the two ESW pumps in each of the two subsystems will receive emergency electrical power from its associated emergency diesel generator. All four pumps will automatically start following a station blackout. The design provides the capability to isolate components or piping so that the ESWS's safety function is not compromised. This includes isolation to deal with leakage and isolation of portions that are not safety related. This satisfies the requirements of GDC 44.

The active components of the ESWS will be capable of being tested during plant operation. Provisions will also be made to allow for inservice inspection of components at appropriate times specified in the ASME Boiler and Pressure Vessel Code, Section XI. This satisfies the requirements of GDC 45 and 46.

The staff concludes that the design criteria and the preliminary design of the essential service water system, which will be finalized for a specific site, are acceptable. However, the staff will review the ESWS in detail against the guidance of SRP Section 9.2.1, "Station Service Water System," during the plant-specific licensing process referencing the RESAR SP/90 design.

9.2.2 Component Cooling Water System (RESAR SP/90 Module 13, Section 9.2.2)

The component cooling water system (CCWS) is designed as a safety-related, closed loop, cooling system that transfers heat from various components in the reactor auxiliary systems to the essential service water system (ESWS) during all phases of operation.

The CCWS will supply cooling to safety-related and non-safety-related plant components during normal operation and to safety-related components during accident and emergency conditions. The system will serve the component cooling water pump motors; the containment fan coolers; the high-head safety injection pumps and motors; the motor-driven emergency feedwater pumps; the reactor coolant pump thermal barrier heat exchangers and motor bearings; the residual heat removal heat exchangers, miniflow heat exchangers, pumps and motors; the air compressor; the charging pumps and motors; the letdown and excess letdown heat exchangers; the reactor coolant pump motor exhaust; the spent fuel pit heat exchangers; the boric acid evaporator; the sampling heat exchanger; the control drive mechanism coolers; the reactor coolant drain tank heat exchanger; the waste evaporator; the seal injection heat exchanger; and miscellaneous waste processing system equipment.

The CCWS will have two totally independent subsystems, each of which will consist of two component cooling water (CCW) pumps, two CCW heat exchangers, one CCW surge tank, and associated piping, valves, and instrumentation.

During normal operation, one CCW pump and one CCW heat exchanger in each of the two subsystems will accommodate the heat removal loads. Following a design-basis event that results in a safety injection signal, the standby CCW pumps will automatically start and headers that are not safety related will be automatically isolated. During the initial phase of the event, the CCW will provide cooling to the containment fan coolers and to safety-grade equipment. Following the initial phase, the CCW will remove heat from the water circulated through the RHR heat exchangers. One of the four pumps will be sufficient to provide the required flow to its associated safety-grade components. This satisfies the requirements of GDC 44.

The CCWS is designed as a seismic Category I, Quality Group C system that will be protected from the effects of natural phenomena, such as earthquakes, tornadoes, hurricanes, floods, and external missiles. It satisfies, therefore, the requirements of GDC 2 and the guidelines of Regulatory Guides 1.26 and 1.29.

The CCWS is designed to remain functional after an SSE and to perform its intended function following the postulated hazards of internal missiles and pipe breaks. This satisfies the requirements of GDC 4.

The RESAR SP/90 application does not state how leakage from the CCWS will be monitored or identified, additionally, the makeup water supply for the CCWS will be from the demineralized water system, which serves no safety function and has no safety design basis. The issues of leakage detection and makeup capability following a postulated design-basis event must be addressed during the FDA stage of review.

The active components of the system will be capable of being tested during plant operation. Provisions also will be made to allow for inservice inspection of components at appropriate times specified in ASME Code, Section XI. This satisfies the requirements of GDC 45 and 46.

Subject to resolution of the items noted above, the staff concludes that the design of the CCWS meets the requirements of GDC 2, 4, 44, and 45 with regard to protection against natural phenomena, decay heat removal capability, and inservice inspection and testing, and the guidelines of Regulatory Guides 1.26 and 1.29 with regard to the system's classification and is acceptable.

9.2.3 Demineralized Water System (RESAR SP/90 Module 13, Section 9.2.3)

The demineralized water system (DWS) will store water for use when makeup is needed within the plant. It will receive filtered and demineralized water from the plant water makeup system and provide demineralized water to the condensate storage tank, reactor makeup water storage tank, component cooling water system, closed cooling water system, auxiliary steam system, diesel-generator cooling water expansion tank, chilled water system, and miscellaneous sampling, flushing, and makeup requirements. The DWS will serve no safety function and will have no safety design basis.

The design of the DWS is the responsibility of the plant-specific applicant referencing the RESAR SP/90 design. The RESAR SP/90 application provides the following interface criteria for the system design:

(1) The system will be capable of maintaining chemistry specifications required by the plant components.

(2) The system will have sufficient capacity to supply the anticipated normal makeup water demand in any 24-hour period.

(3) The system also will have sufficient storage capacity to augment the condensate and reactor makeup water storage facilities to ensure a 3-day supply of anticipated normal makeup demand to both the primary coolant and secondary water systems.

The staff concludes that the interface criteria for the DWS are adequate for the stated system functions and are acceptable.

9.2.4 Potable and Sanitary Water System (RESAR SP/90 Module 13, Section 9.2.4)

The design of the domestic water system is the responsibility of the plant-specific applicant referencing the RESAR SP/90 design and will have no safety-related interface with any of the systems covered under the scope of the RESAR SP/90 NPB. The staff will review this system against the guidance described in SRP Section 9.2.4, "Potable and Sanitary Water System," during the plant-specific licensing process of an application referencing the RESAR SP/90 design.

9.2.5 Ultimate Heat Sink (RESAR SP/90 Module 13 Section 9.2.5)

The basic function of the ultimate heat sink system (UHS) will be to provide a reliable seismic Category I source of cool, or cooled, water for use as essential service water.

The RESAR SP/90 application states that the design of the UHS is site dependent and therefore will be provided by the plant-specific applicant referencing the RESAR SP/90 design. However, Westinghouse has established certain design criteria to ensure that the plant-specific design is compatible with the ESWS design.

The staff concludes that the design criteria provided in the RESAR SP/90 application are adequate to enable the plant-specific applicant to provide an acceptable UHS design. However, the staff will review the UHS against the guidance described in SRP Section 9.2.5, "Ultimate Heat Sink," during the plant-specific licensing process of an application referencing the RESAR SP/90 design.

9.2.6 Condensate Storage and Transfer System (RESAR SP/90 Module 13, Section 9.2.6)

The basic function of the condensate and storage transfer system will be to serve as a reservoir to supply or receive condensate, as required, by the condenser hot well control system. The main component of the system, the condensate storage tank (which will not be safety related), also will serve as a backup source of water for the startup feedwater system.

The RESAR SP/90 application states that the design of the system is the responsibility of the plant-specific applicant. The system has no safety-related interface with any other system covered by the scope of the RESAR SP/90 NPB. The staff will review this system against the guidance described in SRP Section 9.2.6, "Condensate Storage Facilities," during the plant-specific licensing process of an application referencing the RESAR SP/90 design.

9.3 Process Auxiliaries (RESAR SP/90 Module 13, Section 9.3)

9.3.1 Compressed Air System (RESAR SP/90 Module 13, Section 9.3.1)

The basic function of the compressed air system will be to provide a supply of filtered, dry, oil-free air for instrument and control operations. Additionally, the system provides station air for operation of pneumatic tools and other service requirements. It also acts as a backup to supply compressed gas to the main feedwater control valves.

The RESAR SP/90 application states that the design of the system is the responsibility of the plant-specific applicant referencing the RESAR SP/90 design.

The staff will review this system against the guidance described in SRP Section 9.3.1, "Compressed Air System," during the plant-specific licensing process of an application referencing the RESAR SP/90 design.

9.3.2 Process and Post-Accident Sampling Systems (RESAR SP/90 Module 13, Section 9.3.2)

9.3.2.1 Process Sampling System (RESAR SP/90 Module 13, Sections 9.3.2.1, 9.3.2.2)

The staff reviewed the process sampling system in accordance with SRP Section 9.3.2.

The process sampling system will be manually operated and is designed to collect liquid and gaseous samples from the reactor coolant systems and associated auxiliary system process streams during all modes of normal plant operation. A number of sample points will be shared with the post-accident sampling system. Local grab samples also can be taken from the volume control tank of the chemical and volume control system, spent fuel pit, emergency water storage tank, boron recycle evaporator, and liquid waste evaporator. These samples will be used to monitor plant conditions and system performance.

To ensure representative samples, all sample lines will be purged before collection of samples. The purge ensures turbulent flows and minimizes crud accumulation or plateout in the sample lines. The purge flow will be routed to the volume control tank of the chemical and volume control system and reused or it will be routed to the waste processing system for disposal.

The staff determined that the process sampling system meets the requirements of

- GDC 13 by sampling the reactor coolant, the core reflood tanks, the emergency water storage tank, and boron recycle evaporator for boron concentration, which can affect the fission process for normal operation, anticipated operational occurrences, and accident conditions
- GDC 14 by sampling the reactor coolant for chemical impurities to ensure that the reactor coolant pressure boundary will have a low probability of abnormal leakage, rapidly propagating failure, or gross rupture
- GDC 26 by sampling the reactor coolant and boron recycle evaporator for boron concentrations for controlling the rate of reactivity changes
- GDC 60 to control the release of radioactive materials to the environment by providing isolation valves that will fail in the closed position
- GDC 63 by sampling the spent fuel pit water for radioactivity to detect conditions that may result in excessive radiation levels
- GDC 64 by sampling the reactor coolant, the pressurizer, the sump inside containment and the containment atmosphere for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents
- Principals for obtaining representative samples of gases contained in ANSI Standard N13.1-1969

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION
TEST TARGET (MT-3)

150mm

6"

- 10 CFR 20.1(c) and Regulatory Positions 2.d(2), 2.f(3), 2.f(8), and 2.i(6) of Regulatory Guide 8.8 by providing ventilation systems and gaseous radwaste treatment system to contain airborne radioactive materials, liquid radwaste treatment system to contain radioactive material in fluids, spent fuel pit cleanup system to remove radioactive contaminants in the spent fuel pit water, and remotely operated containment isolation valves to limit reactor coolant loss in the event of rupture of a sampling line

- Regulatory Positions C.1, C.2, and C.3 of Regulatory Guide 1.26, Revision 3, and C.1, C.2, C.3, and C.4 of Regulatory Guide 1.29, Revision 3, by designing the sampling lines and components of the process sampling system to conform to the classification of the system up to and including the first isolation valves to which each sampling line and component will be connected, thus meeting the quality standards requirements of GDC 1 and the seismic requirements of GDC 2

The staff concludes that the process sampling system is acceptable.

9.3.2.2 Post-Accident Sampling System (NUREG-0737, II.B.3) (RESAR SP/90 Module 13, Sections 9.3.2.2.2, 9.3.2.2.3)

Subsequent to the Three Mile Island, Unit 2, accident, the need was recognized for a post-accident sampling system to determine the extent of reactor core degradation following a severe reactor accident. Criteria for an acceptable sampling and analysis system are specified in NUREG-0737, Item II.B.3. The system should have the capability to obtain and quantitatively analyze reactor coolant and containment building atmosphere samples without radiation exposure to any individual exceeding 5 rem to the whole body or 75 rem to the extremities (GDC 19) during and following an accident in which there is reactor core degradation. Materials to be analyzed and quantified include certain radionuclides that are indicators of severity of reactor core damage (e.g., noble gases, isotopes of iodine and cesium, and nonvolatile isotopes), hydrogen in the containment atmosphere, and total dissolved gases or hydrogen, boron, and chloride in reactor coolant samples.

The staff reviewed the post-accident sampling system in accordance with SRP Section 9.3.2, the criteria in Item II.B.3 of NUREG-0737, and guidelines of Regulatory Guides 1.97, Revision 3.

The post-accident sampling system is designed to take samples from the reactor coolant loop, emergency water storage tank (containment sump), and containment atmosphere. The time required for taking and analyzing samples will be less than 3 hours after a decision is made to sample. Reactor coolant and containment atmosphere can be sampled following an accident without placing any isolated auxiliary system into service. The reactor coolant sample will be obtained from the hot-leg sampling lines. The emergency water storage tank sample will be taken from the residual heat removal system sampling lines. The post-accident containment atmosphere sampling system will consist of an on-line hydrogen monitor for continuous analysis of hydrogen concentration in the containment air.

Radiation exposure to the operators during post-accident sampling will be kept within acceptable limits by the use of remotely operated valves, ventilation, shielding, and dilution of samples. The ventilation exhaust of the sampling station will be routed to the plant heating, ventilation, and air conditioning system, which will include charcoal adsorbers and high-efficiency particulate air filters. In addition, it will be possible to return to the containment all excess samples and purge volumes from sample points inside the containment building.

The detailed sampling procedures, the procedure for estimating the degree of reactor core damage, the capability of measuring dissolved gases in reactor coolant samples, and the capability of chloride analysis will be developed during the FDA stage of review.

The staff determined that this information is insufficient to complete its review on the post-accident sampling system. Therefore, the staff concludes that the post-accident sampling system design will be addressed during the FDA stage of review. The applicant will be required to provide the sampling procedure, analytical instrumentation, analysis accuracy and ranges, sampling frequency, operator training, methodology for estimating the degree of reactor core damage, capability of measuring dissolved gases and chloride, and provisions to restrict background radiation level during the FDA stage of review.

9.3.3 Equipment and Floor Drainage System (RESAR SP/90 Module 13, Section 9.3.3)

The equipment and floor drainage system (EFDS) will collect, monitor, and direct liquid wastes generated within the plant to the proper area for processing or disposal.

The non-safety-related (Quality Group D, non-seismic Category I) portions of the EFDS will include all piping from equipment and floor drains to the sump and sump pumps, and piping necessary to carry nonradioactive and potentially radioactive effluents through separate subsystems. Potentially radioactive drainage is collected in floor and equipment drain sumps in each building and discharged to the waste processing system, thus satisfying the requirements of GDC 60.

Safety-related portions of the system will include the containment penetration lines and valves for the containment floor and equipment sump and incore instrumentation sump. These portions are seismic Category I, Quality Group B. All this equipment will be located in seismic Category I, flood- and tornado-protected structures. Thus satisfying the requirements of GDC 2.

The capability to isolate components or piping so that the EFDS functions are not compromised is included in the design bases. This includes isolation of components to deal with leakage or malfunctions. Drainage from safety-related equipment rooms is designed to prevent flooding via drain piping backflow. Instrumentation will be provided to detect leakage from safety-related systems and water accumulation that could affect the operation of safety-related equipment. These features satisfy the requirements of GDC 4.

The staff concludes that the design meets the requirements of GDC 2 and 4 with regard to protection against natural phenomena and environmental effects (flooding) and the requirements of GDC 60 with regard to protection against releases of radioactive material to the environment and that the design is acceptable.

9.3.4 Chemical and Volume Control System (RESAR SP/90 Module 13, Section 9.3.4)

The staff reviewed the chemical and volume control system (CVCS) in accordance with SRP Section 9.3.4.

The CVCS is designed to maintain water inventory in the reactor coolant system, to maintain seal water injection flow to the reactor coolant pumps during normal operation and abnormal events, and to control reactor coolant water chemistry conditions, radioactivity level, and soluble neutron absorber (boron) concentration. The CVCS will not provide any accident mitigation function except for the three isolation functions, namely, isolation of the CVCS lines penetrating the containment building, isolation of inadvertent CVCS boron dilution, and isolation of the CVCS lines connected to the reactor coolant system. As indicated in Section 3.2.2 of this SER, the staff will request the applicant to provide the basis for classifying the CVCS seal injection as non-safety related and consequently not conforming to 10 CFR 50, Appendix B.

Two centrifugal charging pumps will supply borated water to the reactor coolant system and reactor coolant pump seals. The pump suction header will be supplied from the volume control tank, the reactor makeup water control, the spent fuel pit, or the emergency boration line. The volume control tank will provide surge capacity for that part of the reactor coolant not accommodated by the pressurizer following load transients. It also will provide a means to introduce hydrogen into the reactor coolant system to scavenge excess oxygen in the coolant. Gaseous fission products in the volume control tank that come from the letdown spray will be removed by stripping with a continuous purge of hydrogen.

Boron concentration changes in the reactor coolant during normal operations will be controlled through makeup and feed and bleed. Boric acid solution of 4 percent by weight will be prepared in the boric acid batching tank. Boric acid also will be supplied from the boron recycle evaporators. All equipment and piping containing the 4-percent boric acid solution will be located in heated rooms or will have recirculation and/or heat tracing to prevent precipation of boric acid.

The CVCS will purify the primary coolant by passing letdown flow through heat exchangers and purification ion exchangers. Mixed-bed demineralizers will be provided to remove ionic corrosion products, certain fission products, and suspended solids from the reactor coolant. Temperature indicators and isolation valves will be provided upstream of the resin beds. The letdown line will be isolated on high letdown temperature. Resin retention screens are designed to withstand differential pressures in excess of normal operational or transient pressure drops across the resin beds.

The staff determined that the proposed CVCS meets the requirements of

- GDC 1 and the guidelines of Regulatory Guide 1.26 by assigning quality group classifications to system components in accordance with the importance of the safety function to be performed

- GDC 2 and the guidelines of Regulatory Guide 1.29 by designing safety-related portions of the system to seismic Category I requirements

- GDC 14 by maintaining reactor coolant purity and material compatibility to reduce corrosion and thus reduce the probability of abnormal leakage, rapid propagating failure, or gross rupture of the reactor coolant pressure boundary

* GDC 29 as related to the reliability of the CVCS to provide negative reactivity to the reactor by supplying borated water to the reactor coolant system in the event of anticipated operational occurrences

* GDC 60 and 61 with regard to confining radioactivity by venting and collecting drainage from the CVCS components through closed systems

With the exception of the staff concern related to the quality group classification for CVCS seal injection (Section 3.2.2 of this SER), the staff concludes that the proposed design of the CVCS is acceptable.

9.4 Heating, Ventilation, and Air Conditioning (HVAC) Systems (RESAR SP/90 Module 13, Section 9.4)

9.4.1 Control Equipment Area Ventilation System (RESAR SP/90 Module 13, Section 9.4.1)

The control equipment area ventilation system will include the main control room ventilation system (MCRVS) and the essential switchgear ventilation system (ESVS). The MCRVS services the main control room and operator convenience facilities (e.g., operator waiting area, kitchen, toilet). The ESVS services all the essential electrical equipment rooms, including relay, battery, inverter, and switchgear rooms.

The MCRVS will include two redundant full-capacity equipment trains, each containing an air handling system (AHS) and an emergency circulation filter system (ECFS). Each AHS will consist of filters, essential chilled water cooling coils for heat removal, and fans for air circulation. The ECFS will consist of high-efficiency particulate air (HEPA) filters, charcoal filters, and fans for emergency air circulation. The MCRVS is designed as fully redundant except for some passive interconnecting duct headers. All essential components are designed as seismic Category I and each train is powered from independent Class 1E emergency power sources. Redundant components will be physically separated and protected from internally generated missiles, pipe break effects, and water spray.

During normal operation, a small portion of outside air will be mixed with return air from the control room, filtered, conditioned and returned to the control room. The outside intake air will be monitored for radiation, smoke, and chlorine by redundant monitors capable of isolating the intake and exhaust paths. The intake and exhaust structures will be protected against wind- and tornado-generated missiles. Because of its location, the system will be inherently protected against flood damage. Normally, the control room temperature will be maintained at approximately 74°F with a design-basis upper limit of 85°F for protection of electrical equipment. Instruments will be provided for fan operating status, filter pressure drop, control room temperature, chilled water flow, and loss of air flow to essential equipment.

In the event of an accident or detection of toxic gases, smoke, or radiation at the AHS air intake, the habitability zone will be isolated from the outside environment by two isolation dampers in series. The emergency circulation system will be automatically activated and the recycled air will pass through a duct-mounted humidity control coil, a recirculation filter bank, and an emergency circulation fan in addition to the normal operating condition filter/conditioning system. Actuation signals from the redundant detection systems will result in operation of all ventilation trains in the event of an accident. Once the initiating event is controlled and the actuation signal reset, unnecessary equipment can be shut down. The recirculation system design has provision for reopening of the intake isolation dampers if a fresh air supply is required.

The ESVS will consist of two 100-percent-capacity redundant trains. Each train of the ESVS will consist of an air-handling unit with medium-efficiency filters and cooling coils served from the essential chilled water system. Inlet air to the relay, switchgear, and inverter rooms will pass through duct-mounted humidity control coils. The battery room of each of the redundant trains will be fitted with two parallel full-capacity exhaust fans to prevent accumulation of toxic fumes or hydrogen. In a loss of coolant accident, only one train of electrical equipment will be required to achieve cold shutdown. All essential components are designed as seismic Category I and each train will be powered from independent Class 1E emergency power sources and served from independent chilled water systems. Intake and exhaust structures will be protected from tornado- or wind-generated missiles. System components will be protected from flood damage by their location in the auxiliary building. Redundant components will be physically separated and protected from internally generated missiles, pipe break effects, and water spray environment. Normally the electrical equipment areas will be maintained at 85°F with a design-basis maximum temperature of 104°F.

The design described above conforms to the requirements of GDC 4, 19, and 60 and the guidelines of Regulatory Guides 1.52, 1.78, and 1.140, "Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants," with regard to ensuring environmental limits for proper operation of plant controls and uninterrupted safe occupancy of the control room and associated areas required to be staffed under all normal and accident conditions, including LOCA conditions.

All essential components of the control room area ventilation system are designed as seismic Category I and are flood and tornado protected. Thus meeting the requirements of GDC 2 and guidelines of Regulatory Guide 1.29.

The staff concludes that the control equipment area ventilation system is in conformance with the requirements of GDC 2, 4, 19 and 60 with regard to protection against natural phenomena, maintaining proper environmental limits for equipment operation, protection to permit access for occupancy of the control room under normal and accident conditions, and capability to control radioactive release and is acceptable.

9.4.2 Reactor External Building Ventilation System (RESAR SP/90 Module 13, Section 9.4.2)

The reactor external building ventilation system is designed as a general supply and exhaust ventilation system that provides heat removal and air exchange for nonessential building areas. The ventilation system will be supplemented by individual cooling units and ventilation fans that serve essential mechanical equipment areas.

The reactor external building general ventilation system will consist of supply air-handling units, particulate exhaust filter units, fans, ductwork, and accessories to provide normal ventilation and building temperature control. Air will be supplied to corridors and exhausted from the individual equipment compartments. More air will be exhausted than supplied to maintain the reactor external building at a slight negative pressure. The system will use a once-through cycle for the ventilation air, will not be safety related, and will not perform functions essential to safe shutdown or post-accident operation. Possible releases from emergency core cooling system leakage and fuel handling accidents will be controlled by the annulus air cleanup and fuel building emergency system, which is described in Section 9.4.5 of this SER.

The essential mechanical equipment room cooling units will consist of a cooling coil with recirculation fan and dampers to remove heat generated within the space. Recirculation cooling units will be provided instead of a once-through ventilation system for the following areas because they may possibly become contaminated:

- safeguard component areas including high-head safety injection pump rooms, residual heat removal/containment spray pump rooms, and associated piping and valve galleries

- charging pump rooms including both centrifugal charging pump rooms and shared service to the positive-displacement charging pump rooms

- component cooling water (CCW) equipment areas including CCW pump rooms, CCW heat exchanger rooms, and essential chilled water system pump and chiller rooms

The essential mechanical equipment room ventilation units will consist of a once-through ventilation cycle using supply fans and exhaust fans to remove heat and maintain space temperature control. All rooms are designed to be clean areas, and the exhaust air will be vented directly to the environment with no filtration or monitoring requirements. Applicable areas are the motor-driven emergency feedwater pump rooms, turbine-driven emergency feedwater pump rooms, and main steam and feedwater piping rooms.

During normal operation of the general ventilation system, outside air will be supplied to the reactor external building by two 50-percent-capacity supply fans. The air is filtered and then conditioned as needed by the heating and cooling coils. The exhaust air will be processed through two 50-percent-capacity particulate filter systems and will be discharged to the unit vent by two of three 50-percent-capacity-exhaust fans. Supply and exhaust fans are electrically interlocked so that the building will remain under a slight negative pressure. In the event of a loss-of-coolant accident, the general ventilation equipment will continue to operate normally, assuming off-site power is still available.

Ducts to areas with essential cooling units will be isolated to enable proper operation of the emergency equipment.

The essential mechanical equipment room cooling and ventilation units will normally operate as required to maintain space temperatures below the design value. The cooling systems will operate based on heat load as indicated by room temperature. In the event of a LOCA, all units will be automatically started and will operate at full capacity throughout the event.

All essential components of the mechanical equipment cooling systems are designed as seismic Category I equipment and will remain functional after a design-basis earthquake. In addition, these subsystems will be housed in seismic Category 1 structures and will be protected from wind- or tornado-generated missiles and flood by virtue of location in the reactor external building. Redundant components will be physically separated and protected from internally generated missiles. Each train will be powered from independent Class 1E power sources. Fan operating status, filter differential pressure, and room temperatures will be monitored in the control room. These systems are designed to ensure that a single failure will not result in the loss of a safety-related function. Thus, the system meets the requirements of GDC 2 and 4 and the guidances of Regulatory Guides 1.26, 1.29, and 1.117.

The staff concludes that the reactor external building ventilation system meets the requirements of the applicable GDC and regulatory guides as described in SRP Sections 9.4.3 and 9.4.4, however, due to commitments the applicant has made for fire protection it is expected that this design will change significantly. The reactor external building ventilation system will be reviewed during the plant-specific licensing process.

9.4.3 Radwaste Building Ventilation System (RESAR SP/90 Module 13, Section 9.4.3)

The radwaste building ventilation system design is not provided in the Westinghouse nuclear power block but will be provided by the plant-specific license applicant. The staff will review this system against the applicable GDC during the plant-specific licensing process of an application referencing the RESAR SP/90 design.

9.4.4 Turbine Building Ventilation System (RESAR SP/90 Module 13, Section 9.4.4)

The turbine building ventilation system design is not provided in the Westinghouse NPB but will be provided by the plant-specific license applicant. The staff will review this system against the applicable GDC during the plant-specific licensing process of an application referencing the RESAR SP/90 design.

9.4.5 Annulus Air Cleanup and Fuel Building Emergency Exhaust System (RESAR SP/90 Module 13, Section 9.4.5)

The annulus air cleanup and fuel building emergency exhaust system is designed as a dual-purpose engineered safety feature that serves two functions: in a post-LOCA mode, the system will exhaust the annular secondary confinement volume including the safeguards component areas and, following a fuel handling accident, the system will exhaust the fuel handling area. In both operating modes the system reduces radiological consequences of the accident.

The system will include two fully redundant trains each comprised of a demister, an electrical heater for relative humidity control, HEPA filters, charcoal filter, and a fan. The system also will include sets of duplicate isolation dampers that allow the system to function in the required mode. In either of the accident modes, dampers will isolate the emergency exhaust system from the nonessential components of the reactor external building ventilation system. Under normal operating conditions, the emergency exhaust system will be in a standby mode. The annulus air cleanup mode will be initiated by a safety injection signal indicating a loss-of-coolant accident. The fuel handling building emergency exhaust mode will be initiated by high radiation levels in the fuel area indicating a fuel handling accident.

In the annulus air cleanup mode, air will be exhausted from the area through the filter bank to the unit vent. When the system pressure has decreased to at least 1/2 inch of water below atmospheric pressure, the system switches to a recirculation flow. In this mode, air drawn from the annulus will be passed through the annulus system filter banks and returned to the annulus. In the fuel handling emergency mode, the system will exhaust air from the fuel handling building through the system filter bank to the unit vent to provide a slight negative pressure relative to the surroundings. When this condition is obtained, the system will switch to a recirculation mode in which air drawn from the fuel handling building will be passed through the filters and returned to the building. In both modes, the negative pressure will be maintained by passing a controlled amount of filtered air to the unit vent. The unit vent will be fitted with appropriate monitors to ensure measurement of the radiation exiting the system.

The annulus air cleanup and fuel building emergency exhaust system is designed as an engineered safety feature in that all essential components will be seismic Category I and that the system will be protected from the effects of wind, tornado, and flood Redundant components will be provided to ensure availability of at least one train in the event of a loss of off-site power. The ductwork, fans, and filters are designed to appropriate standards and operation of the system, including system pressure, fan operation, and carbon bed heating, will be monitored from the control room.

The system described above is designed to withstand the effects of natural phenomena and is therefore consistent with the requirements of GDC 2 and the guidelines of Regulatory Guide 1.29. The system is designed as an engineered safety feature to ensure that it can accommodate normal operation and accident conditions, including a LOCA, and is therefore consistent with the requirements of GDC 4. The system will consist of two fully redundant trains to ensure function in the event of a single failure. The system will control releases of radioactive materials through a filtration and monitoring system that will consist of the components described in Regulatory Guide 1.52. The system also will be consistent with GDC 60 and 61 and is acceptable. Westinghouse indicates that system flow rate, based on radiological factors including filtration efficiency, off-site and on-site doses, will be provided during the FDA stage of review.

Subject to the successful resolution of the system flow, the staff concludes that the annulus air cleanup and fuel building emergency exhaust system is acceptable.

9.4.6 Containment Cooling and Ventilation System (RESAR SP/90 Module 13, Section 9.4.6)

The containment cooling and ventilation system is designed with five sub-systems that will maintain environmental conditions within the containment during normal operating conditions, extended shutdowns, and refueling outages. These subsystems are:

- the containment recirculation cooling system
- the control rod drive mechanism (CRDM) cooling system
- the containment purge system
- the containment air cleanup system
- the digital rod position indication (DRPI) room cooling system

In addition, the containment recirculation cooling system is designed as an engineered safety feature (ESF) and provides a safety-related heat removal function following a LOCA or steamline break accident. The safety-related function of this system is reviewed in Section 6.2.2 of this SER.

The containment recirculation cooling system will consist of four 33-percent-capacity recirculation cooling units designed to maintain containment air temperature between 90°F and 120°F during normal operation. Cool air will be fed to fans in the steam generator and pressurizer compartment, the reactor compartment, the heat exchanger compartment, and the containment dome area. Each of the four units will include a roughing filter, a chilled water cooler, and a recirculation fan. The CRDM system will consist of three 50-percent-capacity cooling units designed to maintain drive mechanism exit air temperature below 170°F. Each unit will include a roughing filter, chilled water cooling coil, fan, and shutoff dampers. The containment purge supply system will consist of two 50-percent-capacity trains that will supply air to the containment during extended shutdowns and refueling outages. The system is designed to maintain containment air temperature between 60°F and 90°F during these periods. Each inlet train will include a supply fan, heating and cooling units for temperature control, roughing and HEPA filters, and dampers for flow control. Each of the inlet and exhaust ducts that penetrate the containment will be fitted with isolation valves inside and outside of the containment. Each exhaust train will include a damper for flow control, roughing, charcoal and HEPA filters, and an exhaust fan. The vent exhaust will be monitored for radioactivity. The containment air cleanup system will consist of two filtration units and two fans that operate during normal operating conditions and extended outages to control fission products within the containment. Each filtration unit will include roughing, charcoal, and HEPA filters. When monitors or sampling indicate high airborne radioactivity levels, the operator will manually activate the units. The DRPI room cooling system will operate under normal conditions to provide cooling for the affected electronics. Instrumentation will be provided to monitor all fan status, filter pressure drop, cooling coil temperature, and containment temperature. Fans will be powered from a reliable non-Class-1E power source.

The design of the systems described above are compatible with the environmental conditions associated with normal operation and are consistent with the requirements of GDC 4 for normal conditions. They will be capable of monitoring for radiation within the containment and for isolation of the containment and their design is consistent with the requirements of GDC 54, 56, and 64. The purge system will provide exhaust gas filtering and is designed consistent with the requirements of GDC 60 for normal operating conditions.

The staff concludes that the containment cooling and ventilation system is acceptable.

9.4.7 Diesel Generator Building Ventilation System (RESAR SP/90 Module 13, Section 9.4.7)

The diesel generator building is designed to provide a suitable working environment under normal and accident conditions for the diesel generator and related equipment areas. The system will ventilate rooms containing the two diesel generator trains. Each room will be serviced by dual 50-percent-capacity inlet and exhaust fans and each inlet will be filtered. Isolation dampers will be provided on each intake and exhaust vent to protect against wind or tornado damage. The system will operate in a once-through mode removing heat generated by the diesels with the exhaust air. When the diesels are not operating, the ventilation system will be normally off, but it may be activated manually to provide cooling. The ventilation system fans will automatically activate in response to building temperature and are designed to maintain room temperature at 120°F or below with outside air temperature of 95°F.

All essential components of the ventilation system are designed as seismic Category I and will remain functional after a design-basis earthquake. Intake and exhaust vents will be protected against wind and tornado missiles and the system will be protected by virtue of location from flood damage. Consequently the system design is consistent with the requirements of GDC 2 and the guidelines of Regulatory Guide 1.29. No high or moderate energy piping will be located in the vicinity of the ventilation equipment and system components will be compatible with environmental conditions. Therefore the system design is consistent with the requirements of GDC 4. The system is designed to maintain function after a single failure and each ventilation train will be powered from an independent Class 1E electrical system. The system will be physically separated from sources of radiation and radioactive materials.

The staff concludes that the system is acceptable.

9.4.8 Essential Service Water Pump Structure Ventilation System (RESAR SP/90 Module 13, Section 9.4.8)

The essential service water pump structure ventilation system design is not provided in the Westinghouse NPB but will be provided by the plant-specific license applicant. The staff will review this system during the plant-specific licensing process of an application referencing the RESAR SP/90 design.

9.4.9 Plant Heating System (RESAR SP/90 Module 13, Section 9.4.9)

The plant heating system design is not provided in the Westinghouse NPB but will be provided by the plant-specific license applicant. The staff will review this system during the plant-specific licensing process of an application referencing the RESAR SP/90 design.

9.4.10 Essential Chilled Water System (RESAR SP/90 Module 13, Section 9.4.10)

The essential chilled water system will supply chilled water to safety-related air handling equipment in the charging pump rooms, the component cooling water pump rooms, the computer room, the switchgear room, and the main control room as well as to equipment that is not safety related throughout the unit. The chilled water system will consist of four sets of chillers, pumps, expansion tanks, and associated piping. The essential equipment served by the essential chilled water system will be separated into redundant trains that will be served by redundant trains of the chilled water system. Upon receipt of a safety injection signal, the isolation valves will separate the redundant trains and stop chilled water flow to nonessential equipment. The essential chilled water system is designed to seismic Category I criteria and will be located in the reactor external building where it will be inherently protected from the effects of wind, tornado, and flood. Each chiller and pump will be powered from independent Class 1E power sources and a single failure will not prevent the system from fulfilling its safety function. Chilled water pumps will be started from the control room and instruments will be provided to indicate low-water flow and high-water supply temperature. The system is designed to meet the requirements of GDC 44 with regard to redundancy and physical independence and GDC 45 and 46 with regard to inspectability and testability.

The staff concludes that the chilled water system design is consistent with the requirements of GDC 2 and the guidelines of Regulatory Guide 1.29 with regard to withstanding the effects of natural phenomena. The system is also designed to be compatible with environmental conditions associated with normal operation and accidents and consistent with the requirements of GDC 4. The system will have component redundancy, will not lose function on a single failure, and will have adequate electrical supply. The system will not be directly involved in the control of releases of radioactive material and therefore the requirements of GDC 60 and the guidelines of Regulatory Guides 1.52 and 1.140 are not applicable. The system is acceptable.

9.5 Other Auxiliary Systems (RESAR SP/90 Module 13, Section 9.5)

9.5.1 Fire Protection Program (RESAR SP/90 Module 13, Section 9.5.1)

Fire protection requirements for nuclear power plants are provided for in 10 CFR 50, Appendix A, Criterion 3, and 10 CFR 50.48.

Criterion 3 of Appendix A to 10 CFR 50 governed fire protection for nuclear power plants and was considered adequate until the Browns Ferry fire of March 22, 1975. This remains the most serious fire to date at a commercial domestic (U.S.) nuclear power plant. A committee was formed to investigate the fire and make recommendations based on their findings. Among the recommendations made by the investigation committee was that specific fire protection guidance should be developed that would supplement the general requirements contained in Criterion 3. That specific guidance was published in Branch Technical Position (BTP) APCSB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants," for new plants docketed after July 1, 1976, dated May 1976 (Revision of Section 9.5.1 of NUREG-75/087 dated May 1, 1976). Following publication of that detailed fire protection guidance, the staff developed Appendix A to BTP APCSB 9.5-1 (published in August 1976) to provide specific fire protection guidance for those plants docketed for construction permit before July 1, 1976. All licensees of operating plants and applicants of plants in various stages of design and construction were requested to review their plants against the guidance contained in Appendix A to BTP APCSB 9.5-1 and identify areas of compliance and noncompliance. For those identified items of noncompliance, each licensee and applicant was asked to propose modifications to achieve compliance or show why compliance was not required.

By mid-1979, most plants had complied with most of the provisions of Appendix A to BTP APCSB 9.5-1. However, 18 open issues existed in various combinations at a total of 33 operating plants. The staff then developed § 50.48 and Appendix R to 10 CFR 50 (published on November 19, 1980 and effective February 17, 1981) as a means of resolving the remaining 15 open issues (reduced from the original 18 open issues) at plants licensed to operate before January 1, 1979. In addition, three sections of Appendix R were considered by the Commission to be so important that those provisions were required for all plants even if the staff had previously approved the design in those areas. The three sections of Appendix R that applied to all plants were III.G (Fire protection of safe shutdown capability), III.J (Emergency lighting), and III.O (Oil collection system for reactor coolant pump). Following publication of § 50.48 and Appendix R to 10 CFR 50, BTP APCSB 9.5-1 was revised (July 1981 as part of NUREG-0800) to include provisions of Appendix R so as to give additional guidance to those applicants that had docketed their application for a construction permit before July 1, 1976, and that were still being completed and preparing for operating licenses.

It is important to note that this subsequent fire protection guidance for operating plants, as well as for plants still being constructed, is derived from and represents deviations from the original guidance (BTP APCSB 9.5-1, May 1, 1976) developed for new plants, the category of plants that the Westinghouse advanced pressurized-water reactor design represents. The intention has always been that when any advanced reactor design was proposed, fire protection would be provided on the basis of the best technology available, not on the basis of methods allowed for plants already operating or in advanced stages of design and construction.

On this basis, the fire protection system of RESAR SP/90 has been evaluated against the criteria of SRP Section 9.5.1 (BTP APCSB 9.5-1, May 1, 1976). Fire protection guidance applicable to advanced reactor design also is contained in supplemental guidance documents that have been issued from time to time. Two examples of such supplemental guidance are the information pertaining to safe shutdown methodology contained in Generic Letter 81-12, dated February 20, 1981, and some important technical information, such as conformance with National Fire Protection Association codes and standards contained in Generic Letter 86-10, dated April 24, 1986.

The issue of fire protection in advanced reactor design will continue to evolve to minimize fire as a significant contributor to the likelihood of severe accidents. The staff expects that current NRC guidance will be enhanced to address this issue, therefore, fire protection will receive an in-depth review during the FDA stage or during the plant-specific licensing process of an application referencing the RESAR SP/90 design based on the then current requirements.

9.5.1.1 General Evaluation Fire Protection Program (RESAR SP/90 Module 13, Sections 9.5.1.1, 9.5.1.2)

The applicant has generally followed the NRC's concept of defense in depth with regard to fire protection. The three steps of defense in depth and the applicant's implementation of these steps follow.

(1) To reduce the possibility of fire starting in the plant, the applicant used fire resistant and fire retardant materials in its design of RESAR SP/90 to minimize and isolate fire hazards. The applicant used low-voltage multiplexed circuits in its design to eliminate the need for cable spreading rooms and substantially reduce the amount of combustible cable insulation and higher voltage ignition sources in the control room.

(2) To promptly detect and suppress fire, the applicant has provided adequate automatic detection and a suitable mix of automatic and manual fire suppression capability in its design.

(3) To ensure that any fire that might occur will not prevent safe shutdown of the plant even if fire detection and suppression efforts should fail, the

applicant provided a fire protection program in its application. However, the staff does not agree in all cases with the effectiveness of the applicant's approach.

The fire protection program described by the applicant is intended to protect safe shutdown capability, prevent release of radioactive materials, minimize property damage, and protect personnel from injury as a result of fire.

In addition to the three aspects of defense in depth outlined above, the applicant also considered such features of general plant arrangement as

- access and egress routes
- equipment locations
- structural design features separating or isolating redundant safety-related systems
- floor drains
- ventilation
- construction materials

The applicant used applicable National Fire Protection Association codes and standards in its design and layout of the facility. However, the applicant must identify any deviations from these codes and standards and describe in the fire hazards analysis the deviations and measures taken to ensure that equivalent protection is provided. The staff will review the fire hazards analysis at the FDA stage or during the review of a plant-specific application referencing the RESAR SP/90 design.

9.5.1.2 Specific Features of Protection

9.5.1.2.1 Protection of Safe Shutdown Equipment (RESAR SP/90 Module 13, Section 9.5.1.3)

The applicant will use 3-hour-rated fire barriers to separate safe shutdown equipment from the remainder of the plant and from redundant systems and components outside of primary containment. Inside containment the applicant proposed using (1) a combination of structural components that do not fully enclose equipment and (2) separation by horizontal distance of more than 20 feet.

In Question 280.1 the staff stated that it does not accept any methods that rely on spatial separation, the use only of automatic detection and suppression, or separation by radiant energy shields for advanced reactors.

Westinghouse has modified Section 9.5.1.1.d of the RESAR SP/90 application in response to Question 280.1. The clarification concerning use of 3-hour-rated fire barriers exclusively for separation of safety-related equipment in the NPB areas outside containment is in accordance with the review criteria and is acceptable. The staff also recognizes the need for open communication between compartments inside containment in order to be able to relieve and equalize pressure following a high-energy line break. Therefore, the use of structural walls inside containment as fire barriers to separate safety-related systems (cabling, components and equipment), even though such walls may not fully enclose the equipment requiring separation, is acceptable in intent. However, care must be taken in actual system layout to ensure that line-of-sight exposure between components requiring separation does not exist and that a sufficient labyrinth exists between the separated components to ensure fire spread does not occur.

Separation only by horizontal distance of more than 20 feet with no intervening combustibles or fire hazards, even with fire detection and automatic fire suppression provided, is not considered acceptable protection for safe shutdown equipment. Since the containment is considered to be a single fire area the separation of redundant shutdown equipment, including associated cables, should be designed to ensure that one shutdown division will remain free of fire damage. Additionally, if credit is to be taken for components that are not safety related or systems in other fire areas that could be relied on to achieve safe shutdown following a fire affecting safety-related equipment, then that combination of components and systems should be explicitly shown. Any supporting analysis also should be provided to support use of components that are not safety related to achieve safe shutdown.

Westinghouse states in Section 9.5.1.1.e of the RESAR SP/90 application that it will use validated analytical techniques in its analysis to determine fire protection to be provided for safe shutdown components. Question 280.2 ended with a parenthetical note: "It is the staff's position that PRA [probabilistic risk assessment] cannot provide the basis for exemption to requirements." The modifications to Section 9.5.1.1.e are satisfactory to the extent that the applicant clearly does not anticipate the routine use of requests for deviation. The applicant's statement on page 24 of its May 1989 response to open items that "there is no intent on the part of Westinghouse to request exemption from fire protection requirements based on PRA" is acceptable.

The applicant has mentioned two areas outside containment and one inside containment that will not conform to the 3-hour-rated fire barrier separation criteria listed above. The three exceptions are discussed below.

(1) The main steam and feedwater tunnel was called out as an exception to separation of redundant safety-related components outside containment.

In March 1989 the staff posed the following questions to the applicant:

- Could fire render these valves inoperable, either open or closed?
- Could fire cause spurious action of these valves?
- If the answer to either or both of these questions is "yes," could the conditions cause or lead to unacceptable consequences vis-a-vis safe shutdown of the plant?

The applicant responded that the active valves, which will be located in the main steam tunnel (MST), will be normally deenergized and, therefore, fire could not cause spurious action of these valves. Fire could, however, render the valves inoperable if the control or power circuits to them were damaged. The applicant clearly states, in answer to the question, that "inability to operate valves in the MST does not inhibit the safe shutdown capability of the plant." The staff considers this response acceptable.

(2) The main control room (MCR) was called out as an exception to separation of redundant safety-related components outside containment.

In March 1989, the staff asked the applicant several specific questions. The applicant responded by providing additional descriptive material in May 1989 in its response to open items. However, the following questions remain to be answered:

- What type of automatically actuated fire extinguishing system would be provided for the MCR and what portion of the MCR would be protected by the system?

- Was operator action contemplated in the design of fire protection for the MCR?

These questions will need to be resolved during the FDA stage of review of the RESAR SP/90 application or during a plant-specific application referencing the RESAR SP/90 design.

(3) The pressurizer relief system was called out as an exception to separation of redundant safety-related components inside containment.

The staff considers the applicant's position that "RCS depressurization is achievable without reliance on pressurizer PORVs (i.e., using the safety grade hot leg letdown valves), although this would be a less desirable mode of operation due to the complex operator action required," unacceptable without further justification. In addition, this position has not been discussed in other pertinent parts of the RESAR SP/90 application; therefore, it cannot be accepted without additional information and clarification. This question will need to be resolved during the FDA stage of review of the RESAR SP/90 application or during a plant-specific application referencing the RESAR SP/90 design.

9.5.1.2.2 Passive Fire Protection Features (RESAR SP/90 Module 13, Section 9.5.1.4.1)

Passive fire protection features for the RESAR SP/90 design consist of building assemblies (such as walls, partitions, floor-ceiling assemblies, columns, beams, doors, and dampers) and insulating materials (such as cable wraps heat resistant coatings). Penetrations through the building assemblies such as doorways, hoistways, stairways, ventilation ducts, and cable trays and conduits are protected by appropriate fire rated doors, dampers, plugs, and seals. In its response to open items in May 1989, Westinghouse stated its intention to select, whenever possible, passive fire protection components of proven design, which have previously been tested and are listed by nationally recognized testing laboratories. In the unlikely event that existing designs cannot meet the functional requirements imposed by Westinghouse, new designs will be developed. Testing of such new designs will be performed by one of the nationally recognized laboratories in accordance with accepted standards in use at the time. This is acceptable to the staff.

The listing of fire barrier locations in Section 9.5.1.4.1.(a) of the RESAR SP/90 application appears satisfactory and is approved pending final design review to ensure it is comprehensive and all safe shutdown and safety-related equipment is properly isolated and segregated by fire rated barriers.

9.5.1.3 Fire Protection System Description (RESAR SP/90 Module 13, Section 9.5.1.4)

9.5.1.3.1 Fire Detection (RESAR SP/90 Module 13, Section 9.5.1.4.2.a)

Automatic fire detection systems, designed and installed in accordance with National Fire Protection Association Standards 72-D and 72-E, will be provided for all significant hazards and safe-shutdown components. Detection capability will be provided for major cable concentrations, safe-shutdown-related major pumps, switchgear, motor control centers, battery and inverter areas, relay rooms, fuel areas, and all other areas containing appreciable in situ or potentially transient combustibles. Detector devices will be selected on the basis of type of anticipated fire and located on the basis of ventilation, ceiling height, ambient conditions, and burning characteristics of the involved materials. Detection systems will alarm and be annunciated in the control room and will give a distinctive audible, and if necessary, visual local alarm.

The staff concludes that the automatic fire detection capability to be provided for RESAR SP/90 meets the guidelines of Section IV.C.1 of BTP 9.5-1 and is acceptable.

9.5.1.3.2 Fire Protection Water Supply System (RESAR SP/90 Module 13, Section 9.5.1.4.2.b)

A dedicated fire protection water supply and distribution system will be designed and installed in accordance with National Fire Protection Association Standards 13, 14, 15, 20, and 24 to meet the anticipated needs for fixed water suppression systems and manual hose stations.

The fire water supply (total capacity and flow rate) will be provided on the basis of the largest expected flow rate for a period of 2 hours, but not less than 300,000 gallons. This flow rate will be based on 1,000 gpm for manual hose streams plus the greater of all sprinkler heads opened and flowing in the largest designed fire area or the largest open head deluge system(s) operating.

Two 100-percent-capacity fire pumps, one electrically motor-driven and powered from onsite emergency power buses and the other diesel engine-driven, will be provided. Each pump will take suction from an ensured water source. When ground level suction tanks are used, a minimum of two 100-percent-capacity tanks will be provided and interconnected so that each pump can take suction from either or both tanks. In addition, connections will be so arranged that a leak in either tank or its piping will not cause the other tank to drain.

The two fire pumps will be located in separate fire areas cut off from each other and from the rest of the plant by 3-hour-rated fire barriers.

The fire-main loop in the yard will be designed and installed with sectional control valves that will deliver total fire flow to all automatic and manual fire suppression systems and manual hose stations even if the shortest portion of the water distribution piping is out of service.

On the basis of the applicant's commitment that the fire water supply and distribution system will conform to the applicable National Fire Protection Association Standards mentioned above, the staff concludes that the system meets the guidelines of Section IV.C.2 of BTP APCSB 9.5-I and is acceptable.

9.5.1.3.3 Water Fire Suppression Systems (RESAR SP/90 Module 13, Section 9.5.1.4.2.c)

Automatic water fire suppression systems will be installed over major fire hazards that will be identified by the fire hazards analysis. The systems will be designed and installed in accordance with National Fire Protection Association Standards 13 and 15.

The applicant states that water shields or buffers will be used, as necessary, to shield safety-related equipment that requires protection from water discharge.

The above are acceptable as general commitments. However, since the fire hazards analysis has not been completed and since no listing has been provided showing the areas, hazards, and safe-shutdown equipment to be protected by fixed water fire suppression systems, the staff cannot give blanket approval at this stage of the review for specific fire protection features that may or may not be provided. For instance, considering that this is an advanced reactor design, the necessity for shields to protect safety-related equipment from water spray should be an exceptional feature and not a widespread practice throughout the plant. The fire hazard analysis will be reviewed during the FDA stage of review or during the plant-specific licensing process of an application referencing the RESAR SP/90 design.

Standpipe and hose stations will be installed throughout the plant on the basis of needs identified in the plant-specific fire hazards analysis. The standpipe systems will be designed and installed in accordance with National Fire Protection Association Standard 14. Each hose station will be equipped with a maximum of 100 feet of 1½-inch hose and adjustable on/off spray nozzle that are listed or approved by a nationally recognized testing laboratory.

Pressure-reducing orifices will be installed at each hose station as required, to ensure that excessive pressures are not delivered to the nozzle.

Exterior hydrants and hose houses will be provided according to needs identified in the fire hazards analysis. They will be designed and equipped in accordance with National Fire Protection Association Standard 24.

Control and sectionalizing valves in the fire water system will be electrically supervised and will be indicated in the main control room.

Piping systems serving standpipes and hose connections for manual fire-fighting, in areas containing equipment required for safe plant shutdown in the event of a safe shutdown earthquake, will be analyzed for SSE loading, and will be provided with supports to ensure system pressure integrity during SSE.

RESAR SP/90 will not include floor penetrations that are susceptible to the potential of channeling water from fire extinguishing operations in one redundant fire area to an adjacent fire area. Floor penetrations will only be used for interconnections within one train of safe-shutdown equipment.

The above details concerning the fire protection water distribution and extinguishing systems conform to the guidelines contained in Sections IV.C.2 and 3 of BTP APCSB 9.5-1 and are acceptable, pending final staff approval of an acceptable plant-specific fire hazard analysis.

9.5.1.3.4 Gaseous Fire Suppression Systems (RESAR SP/90 Module 13, Section 9.5.1.4.2.d)

Fixed Halon 1301 or carbon dioxide fire extinguishing systems, designed and installed in accordance with National Fire Protection Association Standards 12 and 12A will be installed to protect selected hazards that may not practically or effectively be protected by water suppression systems. These gaseous fire suppression systems will be installed as required by the plant-specific fire hazards analysis.

As with the water fire suppression systems described above, since the fire hazards analysis has not been completed, and since no listing has been provided of the areas hazards safe-shutdown equipment to be protected by these gaseous fire suppression systems, the staff cannot give blanket approval at this stage of the review for specific features of protection that may or may not be provided. However, the details described are consistent with the guidelines provided in Section IV.C.4 and 5 of BTP APCSB 9.5-1 and are acceptable. The fire hazards analysis will be reviewed during the FDA stage of review or during the plant-specific licensing process of an application referencing the RESAR SP/90 design.

9.5.1.3.5 Fire Extinguishers (RESAR SP/90 Module 13, Section 9.5.1.4.2.f)

Portable fire extinguishers will be provided in areas with in situ or potentially transient combustibles. Extinguishers will be chosen on the basis of the anticipated type of fire in the area and the effect of the extinguishing agent on equipment in the area. Selection, installation, and maintenance of the portable extinguishers will be done in accordance with provisions of National Fire Protection Association Standard 10. This conforms to the guidelines of Section IV.C.6 of BTP APCSB 9.5-1 and is acceptable.

9.5.1.4 Fire Protection Support Systems (RESAR SP/90 Module 13, Section 9.5.1.4.3)

9.5.1.4.1 Emergency Communication and Lighting (RESAR SP/90 Module 13, Section 9.5.1.4.3.a, 9.5.1.4.3.b)

Portable radio communications will be provided for fire brigade and plant operations personnel during a fire incident. This communication system will have a distinct and separate frequency so that plant security force communications and actuation of protection relays will not be affected. The portable radio communication system will use fixed repeaters, as necessary, to ensure communications capability with any location in the station from the control room. The fixed repeaters will be arranged and protected so that exposure to fire damage will not disable the entire system.

Sealed-beam emergency lights with individual 8-hour battery supplies will be provided in areas that must be occupied for safe shutdown and in routes used for access and egress to these locations. The lighted areas will include areas where operator actions occur if the control room is evacuated. In addition to the sealed-beam 8-hour emergency lights, portable sealed-beam battery-powered hand-held lights will be provided for use by fire brigade and plant operations personnel during a fire incident.

These details conform to the guidelines of Section IV.B.5 of BTP APCSB 9.5-1 and are acceptable. (However, see Secti   9.5 .2.1 of this SER for limitations on allowed operator actions.)

9.5.1.4.2 Emergency Breathing Air (RESAR /90 Module 13, Section 9.5.1.4.3.c)

Emergency breathing air will be provided for fire brigade and control room personnel. The breathing air will be delivered by a self-contained apparatus or a storage reservoir. Full-face positive-pressure masks approved by the National Institute for Occupational Safety and Health will be used by all personnel required to use emergency breathing air.

A minimum of 10 self-contained breathing units will be provided for fire brigade use. Each unit will be provided with two extra air bottles located on site. Rated service life for the self-contained units will be a minimum of 1/2 hour. In addition to the two extra bottles for each self-contained unit, compressors will be provided so that exhausted air bottles may be quickly replenished. The compressors will operate in areas free of dust and contaminants and will be powered from a vital power bu so that breathing air is available if offsite power is lost.

These provisions for emergency breathing air conform to the guidelines contained in Section IV.B.4 of BTP APCSB 9.5-1 and are acceptable.

9.5.1.4.3 Curbs and Drains (RESAR SP/90 Module 13, Section 9.5.1.4.4)

Floor drains and curbs that are sized to remove expected fire fighting water flow will be provided in areas protected by fixed water fire suppression systems or hand-held hose lines if water accumulation will cause unacceptable damage to safety-related equipment. Water drained from areas that may contain radioactivity will be properly collected, analyzed, and treated before being discharged to the environment.

In areas pro' :cted by gas suppression systems, either the floor drains will be designed with adequate liquid seals or the suppression agent supply will be sized to compensate for agent loss through the drainage system.

Floor drains located in areas containing combustible liquids will be designed so that these liquids cannot flow back into safety-related areas through the drainage system.

These provisions for curbs and drains conform to the guidelines contained in Section IV.B.1 of BTP APCSB 9.5-1 and are acceptable.

9.5.1.4.4 Reactor Coolant Pump Oil Collection Systems (RESAR SP/90 Module 13, Section 9.5.1.4.5)

Seismically qualified oil collection systems will be provided for reactor coolant pump motor oil systems. These oil collection systems will protect all potential pressurized and nonpressurized leakage points in the pump lube oil system. A drain line will be provided to transport the largest potential oil leak to a vented closed container. The vent on the closed container will be provided with a flame arrestor if the oil flash point characteristics and proximity of hot surfaces present a flashback hazard.

The reactor coolant pump oil collection systems will be designed so that failure will not lead to a fire during normal or design-basis accident conditions. The oil collection systems also will be designed to withstand a safe shutdown earthquake. The oil collection container is designed to hold the entire lube oil inventory of the reactor coolant pump motor in accordance with 10 CFR 50, Appendix R, Section III.O, and is acceptable.

9.5.1.4.5 Smoke Control (RESAR SP/90 Module 13, Section 9.5.1.4.6)

Smoke will be removed from each fire area by ventilation systems. The removal of smoke by ventilation systems may be achieved either directly or via manual portable smoke venting devices. When using the manual smoke venting devices, smoke will be removed from the fire-involved room or area and placed in an area where exhaust ventilation will be provided. Release of smoke that may contain radioactive materials will be monitored to ensure compliance with applicable guidelines.

The general arrangement of the RESAR SP/90 design of safe-shutdown trains features a high degree of separation with no piping and few cabling interconnections. With such a physical arrangement, the ventilation system can become the most likely pathway for fire propagation and smoke dispersal. RESAR SP/90 will employ separate, dedicated ventilation systems for each of the two areas containing redundant trains of safe-shutdown equipment. This arrangement of the ventilation systems serving the areas containing safe-shutdown equipment will facilitate the venting of smoke originating in one area containing safe shutdown equipment and preclude spreading of this smoke to the redundant area containing safe-shutdown equipment and is acceptable.

9.5.1.4.6 Access/Egress Routes (RESAR SP/90 Module 13, Section 9.5.1.4.7)

Clearly marked fire exit routes will be provided for each fire area. These routes will be designed to comply with applicable life safety codes and standards. These provisions for access and egress routes conform to the guidelines contained in Section IV.B.4.(f) of BTP APCSB 9.5-1 and Section III.G of Appendix R to 10 CFR 50 and are acceptable.

9.5.1.4.7 Construction Materials and Combustible Contents (RESAR SP/90 Module 13, Section 9.5.1.4.8)

Noncombustible materials having radiant energy heat flux equal to or more than 50 kW/cm^2 will be used for interior wall and structural components, thermal insulation, radiation shielding, soundproofing, interior finishes, and suspended ceilings.

Transformers located inside fire areas containing safety-related equipment will be of the dry type, insulated with noncombustible liquid or separated from safety-related equipment by 3-hour-rated fire construction.

These provisions comply with the intent of the guidelines contained in Sections IV.B.1.(d) and (g) of BTP APCSB 9.5-1 to use only noncombustible materials for interior finish and are acceptable.

With regard to noncombustible liquid insulated transformers, care must be taken to ensure that the insulating liquid does not present any unacceptable health hazards to employees in the event of release of the material to the building environment. Consideration of this hazard should be included during the FDA stage of review or during the review of a Plant Specific Application referencing the RESAR SP/90 design.

9.5.1.4.8 Interaction With Other Systems (RESAR SP/90 Module 13, Section 9.5.1.5)

Potential for water spray from fire protection water suppression system piping will be reviewed based on the pipe rupture analysis criteria. Where potential for water spray may result in damage to safety-related equipment, one of the following options will be used to ensure that equipment is not damaged:

- rerouting of fire protection piping
- installation of water spray shields
- qualifying safety-related equipment to withstand effects of water spray

Pipe rupture criteria will be used to ensure that the flood inventory in fire protection piping will not cause damage to safety-related equipment. Drains and sumps in the NPB will be sized to control maximum flood inventory of fire protection piping.

These provisions comply with the guidelines of Section IV.C.3 and IV.B.1(i) of BTP APCSB 9.5-1 and are consistent with the provisions of water fire suppression systems described above in Section 9.5.1.3.3 and are acceptable. However, the same concern discussed in Section 9.5.1.3.3 relative to the case of water spray shields is of concern here. The fire hazards analysis will be reviewed during the FDA stage of review or during the plant-specific licensing process of an application referencing the RESAR SP/90 design to ensure this concern is addressed.

9.5.1.4.9 Preoperational Testing (RESAR SP/90 Module 13, Section 9.5.1.6)

All of the active components of the entire plant fire protection system(s) will pass a preoperational acceptance test in accordance with the appropriate National Fire Protection Association Standard governing design and installation of the system. Components and systems subject to passing the preoperational testing before being placed in service include

- fire pumps - controls, flow volume and pressure

- water distribution - flush and hydrostatic

- control valves

- fire detection and alarm systems including electronic supervision for other fire detection and fire suppression systems

- fire dampers

- water fire suppression systems

- gaseous fire suppression systems

- emergency radio communication systems

* emergency lights

* emergency breathing air systems and components

The above preoperational testing requirements are consistent with the guidance contained in BTP APCSB 9.5-1 and are acceptable.

9.5.1.5 Administrative Controls (RESAR SP/90 Module 13, Section 9.5.1.7)

The description of administrative controls that will be established to govern various details of operations of the plant conform to guidelines of BTP APCSB 9.5-1 and are acceptable. However, a detailed review and acceptance of the administrative controls will be performed during the FDA stage of review or during the plant-specific licensing process of an application referencing the RESAR SP/90 design. Items of interest under the administrative controls review will include

* control of combustible materials such as combustible/flammable liquids and gases, fire retardant treated wood, plastic materials, and dry ion exchange resins

* transient combustible materials and general housekeeping, including health physics materials

* open-flame and hot-work permits and cutting and welding operations

* quality assurance with respect to fire protection system(s) components, installation, maintenance, and operation

* qualification of fire protection engineering personnel, fire brigade members, and fire protection systems(s) maintenance and testing personnel

* instruction, training and drills provided to fire brigade members

9.5.1.6 Summary

Most items pertaining to the staff's review of Section 9.5.1 of the RESAR SP/90 application conform to the guidance contained in one or more of the applicable fire protection guidance documents and are acceptable. The staff review at the next licensing stage will be governed by the results of the fire hazards analysis and the fire protection requirements that are in effect at the time of that review.

9.5.2 Communication Systems (RESAR SP/90 Module 13, Section 9.5.2)

The RESAR SP/90 communication systems will provide reliable, effective intraplant and interplant (plant-to-offsite) communication under both normal plant operation and accident conditions.

9.5.2.1 Intraplant Systems (RESAR SP/90 Module 13, Section 9.5.2.2)

The intraplant communication systems will provide sufficient equipment of various types so that the specific RESAR SP/90 plant will have adequate communications to start up, continue safe operation, or safely shut down. The intraplant systems will include those items listed belo

(1) The public address/emergency evacuation alarm system will consist of handsets, amplifiers, loudspeakers, solid-state tone generator, and other associated equipment. The system will have one dedicated line for paging; it will be divided into two separate zones to minimize interference. Manually operated switches will be provided in each zone to allow plant-wide paging. A number of party lines also will be provided for two-way communication between all important plant areas. The system will be designed to ensure that the paging system output is audible over the expected noise levels under both normal and accident conditions. Headsets will be provided for use in high noise areas. An evacuation alarm will be generated by a solid-state tone generator over the paging system. The alarm will include rotating beam lights for visual indication in areas of high background noise. The power to the system will be from a source available during loss-of-site power.

(2) The private automatic branch exchange (PABX) system will provide uninterrupted private communications at all times between critical plant areas.

The PABX system will be powered from a source available during a loss-of-offsite power and designated telephone will be directly connected to the public telephone system.

(3) The specific RESAR SP/90 plant will have three unitized, sound-powered systems. A multichannel system will be provided between the control area and critical plant areas for startup and maintenance testing. In addition, there will be a two-channel, sound-powered system between the control room, fuel handling area, and the reactor operating floor for refueling operations and another two-channel, sound-powered system between the auxiliary shutdown panel and local control areas for maintaining cold shutdown following control room evacuation.

Conduit for all three sound-powered systems will be seismically installed in all safety-related areas with the emergency system conduits seismically installed in all areas. Heavy-duty, industrial-quality jacks and mounting boxes will be used throughout the plant. All locations served by the emergency system will be served by the maintenance system to provide redundancy.

All sound-powered systems will operate independently of external electrical power and will be available if offsite power is lost. Redundant components and cabling will be provided to prevent loss of service to critical areas as a result of a single failure.

9.5.2.2 Interplant (Plant-to-Offsite) Systems (RESAR SP/90 Module 13, Section 9.5.2.2)

The function of the interplant communication systems will be to provide dependable, reliable plant-to-offsite communications. The interplant communication systems will have the features listed below.

(1) An emergency notification system, with telephones located in the control room and technical support center, will be provided with a communication link to the staff. The system will be independent and will be connected directly to the local long-distance carrier.

(2) Two-way radio links through microwave and/or very-high-frequency radio between the site and local public safety agencies will be provided. This system will have a power source available if offsite power is lost.

(3) A power-line carrier microwave link will be provided with direct communication with the system dispatcher.

(4) The plant PABX telephone system will be connected to public telephone lines.

The communication systems for the RESAR SP/90 design will include all components for intraplant and interplant communications. The staff concludes that the communication systems for RESAR-SP/90, when implemented in accordance with the design criteria and bases contained in SRP Section 9.5.2, will satisfy the staff's requirements and are acceptable for PDA.

9.5.3 Lighting Systems (RESAR SP/90 Module 13, Section 9.5.3)

The lighting system design should provide adequate lighting in all areas of the station and should consist of normal and standby (essential) ac lighting systems, an emergency dc lighting system, and a battery pack standby lighting system. The lighting system is not safety related and therefore serves no safety function.

RESAR SP/90 plant lighting systems are designed to provide adequate lighting during normal plant operation and accident conditions, including the effects of a loss of offsite power. Adequate lighting systems will be provided in areas used during normal, shutdown, and emergency operations, including the appropriate access of exit routes. Lighting intensities are designed to the levels recommended by the Illuminating Engineering Society. The use of high-pressure sodium, fluorescent, and mercury vapor lamps is restricted; these lamps are not used in the following major areas:

- containment
- above the fuel transfer canal
- above the new and spent fuel storage areas
- the radwaste building (only mercury vapor is restricted)

Incandescent lighting is used in these areas except as noted in the radwaste building. The plant lighting systems consist of normal, essential, and emergency lighting.

9.5.3.1 Normal Lighting System (RESAR SP/90 Module 13, Section 9.5.3.2)

The normal lighting system will be supplied from 480/277-Vac lighting load centers and lighting panels fed from non-Class 1E motor control centers through 480-208Y/120-Vac dry-type transformers. Power source for these lighting systems will be taken from the normal non-Class 1E auxiliary power system.

9.5.3.2 Essential Lighting System (RESAR SP/90 Module 13, Section 9.5.3.2.2)

The essential lighting system will be used in conjunction with the normal lighting system especially in main walkways and stairs, Class 1E equipment rooms, switchgear rooms, and areas used for safe shutdown. The power for the essential lighting system will be supplied from non-Class 1E motor control centers backed by the emergency diesel generators.

9.5.3.3 Emergency Lighting System (RESAR SP/90 Module 13, Section 9.5.3.2.3)

The emergency lighting system will be powered from either a 250-Vac bus or from self-contained battery packs witn charger units. Emergency lighting will be provided for the main control board, remote shutdown panels, diesel generator panels, auxiliary feedwater and the access routes between the main control room and the shutdown panel rooms and all areas required for safe shutdown operation.

Those portions of the lighting system that service the main control room, shutdown panel rooms, diesel generator panels and auxiliary feedwater panels will be designed and constructed so that a safe shutdown earthquake (SSE) will not cause any structural failure that could reduce the function of any post-safe-shutdown earthquake item to an unacceptable level or could result in an incapacitating injury to occupants in these areas.

Westinghouse has committed to establish a failure mode and effects analysis for the emergency and essential lighting systems during the plant-specific design stage. The staff concludes the lighting systems are acceptable for the PDA.

9.5.4 Emergency Diesel Engine Auxiliary Support Systems

Each diesel engine will have a number of auxiliary systems, the design of which is the responsibility of the plant-specific applicant. However, Westinghouse has established certain safety design bases to ensure that the plant-specific design is compatible with the general design philosophy in the RESAR-SP/90 application. These include commitments to meet the requirements of the 10 CFR 50, Appendix A, GDC relevant to these systems, and some system-specific requirements based on regulatory policy.

This section applies to the general requirements of all the emergency diesel generator auxiliary support systems. The assessment of system-specific requirements is provided in the SER sections indicated below.

- Emergency Diesel Engine Fuel Oil Storage and Transfer System (9.5.4.1)
- Emergency Diesel Engine Cooling Water System (9.5.4.2)
- Emergency Diesel Engine Starting System (9.5.4.3)
- Emergency Diesel Engine Lubrication System (9.5.4.4)
- Emergency Diesel Engine Combustion Air Intake and Exhaust System (9.5.4.5)

The RESAR-SP/90 design provides for each of the diesel generator auxiliary systems to be associated separately and independently with each one of the diesel generators. The requirements, therefore, of GDC 17 and 44 as they relate to redundancy and single failure are met.

The design bases include protection of the systems from the effects of natural phenomena, the requirement for the systems to remain functional after an SSE, as well as postulated hazards such as internal missiles and pipe breaks, and the requirement that the systems are designed according to classifications indicated in Regulatory Guides 1.26 and 1.29. The requirements therefore of GDC 2 and 4 are met.

The active components of the systems will be capable of being tested during plant operation and will have provisions to allow for inservice inspection at appropriate times. Therefore, the requirements of GDC 45 and 46 are met.

The staff concludes that the design criteria included for the plant-specific design are appropriate and in accordance with the guidelines described in SRP Section 9.5.4 through 9.5.8, and, therefore, are acceptable. However, the staff will review the above cited systems in detail during the plant-specific licensing process of an application referencing the RESAR SP/90 design.

9.5.4.1 Emergency Diesel Engine Fuel Oil Storage and Transfer System (RESAR SP/90 Module 13, Section 9.5.4)

The basic function of this system will be to provide onsite storage and transfer of fuel oil to the diesel engines.

In addition to the general criteria described above, the following system-specific requirements are included in RESAR-SP/90: (1) capacity for providing onsite storage and delivery of fuel for at least 7 days of operation, (2) compliance with the guidelines of Regulatory Guide 1.137, "Diesel Generator Fuel Oil System," and (3) conformance to fire protection and separation criteria.

The staff concludes that the design criteria for the emergency diesel engine fuel oil storage and transfer system are acceptable.

9.5.4.2 Emergency Diesel Engine Cooling Water System (RESAR SP/90 Module 13, Section 9.5.5)

The basic function of this system is to provide cooling water to the emergency diesel engines. The system will be a closed cycle system, and will serve as an intermediate system between the diesel engines and the essential service water system.

In addition to the general criteria described in Section 9.5 4 above, the following system-specific requirements are included in RESAR-SP/90: (1) capacity to remove heat from the diesel engines at the maximum nameplate rating, and (2) ability to maintain the diesel engines in a hot standby condition to ensure quick starting of the engines.

The staff concludes that the design criteria for the emergency diesel engine cooling water system are acceptable.

9.5.4.3 Emergency Diesel Engine Starting System (RESAR SP/90 Module 13, Section 9.5.6)

The basic function of this system will be to provide a reliable method for starting the emergency diesels for all modes of operation. The system will consist of a safety-related portion downstream of and including the air start check valve as well as the remainder of the system, which is not safety related.

In addition to the general criteria described in Section 9.5.4 above, the following system-specific requirements are included in RESAR-SP/90: (1) capability of isolation of the safety-related portion of the system from the remainder when required, and (2) capacity to store sufficient air to allow for at least five consecutive crank cycles of approximately 3 seconds, or 2 or 3 revolutions of the diesel engine without external support or assistance.

The staff concludes that the design criteria for the emergency diesel engine starting system are acceptable.

9.5.4.4 Emergency Diesel Engine Lubrication System (RESAR SP/90 Module 13, Section 9.5.7)

The basic function of the system will be to provide essential lubrication and cooling for the components of the diesel engines.

In addition to the general criteria described in Section 9.5.4 above, the following system-specific requirements are included in RESAR-SP/90: (1) capability to provide adequate lubrication and cooling for at least 7 days of diesel engine operation, (2) capability to maintain the lubricating oil in a warm condition when the engine is on standby, and (3) conformance to fire protection and separation criteria.

The staff concludes that design criteria for the emergency diesel engine lubrication system are acceptable.

9.5.4.5 Emergency Diesel Engine Combustion Air Intake and Exhaust System (RESAR SP/90 Module 13, Section 9.5.8)

The basic function of the system will be to supply combustion air of suitable quality to the diesel engines and exhaust the combustion products from the diesel engines to the atmosphere.

In addition to the general criteria described in Section 9.5.4 above, the following system-specific requirements are included in RESAR-SP/90: (1) capability to enable the continuous operation of the diesel engines at nameplate rating and (2) capability to route exhaust effluents to not impact plant intake air supplies.

The staff concludes that the design criteria for the emergency diesel engine combustion air intake and exhaust system are acceptable.

10 STEAM AND POWER CONVERSION SYSTEM (RESAR SP/90 Modules 6 & 8, Section 10.0)

10.1 Summary Description (RESAR SP/90 Modules 6 & 8, Section 10.1)

The steam and power conversion system is designed to provide heat removal from the steam generators and to generate electricity. This system will consist of the turbine generator, the main steam supply system, the main condenser and associated subsystems, and the condensate and feedwater systems.

10.2 Turbine Generator (RESAR SP/90 Modules 6 & 8, Section 10.1.1.1)

The turbine generator (T-G) system will be designed to extract part of the thermal energy of high-pressure steam from the main steam system and convert it to electrical energy and to provide steam for running the steam generator feedwater pump turbines. The design of the T-G is the responsibility of the plant-specific applicant. However, the RESAR SP/90 application establishes interface requirements to ensure the compatibility between the nuclear power block (NPB) and the T-G system.

The interface requirements specify that the T-G should be designed (1) to be compatible with the NPB instrumentation and control systems; (2) to allow periodic testing of steam valves important to overspeed protection, emergency overspeed trip circuits, and other trip circuits under load; (3) to allow unlimited access to all levels of the turbine area under all operating conditions; and (4) to trip when a reactor trip signal, a manual trip signal from the control room, or a low condenser vacuum signal is received. In addition, all associated piping, valves, and controls should be located within the turbine building and no safety-related systems and components should be located within the turbine building.

The staff concludes that the design criteria provided in the RESAR SP/90 application are adequate to enable the plant-specific applicant to provide an acceptable T-G system. However, the staff will review the detailed T-G system design against the guidance described in SRP 10.2, "Turbine Generator," during the plant-specific licensing process referencing the RESAR SP/90 design.

10.2.1 Turbine Disk Integrity (RESAR SP/90 Modules 6 & 8, Section 10.2.3)

Since the T-G is not part of the NPB for the RESAR SP/90 design, the applicant has not performed an evaluation of the integrity of the turbine disk. The plant-specific applicant will be required to address this issue during its licensing process referencing the RESAR SP/90 design. At that time, the staff will review the probability of turbine disk failure during normal operation, including transients up to design overspeed, to ensure compliance with GDC 4.

10.3 Main Steam Supply System (RESAR SP/90 Modules 6 & 8, Section 10.1.1.2, 10.3)

The main steam supply system (MSSS) is designed to convey steam from the steam generators (SGs) to the T-G system and auxiliary systems for power generation. The review of the safety-related portion of the MSSS including the main steam isolation valves (MSIVs) is presented in Section 10.3.1 below. The portion of the MSSS that is not safety related, will consist of systems and components downstream of the MSIVs up to and including the turbine stop valves and is discussed in Section 10.3.2.

10.3.1 MSSS Up To and Including the MSIVs (RESAR SP/90 Modules 6 & 8, Section 10.3.2)

The portion of the MSSS that includes the main steam lines from the SGs through the steam tunnel to the turbine building side of the steam tunnel/turbine building wall is designed as the steam generator isolation system (SGIS) and is required to function following a DBA to achieve and maintain the plant in a safe shutdown condition. This system will consist of 4 main steam lines (1 for each SG), 4 power-operated relief valves (1 for each main steam line), 20 safety valves (5 for each main steam line), and 4 main steam isolation valves (1 for each main steam line). The SGIS will be located in the reactor external building, which is designed to withstand the effects of earthquakes, floods and tornadoes. The SGIS components are designed as Quality Group B and seismic Category I, thereby satisfying the requirements of GDC 2 and Regulatory Guides 1.26, and 1.29.

Main steam isolation capability will be provided by closure of the gate valve with a pneumatic/hydraulic operator in each steam line located outside of the containment. The MSIVs will automatically close upon receipt of a steam line isolation signal. The valves are designed to close in less than 5 seconds against the flow of a pipe break on either side of the valve. The redundant electrical solenoids will be powered from separate Class 1E sources. A steam line break upstream or downstream of the MSIVs, coupled with an MSIV failure to close, will not result in the blowdown of more than one SG. If a break occurred upstream of an MSIV and an MSIV of an unaffected SG failed to close, blowdown of the unaffected SG through the break would be prevented by closure of the MSIV associated with the affected SG. Blowdown through the turbine and condenser would be prevented by closure of the turbine stop valves and turbine bypass valves, which serve as backup for this type of accident in accordance with the guidelines of NUREG-0138, "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum from Director, NRR, to NRR Staff."

The power-operated atmospheric relief valves (POARVs) are designed as Quality Group B, seismic Category I, and powered by Class 1E dc sources. These valves will be installed to provide decay heat removal during normal or post-accident operation when the MSIVs are closed and the turbine bypass system is not available. Thus, the requirement of GDC 34 and the applicable guidelines of BTP RSB 5-1, "Design Requirements of the Residual Heat Removal System," are satisfied. The POARVs are designed normally closed and will fail closed (safe position). The valves will open and close on high steam line pressure (2-out-of-4 logic). POARVs are designed to be remotely operated from the main control room and at the shutdown control panel. The safety relief valves (SRVs) are designed as spring loaded, Quality Group B, and seismic Category I. The SRVs will discharge directly to the atmosphere through vent stacks.

The POARVs and SRVs will be located outside containment, upstream of the MSIVs, in the reactor external building. The MSIVs, POARVs, and SRVs will be subjected to a preservice testing program during initial startup and an inservice testing program during operation. MSIV closure time and the set pressure of each POARV will be verified.

The SGIS will be located in a structure that is protected from earthquakes, tornadoes, floods, and external missiles. The system components are designed to remain functional after an SSE and to function when called upon after postulated hazards of fires, internal missiles, or pipe breaks. Thus, it will satisfy the requirements of GDC 4, the guidelines of Regulatory Guide 1.117, and BTP ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment."

The staff concludes that the SGIS portion of the MSSS satisfies the requirements of GDC 2, 4, and 34 and the guidelines of Regulatory Guide 1.117 and BTPs ASB 3-1 and RSB 5-1 and is acceptable.

10.3.2 MSSS Portion Downstream of the MSIVs

This portion of the MSSS is not required to support safe shutdown of the reactor. The design of this portion of the MSSS is the responsibility of the plant-specific applicant. However, the RESAR SP/90 application provides system interface requirements that call for the design to be compatible with the SGIS system. The staff will review this portion of the MSSS against the applicable portion of the guidelines as described in SRP Section 10.3, "Main Steam Supply System," when it is available during the plant-specific licensing process referencing the RESAR SP/90 design.

10.3.3 Steam and Feedwater Systems Materials (RESAR SP/90 Modules 6 & 8, Section 10.3.6)

The staff will review the steam and feedwater systems materials when the applicant has submitted them with the detailed designs during the FDA stage.

10.3.4 Secondary Water Chemistry (RESAR SP/90 Modules 6 & 8, Section 10.3.5)

Chemistry control of the water on the shell side of the steam generators (secondary water chemistry) will be accomplished by chemical additives to minimize system corrosion, use of demineralizers and filters to remove chemical impurities in the condensate system, deaeration to minimize dissolved oxygen in the feedwater, and continuous blowdown of the steam generators to minimize concentration of impurities in them.

An all-volatile treatment will be used for chemical addition. Ammonia will be added to adjust the alkaline conditions and hydrazine will be added to scavenge dissolved oxygen in the feedwater. Both ammonia and hydrazine can be injected continuously at the discharge headers of the condensate pumps and will be added, as necessary, for chemistry control. The RESAR SP/90 application provides guidelines for the chemical impurity limits for water in (1) the condensate storage and emergency feedwater tanks; (2) the steam generator blowdown during cold shutdown/wet layup, heatup to less than 5 percent power, and power operation; and (3) the condensate and feedwater during power operation.

The plant-specific applicant will be responsible for secondary water chemistry control. The RESAR SP/90 application provides guidelines that were developed by the Electric Power Research Institute. The staff determined that the proposed guidelines for secondary water chemistry control and the all-volatile treatment of feedwater meet BTP MTEB 5-3 and the requirements of GDC 14 by controlling the water chemistry conditions on the steam generator secondary side to reduce corrosion, thus lowering the probability of abnormal leakage, rapidly propagating failure, and gross rupture of the reactor coolant pressure boundary.

The proposed secondary water chemistry control program is acceptable. However, the staff will review secondary water chemistry during the plant-specific licensing process referencing the RESAR SP/90 design.

10.4 Other Features of Steam and Power Conversion System (RESAR SP/90 Modules 6 & 8, Section 10.4)

10.4.1 Main Condenser (RESAR SP/90 Module 6 & 8, Section 10.4.1)

The main condenser is designed to function as the steam cycle heat sink. During normal operation, it will receive and condense steam from the turbine exhaust system, the turbine bypass system, and the steam generator feedwater pump turbine exhaust. It also will be a collection point for other steam cycle miscellaneous flows, drains, and vents. The main condenser is not designed to affect or support safe shutdown of the reactor or to serve any safety function in the operation of the reactor.

The design of the main condenser is the responsibility of the plant-specific applicant. However, the RESAR SP/90 application provides two system interface requirements: (1) the main condenser should be able to accommodate up to 40 percent of the main steam flow from the turbine bypass system and (2) the main condenser also should provide the surge volume required for the condensate and feedwater system.

The staff concludes that the applicant has provided adequate interface requirements for the plant-specific design of the main condenser. During the plant-specific licensing process referencing the RESAR SP/90 design, the staff will review the system to ensure compliance with the interfacing criteria and the guidance of SRP Section 10.4.1, "Main Condenser."

10.4.2 Main Condenser Evacuation System (RESAR SP/90 Modules 6 & 8, Section 10.4.2)

The main condenser evacuation system, also called the main condenser air removal system (MCARS), will be designed to remove air and other noncondensable gases from the main condenser during startup, normal operation, and cooldown. This system will not perform any safety function in the operation of the reactor.

The design of MCARS is the responsibility of the plant-specific applicant. However, the RESAR SP/90 application provides two system interface requirements: (1) a deaerating feedwater heater (DFH) should be provided for feedwater oxygen removal to keep the oxygen content from exceeding 5 parts per billion during normal operation and 2) the DFH should provide sufficient heating capacity to maintain a minimum feedwater temperature of 450°F under normal operating conditions.

The staff concludes that the applicant has provided the appropriate interfacing requirements to enable the plant-specific applicant to provide an acceptable main condenser evacuation system design. However, the staff will review this system against the guidance of SRP Section 10.4.2, "Main Condenser Evacuation System," during the plant-specific licensing process referencing the RESAR SP/90 design.

10.4.3 Turbine Gland Sealing System (RESAR SP/90 Modules 6 & 8, Section 10.4.3)

The turbine gland sealing system (TGSS) will be designed to prevent the escape of steam from the main generator and the main feedwater pump turbine shafts into the turbine building, as well as to prevent air inleakage into the turbines through the turbine glands. This system serves no safety function in the operation of the reactor.

The design of the TGSS is the responsibility of the plant-specific applicant. The staff will review this system against the guidance described in SRP Section 10.4.3, "Turbine Gland Sealing System," during the plant-specific licensing processing referencing the RESAR SP/90 design.

10.4.4 Turbine Bypass System (RESAR SP/90 Modules 6 & 8, Section 10.4.4)

The turbine bypass system (TBS), also called the steam dump system, will be designed to bypass main steam from the steam generators to the main condenser in a controlled manner to allow the RESAR SP/90 to withstand step load reductions in generator loads and to control reactor pressure during startup, hot shutdown, and reactor cooldown. This system will not serve any safety functions and will not be required for plant shutdown following an accident.

The design of the TBS is the responsibility of the plant-specific applicant; however, RESAR SP/90 application provides the following interface requirements for the system design:

- The TBS should accommodate up to 40 percent bypass of main steam flow to the main condenser.
- The TBS should provide bypass of main steam flow to the main condenser during plant startup, and should allow a normal manual cooldown of the reactor coolant system from a hot shutdown condition to a point where residual heat removal could be initiated.
- The TBS should withstand a 50-percent electrical step-load reduction in generator loads without tripping the reactor.
- The TBS should permit a turbine and reactor trip without lifting the main steam relief and safety valves.

The staff concludes that the applicant has provided the appropriate interfacing requirements for the plant-specific applicant to provide an acceptable design of the TBS. However, the staff will review this system against the guidance as described in SRP Section 10.4.4, "Turbine Bypass System," during the plant-specific licensing process referencing the RESAR SP/90 design.

10.4.5 Circulating Water System (RESAR SP/90 Modules 6 & 8, Section 10.4.5)

The circulating water system (CWS) will be designed to remove heat from the main condenser and to reject heat to the plant's ultimate heat sink. This system will not serve any safety functions during plant accidents and is not required to keep the reactor in a safe shutdown condition.

The design of this system is the responsibility of the plant-specific applicant and there are no interface requirements necessary between the RESAR SP/90 NPB and this system. The staff will review this system against the guidance described in SRP Section 10.4.5, "Circulating Water System," during the plant-specific licensing process referencing the RESAR SP/90 design.

10.4.6 Condensate Cleanup System (RESAR SP/90 Modules 6 & 8, Section 10.4.6)

The condensate cleanup system (CCS) will be designed to remove dissolved and suspended impurities from the condensate. The CCS will not be required for safe shutdown or mitigation of postulated accidents, but will be important in maintaining the secondary water quality.

The plant-specific applicant is responsible for the design of the CCS. The RESAR SP/90 application provides the interface criterion that the CCS is capable of maintaining the quality of the secondary water, including the CCS makeup water and the treated water from the steam generator blowdown processing system, in accordance with the proposed guidelines for secondary water chemistry control, discussed in Section 10.3.4 of this SER.

The staff concludes that the interface criterion is adequate to address the water quality in the CCS and is acceptable. However, the staff will review this system during the plant-specific licensing process referencing the RESAR SP/90 design.

10.4.7 Condensate and Feedwater System (RESAR SP/90 Modules 6 & 8, Section 10.4.7)

The condensate and feedwater system (CFS) is designed to provide feedwater from the condenser to the steam generators at a predetermined pressure and temperature. This system is divided into two portions.

The safety-related portion will include the main feed piping, valves, and associated instrumentation and controls upstream of the steam generator inlet nozzle to the turbine building side of the wall between the steam tunnel and the turbine building.

The portion that is not safety related will consist of the piping, valves, deaerating feedwater heater, pumps, and associated instrumentation and controls from the condenser hotwell up to and including the main feedwater control valves (located upstream of the turbine building/steam tunnel wall). The RESAR SP/90 application indicates that the design of the non-safety-related portion of the CFS, with the exception of the main feedwater control valves, is the responsibility of the plant-specific applicant. The staff will review the non-safety-related portion of this system against the guidances as described in SRP Section 10.4.7, "Condensate and Feedwater System," during the licensing process of the plant-specific application referencing the RESAR SP/90 design.

However, the staff notes that the RESAR SP/90 application addresses provisions for turbine-driven main feedwater pumps within the non-safety-related portion of the CFS. Recent NRR studies have shown that plant trips caused by balance-of-plant systems can be significantly reduced if motor-driven rather than turbine-driven main feedwater pumps are used. The plant-specific applicant should consider this during design of the non-safety-related portion of the CFS.

The safety-related portion of the CFS is designed to Quality Group B and seismic Category I requirements and will be located in the reactor external building.

This building will be protected from the effects of earthquakes, tornadoes, floods, external missiles, and other appropriate natural phenomena. Thus, the requirements of GDC 2 and the guidelines of Regulatory Guide 1.29 are met. This portion of the CFS is designed to remain functional after an SSE and to perform its design function following postulated hazards of internal missiles or pipe breaks. Thus, the requirements of GDC 4 are satisfied.

Each of the four main feedwater lines will be equipped with one main feedwater isolation valve (MFIV) located outside the containment and downstream of the feedwater control valve. The MFIVs will be automatically closed to prevent uncontrolled blowdown from more than one steam generator (SG) during the event of a feedwater line break inside the turbine building. The valves will be bi-directional gate valves with pneumatic/hydraulic operators and will close within 5 seconds upon receipt of a feedwater isolation signal or low-low SG level signal. Main feedwater control valves and associated bypass valves that are not safety related will provide redundant isolation to the MFIVs in the event of a secondary side pipe rupture inside the containment. Main feedwater check valves that will be located inside the containment upstream of the emergency feedwater connection also will serve as backup to the MFIVs in the event of a pipe rupture inside the containment.

During startup and shutdown operations, the non-safety-related startup feedwater system (SFWS) will be used to provide preheated feedwater to the SGs instead of the main feedwater system. The SFWS will be equipped with air-operated control valves that close during normal operation or whenever the emergency feedwater system is operating. This will isolate the SFWS flow. In the event of a failure of the main feedwater system, the MFIVs will automatically close to isolate the main feedwater system and to allow the addition of the SFWS flow to the SGs for reactor heat removal and cooldown. However, if both main feedwater and the startup feedwater systems are not available, the safety-related emergency feedwater system will be available for decay heat removal. (The emergency feedwater system is described in Section 10.4.9.) Thus, the RESAR SP/90 design meets the requirements of GDC 44.

The safety-related portion of the CFS will be located in an accessible area (reactor external building) and in the containment. The CFS will have a testing and inspection program that is designed to allow preservice valve testing, preoperational system testing, and inservice inspections during operation. Therefore, the requirements of GDC 45 and 46 are satisfied.

The CFS is designed with several features to prevent damaging flow instabilities (water hammer) in the main feedwater lines. These features include:

- The SG will be fitted with a top feedring with J-tubes.
- The main feedwater line will be connected to the SG by a downward-turned elbow.
- Cold water injection frequency from the emergency feedwater system will be reduced by increasing the use (manual or automatic) of preheated (greater than 250°F) feedwater from the SFWS.

In addition, the applicant has committed to provide a SG design that will conform to the guidelines of BTP ASB-10-2, "Design Guidelines for Avoiding Water Hammers in Steam Generators."

The staff concludes that the safety-related portion of the CFS satisfies the requirements of GDC 2, 4, 44, 45, and 46 and meets the guidelines of Regulatory Guide 1.29 and BTP ASB-10-2. This portion of the CFS is acceptable.

10.4.8 Steam Generator Blowdown Processing System (RESAR SP/90 Modules 6 & 8, Section 10.4.8)

The staff reviewed the design for the SG blowdown processing system in accordance with SRP Section 10.4.8.

The SG blowdown processing system will be used, in conjunction with the condensate demineralizer, chemical addition, and sampling systems, to control the quality of water on the shell side of the SG. Heat will be recovered from the blowdown and returned to the feedwater system. The blowdown then will be treated to remove impurities before being returned to the condenser hotwell or the condensate storage tank.

All the piping, valves, and controls from the SG blowdown nozzle through the steam tunnel and turbine building wall, including the containment isolation valves, are designed to nuclear safety Class 2, seismic Category I, and Quality Group B.

The equipment and components, which will be downstream of the containment isolation valves and which will process the SG blowdown, are not safety related, and the plant-specific applicant will be responsible for the design of the equipment and components. However, the RESAR SP/90 application provides the interface criteria for the portion of the blowdown processing system that is not safety related. These criteria include (1) conformance with the proposed guidelines for secondary water chemistry control (see Section 10.3.4 of this SER); (2) continuous flows up to 43,500 pounds per hour (nominally 90 gallons per minute) per SG; (3) fabrication codes consistent with Regulatory Position C.1.1 in Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," Revision 1; and (4) compatibility with the interface connections to the SG isolation system.

The design of the SG blowdown processing system meets the primary boundary material integrity requirements of GDC 14 as it relates to maintaining acceptable secondary water chemistry control during normal operation and anticipated operational occurrences to minimize potential corrosion of steam generator tubes and materials, thereby reducing the likelihood and magnitude of primary-to-secondary coolant leakage.

The safety-related portion of the system is designed seismic Category I and Quality Group B (from its connection to the steam generator inside primary containment up to and including the first isolation valve outside containment) in accordance with Regulatory Guides 1.26 and 1.29 because this portion of the SG blowdown processing system is considered an extension of the primary containment. Isolation valves downstream of the outer containment, which will not be safety related, meet the quality standards of Regulatory Position C.1.1 of Regulatory Guide 1.143. Thus, the system meets the quality standards requirements of GDC 1 and seismic requirements of GDC 2.

The staff also determined that the interface criteria are adequate to address the water quality in the SG blowdown.

The staff concludes that the SG blowdown processing system and the interface criteria meet GDC 1, 2, and 14 and the appropriate regulatory positions in Regulatory Guides 1.26, 1.29, and 1.143, and are acceptable. However, the staff will review this system during the plant-specific licensing process referencing the RESAR SP/90 design.

10.4.9 Emergency Feedwater System (RESAR SP/90 Modules 6 & 8, Section 10.4.9)

In a conventional Westinghouse PWR, the auxiliary feedwater system (AFWS) functions to provide a reliable source of water for the SGs during normal shutdowns and accidents. The AFWS functions to remove thermal energy from the reactor coolant system through the SG to the atmosphere. A typical AFWS provides emergency water following any accident. This system also may be used following a reactor shutdown, in conjunction with the condenser dump valves, to cool the reactor coolant system to hot shutdown temperatures, at which point the residual heat removal system will be brought into operation.

For the RESAR SP/90, the above safety and control functions are performed by two secondary side systems: the emergency feedwater system (EFWS), which is discussed below, and the startup feedwater system, which is discussed in Section 10.4.10 of this SER.

The EFWS is designed as a safety-related system to provide feedwater to the steam generators following transients or accidents, such as reactor trip, loss of main feed, steam or feedwater line breaks, SG tube ruptures, and at any time the main and startup feedwater systems are not available.

The EFWS will consist of tw independent, but identical, subsystems. Each will consist of one motor-driven and one turbine-driven feedwater pump, an emergency feedwater storage tank (EFWST), and associated piping, valves, and instrumentation and controls. Each subsystem will be powered by one of two separate safety Class 1E electrical supplies. The EFWS pumps are designed so that, following a feedwater line break event, two pumps can deliver the minimum emergency feedwater flow to at least two unaffected SGs within 1 minute after system actuation with the SG pressure at the lowest pressure set point of the safety relief valves.

During normal plant operation, the EFWS will not be in operation, but will remain in a standby mode ready to deliver emergency feedwater to the SGs. During plant startup and shutdown, the SFWS will normally be used to provide feedwater; the main feedwater system will be used during normal operation. Following most plant transients, such as a reactor trip or loss of main feedwater system, the nonessential SFWS is automatically started to control feedwater flow to the SGs. Only in the event of SFWS failure subsequent to a plant transient, or in a more severe accident, such as a feedwater line break, is the EFWS automatically actuated.

The EFWS is designed with adequate isolation from nonessential portions of the CFS. The non-safety-related SFWS will be located in the turbine building and, therefore, completely separate from the safety-related EFWS. Seismic Category I piping and manual valves will be provided for recirculation of emergency feedwater to the EFWST, and a safety grade cavitating venturi will be provided on the discharge line to each SG to limit loss of feedwater and prevent EFWS pump runout in the event of steamline or feedline breaks. Thus, EFWS functions will not be adversely affected by such accidents. The EFWS will be connected to the main feedwater lines downstream of the safety grade check valves and redundant safety grade air-operated main feedwater isolation valves. These valves will fail closed, close on self-contained high-pressure gas, and serve as a means to isolate the nonessential portions from the essential portions of the main feedwater system. Safety grade check valves also will be provided in the discharge line of each EFWS pump to prevent reverse flow through a failed EFWS pump and at the connection to each main feedwater line to prevent the high temperature and pressure main feedwater flow from discharging into the EFWS when it is not in operation. In addition, a temperature instrument will be provided in the discharge of each EFWS pump, just outside the containment, with readout in the control room to detect significant backleakage from the main feed lines and to alert the operators to possible steam binding of the EFWS pumps. Two normally open safety grade motor-operated gate valves will be provided in a crossover line between the two pumps in each EFWS train. These valves will allow flow from one EFWS pump to feed two SGs. In the event of a failed SG, these valves will close to separate the two pump lines, reducing the chance of not feeding an intact SG. These features provide adequate isolation for the EFWS to remain functional and not be impaired by the failure of nonessential components. Thus, the EFWS design satisfies the isolation requirements of GDC 2 and 44 and the guidelines of Position C.2 of Regulatory Guide 1.29.

The RESAR SP/90 application provides the framework for preoperational testing of the EFWS. Functional, structural, and leaktight integrity of the EFWS components will be demonstrated by periodic operation. The EFWS is designed with the capability for being tested during plant normal operation; the critical valves will be cycled, and the EFWS pumps will run with flow through a mini-flow line to the EFWST, with the lines to the SGs closed. Should it be required, the EFWS pumps also can be run at full flow by opening a bypass of the miniflow orifices. Thus, the functional and surveillance requirements of GDC 46 and the recommendations of NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants," January 1980, concerning generic improvements to the EFWS design, procedures, and reliability are satisfied.

The EFWS components will be located in the reactor external building, which is accessible during normal plant operation to allow periodic inservice inspection. Therefore, the EFWS design meets the requirements of GDC 45 and the recommendations of NUREG-0611 with regard to provisions for inservice inspection.

All essential portions of the EFWS are designed to Quality Group C and seismic Category I, with the exception of the piping and valves (including steam admission air-operated valves) from the SGs, which are designed to Quality Group B, seismic Category I. Thus, the design of the EFWS conforms to the requirements of GDC 2 and the recommendations of Regulatory Guide 1.29 with regard to its seismic classification.

The EFWS (including the EFWST) will be located in the safeguards area outside the reactor containment. The structure is designed to withstand the effects of applicable natural phenomena such as tornadoes, floods, earthquakes, and external missiles. Thus, the requirements of GDC 2 and 4 are satisfied.

The EFWS will remain functional after an SSE or function as intended following postulated hazards such as internal missiles or pipe break. In the event of a main steam line break in the turbine building as a result of a seismic event, the physical separation between the turbine building and the safeguards area will protect the EFWS from the effects of jet impingement or extreme environmental conditions. Thus, the EFWS design satisfies the requirements of GDC 4.

In the event of loss of offsite power (LOSP), the essential EFWS trains are designed to function automatically as required. The turbine-driven pumps will be energized by steam drawn from the main steam lines between the containment penetrations and the main steam isolation valves. Turbine and pump cooling oil will be cooled by emergency feedwater flow through integral heat exchangers. All valves and controls required for operation of the turbine-driven pumps will be independent of ac power and will be either fail-open air-operated valves (steam admission valves) or mechanical/hydraulic valves (steam control valves) that will be powered by a Class 1E power supply. The turbine-driven pumps in both EFWS trains will not be affected by a LOSP. Additionally, the motor-driven EFWS pumps will be powered by independent Class 1E sources. The system meets the requirements of GDC 34 and 44 and the recommendations of NUREG-0611 with regard to the ability of the EFWS to transfer decay heat from the reactor coolant system under a LOSP.

As noted above, the turbine-driven EFWS pumps and associated valves are designed to function independent of all offsite or onsite ac power sources. The pump discharge control and isolation valves will normally be full open. The air-operated steam admission valves will normally be closed, but are designed to fail open on loss of air or electric power. The turbine-driven pumps will be equipped with integral heat exchangers to cool turbine and pump bearing oil using emergency feedwater flow, thus ensuring proper cooling and functioning of the equipment during a loss of all ac power. In addition, the turbine-driven pump in each EFWS train will receive steam from separate SG steam lines (turbine driven pump 3 receives steam from SG C, pump 4 from SG D). Thus, the EFWS design satisfies the requirements of GDC 34 and 44, the guidelines of BTP ASB 10-1, and the recommendations of NUREG-0611 with regard to EFWS power diversity.

The EFWS is designed to function in the event of steamline or feedline rupture. The system will be equipped with four cavitating venturies, one on each discharge line to the SGs. During accidents, these cavitating venturies will choke flow to the SG to prevent pump damage caused by runout. In the event of steamline breaks, the cavitating venturies also will prevent excessive rates of cooldown of reactor coolant system components by choking the emergency feedwater flow to the SGs and, if the break is inside containment, the cavitating venturies will help limit the effect of EFWS on the mass and energy released to the containment. They also prevent overfilling the SGs immediately following an accident (before the operator is able to take control of the situation), and thus limit the chance of steamline flooding. Therefore, the EFWS design meets the requirements of GDC 34 and 44 with regard to its ability to transfer heat under accident conditions and provide isolation to ensure system function.

The EFWS is designed to provide sufficient feedwater to prevent damage to the reactor following a main feedline break inside the containment, or a main steamline break event, as well as to cool down the reactor to a temperature (350°F) below which the residual heat removal system can be operated. Each train of the EFWS will be provided with an EFWST that will be safety grade, seismically qualified, and protected from missiles and natural hazards. The tank will contain an adequate quantity of condensate quality water to enable a cold shutdown with extended hot standby and cooldown times (8 hours hot standby, 6 hours cooldown), and indefinite hot standby (1 day of water supply in the EFWSTs, following by alternate water supply within that timeframe or feed and bleed operations). The RESAR SP/90 application also provides adequate interfacing requirements for the alternate water supply to the EFWS. The EFWST will be equipped with redundant level indicators and alarms. Thus, the EFWS design meets the requirements of GDC 34 and 44 and the recommendations of NUREG-0611 with regard to the ability to perform decay heat removal functions.

The staff concludes that the EFWS meets the requirements of GDC 2, 4, 34, 44, 45, and 46 and the guidelines of Regulatory Guides 1.29, BTPs ASB 10-1 and RSB 5-1, and is acceptable.

10.4.10 Startup Feedwater System (RESAR SP/90 Modules 6 & 8, Section 10.4.10)

The startup feedwater system (SFWS) will be designed to feed the SGs preheated water during normal plant startup and shutdown operations. Under these operating conditions, the steam from the SGs will be sent directly to the main condenser. The SFWS also will be designed to automatically start following a plant transient such as a reactor trip or loss of main feedwater. The SFWS will be a nonessential control grade system, not required to perform mitigation actions following postulated accidents. However, it will provide additional diversity to the essential EFWS and also will serve to minimize the number of EFWS actuations, thus reducing the chance of EFWS failures.

The design of the SFWS is the responsibility of the plant-specific applicant. However, the RESAR SP/90 application provides several interfacing requirements for the SFWS including: the SFWS shall consist of one motor-driven feedwater pump with associated piping, valves, and instrumentation and controls necessary for operation; all SFWS components shall be located in the turbine building; a safety grade isolation valve, flowmeter, and flow control valve shall be provided; the design of the SFWS shall be compatible with the interfacing connections to the SGIS and the EFWS; and the SFWS shall be designed to remove sufficient core heat for all condition I and II events; the SFWS shall automatically start following a plant transient; and the SFWS shall be designed to deliver preheated water to the SGs, but also shall have the capability to take suction from the condensate storage tank, or from the main condenser hotwell. The staff concludes that the applicant has provided adequate interface information for the plant-specific applicant to design the SFWS. However, the staff will review this system during the plant-specific licensing process referencing the RESAR SP/90 design.

10.4.11 Secondary Liquid Waste System

The secondary liquid waste system will be designed to process potentially radioactive condensate demineralizer regeneration wastes and liquid waste found in the turbine building. The design of this system is the responsibility of the plant-specific applicant. The RESAR SP/90 application provides no interfacing requirements or safety design basis. The staff will review this secondary liquid waste system during the plant-specific licensing referencing the RESAR SP/90 design.

11 RADIOACTIVE WASTE MANAGEMENT (RESAR SP/90 MODULE 12, Section 11.0)

The staff evaluated the radioactive waste management systems for the RESAR SP/90 design in accordance with SRP Sections 11.1, 11.2, 11.3, 11.4, and 11.5. Conformance to the acceptance criteria will provide the bases for concluding whether the radioactive waste management systems meet the requirements of 10 CFR Parts 20 and 50.

The following acceptance criteria are dependent on plant-specific design information and site information and will be evaluated during the licensing process for plant-specific applications referencing the RESAR SP/90 design:

- 10 CFR 20.105 provides radiation protection standard permissible levels of radiation in unrestricted areas. 10 CFR 20.106 provides radiation protection standard limiting concentrations for radioactive materials in air and water effluent streams and other requirements concerning radioactivity in effluents to unrestricted areas.

- 10 CFR 20.1 provides that licensees make every reasonable effort to maintain releases of radioactive materials in effluents to unrestricted areas as low as reasonably achievable. 10 CFR 50.34a provides design objectives for equipment to control releases of radioactive materials in effluents. 10 CFR 50, Appendix I, provides numerical guidance on design objectives to meet the requirements that radioactive materials in effluents released to unrestricted areas be kept as low as reasonably achievable (ALARA).

11.1 Source Terms (RESAR SP/90 Module 12, Section 11.1)

Westinghouse calculated the RESAR SP/90 "normal" radiation sources based on the methodology presented in ANSI/ANS 18.1-1984 "American National Standard Radiation Source Term for Normal Operation of Light Water Reactors." The staff finds that this is consistent with the methodology of NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents From Pressurized Water Reactors," 1976, referred to in SRP Section 11.1 and is acceptable. The methodology also is consistent with NUREG-0017, Rev. 1, 1985, which is used by the staff in performing independent calculations to determine compliance with 10 CFR Parts 20 and 50 as they relate to permissible levels of radiation in unrestricted areas and radioactive materials in effluents to unrestricted areas.

11.2 Liquid Waste Management Systems (RESAR SP/90 Module 12, Section 11.2)

The liquid waste management systems will consist of process equipment and instrumentation necessary to collect, process, monitor, and recycle and/or dispose of radioactive liquid wastes. The liquid waste management systems are designed to collect and process wastes on the basis of the origin of the waste in the plant and the expected levels of radioactivity. Liquid waste generally will be processed on a batch basis to permit optimum control of releases. Before liquid waste is released, samples will be analyzed to determine the types and amounts of radioactive material present. On the basis of the results of the analysis, the waste will be recycled for eventual reuse in the plant, retained for further processing, or released to the environment under controlled conditions. A radiation monitor will be placed in the discharge line and will automatically terminate liquid waste discharges if radiation measurements exceed a predetermined level.

The liquid waste management systems will include four subsystems: (1) the boron recycle system (BRS), (2) the steam generator (SG) blowdown processing system, (3) the turbine building floor drain system, and (4) the liquid waste processing system (LWPS).

Water will be processed by the BRS using an evaporator to concentrate boric acid solution to the concentration required in the boric acid tanks. Both evaporator condensate and concentrates normally will be recycled; however, a fraction of the condensate will be discharged for the purpose of avoiding excessive buildup of tritium in the reactor coolant water.

The SG blowdown processing system will treat the blowdown from the SGs by cooling, filtering, and demineralizing it. Normally, it then will be returned to the condensers for reuse as condensate makeup, or it may be discharged to the environment.

The turbine building floor drain system is designed to collect the floor drains and sampling wastes in the turbine building and other miscellaneous drains. The LWPS will collect and process potentially radioactive wastes for recycling, solidification, or release to the environment. On the basis of laboratory analysis of fluid samples taken before discharge, these wastes either will be retained for further processing or released under controlled conditions through the cooling water system, which will dilute the discharge flow. The segregation of equipment drains and waste streams will prevent intermixing of liquid wastes and allow for the recycling of as much reactor grade water entering the system as possible. The LWPS will be divided into the reactor coolant drain tank (RCDT) subsystem, drain channel A, and drain channel B. The RCDT subsystem will collect nonaerated reactor grade effluent from sources inside the containment for recycling. Drain channel A will collect aerated reactor grade effluent that normally can be recycled. Drain channel B will process all effluent that is normally to be discharged to the environment and is not suitable for recycling. The LWPS also will be able to handle and store spent ion exchange resins.

The staff reviewed the liquid waste management systems against SRP Section 11.2, considering the capability of the proposed liquid waste management systems to meet station demands resulting from anticipated operational occurrences. The systems' capacities and design flexibilities are adequate to meet the anticipated needs of the RESAR SP/90 design.

The RESAR SP/90 application states that for the liquid waste management systems there are no deviations from the criteria of applicable regulatory guides, including Regulatory Guide 1.143, this is acceptable.

The system design for meeting Sections II.A and II.D of 10 CFR 50, Appendix I, as it relates to the numerical guides for dose design objectives and limiting conditions for operation to meet ALARA criterion, is site and plant dependent and will be evaluated during the plant-specific licensing process of an application referencing the RESAR SP/90 design. Cost-benefit analyses for the identification of items to be added to the system in accordance with the guidance of Regulatory Guide 1.110, "Cost Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors," are dependent on the completion of offsite dose calculations and will be performed during the plant-specific licensing process of an application referencing the RESAR SP/90 design.

11.3 Gaseous Waste Management Systems (RESAR SP/90 Module 12, Section 11.3)

The design of the gaseous waste management systems includes systems that treat the normal ventilation exhausts, the exhaust from the main condenser air ejectors and associated components, and the gaseous wastes associated with degassing primary coolant, purging the volume control tank, displacing cover gases, purging of equipment, sampling and gas analysis operations, and boron recycle process operations.

The design of normal ventilation systems that will incorporate filters and other features to reduce or limit the release of airborne radioactive material under normal conditions include those for the reactor external building ventilation system, the radwaste building ventilation system, and the turbine building ventilation system.

The exhaust side of the reactor external building ventilation system will consist of two 50-percent-capacity particulate filtration exhaust units and three 50-percent-capacity exhaust fans. The system is designed to operate under normal conditions and to continue operating during a loss-of-coolant accident.

The plant-specific applicant will be responsible for the design of the radwaste building ventilation system and the turbine building ventilation system.

The gaseous waste processing system (GWPS) will consist of a refrigerated waste gas dryer, two charcoal guard beds, six charcoal absorption tanks, a charcoal filter, a gas surge tank, compressors, and interconnecting piping, valves, and instrumentation. This system will collect the gases from degassing the reactor coolant, purging the volume control tank, displacing cover gases, sampling and gas analyses, and from the boron recycle process operation.

Those gaseous waste management systems that are in the containment, reactor external, and fuel handling buildings will be located in seismic Category I structures. All other structures are designed seismic Category II.

For the GWPS a sample will be taken of the gas to be discharged and analyzed. The inventory will be determined from the analysis and the pressure in t[illegible] tank. When the contents of the gas decay tank are released, the local manual [illegible] valve will be opened to the plant vent and the gas discharge valve will be opened. On a high-radiation signal in the vent, the valve will close.

The staff reviewed the gaseous waste management system against the acceptance criteria of SRP Section 11.3, considering the capability of the proposed gaseous radwaste treatment systems to meet the anticipated demands of the plant as a result of operational occurrences. The staff concludes that the system capacity and design flexibility are adequate to meet the anticipated needs of the station.

The system design for meeting 10 CFR 50, Appendix I, as it relates to the numerical guides for design objectives and limiting conditions for operation to meet the ALARA criterion, is site and plant dependent and will be evaluated during the plant-specific licensing process of an application referencing the RESAR SP/90 design. Cost-benefit analyses for the identification of items to be added to the system in accordance with the guidance of Regulatory Guide 1.110 are dependent on the completion of offsite dose calculations, which also will be evaluated during the plant-specific licensing process of an application referencing the RESAR SP/90 design.

The staff will verify compliance with BTP ETSB 11-5, which requires the analysis of the radiological consequences of a single failure of an active component in the waste gas system, for a plant-specific application referencing the RESAR SP/90 design since it involves the analysis of radiological consequences that are dependent on site-specific parameters.

Westinghouse has stated in the RESAR SP/90 application that the guidance of Regulatory Guide 1.143 and the requirements of 10 CFR 50, Appendix I, and 10 CFR 20, Appendix B, Table II, Column 1, with respect to the GWPS have been complied with. Westinghouse has further stated in a response to open items dated May 1989 that the gaseous waste processing system meets the requirements of GDC 60 and 61 and the requirements of 10 CFR 50.34a.

The staff has reviewed the provisions incorporated into the RESAR SP/90 design to control releases resulting from hydrogen explosions in the gaseous waste processing system and concludes that the measures proposed by Westinghouse are adequate to prevent the occurrence of an explosion.

The staff identified certain minor drawing discrepancies and inconsistencies between drawings and text during its review of the RESAR SP/90 application. A check valve shown on the GWPS piping and instrumentation diagram (P&ID) between the charcoal filter and the plant vent appears to indicate flow in the wrong direction. One of two pressure-indicating instruments described in the text as being located upstream of the recycle line compressors to protect the compressors from being operated with the suction line valve closed is not shown on the P&ID. Repeated references are made in the text to oxygen analyzers that are to monitor oxygen concentration on the upstream or suction side of the recycle line compressors and automatically stop the compressors if oxygen concentration is too high. However, on the P&ID the analyzers are shown on the discharge side of the compressors. These discrepancies and inconsistencies, though not significant enough to preclude a favorable evaluation at the preliminary stage, should be corrected before the FDA stage of review of the RESAR SP/90 design.

11.4 <u>Solid Waste Processing System</u> (RESAR SP/90 Module 12, Section 11.4)

The solid waste processing system is designed to collect, process, and package radioactive wastes generated as a result of normal plant operation, including anticipated operational occurrences. The packaged waste will be stored until it is shipped off site to a licensed burial site.

The plant-specific applicant is responsible for the design of the system. However, the RESAR SP/90 application provides design criteria for the plant-specific applicant to meet and states that the system will be designed and constructed in accordance with Regulatory Guide 1.143. The ALARA requirements of GDC 60 and

Regulatory Guide 8.8 will be met by incorporating design features including, but not limited to, remote system operation, remotely actuated flushing, and shielding of components containing radioactive materials.

In addition to the above design criteria provided in the RESAR SP/90 application, the solid waste processing system shall be capable of solidifying radioactive wet wastes to meet 10 CFR 61.56, "Waste Characteristics" and the guidence of the NRC "Technical Position on Waste Form". The plant-specific applicant shall provide a process control program (PCP) to (1) classify solidified waste in accordance with 10 CFR 61.55, "Waste Classification" and the guidence of the NRC "Technical Position on Waste Classification"; (2) establish a manifest tracking system in accordance with 10 CFR 20.311, "Transfer for Disposal and Manifests"; and (3) ensure the system design meets BTP ETSB 11-3; and (4) ensure that the processed wastes meet shipping and transportation requirements during transit and disposal site requirements when received at the disposal site. The staff will review this system against the guidances of SRP Section 11.4, "Solid Waste Processing Systems," during the plant-specific licensing process referencing the RESAR SP/90 design.

11.5 Process and Effluent Radiological Monitoring and Sampling Systems (RESAR SP/90 Module 12, Section 11.5)

The process and effluent radiological monitoring and sampling (PERMS) systems are designed to provide information concerning radioactivity levels in systems throughout the plant, indicate radioactive leakage between systems, monitor equipment and performance, and monitor and control radioactivity levels in plant discharges to the environs.

11.5.1 Effluent Monitoring Systems

Monitors on certain effluent release lines are designed to automatically terminate discharges should radiation levels exceed a predetermined value. Systems that are not amenable to continuous monitoring, or for which detailed isotopic analyses are required, will be periodically sampled and analyzed in the plant laboratory.

The staff has reviewed the RESAR SP/90 design for the general locations, type, and properties of effluent monitors. During this preliminary stage of design, Westinghouse has not provided detailed information that the staff normally reviews, including P&IDs, sampling and maintenance procedures, technical specifications, and other specific information regarding the effluent sampling systems. The staff will review this information when provided by the applicant during the FDA stage of review or during the plant-specific licensing process for the RESAR SP/90 design, as appropriate.

The applicant proposes to use the PERMS systems with digital microprocessors for each self-contained radiation monitor that has a detector and data processing module. The PERMS systems will collect and display all information available from the field-mounted detectors on a screen and hardcopy printer in the control room. The RESAR SP/90 design includes 31 process and effluent radiation monitors and provides information on the type of monitor, general location, expected radiation zone of the detector, major isotopes to be monitored, and detectable radiation ranges.

In addition, the applicant provided the following information concerning each proposed monitor: safety classification, seismic qualification, whether moving filter, pump, or automatic control will be used, and the type of power supply to be used. The conditions of service for the monitors also were specified, such as expected operating process fluid temperature and pressure and ambient conditions at the detectors and relative humidity. Airborne radiological sampling requirements such as sample description, required analysis, collection and sample analysis frequency, and description and purpose for each monitor. In addition to the radiation monitors for normal plant operation, post-accident radiation monitors are specified.

The staff concludes that the effluent radiological monitoring and sampling system is acceptable for preliminary design approval. The staff will review detailed information regarding the effluent radiological monitoring and sampling system when it is submitted by the applicant during the FDA stage of review.

11.5.2 Process Monitoring Systems

Most of the process systems involving gaseous streams for which in-process grab sampling provisons are specified in SRP Section 11.5 generally had no such provisions identified in the RESAR SP/90 application. Westinghouse has stated that these systems are the responsibility of the plant-specific applicant. This section discusses these and other systems for which the in-process monitoring and sampling provisions specified by SRP Section 11.5 are not identified.

The main condenser air removal system (MCARS) is one system for which neither the monitoring nor the sampling provisions required in the process stream are identified. Monitoring and grab sample capabilities; however, are provided in the discharge or effluent stream from the MCARS. The staff will review this information, when provided, during the plant-specific licensing process referencing the RESAR SP/90 design.

The primary means of quantitatively evaluating the isotopic activity levels in process and effluent paths is a program of sampling and laboratory measurements. The RESAR SP/90 application provides the details of the sampling system associated with the PERMS monitors. However, the detailed information on sampling frequencies, required analyses, instrument alarm/type set points, calibration, and sensitivities will be provided by the plant-specific applicant. The staff will review this detailed information during the plant-specific licensing process referencing the RESAR SP/90 design.

For several systems, the RESAR SP/90 application provides grab sample capabilities at monitors located in the effluent streams from the systems, but not in the process streams, as specified by SRP Section 11.5. These systems include the MCARS, as noted previously, the auxiliary building ventilation system, the fuel handling area ventilation system, the radwaste area ventilation system, the evaporator ventilation system, and the pressurizer and boron recycle ventilation system. The staff will review this information, when provided during the plant-specific licensing process.

No provisions are identified for sampling the process portion of the steam generator blowdown ventilation system. The staff will review this information, when provided during the plant-specific licensing process.

The design of the evaporator ventilation systems and the pressurizer and boron recycle ventilation systems is described in appropriate sections of the RESAR SP/90 application. However, the applicant has not identified any provisions for sampling the process portion of these ventilation systems. The staff will review this information, when provided during the plant-specific licensing process.

Finally, no monitoring or sampling provisions have been discussed for the turbine gland seal condenser ventilation system and the mechanical vacuum pump exhaust system. The staff will review these systems during the plant-specific licensing process.

The staff has not been able to conclude that the provisions for monitoring process streams during postulated accidents are in accordance with GDC 64 and Regulatory Guide 1.97. The staff will review process and effluent radiological monitoring and sampling systems when the information is provided during the plant-specific licensing process referencing the RESAR SP/90 design.

12 RADIATION PROTECTION (RESAR SP/90 Module 11, Section 12.0)

The staff evaluated the radiation protection program presented in the RESAR SP/90 application against the review guidelines and criteria set forth in SRP Section 12. The radiation protection measures for the RESAR SP/90 design are intended to ensure that internal and external radiation dose to plant personnel and contractors resulting from plant conditions, including anticipated operational occurrences, will be within applicable limits of 10 CFR 20, "Standard for Protection Against Radiation," and will be as low as is reasonably achievable (ALARA).

The radiation protection design and features described in the RESAR SP/90 application are consistent with the guidelines of Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable" (Rev. 3, June 1978). The staff review consisted of ensuring that the applicant had either committed to following the guidelines of the applicable regulatory guides and staff positions referenced in the appropriate SRP section or provided acceptable alternatives.

12.1 Ensuring That Occupational Radiation Doses Are As Low As Is Reasonably Achievable (RESAR SP/90 Module 11, Section 12.1)

The staff reviewed the design considerations contained in the RESAR SP/90 application against the criteria set forth in SRP Section 12.1.

12.1.1 Policy Considerations (RESAR SP/90 Module 11, Section 12.1.1)

This information will be supplied by the plant-specific applicant during its licensing process referencing the RESAR SP/90 design.

12.1.2 Design Considerations (RESAR SP/90 Module 11, Section 12.1.2)

The applicant's radiation protection design is intended to maintain individual doses and total person-rem doses to plant workers, including construction workers, ALARA and to maintain individual doses within the limits of 10 CFR 20.

The applicant, using feedback information obtained from operating plants and following the guidelines of Regulatory Guide 8.8, incorporated facility and equipment design improvements into the RESAR SP/90 to satisfy its objectives of the radiation protection design. These features include the following for the design of systems and components:

- increase in reliability and maintainability, thereby effectively reducing the maintenance requirements on radioactive components
- reduction of radiation fields to ensure that operation, maintenance, and inspection activities are performed in the minimum radiation field feasible
- reduction of time spent in radiation fields during operation, maintenance, and inspection

- accommodation of remote and semiremote operation, maintenance, and inspection procedures

Some of the design improvements Westinghouse incorporated to reduce operational radiation exposure include

- simpler/faster refueling operation via an integrated head package
- corrosion-resistant steam generators
- steam generator maintenance features
- 18- to 24-month fuel cycles
- increased filtered flow in the chemical and volume control system
- shielding/layout improvements
- simplification of major fluid systems (e.g., reduced valve count)

These design considerations conform to the guidelines of Regulatory Guide 8.8 and the SRP and are acceptable.

12.1.3 Operational Considerations (RESAR SP/90 Module 11, Section 12.1.3)

This information will be supplied by the plant-specific applicant referencing the RESAR SP/90 design.

12.2 Radiation Sources (RESAR SP/90 Module 11, Section 12.2; Module 12, Section 11.0)

The staff reviewed the contained sources and airborne radioactive material source term values provided in the RESAR SP/90 application against the criteria set forth in SRP Section 12.2. These source term values are used as the bases for dose assessment and for design of the shielding.

12.2.1 Contained and Airborne Sources (RESAR SP/90 Module 11, Section 12.2.1; Module 12, Section 11.0)

Inside the containment during power operation, the greatest potential for personnel dose results from nitrogen-16, noble gases, and neutrons. Outside the containment and after shutdown inside the containment, the primary sources of personnel exposure are fission products from defects of the fuel cladding and activation products such as activated corrosion. Most of the airborne radioactivity within the plant will result from equipment leakage. The fission product source term values are based on cladding defects in the fuel rods producing 0.25 percent of the core thermal power. The coolant and corrosion activation product source term values are based on operating experience and reactors with a similar design; allowances are included for the buildup of activated corrosion products. Westinghouse stated that it will provide the plant-specific applicant with numerous reactor radiation source values for the at-power condition; these will include:

- neutron particle fluxes at the inside surface of the primary shield concrete at the core midplane

- gamma-ray energy fluxes at the inside surface of the primary shield concrete at the core midplane

- gamma-ray dose rates at the inside surface of the primary shield concrete

- detailed angular distributions of radiation leakage (neutron and gamma ray) from the reactor pressure vessel for streaming analyses

Specific source information will be provided later by the plant-specific application referencing the RESAR SP/90 design.

The source term values presented in the RESAR SP/90 application are comparable to estimates provided by other applicants with reactors of similar design and are acceptable.

Westinghouse provided a basis for tabulation of maximum expected radioactive airborne concentrations in equipment cubicles, corridors, and operating areas. However, the actual required tabulation of airborne concentrations and the expected airborne radioactivity levels at various plant locations for normal operations and anticipated occurrences, as well as the description of the ventilation system design, will be provided by the plant-specific applicant.

12.3 Radiation Protection Design Features (RESAR SP/90 Module 11, Section 12.3)

The staff reviewed the facility design features contained in the RESAR SP/90 application against the criteria set forth in SRP Section 12.3.

12.3.1 Facility Design Features (RESAR SP/90 Module 11, Section 12.3.1)

The RESAR SP/90 application provides evidence that those functions that contribute to accumulated doses of radiation to workers have been considered in the plant design. Features have been included in the design to help maintain doses ALARA in the performance of those functions. These features will facilitate access to work areas, reduce or allow for the reduction of source intensity, reduce the time required in the radiation fields, and provide for portable shielding and remote-handling tools and are acceptable.

The RESAR SP/90 application developed five radiation zones as a basis for classifying occupancy and access restrictions on various areas within the plant design. On this basis, maximum design dose rates are established for each zone and used as input for shielding of the respective zones. The areas that will have to be occupied on a predictable basis during normal operations and anticipated occurrences are zoned so that exposures are below the limits of 10 CFR 20 and will be ALARA. The zoning system and access control features also meet the posting entry requirements of 10 CFR 20.203 or Standard Technical Specifications and are consistent with Regulatory Guide 8.8. The staff finds this acceptable.

Several features are included in the plant design to minimize the buildup of activated corrosion products, a major contribution to occupational doses. Examples include:

- Corrosion resistant steam generators with low cobalt tubing material will be used. The previous limit on the amount of cobalt in steam generator tubing was 0.1 weight percent; this has been substantially reduced in the RESAR SP/90 design.

- Increased filtered flow in the CVCS will be used.

* Major fluid systems will be simplified (e.g., reduced valve count reduces crud deposition).

* To minimize crud buildup in branch lines, piping connections to reactor coolant loops will be located on or above the horizontal centerline of the pipe wherever possible.

* Demineralizers and piping are designed with provisions for flushing.

* Horizontal and flat-bottom tanks are designed to slope downward to the tank drain to reduce buildup of crud.

The corrosion product control features described in the RESAR SP/90 application are consistent with the guidance of Regulatory Guide 8.8 (Rev. 3) and the SRP and are acceptable.

The design features incorporated in the RESAR SP/90 application for maintaining occupational radiation doses ALARA during plant operation and maintenance also will serve to maintain radiation doses ALARA during decommissioning operations and are, therefore, acceptable.

12.3.2 Shielding (RESAR SP/90 Module 11, Section 12.3.2)

The radiation shielding is designed (1) to provide protection against radiation for operating personnel inside and outside the plant during normal operation, including anticipated occurrences and during reactor accidents and (2) to meet the requirements of the radiation dose rate zone system discussed previously. The following are several of the shielding features incorporated into the RESAR SP/90 design:

* Access labyrinths will be provided for rooms housing equipment that contains high-radiation sources to preclude a direct radiation path from the equipment to accessible areas.

* Radioactive piping will be routed through high-radiation areas where practicable or in shielded pipe chases in low-radiation areas.

* Shielding will be provided for all equipment that is anticipated to be normally radioactive.

* Shielding discontinuities that are caused by shield plugs, concrete hatch covers, and shield doors to high-radiation areas, will be provided with offsets to reduce radiation streaming.

* Shield penetrations will have offsets to minimize radiation streaming between the source and accessible areas. If offsets are not practicable, penetrations will be located as far as possible above the floor elevation to reduce personnel exposure.

These shielding techniques are designed to maintain personnel radiation exposure ALARA. Therefore, the staff concludes that the shielding design objectives are acceptable.

RESAR SP/90 shielding-design methodology included the use of industry-accepted standard computer codes such as ANISN, DOT, SCAP and MORSE. The applicant also used shielding information from operating nuclear plants as input data for the shield design calculations. The staff concludes that the shielding-design methodology presented is acceptable.

Spent fuel is the primary source of radiation during refueling. Because of the high activity of the fission products contained in the spent fuel elements, extensive shielding will be provided for areas surrounding the refueling pool and the fuel transfer canal to ensure radiation levels remain below zone levels specified for adjacent areas. Water shielding will be provided over the spent fuel assemblies during fuel handling. The staff finds this acceptable.

With regard to NUREG-0737, "Clarification of TMI Action Plan Requirements," Item II.B.2, "Plant Shielding," Westinghouse stated that the required information will be provided by the plant-specific applicant. However, in its RESAR SP/90 application, Westinghouse states that the control room shielding combined with other engineered safety features is provided to limit radiation doses for the duration of a loss-of-coolant accident (LOCA) in accordance with 10 CFR 50, Appendix A, GDC 19.

12.3.3 Ventilation (RESAR SP/90 Module 11, Section 12.3.3)

The applicant gave a general description only of the ventilation system, with assurance that the maximum airborne radioactivity level for normal operation, including anticipated operational occurrences, will be within the limits of 10 CFR 20, Appendix B, Table I. The remaining required information and complete description of the ventilation system is to be provided by the plant-specific applicant.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation (RESAR SP/90 Module 11, Section 12.3.4)

12.3.4.1 Area Radiation Monitoring Instrumentation (RESAR SP/90 Module 11, Section 12.3.4.1)

The RESAR SP/90 area radiation monitoring system is designed to

- monitor the radiation levels in areas where radiation levels could become significant and where personnel may be present
- alarm when the radiation levels exceed preset levels to warn of excessive radiation levels
- provide a continuous record of radiation levels at key locations within the plant area

To meet these objectives, the RESAR SP/90 design provides 10 area monitors in those areas where personnel may be present and where radiation levels could become significant. The area radiation monitoring system will be equipped with local and remote audio and visual alarms and a facility for central recording.

The RESAR SP/90 design does not provide area radiation monitors around the fuel storage area to meet the requirements of 10 CFR 70.24 or to be consistent with the guidance of Regulatory Guide 8.12, "Criticality Accident Alarm Systems." In response to the staff's question, Westinghouse stated that a request for exemption from 10 CFR 70.24 is the responsibility of the plant-specific applicant. Westinghouse will supply the applicant with a criticality analysis report, which will provide justification for the exemption.

To meet the criteria of the TMI Action Plan Item II.F.1.3, Westinghouse has committed to providing for the installation of two high-range gamma monitors in the RESAR SP/90 design that measure up to 10^7 R/hr, as required in Table II.F.1-3 of NUREG-0737. Westinghouse provided a description of the location of the high-range monitors.

With the exception of the fuel storage area radiation monitors, the staff finds the area radiation monitoring instrumentation design acceptable.

12.3.4.2 Airborne Radioactivity Monitoring Instrumentation (RESAR SP/90 Module 12, Section 11.5)

The RESAR SP/90 airborne radioactivity monitoring system is designed to

- assist in maintaining occupational radiation exposure to airborne contaminants ALARA

- check on the integrity of systems containing radioactivity

- warn of an inadvertent release of airborne radioactivity to prevent overexposure of personnel

To meet these design objectives, the system will monitor the air within an enclosure by direct measurement of either the enclosure atmosphere or the exhaust air from the enclosure, in compliance with 10 CFR 20 and Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection System." The system will indicate and record the levels of airborne radioactivity and actuate alarms if abnormal levels occur. Alarms will be provided to alert personnel that airborne radioactivity is at or above the selected set point level to ensure that personnel are not subjected to airborne radioactivity above the limits in 10 CFR 20.

The RESAR SP/90 application describes the process and effluent radiation monitoring systems (PERMS) that monitor radiation levels in selected plant process systems and in plant effluents. The airborne radioactivity monitoring instrumentation capable of detecting 10 MPC-hours (maximum permissible concentration) of particulate and iodine radioactivity from any compartment that has a possibility of contamination and that may be occupied by personnel (as discussed in the SRP) was not described in RESAR SP/90. This item should be provided by the plant-specific applicant.

12.3.5 Conclusion

With the exceptions noted above and the limited scope discussed in the report, the staff concludes that the radiation protection design features as described in the RESAR SP/90 application are consistent with ensuring that radiation exposures are ALARA and are acceptable.

12.4 <u>Dose Assessment</u> (RESAR SP/90 Module 11, Section 12.4)

The staff has reviewed the dose assessment for the RESAR SP/90 design against the criteria set forth in SRP Section 12.3.

The RESAR SP/90 application included an assessment of the doses that will be received by plant and contractor personnel. This dose assessment is based on occupancy factors, expected dose rates, expected airborne radioactivity concentrations, and historical information from operating pressurized-water-reactor (PWR) power plants. The dose assessment includes a breakdown of the annual person-rem doses associated with major functions such as routine operations, routine maintenance, inservice inspections, special maintenance, radwaste processing sing, refueling, and health physics. The applicant estimated the total annual collective dose to plant personnel and contractors to be less then 350 person-rems. Westinghouse stated that various design improvements have been incorporated into the RESAR SP/90 plant design and, in addition to these improvements, additional ALARA features will be considered for the final design. With these additional plant features, Westinghouse believes that approximately 215 person-rem per year is a realistic design goal for the annual cumulative plant exposure.

Westinghouse provided a partial description of the improvements that are expected to reduce the occupational radiation exposures. However, the assessment did not assign a numerical value to person-rems saved from each improvement. Historical data were used from other operating PWR facilities to estimate the dose for inservice inspection, special maintenance, radwaste processing, refueling, routine maintenance, reactor operations, and surveillance to identify areas where improvements can be made. Although the person-rem estimate was not made in complete accordance with Regulatory Guide 8.19, "Occupational Dose Assessment in Light-Water Reactor Power Plants Design Stage Man-Rem Estimates," the staff concludes that Westinghouse's dose assessment is acceptable because it meets the intent of the regulatory guide in that it identifies potential unnecessary exposures, minimizes foreseen doses, and leads to an examination of dose-reducing methods and techniques.

Currently, operating PWRs average 336 person-rems per unit annually (based on 1988 data), with particular plants experiencing an average annual dose as high as 1160 person-rems over the plant lifetime. The dose average is based on widely varying yearly doses for PWRs. The staff finds the bases for the RESAR SP/90 exposure estimate acceptable.

12.5 Operational Radiation Protection Program

The staff has not audited the organization, equipment, instrumentation, facilities, and procedures for radiation protection as required by the SRP Section 12.5. Since this section applies entirely to plant operation, Westinghouse stated in its RESAR SP/90 application that the required information will be provided by the plant-specific applicant during the licensing process referencing the RESAR SP/90 design.

13 CONDUCT OF OPERATIONS (RESAR SP/90 Module 13, Section 13.0)

The issues related to conduct of operations will be addressed on plant-specific basis at the appropriate time during the licensing process of an applicant referencing the RESAR SP/90 design.

Appendix O to 10 CFR 50 does not require applicants requesting approval of a standard nuclear reactor power plant design to submit information demonstrating compliance with the requirements of 10 CFR 73. Nevertheless, the staff performed a review that focused on the potential of the design to facilitate demonstration of compliance with existing requirements for protection against radiological sabotage. This included review pertinent to the requirements of 10 CFR 73.55 that could be standard for all utilities referencing this design application and to standard interface requirements for plant-specific items not otherwise within the nuclear power block (NPB) scope of the safety analysis report.

In its review of safeguards and security, the staff placed special attention on how the RESAR SP/90 design would foster the objectives of the Commission's Severe Accident Policy, which states:

> The Commission also recognizes the importance of such potential contributors to severe accident risk as human performance and sabotage. The issues of both insider and outsider sabotage threats will be carefully analyzed and, to the extent practicable, will be emphasized in the design and in the operating procedures developed for new plants.

Generic Issue A-29, "Nuclear Power Plant Design for the Reduction of Vulnerability to Sabotage," is one of the medium-prority generic safety issues for which this policy expects new designs to demonstrate technical resolution.

The staff reviewed relevant sections of the RESAR SP/90 application, especially Modules 2, 3, 8, and 13, and the applicant's responses to requests for additional information that are contained in Amendment 1 to Module 2 dated October 8, 1987.

The staff requested Westinghouse to identify the systems and components within its scope (i.e., NPB), including piping runs and valve motor control centers, that would be considered vital in the sense of 10 CFR 73.2(i). The applicant identified these vital areas as well as the major systems beyond the scope of the NPB that will be outside of containment and that plant-specific applicants also will need to protect as vital equipment. The equipment identified as vital equipment appears to be consistent with Review Guideline 17, "Definition of Vital Areas" contained in Appendix C to NUREG-0416, "Security Plan Evaluation Report," Revision 1, 1979.

The regulatory position in Regulatory Guide 5.65, "Vital Area Access Controls, Protection of Physical Security Equipment, and Key and Lock Controls," September 1986, will need to be considered during the final design approval (FDA) stage of the RESAR SP/90 NPB ducting and ventilation systems, such as the ventilation for the essential mechanical equipment room and the air intake and exhaust ducts for the diesel generators.

Because the RESAR SP/90 design is a standard design application, Westinghouse did not provide a discussion of specific items normally evaluated by the staff in its physical security review. These items include

- the physical security organization
- protected area
- access requirements
- detection aids
- communications
- text and maintenance requirements
- response requirements
- employee screening program

These issues are not a factor during the licensure of the RESAR SP/90 as a standard design, but will have to be addressed during the licensing process of a plant-specific application referencing the RESAR SP/90 design. However, the applicant should consider certain items during the FDA stage, including those items discussed below.

(1) Protected Area

The applicant should evaluate the interface between the onsite electric power systems described in Module 9 of the RESAR SP/90 application and uninterruptible power needed by exterior lighting systems to ensure required illumination of the protected area. This will facilitate demonstration of compliance with 10 CFR 73.55(c)(5) by plant-specific applicants using this standard design.

(2) Access Requirements

The applicant should evaluate the environmental interface criteria for the NPB wiring that may be needed for electric locks and intrusion alarms required on doors into NPB vital areas to ensure that these locks will permit needed access under steam line break or other accident conditions. The applicant also should consider the possible need for interface criteria between the intrusion alarm system for vital areas and NPB sources of electromagnetic interference that could trigger false alarms in this alarm system. This will facilitate demonstration of compliance with 10 CFR 73.55(d)(7) and (8) by plant-specific applicants using this standard design.

(3) Detection Aids

The applicant should evaluate the interface between the onsite electric power systems described in Module 9 and alarm station power supplies to facilitate demonstration of compliance with 10 CFR 73.55(e) by plant-specific applicants using this standard design.

(4) Communications

The applicant should evaluate the interface between security communications systems and the communication systems described in Sections 9.5.1.4.3 and

9.5.2 of Module 13 of the RESAR SP/90 application. The applicant also should evaluate the interface between security communications systems and onsite electric power systems and between security communications and plant components that may be sensitive to radio frequency transmissions. This would help avoid potential electromagnetic interference problems and facilitate demonstration of compliance with 10 CFR 73.55(f) by plant-specific applicants referencing this standard design.

The RESAR SP/90 contains neither design criteria for protection against radiological sabotage nor an analysis of the sabotage vulnerabilities of the design. The Severe Accident Policy Statement specifies that the issues of insider and outsider sabotage threats will be carefully analyzed and, to the extent practicable, will be emphasized in the design and in the operating procedures developed for new plants. This policy calls for demonstration of technical resolution of Generic Safety Issue A-29, which deals with plant design changes that could enhance the inherent resistance of current nuclear power plants to radiological sabotage. The applicant should submit this material during the FDA stage of review.

Nevertheless, at this time, the design features of RESAR SP/90 appear to provide advantages in protection against sabotage threats from insiders and outsiders as compared to the current generation of light-water reactors. These features, which are discussed below, are in addition to the physical security program for protection against sabotage (including employee trustworthiness screening programs, access controls and logging of persons who have had access to vital areas, intrusion detection and assessment systems, and armed response capabilities) that will be the responsibility of individual plant-specific applicants.

The RESAR SP/90 design of the NPB layout provides for physical separation of Train A and Train B safety equipment. This would make it easier for a utility to impose work rule constraints and access controls to protect against a single insider damaging both trains of a safety system. However, this does not resolve Generic Safety Issue A-29 with regard to insider sabotage because there is no requirement for the utility to use such constraints. In addition, there is some concern that such constraints could have an adverse effect on plant and personnel safety, as well as causing a utility some operational penalties. Physical separation of trains also could benefit protection against attack from outsiders because it would force an attacking group either to take longer to disable both trains of the system than if both trains were accessible at the same location or to split the attack force into two weaker parts, while the site's defenders would only need to position themselves to protect one of the two trains.

Protection of vital equipment in the NPB against external adversaries also could be aided by the limited access available for entry to the NPB. It is possible, however, that this advantage will depend on where the security response personnel and equipment are stationed, which is the responsibility of the plant specific applicant. This protection also could work to the disadvantage of the defenders if the adversaries can arrive at and seize control of the access point before the defenders. An interface criterion on security force patrols or stations is necessary to ensure this advantage.

The NPB layout will separate safety and control systems, which could permit fewer persons to be authorized access to safety equipment without interfering with the access of other persons to control equipment. Assuming that there is some probability of one randomly selected individual who has been screened being motivated to perform an act of sabotage, the probability of insider sabotage or tampering is minimized by minimizing the number of such individuals who have access to the vital equipment.

Another design feature that may reduce the risk of radiological release resulting from an insider tampering with safety-related equipment is safety systems with few valves whose positions are not monitored in the control room (i.e., manual valves).

Inherent sabotage resistance, it appears, will benefit from greater redundancy and diversity. For example, sabotage of the decay heat removal capability to mitigate a reactor transient would require disablement of the startup feedwater system and two independent trains of the emergency feedwater system (EFWS) on the secondary side, as well as the feed and bleed capability that has been designed into the primary side.

Safety-related emergency water storage tanks and emergency feedwater storage tanks will be located within the reinforced concrete NPB buildings, which protects them from possible attack from outside the protected area of the site.

The plant is designed to withstand loss of all ac electric power for a minimum of 2 hours, which could help reduce the risk associated with a sabotage-induced station blackout.

Consideration should be given to additional design features that could be beneficial against insider and outsider sabotage. Examples of such features may vary from installing tamper-indication alarms on certain equipment (e.g., manual isolation valves on emergency feedwater system pumps) to adding an alternative shutdown system to be maintained by a dedicated alternative shutdown staff. Other examples that could be considered may be found in NUREG/CR-4462, "A Ranking of Sabotage/Tampering Avoidance Technology Alternatives," January 1986.

The staff concludes that RESAR SP/90 design safeguards against radiological sabotage are in a conditionally acceptable stage of development for preliminary design approval. The principal issues to resolve at the FDA stage involve design considerations given to the risk of insider sabotage and physical protection interface requirements for plant-specific applications. Completion of a security plan, a guard training and qualification plan, and a contingency plan, which will be safeguards information, will be required at the plant-specific stage of review.

The following open issues should be addressed during the FDA phase of review:

(1) In Sections 3.2, "H. External Events, Human Errors, and Sabotage," and 5.1, "29. Issue A-29: Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage," of Module 2, Westinghouse stated that it will perform a sabotage assessment for RESAR SP/90. In Amendment 1 to Module 2, Response 910.2 stated that there was no sabotage assessment included in the RESAR SP/90 application. While the Severe Accident Policy Statement asks that the issues of insider and outsider sabotage be carefully analyzed, it is acceptable that this be presented at the FDA stage.

(2) Responses 910.2 and 910.9 say that sabotage considerations in the FDA stage will be aimed at ensuring compliance with any NRC imposed requirements.

Although waiting for the FDA stage is acceptable, the Severe Accident Policy Statement asks that the applicant propose how the design will resolve generic issues. Section 5.1 of the RESAR SP/90 application should include a means to resolve Generic Issue A-29 that would apply to plant-specific applicants referencing the RESAR SP/90 standard design certification.

(3) Although the plant layout with its separation of trains would make it feasible to prevent the same insider from having authorized access to both trains of safety-related equipment, the staff recommends consideration be given during the FDA stage to additional design features that could be beneficial against insider sabotage. Examples of such features may vary from installing tamper-indication alarms on certain equipment (e.g., manual isolation valves on emergency feedwater system pumps) to adding an alternative shutdown system that could be maintained by a dedicated alternative shutdown staff. Other examples that could be considered can be found in NUREG/CR-4462.

(4) The regulatory position in Regulatory Guide 5.65 will need to be considered at the FDA stage of NPB ducting and ventilation systems. This would include, but not necessarily be limited to, the ventilation for the essential mechanical equipment room and the air intake and exhaust ducts for the diesel generator.

(5) Response 910.9 states that interface requirements for structures, systems, and components outside the NPB will be deferred to the FDA stage. These interface requirements should, as a minimum, include:

- designating as vital those non-NPB systems (identified in response 910.3) that are essential to the functioning of the NPB vital systems, and imposing (where practicable) the same train separation design criteria for these systems as used for NPB systems

- providing uninterruptible power supplies for security equipment, including exterior security lighting

- addressing security communication needs

- providing penetration resistance for the wall between the steam tunnel and the turbine building and any openings in it

- developing criteria to prevent electromagnetic interference between plant and security equipment

- developing environmental interface criteria under which vital area access controls may need to operate in the event of a steamline break or other accident

14 TESTING AND MAINTENANCE

The staff will base its certification of the RESAR SP/90 design, in part, on a probabilistic risk assessment (PRA). The validity of a PRA is highly dependent on the reliability of systems, structures, and components; therefore, the staff requires assurance that programs will be implemented to ensure that the reliability of those systems, structures, and components (assumed in analyses) will be maintained throughout plant life. The applicant will be required to provide a program to ensure design reliability as part of the FDA application.

15 TRANSIENT AND ACCIDENT ANALYSES (RESAR SP/90 Module 10, Section 15.0)

15.1 General Discussion (RESAR SP/90 Module 10, Section 15.0)

The staff reviewed the transient and accident analyses provided in the RESAR SP/90 application in accordance with SRP Section 15. The staff's evaluation of the design criteria of the facility for each of the areas reviewed for the RESAR SP/90 design is given below.

Westinghouse evaluated the ability of the RESAR SP/90 design to withstand anticipated operational occurrences and a broad spectrum of postulated accidents without undue hazard to the health and safety of the public. Westinghouse used the results of these analyses to show conformance with General Design Criteria (GDC) 10 and 15.

Westinghouse used the following classification to divide plant conditions into four categories in accordance with anticipated frequency of occurrence and potential radiological consequences to the public:

Category I - normal operation and operational transients
Category II - faults of moderate frequency
Category III - infrequent faults
Category IV - limiting faults

Category I events are those occurrences that are expected frequently or regularly in the course of normal plant operation, refueling, and maintenance. They must be considered from the point of view of affecting the consequences of faulted conditions. In this regard, analysis of each faulted condition should be based on a conservative set of initial conditions corresponding to adverse conditions that can occur during normal operation.

In Section 15.0.2 of the RESAR SP/90 application, Westinghouse indicates that operation with permissible deviations, such as power operation with a reactor coolant pump out of service, may occur during continued operation as permitted by the plant-specific technical specifications. Since there is no accident analyses performed for N-1 loop operation, the staff asked Westinghouse to provide clarification of this issue to indicate that plant operation with a reactor coolant pump out of service is not permitted. In its response to open items dated August 1989, Westinghouse states that RESAR SP/90 Section 15.0.2 will be revised to delete the reference to power operation with a reactor coolant pump out of service.

Category II events are faults of moderate frequency that result in a reactor trip with the plant being capable of returning to operation. These anticipated operational occurrences are considered to be moderate-frequency events and are expected to occur during the lifetime of the plant. Events of moderate frequency should not result in fuel rod failure or reactor coolant system or secondary system overpressurization. In its August 1989 response, Westinghouse states that the RESAR SP/90 design meets the requirements of GDC 10, 15, and 26 for transients of moderate frequency. Design-basis criteria include:

- Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

- Fuel cladding integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit established in RESAR SP/90 Section 4.4.

- An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

- An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered and is an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. For such accidents, fuel failure must be assumed for all rods for which the DNBR falls below those values cited above for cladding integrity. The radiological consequences of this event should not exceed a small fraction of the guideline limits of 10 CFR 100.

- To meet the requirements of GDC 10, 15, and 26, the positions of Regulatory Guide 1.105, "Instrument Setpoints for Safety Related Systems," Rev. 2, February 1986 are used with regard to their effect on the plant response to the type of transient analyzed.

- The most limiting single failure of plant systems (single failure as defined in the "Definitions and Explanations" of 10 CFR 50, Appendix A) shall be identified and assumed in the analysis and shall satisfy the positions of Regulatory Guide 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems" June 1973.

The staff will evaluate compliance with these criteria to ensure that the RESAR SP/90 design meets the requirements of GDC 10, 15, and 26 for transients of moderate frequency. Additionally, the technical specification nominal and allowable set points will be calculated from the set points used in the applicable safety analysis. These calculations will be performed during the licensing process of a plant-specific application referencing the RESAR SP/90 design. The staff will review those calculations at that time.

Category III events are faults that, on the average, are not expected to occur during a plant lifetime. They may result in the failure of a small fraction of the fuel rods, which could preclude immediate resumption of operation. In its August 1989 response, Westinghouse states that the design meets the specific design-basis criteria for these events. These criteria include

- Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

- An infrequent fault (Category III) will not generate a limiting fault (Category IV) without faults occurring independently.

- For events of infrequent faults (Category III), an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. For such accidents, fuel failure must be assumed for all rods for which the DNBR falls below the 95/95 DNBR limit established in Section

4.4 of the RESAR SP/90 application on the basis of acceptable correlations. The radiological consequences of this accident should not exceed a small fraction of 10 CFR 100 limits. A "small fraction" is defined in SRP Section 15.6.3 as 10 percent, which is 2.5 rem whole body and 30 rem thyroid.

In RESAR SP/90 Section 15.0.2.3, Westinghouse indicates that a complete loss of forced reactor coolant flow is categorized as an infrequent-fault event. However, in accordance with the requirements of SRP Section 15.3.1, the staff concludes that this is a fault of moderate frequency. Westinghouse analyzed the loss of flow transient and concluded that the design meets the criteria for Category II events (faults of moderate frequency). The staff finds this acceptable.

Category IV events are limiting faults that are not expected to occur. These accidents are postulated because their consequences would include the potential for release of significant amounts of radioactive material. Plant design must be such as to preclude a fission product release to the environment in excess of guideline values to 10 CFR 100 that would result in an undue risk to public health and safety. In its August 1989 response, Westinghouse states the RESAR SP/90 design meets the following specific criteria for these events.

- Pressure in the reactor coolant and main steam systems should be maintained below acceptable design limits, considering potential brittle as well as ductile failures.

- A limiting-fault event (Category IV) should not, by itself, generate a more serious condition or result in a loss of function of the reactor coolant system or containment barriers.

- The potential for core damage is evaluated on the basis that it is acceptable if the minimum DNBR remains above the 95/95 DNBR limit established in Section 4.4 of the RESAR SP/90 application on the basis of acceptable correlations. If DNBR falls below these values, fuel failure (rod perforation) must be assumed for all rods that do not meet these criteria. Any fuel damage calculated to occur must be sufficently limited in extent so that the core will remain in place and intact with no loss of core cooling capability.

- The calculated radiological consequences of a limiting-fault accident must be within the limits of 10 CFR 100 guidelines.

- Only safety-grade equipment should be assumed to mitigate the consequences of the accident. Safety-related functions should be accomplished assuming the worst single-failure active component in a safety system.

In its August 1989 response, Westinghouse included a table (86-1) that shows the maximum reactor coolant systems (RCS) pressure is less than the faulted-condition stress limits (Service Level D as defined in ASME Code Section III) of a rod ejection event. The acceptance criteria for this event as defined in SRP Section 15.4.8, dictates that the maximum pressure should be less than the emergency-condition stress limits (Service Level C). Westinghouse should clarify this issue and its definition of acceptable design pressure limits for the RCS and main steam system for Category IV events. These issues will be resolved during the FDA stage of review.

For most transients and accidents that are limited by the DNBR, nominal values of initial conditions are assumed. The allowances on power, temperature, and pressure noted above are determined on a statistical basis and are included in the DNBR limit, as described in WCAP-8567, "Improved Thermal Design Procedure." The staff previously reviewed and approved this Westinghouse topical report.

For accidents that are not limited by the DNBR, or for which the procedure described in WCAP-8567 is not employed, initial conditions are obtained by adding the maximum steady-state errors to rated values. For core power, an allowance of ±2 percent is the calorimetric error used for the design; for average RCS temperature, an allowance of ±4°F is the controller deadband and measurement error used for the design; and for pressurizer pressure, an allowance of ±30 psi is the steady-state fluctuations and measurement error used for the design. The staff finds this approach acceptable provided the plant-specific technical specifications limit the plant operation within the conditions that are bounded by the assumed initial conditions for the applicant's accident analyses in the RESAR SP/90 application. The staff will review the plant-specific technical specifications during the licensing process of a plant-specific application referencing the RESAR SP/90 design.

The core power distribution will be continuously monitored by the integrated protection system. Audible alarms will be activiated in the control room whenever the power distribution exceeds the limits assumed as initial conditions for the events analyzed in Section 15.0 of the RESAR SP/90 application. In the analysis of certain events, conservatism requires the use of large reactivity coefficient values; whereas, in the analysis of other events, conservatism requires the use of small reactivity coefficient values. Conservative combinations of parameters are used for each transient to bound the effects of core life although these combinations may not represent realistic situations. The staff concludes that the above design criteria are conservative and acceptable.

No credit is taken for non-safety-grade system operation if that operation mitigates the results of an accident. The staff requested Westinghouse to supply a listing of assumed single failures that were used in the applicant's accident analyses and the limiting single failure of each event analyzed that results in the peak pressure or limiting fuel performance for that event. In its August 1989 response, Westinghouse provided qualitative justification for its selection of limiting single failures. The staff finds the Westinghouse assessment adequate for the PDA.

For rod ejection accidents (Category IV), the staff requires the analysis to include a loss of off-site power resulting from the turbine trip. This will need to be satisfactorily addressed at the FDA stage of review.

The following groups of reactor trip functions mitigate the consequences of the transients and accidents that were analyzed in the RESAR SP/90 application.

- nuclear startup trips
- nuclear overpower trips
- core heat removal trips
- primary overpressure trips
- loss of heatsink trips
- excessive cooldown trips
- turbine trip

All of the transients and accidents analyzed can be grouped according to the following plant process disturbances: increases in heat removal by the secondary system, decreases in reactor coolant system flow rate, reactivity and power distribution anomalies, increase in reactor coolant inventory, decreases in reactor coolant inventory, radioactive releases from a subsystem or component, and anticipated transients without scram.

Most of the applicant's accident analyses were performed with the assumption that the transient was initiated at full-power operation. In its August 1989 response, Westinghouse provided a qualitative discussion on the consequences of such transients and accidents being initiated at low-power levels or lower modes of plant operation to demonstrate that the accident analyses performed represent the bounding cases. The staff finds the Westinghouse assessment adequate for the PDA.

15.2 Normal Operation and Anticipated Transients (RESAR SP/90 Module 10, Section 15.0.2)

15.2.1 Increase in Heat Removal by the Secondary System (RESAR SP/90 Modules 6 & 8, Section 15.1)

Westinghouse provided analyses of events that would produce increased heat removal by the secondary system.

Decrease in Feedwater Temperature

Westinghouse made a comparative assessment for this event and determined that it would result in a transient very similar to, but less severe than, that discussed later for an excessive increase in secondary steam flow. The Westinghouse analysis for increase in secondary steam flow considered the consequences of a 10-percent step load increase.

Increase in Feedwater Flow

An example of excessive feedwater flow would be a full opening of a feedwater control valve as a result of a feedwater control system malfunction or an operator error. At power this event will cause a greater load demand on the reactor coolant system (RCS). With the plant at no-load conditions, this event may cause a reactivity insertion as a result of a decrease in RCS temperature.

In the RESAR SP/90 application, Westinghouse provided the results of its analyses of this event for both full-power and hot-zero-power cases. Continuous addition of excessive feedwater will be protected by the steam generator high-level trip, which will close all feedwater control and isolation valves and will trip the main feedwater pumps.

The results of the Westinghouse analyses of this event show that the peak RCS pressure will be below 110 percent of the RCS design pressure and the transient DNBR will remain above the minimum DNBR defined in RESAR SP/90 Section 4.4, thus meeting the acceptance criteria for this moderate-frequency event.

Excessive Increase in Steam Flow

In the RESAR SP/90 application, Westinghouse provided the results of an analyses for an event associated with a 10-percent step load increase (which the reactor control system is designed to accommodate). The consequences of any loading rate in excess of this value will be mitigated by the reactor protection system signals of low DNBR, high kW/ft, power range high neutron flux, or low pressurizer pressure. Four cases were analyzed: a 10-percent step load increase associated with minimum moderator feedback and manual reactor control, maximum moderator feedback and manual reactor control, minimum moderator feedback and automatic control, and maximum moderator feedback and automatic control. The event associated with the minimum moderator feedback and automatic control leads to the limiting consequences with regard to the peak RCS pressure and fuel performance. However, the peak RCS pressure will be less than 2300 psia and minimum transient DNBR will be well above the minimum DNBR criterion established for the RESAR SP/90 design in this case. Therefore, the results of the analysis meet the acceptance criteria for this moderate-frequency event.

Inadvertent Opening of a Steam Generator Relief or Safety Valve

In the event of increase in heat removal by the secondary system, the most severe core performance conditions will result from an inadvertent opening of a steam generator relief or safety valve.

The steam release, following the event initiation, will result in an initial increase in steam flow, which will decrease the transient as the steam pressure falls. The excessive cooldown will result in an insertion of positive reactivity. The Westinghouse analysis of this event assumed a rod cluster control assembly is stuck, offsite power is available, and a single active failure occurs in the engineered safety features. A reactor trip will occur from low DNBR, high neutron flux, low pressurizer pressure, or a safety injection signal. Main feedwater isolation and trip of the fast acting main steam isolation valves will be provided to mitigate the effects of the event.

The results of the Westinghouse analysis indicate that the RCS pressure will remain below the initial pressure throughout the transient. Westinghouse asserts that the minimum DNBR will remain well above the limiting value during this event. However, Westinghouse has not provided a DNBR curve to demonstrate acceptability of the minimum DNBR for this transient. This issue must be satisfactorily addressed by the applicant during the FDA stage of review.

Except for the DNBR curve, the staff concludes that the results of the Westinghouse analysis meet the staff's acceptance criteria for this moderate-frequency event and for that of an event with a single active failure.

Spectrum of Steam System Piping Failures

Following a main steam line rupture, the steam release will increase in the beginning of the transient and decrease during the accident as the steam pressure decreases. The rapid cooldown rate will result in an insertion of positive reactivity. With the most reactive rod cluster control assembly (RCCA) assumed stuck in its fully withdrawn position after a reactor trip, there is a possibility that the core will become critical and return to power. The reactor will be ultimately shut down because of the boric acid delivered by the safety injection system. A main steam line rupture accident will lead to a reactor trip from low DNBR, high neutron flux, high linear heat flux, low pressurizer pressure, or a safety injection signal. Main feedwater isolation and trip of the fast-acting main steam isolation valves will be provided to mitigate the consequences of this limiting-fault accident.

Westinghouse indicates that a double-ended main steam line rupture at hot-zero power with no decay heat is the most limiting case for this accident. The results of the analysis are provided in RESAR SP/90 Section 15.1.5 with the following major assumptions:

- The most reactive rod cluster control assembly (RCCA) is assumed stuck in its fully withdrawn position after reactor trip.
- A single failure occurs in the engineered safety features.
- End-of-life shutdown margin at no load with xenon conditions at equilibrium exists at the time the accident is initiated.
- The minimum capability exists to inject boric acid (2500 ppm) solution, which corresponds to the most restrictive single failure in the safety injection portion of the ECCS.
- Perfect moisture separation exists in the steam generator.
- Offsite power is available or offsite power is not available following the accident.

Westinghouse indicates that for both cases (with or without offsite power) analyzed: the reactor core will reach criticality after a reactor trip following the accident; the peak power will remain well below the nominal full-power value; and a shutdown will be achieved after the boron solution of 2500 ppm enters the RCS. Westinghouse states that the results of DNB analysis showed that the minimum DNBR will remain above the limiting value during such an event. However, Westinghouse did not provide the transient curves showing DNBR and nuclear power to support this statement.

The staff requires that Westinghouse provide transient curves of DNBR and nuclear power to demonstrate acceptability of the minimum DNBR for this event. This issue must be satisfactorily addressed by the applicant during the FDA stage of review.

Westinghouse indicates that the steam line rupture cases presented in the RESAR SP/90 application are more limiting than a smaller break occurring at full power with or without offsite power available during the transient. To support this position, Westinghouse referred to WCAP-9226, "Reactor Core Response to Excessive Secondary Steam Releases," for its full spectrum of break sizes. In its response to open items dated August 1989, Westinghouse outlined the similarities between the RESAR SP/90 and WCAP-9226 designs and concluded that the WCAP-9226 results are applicable to the RESAR SP/90. The staff position remains that, at the FDA stage, Westinghouse must perform analysis covering the spectrum of break sizes for the steam line break accidents.

With the exception of the above-identified unresolved items, the staff concludes that the consequences of a postulated main steam line rupture accident meet the relevant requirements set forth in the GDC 27, 28, 31, and 35 with regard to control rod insertability and core coolability. The applicant has met the requirements of GDC 27 and 28 by demonstrating that the resultant fuel damage would be limited such that control rod insertability would be maintained and no loss of core cooling capability would result. The minimum DNBR experienced by any fuel rod would be above the limiting value, resulting in 0 percent of the rods experiencing cladding perforation. The applicant has met the requirements of GDC 31 by demonstrating that the primary system boundary will be capable of withstanding the postulated accident. The applicant has met the requirements of GDC 35 by demonstrating that the emergency cooling systems will be adequate to provide abundant core cooling and reactivity control (via boron injection).

15.2.2 Decrease in Heat Removal by the Secondary System (RESAR SP/90 Modules 6 & 8, Section 15.2)

Westinghouse provided analyses of events that would produce decreased heat removal by the secondary system.

Steam Pressure Regulator Malfunction or Failure That Results in Decreasing Steam Flow

There are no steam generator pressure regulators whose failure could cause a steam flow transient for the RESAR SP/90 design.

Loss of External Electrical Load

Westinghouse made a comparative assessment for this event and determined that it would result in a transient similar to, but less severe than, that presented below for a turbine trip event.

Turbine Trip

For a turbine trip event, the reactor would be tripped directly (unless below 10 percent power) by a signal that would automatically stop the turbine oil pressure and turbine stop valves. The turbine stop valves will close rapidly on one of the turbine trip signals. The loss of steam flow will result in a rapid rise in secondary coolant system pressure and temperature and a reduction of the heat transfer rate in the steam generators, which in turn will cause the RCS pressure and temperature to rise.

The turbine trip transient produces the most limiting RCS pressure among the events that decrease heat removal by the secondary system because the turbine trip results in the most rapid reduction in steam flow and, in turn, a rapid RCS heatup results.

The turbine trip transients were analyzed by using the digital computer code LOFTRAN, which has been reviewed and approved by the NRC staff. Initial reactor power and pressure and RCS temperature were assumed to be at their maximum values consistent with steady-state full-power operation, including allowances for calibration and instrument errors.

The most limiting case analyzed is a turbine trip from full power with minimum moderator feedback and no credit for the affect of pressurizer spray and power-operated relief valves in reducing the RCS pressure. Only the safety valves were assumed to be available.

The results of the Westinghouse analysis showed that the peak RCS pressure during the turbine trip transient will be below 110 percent of the RCS design pressure and the DNBR will increase throughout the transient. Therefore, the staff concludes that the RESAR SP/90 design meets the acceptance criteria for this moderate-frequency event.

Inadvertent Closure of Main Steam Isolation Valves

The inadvertent closure of main steam isolation valves would result in a turbine trip and its consequences would be bounded by the analyses discussed above for the turbine trip event.

Loss of Condenser Vacuum

Loss of the condenser vacuum would result in a turbine trip and its consequences would be bounded by the analysis discussed above for a turbine trip event.

Loss of Nonemergency AC Power to Station Auxiliaries

Westinghouse states that this transient is more severe than the turbine trip event because the decrease in heat removal by the secondary system would be accompanied by a RCS flow coastdown, which further reduces the capacity of the primary coolant to remove heat from the core. However, the peak pressurizer pressure during this transient (shown in Figure 15.2-9 of the RESAR SP/90 application would not be higher than the peak pressurizer pressure during a turbine trip (shown in RESAR SP/90 Figure 15.2-5). Westinghouse did not provide a DNBR transient curve for a loss-of-nonemergency-ac-power event. The staff asked Westinghouse to provide a DNBR transient curve for this event and clarify its position comparing the RCS transient pressure of this event with that for a turbine trip event. In its August 1989 response, Westinghouse provided acceptable information.

Loss of Normal Feedwater Flow

An event resulting from the loss of normal feedwater flow would be mitigated by the reactor trip on low water level in any steam generator and the initiation of the emergency feedwater system (EFWS).

The analysis of this event was performed using the LOFTRAN computer code. The major assumptions were that this is the worst single failure in the EFWS, relief of steam in the secondary system is achieved through the steam generator safety valves, and no credit is taken for the startup feedwater system in mitigating the transient.

The loss of normal feedwater analysis was performed to demonstrate the adequacy of the EFWS to remove long-term decay heat and to prevent excessive heatup of the RCS with possible resultant RCS overpressurization or loss of RCS inventory. The results of the analysis show that the peak RCS pressure will be lower than that for a turbine trip event. Westinghouse states, in RESAR SP/90 Section 15.2.7.1, that the primary system variables never approach a DNB condition during a transient involving the loss of normal feedwater flow. However, a DNBR transient curve was not provided for this event. The staff asked Westinghouse to provide a DNBR transient curve for the event of loss of normal feedwater flow. In its August 1989 response, Westinghouse provided acceptable information.

Feedwater System Pipe Break

A feedwater line-break accident is defined as a break in a feedwater line large enough to prevent the addition of sufficient feedwater to the steam generators to maintain shell-side water inventory in the steam generators.

Westinghouse performed a feedwater system pipe-break analysis using assumptions that would minimize secondary system heat removal capability, maximize heat addition to the primary system coolant, and maximize the calculated primary system pressure. A double-ended rupture of the largest feedwater line was assumed. One train of turbine- and motor-driven pumps was assumed unavailable because of the loss of steam supply to the turbine-drive pump and a single failure of the motor-driven pump. In addition, only one train of the safety injection system was assumed to be available, which is consistent with the single-failure criterion. Emergency feedwater flow was assumed to be supplied to only two intact steam generators.

The feedwater line rupture analysis was performed using the LOFTRAN computer code. The results of the analysis for the accident, with or without offsite power available, show that the peak pressures of the RCS and steam generator will be below 110 percent of the system design pressures.

Westinghouse asserts in RESAR SP/90 Section 15.2.8.3 that there will be no fuel failures following the feedwater line rupture accident. However, a DNBR transient curve for this accident was not provided. The staff asked Westinghouse to provide a DNBR transient curve for the feedwater line rupture accident. In its August 1989 response, Westinghouse provided a DNBR curve that shows that, prior to reactor trip, the DNBR remains approximately equal to its initial value. Following reactor trip, the DNBR increases rapidly. The minimum DNBR never approaches the design DNBR limit.

The staff concludes that the results of the Westinghouse feedwater line rupture accident analysis meet the acceptance criteria for this limiting-fault event.

15.2.3 Decrease in Reactor Coolant Flow Rate (RESAR SP/90 Modules 6 & 8, Section 15.3)

Westinghouse provided analyses of events that would result in a decrease in the reactor coolant flow rate.

Loss of Forced Reactor Coolant Flow

Westinghouse analyzed (1) a partial loss of forced reactor coolant flow, assuming a loss of two reactor coolant pumps with four loops in operation, and (2) a total loss of forced reactor coolant flow, assuming a simultaneous loss of electrical power to all reactor coolant pumps. The results of the total loss of forced reactor coolant flow bound that of the partial loss of forced reactor coolant flow. The staff reviewed these events using the procedures and acceptance criteria set forth in SRP Sections 15.3.1 and 15.3.2.

The loss of offsite power and the resulting loss of all forced reactor coolant flow through the reactor core will cause an increase in the average coolant temperature and a decrease in the margin to DNB. The signals from the low reactor coolant pump speed or the low reactor coolant loop flow will trip the reactor.

The results of the Westinghouse analysis show that the peak RCS pressure during the transient will be below 110 percent of the system design pressure and the transient minimum DNBR will be above the limiting value established in RESAR SP/90 Section 4.4. Thus, no fuel failure is predicted to occur, and core geometry and control rod insertability will be maintained with no loss of core cooling capability. Westinghouse did not address the consequences of a limiting single failure in combination with these transients. The staff asked Westinghouse to provide the results of an analysis to demonstrate that a loss of reactor coolant flow with a limiting single failure will meet the acceptance criteria of an infrequent-fault event. In its August 1989 response, Westinghouse provided an acceptable clarification related to limiting single failures. The staff finds the Westinghouse assessment adequate for the PDA stage.

Reactor Coolant Pump Shaft Seizure and Shaft Break

The reactor coolant pump shaft seizure and shaft break are classified as limiting-fault events. In accordance with GDC 17, such an event should be analyzed assuming loss of offsite power throughout the event and the worst single failure of an active component. The maximum RCS activity and steam generator tube leakage allowed by the Westinghouse Standard Technical Specifications should be assumed. The results of the analysis should demonstrate that offsite doses following the accident are less than the 10 CFR 100 guideline values. The potential for fuel failure should be evaluated on the basis that it is acceptable if the minimum DNBR remains above the 95/95 DNBR limit established in RESAR SP/90 Section 4.4 on the basis of acceptable correlations. If the DNBR falls below these values, fuel failure must be assumed for all rods that do not meet these criteria. Any fuel damage calculated to occur must be sufficiently limited so that the core will remain in place and intact with no loss of core cooling capability. Westinghouse's analysis did not meet the above criteria. The staff asked Westinghouse to provide the results of an acceptable analysis, or provide acceptable justification for deviation. Westinghouse provided the new analysis in its August 1989 response.

In its analysis, Westinghouse assumed the failure of one protection train to be the most limiting single failure. It also was assumed that for a locked rotor case without offsite power, some fuel rod failure would occur and that activity contained in the gaps of the failed rod would be released to the primary coolant and eventually into the environment. The resulting doses were shown to be well within the 10 CFR 100 guidelines. The transient DNBR curve for a locked rotor with offsite power also was provided in the response. However, the staff believes that the most limiting case for the radiological consequence calculations could be a stuck-open power-operated relief valve consequent with fuel failure. This issue will be addressed during the FDA stage of review.

15.2.4 Increase in Reactor Coolant System Inventory (RESAR SP/90 Module 4, Section 15.5; Module 1, Section 15.5)

Westinghouse provided analyses of events, such as inadvertent operation of the emergency core cooling system (ECCS) or malfunction of the chemical and volume control system, that would result in an increase in the reactor coolant inventory.

In the RESAR SP/90 application, Westinghouse states that a spurious safety injection system (SIS) operation at power could be caused by false electrical actuation signals. Following the SIS actuation signal, the safety injection pumps will start and the valves isolating the high-concentration borated water source will open. However, the safety injection pumps have a shutoff head at about 1800 psi and consequently, would provide no flow at normal RCS pressure. Therefore, a spurious safety injection with or without a reactor trip has no effect on the RCS.

In its August 1989 response to specific open items contained in the draft safety evaluation report, Westinghouse provided an analysis addressing inadvertent operation of ECCS during plant startup and shutdown when the RCS pressure is below the ECCS pump shutoff head. In addition, Westinghouse provided a discussion addressing inadvertent operation of the centrifugal charging pumps with letdown lines isolated. The staff concludes that the SP/90 design meets the acceptance criteria for this moderate-frequency event.

15.2.5 Decrease in Reactor Coolant System Inventory (RESAR SP/90 Modules 6 & 8 and Module 1, Section 15.6)

Westinghouse provided analyses of events that would result in a decrease in the reactor coolant inventory.

Inadvertent Opening of Pressurizer Safety or Relief Valve

An accidental depressurization of the RCS could occur as a result of an inadvertent opening of a pressurizer relief or safety valve. A pressurizer safety valve is sized to relieve approximately twice the steam flow rate of a relief valve. Therefore, the most severe core conditions resulting from an accidental depressurization of the RCS are associated with an inadvertent opening of a pressurizer safety valve. During this transient, the RCS pressure would continue to decrease and signals for a low DNBR or low pressurizer pressure would trip the reactor.

The Westinghouse analysis of this event was performed using the LOFTRAN computer code. The results of the analysis show that the DNBR will decrease initially but will increase rapidly following the reactor trip. The DNBR will remain above the limiting value throughout the transient.

In its August 1989 response, Westinghouse provided the results of an analysis for the transient with a limiting single failure. The loss of one protection train was assumed to be the most limiting single failure. The results show no adverse effect as a result of this single failure; therefore, the staff concludes that the results of the Westinghouse analysis meet the acceptance criteria for the moderate frequency event.

Steam Generator Tube Rupture

Westinghouse provided the results of an analysis for a steam generator tube rupture (SGTR) event assuming a complete severance of a single steam generator tube.

Following a design-basis SGTR event, it was assumed that alarms signaling pressurizer low pressure and low level, high steam line radiation, high condenser vacuum pump discharge radiation, and high steam generator blowdown sample radiation will assist the plant operator to determine that a SGTR has occurred. Continued loss of RCS inventory will lead to a reactor trip signal generated by low pressurizer pressure or by a low DNBR signal. The reactor trip will automatically trip the turbine, and a loss of offsite power was assumed coincident with the turbine trip. Shortly after the reactor trip, a safety injection signal will be generated by low pressurizer pressure. The safety injection signal will automatically terminate the supply of main feedwater and initiate the emergency feedwater system. Following the plant trip and loss of offsite power supply, steam discharge will be released to the atmosphere through the steam generator safety and/or power-operated relief valves.

The break flow in the affected steam generator will raise the water level rapidly. The level will continue to rise until one of two parallel overflow valves is opened by two-out-of-four high water level signals. This action will release water from the steam generator to the emergency water storage tank inside the containment. The overflow valve will be closed when the steam generator water level drops to a lower level.

Subsequently, the operator actions will be directed toward equalizing the RCS pressure and the affected steam generator pressure. This includes isolating the affected steam generator from the main steam header, which will cool down the RCS by increasing steam relief from the intact steam generators to the atmosphere until the reactor coolant becomes subcooled relative to the affected steam generator pressure. The RCS will be further depressurized using pressurizer power-operated relief valves. Safety injection will then be stopped, allowing pressure to gradually become equalized across the break, stopping break flow.

Table 15.6-1 of RESAR SP/90 Modules 6 & 8 indicates that a reactor trip will occur 690 seconds after the initiation of a SGTR accident. The results of the Westinghouse generic study, documented in WCAP-11002, "Evaluation of Steam Generator Overfill Due to a Steam Generator Tube Rupture Accident," indicate that a reactor trip will occur at 215 seconds after the initiation of a design-basis SGTR. In the SGTR analysis performed for other Westinghouse four-loop plants, a reactor trip will occur within 100 seconds into the SGTR transient. The longer time delay for a reactor trip following a SGTR accident will lead to a nonconservative calculation of the radiological consequences. This is because the steam release from the affected steam generator will be released to the condenser before the assumed loss of offsite power. Following the loss of offsite power with the condenser unavailable, the steam will be released to the atmosphere through steam generator safety or power-operated relief valves.

The staff asked Westinghouse to provide justification for the 690-second time delay on a reactor trip during an SGTR accident. In its August 1989 response, Westinghouse states that the prolonged reactor trip is due mostly to a slower depressurization rate following a SGTR event and a less limiting reactor protection system set point. Westinghouse further states that the prolonged reactor trip time is not expected to have a significant effect on the dose calculation because an additional conservative assumption was used in the RESAR SP/90 SGTR analysis. This conservative assumption of immediate throttling of the emergency feedwater to the steam generators produced a delayed operator action to isolate the affected steam generator and thereby maximized the integrated primary-to-secondary-side leakage that should result in a conservative offsite dose calculation.

The staff finds that more detailed analyses are necessary to assess the consequence of the delayed reactor trip. The applicant must address this issue during the FDA stage of review.

For an SGTR accident, the most limiting single active failure should be assumed to estimate the radiological consequences of the event. Westinghouse has not provided the results of an analysis of an SGTR accident assuming the power-operated relief valve (PORV) on the affected steam generator fails in its open position, which would result in a continuous release of contaminated steam to the atmosphere. In its August 1989 response, Westinghouse took credit for the block valve to close on low steamline pressure in the case of a stuck open steam generator PORV. The staff finds that this response is not acceptable since the steamline associated with a affected steam generator may not generate a low pressure signal. The applicant must address this issue during the FDA stage of review.

Loss-of-Coolant Accident

In RESAR SP/90 Section 15.6.4.1, Westinghouse defines a large-break loss-of-coolant accident (LBLOCA) as a rupture of the RCS pressure boundary with a total cross-section equal to or greater than 1.0 ft^2. A small-break loss-of-coolant accident (SBLOCA) is defined as a rupture with a total cross-section less than 1.0 ft^2 in which the normal operating charging system flow is not sufficient to sustain pressurizer level and pressure.

The staff reviewed the LOCA analysis and ECCS performance in accordance with the requirements of 10 CFR 50.46; 10 CFR 50, Appendix K; GDC 35; and 10 CFR 100. The results of the LOCA analysis should show that the ECCS satisfies the following criteria:

(1) The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

(2) The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

(3) The calculated total amount of hydrogen generated shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(4) Calculated changes in core geometry shall be such that the core remains amenable to cooling.

(5) After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended time required by the long-lived radioactivity remaining in the core.

These criteria were established to provide significant margin in ECCS performance following a LOCA. Westinghouse states that in all cases, a SBLOCA yields results with more margin to the acceptance criteria limits than a LBLOCA.

Following a SBLOCA, a reactor trip signal will be generated by pressurizer low pressure. A safety injection actuation signal will be generated when the appropriate set point is reached. Borated water will be injected to the reactor core by the ECCS to provide shutdown margin and remove heat from the core to prevent excessive cladding temperatures. In addition, a high containment pressure signal will be generated to start the low-head containment spray pumps. The high-head safety injection pumps will inject borated water from the EWST to the reactor vessel when the RCS pressure is below 1800 psia. When the RCS depressurizes to 600 psia, the accumulators will begin to inject borated water to the RCS cold legs. When the RCS depressurizes to 200 psia, the core reflood tanks will inject borated water to the reactor vessel. The water discharged through the break will be returned to the EWST inside the containment and no switchover from the injection mode to the recirculation mode will be required. Approximately 24 hours after the LOCA, the ECCS will be realigned to supply water to the RCS hot legs to prevent boron stratification in the reactor vessel.

For the LBLOCA analysis, Westinghouse used the 1981 version of the evaluation model, which includes modifications delineated in WCAP-9220-P-A, "Westinghouse ECCS Evaluation Model, Class 2," and WCAP-9221-P-A, "Westinghouse ECCS Evaluation Model, Class 3." In response to the staff's request, Westinghouse has committed to specifically developing an ECCS evaluation model for LBLOCA for the RESAR SP/90 design per the requirement of 10 CFR 50, Appendix K, and submitting it for NRC review in the RESAR SP/90 FDA application. The staff finds this acceptable.

The LBLOCA analysis was performed using three different flow discharge coefficients. The results of the analysis show that the double-ended cold leg break with a Moody break discharge coefficient of 1.0 is the worst case. In this case, the peak cladding temperature is below 2200°F with sufficient margin.

For the SBLOCA analysis Westinghouse used the 1975 version of the ECCS evaluation model as it is documented in WCAP-8970-P-A, "Westinghouse Emergency Core Cooling System Small Break, Class 3," and WCAP-8971, "Westinhouse Emergency Core Cooling System Small Break, Class 2." In response to the staff request, Westinghouse has committed to specifically developing an ECCS evaluation model for SBLOCA for the RESAR SP/90 design and submitting it for NRC review in the RESAR SP/90 FDA application. The staff finds this acceptable.

Westinghouse performed the SBLOCA analysis for three different break sizes: a break of RCS piping of 3-inch, 4.313-inch, and 6-inch diameters. Westinghouse asserts that there was no core uncovery during these postulated SBLOCA.

Westinghouse presented results of the SBLOCA analysis in RESAR SP/90 Module 1, Tables 15.6.4-4 and 15.6.4-5 and Figures 15.6.4-31 through 15.6.4-39. The 4.313-inch-diameter break was identified to be the worst case. Westinghouse provided additional results and discussions in its August 1989 response and its response to staff questions 440.210, 440.220, and 440.221 dated May 1986.

With Westinghouse's commitment to develop RESAR SP/90 design-specific ECCS evaluation models during the FDA stage of review, the staff finds this acceptable for the PDA.

15.3 Reactivity and Power Distribution Anomalies (RESAR SP/90 Module 5, Section 15.4)

As discussed in Section 4.3 of this SER, the RESAR SP/90 neutronic parameters used in transient and accident analyses are generally either more favorable or are within or near the range of parameters used in analyses for typical Westinghouse reactors. Thus the analyses, methodologies, and results of corresponding events for typical Westinghouse reactors and the RESAR SP/90 are very similar and the results of the reviews are similar. Module 5 of the RESAR SP/90 application presents the analyses and results for the usual events related to the control rods and fuel assembly misloading.

Relevant parameters associated with the gray rods and water displacement rods fall within the bounds covered by the control rods and therefore are not explicitly mentioned here.

15.3.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From Zero-Power Conditions (RESAR SP/90 Module 5, Section 15.4.1)

Westinghouse provided the results of its analysis of the consequences of an uncontrolled rod cluster control assembly bank withdrawal at zero power. The staff reviewed the potential for single failures of the reactor control system that could result in uncontrolled withdrawal of control rods under low-power startup conditions in accordance with SRP Section 15.4.1 and the requirements of GDC 10, 20, and 25. The scope of the review included (1) investigations of initial conditions and control rod reactivity worths, the course of the resulting transients or steady-state conditions, and the instrument response to the transient or power maldistribution and (2) examination of the methods used to determine the peak fuel rod response and the input into the analysis (such as power distributions and reactivity feedback effects resulting from moderator and fuel temperature changes).

Such a transient can be caused by a failure of the reactor control rod control systems. The analysis assumed a conservatively small (in absolute magnitude) negative Doppler coefficient and a conservative moderator coefficient. Further, initial conditions at hot-zero power with the reactor just critical were chosen because they are known to maximize the calculated consequences. The reactivity insertion rate was assumed to be equivalent to the simultaneous withdrawal of the two highest-worth banks at maximum speed (45 inches per minute). The withdrawal of gray rods or water displacement rods would be less limiting.

Reactor trip was assumed to occur on the low setting of the power range neutron flux channel at 35 percent of full power (a 10 percent uncertainty has been added to the set point value). The results show that maximum heat flux will be much less than the full power value and average fuel temperature will increase to a value lower than the nominal full-power value. The minimum DNBR at all times will remain above the limiting value of 1.17, using the WRB-2 correlation.

The applicant has met the requirements of GDC 10 in that the specified acceptable fuel design limits will not be exceeded, GDC 20 in that the reactivity control systems will be automatically initiated so that specified acceptable fuel design limits will not be exceeded, and of GDC 25 in that single malfunctions in the reactivity control system will not cause the specified acceptable fuel design limits to be exceeded. These requirements were met by comparing the resulting extreme operating conditions and response for the fuel (i.e., fuel duty) with the acceptance criteria for fuel damage (e.g., critical heat flux, fuel temperatures, and strain limits of the cladding should not be exceeded) to ensure that fuel rod failure will be precluded for this event.

The staff concludes that the applicant's analysis of the maximum transients for single-failure control rod withdrawal from a subcritical or low-power condition are acceptable, that the analytical methods and input data are reasonably conservative, and that specified acceptable fuel design limits will not be exceeded.

15.3.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (RESAR SP/90 Module 5, Section 15.4.2)

Westinghouse provided the results of its analysis of the consequences of controlled withdrawal of a rod bank in the power operating range. The staff reviewed the potential for single failures of the reactor control systems that could result in uncontrolled withdrawal of control rods beyond normal limits under power operation conditions in accordance with SRP Section 15.4.2 and the requirements of GDC 10, 20, and 25.

The scope of the review included (1) investigations of possible initial conditions and the range of reactivity insertions, the course of the resulting transients, and the instrumentation response to the transient and (2) examination of the methods used to determine the peak fuel rod response and the input into the analysis (such as power distributions, rod reactivities, and reactivity feedback effects of moderator and fuel temperature changes).

The effect of such an event is an increase in fuel and coolant temperature (as a result of the core-turbine power mismatch), which must be terminated before exceeding fuel design limits. Reactivity insertion rates, reactivity feedback coefficients, and core power level were analyzed. Protection will be provided by the high neutron flux trip, the N16 DNBR and kW/ft trips, and pressurizer pressure and pressurizer water level trips. In no case did the DNBR fall below the limiting value of 1.17, using the WRB-2 correlation. Adequate fuel cooling will therefore be maintained. The maximum heat flux reached, including uncertainties, did not exceed 118 percent of full power, thus precluding fuel centerline melting.

As discussed in Sections 4.3.1 and 4.3.3 of this SER, the N16 and four-segment excore detector protection system offer improvements over the core delta temperature power level systems and the two-segment excore detectors, respectively. It has been partially approved in connection with the RESAR 414 and Comanche Peak reviews. This withdrawal-at-power review assumes satisfactory operation of the system.

The applicant has met the requirements of GDC 10 in that the specified acceptable fuel design limits will not be exceeded, of GDC 20 in that the reactivity control systems will be automatically initiated so that specified acceptable fuel design limits will not be exceeded, and of GDC 25 in that single malfunctions in the reactivity control system will not cause the specified acceptable fuel design limits to be exceeded. These requirements were met by comparing the resulting extreme operating conditions and response for the the fuel (i.e., fuel duty) with the acceptance criteria for fuel damage (e.g., critical heat flux, fuel temperatures and cladding strain limits should not be exceeded) to ensure that fuel rod failure will be precluded for this event. Therefore, on the basis of its review and pending satisfactory review of the N16 and four-segment excore detector protection system, the staff concludes that the applicant's analysis of maximum transients for single-error control rod malfunctions are acceptable, that the analytical methods and input data are reasonably conservative, and that specified acceptable fuel design limits will not be exceeded.

15.3.3 Rod Cluster Control Assembly Malfunctions (RESAR SP/90 Module 5, Section 15.4.3)

Westinghouse provided the results of its analyses of rod cluster control assembly misalignment incidents including a dropped full-length assembly, a dropped full-length bank, a misaligned full-length assembly, and the withdrawal of a single assembly while operating at power. However, the analysis of the dropped rod or bank event have not been presented and will be addressed du ing the FDA stage of review.

The staff has reviewed the potential for single failures of the reactor control system that could result in a movement or malposition of control rods beyond normal limits in accordance with SRP Section 15.4.3 and the requirements of GDC 10, 20, and 25. The scope of the review included (1) investigations of possible rod malposition configurations, the course of the resulting transients or steady-state conditions, and the instrumentation response to the transient of power maldistribution and (2) examination of the methods used to determine the peak fuel rod response and the input to that analysis (such as power distribution changes, rod reactivities, and reactivity feedback effects as a result of moderator and fuel temperature changes).

Misaligned rods are detectable by: (1) asymmetric power distributions sensed by excore nuclear instrumentation or core exit thermodcouples, (2) rod deviation alarms, and (3) rod position indicators. A deviation of a rod from its bank by about 15 inches or twice the resolution of the rod position indicator will not cause the power distribution to exceed design limits. Additional surveillance will be required to ensure rod alignment if one or more rod position channel is out of service.

For cases where a group is inserted to its insertion limit with a single rod in the group stuck in the fully withdrawn position, analysis indicates that DNB will not occur. The staff reviewed the calculated estimates of the expected reactivity and power distribution changes that accompany postulated misalignments of representative assemblies and concludes that the values used in this analysis conservatively bound the expected values including calculational uncertainties.

The inadvertant withdrawal of a single assembly requires multiple failures in the control system, such as multiple operator errors or deliberate operator actions combined with a single failure of the control system. As a result, the single assembly withdrawal is classified as an infrequent occurrence. The resulting transient from such an event would be similar to that resulting from a bank withdrawal, but the increased peaking factor would cause DNB to occur in the region surrounding the withdrawn assembly. However, less than 5 percent of the rods in the core would experience DNB for such a transient.

The applicant has met the requirements of GDC 10 in that the specified acceptable fuel design limits will not be exceeded, of GDC 20 in that the reactivity control systems will be automatically initiated so that specified acceptable fuel design limits will not be exceeded, and of GDC 25 in that single malfunctions in the reactivity control system will not cause the specified acceptable fuel design limits to be exceeded. These requirements were met by comparing the resulting extreme operating conditions and response for the fuel (i.e., fuel duty) with the acceptance criteria for fuel damage (e.g., critical heat flux, fuel temperatures and clad strain limits should not be exceeded) to ensure that fuel rod failure will be precluded for this event.

The staff concludes that maximum configurations and transients for single-error control rod malfunctions are acceptable, that the analysis methods and input data are reasonable conservative, and that specified acceptable fuel design limits will not be exceeded. Results of the staff's review of the dropped rod or bank event will be discussed during the FDA stage when Westinghouse submits its analysis for the event along with the limiting single failure.

15.3.4 Inadvertent Loading of a Fuel Assembly Into Improper Position (RESAR SP/90 Module 5, Section 15.4.7)

Strict administrative controls in the form of previously approved and established procedures and startup testing will be followed during fuel loadings to prevent operation with a fuel assembly in an improper location of a misloaded burnable poison assembly. Nevertheless, the applicant has performed an analysis of the consequences of a loading error. The staff has reviewed this event in accordance with SRP Section 15.4.7 and the requirements of GDC 13 and 10 CFR 100.

The applicant provided comparisons of power distributions calculated for the nominal fuel loading pattern and those calculated for five loadings with misplaced fuel assemblies or burnable poison assemblies. The selected non-normal loadings represent the spectrum of potential inadvertent fuel misplacements. Calculations included, in particular, the power in assemblies that contain provisions for monitoring with in-core detectors.

As part of the required startup testing, the in-core detector system will be used to detect misloaded fuel before operating at power. The analyses described above shows that all but one of the above misloading events would be detected by this test. In the excepted case, an interchange of Region 1 and 2 assemblies near the center of the core, the increase in the power peaking is approximately equal to the uncertainty in the measurement of the quantity (5 percent). This uncertainty was considered in analysis and shows that this misloading event will not result in unacceptable consequences.

The applicant has met the requirements of GDC 13 with respect to providing adequate provisions to minimize the potential of a misloaded fuel assembly going undetected and of 10 CFR 100 with respect to mitigating the consequences of reactor operations with a misloaded fuel assembly. These requirements have been met by providing acceptable procedures and design features that will minimize the likelihood of loading fuel in a location other than its designated place.

The staff has evaluated the consequences of a spectrum of postulated fuel loading errors and concludes that the analyses show for each case considered that either error would be detectable by the available instrumentation (and hence remediable) or the error would be undetectable, but the offsite consequences of any fuel rod failures would be a small fraction of 10 CFR 100 guidelines. The applicant affirms that the available in-core instrumentation will be used before the start of a fuel cycle to search for fuel loading errors. The staff finds this acceptable.

15.3.5 Rupture of a Control Rod Drive Mechanism Housing Control Assembly Ejection (RESAR SP/90, Module 5, Section 15.4.8)

The mechanical failure of a control rod mechanism pressure housing would result in the ejection of a rod cluster control assembly. For assemblies initially inserted, the consequences would be a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. Although mechanical provisions have been made to make this accident extremely unlikely, the applicant has provided its analysis of the consequences of such an event. The staff has reviewed this analysis in accordance with SRP Section 15.4.8 and the requirements of GDC 28.

Methods used in the analysis are reported in WCAP-7588, Revision 2, "An Evaluation of the Rod Ejection Accident in Westinghouse Reactors Using Spatial Kinetics Methods," which has been reviewed and accepted by the staff. This report demonstrated that the model used in the accident analysis is conservative with regards to a three-dimensional kinetics calculation.

The applicant's criteria for gross damage of fuel are a maximum cladding temperature of 2700°F and an energy deposition of 200 calories per gram in the hottest pellet. These criteria are more conservative than those proposed in Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for PWRs."* Therefore, they are acceptable.

Eight cases were analyzed: beginning-of-cycle at 102 percent and zero power and end-of-cycle at 102 percent and zero power, each with water displacement rods in and out (i.e., voided and unvoided). The highest cladding temperatures, 2499°F, and the highest fuel enthalpy, 165 calories per gram, were reached in the beginning-of-cycle full-power unvoided cases and beginnining-of-cycle full-power voided cases, respectively. The analysis also shows that less than 10 percent of the fuel experiences DNB and less than 10 percent of the hot pellet melts. Analyses were performed to show that the pressure surge produced by the rod ejection will not exceed the reactor coolant system faulted condition stress limits (Service Level D, as defined in ASME Code Section III). However, the acceptance criteria of SRP Section 15.4.8 specifies that maximum pressure should not exceed that value that causes emergency condition stress limits (Service Level C).

Further analyses show that a cascade effect, that is, the ejection of another rod as a result of the ejection of the first rod, is not credible.

Since the calculations resulted in peak fuel enthalpies less than 280 calories per gram, prompt fuel rupture with consequent rapid heat transfer to the coolant from finely dispersed molten UO_2 was assumed not to occur. The pressure surge was, therefore, calculated on the basis of conventional heat transfer from the fuel and resulted in a pressure increase below Service Level D for the maximum control rod worths assumed.

By not satisfying all the regulatory positions in Regulatory Guide 1.77 (i.e., maximum calculated RCS pressure exceeds the value which causes emergency condition stress limit [Service Level C], the applicant has not met the requirements of GDC 28 with respect to preventing postulated reactivity accidents that could result in damage to the reactor coolant pressure boundary greater than limited local yielding or cause sufficient damage that would significantly impair the capability to cool the core. Furthermore, as stated in Section 15.1 of this

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*RG 1.77 has an acceptance criterion of 280 calories per gram energy deposition and no criterion for cladding temperature other than that implicit in requirements for fuel and pressure vessel damage.

SER, for rod ejection accidents, a loss of off-site power should be assumed in addition to the assumed limiting single failure in the safety analyses.

The staff concludes that the rod ejection results are not acceptable. The above two items must be satisfactorily addressed during the FDA stage of review.

15.4 Radiological Consequences of Design-Basis Accidents (RESAR SP/90, Modules 6 & 8, Section 15A)

The staff reviewed the RESAR SP/90 application as it relates to the radiological assessment of design-basis accidents (DBAs). By letters dated April 14 and April 15, 1988, the staff requested the applicant to provide additional information concerning the dose consequences of various DBAs. The staff has reviewed the applicant's response of May 27, 1988. The staff has independently evaluated the potential radiological consequences of these various DBAs using the calculational assumptions from the appropriate SRP sections and regulatory guides. Since this evaluation is based on the preliminary design of the RESAR SP/90 and typical site characteristics, the staff's conclusions are not final regarding the ability of the facility to satisfy the appropriate SRP dose guidance values. The staff will reach its final conclusions regarding this matter during the FDA stage of review and on a plant-specific basis when final site parameters are determined.

15.4.1 Loss-of-Coolant Accident

The limiting fault postulated as the design basis for the containment and its associated engineered safety features (ESFs) and as a demonstration of the adequacy of the distances to the exclusion area boundary (EAB) and low-population zone (LPZ) is a LOCA in conjunction with the release of a substantial fraction of the fission product inventory of the core, as set forth in 10 CFR 100.11(a). The analysis includes the sources and radioactivity transport assumptions specified in Regulatory Guide 1.4, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors," as well as additional guidance contained in the SRP.

This postulated event involves the assumed availability for release from the containment atmosphere of 100 percent of the core's inventory of noble gas and 25 percent of the iodine inventory. Although the containment is assumed to be intact, two pathways are normally analyzed for leakage of fission products, the containment leakage and ESF leakage outside containment. Only the containment pathway was analyzed because the RESAR SP/90 design encompasses the emergency core cooling systems (ECCS) completely within the containment system.

The staff utilized the preliminary design and typical site characteristic data presented in the RESAR SP/90 application. The safety features of the facility include a containment designed to minimize the leakage of fission products from postulated accidents involving the failure of the first two barriers against a release of fission products, that is the fuel cladding and primary pressure boundary. Another engineered safety feature is the containment spray system with a sodium phosphate additive to achieve a slightly basic pH in the water that would collect in the containment sump following a LOCA. The staff's calculation of the consequences of the hypothetical LOCA used the conservative assumptions of Positions C.1.a through C.1.e of Regulatory Guide 1.4, Revision 2. The containment was assumed to leak at a rate of 0.1 percent per day for the first 24 hours and 0.05 percent per day after 24 hours. The analysis took into account radiological decay during holdup in the containment, mixing in the containment, iodine decontamination by the ESF spray system, and a conservative estimate of dispersion of the fission products in the environment. A list of assumptions used in the staff's calculation of the LOCA doses is given in Table 15.1 of this SER.

The staff's calculated doses from the hypothetical LOCA for the 0-2 hour time period at the EAB and the 0-30 day period at the LPZ and control room are given in Table 15.2. The staff concludes that, for the typical site and the preliminary design of ESFs of RESAR SP/90, the radiological consequences of the postulated LOCA fall within the exposure guidelines in 10 CFR 100.11. Based on the preliminary design of the control room habitability system, the postulated LOCA doses to the control room operators are below the guideline dose values of GDC 19. The ability of the RESAR SP/90 design to satisfy the guideline doses of 10 CFR 100 for a site-specific facility will be determined on a case-by-case basis. Similarly, the ability of a site-specific control room habitability system to satisfy the guidelines of GDC-19 will have to be evaluated on a case-by-case basis.

15.4.2 Steam Generator Tube Rupture Accident

Westinghouse has provided analyses of the consequences of a variety of steam generator tube rupture (SGTR) accidents. However, the staff has requested the applicant to analyze the consequences of an SGTR accident with a break at the top of a steam generator tube and an associated scenario that minimizes the inventory of water in the affected steam generator. This SGTR accident has the potential of being the design-basis SGTR because analyses performed for several Westinghouse operating nuclear power plants have indicated that the broken tube could be uncovered, causing a potentially large release of iodine from the primary coolant. This iodine release could be greater than that normally used in the SGTR dose assessment analysis. The Westinghouse Owners Group (WOG) is presently working with the staff to resolve this tube uncovery issue. Following the completion of the WOG efforts, Westinghouse will evaluate this SGTR for the RESAR SP/90 design and include the results as part of the FDA application. The staff will review the WOG resolution efforts and will ensure the WOG results are appropriately incorporated in the evaluation of the radiological consequences for RESAR SP/90 during the FDA stage of review.

15.4.3 Radiological Consequences of the Failure of a Small Primary Coolant Line Outside Containment

The radiological consequences of the failure of certain small lines that carry primary coolant outside the containment building are evaluated to ensure compliance with the exposure guidelines of 10 CFR 100.11. Westinghouse provided an analysis of an accidental break in the chemical and volume control systems (CVCS) letdown line outside containment and downstream of the containment isolation valves as a worst-case failure of one of these lines. This break would release 296 gpm of primary coolant to the auxiliary building before isolation could be expected in 30 minutes.

Table 15.1 Assumptions used in the staff's calculation of loss-of-coolant-accident doses

Parameter, Unit of Measure	Quantity
Power level, MWt	3,800
Operating time, year	3
Fraction of core inventory available for containment leakage, %	
Iodine	25
Noble gases	100
Initial iodine composition in containment atmosphere, %	
Elemental	91
Organic	4
Particulate	5
Containment leak rate, %/day	
0-24 hr	0.1
After 24 hr	0.05
Total containment vo' me, ft³	2.75×10^6
Sprayed volume	2.2×10^6
Unsprayed volume	5.5×10^5
Containment mixing rate from cooling fan operation, ft³/min	90,000
Containment spray system	
Maximum elemental and particulate iodine decontamination factors, respectively	100
Spray removal coefficients, %/hr	
Elemental iodine	5.5
Particulate iodine	1.5
Organic iodine	0
Relative concentration values, sec/m³	
0-2 hr at the exclusion area boundary (EAB)	2.0×10^{-4}
0-8 hr at the low-population zone (LPZ) boundary	7.0×10^{-5}
8-24 hr at the LPZ boundary	2.0×10^{-5}
24-96 hr at the LPZ boundary	9.0×10^{-6}
96-720 hr at the LPZ boundary	3.0×10^{-6}
Control room habita''ility, sec/m³	
0-8 hr at the control room intake	4.0×10^{-3}
8-24 hr at the control room intake	2.8×10^{-3}
24-96 hr at the control room intake	2.0×10^{-3}
96-720 hr at the control room intake	1.5×10^{-3}
Control room volume, ft³	1.75×10^5
Recirculation flow rate, ft³/min	2.5×10^4
Filtered flow rate, ft³/min	5.0×10^2
Unfiltered inleakage, ft³/min	1×10
Total iodine removal efficiency	0.95

Table 15.2 Radiological consequences of design-basis accidents as calculated by the staff

Postulated Accident	Exclusion-Area-Boundary Dose, Rem		Low-Population-Zone Dose, Rem		Control Room Dose, Rem		
	Thyroid	WB*	Thyroid	WB	Thyroid	WB	Skin
Loss of coolant, containment leakage	1×10^2	≤1.0	1×10^2	≤1.0	30	0.7	20
Steamline break outside containment:							
Long-term operation case (DEI-131 at 1 μCi/gm)	1.2	≤1.0	2.1	≤1.0			
Short-term operation case (DEI-131 at 60 μCi/gm)	1.4	≤1.0	1.2	≤1.0			
Control rod ejection	28	≤1.0	15	≤1.0			
Fuel handling accident in fuel handling area	58	≤1.0	20	≤1.0			
Fuel handling accident inside containment	70	≤1.0	25	≤1.0			
Small-line break	9	≤1.0	3	≤1.0			

*WB = whole body.

The staff has performed an independent assessment of the potential dose consequences of the release of primary coolant outside the containment. The staff assumed that 30 minutes would elapse before isolation of the CVCS line break. Thus, a total of 8885 gallons of primary coolant could be released. It was estimated that 18 percent of the hot reactor coolant would flash into steam upon entering the auxiliary building atmosphere and assumed that a proportional fraction of the iodine dissolved in the coolant would become airborne in gaseous or particulate form. The staff further assumed that this airborne iodine could, without delay or effective filtration, escape directly to the environment at ground level. Other assumptions are given in Table 15.3 of this SER.

Table 15.3 Assumptions used by the staff for estimating accidents involving small-line breaks outside containment

Parameter, Unit of Measure	Quantity
Coolant released, lb	74,100
Fraction of coolant released flashed to steam, %	17.9
Coolant contaminant concentration, μCi/gm	1.0
Spiking factor (iodine release rate multiplier)	500

The staff concludes that the consequences of a postulated small-line failure outside the containment, assuming the primary coolant equilibrium iodine concentration permitted by the Westinghouse Standard Technical Specifications in combination with an accident-generated iodine spike, do not exceed a small fraction of the exposure guidelines of 10 CFR 100.11. These conclusions will have to be re-evaluated during the FDA stage of review and on a case-by-case basis using site-specific information during the licensing process of a plant-specific application referencing the RESAR SP/90 design.

15.4.4 Control Rod Ejection Accident

The staff reviewed Westinghouse's analysis of radiological consequences and the Westinghouse Standard Technical Specifications for the primary-to-secondary leakage in the steam generator and performed independent dose calculations using guidance of Regulatory Guide 1.77 Appendix B, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," May 1974.

A nonmechanistic rupture of a control rod drive housing was postulated for the RESAR SP/90 design. Because of the resultant opening in the pressure vessel, primary coolant will be lost to the containment with concurrent rapid depressurization of the reactor pressure vessel. Reactor trip, will be initiated by one of several trip signals, and was assumed to occur rapidly.

Ejection of a control rod will result in rapid reactivity insertion. Westinghouse conservatively assumed that 10 percent of the fuel elements will experience cladding failure, releasing the volatile fission products in the fuel-cladding gap. In addition, 0.25 percent of the fuel rods may experience fuel melting. The fission products released as a result of this damage to the fuel were assumed to be released with the primary coolant. The release to the environment may occur to either of two pathways: the first pathway would involve a release of coolant carrying fission products to the primary containment, which was then assumed to leak to the atmosphere at the design leak rate of the containment (0.1 percent per day). In the second pathway, activity would reach the secondary coolant via steam generator tube leaks. A maximum of 1-gpm primary-to-secondary leak rate was assumed. With loss of offsite power (assumed to occur as a result of reactor/turbine trip) and subsequent steam venting, some of the iodine transferred to the shell (secondary) side would be available for leakage to the environment.

In considering the consequences of this postulated event, the staff calculated the doses from the activity available for release separately for each of the above pathways. The staff would expect the actual consequences to be some combination of these pathways. The assumptions used in calculating the radiological consequences are presented in Table 15.4 and the resultant doses are given in Table 15.2 of this SER.

The staff has reviewed Westinghouse's analysis of the radiological consequences following a postulated control rod ejection accident and concludes that, for the typical site and preliminary design, the postulated control rod ejection accident doses are well within (less than 25 percent) the dose guidelines of 10 CFR 100.11.

These conclusions will have to be re-evaluated during the FDA stage of review and on a case-by-case basis using site-specific information during the licensing process of a plant-specific application referencing the RESAR SP/90 design.

15.4.5 Fuel-Handling Accident

In its evaluation of the fuel-handling accident, the staff based its criteria and methodology on the guidance of 10 CFR 100, GDC 61(3), Positions C.1.a through C.1.f 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 SRP Section 15.7.4. The staff assumed that a single fuel assembly is dropped into the fuel pool during refueling operations and that all of the fuel rods in the assembly would be damaged, releasing radioactive materials in the fuel gaps into the pool. The staff's analysis of the fuel handling accident in containment assumed that the fuel assembly is dropped into the core, releasing the gap activity of 1.2 assemblies into the pool.

The estimated offsite doses for the postulated fuel handling accidents inside containment and inside the fuel building are shown in Table 15.2. The list of assumptions and parameters used in the staff's analysis are given in Tables 15.5 and 15.6 of this SER. The staff concludes that the potential doses for the fuel-handling accidents are well within the guidelines values given in 10 CFR 100.11.

With regard to a potential spent-fuel cask-drop accident, the crane (used in handling the cask) will be designed to prevent a cask from being moved over the spent fuel in the pool. The staff concludes that the likelihood of a spent-fuel cask-drop accident is sufficiently small that no radiological consequence analysis is required. Section 9.1.5 of RESAR SP/90 application provides an evaluation of this crane as part of the RESAR SP/90 overhead heavy-load-handling system.

Table 15.4 Assumptions used by the staff for estimating the radiological consequences following a postulated control-rod-ejection accident

Parameter, Unit of Measure	Quantity
Power level, MWt	3800
Primary-to-secondary leak rate, gpm	1.0
Fraction of the fuel rods experiencing cladding failure	0.1
Fraction of noble gas and iodine inventory in gap of failed rods	0.1
Fraction of the fuel rods experiencing fuel melting	0.0025
Fraction of iodine inventory released from rods experiencing melting	0.5
Fraction of iodine entering steam generator secondary side released environs	0.1
Time of primary system release, sec	10000
Fraction of iodine plated out in containment, %/day	0.5
Design leak rate of containment, %/day	0.1
Iodine concentration (DEI-131) in the secondary coolant, μCi/gm	0.1

Table 15.5 Assumptions used by the staff for estimating the radiological consequences following a postulated fuel-handling accident in fuel-handling building

Parameter, Unit of Measure	Quantity
Power level, MWt	3800
Number of fuel assemblies	1.0
Radial peaking factor of damaged rod	1.65
Shutdown time, hr	100
Inventory released from damaged rods (iodine and noble gases), %	10
Kr - 85, %	30
Pool decontamination factors	
Iodine	100
Noble gases	1
Iodine removal efficiencies for auxiliary building gas treatment system (spent fuel pool area), %	
Elemental	no filters assumed
Organic	no filters assumed

Table 15.6 Assumptions used by the staff for estimating the radiological consequences following a postulated fuel-handling accident inside containment

Parameter, Unit of Measure	Quantity
Power level, MWt	3800
Number of fuel assemblies damaged	1.2
Radial peaking factor of damaged rod	1.65
Shutdown time, hr	100
Inventory released from damaged rods (iodine and noble gases), %	10
Kr - 85, %	30
Reactor water cover decontamination factors	
Iodine	100
Noble gases	1
Iodine fractions released from reactor water cover, %	
Elemental	75
Organic	25
Iodine removal efficiencies for containment effluent, %	
Elemental	no filters assumed
Organic	no filters assumed

A re-evaluation of spent fuel-handling accidents will be conducted during the FDA stage of review and on a case-by-case basis using site-specific information during the licensing process of a plant-specific application referencing the RESAR SP/90 design.

16 TECHNICAL SPECIFICATIONS

The technical specifications in a license define certain features, characteristics, and conditions governing operation of a facility that cannot be changed without prior approval of the staff. Included will be sections covering definitions, safety limits, limiting safety system settings, limiting conditions for operation, surveillance requirements, design features, and administrative controls.

Because of the stage of the design of RESAR SP/90, the applicant has not provided specific information for technical specifications. By letter dated November 22, 1988, the staff indicated that proposed technical specifications representative of the RESAR SP/90 design should be submitted for review and approval by the staff as part of the FDA submittal and will be included in the design certification process. The applicant should identify design features that are necessary for testing and maintenance during operation without challenging safety systems.

Where practicable, the technical specifications should be developed on the basis of risk and reliability considerations. During the course of its review of the RESAR SP/90 PDA application, the staff identified certain issues that must be included in the technical specifications as a condition of staff acceptance. These issues are listed below and are discussed further in the sections of this SER as indicated in parentheses.

- inservice testing of pumps and valves (3.9.6)
- control rod insertion operating limits (4.3.3)
- reactor coolant system flow monitoring (4.4.3)
- thermal design flow determination (4.4.3)
- testing of the emergency core cooling system (6.3.4)
- steady-state errors for core power, average RCS temperature, and pressurizer pressure (15.1)
- maximum RCS activity and steam generator tube leakage (15.4)
- primary coolant equilibrium iodine concentration (15.4)

New standard technical specifications are being developed by the vendor owners groups and the NRC to implement the Commission's interim policy statement on technical specification improvement. The new standard technical specifications will be the model for the staff's review of the technical specifications submitted during the FDA stage. Some of the items identified above may not be included in the new standard technical specifications and will, therefore, be reviewed again during the FDA stage.

17 QUALITY ASSURANCE (RESAR SP/90 Module 3, Section 17.0)

17.1 General

The description of the quality assurance program for the design and design assurance of RESAR SP/90 references WCAP-8370, Revision 9A, Amendment 1, "Westinghouse Water Reactor Divisions Quality Assurance Plan." The staff has further evaluated the most current version of the quality assurance program description, which is contained in Revision 11 of WCAP-8370, "Westinghouse Energy Systems Business Unit/Nuclear Fuel Business Unit Quality Assurance Plan," against the acceptance criteria given in SRP Section 17.1. The staff's evaluation was based on a review of the information provided in the RESAR SP/90 application and discussions with representatives from Westinghouse. The staff assessed the Westinghouse quality assurance program to determine if it complies with the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Plants and Fuel Reprocessing Plants," and the applicable quality assurance related regulatory guides listed in SRP Section 17.1.

17.2 Organization

The organization responsible for the design and design verification of RESAR SP/90 is the Energy Systems Business Unit (ESBU) of the Westinghouse Energy and Utility Systems Group as shown in Figure 17.1 of this SER. The autnority and responsibility of each division in the ESBU is established by the Vice President and General Manager of the unit. The Vice President and General Manager also assigns responsibility for the establishment of the ESBU quality assurance plan and for assuring the effective implementation of the commitments of WCAP-8370 within the unit. Statements of activity, scope, and function are issued for each division in the unit. Required documentation of specific organizational detail, including the authority and responsibilities of organizations performing activities covered by WCAP-8370, is established and maintained.

The ESBU and each subtier division general manager, as applicable, is responsible for establishing and implementing a quality assurance program that meets the requirements of WCAP-8370. Responsibility for documenting the quality assurance program is assigned to a Quality Assurance (or similar title) Manager. Figure 17.2 shows a typical Westinghouse divisional organization for quality assurance. Regardless of variation to Figure 17.2 the Quality Assurance Manager always has direct access to the General Manager for resolution of quality problems. As indicated in Figure 17.2, each Quality Assurance Manager is sufficiently free from direct pressure for cost/schedule and has the authority to stop unsatisfactory work or otherwise control further processing, delivery, or installation of nonconforming materials. The Quality Assurance Manager has access to higher management levels to assure the ability to identify quality problems; initiate, recommend, or provide solutions through designated channels; and verify implementation of solutions.

Functional organizational responsibilities are described below.

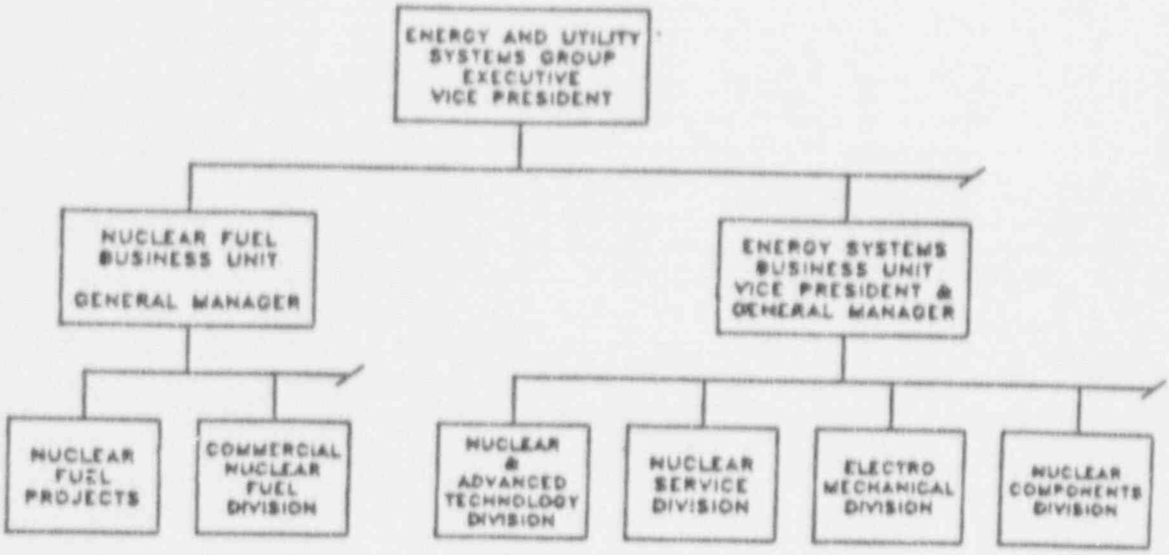

Figure 17.1 Westinghouse Energy and Utility Systems Group organization

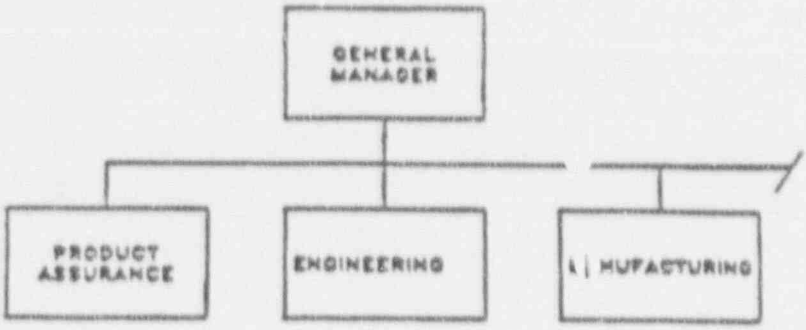

Figure 17.2 Typical division organization for quality assurance

Engineering groups within the ESBU are responsible for performing the various technical functions associated with the design and specification of the SP/90 and for technical follow-up of the remainder of the design cycle. These engineering groups also are responsible for providing safety analyses including safety evaluations of system and equipment design and safety performance criteria.

Manufacturing groups are generally responsible for the manufacture, fabrication, construction, testing and/or servicing of hardware. Thus, they are not directly involved in the RESAR SP/90 design.

Quality assurance groups are responsible for performing verification of the implementation of the quality assurance program and reporting the degree of compliance to Westinghouse management and to ensure that the quality assurance program is established and effective. Quality assurance activities for the SP/90 application include, as applicable, review of drawings, specifications, and procedures; surveillance and audits of suppliers; schedule of and participation in internal audits; and development and maintenance of specific quality assurance program documents.

Future organizational changes will be reported to the staff under the requirements of WCAP 8370/7800, Section 17.1.1 (last notification made January 25, 1990 by Westinghouse Letter NS-NRC-90-3487).

17.3 Quality Assurance Program

Westinghouse has provided a cross-reference to identify the manuals that implement each of the criteria of 10 CFR 50, Appendix B. The quality assurance program commits Westinghouse to meet the requirements of Appendix B to 10 CFR 50. Also, Westinghouse has committed to comply with the regulatory positions of applicable NRC regulatory guides and ANSI standards listed in Table 17.1, with some exceptions and clarifications as described in WCAP-8370, which the staff has found acceptable. The quality assurance program applies to all safety-related items and services engineered, procured, and manufactured by Westinghouse, including nuclear fuel assemblies. Westinghouse has indicated that it will apply applicable 10 CFR 50, Appendix B, quality assurance criteria to structures, systems, and components important to safety, recognizing that important-to-safety items include equipment that has been historically classified as non-nuclear safety. The staff will require clarification of this intent during the review of an FDA application or a plant-specific application. Highlights of the quality assurance program are described below.

Procedures require formal training and indoctrination of personnel performing activities affecting quality to ensure they will be suitably trained and their proficiency will be maintained.

The quality assurance program provides a system for design control that is documented and controlled by procedures and instructions. These procedures and instructions describe the responsibilities and interfaces of each organizational unit that has an assigned responsibility. Distribution lists and master lists of project drawings and specifications will be maintained to ensure timely and accurate access to latest applicable documents. Procedures will be established for verifying designs.

Table 17.1 Regulatory guidance applicable to the quality assurance program

Regulatory Guide	Title	Revision, date
1.26	Quality Group Classifications, and Standards for Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants	Rev. 3, 2/76
1.28	Quality Assurance Program Requirements (Design and Construction)	Rev. 3, 8/85
1.29	Seismic Design Classification	Rev. 3, 9/78
1.30	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electrical Equipment	Rev. 0 8/11/72
1.37	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants	Rev. 0, 3/16/73
1.38	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants	Rev. 2, 5/77
1.39	Housekeeping Requirements for Water-Cooled Nuclear Power Plants	Rev. 2, 9/77
1.58	Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel	Rev. 1, 9/80
1.64	Quality Assurance Requirements for the Design of Nuclear Power Plants	Rev. 2, 6/76
1.74	Quality Assurance Terms and Definitions	Rev. 0, 2/74
1.88	Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records	Rev. 2, 10/76
1.123	Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants	Rev. 1, 7/77
1.144	Auditing of Quality Assurance Programs for Nuclear Power Plants	Rev. 1, 8/80
1.146	Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants	Rev. 0, 9/80

Westinghouse has established and documented measures for the preparation, review, approval, and control of procurement documents. These measures provide assurance that the procurement documents will include or reference regulatory requirements, design bases, and quality requirements. Westinghouse quality assurance organizations will review and approve purchase specifications before they are issued. These reviews by qualified engineering and quality assurance personnel will provide assurance that quality requirements are complete and correctly stated. The reviews also will ensure that the quality requirements can be controlled by the supplier/manufacturer and verified by Westinghouse quality assurance personnel. Westinghouse requires that its suppliers/manufacturers will identify and control items that they supply, and Westinghouse quality assurance personnel will inspect the marking of items before shipment.

Westinghouse requires that in-process and final inspections will be performed in accordance with procedures submitted to and found acceptable by Westinghouse. Procedures require that inspection personnel will be qualified and that records of qualification will be maintained and that inspection personnel will be organizationally independent from personnel who perform the work being inspected.

Westinghouse suppliers/manufactures will be required to maintain a system providing for identification, documentation, and control of nonconforming items to prevent inadvertent use. Westinghouse quality assurance personnel will review and approve nonconformance actions. Westinghouse engineering personnel will evaluate and disposition nonconformances, and Westinghouse quality assurance personnel will review these actions. Westinghouse quality assurance personnel also will verify proper corrective action.

Westinghouse will execute a comprehensive system of planned and documented audits to verify product quality and compliance with the quality assurance program. The audits, with pre-established check lists, will ensure compliance with all aspects of 10 CFR 50, Appendix B, including the quality-related aspects of design, procurement, manufacture, storage, shipment, and reactor site activities. The Westinghouse quality assurance program requires that suppliers/manufacturers also will audit their own and their subvendor's operations to verify conformance with quality requirements. The audits will include quality-related practices, procedures, instructions, and conformance with the quality assurance program.

Westinghouse quality assurance personnel will conduct audits of the Westinghouse suppliers/manufacturers and selected subvendors. Written reports will be forwarded to management of the area audited and to Westinghouse management. Follow-up audits will ensure corrective action.

17.4 Conclusion

The staff concludes that the Westinghouse quality assurance program description is in compliance with the current applicable NRC regulations. Further, the quality assurance organizations are structured such that they can effectively carry out their responsibilities related to quality without undue influence from other groups. With the applicant's commitment to meet applicable regulatory requirements, the staff finds this program acceptable.

18 CONTROL ROOM DESIGN/HUMAN FACTORS ENGINEERING (RESAR SP/90 Module 15, Sections 7.8, 18.2, 18.3, and 18.4)

18.1 Introduction

Module 15, "RESAR SP/90 Control Room/Human Factors Engineering," describes the analytical and design processes being used to support the design of the control room for RESAR SP/90. The staff reviewed the submittal against the requirements of 10 CFR 50.34(f)(2), sections (iii), (iv), and (v), and SRP Section 18.1. Meetings also were held on September 11, 1986, and November 19, 1987, to discuss various aspects of the design process. Because of the stage of the design of RESAR SP/90, the scope of the review was limited to the main control room design and did not include remote shutdown panels or local control stations outside the main control room.

18.2 Evaluation

The purpose of the staff's review was to verify that Westinghouse has (1) appropriately incorporated accepted design processes and human factors engineering principles and criteria during its preliminary design phase and (2) performed comprehensive systems analysis to develop the design of the control room. Since this evaluation is based on the preliminary design of the Westinghouse advanced control room, the staff's conclusions are not final and will be reviewed further during the FDA stage and the plant-specific licensing process.

18.2.1 Establishment of a Qualified Multidisciplinary Design Team

The Westinghouse RESAR SP/90 control room design team includes experts in the fields of plant systems, instrumentation, human factors, operations, artificial intelligence, and computer systems design. The staff finds the composition of the design team acceptable.

18.2.2 Analytical Basis for Design of the Control Room

The Westinghouse design team performed numerous analyses to develop design requirements for the RESAR SP/90 control room. These analyses address both the plant process/safety functions and operator functions which are broken down to detailed operator information and action requirements that will later provide the basis for specifying the displays, controls, and layout of a RESAR SP/90 control room. Because the analytical methods and results are proprietary, they are not described here. However, the staff finds that the methods used by the Westinghouse design team follow the preferred top-down systems/function/task analysis recommended by the staff in NUREG-0700, "Guidelines for Control Room Design Reviews."

The applicant conducted a system analysis, using acceptable methods to identify man/machine interface requirements, including allocation of functions to man and machine (manual and automatic) and identification of information and controls to be provided to the operators. The applicant demonstrated that all the necessary information and controls needed for normal, abnormal, and emergency operation of the plant will be identified and provided.

The applicant further demonstrated that the design of the control room will comply with accepted human factors engineering principles and will address:

- control room work space
- workspace environment
- annunciator warning systems
- controls
- visual displays
- process computers
- panel layout
- control display integration

In addition the analyses take advantage of existing research in operator decision-making models to optimize the man/machine interface. Application of the results should enhance the operational environment significantly. Further analysis will be needed to finalize and specify the control room design. However, at this stage of design, the staff concludes that Westinghouse has applied analytical techniques that reflect the state of the art in human factors analysis. When the results of these analyses are applied during the FDA and plant-specific stages of design, they should result in a design that easily allows operators to control plant processes and monitor the safety status of the plant and the operability of safety systems.

18.2.3 Human Factors Criteria

Westinghouse has committed to using the criteria and guidance of NUREG-0700, where applicable. In cases where advanced technology has overtaken the guidance of NUREG-0700, the design team has referenced other acceptable sources of guidance. In cases where little or no guidance exists, such as comparative error rates between new forms of controllers (e.g., touch panel vs. track ball), the design team should make a defensible judgment based on existing research/experience and the results of their in-house test programs and design verification and validation (V&V) program.

The staff concludes that Westinghouse is incorporating acceptable human factors criteria in the design of the RESAR SP/90 control room. Documentation of the design and analytical processes used to support certain design choices will be audited during the FDA stage of review.

18.3 Conclusion

The staff concludes that the preliminary design analysis submitted by Westinghouse in support of the design of the RESAR SP/90 control room is acceptable for PDA.

19 PROBABILISTIC RISK ANALYSIS (RESAR SP/90 Module 16)

19.1 Introduction

Westinghouse submitted its "Probabilistic Safety Study," and amendments (referred to herein as the probabilistic risk assessment [PRA]) between June 28, 1985, and October 8, 1987. The SP/90 PRA is a level 3 PRA; that is, it includes core damage sequence analyses, applicable containment response analyses, and a representative offsite consequence analysis. Volumes 1 and 2 of Module 16 contain the core melt accident analysis of the RESAR SP/90 design. The mean core melt frequency was calculated for accidents resulting from internally initiated events. Volumes 3 and 4 of Module 16 contain the core melt and containment response analysis, offsite consequence analysis, and plant risk and uncertainty analysis. At this time, the core damage sequence analysis is limited to transients and loss-of-coolant accidents; it excludes external events such as internal flooding, fire seismic events, and high winds.

The overall objectives of the review of SP/90 PRA were to

- assess the reasonableness of core damage frequency estimates by identifying major optimisms and pessimisms with respect to assumptions made, data used, and major events omitted

- identify dominant sequences and contributors to core damage

- assess the Westinghouse analyses of the core melt accident and the containment response and obtain insights into the strengths and weaknesses of the proposed containment structure for early and late failure

- provide an estimate of the source term and risk of the RESAR SP/90 design and an evaluation of the reasonableness of the risk estimates provided by Westinghouse for two representative sites

As part of the review process, the NRC staff contracted Brookhaven National Laboratory (BNL) to perform a technical review of the RESAR SP/90 PRA. The BNL review considered core damage sequence analyses and the containment response and offsite consequence analyses resulting from internal events only. The staff sent requests for additional information to the applicant and conducted meetings with Westinghouse to discuss and clarify review-related information. The staff and BNL have evaluated the information obtained during these meetings and the formal responses to the staff questions as part of their review.

Results of the BNL review are documented in NUREG/CR-5177, "A Review of the WAPWR/SP/90 Probabilistic Safety Study," September 1988. Volume 1 covers the review of the core melt accident analysis and Volume 2 covers the review of the containment and risk portion of the PRA.

Contrary to the standard NRC licensing evaluations that compare the plant design and its transient or accident response to deterministic criteria described in the Standard Review Plan, this review is a technical assessment of the assumptions and methods used in the RESAR SP/90 PRA as well as a re-evaluation of the SP/90 PRA results through independent studies using alternative methods. The staff based its evaluation on the findings of the BNL review. The staff evaluation is documented in two parts; Sections 19.3 and 19.4 cover the core melt accident analysis (front-end of PRA) and the containment consequence analysis (back-end of PRA), respectively. A list of PRA issues that have not been resolved is presented in Section 19.6. However, at this stage of the design, these issues should not prevent issuance of the PDA.

19.2 Unique Features of the RESAR SP/90 Design

RESAR SP/90 is a pressurized light-water reactor (PWR) rated at a nominal 3800 MWt. The nuclear steam supply system (NSSS) is a Westinghouse four-loop design that will have a core containing 193 fuel assemblies with a 19x19 fuel matrix. The containment building is a large, dry spherical design with a steel shell containment structure and an outer concrete wall separated by an annular volume. A comparison of major design characteristics between the RESAR SP/90 and six other PWR plants is given in Table 19.1 of this SER. The six other plants are all Westinghouse-designed PWR reactors with large, dry (atmospheric or subatmospheric) containments. Only the South Texas plant has a thermal power equivalent to that of the RESAR SP/90 design. It appears that there are several major design differences between RESAR SP/90 and conventional PWRs that could influence the overall plant response to severe accident conditions. These are discussed briefly in the following sections.

The RESAR SP/90 is designed with a large emergency water storage tank (EWST) located within the containment building. This water supply will offer two advantages under severe accident conditions. First, the pressurizer relief tank (PRT) blowdown will be directed into the EWST. Most PWR plants discharge directly to the containment atmosphere. Thus, for sequences with total loss of onsite and offsite power, the PRT discharge will not usually pressurize the RESAR SP/90 containment as quickly as it would the containments of other large, dry PWRs. The capacity of the EWST will be comparable with a typical BWR Mark III containment suppression pool. Like a BWR suppression pool, the EWST will function as an effective energy and fission product removal system for transient-induced accident sequences. The second advantage is that the emergency core cooling system (ECCS) for the RESAR SP/90 design will always take suction from the EWST, so the EWST, in effect, also will function as the containment sump. As such, contributions to core melt frequency caused by the failure to switch from injection to recirculation mode have been eliminated by the design. In addition to the accumulators, which are common to all PWRs, RESAR SP/90 also incorporates four core-reflood tanks (CRTs) in its design, which will passively inject directly into the cold legs of the reactor vessel. The combination of accumulators and CRTs, when actuated, will significantly increase the water inventory available for emergency core cooling. The large amount of water inventory in the reactor vessel will delay the time of core uncovery in, for example, small LOCAs. In the event of reactor vessel failure at high pressure (transients and small-break LOCA scenarios), the discharge of all accumulators and CRTs after vessel failure and depressurization will affect the corium/concrete interaction in the reactor cavity region because of the large quantity of water on the cavity floor.

The RESAR SP/90 core design is larger in diameter, contains more fuel rods (19x19 fuel array), and has a greater mass of fuel and cladding material (UO_2

Table 19.1 A comparison of proposed RESAR SP/90 design characteristics with other plants[a]

Design Parameters	Zion-1/2	Indian Point-2	Indian Point-3	Surry	Millstone-3	South Texas	RESAR SP/90
Thermal Power (MWt)	3,250	3,030	2,760	2,441	3,411	3,893	3,800
Containment Building:							
free volume (ft^3)	2.72E+6	2.61E+6	2.61E+6	1.8E+6	2.3E+6	3.41E+6	3.09E+6
design pressure (psia)	62	62	62	60	60	72	60
operating pressure (psia)	15	14.7	14.7	9/11	9.1/12.7	15	15
operating temperature (°F)	120	120	120	80/120	80/120	122	100
Primary System:							
volume water (ft^3)	12,710	11,347	11,347	9,167	11,671	12,700	16,845
volume steam (ft^3)	720	720	720	520	720	720	1,000
mass of UO_2 in core (lb)	216,600	216,600	215,800	175,600	222,739	259,860	298,200
mass of steel in core (lb)	21,000	20,407	20,407	36,300	--	--	--
mass of Zr in core (lb)	44,500	44,600	41,993	36,300	45,296	54,840	84,063
mass of bottom head (lb)	87,000	78,130	78,130	87,000	87,000	77,000	--
bottom head diameter (ft)	14.4	14.7	14.7	14.4	14.4	14.4	16.66
bottom head thickness (ft)	0.45	0.44	0.44	0.42	0.45	0.45	0.56
High-Head ECC Pumps:							
total flow rate (gpm)	800 (2 pumps)	1,200 (3 pumps)	1,200 (3 pumps)	--	850 (2 pumps)	1,600 (3 pumps)	4,000 (4 pumps)
Accumulator Tanks:							
total mass of water (lb)	208,000	173,000	173,000	173,000	348,000	225,000	434,000
initial pressure (psia)	665	665	665	665	600	715	615
temperature (°F)	150	150	150	120	80	120	100-150
Refueling Water Storage Tank:							
mass (lb)	2.89E+6	2.89E+6	2.89E+6	2.89E+6	1.0E+7	4.2E+6	4.9E+6[b]
initial pressure (psia)	14.7	14.7	14.7	14.7	14.7	14.7	14.7
Reactor Cavity:							
configuration	flooded	flooded	flooded	dry	dry	dry/flooded	dry/flooded
concrete material	limestone	basaltic	basaltic	basaltic	basaltic	basaltic	limestone/ basaltic[c]

Notes at end of table.

Table 19.1 (Continued)

Design Parameters	Zion-1/2	Indian Point-2	Indian Point-3	Surry	Millstone-3	South Texas	RESAR SP/90
Core Reflood Tank:							
total mass (lb)	--	--	--	--	--	--	347,000
initial pressure (psia)	--	--	--	--	--	--	215
Low-Head ECC Pumps:							
total flow rate (gpm) (design)	6,000 (2 pumps)	6,000 (2 pumps)	6,000 (2 pumps)	6,000 (2 pumps)	8,000 (2 pumps)	8,700 (3 pumps)	7,000 (2 pumps)
containment spray (CS) design flow rate (gpm)	9,000[d] (3 pumps)	5,200 (2 pumps)	5,200 (2 pumps)	7,000[e] (4 pumps)	4,000/3,950[f] (6 pumps)	2,200[g] (3 pumps)	3,000[h] (2 pumps)
containment set point of CS initiation (psia)	38	38	38	38	40	40	40
Containment Building Fan Coolers:							
total capacity (Btu/hr)	4.05E+8	3.8E+8	3.8E+8	NA(9)	NA[i]	5.70E+8[j]	--

[a]Source: NUREG/CR-5177, Volumes 1 and 2, September 1988.

[b]Emergency water storage tank inside containment.

[c]Site specific.

[d]Additional CS pump driven by dedicated diesel.

[e]Two pumps inside and two pumps outside containment.

[f]Two quench spray pumps (4,000 gpm one pump, 6,000 gpm both pumps). Four recirculation spray pumps (outside containment).

[g]Per pump (three pumps).

[h]Per pump (four pumps available, two pumps are required).

[i]Not safety related.

[j]For six fans, however, three out of six is sufficient for design basis.

and Zr) than the conventional 17x17 fuel assembly design. The additional fuel loading will result in a significant reduction in average power density. The lower average power density will reduce the rate of fuel temperature increase during the early core heatup. However, the increases of total fuel surface area and mass of zirconium cladding have the potential for enhancing the metal/steam reaction and increasing the total hydrogen production during the later stages of core heatup. In the event of vessel failure, the large mass of core material will influence the potential corium/concrete interactions and the direct containment heating mode of containment failure.

The RESAR SP/90 reactor vessel design is similar to other conventional PWRs except for differences in the design of the region from the upper core plate to the outlet nozzle. In the RESAR SP/90 design, the thickness of the upper core plate has been increased, more control elements are used, and there is a new calandria structure in the upper-plenum region. The new design not only increases the internal structural mass in the upper plenum, which could influence the transport of fission products in the event of a core meltdown, but also significantly increases the water inventory in the reactor vessel. The larger water inventory would delay the time of core uncovery and thus the time at which the core might melt and fail the reactor vessel.

There is a larger heat capacity in the RESAR SP/90 containment design because of the large increase in the containment volume and surface area. In addition, the steel liner, concrete, and equipment provide significant heat sinks. Comparison with the Zion containment shows that RESAR SP/90 has a 41-percent increase in heat capacity and a 75-percent increase in surface area, which is mainly because of the presence of additional equipment. In addition, the thermal power of RESAR SP/90 is only 17 percent higher than that of Zion.

19.3 Core Melt Accident Analysis

The staff reviewed the core melt accident analysis (front-end of PRA) of the RESAR SP/90 design which is contained in Volumes 1 and 2 of the Westinghouse Module 16 submittal (Sections 1 through 4). The staff's findings are based on the BNL technical review (NUREG/CR-5177).

(1) The total core damage frequency in the SP/90 PRA is about 2.0E-6 per reactor year; the corresponding BNL estimate is about 6.0E-6 per reactor year.

(2) The difference in these two estimates is only a factor of 3. Such a difference is not greatly significant considering the level of uncertainty around the estimates. BNL, however, performed sensitivity analyses as part of the review process, using critical modeling assumptions and core cooling success criteria. The results show that the core damage frequency contribution from internal events should not exceed an upper bound of about 2E-5 per reactor year.

(3) The review process consisted of identifying optimisms and pessimisms with respect to data and modeling assumptions and identifying any omitted events or sequences. Some of the omissions identified by the review include the following:

- dependency of the integrated protection system (IPS)

- common-mode failures of the air-operated valves in the emergency feedwater system (EFW)

- inclusion of the EFW system failures following a steam generator tube rupture (SGTR) event combined with a failure to scram

BNL has not identified any major pessimism that could reduce the reported frequency estimates in the PRA. This aspect of the review was completed carefully, considering the increased level of redundancy in safety systems and some unique design features applicable to the RESAR SP/90 design. However, BNL did identify many of the optimisms and corrected them in its evaluation. Some of these corrections include the following:

- revised the failure probability to trip the reactor following the failure-to-scram events

- revised the support system failure probability for certain anticipated transient without scram (ATWS) events

- revised treatment of the probability for the support-state dependencies for various initiating events

Table 19.2 shows the effect of these changes on individual sequence frequencies and frequency estimates. Table 19.3 itemizes significant changes made to the system unavailability quantification process.

(4) Table 19.2 also shows the 15 dominant sequences extracted from the BNL review. These 15 sequences contributed more than 90 percent of the total core damage frequency. About 48 percent of the total core damage frequency results from one single sequence. This sequence is basically a transient involving the total loss of IPS and failure to recover the system within 3 hours. The sequence contribution to total core damage frequency in the original SP/90 PRA is either not accounted for or not modeled explicitly. The second BNL sequence is an ATWS followed by failure to provide manual trip in a timely fashion and failure to provide long-term shutdown capability. The frequency estimate is about 6E-7 per reactor year. The SP/90 PRA also modeled such a sequence, but on the basis of its review, BNL revised the PRA frequency estimate upward. The third BNL sequence is a loss of offsite power followed by failure of the onsite ac power system, which results in a seal LOCA and failure to recover onsite ac power in 3 hours. The frequency estimate is about 5E-7 per reactor year. The SP/90 PRA also modeled such a sequence and produced an estimate very close to the BNL estimate. Although the RESAR SP/90 design features include a redundant backup seal cooling system, the sequence involving failure of the ac power system and seal failures contributes moderately (about 9 percent) to the total core damage frequency. The other 12 sequences shown in Table 19.2 seem to be uniformly distributed with respect to the frequency estimates. The SP/90 PRA has given similar results.

(5) The interfacing LOCA contribution to the total core damage frequency is negligible for the RESAR SP/90 design. The frequency estimate is about 4E-8 per year, well below the typical estimate for the "Reactor Safety Study" (RSS, NUREG-75-014 [a.k.a. WASH-1400], 1975). This low contribution

Table 19.2 Frequency of dominant sequences for RESAR SP/90 design

No.	Sequence description	Mean frequency estimate per year[a]	See notes
1.	A transient followed by a failure of IPS signal and failure to recover within 3 hours. (TR75)	3E-6	b
2.	An ATWS event followed by a failure of the manual reactor trip and failure to provide long-term shutdown capability. The primary pressure-relief, secondary cooling, and containment heat removal functions are successful. (ATW 15)	6E-7	c
3.	A total loss of ac power followed by reactor coolant pump (RCP) seal failures and failure to recover ac power within 3 hours. The secondary cooling is successful. (LSP 61)[d]	5E-7	e,f
4.	A total loss of ac power followed by failure of secondary cooling and failure to recover ac power within 3 hours. The secondary cooling and containment heat removal functions are successful. (ATW 19)	3E-7	e
5.	An ATWS event followed by failure to manually trip the reactor and failure to relieve primary system pressure. The secondary cooling and containment heat removal functions are successful. (ATW 19)	3E-7	c
6.	An SGTR event followed by failure to scram automatically and operator failure to manually trip the reactor. (ATW 54)[g]	3E-7	c,h
7.	A loss of auxiliary cooling with a subsequent RCP seal failure or a consequence LOCA due to stuck-open safety-relief valves (SRVs). The secondary cooling function is successful. (LC 11)	2E-7	f
8.	An ATWS event followed by a failure of the operator to manually trip the reactor and a loss of the secondary cooling system. The containment heat removal function is successful. (ATW 23)	2E-7	c,i

Notes at end of table.

Table 19.2 (Continued)

No.	Sequence description	Mean frequency estimate per year[a]	See notes
9.	A loss of the auxiliary cooling system followed by loss of the secondary cooling function. (LC 12)	1E-7	i
10.	A postulated vessel rupture event. The containment heat removal function is successful. (VEF 1)	1E-7	j
11.	A loss-of-offsite power event followed by loss of the service water/component cooling water (SW/CCW) systems and RCP seal failures or a consequential LOCA resulting from a stuck-open SRV. The secondary cooling is assumed to be successful. (LSP 77)	8E-8	e,f
12.	A transient followed by the loss of SW/CCW systems and RCP seal failures or a consequential LOCA resulting from stuck-open SRVs. The secondary cooling is assumed to be successful. (TR 77)	7E-8	e,f
13.	A loss-of-offsite power event followed by the loss of SW/CCW and failure of secondary cooling with an intact primary system. (LSP 78)	5E-8	e
14.	A transient followed by the loss of SW/CCW and failure of secondary cooling, with an intact primary system. (TR 78)	4E-8	e
15.	A loss-of-offsite power event followed by a loss of IPS signal and fail to recover the IPS within 3 hours. (LSP 75)	3E-8	b

[a] The frequency estimates correspond to the estimates requantified by the BNL review.

[b] An added sequence in the BNL review to account for the dependency of the IPS failures.

[c] The frequency estimate accounts for the increased probability of operator failure to scram the reactor.

[d] The sequence also consists of failure contributions resulting from a stuck-open SRV.

[e] The frequency estimate accounts for the revised probability of support systems, depending on the type of initiating event.

Table 19.2 (Continued)

[f]The frequency estimate accounts for the revised probability estimate for the consequential LOCA.

[g]The sequence also consists of failure contributions resulting from a secondary-side break.

[h]An added sequence in the BNL review to account for the methodological difference (consideration of the initiating events).

[i]The frequency estimate accounts for the revised probability estimates to account for omitted faults in the EFW system.

[j]The frequency estimate is the same as that of the "Reactor Safety Study" (NUREG-75-014, 1975).

Table 19.3 Changes to system modeling made as part of BNL review

System	Changes made to fault trees
Onsite ac power	Added dependency of dc power Added dependency of service water/ component cooling water (SW/CCW) systems
Service water/component cooling water	Added common-cause failures to start four pumps during loss-of-ac-power event Added dependency of integrated protection system (IPS)
High-head safety injection for small and large LOCA events	Added SW/CCW cooling failures Added dependency of IPS Added dependency of ac and dc system interfaces with IPS and SW/CCW systems
Emergency feedwater system:	
All transients	Added IPS dependence Added dependency of ac and dc for the emergency feedwater system pumps
Steam generator tube rupture (SGTR) event	Revised system logic to account for check valve failures in cross-connect lines
Secondary-side break event	Same as SGTR event above
Containment fan cooling	Added common-cause contribution to account for start and run failures of four fan coolers
Containment spray	Added dependency of IPS Added dependency of ac and dc power with other interfacing systems, SW/CCW
Feed-and-bleed cooling	Added hardware failures and maintenance availability of high-head safety injection system during small LOCA events Added hardware failures of power-operated relief valve and block valve

is primarily the result of a unique feature of the RESAR SP/90 design: the EWST needed for the suction of low- and high-pressure system pumps is located inside the containment. Because of this arrangement, the rupture of low-pressure system piping (4-inch residual heat removal suction-side motor-operated valve failures) would be more likely to result in a LOCA inside the containment, and the probability of this event has been treated accordingly by the SP/90 PRA and BNL.

(6) The contribution of LOCA-induced accidents to total core damage frequency also is lower than the corresponding estimates of the RSS. This is primarily because of the credit taken for the fourth high-head safety injection pump, low-pressure injection pump, and added second turbine-driven pump.

Although the common-cause failures have been treated in the SP/90 PRA, the common-cause contribution resulting from steam binding of the emergency feedwater system is not addressed explicitly.

(7) The frequency contribution of SGTR events to total core damage frequency is small (less than 5 percent), and the frequency estimate is about 3E-7 per reactor year. The secondary-side atmospheric dump valves are equipped with block valves to isolate the break flow and unwanted radioactive release. The BNL estimate of 3E-7 per year for SGTR events is the result of modeling aspects that were not adequately considered in the SP/90 PRA. The identified sequence is basically an SGTR event followed by the failure of automatic scram and a failure of the operator to manually trip. The contribution of such a scenario may not increase the total core damage frequency significantly but could increase the risk of early fatality.

(8) The frequency contribution of reactor pressure vessel failure is about the same as estimated in the RSS study, about 1E-7 per year. However, the pressure vessel of the RESAR SP/90 design will have significant design improvements to reduce the probability of vessel failure as a result of pressurized thermal shock phenomena.

(9) The failure contribution resulting from secondary-side steam line break events was considered, along with isolation and mitigation capabilities. The frequency contribution resulting from steam line break events is estimated to be about 2E-8 per year by BNL, and about 1E-10 per year by the SP/90 PRA. BNL's increased frequency estimate comes from the revised treatment of main steam isolation valve (MSIV) failures that derives from other PRAs reviewed by the staff. Both the applicant's and BNL's frequency estimates are based on a value of 8.2E-4 per demand, but this MSIV unavailability estimate may be questionable, considering the existing qualification requirements for MSIVs to close following postulated large-break events. Recent operating experience seems to indicate a higher MSIV failure-to-close probability than the estimate reported in the SP/90 PRA. The main steam isolation features of the RESAR SP/90 design are not unique. Thus, the applicant should reassess the basis for the estimate of 8.2E-4 per demand for the MSIV failure-to-close event following a break outside the containment using available information from other plants.

(10) The BNL review also examined sequences with very low frequencies (below 1.0E-8 per year) for the appropriateness of modeling assumptions and support system availability, particularly for ATWS and SGTR events. The

review revised the frequency estimates upward by at least 2 orders of magnitude. At this PDA stage, the details of the support systems are, to a great extent, not available. As the design details of these support systems become available, the FDA review should consider the detailed information and obtain revised frequency estimates as appropriate.

The staff concurs with the BNL review results except as noted above. The staff further agrees with BNL that additional information is needed to better characterize the accident frequency applicable to the RESAR SP/90 design, certain critical design-related information that affects the quantification of core damage frequency results is needed in the future, as it evolves. These modeling issues and additional information needed include the following:

- treatment of load-sequencing failures following the loss-of-ac-power event and ESF failures
- treatment of subtle ac and dc interactions and their impact on safety systems, particularly during the LOCA sequences
- estimate of mission times for diesels and their auxiliaries (sequence dependent)
- details of unavailability analyses of the IPS, as it exists/evolves (event dependent)
- adequacy of modeling of various loads connected to all dc buses
- treatment of loss-of-air scenarios and impact of loss of air on auxiliary cooling and secondary cooling systems and resulting sequences
- details of IPS and ESF
- details of ac and dc load distribution and control information
- ESF single failure analyses
- air system details (including air pressure decay behavior for recovery analyses)
- details of diesel generator and auxiliaries
- EFW check valve details
- scoping-type external event analyses, as they evolve

The staff concludes that additional modeling details of the above systems will enhance full understanding of the strength and weaknesses of the RESAR SP/90 design. Alternatively, the applicant could consider these comments and update the existing version of the PRA, as the design evolves into the FDA stage of review.

19.4 Core and Containment Consequence Analysis

This section discusses the review of the core and containment analyses, consequence analysis and plant risk assessment (back-end of PRA). These areas are contained in Volumes 3 and 4 of the Westinghouse Module 16 submittal (Sections 5 through 7). The staff's findings are based on the BNL technical review (NUREG/CR-5177).

19.4.1 High-Pressure Molten-Core Ejection

An important phenomenological issue has been identified in the past several years, which is referred to as high-pressure molten-core ejection. This occurs when the reactor vessel is breached at a reactor pressure of several hundred pounds per square inch or higher and the molten-core debris is expelled from the vessel, forming hot, high-velocity aerosols. In this circumstance, energy may be transferred very efficiently from the small corium particles directly to the containment (direct containment heating). The energy transfer mechanism is extremely complex, and there are additional potential energy sources provided by the oxidation of available metallic zirconium and the combustion of hydrogen.

This issue was studied as part of the NRC severe-accident research programs, the Industry Degraded-Core Rulemaking (IDCOR) program, and other programs. It has been one of the key areas of NRC research. Experimental evidence to date does not provide a basis for neglecting this effect, which could occur when the core melts in a pressurized vessel.

In addition, the staff has identified a high-pressure core melt scenario that postulates SGTR before the core melts through the reactor pressure vessel (RPV) bottom head. In this high-pressure core melt accident scenario, the potential exists for recirculation of hot steam, hydrogen, and aerosol flow inside the reactor primary system as a result of natural circulation. The establishment of natural circulation could lead to a significant temperature increase in the hot legs of the primary system, including the steam generators and pressurizer, and could potentially lead to SGTR events before the core melts through the reactor pressure vessel bottom head. Should the steam generator tubes fail, the scenario could lead to an early release of fission products directly to the atmosphere. Such a scenario also could depressurize the primary system sufficiently to preclude the direct heating effects postulated subsequent to the reactor pressure vessel bottom-head failure. The NRC and industry are continuing the study of the natural circulation issue following the high-pressure accident scenarios by means of tests and code development.

Westinghouse performed the containment loading analysis for RESAR SP/90 using the industry-developed modular accident analysis program (MAAP) computer code, which does not consider these phenomena. The MAAP code was not submitted to the staff as part of the SP/90 PRA review process (Docket 50-601), and consequently, BNL's review of the PRA used the source-term code package (STCP) and CONTAIN computer codes, which were developed for the NRC to perform similar calculations. During the past several years, the modeling of all these codes has been the subject of review as part of the NRC's interactions with the representatives of the IDCOR and Electric Power Research Institute (EPRI) program on the severe-accident risks of existing plants. During these interactions with IDCOR representatives, the staff identified technical areas, such as the high-pressure molten-core ejection and the natural circulation concerns discussed above, where MAAP calculations need to be supplemented by additional analysis. Although this area is still under study, the applicant should explore the possibility of modeling the natural-circulation scenario along with the direct heating effect for all high-pressure core melt accident scenarios for RESAR SP/90 to support the FDA application.

Because the radiological consequences of high-pressure molten-core ejection are considered more severe than those of the steam generator tube failure scenarios discussed above, BNL performed its evaluation considering the effects of direct containment heating. To provide a basis of comparison to support the staff's review of the SP/90 PRA, BNL also evaluated the design without including the effects of high-pressure molten-core accident scenarios. The results of these analyses are discussed throughout this report.

19.4.2 Containment Response to Severe Accidents

In the containment response analyses of the SP/90 PRA, the applicant used a simplified containment event tree (ARBRE computer code) to estimate the containment failure probabilities of various modes following a core damage accident. The staff concludes that the use of such a systematic method of estimating containment failure probability is reasonable and that the event tree is a good display tool that provides an integrated approach to combining various failures and successes of different containment mitigation features. BNL used basically the same approach in its evaluation, using the EVNTRE code to estimate the probability of various containment failure modes. BNL also made an effort to model the impact of the direct containment heating effect through detailed modeling of the containment event tree (CET), and it estimated the containment failure probability of various failure modes accounting for the direct containment heating effect. BNL's calculated frequency estimates and the definitions of severe accident sequence groups (various plant damage states) are provided in Table 19.4. Also, a summary of containment failure probabilities for various containment failure modes is provided in Table 19.5. As can be seen, the two studies agree within reason (considering the uncertainties involved) when direct containment heating is not included in the analysis although the BNL analysis does not give as high a weight for the case of no containment failure for the SEF sequence. For the TE, SE, and TEFC plant damage states with direct heating, the most likely outcome is early containment failure. The second most likely outcome is late containment failure. Tables 19.6 and 19.7 show peak containment pressures, temperatures and times of critical events for various plant damage states.

BNL estimated in 1987 that the probability of early containment failure (within an hour or so of vessel breach) is approximately 0.5-0.75, given a high-pressure molten-core scenario. When high-pressure molten-core ejection is not included in the evaluation, BNL predicts that the likelihood of early containment failure is very small. On the basis of the BNL estimates, the frequency of sequences that could lead to early containment failure (those at sufficiently high reactor pressure) is approximately 4E-6 per reactor year.

Because of the preliminary stage of the design of RESAR SP/90, neither Westinghouse nor BNL made a detailed assessment of the containment structural performance under severe accident conditions. The staff believes the applicant should perform and submit such an analysis during the FDA stage of review. The potential containment loading conditions resulting from high-pressure molten-core ejection should be considered in performing this analysis.

19.4.3 Transport of Radionuclides Following a Severe Accident

BNL did not evaluate the adequacy of the MAAP computer code with regard to the transport of fission products. However, BNL used the CONTAIN and STCP computer

Table 19.4 BNL definitions of severe accident sequence groups and estimated frequencies applicable to RESAR SP/90

Designator*	Characteristics of the severe-accident sequence group	Estimated frequency per reactor-year
TE	Transient followed by release of primary coolant inventory through the pressurizer safety and relief valves (S/RVs) to the EWST with possible early core melting	4E-6
TEFC	Same as TE with containment sprays and fan coolers operating	6E-7
SE	Small LOCAs including pump seal failures with possible early core melting	2E-7
SEFC	Small as SE with containment sprays and fan coolers operating	5E-7
SEF	Same as SE with only containment fan coolers operating	1E-8
AE	Large LOCAs, including pressure vessel failure with possible early core melting	1E-7
AEFC	Same as AE with containment fan coolers and sprays operating	1E-7
V	Interfacing LOCAs, including pipe breaks outside the containment with possible early core melting	4E-8
V2E	Steam generator tube rupture events with release of primary inventory through the secondary S/RVs with possible early core melting	3E-7
V2L	Steam generator tube rupture events with release of primary inventory through the secondary S/RVs with possible late core melting	1E-8

*Plant damage state.

Table 19.5 Containment failure modes and probability estimates for RESAR SP/90 design

	Plant damage state[a]							
Failure modes	TE	SE	AE	TEFC	SEFC	AEFC	SEF	V, V2E, V2L
1. Bypass failures	NA	NA	NA	NA	NA	NA	NA	1.0
NA[b] NA	NA	NA	NA	NA	NA	NA	NA	1.0
NA NA	NA	NA	NA	NA	NA	NA	NA	NA
2. Unisolated containment,	-[c]	-	NA	1.0E-4	1.0E-4	1.0E-4	NA	NA
spray operable	-	-	NA	2.5E-3	2.5E-3	2.5E-3	NA	NA
	-	-	NA	7.0E-4	2.0E-4	NA	NA	NA
3. Unisolated containment,	1.0E-4	1.0E-4	1.0E-4	-	-	-	1.0E-4	NA
spray inoperable	2.5E-3	2.5E-3	2.5E-3	5.0E-5	5.0E-5	5.0E-5	2.5E-3	NA
	7.0E-4	1.0E-4	NA	1.0E-5	5.0E-6	NA	3.0E-4	NA
4. Early failure, spray	-	-	NA	1.4E-4	1.4E-4	1.4E-4	NA	NA
operable	-	-	NA	-	2.2E-5	-	NA	NA
	NA	NA	NA	0.73	0.52	NA	NA	NA
5. Early failure, spray	2.4E-4	4.0E-4	1.0E-4	-	-	-	1.4E-4	NA
inoperable	1.0E-4	2.6E-4	1.0E-4	1.0E-4	1.2E-4	1.0E-4	1.0E-4	NA
	0.73	0.99	NA	NA	0.42	NA	0.89	NA
6. Late failure	0.50	0.50	0.51	5.0E-4	5.2E-4	5.0E-4	5.2E-4	NA
	0.75	0.83	0.81	0.01	0.01	0.01	0.37	NA
	0.20	0.01	NA	7.0E-3	1E-3	NA	0.08	NA
7. Base-mat failure	5.0E-3	5.0E-3	5E-3	1.0E-4	1.0E-4	1.0E-4	-	NA
	2.5E-3	1.6E-3	2E-3	0.01	7.7E-3	8E-3	4E-3	NA
	7.0E-4	2.0E-5	NA	3E-3	4.0E-4	NA	4E-6	NA
8. No failures	0.495	0.495	0.48	0.99	0.99	0.99	0.99	NA
	0.250	0.160	0.18	0.97	0.98	0.98	0.62	NA
	0.70	0.002	NA	0.26	0.06	NA	0.03	NA

[a]For descriptions of various plant damage states, refer to Table 19.4.

For each failure mode, the first row corresponds to the applicant's estimate; the second row corresponds to BNL's estimate without accounting for the direct heating effect mode of failure; and the third row corresponds to BNL's estimate including the direct heating effect mode of failure.

[b]NA = case is not applicable.

[c]- = insignificant.

Table 19.6 Containment loading analysis of RESAR SP/90 design

Plant damage state[a]	Computer code	Time[b] (hrs)	Peak containment[c] Pressure (PSI)	Temperature (°F)
TE	MAAP	44	56	400
	STCP	44	58	280
AE	MAAP	47	64	360
	STCP	47	110	315
AEFC	MAAP	N/A[d]	62	324
	STCP	N/A	N/A	N/A
SE	MAAP	38	93	708
	STCP	38	104	380
SEFC	MAAP	N/A	38	560
	STCP	N/A	N/A	N/A

[a]See Table 19.4 for description of various plant damage states.

[b]Following accident initiation.

[c]Estimates correspond to upper compartment bulk average temperature.

[d]N/A = not available.

Table 19.7 Times of critical events for TE and AE plant damage states RESAR SP 90

Event	Computer code	Time (minutes) TE	AE
Steam generator dryout	MAAP	141	129
	STCP	132	N/A[b]
Core uncovery	MAAP	194	141
	STCP	222	96
Onset of core melt	MAAP	N/A	N/A
	STCP	284	151
Support plate failure	MAAP	355	293
	STCP	327	199
Reactor pressure vessel failure	MAAP	356	294
	STCP	343	221[c]
Cavity dryout	MAAP	810	294
	STCP	1243[d]	282

[a]See Table 19.4 for descriptions of various plant damage states.
[b]N/A = not available
[c]In the STCP computer code, early prediction of reactor pressure vessel failure subsequent to support plate failure is also noteworthy.
[d]In the STCP computer code, late prediction of cavity dryout is due to the assumed heat transfer model in the CORCON-MOD 2 version of the MAAP computer code.

codes to independently estimate release fractions of fission products and compared its estimates with the corresponding estimates by MAAP, which may be found in NUREG/CR-5177. The CONTAIN and STCP estimates are given in Table 19.8 for the TE plant damage state (no failure, late failure, and early failure release modes). The following are the staff's general observations regarding these results:

- The ex-vessel release of barium, strontium, and "refractories" (the RSS ruthenium and lanthanum groups for MAAP and CONTAIN computer codes, ruthenium only for STCP) is at least three orders of magnitude lower in the applicant's analysis (MAAP) than in the CONTAIN or STCP computer runs. This is because MAAP predicts relatively little corium/concrete interaction, which acts as the driving force for such releases.

- Although the early release fractions for cesium iodine and cesium hydroxide in MAAP are somewhat lower than the estimates predicted by BNL's analysis, late release fractions for cesium iodine and cesium hydroxide are much higher in MAAP than in BNL's analysis because MAAP models cesium and iodine revaporization from the reactor coolant system (RCS) late in the core-melt-accident progression while the STCP does not.

BNL judged that differences in estimates of late release fractions predicted by the MAAP analyses are not significant to warrant including them in BNL's additional consequence calculations. However, BNL performed detailed calculations for various severe accident conditions to establish a set of representative source terms for the RESAR SP/90 design. To manage the large number of scenarios resulting from the severe accident analyses, BNL grouped the various end states of the containment that have similar radionuclide release characteristics into containment failure "bins." Each bin involves combinations of the accident sequence characteristics that have the greatest influence on source term behavior. Table 19.9 identifies the containment and system conditions that define these bins. Table 19.10 provides the results of BNL's calculations for various severe accident conditions.

BNL's reassessment of the overall source term and risk for RESAR SP/90 took into account the effect of the unmodeled physical processes that Westinghouse did not include in its evaluation.

19.4.4 Conditional Consequence and Risk Estimates

As part of the BNL review of the conditional consequence and risk estimates applicable to the RESAR SP/90 design, BNL used the MELCOR accident consequence code system (MACCS) computer code (Version 1.4) developed by the NRC as part of the Severe Accident Research Program. The applicant used a modified CRAC2 computer code to estimate the public risk, using the Salem site and associated meteorology. BNL did not perform a detailed uncertainty analysis of the severe accident consequence estimates as part of its review because that was beyond the review scope. Basically, BNL's consequence and risk estimates are point estimates. However, BNL performed a detailed parametric sensitivity study to include the effect of variations in release time, release duration, etc., on the conditional consequence estimates for important severe accident scenarios.

The results of these sensitivity studies are documented in Appendix E of Volume 2 of NUREG/CR-5177. The applicant did not perform and document such detailed

Table 19.8 Comparison of estimates of fission product release fractions to environment by CONTAIN and STCP computer codes for TE*

Fission product group	Early containment failure**	Late containment failure**		No containment failure**	
	CONTAIN	STCP	CONTAIN	STCP	CONTAIN
Xe, Kr	0.99	0.95	0.93	2.3(-3)	1.3(-3)
CsI	1.2(-1)	3.2(-4)	1.6(-5)	2.2(-5)	1.6(-5)
CsOH	7.8(-2)	2.0(-4)	1.1(-5)	1.4(-5)	1.1(-5)
Te	1.8(-1)	1.2(-2)	5.7(-3)	5.9(-5)	2.5(-5)
Ba, Sr	1.9(-1)	4.7(-4)	3.8(-4)	9.8(-5)	1.5(-5)
Refractory	1.3(-2)	9.7(-9)†	2.6(-5)†	3.7(-10)†	1.1(-6)†
Failure time (hr)	5.7	34	33	--	--
End of calculation (hr)	42	48	42	48	33

*See Table 19.4 for description of TE plant damage state.

**MAAP computer code estimates may be found in NUREG/CR-5177, September 1988.

†The difference between STCP and CONTAIN is due mainly to neglecting the lanthanum release in the STCP computer code calculations.

Table 19.9 Accident sequence characteristics of various containment failure bins applicable to the RESAR SP/90 design

Bin	Characteristics
1	Containment failure at or before the time of reactor-vessel breach with reactor coolant system (RCS) at high pressure (prior to vessel breach) and with the containment spray system failing to operate through the period of core/concrete interactions. The failure modes include early burning of hydrogen generated during the core degradation process; failure at the time of vessel breach as a result of the steam spike generated when the core debris is discharged into the water in the cavity or as a result of an in-vessel steam explosion; or failure as a result of burning of the hydrogen released at the time of vessel breach.
2	Early failure of containment, as in bin 1, but with the containment spray system operating through the period of core/concrete interactions.
3	Containment rupture at or before the time of vessel breach with the RCS not at high pressure and with the containment spray system not operating through the period of core/concrete interactions. The failure causes are similar to those in bin 1.
4	Early failure of containment, as in bin 3, but with the containment spray system operating through the period of core/concrete interactions.
5	Failure of containment before core damage as a result of failure to remove heat from containment during operation of the emergency core cooling systems; failure of core cooling is a consequence of containment failure.
6	Large pre-existing leakage or failure to isolate containment with the spray system not operating.
7	Large pre-existing leakage or failure to isolate containment with the spray system operating.
8	Late containment failure as a result of overtemperature or overpressure caused by the generation of steam or the burning of hydrogen and/or other flammable gases. Either the spray system is operable or the sprays fail, but the debris bed is nevertheless permanently coolable.
9	Late containment failure, as in bin 8, but the spray system is not available.
10	Long-term overtemperature or overpresurization of containment as a result of the generation of steam and noncondensible gases as a consequence of core/concrete interactions resulting in leakage sufficient to prevent further pressurization but insufficient to produce rapid depressurization.
11	Containment bypass as a result of steam generator tube rupture (SGTR).
12	Containment bypass (interfacing system's loss-of-coolant accident, but with little or not submergence at the point of release.

Table 19.9 (Continued)

Bin	Characteristics
13	Base-mat melt-through with containment spray system unavailable.
14	Base-mat melt-through with containment spray system available.
15	Intact containment throughout the accident with the release equivalent to the design-basis leakage.
16	Containment failure at the time of vessel breach as a result of direct heating with the RCS previously at high pressure. The spray system fails before or at the time of containment failure.
17	Containment failure as a result of direct heating, as in bin 16, but the sprays continue to operate after vessel breach.
18	Containment failure as a result of direct heating at vessel breach with the RCS at moderate pressure and with the sprays failing at the time of vessel breach.
19	Containment failure as a result of direct heating at vessel breach with the RCS at moderate pressure and with the sprays continuing to operate after vessel breach.

Table 19.10 A summary of fractional radioactive release that could result from various types of potential severe accidents - RESAR SP/90[a,b]

Bin[c]	Accident release group								
	XE	I	CS	TE	SR	RU	LA	CE	BA
1	0.10E+01	0.12E+00	0.80E-01	0.14E+00	0.84E-02	0.15E-05	0.14E-06	0.14E-06	0.84E-02
	0.00E+00	0.71E-01	0.76E-01	0.36E-01	0.19E+00	0.15E-06	0.18E-01	0.18E-01	0.19E+00
3	0.10E+01	0.43E+00	0.43E+00	0.65E-01	0.97E-02	0.15E-05	0.14E-06	0.14E-06	0.97E-02
	0.00E+00	0.39E-01	0.39E-01	0.36E-01	0.19E+00	0.15E-06	0.18E-01	0.18E-01	0.19E+00
4	0.10E+01	0.43E+00	0.43E+00	0.69E-01	0.97E-02	0.15E-05	0.14E-06	0.14E-06	0.97E-02
	0.00E+00	0.39E-01	0.39E-01	0.60E-03	0.20E-02	0.17E-08	0.17E-03	0.17E-03	0.20E-02
6	0.10E+01	0.15E-03	0.10E-03	0.18E-03	0.11E-04	0.19E-08	0.18E-09	0.18E-09	0.11E-04
	0.00E+00	0.71E-01	0.76E-01	0.36E-01	0.19E+00	0.15E-06	0.18E-01	0.18E-01	0.19E+00
7	0.10E+01	0.14E-01	0.14E-01	0.23E-02	0.32E-03	0.49E-07	0.47E-08	0.47E-08	0.32E-03
	0.00E+00	0.13E-02	0.13E-02	0.26E-03	0.40E-05	0.33E-11	0.33E-06	0.33E-06	0.40E-05
8	0.10E+01	0.71E-01	0.76E-01	0.28E-03	0.51E-05	0.19E-09	0.33E-06	0.33E-06	0.51E-05
9	0.10E+01	0.72E-01	0.76E-01	0.16E-01	0.38E-03	0.22E-08	0.36E-04	0.36E-04	0.38E-03
10	0.10E+01	0.15E-03	0.10E-03	0.18E-03	0.11E-04	0.19E-08	0.18E-09	0.18E-09	0.11E-04
	0.00E+00	0.71E-01	0.76E-01	0.52E-02	0.12E-03	0.10E-09	0.12E-04	0.12E-04	0.12E-03
11	0.10E+01	0.80E-02	0.80E-02	0.41E-02	0.25E-03	0.40E-07	0.40E-08	0.40E-08	0.25E-03
	0.00E+00	0.48E-02	0.48E-02	0.36E-02	0.19E-01	0.15E-07	0.18E-02	0.18E-02	0.19E-01
12	0.10E+01	0.80E-01	0.80E-01	0.41E-01	0.25E-02	0.40E-06	0.40E-07	0.40E-07	0.25E-02
	0.00E+00	0.48E-01	0.48E-01	0.36E-01	0.19E+00	0.15E-06	0.18E-01	0.18E-01	0.19E+00
13	0.10E+01	0.60E+04	0.40E-04	0.12E+03	0.15E-03	0.86E-09	0.14E-04	0.14E-04	0.15E-03
14	0.10E+01	0.60E+04	0.40E+04	0.79E-04	0.20E-04	0.75E-09	0.13E-05	0.13E-05	0.20E-04

Footnotes at end of table.

Table 19.10 (Continued)

Bin[c]	Accident release group								
	XE	I	CS	TE	SR	RU	LA	CE	BA
15	0.30E-01 0.00E+01	0.60E-04 0.00E+00	0.40E-04 0.00E+00	0.71E-04 0.16E-04	0.42E-06 0.49E-04	0.74E-09 0.40E-10	0.70E-10 0.48E-05	0.70E-10 0.48E-05	0.42E-05 0.49E-04
16	0.10E+01 0.00E+00	0.12E+00 0.71E-01	0.80E-01 0.76E-01	0.22E+00 0.12E-01	0.45E-01 0.66E-01	0.73E-01 0.54E-07	0.38E-01 0.64E-02	0.38E-01 0.64E-02	0.45E-01 0.66E-01
18	0.10E+01 0.00E+00	0.43E+00 0.39E-01	0.43E+00 0.39E-01	0.11E+00 0.25E-01	0.27E-01 0.13E+00	0.35E-01 0.10E-06	0.18E-01 0.13E-01	0.18E-01 0.13E-01	0.27E-01 0.13E+00
19	0.10E+01 0.00E+00	0.43E+00 0.39E-01	0.43E+00 0.39E-01	0.92E-01 0.52E-03	0.20E-01 0.18E-02	0.16E-01 0.15E-08	0.10E-01 0.15E-03	0.10E-01 0.15E-03	0.20E-01 0.18E-02

[a] Symbols for the release group:

XE - xenon	TE - telurium	LA - lanthanum
I - iodine	SR - strontium	CE - cerium
CS - cesium	RU - ruthenium	BA - barium

[b] Release fraction is defined as fraction of the core inventory of radionuclides released to the environment unadjusted for radioactive decay within the plant systems and components.

[c] For accident bin characteristics, refer to Table 19.9 of this SER.

sensitivity analysis as part of the PDA application. Thus, the staff concludes that BNL's independent assessment of the conditional consequence estimates, making use of state-of-the-art methods generally accepted by the staff, is more representative of the proposed RESAR SP/90 design and containment than is the applicant's consequence estimates. Table 19.11 of this SER provides the summary of release characteristics that formed the input to BNL's estimate of severe-accident consequences for the RESAR SP/90 design. Tables 19.12 and 19.13 document the conditional consequences for a typical low-population-density and a high-population-density site, respectively. The most significant consequences result from early containment-failure scenarios that are dominated primarily by the direct containment heating effect. Furthermore, marked differences in consequence estimates are evident between the high-population- and low-population-density sites.

To estimate the risk to the public resulting from potential severe accidents, BNL made use of the estimated conditional consequences in terms of early fatalities, latent cancer fatalities, and population dose (Tables 19.12 and 19.13), the conditional probability of the accident release modes (Table 19.5), and the frequency of various severe-accident bins (Table 19.4). A summary of risk estimates in terms of early fatalities, latent cancer fatalities, and population dose for low- and high-population-density sites is provided in Table 19.14. Table 19.14 also provides severe-accident risk estimates with the direct containment heating effect. The risk estimates shown in Table 19.14 are due only to internal events, that is transients and LOCAs. The portion of risk resulting from external events such as fires, floods, and seismic events should be provided during the FDA stage of review.

19.5 Conclusions

The staff's insights and issues resulting from the review of the estimated risk of potential severe accidents for the RESAR SP/90 design follow:

(1) A major severe accident issue concerns the potential for high vessel-pressure core melt scenarios, particularly high-pressure blowdown that results in direct containment heating. BNL analyses determined that there is up to a 73 percent chance of failing the containment at an early stage following such a high-pressure molten-core scenario, based on calculations performed in 1987. The applicant has documented only point estimates of the containment failure probability and risk of early and latent cancer fatalities applicable to the proposed RESAR SP/90 containment. These point estimates of risk do not include the effect of high-pressure molten-core ejection and resulting rapid containment failure scenarios.

The staff concludes that the point-estimate risk as documented in the RESAR SP/90 PRA does not show risk results representative of the proposed RESAR SP/90 design because of the possible effect of the direct containment heating phenomenon. The staff has, as part of this PRA review and the NUREG-1150 (second draft) effort, performed containment response analyses with regard to the direct-containment-heating (DCH) phenomenon and the related issues of "induced" reactor coolant system failures as a result of natural circulation heating during severe accidents. From this, the staff concludes that there are considerable uncertainties involved in assigning a probability estimate to the induced reactor coolant system failure DCH phenomena

Table 19.11 A summary of release characteristics to be input to consequence estimates for RESAR SP/90 design

Bin[a]	Puff[b]	Time of release (hr)[c]	Duration of release (hr)	Warning time (hr)[d]	Energy of release (Btu/hr)	Elevation of release (m)
1	1	5.50	0.40	2.00	0.24E+09	10.00
	2	6.00	1.30		0.30E+06	
2	1	5.50	0.40	2.00	0.24E+09	10.00
	2	6.00	1.30		0.30E+06	10.00
3	1	5.50	0.40	1.90	0.24E+09	10.00
	2	6.00	1.30		0.30E+06	
4	1	5.50	0.40	1.90	0.84E+08	10.00
	2	6.00	1.30		0.10E+07	
5	1	1.50	2.00	2.00	0.10E+07	10.00
	2	3.50	10.00		0.10E+06	
6	1	5.50	0.40	0.00	0.24E+07	10.00
	2	6.00	1.30		0.10E+06	
7	1	5.50	0.40	0.00	0.10E+06	10.00
	2	6.00	1.30		0.00E+00	
8	1	33.00	8.00	24.00	0.12E+07	10.00
9	1	33.00	8.00	24.00	0.12E+08	10.00
10	1	5.50	0.40	2.00	0.10E+06	10.00
	2	6.00	1.30		0.10E+07	
11	1	2.50	0.50	1.00	0.10E+07	0.00
	2	3.50	5.00		0.10E+06	
12	1	1.50	0.50	1.00	0.10E+08	0.00
	2	3.50	7.50		0.10E+06	
13	1	48.00	1.00	24.00	0.00E+00	0.00
14	1	48.00	1.00	24.00	0.00E+00	0.00
15	1	5.50	0.40	0.00	0.70E+04	10.00
	2	6.00	10.00		0.70E+04	
16	1	5.50	0.40	0.50	0.84E+09	10.00
	2	6.00	1.30		0.30E+06	

Footnotes at end of table.

Table 19.11 (Continued)

Bin[a]	Puff[b]	Time of release (hr)[c]	Duration of release (hr)	Warning time (hr)[d]	Energy of release (Btu/hr)	Elevation of release (m)
17	1	5.50	0.40	0.50	0.84E+08	10.00
	2	6.00	1.30		0.30E+05	
18	1	5.50	0.40	0.50	0.84E+08	10.00
	2	6.00	1.30		0.30E+06	
19	1	5.50	0.40	0.50	0.84E+08	10.00
	2	6.00	1.30		0.30E+05	
LCF	1	36.4	10.0	24.0	0.59E+07	10.00
V	1	1.5	7.5	1.0	0.46E+05	0.00
NCF[e]		1.4	10.0	0.62	0.96E+03	10.00

[a]For accident bin characteristics, refer to Table 19.9.

[b]Puff 1 - Characteristics corresponding to the release at the early phase of melting.
Puff 2 - Characteristics corresponding to the release at the late phase of melting at time of containment failure.

[c]Time of release is defined as the time interval between the potential severe accident and the release of radioactive material from the containment building to the environment.

[d]Warning time is defined as the time interval between the awareness of the impending severe accident and the possible release of radioactive materials from the containment building to the environment.

[e]LCF = late containment failure.
V = event V.
NCF = no containment failure.

Table 19.12 BNL conditional consequence estimates of potential severe accidents at a typical low-population-density RESAR SP/90 site[a,b]

Bin[c,d]	Evacuation	Prompt fatalities	Early injuries	Latent cancer deaths	Population dose (P-SV)[e] 0-50 mi	0-350 mi
1	None	9.87	175.5	7170.0	1.77E+05	7.90E+05
	Evacuation	0.98	103.4	6970.0	1.69E+05	7.82E+05
3	None	16.0	416.2	9450.0	1.86E+05	8.75E+05
	Evacuation	NC[f]	NC	NC	NC	NC
4	None	1.53	298.3	7590.0	5.80E+04	5.02E+05
	Evacuation	NC	NC	NC	NC	NC
6	None	5.83	34.0	5270.0	1.68E+05	6.92E+05
	Evacuation	0.151	5.18	5090.0	1.60E+05	6.83E+05
7	None	0.001	0.26	753.0	1.90E+04	4.81E+04
	Evacuation	NC	NC	NC	NC	NC
8	None	0.0	0.36	58.0	6.02E+04	1.91E+05
	Evacuation	NC	NC	NC	NC	NC
9	None	0.0	0.98	3660.0	6.56E+04	2.22E+06
	Evacuation	0.0	0.005	3660.0	6.55E+04	2.22E+05
10	None	0.021	12.1	2630.0	4.34E+04	1.56E+05
	Evacuation	0.0	0.003	2620.0	4.33E+04	1.56E+05
11	None	0.04	1.3	1240.0	5.17E+04	2.02E+05
	Evacuation	NC	NC	NC	NC	NC
12	None	2.5	33.7	6540.0	1.90E+05	9.78E+05
	Evacuation	0.005	14.9	6480.0	1.86E+05	9.74E+05
13	None	0.0	0.0	14.8	1.72E+03	3.11E+03
	Evacuation	NC	NC	NC	NC	NC
14	None	0.0	0.0	6.1	3.64E+02	6.39E+02
	Evacuation	NC	NC	NC	NC	NC
15	None	0.0	0.0	7.4	7.62E+02	1.27E+03
	Evacuation	0.0	0.0	7.3	7.60E+02	1.27E+03
16	None	27.6	340.2	10900.0	2.17E+05	8.40E+05
	Evacuation	23.0	305.2	10800.0	2.14E+05	8.40E+05

Footnotes at end of table.

Table 19.12 (Continued)

Bin[c,d]	Evacuation	Prompt fatalities	Early injuries	Latent cancer deaths	Population dose (P-SV)[e] 0-50 mi	0-350 mi
18	None	81.0	642.0	10300.0	2.32E+05	8.40E+05
	Evacuation	31.3	495.0	10100.0	2.22E+05	8.30E+05
19	None	27.2	436.2	7610.0	1.03E+05	5.52E+05
	Evacuation	7.49	309.1	7540.0	1.01E+05	5.49E+05
LCF	None	0.01	10.6	6530.0	9.04E+04	3.85E+05
	Evacuation	0.0	1.09	6520.0	9.01E+04	3.85E+05
NCF	None	0.0	0.09	6.8	2.64E+02	3.07E+02
	Evacuation	0.0	0.0	6.5	2.49E+02	3.55E+02
V	None	0.2	8.2	3570.0	6.64E+04	2.15E+05
	Evacuation	0.0	0.62	3570.0	6.62E+04	2.15E+05

[a]The relatively insignificant change in conditional consequence estimates as a result of the presence versus absence of evacuation differences should be noted for bins 16 through 19. These bins (16, 17, 18, and 19) are modeled separately to account for the direct heating effect of the proposed RESAR SP/90 containment following a high-pressure core-melt scenario.

[b]The last three bins are reported to show the significance of conditional consequence estimates on the basis of release terms provided by the applicant.

[c]For accident bin characteristics, refer to Table 19.9.

[d]Although source terms for bins 2, 5, and 17 were calculated, consequence estimates were not estimated because of an insignificant/nonexistent accident probability.

[e]P-SV is person-Sieverts, to read the entries in person-rem, a multiplier of 100 should be used.

[f]NC = Not calculated.

Table 19.13 BNL conditional consequence estimates of potential severe accidents at a typical high-population-density RESAR SP/90 site[a,b]

Bin[c,d]	Evacuation	Prompt fatalities	Early injuries	Latent cancer deaths	Population dose (P-SV)[e] 0-50 mi	Population dose (P-SV)[e] 0-350 mi
1	None	479.0	1162.0	14200.0	4.54E+05	2.23E+06
	Evacuation	0.19	74.1	12000.0	2.88E+05	2.07E+06
6	None	418.0	610.0	11500.0	4.28E+05	2.07E+06
	Evacuation	0.0	1.76	9440.0	2.64E+05	1.91E+06
9	None	1.57	67.8	6000.0	1.54E+05	3.63E+05
	Evacuation	0.0	0.0	5920.0	1.52E+05	3.61E+05
10	None	39.3	197.5	5120.0	1.22E+05	3.05E+05
	Evacuation	0.0	0.031	5040.0	1.20E+05	3.02E+05
12	None	348.0	856.0	13000.0	4.88E+05	2.57E+06
	Evacuation	51.8	450.0	12000.0	3.76E+05	2.47E+06
15	None	0.0	0.0	14.2	1.14E+03	2.29E+03
	Evacuation	0.0	0.0	13.7	1.09E+03	2.25E+03
16	None	270.0	1258.0	23900.0	6.16E+05	2.26E+06
	Evacuation	73.7	670.6	22700.0	5.49E+05	2.19E+06
18	None	940.0	2495.0	22600.0	6.23E+05	2.37E+06
	Evacuation	81.6	879.0	20600.0	4.62E+05	2.21E+06
19	None	282.0	2057.0	15900.0	2.79E+05	1.25E+06
	Evacuation	21.3	592.0	15200.0	2.50E+05	1.22E+06
LCF	None	31.0	142.5	10800.0	2.11E+05	6.31E+05
	Evacuation	0.0	0.91	10600.0	2.06E+05	2.26E+05
NCF	None	2.61	17.6	21.7	9.51E+02	1.10E+03
	Evacuation	0.0	0.22	14.9	6.64E+02	8.16E+02
V	None	129.0	503.4	6650.0	1.72E+05	3.96E+05
	Evacuation	2.26	75.8	6470.0	1.65E+05	3.90E+05

[a]The bins 16, 17, 18, and 19 are modeled separately to account for the direct heating effect of the proposed SP/90 containment following a high-pressure core melt scenario. The high conditional consequence estimates associated with these bins are noteworthy.

[b]The last three bins are reported to show the significance of conditional consequence estimates based on release terms provided by the applicant.

[c]For bin characteristics, refer to Table 19.9.

Table 19.13 (Continued)

[d]Although the source term for bins 2, 3, 4, 5, 7, 8, 11, 13, 14, and 17 were calculated, consequence estimates were not documented due to insignificant/nonexistent probability.

[e]P-SV is person-Sieverts; to read the entries in person-rem, a multiplier of 100 should be used.

Table 19.14 A summary of potential severe accident risk estimates of the proposed RESAR SP/90 design

Case	Early* fatality	Latent* fatality	Dose**
Low-population-density site:			
with evacuation			
DCH† impact	9.3E-5	4.7E-2	3.60
no DCH impact	2.3E-9	1.1E-2	0.67
no evacuation			
DCH impact	1.3E-4	4.7E-2	3.60
no DCH impact	1.8E-7	1.2E-2	0.73
High-population-density site:			
with evacuation			
DCH impact	2.9E-4	9.6E-2	9.10
no DCH impact	1.5E-6	1.9E-2	1.20
no evacuation			
DCH impact	1.3E-3	1.0E-1	9.50
no DCH impact	7.8E-5	1.9E-2	1.20

*Per reactor year.
**Estimates are based on a 350-mile exposure zone and are expressed in units of person-rem.
†DCH = Direct containment heating effect failure mode.

applicable to the RESAR SP/90 design. Therefore, the staff believes that the applicant should, as part of the revised PRA activities, perform a design-specific uncertainty and/or sensitivity analysis of the two phenomena and their effect on early containment failures and risk, and should submit this during the FDA stage of review.

(2) Neither the applicant nor BNL has performed a detailed analysis of the structural integrity of the proposed Westinghouse containment. Both Westinghouse and BNL made use of previously performed Zion containment pressure integrity analyses to make judgments on the adequacy of the RESAR SP/90 containment. Because the proposed Westinghouse containment is different (steel sphere versus concrete cylinder) and is to be standardized, the staff concludes that the applicant must submit detailed analyses of structural integrity under severe accident conditions to support the FDA application. Such detailed analyses should provide information regarding the location of failures, penetration integrity, pool drainage behavior, etc., that are critical to staff decisions on regulatory issues.

(3) The modeling of the MAAP, STCP, and CONTAIN computer codes has been the subject of review as part of the NRC's interactions with IDCOR and EPRI representatives. Recently, the staff has been collaborating with EPRI in reviewing the MAAP 3-B computer code to determine how MAAP and NRC codes treat those phenomena important to severe accidents. The results of this collaboration will identify technical issues treated similarly by both NRC codes and the MAAP and technical issues that are treated differently, and consequences of these differences on overall risk, if possible. This work will be completed in 1991. The staff suggests that differences identified in this process as potentially significant be specifically addressed by the applicant during the FDA stage of review.

(4) Since this PRA addresses only internal events (transients and LOCAs), the portion of risk as a result of external events, such as fires, floods, seismic events, and high winds, must be satisfactorily addressed to support the FDA application.

(5) A sensitivity analysis was performed by BNL to examine the effect of early containment failure that would result from the direct containment heating phenomenon. When the direct containment heating effect as a containment failure mode is included (with an associated probability of occurrence), the early fatality risk associated with the RESAR SP/90 design at two possible sites (a high-population-density site such as Zion and a low-population-density site such as Salem) increases by a factor of 3 (with evacuation) to 10 (without evacuation).

BNL compared the risk estimates for the RESAR SP-90 design with the safety goals documented in the Commission's Safety Goal Policy. The comparison is provided in Table 19.15. The safety goal comparison indicates that the risk estimates for the RESAR SP/90 design are significantly below the Commission's safety goal quantitative health objectives as documented in the Federal Register (51FR30028, dated August 21, 1986). Also, BNL compared the reviewed risk estimates for the RESAR SP/90 design with the design goal documented in the Electric Power Research Institute (EPRI) Advanced Light Water Reactor (ALWR) Requirements Document (i.e., a mean frequency of 1E-6

Table 19.15 Comparison of reviewed risk estimates of the RESAR SP-90 design with safety goals

	Individual risk within 1 mile[a]	Risk of cancer within 10 miles[a]
USNRC Safety Goal[b]		
	5E-7	2E-6
Reviewed Risk Results of the SP-90 Design[c]		
Low-population-density site (Salem site)		
With evacuation measures		
With DCH impact[d]	e	1E-8
Without DCH impact	e	9E-10
High-population-density site (Zion site)		
With evacuation measures		
With DCH impact	6E-9	2E-8
Without DCH impact	2E-10	1E-9

[a]Fatalities per reactor year.

[b]These estimates are basically the same as the goal estimates published in the Federal Register (51FR30028, August 21, 1986).

[c]These estimates are basically point estimates and are extracted from BNL's analyses.

[d]DCH = direct containment heating.

[e]Not calculated.

or less of obtaining a dose of 25 rem or greater at one-half mile distance from the plant). Offsite consequence calculations were performed (by MACCS, Version 1.4, computer code) using a RESAR SP-90 design-specific core inventory and meteorology characteristics of the Washington, D.C., area to obtain the whole-body dose in a 24-hour period (following a postulated release) at a distance of 0.5 mile from the proposed plant. Accordingly, the frequency of the dose exceeding 25 rem at a 0.5-mile radius was estimated, and the results are provided in Table 19.16 for all release bins. In summary, the frequency of the dose exceeding 25 rem at a 0.5 mile from the proposed plant is estimated to be 4E-6 reactor year if an early containment failure as a result of the DCH effect is not assumed. (In these cases, late containment failure occurred with relatively high probability and with resulting doses greater than 25 rem.) The major contributors to the severe release frequency are the postulated accident bins 9 and 10. These accident bins primarily involve transient-induced sequences followed by late containment failures as a result of overtemperature and overpressure buildup inside the containment caused by the generation of flammable and noncondensible gases. The scenario also involved coolable core debris with an inoperable containment spray system and containment fan coolers. In accident bin 9, the pressure and temperature rise is due to postulated hydrogen burning. In accident bin 10, the generation of steam and noncondensible gases is due to postulated core/concrete interactions subsequent to a postulated vessel failure event.

The result is about 5E-6 per reactor year (see Table 19.16) if an early containment failure resulting from the DCH effect is assumed to occur (with an assigned conditional probability, see Table 19.5) for the postulated high-pressure core-melt sequences. The staff will report again on its findings of the risk comparison after the applicant has submitted additional information during the RESAR SP/90 FDA stage of review.

(6) The staff made an attempt to evaluate the RESAR SP/90 design against the design goals developed by the EPRI ALWR Requirements Document. As part of the design evaluation of the future evolutionary designs, EPRI proposed an "investment protection" goal for all evolutionary designs including the RESAR SP/90 design. The investment protection goal states that the cumulative mean frequency of postulated core damage sequences should be less than 1E-5 per reactor year. This goal is basically geared towards the core-damage-prevention concept. With the exception of external events (which were not considered in the PDA application), the RESAR SP/90 design meets this investment protection goal. The staff notes that the RESAR SP/90 design has additional trains of high- and low-pressure coolant makeup systems. In particular, the emergency water storage tank (EWST) for the RESAR SP/90 design will be located inside the containment. Although the risk analyses of external events are not within the scope of the PDA application, the applicant should submit these analyses to the staff during the FDA stage of review.

Also, one of the two staff-recommended design goals on containment performance states that the conditional failure probability of the containment should be less than 1 in 10 when weighted over credible core damage sequences. The applicant has not evaluated its design against this goal, although the applicant has agreed to meet the above design goal for the RESAR SP/90 design during the FDA stage of review.

Table 19.16 Comparison of reviewed risk estimates of the RESAR SP/90 design by accident bins with ALWR safety goal limits

Bin[a]	Dose at 0.5 mile (person rem)[b]	Frequency (without DCH impact)	Frequency (with DCH impact)[c]
	EPRI ALWR Safety Goal[d]		
	25	1 E-6	1 E-6
	Reviewed Risk Results of the SP-90 Design[e]		
1	7780	5E-10	Nil
3	7430	7E-11	Nil
4	1390	1E-11	Nil
6	6870	1E-8	3E-9
7	695	3E-9	6E-10
8	403	1E-9	3E-10
9	288	2E-6	4E-7
10	1610	2E-6	4E-7
11	1390	3E-7	3E-7
12	6960	3E-8	3E-8
13	26	1E-8	3E-9
14	20	1E-8	2E-9
15	13	2E-6	5E-7
16	3800	NA	4E-6
18	7480	NA	2E-7
19	3070	NA	2E-7

[a]For accident bin characteristic, refer to Table 19.9.

[b]Estimates are mean values.

[c]Frequency per reactor year. DCH = direct containment heating.

[d]These estimates are basically the same as the safety estimates documented in the ALWR Requirements Document. The ALWR safety limit is characterized in the form of a statement that the frequency of the whole-body dose at 0.5 mile distance from the center of the plant resulting from a postulated severe accident should not exceed an estimate of 1E-6 per reactor year.

[e]These estimates are basically point estimates and are extracted from the BNL analyses.

In addition, the staff has, as part of the review of the EPRI ALWR Requirements Document, proposed additional "general criteria" for containment performance during postulated severe accident challenges for evolutionary ALWRs, including the RESAR SP/90 design. The first staff criterion states that the containment should maintain its role as a reliable leak-tight barrier by ensuring that the containment stresses do not exceed ASME Service Level C limits for a minimum period of 24 hours following the onset of a postulated core damage event. The second staff criterion states that the estimated unreliability of the mitigation systems (including the decay heat removal systems and reactor cavity flooding system), from the onset of core damage to prevention of significant releases, should not exceed 0.1. Currently, the staff permits the applicant to choose to use one or the other of the two criteria.

The applicant has not assessed these containment mitigation systems against the staff's proposed general criteria and has not evaluated the need for such systems in the context of such containment performance criteria. Therefore, the applicant should perform design evaluations against the staff's proposed general criteria, including the containment failure definitions, and should submit them during the FDA stage of review.

(7) The staff evaluated the adequacy of certain improvements to the current RESAR SP/90 design, such as the RCS depressurization system, the cavity flooding system, and dc-powered igniters.

RCS depressurization system: The staff believes that the Westinghouse-proposed RCS depressurization system along with established emergency operating procedures will result in a lower conditional probability for occurrence of the DCH effect following a postulated high-pressure molten-core ejection accident and a lower net risk. Therefore, this system provides a viable method of minimizing the potential for an early containment failure following a postulated core-melt accident.

Cavity flooding system: The applicant has improved the current RESAR SP/90 design of the cavity flooding system in such a manner that water makeup could be provided to the bottom of the reactor vessel from the EWST located inside the containment. The staff believes that this will be beneficial in mitigating the consequences (primarily overtemperature failures of the containment) of ex-vessel debris. Therefore, this system provides a viable method of preventing some postulated containment failures following a postulated core-melt accident. However, additional production of hydrogen resulting from flooding the reactor cavity could result in an additional overpressurization effect and could cause an earlier challenge to the containment (as defined in the current version of the RESAR SP/90 design). Therefore, the staff suggests that the applicant evaluate the net-risk significance of this containment mitigation system (in combination with the dc-powered igniters) as part of future revisions to the SP/90 PRA.

DC-Powered Igniters: The applicant plans to place dc-powered igniters inside the containment. The staff believes that the effect of the dc-powered igniters will be significant with regard to preventing hydrogen-induced containment failure during the ex-vessel debris interaction (both dry cavity and wet cavity because of the reactor cavity flooding) scenario following a postulated core-melt accident, provided the igniters are located in appropriate locations, including the ceilings of the EWST and the bottom area of the reactor cavity. The staff suggests that the applicant evaluate the net-risk significance of this containment mitigation system (in combination with the reactor cavity flooding system) as part of future revisions to the SP/90 PRA.

19.6 Outstanding Issues To Be Resolved

The applicant should include the information listed below in its RESAR SP/90 FDA application. The sections of this SER in which these issues are discussed are given in parentheses.

- detailed design information and an appropriate core melt accident analysis to obtain revised core damage frequencies (Section 19.3)
- a quantitative reliability analysis of the system and related references as well as discussions on the reliability of the as-built IPS with respect to the RESAR SP/90 design requirement and EPRI ALWR Requirement Document (Section 19.3)
- additional discussions concerning to ATWS sequence frequency estimates addressing the consistency between EPRI ALWR Requirements Document and the RESAR SP/90 design requirements with regard to the number of reactor trips per year (Section 19.3)
- modeling details and discussions relating to the SGTR event followed by the failures of an automatic and manual scram in view of any improvements made in the RESAR SP/90 design-specific control room features and other human reliability improvements, including discussions on station blackout enhancement capabilities and its effect on accident sequence frequency estimates (Section 19.3)
- modeling details of the natural circulation scenario along with the direct heating effect for all high-pressure core melt accidents (Section 19.4.1)
- analysis of detailed containment structural performance under severe accident conditions, including the containment loading conditions resulting from high-pressure molten-core ejection accidents (Section 19.4.2)
- risk analysis as a result of external events such as fires, floods, seismic events and high winds (Section 19.4.4)
- additional discussions regarding the differences identified in comparing MAAP and NRC computer codes with regard to the treatment of the technical issues as well as consequences of these differences on overall risk (Section 19.5)
- an evaluation of the RESAR SP/90 containment design against the EPRI ALWR design goal of 0.1 conditional containment failure probability (Section 19.5)

- an evaluation of the RESAR SP/90 containment design and mitigation systems against the general criteria for containment performance proposed by the staff, including the containment failure definitions (Section 19.5)

- an evaluation of the net-risk significance of the containment mitigation systems such as the RCS depressurization system, cavity flooding system and dc-powered ignitors (Section 19.5)

20 REGULATORY DEPARTURE ANALYSIS

20.1 Introduction

Based on operating experiences and a number of studies (e.g., probabilistic risk assessments [PRAs]), the staff has identified the following safety issues that are significant in evaluating the acceptability of future designs, which include the Westinghouse RESAR SP/90.

- public safety goals for evolutionary light-water reactors
- source term
- anticipated transients without scram
- mid-loop operation
- station blackout
- fire protection
- intersystem LOCA
- hydrogen generation and control
- core-concrete interaction - ability to cool core debris
- high-pressure core-melt ejection
- containment performance
- advanced boiling-water reactor (ABWR) containment vent design
- equipment survivability
- operating-basis earthquake/safe-shutdown earthquake
- inservice testing of pumps and valves

Except for the ABWR containment vent design, these safety issues are pertinent to the RESAR SP/90 design.

These issues are discussed in this section in the context of certification of the evolutionary designs. Discussions on each issue include the current regulatory requirement or interpretation, the staff's proposed departure from current requirement, and the basis for the proposed departure. These issues were identified to the Commission in SECY-90-016 dated January 12, 1990.

Note that these issues are considered fundamental to agency decisions on the acceptability of the evolutionary ALWR designs. The Commission has reviewed these issues and addressed them in the Staff Requirements Memorandum dated June 26, 1990. The Commission responses to the staff proposals are included in the appropriate sections. The Commission may at some future point decide that certain issues involve policy questions that the Commission may wish to reconsider.

20.2 Evolutionary ALWR Certification Issues

20.2.1 General Issues

(1) ALWR Public Safety Goal

The EPRI Requirements Document proposes that the evolutionary ALWRs comply with the following public safety goals:

- The frequency of core damage will be less than 1.0X10E-5 events/per reactor year. (Note: EPRI refers to this as a "quantitative investment protection goal.")

- Whole-body dose at an assumed 0.5-mile site boundary must be less than 25 rem for events whose cumulative frequency exceeds 1.0X10E-6 per reactor year.

In its letter of March 20, 1990, Westinghouse stated its position on severe accident policy issues: Westinghouse committed to meet the EPRI ALWR public safety goals and to perform a detailed PRA for both internal and external events to demonstrate compliance with these goals at the FDA stage of review of the RESAR SP/90 design.

The staff was reviewing the proposed ALWR public safety goals to ensure they are consistent with the Commission's Safety Goal Policy Statement, which proposes qualitative as well as quantitative safety goals for future reactor designs. The current regulations do not specify requirements in numerical terms of frequency of core damage or large release events. However, the Commission, in its Safety Goal Policy Statement, proposed that the staff examine a general performance guideline that "the overall mean frequency of a large release of radioactivity to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation."

The staff was considering the use of the following quantitative objectives in its implementation of the safety goal policy for future standardized plants:

- The mean core damage frequency target for each design should be less than 1.0X10E-5 event per reactor year.

- The overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation where a large release is defined as one that has a potential for causing an offsite early fatality.

However, the Commission has disapproved the use of 1.0x10E-5 events per reactor year as a core damage frequency for advanced designs. The Commission supports the the use of 1.0x10E-4 events per reactor year as a core damage frequency goal. Although the Commission strongly supports the use of information and experience gained from the current generation of reactors as a basis for improving the safety performance of new designs, the NRC should not adopt industry objectives as a basis for establishing new requirements. However, after applying the criteria of 10 CFR Part 52 (and in view of the uncertainties associated with PRAs), if the staff concludes that additional requirements are needed on the basis of operational experience to provide assurance that future designs will meet the safety goal policy statement, then the staff should provide those additional requirements to the Commission for consideration as they are identified.

The Commission approved the overall mean frequency of a large release of radioactive material to the environment from a reactor accident as less than one in one million per year of reactor operation.

(2) Source Term

The staff's methodology for determining compliance with the siting requirements of 10 CFR Part 100 has been based on the TID-14844 source term (U.S. Atomic Energy Commission, "Calculation of Distance Factors for Power and Test Reactor Sites," TID-14844, March 23, 1962). This methodology, which involves calculation of offsite dose for comparison against Part 100 dose criteria (i.e., criteria for establishing the size of the exclusion area and the low-population zone), is widely acknowledged to utilize conservative assumptions. At the time this approach was developed, these conservatisms were considered appropriate. They were based on uncertainties associated with accident sequences and equipment performance and were used as a means to ensure that future plant sites would be essentially equivalent to sites approved up until that time. The conservatisms initially included in the methodology have been essentially retained up to this time.

EPRI has stated that the evolutionary ALWR licensing design-basis requirements as well as design enhancements related to severe accidents should be based on the full body of current knowledge regarding accident source terms. It believes that the evolutionary designs should be evaluated based on a realistic treatment of fission product source terms, including the extensive research that has been done on fission product behavior since TID-14844 was issued, and especially since the Three Mile Island accident in 1979. EPRI's view is that this approach will result in designs that are improved and provide enhanced safety protection. EPRI has identified this as a plant optimization issue.

The severe accident source term calculations for the RESAR SP/90 design were performed using Modular Accident Analysis Program (MAAP) computer code. Westinghouse has indicated that no early containment failure modes can be identified for the RESAR SP/90 design, thereby precluding the potential for "large" fission product releases. Consequently, Westinghouse believes that the differences in source term methodologies do not affect severe accident risk predictions. If source term methodologies were to be modified after the PDA, Westinghouse would address such modifications in the FDA submittal.

The staff is considering the effects of separating site certification from plant design certification for future reactors. Under this plan, reactor site characteristics would be reviewed separately from the reactor design without using source terms or dose calculations. This would require revision to Part 100 and other regulatory staff practices. The results of such a study will establish appropriate guidelines for any future plant license applications. In the interim, however, the staff will adopt the following approach for evolutionary ALWRs:

- Ensure that evolutionary designs meet the requirements of 10 CFR 100.

- Consider deviations from current methodology used to calculate Part 100 doses on a case-by-case basis using engineering judgment and updated information on source term and equipment reliability, even though it is recognized that such deviations could affect plant design features.

- Do not modify current siting practice, even though it is recognized that such deviations could result in calculated low-population zones and exclusion areas that are smaller than those that have been approved for currently operating reactors.

- Continue to interact with EPRI and the evolutionary ALWR vendors to reach agreement on the appropriate use of updated source term information for severe accident performance considerations.

The Commission has approved the staff's approach to source term with the addition of the following element: On an expedited basis, incorporate appropriate changes to regulations,regulatory practices, and the review process resulting from source term research.

20.2.2 Preventive Feature Issues

(1) <u>Anticipated Transient Without Scram (ATWS)</u>

The ATWS rule 10 CFR 50.62 was promulgated to reduce the probability of an ATWS event and to enhance mitigation capability if such an event occurred.

EPRI has indicated that its approach to resolving the ATWS issue is compliance with the ATWS rule. Design requirements beyond those which would be required to meet the rule have not been proposed.

Westinghouse has concluded that a diverse scram system is unnecessary for the RESAR SP/90 design because of (1) the high reliability of the integrated reactor protection system (IPS), (2) a turbine trip and emergency feedwater actuation that is independent of the IPS, (3) the ability to manually trip the rod control motor generators from the main control board, and (4) a highly negative moderator temperature coefficient. Westinghouse has committed to provide a detailed analysis during the FDA stage of review to demonstrate that the consequences of an ATWS are acceptable.

The staff believes, notwithstanding the Westinghouse position on diverse scram systems, that all future evolutionary ALWR designs should be required to provide a diverse scram system unless the ALWR vendor can demonstrate that the consequences of an ATWS are acceptable. The ATWS rule presently requires a diverse scram system for all (Combustion Engineering, Babcock and Wilcox, and General Electric) LWR designs except Westinghouse PWRs. The staff has determined that previous Westinghouse designs had adequate ATWS capability and backfit could not be justified. The staff believes that evolutionary ALWR designs should provide diverse methods of inserting control rods to mitigate a potential ATWS and to ensure a safe reactor shutdown. The staff considers that diverse scram capability is a worthwhile measure of prevention for all evolutionary ALWRs, especially when incorporated into the initial design.

The Commission has approved the staff position. However, if the applicant can demonstrate that the consequences of an ATWS are acceptable, the staff should accept the demonstration as an alternative to the diverse scram system.

The staff also should retain the flexibility to accept designs with non-diverse scram logic in those instances where it is demonstrated to the staff's satisfaction that the reliability of the scram function is such that the risk from ATWS is insignificant.

(2) Mid-Loop Operation

The staff is concerned that decay heat removal capability could be lost when a PWR is shut down for refueling or maintenance and drained to a reduced reactor coolant system (RCS) or "mid-loop" level. For example, a significant problem has been the loss of residual heat removal (RHR) suction because of air-binding of the RHR pumps, which is usually caused by an uncontrolled low-loop level and consequent air ingestion into the pump suction.

The EPRI Requirements Document specifies requirements consistent with measures applicable to operating reactors as described by the administrative procedures identified in Generic Letter 88-17, but does not specify design modifications to address the root cause of this event.

Westinghouse has committed to install a vortex breaker at the RHR hot-leg connections to significantly reduce air entrainment during mid-loop operation. This feature, in conjunction with other RESAR SP/90 design features, should greatly reduce concerns over mid-loop operation.

The staff expects improvements in instrumentation in many existing PWRs, but does not require specific modifications to the nuclear steam supply system (NSSS) to correct mid-loop problems. However, the staff believes that physical modifications such as those proposed by Westinghouse, may be necessary to essentially eliminate any concerns with mid-loop operation for future evolutionary pressurized ALWRs. Mid-loop operation is not explicitly covered by current regulations, however imposition of such requirements would exceed current staff licensing practices. The staff's position is that evolutionary PWR vendors propose design features to ensure high reliability of the shutdown decay heat removal system.

The Commission has approved the staff's proposed position, with the ACRS recommendation of April 26, 1990, that four additional specific requirements be considered for mid-loop operation:

- Design provisions should be made to help ensure continuity of flow through the core and residual heat removal system with low liquid levels at the junction of the RHR system suction lines and the RCS.
- Provisions should be made to ensure availability of reliable systems for decay heat removal.
- Instrumentation for reliable measurements of liquid levels in the reactor vessel and at the junction of the RHR system suction lines and the RCS should be included in the design.
- Provisions should be made for maintaining containment closure or for rapid closure of containment openings.

(3) Station Blackout

The station blackout rule (10 CFR 50.63) allows utilities several design alternatives to ensure that an operating plant can safely shut down in the event that all ac power (off site and on site) is lost.

The EPRI Requirements Document provides for improvements in offsite power reliability, onsite power reliability and capacity, and station blackout coping capability. EPRI is also proposing that a large capacity, diverse alternate ac power source (combustion turbine generator) with the capability to power one complete set of normal safe shutdown loads be included in evolutionary ALWR designs.

The RESAR SP/90 emergency feedwater system design includes two turbine-driven pumps that are independent of ac and dc power. The electrical design includes two full-capacity emergency diesel generators. In addition, it includes a backup seal injection pump powered by a small dedicated diesel generator that has enough capacity to also charge the station batteries. The reactor coolant pump (RCP) seals are equipped with an improved O-ring material. In its letter of March 20, 1990, Westinghouse committed to include a non-Class 1E alternate ac power source in addition to the redundant Class 1E emergency diesel generators for the SP/90 RESAR design.

The staff believes that the preferred method of demonstrating compliance with 10 CFR 50.63 is through the installation of a spare (full capacity) alternate ac power source of diverse design that is consistent with the guidance in Regulatory Guide 1.155, "Station Blackout," and is capable of powering at least one complete set of normal safe shutdown loads.

The Commission has approved the staff's position that the evolutionary ALWRs have an alternate ac power source of diverse design that is capable of powering at least one complete set of normal shutdown loads.

(4) Fire Protection

The staff has concluded that fire protection issues that have been raised through operating experience and through the NRC's external events program must be resolved for evolutionary ALWRs. To minimize fire as a significant contributor to the likelihood of severe accidents for advanced plants, the staff concludes that current NRC guidance must be enhanced. Therefore, the evolutionary ALWR designers must ensure that safe shutdown can be achieved, assuming that all equipment in any one fire area will be rendered inoperable by fire and that re-entry into the fire area for repairs and operator actions if not possible.

Because of its physical configuration, the control room is excluded from this approach, provided an independent alternative shutdown capability that is physically and electrically independent of the control room is included in the design. Evolutionary ALWR designers must provide fire protection for redundant shutdown systems in the reactor containment building that will ensure, to the extent practicable, that one shutdown division will be free of fire damage. Additionally, the evolutionary ALWR designers must ensure that smoke, hot gases, or the fire suppressant will not migrate into other fire areas to the extent that they could adversely affect safe-shutdown capabilities, including operator actions. Because the layout of a nuclear plant if design-specific, plant-specific design details will be reviewed by the staff on an individual basis. The staff will require a plant-specific applicant to provide a description of safety-grade provisions for the fire-protection systems to ensure that the remaining shutdown capabilities are protected, as well as demonstration that the design complies with the migration criteria discussed above.

The EPRI Requirements Document for ALWRs indicates that fire protection will be as specified in 10 CFR 50.48 and Appendix R to 10 CFR 50. It states that for equipment in the same general area, a 3-hour fire barrier will be used instead of physical separation unless it is "impractical or less safe." However, no guidelines are provided in the Requirements Document as to the application of these criteria.

Westinghouse has indicated that the RESAR SP/90 fire protection designs are consistent with the staff's proposed enhancements.

Appendix R to 10 CFR 50 was promulgated for plants that were in operation before January 1, 1979. Subsequently, PRAs performed on more than a dozen plants have shown that fire is a significant contributor to core damage. The staff believes that in keeping with the Commission's desire for enhanced safety for evolutionary ALWRs, fire protection requirements should reflect experience from operating reactors and the greater understanding of severe accidents that has been acquired since Appendix R was promulgated.

The Commission has approved the staff's position on fire protection.

(5) Intersystem LOCA

Future evolutionary ALWR designs can reduce the possibility of a LOCA outside containment by designing (to the extent practicable) all systems and subsystems connected to the RCS to an ultimate rupture strength at least equal to the full RCS pressure.

For both BWRs and PWRs, EPRI states that low-pressure systems that could be overpressurized by the RCS should be designed with sufficient margin to withstand full RCS pressure without structural failure.

For PWRs, relief valves sized to protect against overpressure transients should be provided on the RHR system. RHR suction valves should be provided with permissive interlocks to prevent opening if RCS pressure exceeds RHR design pressure.

Westinghouse has indicated that, should the isolation valves of the RESAR SP/90 design fail, the design pressure of the piping outside of the containment will be sufficient to withstand primary side pressure or will be vented to the emergency water storage tank (EWST).

The staff concludes that designing, to the extent practicable, low-pressure systems to withstand full RCS pressure is an acceptable means of resolving this issue. However, the staff believes that for those systems that have not been designed to withstand full RCS pressure, evolutionary ALWRs should provide (a) the capability for leak testing of the pressure isolation valves, (b) valve position indication that is available in the control room when isolation valve operators are deenergized, and (c) high-pressure alarms to warn control room operators when rising RCS pressure approaches the design pressure of attached low-pressure systems and both isolation valves are not closed.

The staff notes that for some low-pressure systems attached to the RCS, it may not be practical or necessary to provide a higher system ultimate pressure capability for the entire low-pressure system that is connected. The staff will evaluate these exceptions on a case-by-case basis during specific design certification reviews.

The Commission has approved the staff's position on intersystem LOCA provided that, as recommended by the ACRS, all elements of the low pressure system are considered (e.g. instrument lines, pump seals, heat exchanger tubes, and valve bonnets).

20.2.3 Mitigative Feature Issues

(1) Hydrogen Generation and Control

The Commission's Severe Accident and Standardization Policy Statements provide that future designs should address the provisions of 10 CFR 50.34(f). The Commission's stated policy has been codified in 10 CFR 52 to require the technically relevant provisions of 10 CFR 50.34(f) be met. Specifically, in order that containment integrity be maintained, 10 CFR 50.34(f)(2)(ix) requires future designs to provide a system for hydrogen control that can safely accommodate hydrogen generated by the equivalent of a 100-percent fuel-cladding metal-water reaction. In addition, the regulation requires this system to be capable of precluding uniform concentrations of hydrogen from exceeding 10 percent (by volume), or an inerted atmosphere within the containment must be provided.

The EPRI Requirements Document specifies that containment and combustible gas control systems should be designed to accommodate 75 percent in-vessel zirconium-water reaction of the active fuel cladding and 13 percent containment uniform hydrogen concentration. It states that 75-percent cladding oxidation is believed to be a conservative upper limit on the amount of hydrogen generated in a degraded-core situation, including recovery. EPRI has identified this as an optimization issue.

In its letter to the NRC dated March 20, 1990, Westinghouse proposes, for the RESAR SP/90 design, to mitigate the effects of a 100-percent metal-water reaction and to preclude uniform hydrogen concentration from exceeding 10 percent (by volume) through the use of hydrogen igniter and hydrogen recombiner systems.

Aside from the issue of regulatory compliance and applicability and because of the uncertainties in the phenomenological knowledge of hydrogen generation and combustion, the staff concludes that compliance with the criteria of 10 CFR 50.34(f) remains appropriate for combustible gas control design in ALWRs. Research (discussed in NUREG/CR-4551, "Evaluation of Severe Accident Risks and the Potential for Risk Reduction," February 1987) indicates that in-vessel hydrogen generation associated with core-damage accidents may range from approximately 40- to 95-percent-equivalent active cladding oxidation. The amount of cladding oxidation is dependent on a variety of parameters related to sequence progression: reactor coolant system pressure, reflood timing and flow rates, as well as core-melt progression phenomena. Thus, a 75-percent-equivalent cladding reaction continues to be viewed as a reasonable design basis for hydrogen generation for severe accidents in which the reactor pressure vessel (RPV) remains intact.

However, it is the staff's view that ALWRs should provide protection for hydrogen generation resulting from a wider spectrum of accidents, i.e., full core-melt accidents with RPV failure. In that context, it is also necessary to consider ex-vessel hydrogen generation as a result of core debris reacting with available water or core-concrete interactions. Calculations using the CORCON models indicate that if the core debris is cooled in relatively rapid fashion (1-2 hours), additional hydrogen generation will be less than that equivalent to a 25-percent cladding oxidation reaction. This relatively limited ex-vessel reaction is conditional on the existence of a coolable debris bed and the availability of sufficient water. If extensive core-concrete interaction occurs as a result of the absence of cavity flooding, more hydrogen generation should be considered. Considering the effects discussed above, the staff concludes that an equivalent 100-percent cladding oxidation reaction is an appropriate deterministic design criteria and a reasonable surrogate for the combination of both in-vessel and ex-vessel hydrogen generation.

Because of the uncertainties in the phenomenological knowledge of hydrogen generation and combustion, it is still the staff's position that, as a minimum, evolutionary ALWRs should be designed to (1) accommodate hydrogen equivalent to 100-percent metal-water reaction of the fuel cladding and (2) limit containment hydrogen concentration to no greater than 10 percent. Furthermore, because hydrogen control is necessary to preclude local concentrations of hydrogen below detonable limits and given uncertainties in present analytical capabilities, the staff concludes evolutionary ALWRs should provide containment-wide hydrogen control (e.g., igniters, inerting) for severe accidents. Additional advantages of providing hydrogen control mitigation features (rather than reliance on random ignition of richer mixtures) includes the lessening of pressure and temperature loadings on the containment and essential equipment. The staff's position is that the requirements of 10 CFR 50.34(f)(2)(ix) remain unchanged for evolutionary ALWRs.

The Commission has approved the staff's position that the requirements of 10 CFR 50.34 (f) (2) (ix) should remain unchanged for evolutionary plants. The staff should seek additional technical information, as suggested by the ACRS, and if reconsideration is warranted, the Commission should be advised

(2) Core-Concrete Interaction - Ability To Cool Core Debris

In the unlikely event of a severe accident in which the core has melted through the reactor vessel, it is possible that containment integrity could be breached if the molten core is not sufficiently cooled. In addition, interactions between the core debris and concrete can generate large quantities of additional hydrogen and other non-condensible gases, which could contribute to eventual overpressure failure of the containment.

The EPRI Requirements Document contains a number of design features that are intended to mitigate the effects of a molten core. To promote long-term debris coolability, it states that the cavity floor should be sized to provide 0.02 m^2/MWt. It also specifies that the containment should be designed to ensure adequate water supply to the floor and that an alternate means of introducing water into the containment, independent of normal and emergency ac power, should be provided. Passive schemes for providing flooding of the floor areas beneath the vessel are proposed and described in general terms for both BWRs and PWRs. EPRI also states that the steel shell or liner of the containment should be protected from core debris by at least 3 feet of concrete.

In its letter of March 20, 1990, Westinghouse agrees to comply with the EPRI core-debris dispersal criteria of 0.02 m^2/MWt and to incorporate a drain from the in-containment emergency water storage tank (EWST) to the reactor cavity of the RESAR SP/90 design to ensure flooding of the lower cavity in the event of a severe accident. This alternate water supply will be remotely actuated from Class 1E dc power supplies by manual operator action specified in emergency response guidelines.

The staff believes that an acceptable resolution to this issue can be provided by the evolutionary ALWR vendors if their designs

- provide sufficient reactor cavity floor space to enhance debris spreading
- provide for quenching debris in the reactor cavity

The staff is evaluating the specific cavity sizing criteria (0.02 m^2/MWt) proposed in the Requirements Document. The issue of debris coolability is an area in which there is active ongoing experimental research including relatively large scale testing jointly sponsored by EPRI and NRC. Additionally, without assurance of core debris coolability, the level of protection afforded by a 3-foot thickness of concrete and the issue of vessel pedestal attack (ablation of concrete supporting the reactor vessel by the molten core debris) require further evaluation. The staff will continue to evaluate the issue of core debris coolability and the specific cavity sizing criteria (0.02m^2/MWt) proposed by EPRI as more data and information becomes available. The staff intends to assess the debris flooding schemes proposed by EPRI on a design-specific basis. Design-specific approaches to resolve this issue will be evaluated by the staff on a case-by-case basis to ensure compliance with the above criteria.

The Commission has approved the staff's position.

(3) High-Pressure Core-Melt Ejection

One effect of a severe accident that could possibly result in containment failure is the phenomenon of direct containment heating (DCH). The staff is concerned that this phenomenon might occur from the ejection of molten-core debris under high pressure from the reactor vessel resulting in wide dispersal of core debris and extremely rapid addition of energy to the containment atmosphere.

To limit DCH, the Requirements Document states that the cavity/pedestal/drywell configuration should be designed to preclude entrainment of core debris by gases ejected from a failed reactor vessel. It also states that a safety-grade RCS safety depressurization and vent system will be provided. The staff review has concluded that reactor vessel depressurization capability and cavity design features to entrap ejected core debris constitute an acceptable approach to the issue of high-pressure core melt ejection.

In its letter of March 20, 1990, Westinghouse indicates that the configuration of the cavity of the RESAR SP/90 containment will prevent core debris from entering the upper containment. In addition, the RESAR SP/90 design includes safety-grade ac-independent pressurizer power-operated relief valves that have the capability to reduce the reactor coolant system pressure to less than 200 psig at reactor vessel failure for severe accident scenarios. The initiation of intentional depressurization will be a manual action that will be included in the plant emergency operating procedures.

The staff concludes that, during a high-pressure core-melt scenario, a depressurization system should provide a rate of RCS depressurization to preclude molten-core ejection and to reduce RCS pressure sufficiently to preclude creep rupture of steam generator tubes. Primary systems of evolutionary ALWRs should have the capability to be depressurized after loss of decay heat removal. In addition, the staff concludes that the Requirements Document should include a requirement that reactor cavities be arranged in such a manner that high-pressure core debris ejection resulting from vessel failure will not impinge on the containment boundary. The staff concludes that ALWR designs should include a depressurization system and cavity design features to contain ejected core debris.

The Commission has approved the staff's position that the evolutionary ALWR designs include a depressurization system and cavity design to contain core debris. The cavity design, as a mitigating feature, should not unduly interfere with operations including refueling, maintenance, or surveillance activities.

(4) Containment Performance

The containment function, i.e., maintenance of a strong leak tight barrier against radioactive release, is faced with distinct challenges as a result of a severe accident. These challenges may be roughly divided into two categories, energetic or rapid energy releases and slower, gradually evolving releases to the closed containment system. Examples of containment loadings that fall into the first category include high-pressure core-melt ejection with direct containment heating, hydrogen combustion, and the initial release of stored energy from the RCS. Slow energy releases to the containment are typified by decay heat and noncondensible gas generation. Engineering practice in containment design calls for passive capability in dealing with energetic energy releases, where practicable, while long-term energy releases may be controlled by both passive means as well as through active intervention.

In view of the low probability of accidents that would challenge the integrity of the containment, the staff concludes that the probability of failure of the mitigation systems (those systems that can reduce the consequences of a core damage accident), from the onset of core damage to loss of containment integrity resulting in an uncontrollable leakage substantially greater than the design basis leakage, should not exceed approximately 0.1. However, the staff intends to ensure that the containment can deal with all credible challenges and does not intend to apply this conditional containment failure probability (CCFP) guideline in a manner that could be interpreted to potentially detract from overall safety. The staff will accept a CCFP of 0.1 or a deterministic containment performance goal that offers comparable protection. For this reason, the staff concludes that the following general criterion for containment performance during a severe-accident challenge would be appropriate for the evolutionary ALWRs in place of a CCFP:

> The containment should maintain its role as a reliable leak-tight barrier by ensuring that containment stresses do not exceed ASME Service Level C limits for a minimum period of 24 hours following the onset of core damage and that following this 24-hour period the containment should continue to provide a barrier against the uncontrolled release of fission products.

Maintaining containment integrity for a minimum period (e.g., 24 hours) is based on providing sufficient time for the remaining airborne activity in the containment (principally noble gases and iodine) to decay to a level that would not exceed 10 CFR 100 dose guideline values when analyzed realistically, if controlled venting were to occur after that time. During this period, containment integrity should be provided, to the extent practicable, by the passive capability of the containment itself and any related passive design features (e.g., suppression pool). The staff further believes that following this period, the containment should continue to provide a barrier against the uncontrolled release of fission products. However, in keeping with the concept of allowing for intervention in coping with long-term or gradual energy release, the staff believes that after this minimum period, the containment design may use controlled, elevated venting to reduce the probability of a catastrophic failure of the containment. Alternatively, a design may use diverse containment heat removal systems or rely on the restoration of normal containment heat removal capability if sufficient time is available for major recovery actions (e.g., 48 hours).

EPRI has indicated that the ALWR publ safety criteria do not contain explicit criteria for conditional probability of containment failure or other mitigation features since the ALWR Steering Committee believes that such criteria could potentially distort the balance in safety design and inhibit innovative improvements in core protection features. However, EPRI has not yet indicated its position on an alternate containment performance goal.

Westinghouse has not yet committed to a specific containment performance goal for RESAR SP/90 design. However, in its letter dated March 20, 1990, Westinghouse indicates that the SP/90 design includes several features to minimize the potential for large fission product releases in the event of a severe accident. These features are primarily aimed at the prevention and/or mitigation of severe accident phenomena that can threaten containment integrity early in a severe accident scenario, such as direct containment heating, core-concrete interactions, and hydrogen deflagrations. In addition, the RESAR SP/90 design includes containment sprays and safety grade fan coolers to prevent containment pressurization for severe accident sequences in which those systems are available. It is expected that the mitigation features discussed by Westinghouse would lead to a CCFP of less than 0.1 for all credible accident scenarios.

Defense in depth, a long standing fundamental principle of reactor safety, results in the concept that multiple barriers should be provided to ensure against any significant release of radioactivity. In its Severe Accident Policy Statement, the Commission indicated that it "... fully expects that vendors engaged in designing new (or custom) plants will achieve a higher standard of severe accident safety performance than their prior designs." A defense-in-depth approach reflects an awareness of the need to make conservative safety judgments in the face of uncertainties; in other words, more than one method of protection should be used. In that regard, the reactor containment boundary should serve as a reliable barrier against fission product release for credible severe-accident phenomena/challenges. Special effort should be made to eliminate or further reduce the likelihood of a sequence that could bypass the containment. The continued reliance on the traditional principle of containment of fission products following an accident is seen as a logical and prudent approach to addressing reasonable questions that will persist regarding the ability to accurately predict certain aspects of severe accident behavior. In order to ensure balance between prevention and mitigation, some criteria on containment performance are appropriate. Accordingly, a general goal of limiting the conditional containment failure probability to less than 1 in 10 when weighted over credible core-damage sequences would constitute appropriate attention to the defense-in-depth philosophy. Alternatively, a deterministic containment performance goal that provides comparable protection would be appropriate.

Probabilistic risk assessment is a very powerful tool that permits systematic integrated assessment of design strengths and weaknesses. However, because very low frequency scenarios (approximately 1.0 X 10E-6 per reactor-year) are being addressed, it is important to recognize the large uncertainties in the quantification of these scenarios. The overall uncertainties in severe accident behavior are driven largely by insufficient data for assessing common-cause failures, difficulty in quantification of the potential for human errors, and questions about completeness of analyses and uncertainties in phenomenological behavior. For this reason, the staff considers it acceptable to utilize a deterministic containment performance criterion that would provide a level of containment performance comparable to that which could be demonstrated using a probabilistic containment failure goal of 0.1, given a severe accident.

The staff's position is to use a CCFP of 0.1 or a deterministic containment performance goal that offers comparable protection in the evaluation of evolutionary ALWRs.

The Commission has approved the use of a 0.1 CCFP as a basis for establishing regulatory guidance for evolutionary ALWRs. This objective should not be imposed as a requirement in and of itself. The use of the CCFP should not discourage accident prevention and the staff should review suitable alternative deterministically established containment performance objectives providing comparable mitigation capability if submitted by applicants. Any such alternatives should be submitted to the Commission following staff review.

(5) Equipment Survivability

With regard to the Commission's request concerning the measures to ensure that systems and equipment required only to mitigate severe accidents are available to perform their intended function (e.g., environmental qualifications), the staff believes that prevention and mitigation features provided for protection against severe accident consequences only (not required for design-basis accidents) need not be subject to (a) the 10 CFR 50.49 environmental qualification requirements, (b) all aspects of quality assurance requirements of Appendix B to 10 CFR 50, or (c) the redundancy/diversity requirements of Appendix A to 10 CFR 50. It is the staff's judgment that severe core-damage accidents should not be design-basis accidents (DBAs) in the traditional sense in which DBAs have been treated in the past.

Notwithstanding that judgment, however, mitigation features must be designed so there is reasonable assurance that they will operate in the severe accident environment for which they are intended and over the time span for which they are needed. In instances where safety-related equipment, (which is provided for DBAs) is relied upon to cope with severe accident situations; there also should be a high confidence that this equipment will survive severe accident conditions for the period that is needed to perform its intended function. However it is not necessary for redundant trains to be qualified to meet this goal.

During the review of a specific ALWR design the credible severe-accident scenarios, the equipment needed to perform mitigative functions, and the conditions under which the mitigative systems must function, will be identified. Equipment survivability expectations under severe accident conditions should include consideration of the circumstances of applicable initiating events (e.g., station blackout, earthquakes) and the environment (e.g., pressure, temperature, radiation) in which the equipment is needed to function. The required system performance criteria will be based on the results of these design-specific reviews. In addition, the staff concludes that severe-accident mitigation equipment for evolutionary ALWRs should be capable of being powered from an alternate power supply as well as from the normal Class 1E onsite systems. Appendices A and B to Regulatory Guide 1.155 provide guidance on the type of quality assurance activities and specifications that the staff concludes are appropriate for equipment utilized to prevent and mitigate the consequences of severe accidents.

Westinghouse commits to identify equipment used in the mitigation of severe accidents and to define a program to ensure that such equipment will perform reliably under severe accident conditions; this commitment will be implemented during the FDA stage of review.

The staff's position is that features provided only for severe-accident protection need not be subject to the environmental qualification requirements of 10 CFR 50.49, quality assurance requirements of Appendix B to 10 CFR 50, and redundancy/diversity requirements of Appendix A to 10 CFR 50.

The Commission has approved the staff's position.

20.2.4 Non-Severe Accident Issues

The following issues, which are not normally considered through PRA analysis or not considered as severe accident issues for the evolutionary ALWRs, but are addressed here because either the staff's positions or the vendor requests differ from past practices.

(1) Operating-Basis Earthquake (OBE)/Safe-Shutdown Earthquake (SSE)

Currently, 10 CFR 100 requires that the magnitude of the OBE be at least one-half that of the SSE. It has been an industry-wide experience that such a requirement leads to a design that is governed by the OBE requirements and produces unnecessary and inconsistent margins for the SSE loading. This requirement was included in the regulation when the staff did not have substantial experience with the seismic resistance of plants that incorporated OBE design at half the SSE value. Since then a number of research programs have been conducted including a large industry effort on testing and observation of actual earthquake experience of industrial facilities. Consequently, the NRC-funded Piping Review Committee concluded that the OBE criteria at existing plants are too high (thereby unnecessarily controlling the design of some safety systems) and recommended that the OBE be decoupled from the design-basis of the SSE. Certain interim measures, such as allowing somewhat higher damping values for piping analysis, have been taken to partially implement the Piping Review Committee recommendations (NUREG-1061, "Report of U.S. Nuclear Regulatory Commission Piping Review Committee," 1984). But the complete implementation of the recommendations would involve a revision of Appendix A to 10 CFR 100. Because of higher priority work, the effort on revision of this regulation has been postponed. It should be noted that the Commission has, in certain site-specific cases, previously approved OBEs of less than one-half the SSE.

EPRI has requested that NRC regulations be changed to reduce the magnitude of the OBE as it relates to the SSE as a basis for the design. All evolutionary ALWR vendors, including Westinghouse, agree with the request. EPRI has identified this as an optimization issue.

The staff agrees that the OBE should not control the design of safety systems. However, a staff position on this issue to be applied generically to all future designs has not yet been fully developed. For the evolutionary reactors, the staff will consider requests to decouple the OBE from the SSE on a design-specific basis. Such a decision would require an exemption to the Commission's regulations.

The Commission has approved the staff's position.

(2) Inservice Testing of Pumps and Valves

The ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," has been used to establish past testing requirements for ASME Code Class 1, 2, and 3 safety-related pumps and valves. These requirements provide certain information on the operational readiness of the components, but in general, do not necessarily verify the capability of the components to perform their intended safety function. It is the staff's judgment that the Code does not ensure the necessary level of component operability that is desired for the evolutionary ALWR designs. The staff believes that the aspects of pump and valve testing and inspection listed below are necessary to provide an adequate level of assurance of operability. The following provisions should be applied to all safety related pumps and valves and not limited to ASME Code Class 1, 2, and 3 components.

- Piping design should incorporate provisions for full-flow testing (maximum design flow) of pumps and check valves.

- Designs should incorporate provisions to test motor-operated valves under design-basis differential pressure.

- Check valve testing should incorporate the use of advanced non-intrusive techniques to address degradation and performance characteristics.

- A program should be established to determine the frequency necessary for disassembly and inspection of pumps and valves to detect unacceptable degradation that cannot be detected through the use of advanced, non-intrusive techniques.

The ACRS recommended that the requirements of Generic Letter 89-10 be applied to evolutionary ALWRs and that the staff resolve the issues of check valve testing and surveillance indicating how it will be applied to evolutionary ALWRs. The ACRS further recommended that the staff consider industry proposed alternative ways of meeting inservice and surveillance requirements. The staff agrees with these recommendations.

In its letter of March 20, 1990, Westinghouse described the design of integrated safeguards and emergency feedwater systems for the RESAR SP/90 that contain features that partially satisfy the staff's proposed measures to ensure operability of pumps and valves.

The Commission has approved the staff's position as supplemented by the staff's response to the ACRS recommendations. The Commission noted that consideration be given to the practicality of designing testing capability into the systems, particularly for large pumps and valves.

These issues and any others that arise from reviews of other ALWR designs will be addressed during the FDA stage of review.

21 REVIEW BY THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Advisory Committee on Reactor Safeguards (ACRS) completed its review of the application for a preliminary design approval (PDA) for the Westinghouse Reference Safety Analysis Report (RESAR) for the SP/90 nuclear power block (NPB) at its 367th meeting held November 8-10, 1990. A copy of the ACRS report dated December 12, 1990, is attached as Appendix F. The actions the staff has taken or plans to take in response to the committee's comments and recommendations are described in the paragraphs below.

(1) The committee stated that, in view of the many open items and its concerns regarding the SER and the many unresolved severe accident issues, the conclusions stated in the draft SER were stated too strongly and should be revised.

The committee indicated this view to the staff during an ACRS subcommittee meeting held on September 20, 1990. In response to the committee's recommendation, the staff revised the SER and provided the revised conclusions section to the subcommittee chairman.

(2) The committee expressed the belief that some portions of the SER do not meet the standards that should be applied to a current SER. In addition, the committee stated that this problem appeared to be caused by the staff's reliance on the July 1981 Standard Review Plan (SRP) (NUREG-0800) and that the SRP needs to be updated to reflect the current situation for the licensing of ALWRs.

In its review of the RESAR SP/90 design, the staff used the SRP of July 1981, which was the last revision to the SRP as a whole, specific sections have been updated by individual technical branches as recently as 1990. However, the staff does agree with the committee on the need to update the SRP to current standards. The staff has initiated efforts in the past few months to upgrade the SRP. During the upgrade process, the staff will identify the standards that should be applied to current SERs to further define the standard form and content of future ALWR reviews.

The staff believes that the committee reached this conclusion partly because, although the information contained in the SER accurately portrayed the information contained in the RESAR SP/90 submittal, this information did not necessarily reflect what Westinghouse had committed to change in its design. This problem arose because Westinghouse made commitments in its responses to open items in areas such as fire protection but did not amend its submittal to show a new design. Although the staff's review may appear to be outdated, the SER is accurate in describing the content of the current submittal. The staff believes that there is reasonable assurance that the activities authorized by licenses or permits referencing the RESAR SP/90 design, subject to the approval of the balance of plant design and the resolution of the open items described in the SER, can be conducted without endangering the health and safety of the public.

(3) The committee stated that the staff should have developed improved standards for the review of computer-based integrated protection systems, particularly the verification and validation of the software employed with such systems. The committee recommended that the staff seek additional personnel with expertise in computer science so that appropriate standards can be developed for the review of computer-based reactor protection systems.

In its review of the RESAR SP/90, the Instrumentation and Control Systems Branch used Revision 1 to Chapter 7 of the SRP, dated February 1984, and determined that the design meets all of the current criteria for instrumentation and control systems important to safety as detailed in Table 7-1 of the SER. In addition, to this list of criteria, the staff also required Westinghouse to conform to Regulatory Guide 1.152, "Criteria for Programmable Digital Computer System Software In Safety-Related Systems of Nuclear Power Plants." Regulatory Guide 1.152 endorses ANSI/IEEE-ANS-7.4.3.2, "Application Criteria for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations," 1982. ANSI/IEEE-ANS-7.4.3.2 is the only software verification and validation standard that is endorsed by the staff for the review and acceptance of microprocessor-based systems similar to RESAR SP/90.

The staff's review of RESAR SP/90 instrumentation and control systems began in late 1984 and was finished in April 1988. During that time, the staff participated in the Institute of Electrical and Electronics Engineers (IEEE) Standard Working Group SC-6.4 (verification and validation). In November 1984, the staff endorsed the conclusions of that group with Regulatory Guide 1.152. The staff is also concerned about software verification and validation and the need to improve standards, and continues to participate in the development of standards and the revision of ANSI/IEEE-ANS-7.4.3.2.

Additionally, the staff has begun a program to develop a standard related to digital systems and software development. The staff has identified 17 issues that should be addressed in the program. These issues include the testing of critical software (reverse engineering and algebraic specification) and the use of artificial intelligence and expert systems.

The staff also believes that it needs additional personnel with expertise in computer science. The staff has hired and continues to recruit additional staff members who are familiar with digital and microprocessor-based systems.

(4) The committee stated that Westinghouse committed to follow applicable codes, standards, and regulatory guides for materials used in the fabrication of pressure boundary components. However, many of these codes, standards, and regulatory guides that Westinghouse committed to follow are not representative of current industry practice. Westinghouse has developed internal specifications for pressure boundary materials that presumably do reflect current industry practice but did not submit these for the staff's review.

The staff is undertaking an effort to update the regulatory guides to include the most current applicable codes and standards. During the time that this effort is in progress, current reviews will rely on the existing guidance. However, in recognition of the concern stated by the ACRS, the staff will continue to accept deviations to existing guidance when the applicant can demonstrate that the methods used are equivalent to or better than those contained in the current regulatory guidance.

(5) The committee stated that the staff's review of the water displacer rods (WDRs) was not complete enough even though Westinghouse submitted appropriate information. Specifically, the committee noted that the SER does not discuss the pressure boundary integrity or the potential for reactivity insertion accidents for the WDRs. The committee further stated that new features of this kind should be thoroughly reviewed at an early stage of review.

The staff believes that it adequately reviewed the information provided in the RESAR SP/90 application related to the WDRs, based upon the stage of development of the design, and included the results of that review in the SER. The standards for SER form and content result in evaluations that are separated by review disciplines rather than by single systems or components. For example, the evaluations of American Society of Mechanical Engineers (ASME) Code Class components and reactor coolant pressure boundary integrity are found in Sections 3 and 5 of the SER and were provided in RESAR SP/90 modules 5 and 7.

Westinghouse described the reactor coolant pressure boundary integrity and construction materials for the WDRs under the heading of Displacer Rod Drive Mechanisms (DRDMs).

The reactivity effects of the WDRs are described in Section 4 of the SER in relation to normal operation and in Section 15 of the SER related to reactivity and power distribution anomalies. The references to the WDRs in these instances are brief and essentially stated that the conditions that may be caused by WDR anomalies are bounded by the more reactive control rods.

(6) The committee stated that it had found errors and inconsistencies in the SER. The committee concluded that Westinghouse should review the SER to identify and correct any additional errors.

During its meeting of September 20, 1990, the ACRS Subcommittee on Advanced Pressurized Water Reactors identified three apparent inconsistencies in the SER. The staff's investigation revealed that the inconsistencies arose because Westinghouse had made committments regarding fire protection that were not addressed by the design documentation that the technical branches reviewed. The staff revised the SER to address these inconsistencies and initiated two new open items. The staff submitted the revised portions of the SER to the subcommittee chairman.

Before the SER is published, the staff will request a review of the SER by Westinghouse to ensure that material proprietary to Westinghouse has not been included in the SER. The staff practice does not include a technical review of an SER by the applicant before publication.

(7) The committee reviewed the draft PDA document and stated its concern that the PDA does not describe how the preliminary design information would be used in a future application for final design approval (FDA) and/or design certification. In addition, the committee stated that the requirement in the PDA that the staff and the ACRS rely upon this information in their reviews of any individual facility construction permit (CP) application could result in problems for the staff in a contested CP application.

On the advice of the Office of the General Counsel (OGC), the staff has added a paragraph to the draft PDA that states this PDA is not a certified design within the meaning of 10 CFR Part 52. In addition to restricting the use of the PDA to individual facility construction permits, this statement should satisfactorily separate this PDA from future applications under 10 CFR Part 52.

The requirement in the PDA that the staff and the ACRS rely upon this information in their review of "any individual facility license application" is taken from 10 CFR Part 52, Appendix O, which details the requirements upon completion of the review and was amended to say "individual facility construction permit applications" to comply with the conditions of the draft PDA. The staff is familiar with this language and does not foresee problems in a future CP application. Therefore, the staff is also satisfied with the OGC advice that this requirement would apply only to reviews of a CP application and that both the staff and the ACRS would be able to revisit any issue in their review of any application that would lead to an operating license.

(8) The committee expressed two concerns related to the SP/90 design features. First, the emergency diesel generators (EDGs) would be located on the same floor and corridor as the main control room, although the operation of these EDGs has a potential for fire and/or explosions. Second, the committee recommended timely development of a Commission position on the cavity floor area beneath the reactor vessel because the requirements could significantly affect the containment sizing and layout.

The staff has not in the past considered, and does not now consider, credible an explosion in the EDG room of sufficient size to cause catastrophic failure of the reinforced concrete enclosures of these rooms. In addition, the applicant has committed to ensure that a fire in any area of the plant (including the EDG room) will not adversely affect the ability to bring the plant to safe shutdown. The staff will perform a final review at the FDA stage to ensure that this design commitment is met by the applicant.

The ACRS concern has been provided to the applicant, by copy of the ACRS letter, in the belief that any improvements providing potential benefit should be included in the plant design once they have been identified.

The staff agrees with the committee on the importance of reaching a timely decision on the issue of the cavity floor area and has undertaken several initiatives to bring this to closure. The ACRS will be kept informed of the staff's progress and will be an integral part of reaching a satisfactory conclusion.

In their letter to the EDO dated February 12, 1991, the ACRS stated that the staff's response to the committees concern over the location of the emergency diesel generator (EDG) did not address the large door that separates the EDG from the corridor leading to the control room. The committee asked that the staff expand their reply to include consideration of this door and provide views on the size of a fire or explosion that the staff would consider credible. The committee also requested that the staff estimate the structural capability of this door under differential pressure conditions and address the issue of a fire that could result from combustibles such as fuel oil that may flow under the door and into the corridor. A copy of this letter is attached in Appendix F.

As part of the review of the preliminary SP/90 design, the staff has not determined the size of the explosion that could occur in the EDG room and the effect of the postulated explosion on the structural integrity of the door between the EDG room and the corridor leading to the control room. If further development of the SP/90 design or any other future designs specify the EDG at that location, the staff would consider several aspects of the design in order to preclude breaching the entrance to the building corridor.

In such a future design, the staff would examine the ventilation intakes and exhausts to the EDG room for rigidity and surface area to determine their ability to relieve pressure during a maximum credible explosion. The intakes and exhausts also would be examined to determine if they would allow an explosive gas mixture to accumulate even if the fans fail. If the design could not ensure that the EDG room doors do not fail, the staff would require additional design changes, including placing the EDGs in a hardened enclosure outside of the reactor auxiliary building.

The SP/90 preliminary design submittal did not provide for postulated fuel oil leaks inside the EDG rooms. However, any further development of the design should provide for postulated fuel oil leaks, the consequences associated with those leaks, and mitigating design techniques such as day tank alarms and sump alarms. The existing design also does not clearly specify the use of curbing to prevent oil leakage from contacting the closure door seals. The staff would review this issue if the design is advanced beyond the current preliminary stage.

The staff also would review future detailed designs to determine the manner in which each of the following would affect the plant's ability to respond to such events: the automatic fire detection and suppression capability provided in the EDG room, the relatively low volatility of diesel fuel, and the amount of diesel fuel that may be available for fire and explosion.

In developing the design, Westinghouse placed the EDG room to comply with the wishes of its Japanese partners early in the design process. Westinghouse has informed the staff that the Japanese no longer use this design concept and that future Westinghouse designs will not place the EDG rooms on the control room corridor. Both Westinghouse and the Japanese now prefer separate, hardened enclosures. On the basis of this information and the concerns that the committee previously supplied to Westinghouse, the staff expects that future submittals will not place the EDGs near the control room.

The staff will review future designs and any undated SP/90 submittals to ensure that the EDGs do not endanger the control room or other safety-related equipment rooms.

22 COMMON DEFENSE AND SECURITY

Common defense and security will be addressed during the plant-specific licensing process of an application referencing the RESAR SP/90 design.

23 FINANCIAL QUALIFICATIONS

The financial qualifications of the applicable utility will be addressed during the plant-specific licensing process of an application referencing the RESAR SP/90 design.

24 FINANCIAL PROTECTION AND INDEMNITY REQUIREMENTS

The indemnity requirements of 10 CFR 140 will be addressed during the plant-specific licensing process of an application referencing the RESAR SP/90 design.

25 CONCLUSIONS

On the basis of its evaluation of the RESAR SP/90 preliminary design as set forth in the preceding sections of this SER, the staff concludes, subject to the conditions discussed in this report and for the porti of the nuclear power block (NPB) design covered by RESAR SP/90 applica on, that

- Westinghouse has adequately described, analyzed, and evaluated the proposed design including, but not limited to, the principal engineering criteria for the design; the interface information necessary to ensure compatibility between the submitted design and the balance of the nuclear power block, 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. Westinghouse has acceptably identified the major features and components incorporated in its design for the protection of the health and safety of the public.

- Westinghouse will be required to provide additional technical or design information as may be required to complete the safety evaluation during the FDA stage of review. The staff concludes that open issues identified throughout this report are not of a nature as to prevent issuance of a PDA since they can be resolved during the review for an individual facility construction permit application or during the FDA stage of review.

- Safety features or components that require research and development have been identified by Westinghouse and have been described. Westinghouse will be required to conduct research and development programs reasonably designed to resolve safety questions associated with those safety features or components in support of its FDA application.

- There is reasonable assurance that the activities authorized by licenses or permits referencing the RESAR SP/90 design, subject to the approval of the balance of plant design and the resolution of the open items described in this SER can be conducted without endangering the health and safety of the public; therefore, the issuance of a PDA in accordance with 10 CFR 52 Appendix O will not be inimical to the common defense and security or to the health and safety of the public.

- A PDA issued on the basis of this SER should be limited for reference ability to individual facility applications for construction permits and has no basis for reference ability in applications made for final design approval or design certification under 10 CFR 52.

APPENDIX A

CHRONOLOGY

October 24, 1983	Letter from applicant submitting proprietary and nonproprietary versions of RESAR SP/90 Module 1, "Primary Side Safeguards System," and the application for a preliminary design approval (PDA) of the RESAR SP/90 design.
November 16, 1983	Letter from applicant clarifying the PDA application of October 23, 1984, for RESAR SP/90.
November 30, 1983	Letter from applicant submitting proprietary and nonproprietary versions of RESAR SP/90 PDA Module 2, "Regulatory Conformance."
January 25, 1984	Letter to applicant regarding RESAR SP/90 PDA application.
January 30, 1984	Letter from applicant submitting comments on NUREG-1070, "Severe Accident Program for Nuclear Power Plant Regulation."
February 9, 1984	Letter from applicant responding to an NRC letter of January 25, 1984, regarding standardization and severe accident considerations.
May 2, 1984	Letter to applicant regarding relationship among PDAs, severe accident policy, and standardization policy.
May 7, 1984	Letter from applicant submitting additional proprietary and nonproprietary copies of RESAR SP/90 Module 1.
May 16, 1984	Letter from applicant submitting proprietary and nonproprietary versions of RESAR SP/90 Module 3, "Introduction and Site," and an application for a PDA.
May 30, 1984	Meeting with applicant to discuss review of RESAR SP/90 modules for PDA (summary dated June 5, 1984).
June 12, 1984	Letter to applicant granting proprietary classification for RESAR SP/90 Module 1.
June 12, 1984	Letter to applicant granting proprietary classification for RESAR SP/90 Module 2.
June 12, 1984	Letter to applicant granting proprietary classification for RESAR SP/90 Module 3.

June 18, 1984	Letter to applicant accepting application for PDA for formal docketing.
July 11, 1984	Letter from applicant submitting proprietary and nonproprietary versions of RESAR SP/90 Module 4, "Reactor Coolant System."
July 23, 1984	Letter from applicant submitting proprietary version of errata to RESAR SP/90 Module 4.
July 31, 1984	Letter from applicant submitting proprietary and nonproprietary versions of RESAR SP/90 Module 5 (Volumes 1 and 2), "Reactor System."
August 1, 1984	Letter to applicant granting proprietary classification for RESAR SP/90 Module 4.
August 2, 1984	Letter from applicant concerning consolidation of RESAR SP/90 Modules 6 and 8.
August 3, 1984	Meeting with applicant to discuss scope of probabilistic risk assessment (PRA) associated with RESAR SP/90 (summary dated August 15, 1984).
August 16, 1984	Letter to applicant granting proprietary classification for RESAR SP/90 Module 5.
August 24, 1984	Letter from applicant regarding scope and schedule for PRA.
August 27, 1984	Letter to applicant requesting additional information on RESAR SP/90 Module 2.
September 12, 1984	Letter to applicant requesting additional information on RESAR SP/90 Module 3.
October 4, 1984	Letter to applicant providing comments on RESAR SP/90 Module 2.
October 17, 1984	Letter to applicant requesting additional information on RESAR SP/90 Module 5.
November 9, 1984	Letter from applicant submitting proprietary and nonproprietary versions of RESAR SP/90 Module 6/8, "Secondary Sides Safeguard System/Steam Power Conversion System."
November 16, 1984	Meeting with applicant to discuss review schedule of PRA (summary dated November 28, 1984).
November 30, 1984	Letter from applicant submitting proprietary and nonproprietary versions of Amendment 1 to RESAR SP/90 Module 5.
November 30, 1984	Letter to applicant granting proprietary classification for RESAR SP/90 Module 6/8.

December 14, 1984	Letter to applicant requesting additional information on Section 5.4.7 of RESAR SP/90 Module 1.
December 24, 1984	Letter to applicant requesting additional information on Section 6.3 of RESAR SP/90 Module 1.
January 30, 1985	Letter from applicant submitting proprietary and nonproprietary versions of Amendment 1 to RESAR SP/90 Module 3.
January 31, 1985	Letter from applicant submitting proprietary and nonproprietary versions of RESAR SP/90 Module 7, "Structural/ Equipment Design."
January 31, 1985	Letter to applicant concerning report by Fire Protection Policy Steering Committee (Generic Letter 85-01).
February 13-14, 1985	Meeting with applicant to discuss request for additional information (RAI) on RESAR SP/90 Module 1 (summary dated March 22, 1985).
February 26, 1985	Letter to applicant granting proprietary classification for Amendment 1 to RESAR SP/90 Module 5.
February 27, 1985	Letter from applicant submitting proprietary and nonproprietary versions of RESAR SP/90 Module 9, "Instrumentation & Controls and Electric Power."
March 7, 1985	Meeting with applicant to discuss PRA methodology for RESAR SP/90 (summary dated March 20, 1985).
March 15, 1985	Letter from applicant transmitting proprietary versions of RESAR SP/90 modules to Brookhaven National Laboratory.
April 16, 1985	Letter from applicant submitting proprietary and nonproprietary versions of Amendment 2 to RESAR SP/90 Module 5.
June 27, 1985	Letter to applicant requesting additional information on Section 3.2 of RESAR SP/90 Module 7.
June 28, 1985	Letter from applicant submitting proprietary and nonproprietary versions of RESAR SP/90 Module 16 (Volumes 1 and 2), "Probabilistic Safety Study."
July 10, 1985	Letter to applicant granting proprietary classification to Amendment 1 of RESAR SP/90 Module 3.
July 10, 1985	Letter to applicant granting proprietary classification to RESAR SP/90 Module 7.
July 12, 1985	Letter from applicant submitting proprietary and nonproprietary versions of RESAR SP/90 Module 11, "Radiation Protection."
August 2, 1985	Letter to applicant granting proprietary classification to RESAR SP/90 Module 9.

August 5, 1985	Letter to applicant granting proprietary classification to Amendment 2 of RESAR SP/90 Module 5.
August 13, 1985	Letter to applicant granting proprietary classification to RESAR SP/90 Modules 11 and 16.
August 20, 1985	Letter to applicant regarding interpretation of the Severe Accident Policy Statement for RESAR SP/90.
August 23, 1985	Letter to applicant requesting additional information on RESAR SP/90 Module 1.
September 13, 1985	Letter from applicant submitting proprietary and nonproprietary versions of RESAR SP/90 Module 16 (Volumes 3 and 4), "Probabilistic Safety Study."
September 20, 1985	Letter from applicant submitting proprietary and nonproprietary versions of RESAR SP/90 Module 12, "Waste Management."
September 24, 1985	Meeting with applicant to discuss RAI on RESAR SP/90 Module 1.
October 2, 1985	Letter to applicant requesting additional information on RESAR SP/90 Module 11.
March 14, 1986	Letter from applicant submitting proprietary and nonproprietary versions of Amendment 1 to Module 11.
April 2, 1986	Letter to applicant requesting additional information on RESAR SP/90 Module 9.
April 16, 1986	Letter to applicant requesting additional information on RESAR SP/90 Module 6/8.
April 23-24, 1986	Meeting with applicant to discuss RESAR SP/90 Module 7.
May 28, 1986	Letter to applicant requesting additional information on RESAR SP/90 Module 16.
May 28, 1986	Letter from applicant transmitting upper calandria design drawings to Brookhaven National Laboratory (BNL).
May 30, 1986	Letter from applicant submitting proprietary and nonproprietary versions of Amendment 1 to RESAR SP/90 Module 7.
June 4, 1986	Letter from applicant submitting proprietary and nonproprietary versions of Amendment 3 to RESAR SP/90 Module 5.

June 9 1986	Letter from applicant submitting proprietary and nonproprietary versions of Amendment 1 to RESAR SP/90 Module 1.
July 8, 1986	Meeting with applicant to discuss outline and format for RESAR SP/90 Module 15 (summary dated July 29, 1986).
July 11, 1986	Letter to applicant requesting additional information on RESAR SP/90 Module 16.
July 14, 1986	Letter from applicant submitting proprietary and nonproprietary versions of Amendment 1 to RESAR SP/90 Module 9.
July 19, 1986	Letter to applicant granting proprietary classification to Amendment 1 of RESAR SP/90 Module 1.
August 5, 1986	Letter to applicant granting proprietary classification to Amendment 3 of RESAR SP/90 Module 5.
August 25, 1986	Letter from applicant submitting proprietary and nonproprietary versions of Amendment 1 to RESAR SP/90 Module 16.
August 29, 1986	Letter from applicant submitting proprietary and nonproprietary versions of Amendment 2 to RESAR SP/90 Module 9.
September 11, 1986	Meeting with applicant to discuss content for RESAR SP/90 Module 15 (summary dated November 3, 1986).
September 25, 1986	Meeting with Advisory Committee on Reactor Safeguards (ACRS) Subcommittee on advanced reactors to discuss status of design of RESAR SP/90.
October 27, 1986	Letter from applicant submitting proprietary and nonproprietary versions of RESAR SP/90 Module 15, "Control Room/ Human Factors Engineering."
October 29, 1986	Letter from applicant submitting proprietary and nonproprietary versions of RESAR SP/90 Module 13, "Auxiliary Systems."
October 30, 1986	Letter from applicant submitting proprietary and nonproprietary versions of Amendment 2 to RESAR SP/90 Module 16.
November 3, 1986	Letter from applicant submitting proprietary design drawings for use in review of RESAR SP/90 Module 16.
November 18, 1986	Letter from applicant submitting proprietary and nonproprietary versions of Amendment 1 to RESAR SP/90 Module 6/8.
January 2, 1987	Letter to applicant granting proprietary classification to Amendment 2 of RESAR SP/90 Module 9.
January 15, 1987	Letter from ACRS regarding improved safety for future light water reactors (LWRs).

January 27, 1987	Letter to applicant granting proprietary classification to Amendment 1 of RESAR SP/90 Module 6/8.
February 10, 1987	Letter to applicant granting proprietary classification to Amendment 2 of RESAR SP/90 Module 16.
March 9, 1987	Letter from applicant submitting proprietary and nonproprietary versions of RESAR SP/90 Module 10, "Containment Systems."
April 30, 1987	Letter to applicant transmitting draft reports on results of reviews of probabilistic safety study (PSS).
May 13, 1987	Letter to applicant granting proprietary classification to RESAR SP/90 Module 10.
June 1, 1987	Letter from applicant submitting slides for plant design overview meeting.
June 3, 1987	Meeting with applicant to discuss plant design overview (summary dated June 10, 1987).
June 9, 1987	Letter from applicant regarding review schedule for RESAR SP/90.
June 15, 1987	Letter from applicant providing comments on the BNL draft report regarding the PSS (front-end or core melt frequency evaluation).
June 19, 1987	Letter to applicant regarding review approach and schedule.
June 22, 1987	Letter to applicant providing notification of NRC reorganization.
June 30, 1987	Letter to applicant requesting additional information on the chemical engineering aspects of the design.
July 6, 1987	Letter to applicant requesting additional information on the sabotage protection features of the design.
July 24, 1987	Letter to applicant requesting additional information on the PSS (back-end or consequence analysis through release to environment).
July 30, 1987	Letter to applicant requesting additional information on the hydrological aspects of the design.
July 30, 1987	Letter to applicant granting proprietary classification to the June 15, 1987 comments on PSS.
August 12, 1987	Letter from applicant providing comments on the January 15, 1987 ACRS letter regarding improved safety for future LWRs.

September 3, 1987	Meeting with applicant to discuss general background information regarding the RESAR SP/90 design (summary dated September 24, 1987).
September 16, 1987	Letter from applicant providing comments on the draft BNL report regarding the PSS (back-end).
September 22, 1987	Letter to applicant requesting additional information on the structural engineering aspects of the design.
September 29, 1987	Letter to applicant regarding the relationship between the operating-basis earthquake (OBE) and the safe-shutdown earthquake (SEE).
October 1, 1987	Letter to applicant granting proprietary classification to the September 16, 1987 comments on the PSS.
October 8, 1987	Letter from applicant submitting proprietary and nonproprietary versions of Amendment 1 to RESAR SP/90 Module 2.
October 8, 1987	Letter from applicant submitting proprietary and nonproprietary versions of Amendment 3 to RESAR SP/90 Module 16.
October 23, 1987	Letter to applicant requesting additional information regarding the electrical engineering aspects of the design.
October 26, 1987	Letter to applicant requesting additional information regarding the materials engineering aspects of the design.
October 26, 1987	Letter from applicant submitting proprietary and nonproprietary versions of Amendment 2 to RESAR SP/90 Module 3.
November 6, 1987	Meeting with ACRS to discuss its letter of January 15, 1987, regarding improved safety for future LWRs.
November 19, 1987	Meeting to discuss Westinghouse advanced control room design (summary dated December 11, 1987).
November 23, 1987	Letter from applicant submitting proprietary and nonproprietary versions of Amendment 2 to RESAR SP/90 Module 1 and Amendment 1 to RESAR SP/90 Module 13.
December 1, 1987	Meeting with the Commission to discuss status of the review of the RESAR SP/90 application.
December 7, 1987	Letter to applicant granting proprietary classification to Amendment 1 to RESAR SP/90 Module 2.
December 7, 1987	Letter to applicant granting proprietary classification to Amendment 3 to Module 16.
December 7, 1987	Letter to applicant granting proprietary classification to Amendment 2 to RESAR SP/90 Module 3.

December 9, 1987	Letter to applicant requesting additional information regarding mechanical engineering aspects of the design.
December 17, 1987	Letter to applicant requesting additional information regarding plant systems aspects of the design.
December 17, 1987	Letter from applicant submitting proprietary and nonproprietary versions of Amendment 3 to RESAR SP/90 Module 9.
December 29, 1987	Letter to applicant concerning proprietary material provided during meeting on November 19, 1987.
December 29, 1987	Letter to applicant granting proprietary classification to Amendment 2 to RESAR SP/90 Module 1 and to Amendment 1 to RESAR SP/90 Module 13.
January 7, 1988	Letter from applicant submitting proprietary and nonproprietary versions of Addenda 1 and 2 to RESAR SP/90 PDA application.
January 12, 1988	Letter to applicant granting proprietary classification to Amendment 3 to RESAR SP/90 Module 9.
February 11, 1988	Letter from applicant responding to question (Q) 210.51.
February 11, 1988	Letter to applicant requesting additional information regarding fire protection features of the design.
February 19, 1988	Letter to applicant concerning Section 11.5 of RESAR SP/90 Module 12.
February 27, 1988	Letter to applicant requesting additional information on severe accident design goals of the RESAR SP/90.
March 2, 1988	Letter to applicant requesting additional information regarding the reactor systems aspects of the design.
March 15, 1988	Letter to applicant providing two additional requests for information regarding the reactor systems aspects of the design.
March 15, 1988	Letter to applicant informing Westinghouse of physical relocation of the Office of Nuclear Reactor Regulation.
March 21, 1988	Letter to applicant forwarding draft SER regarding the "front end" (core melt frequency evaluation) portion of the probabilistic risk assessment (PRA).
March 22, 1988	Letter from applicant submitting proprietary and nonproprietary versions of Amendment 1 to RESAR SP/90 Module 12.
March 22, 1988	Letter from applicant submitting proprietary and nonproprietary versions of Amendment 2 to RESAR SP/90 Module 2.

March 24, 1988	Letter from applicant submitting proprietary and nonproprietary versions of Amendment 2 to RESAR SP/90 Module 13.
March 28, 1988	Letter to applicant forwarding draft Brookhaven National Laboratory (BNL) report concerning the PRA.
March 31, 1988	Meeting with applicant to discuss review results of PRA (summary dated April 21, 1988).
April 6, 1988	Letter from Electric Power Research Institute (EPRI) concerning the EPRI advanced light-water reactor (ALWR) public safety criteria.
April 6, 1988	Meeting with Advisory Committee on Reactor Safeguards (ACRS) to discuss review results of the PRA.
April 14, 1988	Letter to applicant concerning review schedule for the RESAR SP/90 application.
April 14, 1988	Letter to applicant requesting additional information regarding the quality assurance aspects of the design.
April 14, 1988	Letter to applicant requesting additional information regarding the radiological consequences of design-basis accidents.
April 20, 1988	Letter from applicant submitting proprietary and nonproprietary versions of Amendment 2 to RESAR SP/90 Module 13.
April 21, 1988	Meeting with applicant to discuss reactor systems concerns (summary dated April 27, 1988).
April 25, 1988	Letter to applicant requesting additional information regarding the radiological consequences of design-basis accidents.
April 28, 1988	Letter to applicant requesting that RESAR SP/90 Module 2 be updated.
May 13, 1988	Letter from applicant submitting proprietary and nonproprietary versions of Amendments 1a and 3 to RESAR SP/90 Module 1.
May 13, 1988	Letter from applicant requesting withdrawal of April 20, 1988, version of Amendment 2 to RESAR SP/90 Module 13.
May 13, 1988	Letter from applicant submitting Addenda 3 and 5 to RESAR SP/90.
May 13, 1988	Letter from applicant submitting proprietary and nonproprietary versions of Amendment 1a to RESAR SP/90 Module 3.
May 13, 1988	Letter from applicant submitting proprietary and nonproprietary versions of Amendment 1 to RESAR SP/90 Module 4.

May 13, 1988	Letter from applicant submitting proprietary and nonproprietary versions of Amendment 3 to RESAR SP/90 Module 13.
May 13, 1988	Letter from applicant submitting Addendum 4 to RESAR SP/90.
May 23, 1988	Letter from applicant submitting Addendum 7 to RESAR SP/90.
May 27, 1988	Letter from applicant submitting Addendum 8 to RESAR SP/90.
June 8, 1988	Letter to applicant granting proprietary classification to Amendment 2 to RESAR SP/90 Module 2.
June 8, 1988	Letter to applicant granting proprietary classification to Amendment 2 to RESAR SP/90 Module 13.
June 8, 1988	Letter to applicant granting proprietary classification to Amendment 1 to RESAR SP/90 Module 12.
June 8, 1988	Letter to applicant granting proprietary classification to Amendments 1a and 3 to RESAR SP/90 Module 1.
June 8, 1988	Letter to applicant granting proprietary classification to Westinghouse's presentation to the ACRS.
June 8, 1988	Letter to applicant granting proprietary classification to Amendment 3 to RESAR SP/90 Module 13.
June 10, 1988	Letter to applicant transmitting portions of the Draft Safety Evaluation Report (DSER) of the RESAR SP/90 design.
June 14, 1988	Letter from applicant submitting Addendum 9 to RESAR SP/90.
June 16, 1988	Meeting with applicant to discuss the quality assurance program for RESAR SP/90 (see summary dated June 30, 1988).
June 23, 1988	Letter from applicant submitting Addendum 10 to RESAR SP/90.
July 1, 1988	Letter to applicant concerning review of Revision 11/7 of the Westinghouse Quality Assurance Topical.
July 6, 1988	Letter from applicant submitting Addendum 6 to RESAR SP/90.
July 27, 1988	Letter from applicant submitting proprietary and nonproprietary versions of Amendment 2 to RESAR SP/90 Module 11.
August 5, 1988	Letter from applicant submitting proprietary and nonproprietary versions of Amendment 4 to RESAR SP/90 Module 13.
August 25, 1988	Letter from applicant submitting proprietary and nonproprietary versions of Amendment 3 to RESAR SP/90 Module 3.
September 19, 1988	Letter from applicant submitting proprietary and nonproprietary versions of Amendment 4 to RESAR SP/90 Module 1.

October 6, 1988	Letter from applicant submitting Revision 11/7 to the Westinghouse Quality Assurance Topical.
October 25, 1988	Letter from applicant responding to quality assurance concerns discussed in the DSER of June 10, 1988.
October 26, 1988	Letter from applicant submitting proprietary and nonproprietary versions of Amendment 2 to RESAR SP/90 Module 4.
October 27, 1988	Memorandum concerning review of resolution of unresolved safety issues/generic safety issues (USIs/GSIs) for RESAR SP/90.
November 8, 1988	Letter from applicant submitting proprietary and nonproprietary versions of Amendment 4 to RESAR SP/90 Module 5.
November 22, 1988	Letter to applicant concerning scope of design and staff review for the RESAR SP/90.
January 18, 1989	Letter to applicant granting proprietary classification to Amendment 4 to RESAR SP/90 PDA Module 13, "Auxiliary Systems."
January 18, 1989	Letter to applicant granting proprietary classification to Amendment 4 to RESAR SP/90 PDA Module 1,"Primary Side Safeguards System."
January 18, 1989	Letter to applicant granting proprietary classification to Amendment 2 to RESAR SP/90 PDA Module 12, "Waste Management," and Amendment 2 to Module 11, "Radiation Protection."
January 18, 1989	Letter to applicant granting proprietary classification to Amendment 2 to RESAR SP/90 PDA Module 4, "Reactor Coolant System."
January 18, 1989	Letter to applicant granting proprietary classification to Amendment 3 to RESAR SP/90 PDA Module 3, "Introduction and Site."
February 2, 1989	Letter to applicant requesting additional information on GSIs 70, 105, 113 and 135 and staff questions 730.1-730.4.
February 14, 1989	Letter from applicant submitting Amendments 1a and 2 to RESAR SP/90 PDA Module 7, "Structural/Equipment Design."
March 9, 1989	Letter to applicant transmitting portions of the DSER concerning RESAR SP/90.
May 8, 1989	Meeting with applicant to discuss vendors standardized plant programs (summary dated June 12, 1989).

June 9, 1989	Letter from applicant submitting response to NRC DSER open issues 42-81.
June 28, 1989	Letter from applicant submitting response to NRC DSER open issues 1-41.
July 6, 1989	Letter to applicant granting proprietary classification to Amendment 1a and 2 to RESAR SP/90 PDA Module 7, "Structural/Equipment Design."
July 14, 1989	Meeting with applicant to discuss severe accident and major licensing issues for RESAR SP/90 (summary dated October 19, 1989).
August 29, 1989	Letter to applicant granting proprietary classification to their response to staff questions 220.1-220.9 and 252.1-252.14. Addendum 1 and 2 to RESAR SP/90 PDA application.
September 12, 1989	Letter to applicant granting proprietary classification to responses to DSER open issues 1-41.
September 28, 1989	Meeting with Advisory Committee on Reactor Safeguards (ACRS) Subcommittee on Advanced Pressurized Water Reactors to discuss review status and design details of RESAR SP/90.
October 3, 1989	Letter from applicant submitting Amendment 3 to RESAR SP/90 PDA Module 2, "Regulatory Conformance."
October 17, 1989	Letter from applicant submitting response to NRC DSER open issues 82-107.
October 17, 1989	Letter to applicant granting proprietary classification to responses to DSER open issues 82-107.
November 3, 1989	Meeting with Advisory Committee on Reactor Safeguards (ACRS) Subcommittee on Advanced Pressurized Water Reactors to discuss review status and design details of RESAR SP/90.
November 9, 1989	Letter to applicant granting proprietary classification to Amendment 3 to RESAR SP/90 PDA Module 2, "Regulatory Conformance."
November 16, 1989	Meeting with Advisory Committee on Reactor Safeguards (ACRS), full committee to discuss status of standardized pressurized water reactor designs.
January 10, 1990	Meeting with Advisory Committee on Reactor Safeguards (ACRS), Subcommittee on Advanced Pressurized Water Reactors to discuss review status and design details of RESAR SP/90.

March 6, 1990	Meeting with Advisory Committee on Reactor Safeguards (ACRS) Subcommittee on Advanced Pressurized Water Reactors to discuss review status and design details of RESAR SP/90.
March 20, 1990	Letter from applicant submitting Westinghouse RESAR SP/90 position on "Severe Accident Policy Issues" presented in February 9, 1990, ACRS full committee meeting.
September 20, 1990	Meeting with Advisory Committee on Reactor Safeguards (ACRS) Subcommittee on Advanced Pressurized Water Reactors to discuss review status and design details of RESAR SP/90.
November 9, 1990	Meeting with Advisory Committee on Reactor Safeguards (ACRS) full committee to discuss the PDA and request the ACRS letter to the Commission detailing the ACRS comment on RESAR SP/90.
December 12, 1990	Letter from ACRS to Chairman Carr stating the ACRS comments and recommendations on the Preliminary Design Approval for the RESAR SP/90 design.
January 14, 1991	Letter from James M. Taylor, Executive Director for Operations, to ACRS responding to the ACRS letter of December 12, 1990.
February 12, 1991	Letter from ACRS to James M. Taylor, Executive Director for Operations, responding to the letter from James M. Taylor to the ACRS of January 14, 1991.
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APPENDIX B

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APPENDIX C

UNRESOLVED SAFETY ISSUES

NRC's policy statement on severe reactor accidents regarding future designs and existing plants (50 FR 32138) states that the Commission believes that a new design for a nuclear power plant can be shown to be acceptable for severe accident concerns if it meets the following criteria and procedural requirements:

- demonstration of compliance with the procedural requirements and criteria of the current Commission regulations, including the Three Mile Island requirements for new plants as reflected in the construction permit rule [10 CFR 50.34(f); 47 FR 2286]

- demonstration of technical resolution of all applicable unresolved safety issues and the medium- and high-priority generic safety issues, including a special focus on ensuring the reliability of decay heat removal systems and the reliability of both ac and dc electrical supply systems

- completion of a probabilistic risk assessment (PRA) and consideration of the severe accident vulnerabilities the PRA exposes along with the insights that it may add to the assurance of no undue risk to public health and safety

- completion of a staff review of the design with a conclusion of safety acceptability using an approach that stresses deterministic engineering analysis and judgment complemented by PRA

Westinghouse has addressed the above requirements in the RESAR SP/90 PDA application. The first two requirements are discussed in Amendment 3 to Module 2 and the staff's response is contained in this appendix. The third requirement is addressed in Section 19 of the safety evaluation report (SER) and the last requirement is addresed throughout the SER.

Section 3 of Amendment 3 to Module 2 contains the discussion and RESAR SP/90 responses to each item applicable to pressurized-water reactors (PWRs) contained in 10 CFR 50.34(f). These items are the additional TMI-related requirements.

Section 4 of Amendment 3 to Module 2 contains the discussion and RESAR SP/90 responses to the unresolved safety issues (USIs). These USIs are identified through NRC's continuous evaluation of information relating to the safety of nuclear power plants. Sources of this information include operational experience, research results, staff and Advisory Committee on Reactor Safeguards (ACRS) safety reviews, and vendor, architect engineer, and utility design reviews. There are a total of 27 USIs as listed in NUREG-0933, "A Prioritization of Generic Safety Issues," August 1987. Eight of these USIs are not applicable to the RESAR SP/90 design. Technical resolutions to all 27 USIs have been completed by the NRC and issued to the industry for its response and proposal of implementation.

Section 5 of Amendment 3 to Module 2 contains the discussion and RESAR SP/90 responses to the generic safety issues (GSIs). It lists the GSIs in Categories A through C as reported in NUREG-0410, "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants," as well as the more recent GSIs identified in NUREG-0933. NUREG-0933 establishes a priority ranking of high, medium, low, and drop for previously categorized and newly identified GSIs.

Since all the USIs have been resolved technically by the staff and are now in the implementation stage by the vendors and utilities, future safety issues will be identified as GSIs and categorized by the Office of Research at NRC.

Considering the limited amount of design information available at the preliminary design approval (PDA) stage, staff review of the TMI-related requirements, the USIs, and GSIs has been minimal. A detailed review and resolution of these issues will be performed at the final design approval (FDA) stage. At that time, detailed design information will enable the staff to ensure that the RESAR SP/90 design incorporates features that can adequately address all the safety issues included in the TMI requirements, USIs, and GSIs.

APPENDIX D

PRINCIPAL CONTRIBUTORS

NRC Personnel	Review Area
Brammer, H.	Mechanical Engineering
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Gill, A.	Plant Systems
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APPENDIX E

ABBREVIATIONS

ABWR	advanced boiling water reactor
ACI	American Concrete Institute
ACRS	Advisory Committee on Reactor Safeguards
AFWS	auxiliary feedwater system
AHS	air handling system
AISC	American Institute of Steel Construction
AISI	American Iron and Steel Institute
ALARA	as low as is reasonably achievable
ALWR	advanced light-water reactor
ANS	American Nuclear Society
ANSI	American National Standards Institute
APCS	automatic power control system
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	anticipated transient without scram
BCS	boron control system
BMI	bottom-mounted instrumentation
BNL	Brookhaven National Laboratory
BOP	balance of plant
BRS	boron recycle system
BTP	branch technical position
Btu	British thermal unit
CAOC	constant axial offset control
CCFP	conditional containment failure probability
CCS	condensate cleanup system
CCW	component cooling water
CCWS	component cooling water system
CET	containment event tree
CFR	Code of Federal Regulations
CFS	condensate and feedwater system
CHF	critical heat flux
CRDM	control rod drive mechanism
CRDS	control rod drive system
CRT	core reflood tanks
CS	containment spray
CVCS	chemical volume and control system
CWS	circulating water system
DBA	design-basis accident
DC	design criteria
DCH	direct containment heating
DFH	deaerating feedwater heater

DNB	departure from nucleate boiling
DNBR	departure from nuclear boiling ratio
DRDM	displacer rod drive mechanism
DWS	demineralized water system
EAB	exclusion area boundary
ECCS	emergency core cooling system
ECFS	emergency circulation filter system
EFDS	equipment floor and drainage system
EFWS	emergency feedwater system
EFWST	emergency feedwater storage tank
EPRI	Electric Power Research Institute
ESBU	Energy Systems Business Unit
ESF	engineered safety feature(s)
ESFAC	engineered safety features actuation cabinet
ESVS	essential switchgear ventilation system
ESW	essential service water
ESWS	essential service water system
EWST	emergency water storage tank
FDA	final design approval
GDC	general design criteria
GWPS	gaseous waste processing system
HEPA	high-efficiency particulate air
HHSI	high-head safety injection
HVAC	heating, ventilation, and air conditioning
IC	interface criteria
I&C	instrumentation and control
ICSB	Instrumentation and Control Systems Branch
IDCOR	Industry Degraded-Core Rulemaking
IEEE	Institute of Electrical and Electronics Engineers
IFBA	integral fuel burnable absorber
IPC	integrated protection cabinets
IPS	integrated protection system
ISI	inservice inspection
ISS	integrated safeguards system
LBB	leak before break
LBLOCA	large-break loss-of-coolant accident
LLHS	light load handling system
LOCA	loss-of-coolant accident
LOSP	loss of offsite power
LPZ	low-population zone
LTOP	low-temperature overpressure protection
LWPS	liquid waste processing system
MAAP	modular accident analysis program (computer code)
MACCS	MELCOR accident consequence code system (computer code)
MCARS	main condenser air removal system
MCR	main control room
MCRVS	main control room ventilation system

MeV	million electronvolts
MFIV	main feedwater isolation valve
MHI	Mitsubishi Heavy Industries
MPC	maximum permissible concentration
MSIV	main steam isolation valve
MSLB	main steam line break
MSSS	main steam supply system
MST	main steam tunnel
MWD/MTU	megawatt day/per metric ton of uranium
MWt	megawatts thermal
NDRC	National Defense Research Committee
NEMA	National Electric Manufacturers Association
NFSF	new-fuel storage facility
NNS	non-nuclear safety
NPB	nuclear power block
NPSH	net positive suction head
NSSS	nuclear steam supply system
OBE	operating-basis earthquake
OFA	optimized fuel assembly
ORNL	Oak Ridge National Laboratory
PAMS	post-accident monitoring system
P&ID	piping and instrumentation diagram
PC	pressurizer controls
PCP	process control program
PDA	preliminary design approval
PERMS	process and effluent radiological monitoring and sampling
PMP	probable maximum precipitation
POARV	power-operated atmospheric relief valve
PORV	power-operated relief valve
PRA	probabilistic risk assessment
PRT	pressurized relief tank
PSD	power spectral density
PSI	preservice inspection
PWR	pressurized-water reactor
QAC	quality assurance criteria
RAI	request for additional information
RC	rod control (system)
RCCA	rod cluster control assembly
RCDT	reactor coolant drain tank
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RESAR SP/90	Westinghouse Reference Safety Analysis Report
RG	regulatory guide
RHR	residual heat removal
RHRS	residual heat removal system
RPV	reactor pressure vessel
RSS	Reactor Safety Study (NUREG-75-014, 1975)
RT_{NDT}	nil-ductility reference temperature
RVLIS	reactor vessel level instrumentation system

SAR	safety analysis report
S/RV	safety and relief valve
SBLOCA	small-break loss-of-coolant accident
SCA	safeguard component areas
SER	safety evaluation report
SFPCS	spent-fuel pit cooling and cleanup system
SFSF	spent-fuel storage facility
SFWS	startup feedwater system
SG	steam generator
SGIS	steam generator isolation system
SGTR	steam generator tube rupture
SIS	safety injection system
SLC	secondary loop controls
SMACNA	Sheet Metal and Air Conditioning Contractor's National Association
SRP	Standard Review Plan (NUREG-0800)
SRSS	square root of the sum of the squares
SRV	safety relief valve
SSC	structures, systems, and components
SSCV	spherical steel containment vessel
SSE	safe-shutdown earthquake
STCP	source-term code package (computer code)
SW	service water
TBS	turbine bypass system
TGSS	turbine gland sealing system
T-G	turbine generator
TMI	Three Mile Island
UHS	ultimate heat sink
WAPWR	Westinghouse advanced pressurized-water reactor
WDR	water displacement rod
WOG	Westinghouse Owners Group
ZPA	zero period acceleration

APPENDIX F

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS REPORT

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

December 12, 1990

The Honorable Kenneth M. Carr
Chairman
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Chairman Carr:

SUBJECT: WESTINGHOUSE'S APPLICATION FOR PRELIMINARY DESIGN APPROVAL FOR THE RESAR SP/90 DESIGN

During the 367th meeting of the Advisory Committee on Reactor Safeguards, November 8-10, 1990, we completed our review of Westinghouse's application for Preliminary Design Approval (PDA) for the Westinghouse Reference Safety Analysis Report (RESAR SP/90) nuclear power block (NPB). We heard presentations from the NRC staff and the applicant concerning the staff's draft Safety Evaluation Report (SER) (NUREG-1413) for this PDA during our meeting. Representatives of the staff and of the Office of the General Counsel (OGC) discussed the related draft PDA document. Our Subcommittee on the Advanced Pressurized Water Reactors has held a series of meetings with the staff and representatives of the applicant regarding this matter over the past two and a half years. We also had the benefit of the documents referenced.

1.0 <u>Scope and History of RESAR SP/90 Application</u>

The RESAR SP/90 is an evolutionary (as contrasted with passive) Advanced Light-Water Reactor (ALWR) design for a single-unit NPB, rated at a reactor power of 3800 MWt. Although many basic design decisions were made by Westinghouse prior to completion of the EPRI ALWR Utility Requirements Document, the design of this four-loop pressurized water reactor generally conforms to the EPRI requirements for such designs.

RESAR SP/90 NPB contains preliminary design information for the portion of the design that encompasses NPB buildings, structures, systems, and components. Specifically excluded from the scope are the turbine building, the waste disposal building, the service building, the administration building, the service water/cooling water structure, and the ultimate heat sink. These features will be the design responsibility of an applicant proposing to build a facility referencing the RESAR SP/90 design. Interface information addressing the pertinent safety-related design requirements necessary to ensure the compatibility of the referenced system with the plant-specific portion of the facility has been included in the RESAR SP/90 application.

On October 24, 1983, Westinghouse submitted an application for a PDA for RESAR SP/90 NPB design in accordance with 10 CFR Part 50, Appendix O, "Standardization of Design: Staff Review of Standard Designs," which was the then existing regulatory basis for this type of application. The application was docketed on November 30, 1983 (Docket No. 50-601). The RESAR SP/90 application describing the design of the NPB was submitted in modular form during the period from October 23, 1983 to March 9, 1987. In addition, the information in RESAR SP/90 has been supplemented by 47 amendments to these modules.

2.0 Regulatory Background

Before the promulgation of 10 CFR Part 52 in May of 1989, the review of RESAR SP/90 had been performed by the staff pursuant to Appendix O to 10 CFR Part 50, using a procedure similar to that used for custom plant reviews for which guidance to staff reviewers is provided in the Standard Review Plan. This evaluation was analogous to a construction permit (CP) licensing review for a specific facility and conducted with the intent that, following satisfactory completion of the reviews performed by the staff and the ACRS, a PDA could be issued by the staff. The promulgation of 10 CFR Part 52 resulted in the transfer of Appendix O to 10 CFR Part 52; hence a PDA can now be issued for this application pursuant to 10 CFR Part 52. A PDA is optional for a Final Design Approval (FDA) and/or Design Certification under the provisions of 10 CFR Part 52.

3.0 The Staff's SER and the PDA

The SER and PDA represent the first stage of the staff's review of the design, construction, and operation of the RESAR SP/90 design. During our meetings, we learned that there is no prospective CP applicant nor does Westinghouse intend to apply for an FDA and/or Design Certification of the RESAR SP/90 design until there is a proven interest on the part of a domestic or foreign utility. The staff's SER summarizes the results of the staff's radiological safety review of the RESAR SP/90 NPB design and delineates the scope of the technical details considered in evaluating the proposed design. This review took place over the period of October 1983 to October 1989 (the date on which the staff decided to close its review). Environmental aspects were not considered in the staff review of RESAR SP/90, but would be addressed in a utility's plant-specific application.

3.1 <u>Comments on the Staff's SER</u>

There are 170 open items that will require resolution during the review of a plant-specific application for an Operating License (OL). Most of these appear to be the kind of open issues expected at this stage of the design. Of the 170 open items, 17 are site specific, 110 involve information in the scope of an OL or FDA and/or Design Certification application, and 43 had not been resolved by the staff when it closed its review in October 1989. (Westinghouse submittals on many of these 43 open items, including its proposed resolution of Generic Safety Issues, Unresolved Safety Issues, post-TMI regulatory requirements, and outstanding PRA issues are yet to be reviewed by the staff.) In view of these open items and our concerns regarding the SER and the many unresolved severe accident issues, we indicated to the staff that its conclusions on page 25-1 of the draft SER were stated too strongly. The staff agreed to revise this language.

The Committee is not of one mind regarding the issuance of a PDA for the RESAR SP/90. On the one hand, there is merit to the argument that Westinghouse's application for the RESAR SP/90 PDA was made in good faith under a different set of regulations and that it is now appropriate to document the reviews that have taken place to date and issue the PDA for potential future use as a reference design for an individual plant CP application or as the starting point for an FDA and/or Design Certification application. Both Westinghouse and the staff advocate this approach; neither believes that it can devote further resources to this effort.

On the other hand, we view the RESAR SP/90 SER as a mixed bag of staff evaluations that were performed over the seven-year period since the application was filed. Some are current and well done; others are poorly done and/or were performed years ago and do not meet the standards that we believe should be applied to a current SER. A major contributor to this problem appears to be the staff's reliance on the July 1981 Standard Review Plan (SRP) (NUREG-0800) in performing this review. This SRP needs updating to reflect the current situation for the licensing of ALWRs.

Some examples of our concerns with the staff's SER are:

3.1.1 SER Chapter 7, Instrumentation and Controls, references a staff review that was performed in 1979 for the Westinghouse RESAR 414 design. The staff concluded that the computer based integrated reactor protection system design for RESAR SP/90 is acceptable for a PDA on the basis of the "similarity" of the RESAR 414 design to that proposed for RESAR SP/90. It is our view that the staff should have developed improved standards for the review of such systems during this 11-year period. We are particularly concerned about the verification and validation of the software employed with computer based reactor protection systems. It appears that there is a need to augment existing staff resources with expertise in the computer science area so that appropriate standards can be developed for the review of computer based reactor protection systems. All of the proposed evolutionary and passive ALWRs employ such systems.

3.1.2 For materials used in the fabrication of pressure boundary components, Westinghouse has committed to follow applicable codes, standards, and regulatory guides. Many of these are not representative of current industry practice for such materials. We learned that Westinghouse has developed internal specifications for pressure boundary materials that presumably do reflect current industry practice. These were not submitted for the staff's review.

3.1.3 The proposed design employs water displacer control rods and associated control rod drive mechanisms, which is a new feature for Westinghouse plants. The SER describes the function of and strategy for use of these control rods. The SER, however, does not discuss the pressure boundary integrity of these new control rod drive mechanisms or the potential for reactivity insertion accidents that could result from misoperation of these control rods. Although Westinghouse submitted information on these subjects, the staff has not completed its review of this information. In general, we believe that new features of this kind should be thoroughly reviewed at an early stage of review.

3.1.4 Our review, which represents only a sampling effort, revealed a number of factual errors and inconsistencies in the SER; the staff has agreed to correct these errors. We believe that a review of the draft SER by Westinghouse, which has not yet had access to this predecisional document, would reveal additional errors that should be corrected. We recommend that this be done.

3.2 <u>Comments on the PDA Document</u>

The PDA states that the preliminary design information contained in RESAR SP/90 "complies with the requirements of 10 CFR Part 52, Appendix O . . . and is acceptable for incorporation by reference in applications for individual construction permits . . . ." The PDA does not describe how this preliminary design information would be used in a future FDA and/or Design Certification application.

We were told by OGC that this results from the fact that Westinghouse has not made an application under 10 CFR Part 52.

Given the quality of the SER for this PDA, we are concerned with the language of the PDA that requires the staff and ACRS to utilize and rely on the "approved preliminary design" in their reviews of any individual facility construction permit application " . . . unless significant information which substantially affects the determination set forth in this PDA, or other good cause, is present." OGC advised us that this requirement would apply only to the staff and ACRS reviews of a CP application and that both entities would be able to revisit any issue in their review of any type of application that would lead to an OL. This is satisfactory to us but could present problems for the staff in dealing with a contested CP application.

4.0 Comments on the SP/90 Design

We have two concerns regarding SP/90 design features:

4.1 Our review of the NPB layout indicates that Westinghouse has provided many desirable features from the standpoint of separation of equipment trains for protection against fires and industrial sabotage. However, we are concerned about the location of the emergency diesel generators (EDGs) on the same floor and corridor from the control room. We believe that another location for the EDG room should be specified in view of the potential for fire and/or explosions associated with the operation of large diesel generators.

4.2 The proposed RESAR SP/90 design employs a spherical containment. To deal with core/concrete interaction, the layout of the containment employs a cavity floor area beneath the reactor vessel that is based on the EPRI requirement of 0.02 m^2 per MWt. If a larger area is required, major changes to the containment sizing and layout may be needed. Timely development of a Commission position on this issue is important not only to this design but also to the design of all of the ALWRs.

5.0 ACRS Recommendations on the Issuance of a PDA

We believe, subject to the above comments, that the proposed design of the RESAR SP/90 NPB can be successfully completed and used in an application for an individual plant CP. Accordingly, we recommend that a PDA be issued for the proposed Westinghouse RESAR SP/90 NPB.

6.0 Concluding Remarks

Finally, we wish to commend the Westinghouse Electric Corporation, the Japanese APWR program participants, the EPRI ALWR Utility Steering Committee, and the EPRI staff for the effort they have expended in the development of this evolutionary design. The RESAR SP/90 design represents an important step forward in providing improved LWR designs that incorporate many of the lessons related to safety, performance, and reliability that have been learned by the nuclear power industry over the past 30 years.

Sincerely,

Carlyle Michelson

Carlyle Michelson
Chairman

References:

1. U.S. Nuclear Regulatory Commission, Draft NUREG-1413, "Safety Evaluation Report Related to the Preliminary Design of the Standard Nuclear Steam Supply Reference System, RESAR SP/90" (Predecisional)
2. Draft Westinghouse Electric Corporation, Docket No. 50-601, Reference Safety Analysis Report (RESAR SP/90 Nuclear Power Block Standard Design), Preliminary Design Approval (PDA) (Predecisional) (Discussed during the November 8-10, 1990 ACRS full Committee meeting)
3. Letter NS-EPR-2675 dated November 1, 1982 from E. P. Rahe, Jr., Westinghouse Electric Corporation, to F. Miraglia, U.S. Nuclear Regulatory Commission, Subject: Westinghouse Advanced Pressurized Water Reactor Licensing Control Document

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

February 12, 1991

Mr. James M. Taylor
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Taylor:

SUBJECT: EDO RESPONSE TO ACRS REPORT DATED DECEMBER 12, 1990 ON THE PRELIMINARY DESIGN APPROVAL FOR THE RESAR SP/90 DESIGN

In our December 12, 1990 report to Chairman Carr regarding Westinghouse's Application for Preliminary Design Approval for the RESAR SP/90 Design, we expressed a concern (item 4.1) about the location of the emergency diesel generator (EDG) on the same floor and corridor as the control room.

In our report we stated that, "We believe that another location for the EDG room should be specified in view of the potential for fire and/or explosions associated with the operation of large diesel generators."

Item 8 of the enclosure to your response of January 14, 1991 states that, "The staff has not in the past considered, and does not now consider, credible an explosion in the EDG room of sufficient size to cause catastrophic failure of the reinforced concrete enclosure of these rooms."

Your response did not address the large door that separates the EDG from the corridor leading to the control room. We ask that you expand your reply to include consideration of this door and give us your views on the size of a fire and/or explosion that you would consider credible, and some estimate of the structural capability of this door under differential pressure conditions. Also, we ask that you address the potential for a fire resulting from combustibles such as fuel oil that may flow under the door into the corridor.

Sincerely,

David A. Ward

David A. Ward
Chairman

NRC FORM 335 (2-89) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

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10. SUPPLEMENTARY NOTES
Docket No. 50-601

11. ABSTRACT (200 words or less)

This report provides the results of the NRC staff review of the Westinghouse Electric Corporation for a preliminary design approval of the SP/90 reactor contained in its reference safety analysis report. The standard safety analysis report describing the design of the facility was submitted from October 24, 1983 through March 9, 1987. Staff of the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, prepared this safety evaluation report of the RESAR SP/90. Based on its review, the staff concludes that there are open issues that, because of the stage of the design, have not been resolved at this stage of review. These issues are discussed in detail throughout this report, and a summary is provided in Section 1.6 of this report.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)
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