1.0 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 INTRODUCTION

The Westinghouse Electric Corporation (hereinafter referred to as Westinghouse) has developed this Reference Safety Analysis Report (RESAR-SP/90) for the Westinghouse Advanced Pressurized Water Reactor (WAPWR) as part of its continuing efforts toward design and licensing standardization of nuclear power plants. RESAR-SP/90 is a standard safety analysis report submitted initially for Preliminary Design Approval (PDA) in accordance with Appendix O, "Standardization of Design; Staff Review of Standard Designs," to Part 50 of Title 10 of the Code of Federal Regulations (hereinafter referred to as 10CFR). The ultimate objective is to obtain a Final Design Approval (FDA) of RESAR-SP/90 followed by a rulemaking proceeding and design certification.

1.1.1 Plant Description

1.1.1.1 Scope

The WAPWR is a single unit Nuclear Power Block (NPB) design for a 3816 MWt, four loop pressurized water reactor. The scope of the WAPWR NPB design includes all buildings, structures, systems and components that are essential to the safe and proper operation of the nuclear power plant. Specially 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.

The key areas that are included in the NPB scope are; (1) the containment building, (2) the fuel handling facilities, (3) the mechanical safeguards equipment area, (4) the auxiliary systems area, (5) the instrumentation and controls area, (6) the control room, (7) the electrical power distribution equipment area, (8) the emergency diesel generator area, and (9) the technical support center.

Table 1.1-1 provides a detailed listing of the building/structure, systems and components that are included in the NPB scope, or interface with the NPB. For those systems which are not in the NPB scope, this table defines the type of interface information that will be provided in RESAR-SP/90. Both the design criteria and interface criteria that will be provided for the non-NPB scope items are considered pertinent safety related requirements.

1.1.1.2 Power Levels

The WAPWR NPB design includes a nuclear steam supply system with a thermal rating of 3816 megawatts, which includes a core thermal power of 3800 megawatts plus 16 megawatts from reactor coolant pump heat. The core thermal power level of 3800 megawatts is the licensed power level for the WAPWR NPB design. However, major components of the WAPWR design have been sized for a stretch core thermal rating of 4200 megawatts.

For a 3816 MWt NSSS output the steam and power conversion system would have a corresponding maximum output of approximately 1350 MWe with a steam pressure of 1024 psia.

The following power levels are assumed in the accident analyses for the WAPWR design:

- A core thermal power level of 3800 megawatts is assumed in the accident analyses for events which are departure from nucleate boiling (DNB) limited and initiated at full power.

- A core thermal power level of 3876 megawatts, which is 1.02 times the core thermal power level, is assumed in the accident analyses for events which are not DNB limited and initiated at full power.

- Analyses for events which result in demonstrating acceptability of the site radiological consequences are analyzed at a core thermal power level of 3876 megawatts.

These assumed power levels are in accordance with the recommendations of Regulatory Guide 1.49, Revision 1, "Power Levels of Nuclear Power Plants," for the licensed core power level of 3800 megawatts.

1.1.1.3 Interface Information

Interface information is provided to address the pertinent safety-related requirements that must be met by the unspecified portions of a nuclear facility design to ensure that structures, systems, and components within the standard design will perform their intended safety functions.

Westinghouse has specified the following types of interface information.

- Design Criteria

  Interface information of a detailed nature such that Westinghouse is essentially specifying the actual design for a specific structure, system or component outside the scope of the NPB.

- Interface Criteria

  Interface information of a general nature such that Westinghouse is specifying the functional requirements for a specific structure, system, or component outside the NPB scope while the actual design is unspecified.

Table 1.1-1 defines the major plant systems and physical areas within the WAPWR NPB scope, and identifies the type of interface information provided in RESAR-SP/90 for those plant areas outside this scope. See Section 1.9 for a further discussion of interface information.

1.1.2 Format and Content

1.1.2.1 Regulatory Guide 1.70

RESAR-SP/90 is written to be consistent with the format and content recommendations of Regulatory Guide 1.70, Revision 3, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants - LWR Edition."

In general RESAR-SP/90 is structured to provide information on a system-by-system modular basis during the development of the PDA document. The information technique provides much of the information associated with a particular system or major component in one package. Note that selected subject areas such as Instrumentation and Controls, and Structural and Equipment design are submitted in separate modules in order to facilitate the review process. See Table 1.1-2 for the various topics of the modules.

Upon completion of the modular submittals, the modules will be consolidated into a PDA-level document.

1.1.2.2 Standard Review Plan

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition," was specifically considered in the preparation of RESAR-SP/90.

In accordance with the regulations of 10CFR50.34(g) and the regulatory guidance of draft NUREG-0906, "Guidance for Implementation of 10CFR50.34(g)," RESAR-SP/90 includes an identification and description of all differences in design features, analytical techniques, and procedural measures for the WAPWR and those corresponding features, techniques, and measures given in the Standard Review Plan acceptance criteria. Where such differences exist, an evaluation is provided that discusses how the alternatives provide an acceptable method of complying with the NRC regulations that underly the corresponding Standard Review Plan acceptance criteria.

Evaluations of the level of compliance with the Standard Review Plan acceptance criteria and justification for any deviations are presented in appropriate locations throughout RESAR-SP/90. Section 1.8 of each module includes a tabulated summary of all deviations from the Standard Review Plan acceptance criteria which is cross-referenced to the detailed evaluations in the appropriate RESAR section.

1.1.2.3 Regulations and Regulatory Guidance

10CFR Parts 0 through 199 include those NRC regulations that must be met in the design, construction, and operation of all nuclear power plants. The "Regulatory Conformance" module discusses the regulations of 10CFR with respect to the WAPWR design.

Since many of the NRC regulations of 10CFR are written in a general nature, the NRC has provided regulatory guidance for implementing specific regulations. This regulatory guidance is in the form of regulatory guide positions, Standard Review Plan acceptance criteria, branch technical positions, NUREG reports, etc. The applicability of this guidance to the WAPWR design is discussed in RESAR-SP/90 PDA Module 2, "Regulatory Conformance".

In addition to the discussion of regulations, the "Regulatory Conformance" module includes a discussion of proposed regulations, unresolved safety issues, generic safety issues, and other licensing issues in relation to the WAPWR NPB design; consistent with the requirements of the NRC's severe accident policy statement.

1.1.3 RESAR-SP/90 Usage, Implementation, and Backfitting

1.1.3.1 Usage

The NPB design described in this document is intended to be incorporated by a utility application for a combined construction permit and operating license. The design may be adopted or modified for a specific application without prior NRC approval provided the modification is appropriately evaluated and found

not to involve a change in the technical specifications or an unreviewed safety question consistent with the regulations of 10CFR 50.59. Design modifications that involve a change in the technical specifications or an unreviewed safety question could also be implemented in the WAPWR NPB design. However, unless it is determined that the modifications are appropriate for backfitting to the base WAPWR design (refer to Section 1.1.3.3), these types of design modifications are discouraged since:

- They can only be implemented after NRC review and approval.

- Optimum utilization of RESAR-SP/90 for one-step licensing could be jeopardized depending on the extent of the modifications.

1.1.3.2 Implementation

The WAPWR NPB (including the specified interface criteria and design criteria for those systems, structures and components outside the NPB scope) presented in RESAR-SP/90 documents the plant design from the standpoint of structures, systems, and components important to safety. The major remaining aspects and areas of an application required to be reviewed by the NRC can be characterized as "site-related" and "facility-related".

- Site-Related - refers to the actual plant specific site characteristics such as population, meteorology, seismology, hydrology, etc.

- Facility-Related - refers to the specific applicant's organizational structure, quality assurance programs, personnel qualifications and training programs, emergency plans, security plans, etc.

Optimal one-step licensing utilization of RESAR-SP/90 (once certified through rulemaking) would be as follows:

- A utility would tender an application for a Class 103 license (combined construction permit and operating license) in accordance with the procedural regulations of 10CFR 50.30.

This application would include:

a. The general information required by 10CFR 50.33.

b. The antitrust information required by 10CFR 50.33a.

c. Site-related and facility-related information required by 10CFR 50.34. (Certain site-related aspects could be incorporated by reference if the site was previously approved in accordance with the provisions of Appendix Q to 10CFR Part 50. Facility-related aspects would be best specified by NRC rule such that the utility could provide appropriate information to demonstrate conformance).

d. Reference to the NPB design documented in RESAR-SP/90 for the remaining information required by 10CFR 50.34.

- The NRC staff, ACRS, ASLB, and appropriate state and local authorities would review any applicable portions of the applications not related to the certified NPB design documented in RESAR-SP/90.

- Upon a favorable decision by the Commission, a combined construction permit and operating license would be issued to the applicant.

In addition to the appropriate permit/license conditions in accordance with the regulations of 10CFR 50.54 and 10CFR 50.55, the combined construction permit and operating license would specifically include the following conditions:

- The utility must certify the implementation of the extensive checks, tests, and inspections described in RESAR-SP/90.

- The utility must certify that Westinghouse has defined a program to verify that all interface criteria and design criteria specified in RESAR-SP/90 have been met by the interfacing structures, systems and components design.

1.1.3.3 Backfitting

The regulations of 10CFR 50.109 provide for backfitting (i.e., the addition, elimination, or modifications of structures, systems, or components) if the Commission finds that such action will provide substantial, additional protection which is required for public health and safety or the common defense and security.

Two concepts have been incorporated in RESAR-SP/90 to minimize the need for future backfitting of the design. First, all currently identified safety issues as documented and summarized in RESAR-SP/90 PDA Module 2, "Regulatory Conformance" have been evaluated by Westinghouse and addressed in the WAPWR design. Second, the integrated design and siting probabilistic risk assessment for the WAPWR (presented in RESAR-SP/90 PDA Module 16, "PRA/Severe Accidents") demonstrates that the design has a sufficiently low risk. The concept of addressing potential future regulatory requirements as an integral part of the WAPWR design process combined with the demonstrated low risk design renders the WAPWR NPB design immune to reevaluation in light of regulatory changes during the effective certification period.

Although believed to be unlikely, backfitting of the WAPWR design documented in RESAR-SP/90 could occur only as a result of those rare instances where it is judged by the Commission, Westinghouse, or the utility utilizing the WAPWR design that an emergency action is needed to protect the health and safety of the public. These instances could result through identification of a substantial safety hazard (as defined by 10CFR 21.3(k)), a significant deficiency (as defined by 10CFR 50.55(e)), or an unreviewed safety question (as defined in 10CFR 50.59(a)).

1.1.4 Amending RESAR-SP/90

When it becomes necessary to submit additional information or revise information presently contained in RESAR-SP/90 the following procedures will be followed:

- Whenever there is a change in the information contained on any of the RESAR-SP/90 text, those pages affected will be marked with the amendment number and date at the bottom of the page and a vertical line extending the length of the change labeled with the amendment number in the margin opposite the binding margin.

- Figures will be revised by indicating the amendment number and date in the lower right hand corner. Notation will be made of any major revisions made in that amendment.

- To make incorporation easier, each amendment will include a detailed instruction sheet to permit the reader to correctly delete, replace, or add material.

- The Table of Contents will be revised as necessary for future amendments.

TABLE 1.1-1

WAPWR NUCLEAR POWER BLOCK DEFINITION

	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

(1) NPB = In WAPWR Nuclear Power Block Scope
(2) DC = Design criteria (see Section 1.1.1.3, "Interface Information" for definition)
(3) IC = Interface criteria (see Section 1.1.1.3, "Interface Information", for definition)

TABLE 1.1-1 (Continued)

WAPWR NUCLEAR POWER BLOCK DEFINITION

	NPB	DC	IC
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		
- 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 CONTROL			
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 (TSC)	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 auxiliary systems I&C			
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		

TABLE 1.1-1 (Continued)

WAPWR NUCLEAR POWER BLOCK DEFINITION

	NPB	DC	IC
- Essential mechanical equipment	x		
- Diesels and auxiliaries	x		
- Essential electrical equipment	x		
- Control room	x		
- Fuel handling and storage areas	x		
Waste disposal building			x
Necessary equipment supports and restraints	x		
Turbine building			x
Service water structure and dams			x

TABLE 1.1-2

PDA

MODULE TOPICS

Module	Topic
1	Primary Side Safeguards System
2	Regulatory Conformance
3	Introduction and Site
4	Reactor Coolant System
5	Reactor System
6	Secondary Side Safeguards System
7	Structural/Equipment Design
8	Steam and Power Conversion
9	I&C and Electric Power
10	Containment Systems
11	Radiation Protection
12	Waste Management
13	Auxiliary Systems
14	Initial Test Program
15	ACR/Human Factors
16	PRA/Severe Accidents

1.2 GENERAL PLANT DESCRIPTION

1.2.1 Site

The WAPWR is a standardized design employing a site envelope concept. Bounding site-related parameters (i.e., interface criteria between the nuclear power block and site) which relate to seismology, hydrology, meteorology, geology, heat sink parameters, and other site-related aspects impacting the plant design are provided in Section 1.9 and Chapter 2 of this module as appropriate. The magnitudes of the parameters are selected to suit a wide range of potential sites.

For those sites having values which extend beyond the range of the selected parameter, such site specific values would have to be demonstrated to be acceptable by the specific applicant(s) referencing the WAPWR NPB design documented in RESAR-SP/90. Additional site-specific information needed to complete the application for a plant specific Construction Permit or Operating License application will be provided by the applicant.

1.2.2 Principal Design Criteria

RESAR-SP/90 is designed to comply with 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants." The specific applications of General Design Criteria to RESAR-SP/90 are discussed in section 3.1 of the "Structural/Equipment Design" module.

1.2.3 Plant Description

The plot plan for the WAPWR nuclear power plant facility is shown on Figure 1.2-1. This plot plan shows the following major building and structures: 1) the containment building, 2) the reactor external building, 3) the turbine building, 4) the waste disposal building, 5) the service building, and 6) the administration building. Not shown on this plot plan are the service water/cooling water building/structures and the ultimate heat sink. The containment building and the reactor external building essentially contain all the

buildings, structures, systems and components that are included in the WAPWR NPB scope. All the other major buildings listed above and shown on Figure 1.2-1 are excluded from the NPB scope. There are a few exceptions, however, where the NPB scope waste processing system would be located in the waste disposal building and the NPB scope Technical Support Center (TSC) would be located in the service building.

The general arrangement drawings for the NPB scope containment building and reactor external building are provided in Figure 1.2-2, sheets 1 through 9. As shown on these drawings, the containment and the reactor external building
(a,c) are constructed on a [ ] meter integral (common) basemat. The [ ] meter
(a,c) diameter spherical steel containment vessel (SSCV), a [ ] meter diameter reinforced concrete shield building and a wrap-around reactor external building (RE/B) represents the major structure located on this common
(a,c) basemat. The two key reference elevations for the nuclear island are the [ ]
(a,c) meter operating floor elevation and the [ ] meter grade elevation. The [ ] meter grade elevation corresponds to the containment floor elevation.

(a,c) The [ ] meter diameter SSCV contains the WAPWR 3816 MWt four loop reactor coolant system (RCS). The major components of the RCS are the reactor vessel, four reactor coolant pumps, four steam generators, the pressurizer, and the pressurizer relief tank (PRT). Several major components of the engineered safety systems are also located in the SSCV, such as four containment recirculation units, four accumulators, four core reflood tanks, four residual
(a,c) heat removal (RHR) heat exchangers, and a [ ] gallons emergency water storage tank (EWST).

The capacity of the EWST is dictated by the water volume required to fill the relatively large refueling canal associated with the WAPWR RCS. Section views
(a,c) A-A and B-B on sheets 8 and 9 of Figure 1.2-2 and the plan elevation [ ] meters on sheet 3 depict the conical shape of the EWST. The EWST is a stainless steel lined tank that is located below the nominal containment floor
(a,c) level of elevation [ ] meters. A 2500 ppm boron concentration is maintained in this tank during normal plant operation. In the event of an accident, the four integrated safeguards system (ISS) low head and four ISS high head pumps

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 are flooded, would water return to the EWST via the [ ]inch diameter spillways shown on plan elevation [ ]meters (sheet 4 (a,c) of Figure 1.2-2).

The reactor external building (RE/B) essentially contains all the NPB scope systems and components not located inside the SSCV. The RE/B is located on the [ ]meters common basemat and it extends 360° around the secondary (a,c) containment (shield building). The equipment located in the RE/B has been arranged to: 1) separate the non-safety equipment from the safety related equipment; 2) separate the Train A components from the Train B components; and 3) separate the radioactive (dirty) components from the non-radioactive (clean) components.

The RE/B general arrangement drawings (Figure 1.2-2, sheets 1 thru 9), show the safety related equipment generally located between building column line (A) and (H). The non-safety related equipment is generally located from column line (H) to column line (Q). For RE/B electrical train separation, Train A equipment has generally been located to the right of the RE/B centerline and Train B equipment is located to the left of the RE/B centerline. The majority of non-safety related component areas are located in radioactive control areas and the majority of safety-related component areas are located in non-radioactive control areas. The only safety-related component areas that are classified as dirty areas are the four ISS safeguard component areas (SCA) located in the shadow area beneath the sphere, between elevation [ ]meters and elevation [ ]meters. (a,c)

It should be noted that the RE/B boundary does include the building volume commonly referred to as the shadow area beneath the sphere. This building volume below elevation [ ]meters and between the primary containment (SSCV) (a,c) and the secondary containment (shield building) is subdivided into seven dedicated and totally separated zones. One of these seven zones is dedicated to the non-safety related chemical and volume control system (CVCS) pumps, valves, and piping. Two of the zones are dedicated to the two emergency

feedwater system (EFWS) subsystems, while the remaining four zones serve as the four ISS safeguard component areas (SCA). Sheets 1, 2 and 4 of Figure 1.2-2 depict the complete separation of these seven zones at elevation[
(a,c)]meters.

Several key areas of the RE/B area: 1) the main control room (MCR) located at
(a,c) elevation[ ]meters in the southeast corner of the RE/B, 2) the Train A and B
(a,c) diesel generator rooms located at elevation[ ]meters and in separate wings
of the RE/B, 3) the Train A and B Class IE switchgear rooms located at
(a,c) elevation[ ]meters and in separate wings of the RE/B; 4) the fuel handling
(a,c) area located in the north wing of the RE/B and extending from elevation[84.9]
(a,c) meters (grade) to elevation[ ]meters; 5) the main steam tunnel located in
(a,c) the south wing of RE/B and extending from elevation[ ]meters to elevation
(a,c) [ ]meters; 6) the electrical penetration areas located on elevation[ ]
meters and in the southeast and southwest quadrants of the RE/B; 7) the emergency feedwater storage tanks located in the south wing of the RE/B and
(a,c) extending from elevation[ ]meters to elevation[ ]meters; 8) the CCW heat
(a,c) exchangers located in the south wing of the RE/B at elevation[ ]meters; 9)
(a,c) the CCW pumps located directly below the CCW heat exchangers at elevation[ ]
(a,c) meters, and 10) the majority of the HVAC equipment located at elevation[ ]
meters.

It should be noted that the space between the primary containment building (SSCV) and the secondary containment building (shield building), above
(a,c) elevation [ ] meters is not considered part of the RE/B. This space is
(a,c) designated the annulus area. It should also be noted that the [ ] meter
elevation coincides with the top of the concrete cradle which supports the spherical containment. Therefore, the building volume below the concrete cradle is considered part of the RE/B while the building volume above the concrete cradle is considered the annulus area.

1.2.3.1 Reactor System

The WAPWR reactor system consists of the equipment and components constituting the operating nuclear reactor. It includes the reactor vessel, integrated

head, reactor internals, control rod drive mechanisms, displacer rod drive mechanism and the reactor core; including fuel assemblies, water displacer rod assemblies, gray rod assemblies, and rod cluster control assemblies.

The primary features of the WAPWR reactor design are:

1) LOW POWER DENSITY - Low power density refers to the significantly reduced power density in this core design compared to other contemporary PWR core designs. The WAPWR core is increased in diameter, contains more fuel rods (19x19 fuel array), and has more weight of fuel. The additional fuel loading results 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 reduces peak clad temperature in a large break Loss of Coolant Accident (LOCA) significantly. The lower average rod heat flux provides additional DNB margin.

   For a given burnup, the increase in fuel loading reduces the fraction of the total core loading which must be replaced at the end of a fuel cycle. The result for the same energy extraction is a reduction in the required feed enrichment. The low power density results in a lower cycle burnup (MWD/MTU) because of the additional fuel loading, which increases the number of zones or reduces the fraction of the core replaced. This results in a lower core average burnup at end of cycle which reduces the required feed region enrichment.

2) MODERATOR CONTROL SYSTEM - The moderator control concept controls excess reactivity by varying the amount of moderator in the core instead of using control poisons for neutron absorption. This control of reactivity is 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 occurs and neutrons 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 are 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 is that the amount of fissile uranium and plutonium remaining at end of life is about the same as in a poison-controlled core; however, the initial core feed enrichment is much lower, which results in an additional savings in ore and enrichment (separative work) requirements.

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

3) RADIAL NEUTRON REFLECTOR - The radial neutron reflector consists of a close-packed array of stainless steel rods assemblied in cans and located on the core periphery. It replaces the current baffle-former structure located between the barrel and the fuel. Its benefit is a reduction in net neutron leakage which increases core reactivity and reduces feed enrichment requirements. The result is a substantial savings in ore use with a potential for increased benefit with a low leakage fuel management scheme. The reflector design also helps to reduce reactor vessel fluence levels.

The advanced reactor core utilizes U-238 enriched with approximately [
(a,c)] weight percent of U-235. Fuel rods are comprised of stacked ceramic
(a,c) UO_2 pellets clad in Zircaloy tubing, with an active fuel length of [ ]
inches. The fuel rods are arranged in a 19x19 array to make up the fuel assembly as shown in Figure 1.2-3.

The WAPWR reactor vessel is a large cylindrical pressure vessel with a welded hemispherical bottom head and removable flanged and gasket upper head (see Figure 1.2-4) which functions to contain and support the operating reactor core, and to provide for insertion and removal of the components and instrumentation used to control reactor power level and monitor reactor core operation. Specifically, it houses the core, core support structures, rod cluster

control assemblies, displacer rod assemblies and other components directly associated with the core. The rod cluster control assemblies, displacer rod assemblies and gray rod assemblies are operated by sealed drive rod mechanisms mounted on the vessel head. The vessel head has 185 penetrations arranged in a square pattern to accommodate the two types of drive rod mechanisms. Sixty-one nozzles penetrating the bottom head provide for connection of the bottom mounted in-core instrumentation conduits.

The integrated head package (IHP) is a system that combines the head lifting rig, mechanism seismic supports, lift rods, reactor vessel missile shield, CRDM cooling system, and the power and instrumentation cabling into the efficient package. Mounted directly on the reactor vessel head, the system minimizes the time, manpower, and radiation dosage associated with head removal and replacement during a refueling (see Figure 1.2-5).

The WAPWR reactor utilizes a control element (either a rod control cluster, gray rod cluster, or water displacer rod cluster) over 185 of the 193 fuel assemblies. Therefore, a rod drive mechanism is required to move each of those 185 control elements. The control rod clusters and gray rod clusters which are used for control of reactor power and for shutdowns are positioned using the conventional, magnetic jack type drive mechanism. They provide stepwise movement of the control rods. The 88 water displacer rod clusters are 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 is composed of a pressure housing, a hydraulic cylinder, the mechanical latching device and a vent system (see Figure 1.2-6).

The reactor internals for the WAPWR perform functions similar to those in conventional pressurized water reactors: core support, flow direction, and guidance and protection of control rods. The WAPWR reactor internals however have additional functions since gray rods and water displacer rods are employed in addition to control rods. These are described in detail in RESAR-SP/90 PDA Module 5, "Reactor System".

The similarity of the WAPWR and the 4XL Models is illustrated in Figure 1.2-7, 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 is provided 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 is much thicker to accept axial loading permitting the elimination of support columns.

1.2.3.2 Reactor Coolant System

The WAPWR Reactor Coolant System (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 (a,c)[ ]ft^3 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 displacer rod drive mechanism (DRDM) vent system, used in connection with operation of the DRDM's. The DRDM's are fully described in Section 3.9 of RESAR-SP/90 PDA Module 5, "Reactor System". All system equipment is 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 1.2-8.

During operation, the reactor coolant pumps circulate pressurized water through the reactor vessel and the coolant loops. The water, which serves as the reactor coolant, the moderator and the solvent for boric acid (chemical shim control) is heated as it passes through the core. It then flows to the steam generators, where the heat is transferred to the steam system, and returns to the reactor coolant pumps to repeat the cycle. After the reactor has been shut down, the reactor coolant is circulated by the residual heat

removal subsystem equipment of the primary side safeguards system to remove the heat generated in the fuel from fission product decay. The RCS pressure boundary also serves as a second barrier, after the fuel cladding, against fission product release to the environment.

RCS pressure is controlled by operation of the pressurizer, where water and steam are maintained in equilibrium (saturated conditions) by electrical heaters and a water spray. Steam can be formed by the heaters and condensed by the pressurizer spray to control pressure variations due to contraction and expansion of the reactor coolant. Spring-loaded safety valves and power-operated relief valves are connected to the pressurizer and discharge to the pressurizer relief tank. In the event of safety valve or PORV operation, the steam discharged is condensed and cooled in the Pressurizer Relief Tank (PRT) by mixing with the water normally present in the tank.

The RCS is serviced by a number of auxiliary systems, including the chemical and volume control system (CVCS), the integrated safeguards system (ISS), the main steam and feedwater systems (MS and FWS), and the secondary side safeguards system (SSSS).

1.2.3.3 Containment Safeguards System

1.2.3.3.1 Safety Features

The safety features limit the potential radiation exposure to the public and to plant personnel following an accidental release of radioactive fission products from the reactor system, particularly as the result of a loss-of-coolant accident (LOCA). These safety features function to localize, control, mitigate, and terminate such accidents, ensuring that 10 CFR 100 guidelines are not exceeded. The safety features include the following systems:

- o Emergency core cooling system (ECCS)

- o Containment spray system

- o Containment fan cooler system

- o Annulus air cleanup system

- o Hydrogen recombiners

1.2.3.3.1.1 Emergency Core Cooling System

The ECCS function is provided by the Integrated Safeguards System (ISS), which injects borated water into the reactor coolant system following a LOCA. This provides cooling to limit core damage, metal-water reactions, and fission product release and ensures adequate shutdown margin regardless of temperature. The ISS also provides continuous long term, post-accident cooling of the core by recirculating borated water between the in-containment Emergency Water Storage Tank (EWST) and the reactor core. See Section 1.2.3.4 of this module for a more detailed discussion of the ISS.

1.2.3.3.1.2 Containment Heat Removal System

The functional performance objective of the containment heat removal system, as an engineered safety features system, is to reduce the containment temperature and pressure following a LOCA or main steam line break (MSLB) inside containment accident, by removing thermal energy from the containment atmosphere. These cooling systems also serve to limit offsite radiation levels by reducing the pressure differential between the containment atmosphere and the external environment, thereby diminishing the driving force for the leakage of fission products from the containment to the environment.

Two separate systems are utilized to perform the containment heat removal function: the containment spray portion of the ISS and the containment fan cooler system. Those components within the ISS that perform a containment spray function are the four low head pumps, the Emergency Water Storage Tank (EWST) and the associated valves, piping, and instrumentation. Within the containment spray ring headers are used to provide containment atmosphere coverage. The containment fan cooler system consists of four fan coolers, which are cooled by component cooling water.

These two systems combined with the containment passive heat sinks are capable of removing sufficient sensible heat and subsequent decay heat from the containment following the hypothesized LOCA or main steam line break accident to maintain the containment design pressure, in accordance with 10CFR50, Appendix A, General Design Criteria 38, "Containment Heat Removal."

1.2.3.3.1.3 Annulus Air Cleanup System

The Annulus Air Cleanup System collects and processes potential airborne contamination due to leakage from the steel containment system. This filtration limits the environmental activity levels following an accident. This system also collects and processes potential airborne contamination resulting from leakage in the ISS recirculation paths outside containment.

1.2.3.3.1.4 Hydrogen Recombiners

Fully redundant electrical hydrogen recombiners inside the containment reduce the percentage of hydrogen in the post-accident containment atmosphere to below combustible levels.

1.2.3.4 Primary Side Safeguards Systems

The Primary Side Safeguards System (PSSS) for the WAPWR is the Westinghouse Integrated Safeguards System (ISS).

The ISS consists of four identical and totally separated mechanical subsystems, which are powered from either two or four separate and redundant emergency electrical power trains, and receive actuation signals from either two or four separate and redundant actuation cabinets. One subsystem is shown schematically in Figure 1.2-9 in its normal valve alignment. Figure 1.2-10 provides a simplified schematic of the four mechanical subsystems.

The basic configuration consists of:

- Four pumping modules each containing one high head and one low head pump
- An emergency water storage tank located inside the containment building
- Four accumulators
- Four core reflood tanks
- Four residual heat removal heat exchangers

The accumulators, core reflood tanks and heat exchangers are located inside the containment building. It is proposed that the four pumping modules be housed in Containment Pressure Pump Enclosures (CPPE's) in order to encompass all piping and components associated with any post accident recirculation of highly radioactive fluid within a containment boundary. This total containment encapsulation concept for the ISS eliminates the potential for post accident releases of highly radioactive liquid or gases into the auxiliary building and subsequently into the environment.

The four pumping modules, are totally independent and identical to each other. The ISS concept recommends that two of the modules be located on the opposite side of the containment 180 degrees apart from the other two modules. This arrangement is shown in Figure 1.2-2 (Sheet 1). Each pumping module contains one low head pump, one low head pump miniflow heat exchanger, one high head pump and the associated piping and valves necessary for these pumps to perform their intended safety functions.

The high head pumps, which perform the safety injection function, are aligned to take suction from the Emergency Water Storage Tank (EWST) and to deliver coolant to the reactor coolant system via the residual heat removal (RHR) heat exchangers and the four separate reactor vessel injection nozzles. The EWST is located at a low elevation inside the containment building.

The low head pumps are primarily residual heat removal pumps which are used for plant cooldown and during refueling operations to remove decay heat. During residual heat removal operation they take suction from the reactor coolant system hot legs and recirculate coolant through the core via the RHR heat exchangers and the four reactor vessel injection nozzles. However, during a LOCA or steam break accident, these pumps function as containment spray pumps. They are aligned to take suction from the EWST and deliver to the containment spray ring headers on receipt of a high containment pressure actuation signal.

The four core reflood tanks provide a supplemental injection flow to the reactor coolant system (RCS) during the post accident reflood phase following an intermediate to large LOCA. The core reflood tanks represent passive injection subsystems which deliver coolant to the RCS via the four reactor vessel injection nozzles. These passive subsystems are low pressure, high resistance, low flow systems.

The four accumulators provide rapid reflood of the reactor vessel lower plenum and downcomer volumes following an intermediate to large LOCA. The accumulator represents a passive injection subsystem which delivers coolant to the RCS via the four RCS cold legs. The accumulators are high pressure, low resistance, high flow systems.

In the event that the normal chemical and volume control system (CVCS) letdown/boration capability was not available, feed and bleed emergency letdown/boration operation would be utilized to achieve a cold shutdown boration of the RCS prior to emergency plant cooldown operations. The CVCS letdown heat exchangers, located inside the containment would permit the letdown flow to be subcooled before it is released into the EWST. The high head pumps would be used during this operation to provide the makeup/borated coolant to the RCS from the EWST.

If four emergency electrical power trains are provided, the electrical loads associated with each ISS subsystem are assigned to one of the four separate and redundant load groups or emergency electrical power trains. These loads

are connected to four separate safeguards vital busses. Each safeguards vital bus is connected to an offsite power source. However, in the event that off-site power is lost, each vital bus is automatically connected to one of four emergency diesel generators.

However, if only two emergency electrical power trains are provided, the electrical load associated with two of the four ISS subsystems are assigned to one of the two separate and redundant load groups or emergency electrical power trains while the electrical loads associated with the other two ISS subsystems are assigned to the second emergency electrical power train. Only two safeguards vital busses and two emergency diesel generators would be associated with two emergency electrical trains.

1.2.3.5 Secondary Side Safeguards System

The primary function of the Secondary Side Safeguards System is to remove heat from the core through the steam generators during any plant condition when the normal secondary side systems (Main Steam and Feedwater Systems) are not available. This function is met by a combination of two secondary side systems, a Startup Feedwater System (SFWS) and an Emergency Feedwater System (EFWS).

The EFWS is a fully safety grade system while the SFWS is a control grade system. The EFWS is designed to meet all the required safety criteria as defined in the "Secondary Side Safeguards System" module. The SFWS, although not required to mitigate the consequences of postulated accidents, provides additional reliability and diversity of the EFWS. The SFWS also serves to minimize the number of EFWS actuations required which enhances the reliability of the EFWS. Figures 1.2-11 and 1.2-12 illustrate the general layout of the SFWS and EFWS equipment.

During normal power operation the steam generators are fed by the main feedwater pumps with steam sent to the turbine. A Startup Feedwater System (SFWS) feeds the steam generators during normal plant startup and shutdown. Under

these conditions steam from the steam generators is sent to the main condenser. The SFWS is also actuated automatically to provide feed following a reactor trip, loss of main feed, loss of offsite power, and other anticipated transients.

The Emergency Feedwater System (EFWS) consists of two totally independent and completely separated subsystems each of which receives electrical power from one of two separate safety class IE electrical power trains. Each subsystem consists of an emergency feedwater storage tank, one motor driven emergency feedwater pump, one turbine driven emergency feedwater pump and the required piping, valves, instruments and controls necessary for system operation.

The motor driven and turbine driven pumps and the EFWST's are located in the safeguards area. The pumps in one subsystem are located on the opposite side of the reactor containment as the pumps in the other subsystem. The use of both motor driven and turbine driven pumps satisfies the requirement that the pumps be powered by diverse power sources. The turbine driven pumps are not dependent on A.C. power. When in operation, the emergency feedwater pumps take suction from the emergency feedwater storage tanks and discharge the water into the main feedline downstream of the isolaton valve.

The pumps are sized such that any two of the four pumps delivering to any two of the four steam generators provides the minimum required emergency feedwater flow.

1.2.3.6 Plant Instrumentation and Control Systems

The plant instrumentation and control systems are described in detail in Chapter 7 of RESAR-SP/90 PDA Module 9, "I&C and Electrical Power".

1.2.3.6.1 Integrated Control System

The purpose of the WAPWR integrated control system is to regulate and maintain the plant operating conditions within prescribed limits over the entire operating range. Those functions which are monitored and controlled include RCS

temperature, neutron power distribution, RCS pressure, pressurizer water level, steam generator water level, and steam dumps.

The integrated control system is comprised of the following systems which perform control functions in order to maintain safe conditions during startup, operation, and shutdown:

- <u>Advanced Power Control System (APCS)</u>

  The advanced power control system provides an integrated control of these systems such that the core axial power distribution and other parameters are maintained automatically. Rather than controlling just a single mechanism such as boron concentration or control rod position, this control system will provide an integrated response to reactivity control using the following subsystems:

  Rod Control System

  The rod control system is designed to maintain nuclear power and reactor coolant temperature, without challenging the protection systems, during normal operating transients. To maintain temperature within a desired control band, neutron absorbing control rods are inserted or withdrawn from the core.

  Boron Control System

  The boron control system maintains the reactor coolant boron concentration either automatically as directed by the APCS or by the operator in such a manner that the axial nuclear power distribution and other operating conditions are maintained.

  Gray Rod Control System

  The gray rod assemblies are used in conjunction with control rods and other mechanisms for controlling reactivity. They are either full inserted or withdrawn under automatic control.

- Pressurizer Pressure Control

  The pressurizer pressure control system acts to maintain or restore the pressurizer pressure to the nominal operating value during normal operation or following transients.

- Pressurizer Water Level Control

  The pressurizer water level control system regulates and maintains or restores pressurizer water level to its required value.

- Steam Generator Water Level Control System

  The steam generator water level control system maintains the steam generator water level within operating limits during steady state operation, and during normal transients. The water level control system also restores normal water level following a plant trip.

- Steam Dump Control

  The steam dump control system controls an intentional release of steam bypassing the turbine to prevent a reactor trip following a sudden loss of electrical load. The system ensures that stored energy and residual heat are removed following a reactor trip so that the plant can be brought to equilibrium no-load conditions without actuation of the steam generator safety valves. The steam dump control system is also used for maintaining the plant at no-load or low load conditions and to facilitate controlled cooldown of the plant.

1.2.3.6.2 Integrated Protection System (IPS)

During normal operation, administrative procedures and the plant control systems serve to maintain the reactor in a safe state, and in the case of a fault serve to prevent damage to the three barriers (fuel clad, reactor coolant system and reactor containment building) to avoid a release of radioactive material. Certain accident conditions may occur which can cause one or more of the

three barriers to be threatened. The integrated protection system (IPS) monitors plant parameters and automatically initiates various protective functions to maintain the integrity of any of the three barriers. The IPS performs its functions by monitoring the plant parameters using a variety of sensors, performing calculations, comparisons and logic based on those sensor inputs and actuating a variety of equipment if parameter setpoints are exceeded. A detailed description of the IPS can be found in RESAR-SP/90 PDA Module 9, "I&C and Electric Power".

1.2.3.7 Steam and Power Conversion

The WAPWR nuclear steam supply system (NSSS) is designed to deliver steam via the main steam supply system to a turbine-generator unit located in the adjacent turbine generator building. It is expected that the turbine generator would be of conventional design with a gross electrical rating of approximately 1350 MW.

The turbine-generator building (or turbine-generator "island") which houses the steam and power conversion systems and equipment is designed and furnished by the plant specific applicant. The main systems/equipment are:

- o Turbine-generator unit and auxiliary systems
- o Main-steam supply system
- o Main condensors and steam dump system
- o Circulating water system
- o Condensate and main feedwater system
- o Startup feedwater system

There are major interfaces between the NSSS and the power conversion systems including the main steam and main feedwater systems which interconnect to the nuclear steam generators. All safety related (Class II) equipment in these two systems is located within the NPB.

The steam and power conversion systems are described in more detail in RESAR-SP/90 PDA Module 8, "Steam and Power Conversion". Interface areas for systems and equipment with the NPB are included in Tables 1.1-1 and 1.9-1.

1.2.3.8 Auxiliary Systems

1.2.3.8.1 Chemical and Volume Control System (CVCS)

The chemical and volume control system (CVCS) provides the chemical, volume and boron control functions. The reactor coolant system is continuously purified; first a small fraction of the reactor coolant is removed through the letdown system and the letdown fluid is cooled in the regenerative heat exchanger. From there, the coolant flows to a letdown heat exchanger, where it is cooled further, and is depressurized. After exiting containment it is depressurized further and then passed through demineralizers where corrosion and fission products are removed. The coolant then passes through a filter and sprayed into the volume control tank, from which it is returned to the reactor coolant system by the charging system. The system is not relied on to mitigate the consequences of any postulated accidents and therefore is not, for the most part, required to be safety grade. However, a small portion of the CVCS letdown line which is used as part of the emergency letdown path inside containment must be safety grade and a few components of the CVCS must meet safety grade isolation requirements.

The CVCS is designed to be very reliable so that it could in fact be used, although not required, during many postulated accidents to eliminate the need for actuation of the engineered safeguards systems. Thus, the CVCS can be used as additional means of providing emergency boration or emergency makeup water to the reactor coolant system, diverse from the dedicated safety grade ISS. This enhances the overall reliability of the emergency boration and makeup functions of the plant.

The CVCS serves to provide several functions during normal operation including charging and letdown; a means for filling, draining and pressure testing of the reactor coolant system; reactor coolant pump seal injection; reactor coolant purification; chemistry control; boron control; and reactor coolant inventory control and makeup. It also provides a redundant control grade seal water injection capability which is independent of all AC power. See Figure 1.2-13 for a flow diagram of the CVCS.

1.2.3.8.2 Residual Heat Removal System (RHR)

The functions performed by a conventional nuclear plant residual heat removal (RHR) system are integrated within the ISS of the WAPWR. Those components within the ISS that perform an RHR function are the four low head pumps, four RHR heat exchangers, and the associated valves, piping, and instrumentation. Refer to 1.2.3.4 of this module and section 5.4.7 of RESAR-SP/90 PDA Module 1, "Primary Side Safeguards System" for further discussion.

1.2.3.8.3 Fuel/Component Storage and Handling Systems

A fuel handling, transfer and storage system provides for the safe handling, transfer, and storage of components, fuel assemblies and control element assemblies for refueling or maintenance purposes. This system provides for the assembly, disassembly, and storage of the reactor vessel head and internals, and includes:

1. A refueling machine (in-containment)
2. A fuel transfer system
3. A vessel head single-ring multi-stud tensioner
4. A fuel transfer tube sleeve grid quick opening hatch
5. A (spent fuel pit) fuel handling machine and fuel storage racks in the fuel handling building
6. Integrated head package
7. Separate upper internals and lower internals lifting rigs.
8. Various tooling required for initial assembly and core load

1.2.3.8.4 Component Cooling Water System

The component cooling water system (CCWS) is a closed loop circulating water system serving heat exchangers whose operation is required for the operation and safe shutdown of the reactor. Heat is removed from the closed loop by the service water system. Radiation monitors are provided to detect any radioactive leakage into the component cooling system.

The CCWS consists of two independent trains with two pumps per train. Each train serves both safety and non-safety related loads, with the non-safety related loads and associated piping automatically isolated during accident conditions. Corrosion-inhibited demineralized water is circulated through the RHR pumps and water coolers, the HHSI pump and motor coolers, the RHR pump mini flow heat exchangers, the charging pump and motor coolers, the RHR heat exchangers, the seal injection heat exchanger, the spent fuel pit heat exchangers, the instrument air package coolers, the reactor coolant pump and motor coolers, the containment fan coolers, and other miscellaneous non-safety related equipment coolers.

1.2.3.8.5 Sampling Systems

Four sampling systems are provided: nuclear sampling system - liquid, nuclear sampling system - gaseous, turbine plant sampling system, and post-accident sampling system. These systems are used for determining both chemical and radiochemical conditions of the various fluids used in the plant.

1.2.3.8.6 Emergency Diesel Generators and Support Systems

The emergency diesel generators provide emergency standby power to the Class 1E electrical power system. Upon loss of offsite power, either diesel generator source and its associated bus has the capacity to power the equipment required to safely shut down the reactor and mitigate the consequences of design basis accidents.

The plant electric power system and the diesels are described in detail in RESAR-SP/90 PDA Module 9, "I&C and Electric Power".

1.2.3.8.7 Plant Ventilation Systems

Within the nuclear power block, separate ventilation systems are provided for the containment and reactor external buildings.

The containment is normally cooled by the Containment Fan Cooler System; this system is safety grade and is also used for post-accident containment cooling. The containment further includes a system to cool the drive mechanisms located on the reactor vessel head, a reactor cavity cooling system, a preaccess filtration system, and a purge system.

The HVAC system for the reactor external building is subdivided into a number of smaller subsystems, each tailored to a specific area. The Reactor External Building Ventilation system basically serves the radiation controlled zone, including the fuel handling and annulus areas. However, these areas are switched to the Annulus Air Cleanup System during accident conditions; this system includes filters for control of radioactive leakage. The clean area of the reactor external building includes safety class, redundant ventilation systems for the diesel-generator rooms, the safety related equipment rooms and the main control room.

1.2.3.8.8 Plant Fire Protection System

The major fire protection system contains fire pumps which supply the various hydrants, hose stations, sprinklers, and deluge systems. Hydrants and hose stations are manually operated; the sprinkler and deluge systems are a combination of automatic and manually actuated systems. In addition to these facilities, chemical fire-extinguishing equipment is provided to accommodate special requirements for various classes of hazards. Noncombustible and fire-resistant materials are selected for use wherever practical throughout the facility, particularly in critical portions of the plant such as the containment, control room, and components of the safety features system. Safety trains are separated by fire walls.

1.2.3.8.9 Service Water System

Cooling water to equipment and systems within the NPB (the diesel generators, the component cooling water heat exchangers and to the essential chiller units) is supplied by service water pumps taking suction from the body of water that serves as the ultimate heat sink (UHS). Two separate piping trains

with two pumps per train supply cooling water flow to the respective and parallel arranged equipment groups. Water is pumped from the UHS, supplied to the equipment groups, and returned to the UHS forming a once-through cooling system configuration. The service water system pumps are located outside the NPB, with piping brought into the NPB as required.

1.2.3.9 Facility Arrangement

As stated earlier, Figure 1.2-2 (sheets 1 through 9) provide the general arrangement of major structures and equipment including plan and elevation drawings.

(a,c)

Also Available On
Aperture Card

TI
APERTURE
CARD

FIGURE 1.2-1 WAPWR PLOT PLAN

8405230322 -01

MARCH, 1984

(a,c)

Also Available On
Aperture Card

TI
APERTURE
CARD

FIGURE 1.2-2 WAPWR PLANT LAYOUT
(SHEETS 1 THROUGH 9) PROPRIETARY

MARCH, 1984

8405?30322-02

KEY TO FIGURE 1.2-2 ABBREVIATIONS

(Page 1 of 7)

Sheet-1	Plan at Elev. 72.0 m
HHSI PUMP	High Head Safety Injection Pump
RHR/CS PUMP	Residual Heat Removal and Containment Spray Pump
MDEFP PUMP	Motor Driven Emergency Feedwater Pump
TDEFP	Turbine Driven Emergency Feedwater Pump
R/B CDT	Reactor Building Containment Drain Tank
R/B CDP	Reactor Building Containment Drain Pump
Sheet-2	Plan at Elev. 77.4 m
BCP	Back-Up Battery Charger Panel
CIV	Computer Inverter Cubicle
IV	Inverter Cubicle
BTC	Back-Up Transformer Cubicle
CVT	Const. Voltage Trans. Cubicle
BAT	Boric Acid Tank
BA TRANSF PUMP	Boric Acid Transfer Pump (2)
RHT	Recycle Holdup Tank
REFP	Recycle Evaporator Feed Pump
SFPHE	Spent Fuel Pit Heat Exchanger
PD CHARG P	Positive Displacement Charging Pump
AUX STM COND RETURN TR	Auxiliary Steam Condensate Return Tank
SFPP	Spent Fuel Pit Pump
SFPSP	Spent Fuel Pit Skimmer Pump

Sheet-2 Cont.	
* EWRPP	Emergency Water Recirculation and Purification Pump
DRYER	Instrument Air Dryer
IA	Instrument Air
REC	Instrument Air Reservoir
IAC	Instrument Air Cooler
CCW	Component Cooling Water Pump
* EFWST	Emergency Feedwater Storage Tank
CHEM DR TK	Chemical Drain Tank
CDP	Chemical Drain Pump
ADT	Auxiliary Drain Tank
WHTRP	Waste Holdup Tank Recirculation Pump
WHT	Waste Holdup Tank
SRT	Spent Resin Tank
SR SLUICE PUMP'S	Spent Resin Sluice Pump
CV SUMP PUMP	Containment Vessel Sump Pump
RC DRAIN TANK PUMP	Reactor Coolant Drain Tank Pump
Sheet-3	Plan at Elev. 83.2 m
CV RCDT HT EXCH	Reactor Coolant Drain Tank Heat Exchanger
RCDT	Reactor Coolant Drain Tank
EWST	Emergency Water Storage Tank
Sheet-4	Plan at Elev. 84.8 m
BAT	Boric Acid Tank
RH TK	Recycle Holdup Tank
VCT	Volume Control Tank

Sheet-4 Cont.	
PMW PUMP	Primary Makeup Water Pump
PASS DRAIN TK PKG.	Post Accident Sampling System
CH WTR PUMP	Chillers Water Pump
CCW HX	Component Cooling Water Heat Exchanger
D/G	Diesel Generator
EFWST	Emergency Feed Water Storage Tank
Sheet-5	Plan at Elev. 92.8m
IVC	Inverter Cubicle
IVA	Inverter Cubicle
BCP	Battery Charger Panel
BTC	Back-Up Transformer Cubicle
SWGR	Switchgear
PASS EQ ROOM	Post Accident Sampling System Equip. Room
CVCS & R/D CONTROL ROOM	Control Volume Chemical Makeup
D.G. AUX	Diesel Generator Auxiliary
EM PNL ROOM	Emergency Panel
BAT. "A" ROOM	Battery "A" Room
INV "A" ROOM	Inverter "A" Room
SWGR "A" ROOM	Switchgear "A" Room
SI HX	Seal Injection Heat Exchanger
BRS EVAP	Boron Recycle System Evaporator
LD TK	Day Tank
V.R.	Valves Room
ICIS D/U	Incore Instrument Drive Unit
RHR HX	Residual Heat Removal Heat Exchanger

Sheet-5 Cont.	
ACCUM TANK	Accumulator Tank
LET DOWN HX	Letdown Heat Exchanger
RCP	Reactor Coolant Pump
UPR INTL	Upper Internals
LWR INTL	Lower Internals
S/G	Steam Generator
R/V	Reactor Vessel
CRT	Core Reflood Tank
PRESS	Pressurizer
ICIS	In Core Instrumentation
Sheet-6	Plan at Elev. 100.0m
R/V	Reactor Vessel
S/G	Steam Generator
RCP	Reactor
C/R	Control Room
CRDM POWER RM	Control Rod Drive Mechanism Power Room
CRDM COOLING UNIT	Control Rod Drive Mechanism Cooling Unit
MCC ROOM	Motor Control Center Room
D/G Control Room	Diesel Generator Control Room
D/G Control Room	Diesel Generator Control Room
M/G SET	Motor/Generator Set
COT	Coolant Oil Tank
FOT	Fuel Oil Tank
FO FILTER	Fuel Oil Filter
AIR RCVR & COMP	Receiver and Compressor

Sheet-6 Cont.

EGB	Emergency Diesel Generator Control Board
GCC	Emergency Diesel Generator Motor Control Center
ACB	Auxiliary Control Board
MCB	Main Control Board
TAB	Turbine Gen. Aux. Board
HSB	House Service Board
VB	Ventilation Board
TMB	Transmission Board
CMB	Common Board
ECC	Engineer CRT Console
TW	Typewriter
LCP	Line Printer

Sheet-7	Plan at Elev. 107.6m
RE/B	Reactor External Building
D/G	Diesel Generator
MCR	Main Control Room
SWGR	Switchgear
CONT'MT	Containment
EFWP	Emergency Feedwater Pump
T/D	Turbine Driven
M/D	Motor Driven
M/S	Main Steam
COMP. RM.	Computer Room
R/B	Reactor Building

Sheet-8	Section A-A
CONT	Containment
M.S. TUNNEL	Main Steam Tunnel
CORR.	Corridor
P.S.	Pipe Space
S.W. PIPE TUNNEL	Service Water Pipe Tunnel
M.S.	Main Steam
F.W.	Feed Water
H.W.L	High Water Level
EFWST	Emergency Feed Water Storage Tank
TDEF PUMP	Turbine Driven Emergency Feedwater Pump
TDEF V.R.	Turbine Driven Emergency Feedwater Valve Room
EWST	Emergency Water Storage Tank
TK	Tank
PRESZ	Pressurizer
I.C.I.S. D/U	Incore Instrument Drive Unit
R/V	Reactor Vessel
S/G	Steam Generator
REFP	Recycle Evaporator Feed Pump
V.R.	Valve Room
MH	Main Hook

Sheet-9	Section B-B
D/G	Diesel Generator
D/G Aux	Diesel Generator Auxiliary
V.R.	Valve Room

Sheet-9 Cont.

RHR/CSP	Residual Heat Removal and Containment Spray Pump
EWST	Emergency Water Storage Tank
CRT	Core Reflood Tank
CONT RECIRC. UNIT	Containment Recirculation Unit
S/G	Steam Generator
R/V	Reactor Vessel
M.H.	Max. High
WHT	Waste Holdup Tank
RCP	Reactor Coolant Pump
T.O.R.	Top of Rails
SW	Service Water
PS	Pipe Space

(a,c)

TI
APERTURE
CARD

Also Available On
Aperture Card

FIGURE 1.2-3 FUEL
ASSEMBLY OUTLINE

8405230322-03

MARCH, 1984

(a,c)

FIGURE 1.2-4 REACTOR VESSEL

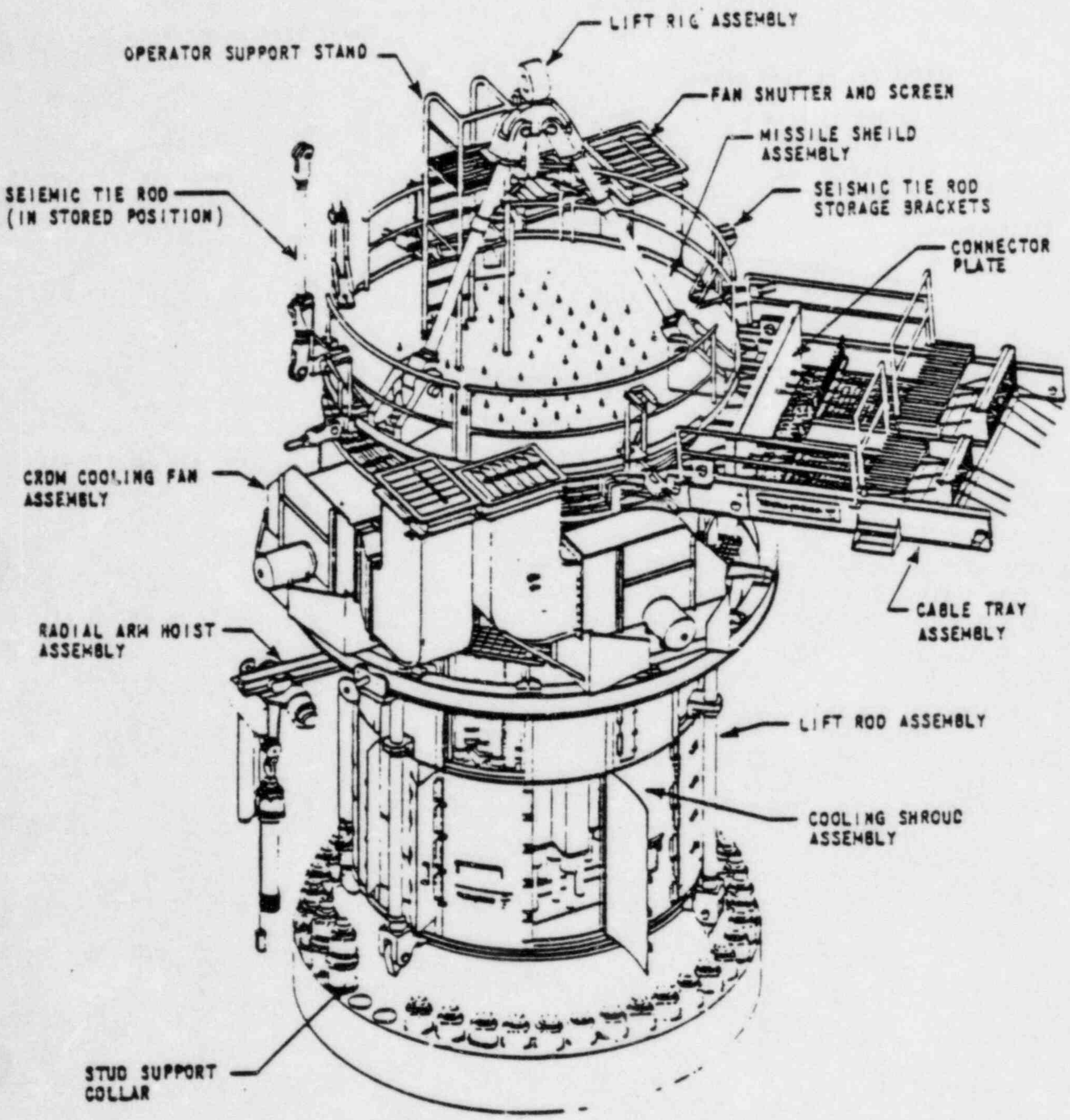

FIGURE 1.2-5 INTEGRATED HEAD PACKAGE

(a,c)

FIGURE 1.2-6 DISPLACER ROD DRIVE MECHANISM

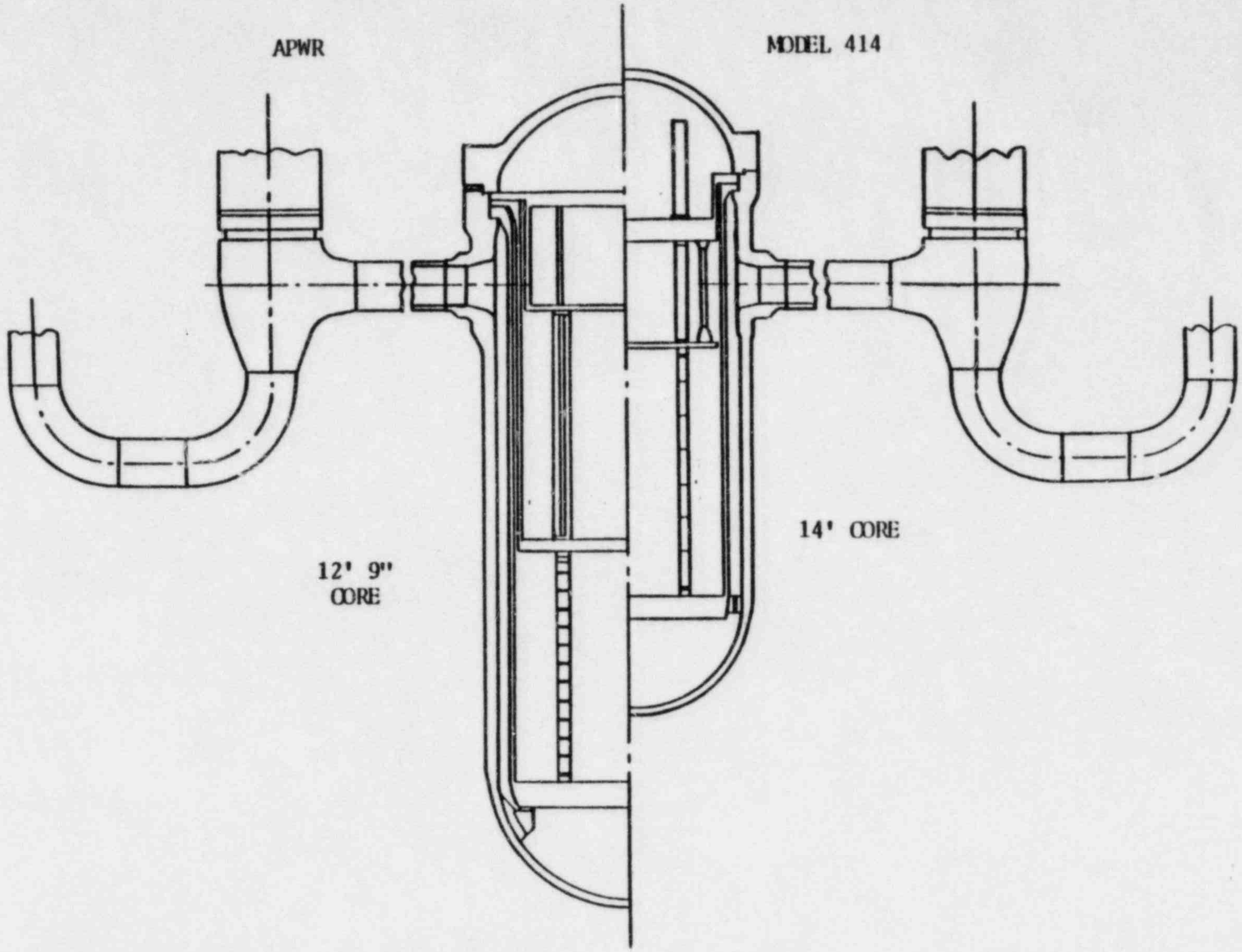

FIGURE 1.2-7 COMPARISON OF 414 AND WAPWR

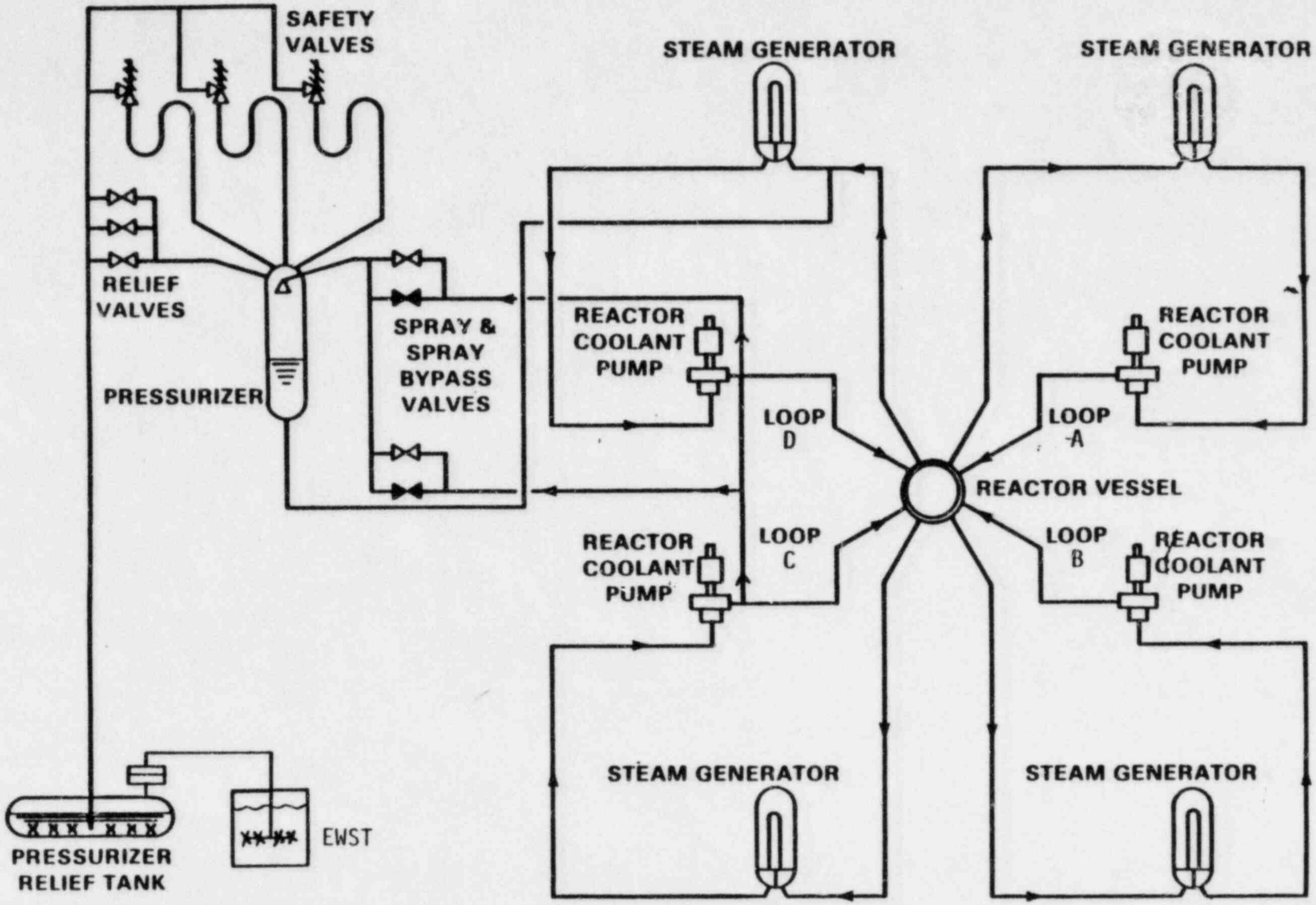

FIGURE 1.2-8 REACTOR COOLANT SYSTEM

(a,c)

FIGURE 1.2 -9 ONE OF FOUR INTEGRATED SAFEGUARDS SUBSYSTEMS NORMAL VALVE ALIGNMENT

(a,c)

FIGURE 1.2-10 INTEGRATED SAFEGUARDS (FOUR MECHANICAL SUBSYSTEMS)

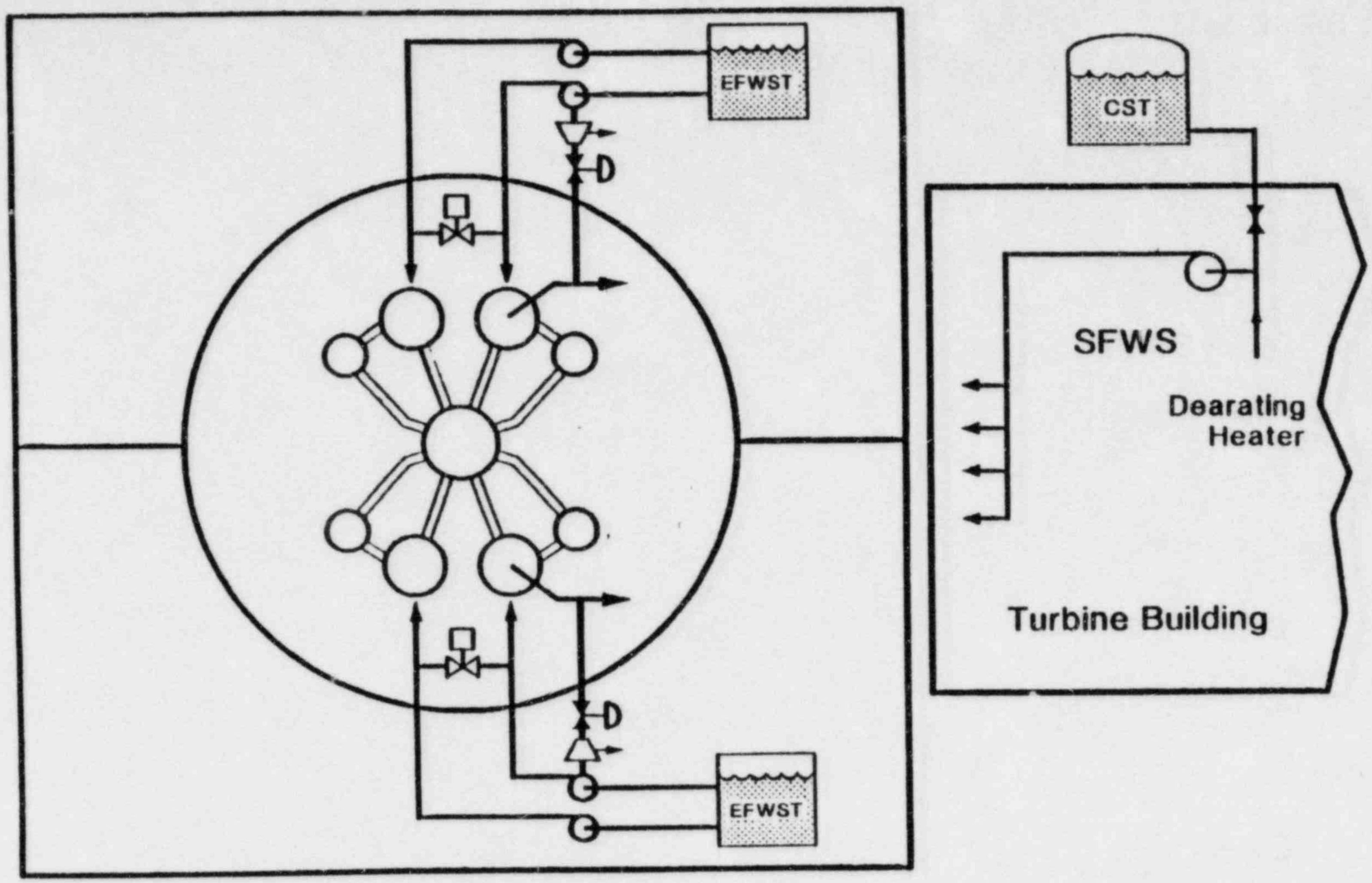

FIGURE 1.2-11 GENERAL LAYOUT OF SFWS/EFWS EQUIPMENT

(a,c)

FIGURE 1.2-12 EMERGENCY FEEDWATER SYSTEM

(a,c)

FIGURE 1.2-13 CHEMICAL VOLUME AND CONTROL SYSTEM

1.3 COMPARISON TABLES

1.3.1 Comparisons with Similar Facility Designs

Tables presenting a design comparison of the major parameters and features of the WAPWR with RESAR-414 (Docket No. STN-50-572; PDA-13), RESAR-3S (Docket No. STN-50-545; PDA-7), and RESAR-41 (Docket No. STN 50-480; PDA-3) are found in the particular module describing the parameters or features.

1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

1.4.1 Applicant

Refer to the plant specific applicant's safety analysis report for a discussion of the applicant's organization, qualifications, and experience.

1.4.2 Westinghouse

Westinghouse is to supply those structures, systems, and components as described in Section 1.1.1.1, "Scope." Westinghouse also will supply interface and design criteria as specified in Section 1.1.1.1. Chapter 17.0 in the RESAR-SP/90 integrated PDA submittal provides a discussion of the Westinghouse organizational structure.

Westinghouse has designed, developed, and manufactured nuclear facilities since the 1950s, beginning with the world's first large central station nuclear plant (Shippingport), which has produced power since 1957. Nuclear steam supply systems for more than 100 commercial nuclear power plants with a combined electrical generating capacity in excess of 90,000 MW have been completed or are presently under contract.

Westinghouse pioneered new nuclear design concepts, such as chemical shim control of reactivity and the rod cluster control concept, throughout the last two decades. Among the company's own related manufacturing facilities are the commercial nuclear fuel fabrication facility at Columbia, South Carolina, and nuclear component manufacturing facilities at Tampa, Florida; Pensacola, Florida; Blairsville, Pennsylvania; and Cheswick, Pennsylvania.

Among the Westinghouse PWR plants in operation and under construction in the United States, and around the world, several are unique because of the extent of involvement by Westinghouse, not only as the NSSS supplier, but as the organization responsible for delivery of the entire plant on a "turnkey" basis. For example KORI units 1 and 2, KRSKO, Angra and Napot Point plants are units

for which Westinghouse was contracted by the utility to perform the design, procurement, erection, installation, and startup for the nuclear island and all its ancillary features.

Other U.S. plants which employed Westinghouse under the turnkey concept include R. E. Ginna, H. B. Robinson Unit 2, Point Beach Unit 1 and 2, and Indian Point Units 2 and 3. Likewise for Mihamma Unit 1, Takahama Unit 1, Ringhals Units 2 and 3 and Ohi Units 1 and 2; Westinghouse had an increased scope beyond that of the NSSS vendor.

1.4.3 Other Agents and Contractors

Refer to the plant specific applicant's safety analysis report for a discussion of the organization, qualifications, and experience, as well as the scope of effort; of those agents and contractors other than Westinghouse who will supply equipment and services for plant areas outside the NPB.

1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

The purpose of this section is to present a description of the safety-related research and development programs which are being carried out for, or by, or in conjunction with, Westinghouse and which are applicable to the RESAR-SP/90 scope.

Each of the research and development programs applicable to the RESAR-SP/90 scope are described in the pertinent PDA module. This description includes a summary of the program purpose, pertinent results to date, the facility in which the testing is performed (if applicable), and the status of the research and development effort.

The technical information generated by these research and development programs will be used either to demonstrate the safety of the design and more sharply define margins of conservatism, or will lead to design improvements.

Progress in these development programs is reported on a timely basis. New safety-related research and development programs will also be described in future amendments to RESAR-SP/90, if applicable.

1.6 MATERIAL INCORPORATED BY REFERENCE

Each PDA module incorporates, by reference, certain topical reports which have been previously filed with the U.S. NRC in support of other Westinghouse applications. These topical reports are listed in Section 1.6 of each module, as applicable.

1.7 DRAWINGS AND OTHER DETAILED INFORMATION

Refer to Section 1.7.1 and Section 1.7.2 of each PDA module for a list of pertinent electrical, instrumentation, and control drawings, and piping and instrumentation drawings, respectively.

1.8 CONFORMANCE WITH THE STANDARD REVIEW PLAN

In accordance with 10CFR50.34(g), Table 1.8-1 of each PDA module identifies and evaluates deviations from the acceptance criteria of those sections of the NRC Standard Review Plan (NUREG-0800) pertinent to the subject module. Table 1.8-1 provides this list for the "Introduction and Site" module.

TABLE 1.8-1

STANDARD REVIEW PLAN DEVIATIONS

SRP Acceptance Criteria	Deviation	Section
SRP 2.5.2: The minimum value of the acceleration level for the OBE is currently one-half the reference acceleration for seismic design corresponding to the SSE.	The OBE has been established as 1/3 of SSE	2A.5.2

1.9 STANDARD DESIGNS

1.9.1 Interfaces

This section identifies safety related interfaces between the NPB and the site specific items which will be covered by the utility's SAR referencing RESAR-SP/90. The safety related interfaces have been selected based on review of all interfaces between the NPB and items within the utility's scope. Satisfaction of all the interfaces will ensure that systems, components and structures within the NPB will perform their safety functions. The magnitudes of the interface parameters have been selected to suit a wide range of potential sites. Values outside the range of specified acceptable parameters may be demonstrated to be acceptable; such cases will be documented as exceptions in Section 1.9.2 of the site specific SAR.

Table 1.9-1 provides a listing of specific systems and structures within the NPB that interface with portions of the plant to be provided by the utility. This table identifies the PDA module where the interface requirements are described. These interfaces will generally be satisfied by engineered systems and structures provided by the utility.

Table 1.9-2 provides a listing of programs and analyses to be developed on a site specific basis that must interface with programs initiated during the design of the NPB. This table identifies the PDA module where the interface requirements are described.

Table 1.9-3 provides a listing of site parameters upon which the standard design is based. Typically these parameters must be satisfied by site selection. This table identifies the site parameter, the magnitude specified in design of the NPB and the PDA module where the interface requirements is further described.

1.9.2 Exceptions

Refer to the plant specific applicant's SAR for a listing of and justification for any exceptions the applicant has taken to the design and interface criteria in RESAR-SP/90.

TABLE 1.9-1

INTERFACE AREAS FOR SYSTEMS, COMPONENTS AND STRUCTURES

System Component or Structure	PDA Module
Service Building	7
Waste Management Building	7
Turbine Building	7
Instrumentation and Control Systems	9
Onsite AC Power Systems	13
Fire Protection for Cable Systems	13
Station Service Water System	13
Cooling System for Reactor Auxiliaries	13
Demineralized Water Makeup System	13
Potable and Sanitary Water Systems	13
Condensate Storage Facilities	13
Compressed Air Systems	13
Auxiliary Gas System	13
Waste Building HVAC	13
Fire Protection System	13
Communication Systems	13
Diesel Generator Fuel Oil Storage and Transfer System	13
Diesel Generator Cooling Water System	13
Emergency Response Facility	13
Main Steam Supply System	8
Condensate and Feedwater System	8
Steam Generator Blowdown System	8
Auxiliary Feedwater System	8
Liquid Waste Management Systems	12
Process and Effluent Radiological Monitoring and Sample Systems	12

TABLE 1.9-2

INTERFACING PROGRAMS AND ANALYSES

Program/Analysis	PDA Module
Inservice Inspection Program	7
Initial Test Program	14
Industrial Security	16
10CFR Part 50 App. I	11
Probabilistic Risk Analysis	16
Soil-Structure Interaction Analysis	7
Turbine Missile Analysis	7

TABLE 1.9-3

SITE INTERFACE PARAMETERS

Consideration	Parameter	PDA Module
1. Operating Basis Wind	100 yr. fastest mile wind speed $\leq$ 130 mph	3
2. Tornado Wind Speed	$\leq$ 320 mph	3
3. Tornado Missiles	$\leq$ ANSI/ANS 2.3 - 1983 Standard Design Missile Spectrum for Wind Velocity of 320 mph	3
4. Safe Shutdown Earth-quake	$\leq$ 0.3G Horizontal ZPA with Reg. Guide 1.60 Spectra	3
5. Operating Basis Earth-quake	$\leq$ 0.1G Horizontal ZPA with Reg. Guide 1.60 Spectra	3
6. Soil Shear Wave Velocity (V)	$\leq$ 1000 ft/sec	3
7. Soil Bearing Strength	Must be capable of supporting NPB (8 KSF static bearing pressure) under all specified conditions	3
8. Flood Level	$\leq$ Finished Grade	3
9. Safety Related Cooling Water	Max. temperature at intakes $\leq$ 95°F. Flow (later)	13
10. Air Temperature	Minimum $\geq$ (-25° F) Maximum $\leq$ (100°F)	7 13
11. Probable Max. Pre-cipitation	$\leq$ 12 inches/hr	13
12. Snow Load	$\leq$ 80 psf	7
13. Accidents External to Plant	Any accident for which the consequences exceed Part 100 guidelines must have a low probability of occurrence.	3
14. Population Distribution	Population distribution must be within the bounds used in the PRA analysis	16