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.

1.2 GENERAL PLANT DESCRIPTION

1.2.2 Principal Design Criteria

RESAR-SP/90 is designed to comply with 10CFR 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 PDA Module 7, "Structural/Equipment Design." Those General Design Criteria applicable to this module are listed in Section 3.1 of this module.

1.2.3 Plant Description

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 mechanisms 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 low 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 the 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) [ ] 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 assembled in enclosures 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 utilized U-238 enriched with approximately [ ] weight percent of U-235. Fuel rods are comprised of stacked ceramic 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 1.2-1. (a,c) (a,c)

The WAPWR reactor vessel (which is described in detail in Section 5.3 of RESAR-SP/90 PDA Module 4, "Reactor Coolant System") is a large cylindrical pressure vessel with a welded hemispherical bottom head and removable flanged and gasket upper head (see Figure 1.2-2) 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 1.2-3).

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 in 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 1.2-4).

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.

The similarity of the WAPWR and the 4XL Models is illustrated in Figure 1.2-5, 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.

Figure 1.2-1 Fuel Assembly Outline (PROPRIETARY)

Figure 1.2-2 Reactor Vessel (PROPRIETARY)

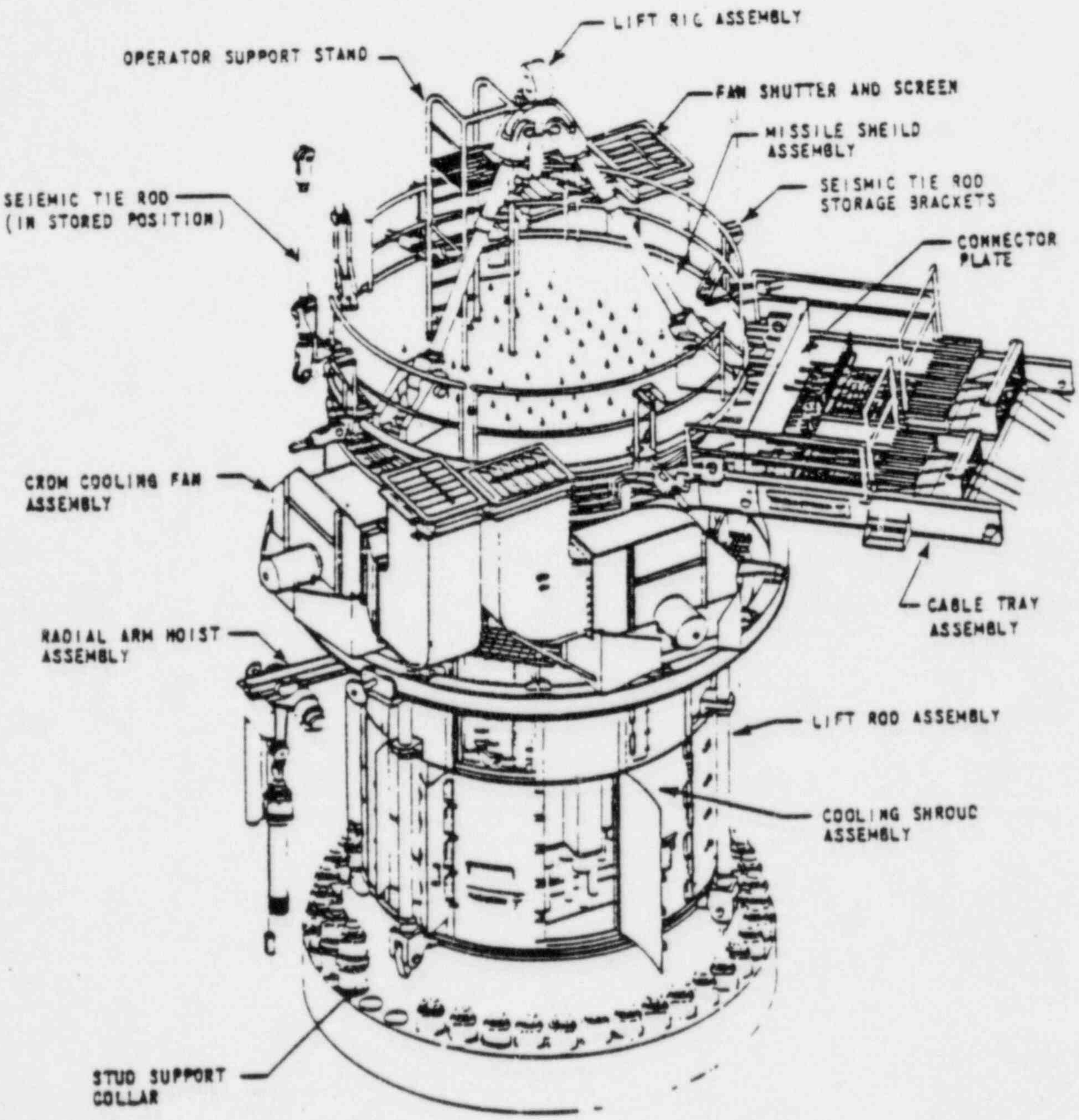

FIGURE 1.2-3 INTEGRATED HEAD PACKAGE

(a,c)

FIGURE 1.2-4 DISPLACER ROD DRIVE MECHANISM

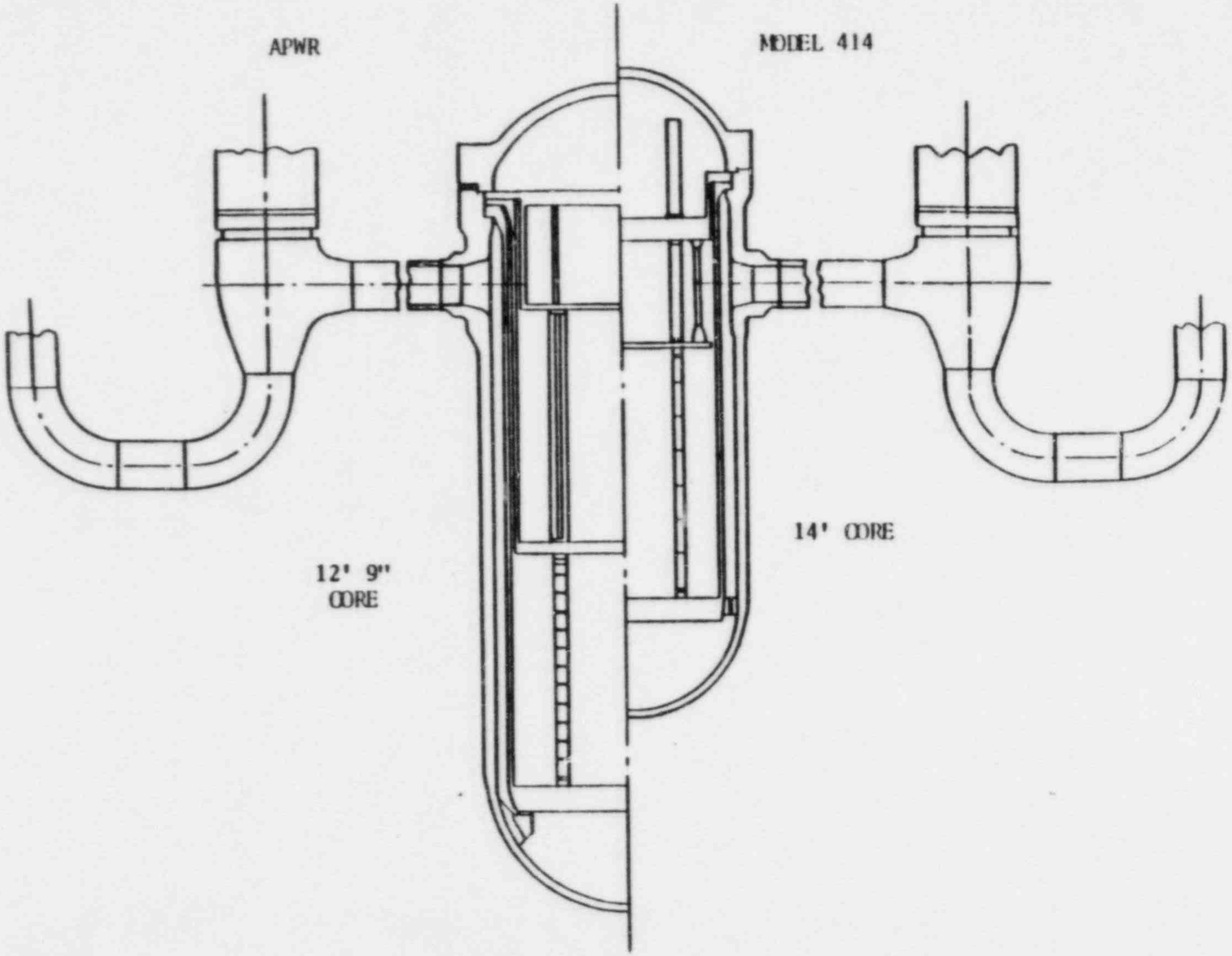

FIGURE 1.2-5 COMPARISON OF 414 AND WAPWR

1.3 COMPARISON TABLES

1.3.1 Comparison With Similar Facility Designs

Table 1.3-1 presents a design comparison of the major parameters and features of the WAPWR reactor system 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).

TABLE 1.3-1

DESIGN COMPARISON

Parameter or Feature	RESAR-SP/90	RESAR-414	RESAR-3S	RESAR-41
Reactor core heat output (MWt)	3800	3800	3411	3800
Total thermal flow rate (10^6 lb/hr)	150.6	150.5	140.3	144.7
Reactor coolant system temperatures (°F)				
1. Core outlet	625.4	627.4	620.8	626.6
2. Vessel outlet	621.1	624.9	618.3	624.0
3. Core average	596.2	597.5	589.4	593.2
4. Vessel average	590.1	596.1	588.2	591.8
5. Core inlet	559.1	563.8	558.1	559.8
6. Vessel inlet	559.1	563.8	558.1	559.8
Average linear power (kW/ft)	5.06	5.20	5.44	5.33
Peak linear power for normal operation (kW/ft)	13.2	14.0	12.6	13.3
Heat flux hot channel factor, F_Q	2.60	2.70	2.32	2.50
Fuel assembly array	19 x 19	17 x 17	17 x 17	17 x 17
Number of fuel assemblies	193	193	193	193

TABLE 1.3-1 (cont)

DESIGN COMPARISON

Parameter or Feature	RESAR-SP/90	RESAR-414	RESAR-3S	RESAR-41
Uranium dioxide rods per assembly	296	264	264	264
Fuel weight as uranium dioxide (lb)	[ ] (a,c)	259,860	222,739	253,675
Number of grids per assembly	10 (8 Zircaloy, 2 Inconel)	9-Type R	8-Type R	9-Type R
Rod cluster control assemblies				
1. Number of full/part length	69/0	57/0	53/8	61/8
2. Absorber material	B_4C and Ag-In-Cd	Ag-In-Cd	Ag-In-Cd	Ag-In-Cd
3. Clad material	Stainless Steel	Stainless Steel	Stainless Steel	Stainless Steel
4. Clad thickness (in.)	0.063	0.0385	0.0185	0.0185
Equivalent core diameter (in.)	156.7	132.7	132.7	132.7
Active fuel length (in.)	[ ] (a,c)	168	143.7	164
Fuel enrichments (weight percent)	[ ] (a,c)			
1. Region 1		1.60	2.10	2.10
2. Region 2		2.40	2.60	2.60
3. Region 3		3.10	3.10	3.10

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 reactor system are described below. 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, as appropriate.

1.5.1 Fuel System Tests

1.5.1.1 Fuel Assembly Tests

1.5.1.1.1 Fuel Assembly Structural Tests

Test Purpose

The purpose of these tests is to provide lateral and axial stiffnesses, vibratory characteristics and impact strengths of the prototype WAPWR fuel assembly.

Results

(Later)

Test Facility

These tests will be conducted at the Mitsubishi Atomic Power Industries (MAPI) Nuclear Development Facility.

Status

The fuel assembly is being manufactured and the test fixtures are being constructed.

1.5.1.1.2 Fuel Assembly Hydraulic Flow Test

Test Purpose

The purpose of this test is to obtain fuel assembly pressure drops, lift forces, and fuel rod and assembly vibration and fretting wear data with a prototype WAPWR fuel assembly in a full flow loop.

Results

(Later)

Test Facility

This test will be conducted at the Westinghouse Fuel Assembly Test System (FATS) hydraulic loop at the Forest Hills Laboratory; Forest Hills, PA.

Status

The fuel assembly and loop internals are currently being manufactured.

1.5.1.2 Core Components Tests

1.5.1.2.1 Rod Cluster Control and Gray Rod Assembly Tests

Test Purpose

The purpose of these tests is to submit prototype WAPWR drive line components to simulated plant conditions to determine hydraulic, vibratory and wear characteristics.

Results

(Later)

Test Facility

These tests will be conducted at the Westinghouse D-Loop test facility at the Forest Hills Laboratory; Forest Hills, PA.

Status

The rod clusters and loop internals are currently being manufactured.

1.5.1.2.2 Water Displacer Rod Assembly Tests

Test Purpose

The purpose of these tests is to submit prototype drive line components to simulated plant conditions to determine hydraulic, vibratory and wear characteristics.

Results

(Later)

Test Facility

These tests will be conducted at the Mitsubishi Atomic Power Industries (MAPI) Nuclear Development Facility.

Status

The test loop is currently being constructed.

1.5.1.3 Drive Mechanism Tests

1.5.1.3.1 Control Rod Drive Mechanism Tests

Test Purpose

The purpose of these tests is to measure the electrical, mechanical, hydraulic and thermal performance of the electro-mechanical stepping mechanism and its rod position indicator system in drive line operation representing accelerated design lifetime in an operating plant. The testing program includes approximately 230 control rod drops and 7.5×10^6 mechanism steps. Rod position readout, rod drop time, electrical and mechanical operation of latching and unlatching, mechanism operating temperature and heat load measurements are made during cycling followed by disassembly for inspection and wear measurements.

Results

(Later)

Test Facility

These tests will be conducted at the Westinghouse D-Loop test facility at the Forest Hills Laboratory, Forest Hills, PA. Prototype hardware of an entire control rod drive line will be tested in the loop at operating plant pressure, temperature, drive line flow rate and water chemistry.

Status

Testing is to be conducted in 1985.

1.5.1.3.2 Water Displacer Rod Drive Mechanism Tests

Test Purpose

The purpose of these tests is to measure the mechanical and hydraulic performance of the hydraulic mechanism and its rod position indicator system in drive line operation representing accelerated design life time in an operating plant. The testing program includes approximately 200 insertion and withdrawal cycles. Rod position readout, insertion and withdrawal velocity, and piston ring pressure drop and leakage rate will be measured during cycling followed by disassembly for inspection and wear measurements.

Results

(Later)

Test Facility

These tests will be performed at the D-Loop facility at Mitsubishi, Takasago Institute in Japan. Prototype hardware of an entire displacer rod drive line will be tested in the loop at operating plant pressure, temperature, drive line flow rate and water chemistry.

Status

Testing is to be conducted in 1986.

1.5.2 Reactor Internals Design Verification Tests

Test Purpose

The purpose of these tests is to measure mechanical and hydraulic performance and operating characteristics of the interaction of internals components at representative plant operating conditions. The primary areas of investigation include overall drive line performance, hydraulic losses, flow and pressure distribution, heat transfer, flow induced vibration and hydraulic loads, operating natural frequencies and component wear. Testing programs include full scale, partial and complete driveline assemblies and a sub-scale model of the reactor internals.

Results

(Later)

Test Facility

Test loops at Westinghouse, Forest Hills Laboratory and Mitsubishi, Takasago Institute in Japan will be used. Prototype and sub-scale model components will be tested at representative operating plant pressure, temperature, flow rate and water chemistry.

Status

Testing to be conducted in 1985 and 1986.

1.6 MATERIAL INCORPORATED BY REFERENCE

The WAPWR reactor system module incorporates, by reference, certain topical reports. The topical reports, listed in Table 1.6-1, have been filed previously in support of other Westinghouse applications.

The legend for the review status code letter follows:

A - U.S. Nuclear Regulatory Commission review complete; USNRC acceptance letter issued.

AE - U.S. Nuclear Regulatory Commission accepted as part of the Westinghouse emergency core cooling system (ECCS) evaluation model only; does not constitute acceptance for any purpose other than for ECCS analyses.

B - Submitted to UNRC as background information; not undergoing formal USNRC review.

O - On file with USNRC; older generation report with current validity; not actively under formal USNRC review.

U - Actively under formal USNRC review.

TABLE 1.6-1
MATERIAL INCORPORATED BY REFERENCE

Westinghouse Topical Report No.	Title	Revision Number	SAR Section Reference	Submitted to the NRC	Review Status
WCAP-2048	The Doppler Effect for a Non-Uniform Temperature Distribution in Reactor Fuel Elements	Rev 0	4.3	7/62	0
WCAP-2850-L(P) WCAP-7916	Single-Phase Local Boiling and Bulk Boiling Pressure Drop Correlations	Rev 0	4.4	5/66	0
WCAP-2923	In-Pile Measurement of UO_2 Thermal Conductivity	Rev 0	4.4	3/66	0
WCAP-3269-8	Hydraulic Tests of the San Onofre Reactor Model	Rev 0	4.4	6/64	0
WCAP-3269-26	LEOPARD - A Spectrum Dependent Non-Spatial Depletion Code for the IBM-7094	Rev 0	4.3, 15.4	9/63	0
WCAP-3385-56	Saxton Core II Fuel Performance Evaluation	Rev 0	4.3	7/70	0
	WCAP-3385-56, Part II, Evaluation of Mass Spectrometric and Radiochemical Materials Saxton Plutonium Fuel				
WCAP-3680-20	Xenon-Induced Spatial Instabilities in Large Pressurized Water Reactors (EURAEC-1974)	Rev 0	4.3	3/68	0
WCAP-3680-21	Control Procedures for Xenon-Induced X-Y Instabilities in Large Pressurized Water Reactors (EURAEC-2111)	Rev 0	4.3	2/69	0
WCAP-3680-22	Xenon-Induced Spatial Instabilities in Three Dimensions (EURAEC-2116)	Rev 0	4.3	9/69	0

TABLE 1.6-1 (cont)
MATERIAL INCORPORATED BY REFERENCE

Westinghouse Topical Report No.	Title	Revision Number	SAR Section Reference	Submitted to the NRC	Review Status
WCAP-3966-B	Pressurized Water Reactor P_H Reactivity Effect Final Report (EURAEC-2074)	Rev 0	4.3	10/68	0
WCAP-3726-1	PUO_2 -UO_2 Fueled Critical Experiments	Rev 0	4.3	7/67	0
WCAP-6065	Melting Point of Irradiated UO_2	Rev 0	4.4	2/65	0
WCAP-6069	Burnup Physics of Heterogeneous Reactor Lattices	Rev 0	4.4	6/65	0
WCAP-6073	LASER - Depletion Program for Lattice Calculations Based on MUFT and THERMOS	Rev 0	4.3	4/66	0
WCAP-6086	Supplementary Report on Evaluation of Mass Spectrometric and Radiochemical Analyses of Yankee Core I Spent Fuel, Including Isotopes of Elements Thorium through Curium	Rev 0	4.3	8/69	0
WCAP-7015	Subchannel Thermal Analysis of Rod Bundle Cores	Rev 1	4.4	2/14/69	0
WCAP-7048-P-A(P) WCAP-7757-A	PANDA Code	Rev 0	4.3	1/9/75	A
WCAP-7208(P) WCAP-7811	Power Distribution Control of Westinghouse Pressurized Water Reactors	Rev 0	4.3	9/68	0
WCAP-7213-P-A(P) WCAP-7758-A	TURTLE 24.0 Diffusion Depletion Code	Rev 0	4.3	1/9/75	A

TABLE 1.6-1 (cont)
MATERIAL INCORPORATED BY REFERENCE

Westinghouse Topical Report No.	Title	Revision Number	SAR Section Reference	Submitted to the NRC	Review Status
WCAP-7267-L(P) WCAP-7809	Core Power Capability in Westinghouse PWRs	Rev 0	4.3	10/69	O
WCAP-7308-L(P) WCAP-7810	Evaluation of Nuclear Hot Channel Factor Uncertainties	Rev 0	4.3	1/9/70 12/16/71	U
WCAP-7359-L(P) WCAP-7838	Application of THINC Program to PWR Design	Rev 0	4.4	9/8/69 1/17/72	O
WCAP-7588	Evaluation of Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods	Rev 1A	15.4	1/7/75	A
WCAP-7667-P-A(P) WCAP-7755-A	Interchannel Thermal Mixing With Mixing Vane Grids	Rev 0	4.4	1/27/75	A
WCAP-7695-P-A(P) WCAP-7958-A	DNB Tests Results for New Mixing Vane Grids (R)	Rev 0	4.4	1/21/75	A
WCAP-7706-L(P) WCAP-7706	An Evaluation of Solid State Logic Reactor Protection in Anticipated Transients	Rev 0	4.6	9/2/71	O
WCAP-7800	Nuclear Fuel Division Quality Assurance Program Plan	Rev 5A	4.2	11/20/79	A
WCAP-7907-P-A	LOFTRAN Code Description	Rev 0	15.0, 15.4	10/11/72	A
WCAP-7908	FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod	Rev 0	15.0, 15.4	9/20/72	U
WCAP-7912-P-A(P) WCAP-7912-A	Power Peaking Factors	Rev 0	4.3	1/16/75	A

TABLE 1.6-1 (cont)
MATERIAL INCORPORATED BY REFERENCE

Westinghouse Topical Report No.	Title	Revision Number	SAR Section Reference	Submitted to the NRC	Review Status
WCAP-7941-P-A(P) WCAP-7959-A	Effect of Axial Spacing On Interchannel Thermal Mixing with the R Mixing Vane Grid	Rev 0	4.4	1/27/75	A
WCAP-7956	THINC-IV - An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores	Rev 0	4.4	10/22/73	A
WCAP-7964	Axial Xenon Transient Tests at Rochester Gas and Electric Reactor	Rev 0	4.3	6/15/71	O
WCAP-7979-P-A(P) WCAP-8028-A	TWINKLE - A Multidimensional Neutron Kinetics Computer Code	Rev 0	15.0, 15.4	1/7/75	A
WCAP-7988-P-A(P) WCAP-8030-A	Application of Modified Spacer Factor to L-Grid Typical and Cold Wall Cell DNB	Rev 0	4.4	1/75	A
WCAP-8054(P) WCAP-8195	Application of THINC-IV Program to PWR Design	Rev 0	4.4	12/7/73 1/11/74	A
WCAP-8174 P-A(P) WCAP-8202-A	Effect of Local Heat Flux on DNB in Nonuniformly Heated Rod Bundles	Rev 0	4.4	2/75	A
WCAP-8183	Operational Experience with Westinghouse Cores (up to December 31, 1982)	Rev 12	4.2	8/83	B
WCAP-8218 P-A(P) WCAP-8219-A	Fuel Densification Experimental Results and Model for Reactor Application	Rev 0	4.2, 4.3, 4.4	3/6/75	A
WCAP-8298-P-A(P) WCAP-8299-A	The Effect of 17 x 17 Fuel Assembly Geometry on Interchannel Thermal Mixing	Rev 0	4.4	1/28/75	A
WCAP-8305	LOCTA-IV Program: Loss of Coolant Transient Analysis	Rev 0	15.0	6/74	AE

TABLE 1.6-1 (cont)
MATERIAL INCORPORATED BY REFERENCE

Westinghouse Topical Report No.	Title	Revision Number	SAR Section Reference	Submitted to the NRC	Review Status
WCAP-8306	SATAN-VI Program: Comprehensive Space-Time Dependent Analysis of Loss-of-Coolant	Rev 0	15.0	7/12/74	AE
WCAP-8330	Westinghouse Anticipated Transients Without Trip Analysis	Rev 0	4.3, 4.6, 15.4	9/25/74	U
WCAP-8359	Effects of Fuel Densification Power Spikes on Clad Thermal Transients	Rev 0	4.3	7/2/74	AE
WCAP-8370	Westinghouse Water Reactor Divisions Quality Assurance Plan	Rev 9A	17	11/14/77	A
WCAP-8377(P) WCAP-8381	Revised Clad Flattening Model	Rev 0	4.2	8/7/74 8/6/74	A
WCAP-8385(P) WCAP-8403	Power Distribution Control and Load Following Procedures	Rev 0	4.3	10/9/74	A
WCAP-8453-A(P) WCAP-8454	Analysis of Data from Zion (Unit 1) THINC Verification Test	Rev 0	4.4	5/10/76	A
WCAP-8498	Incore Power Distribution Determination in Westinghouse Pressurized Water Reactors, Program Summaries - Fall 1974	Rev 0	4.3	7/22/75	U
WCAP-8567-P(P) WCAP-8568	Improved Thermal Design Procedure	Rev 0	4.4, 15.0	7/75	A
WCAP-8575(P) WCAP-8576	Augmented Startup and Cycle 1 Physics Program Supplement 1	Rev 0	4.3	6/76	U
WCAP-8584(P) WCAP-8760	Failure Mode and Effects Analysis (FMEA) of Engineered Safeguard Features Actuation System	Rev 0 Rev 1	4.6	4/23/76 2/80	U

TABLE 1.6-1 (cont)
MATERIAL INCORPORATED BY REFERENCE

Westinghouse Topical Report No.	Title	Revision Number	SAR Section Reference	Submitted to the NRC	Review Status
WCAP-8682(P) WCAP-8683	Experimental Verification of Wet Fuel Storage Criticality Analyses	Rev 0	4.3	3/18/76	B
WCAP-8691(P) WCAP-8692	Fuel Rod Bowing	Rev 1	4.2, 4.4	7/79	A
WCAP-8708-P-A (P), Volumes I and II WCAP-8709-A, Volumes I & II	MULTIFLEX - FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic Structure System Dynamics	Rev 0	3.9	9/16/77	A
WCAP-8720(P) WCAP-8785	Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations	Rev 0	4.2	11/2/76	A
WCAP-8762(P) WCAP-8763	New Westinghouse Correlation WRB-1 for Predicting Critical Heat Flux in Rod Bundles with Mixing Vane Grids	Rev 0	4.4	7/76	A
WCAP-8768	Safety-Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries - Winter 1977 through Summer 1978	Rev 2	4.2, 4.3	9/28/78	B
WCAP-8768	Safety-Related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries - Spring 1976	Rev 0	4.3	6/17/76	B
WCAP-8846-A	Hybrid B_4C Absorber Control Rod Evaluation Report	Rev 0	4.2, 15.0	10/77	A
WCAP-8963(P) WCAP-8964	Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis	Rev 0	4.2	3/31/77 8/11/77	A

TABLE 1.6-1 (cont)
MATERIAL INCORPORATED BY REFERENCE

Westinghouse Topical Report No.	Title	Revision Number	SAR Section Reference	Submitted to the NRC	Review Status
WCAP-8976	Failure Mode and Effects Analysis (FMEA) of Solid State Full-Length Rod Control System	Rev 0	4.6	10/26/77	U
WCAP-9000-L(P)	Nuclear Design of Westinghouse Pressurized Water Reactors with Burnable Poison Rods	Rev 1	4.3	7/69	O
WCAP-9004(P) WCAP-7836	Inlet Orificing of Open PWR Cores	Rev 0	4.4	1/17/72	B
WCAP-9105(P)	Axial Power Distribution Monitoring Using Four-Section Ex-Core Detectors	Rev 0	4.3	7/77	U
WCAP-9179(P) WCAP-9224	Properties of Fuel and Core Component Materials	Rev 1	4.2	8/2/78	A
WCAP-9485(P) WCAP-9486	Paladon - Westinghouse Nodal Computer Code	Rev 0	4.3	12/78	A
WCAP-10444(P)	Westinghouse Reference Core Report - VANTAGE 5 Fuel Assembly	Rev 0	4.2, 4.3, 4.4	12/83	U

1.8 CONFORMANCE WITH THE STANDARD REVIEW PLAN

In accordance with 10CFR50.34(g), Table 1.8-1 identifies and evaluates deviations from the acceptance criteria of those sections of the NRC Standard Review Plan (NUREG-0800) pertinent to the Reactor System. In addition a listing of NRC Division 1 Regulatory Guides pertinent to the WAPWR Reactor System, and a discussion of the extent to which the WAPWR complies with the regulatory positions of these Regulatory Guides is given in Table 1.8-2.

TABLE 1.8-1

STANDARD REVIEW PLAN DEVIATIONS

SRP Acceptance Criteria	Deviation	Section

(To date, no deviations from the acceptance criteria of those SRP sections applicable to the WAPWR Reactor System have been identified).

TABLE 1.8-2

CONFORMANCE TO US NRC REGULATORY GUIDES
APPLICABLE TO THE WAPWR REACTOR SYSTEM

REGULATORY GUIDE 1.13, REVISION 1, DECEMBER 1975,
SPENT-FUEL STORAGE FACILITY DESIGN BASIS

Westinghouse conforms to the regulatory position of this regulatory guide.

REGULATORY GUIDE 1.20, REVISION 2, MAY 1976,
COMPREHENSIVE VIBRATION ASSESSMENT PROGRAM FOR
REACTOR INTERNALS DURING PREOPERATIONAL AND
INITIAL STARTUP TESTING

The comprehensive vibration assessment program for the WAPWR internals during preoperational and initial startup testing will conform with the recommendations of this guide.

REGULATORY GUIDE 1.29, REVISION 3, SEPTEMBER 1978,
SEISMIC DESIGN CLASSIFICATION

The WAPWR conforms with this regulatory guide as shown in Table 3.2-1 of RESAR-SP/90 PDA Module 7, "Structural/Equipment Design".

With regard to regulatory position C.1, each nuclear steam supply system (NSSS) component important to safety is classified as Safety Class 1, 2, or 3; these classes are qualified to remain functional in the event of the SSE, except where exempted by meeting all of the below requirements. Portions of systems required to perform the same safety function as required of a safety class component which is part of that system shall be likewise qualified or granted exemption. Conditions to be met for exemption are:

- Failure would not directly cause a Condition III or IV event (as defined in ANSI N18.2-1973).

TABLE 1.8-2 (Continued)

- o There is no safety function to mitigate, nor could failure prevent mitigation of, the consequence of a Condition III or IV event.

- o Failure during or following any Condition IV event would result in consequences no more severe than allowed for a Condition III event.

- o Routine post-seismic procedures would disclose loss of the safety function.

REGULATORY GUIDE 1.65, OCTOBER 1973, MATERIALS AND INSPECTIONS FOR REACTOR VESSEL CLOSURE STUDS

The WAPWR conforms with this guide except for two points. The use of modified SA-540 Grade 824 material as specified in ASME Boiler and Pressure Vessel Code Case 1605 is not specified in the guide but is used by Westinghouse. The use of this Code Case has been approved by the NRC via Regulatory Guide 1.85.

The maximum limit of 170 ksi ultimate tensile strength is not explicitly specified by Westinghouse as required by the guide. Westinghouse does specify fracture toughness of 45 ft/lb and 25 mils lateral expansion as required by the ASME Code and 10 CFR 50, Appendix G. These requirements also result in strength levels below the maximum limit, as demonstrated by the actual stud material properties for the WAPWR.

REGULATORY GUIDE 1.70, REVISION 3, NOVEMBER 1978, STANDARD FORMAT AND CONTENT OF SAFETY ANALYSIS REPORTS FOR NUCLEAR POWER PLANTS

The format and content of the RESAR-SP/90 PDA Modules meet the intent of this regulatory guide.

TABLE 1.8-2 (Continued)

REGULATORY GUIDE 1.77, MAY 1974, ASSUMPTIONS USED FOR EVALUATING A CONTROL ROD EJECTION ACCIDENT FOR PRESSURIZED WATER REACTORS

Westinghouse conforms as discussed in Subsection 15.4.8.

REGULATORY GUIDE 1.99, REVISION 1, APRIL 1977, EFFECTS OF RESIDUAL ELEMENTS ON PREDICTED RADIATION DAMAGE TO REACTOR VESSEL MATERIALS

There are two primary issues with the guide:

1. The guide provides a procedure and curves for predicting radiation damage (as relating to the shift of the reference temperature, RT_{NDT}), in terms of chemistry (Cu and P) and fluence. This guide's procedure differs significantly from the one used by the WAPWR.

   Since the adjustments in reference temperature obtained from the radiation damage curves are used in developing heatup and cooldown limits for plant operation, the use of the curves in the guide could result in over-conservative heatup and cooldown limits during plant life.

2. The guide restricts the end of life transition temperature to 200°F maximum. Control of residual elements such as copper, phosphorus, sulfur, and vanadium in the reactor vessel beltline materials of new plants to levels that result in a predicted adjusted reference temperature of less than 200°F at end of life is considered technically unnecessary and could lead to unnecessary changes in chemistry (Cu and P) requirements with corresponding adverse impact on cost and materials availability.

TABLE 1.8-2 (Continued)

One additional feature of the guide constitutes a lesser but nevertheless important issue:

1. Figure 2 of the guide presents a curve which gives the decrease of upper shelf impact energy with fluence as a function of Cu content. Although it appears that the prescribed relationship does not predict unacceptable drops in upper shelf toughness for vessels with controlled chemistry the curves are nevertheless over conservative.

The WAPWR position with respect to each of the guide positions is as follows:

1. The basis as well as the scope of the guide for predicting adjustment of reference temperature as given in regulatory position C.1 are inappropriate since the data base used was incomplete and included some data which were not applicable.

2. The WAPWR design is consistent with the guide position C.2a. However, with respect to guide position C.2b, Westinghouse believes that Figure 2 of the guide is incorrect since the upper shelf energy for 6-in.-thick American Society of Testing Materials (ASTM) A302B reference correlation monitor material reported by Hawthorne indicates essentially a constant upper shelf at fluences above ~ 1 x 10^{19} n/cm^2.[(a)]

3. The Westinghouse position with reference to the guide position C.3, controlling residual elements to levels that result in a predicted adjusted reference temperature of less than 200°F at end of life, is that the stresses in the vessel can be limited during operation in order to comply with the requirements of Appendix G to 10 CFR 50 even though the end of life adjusted reference temperature may exceed 200°F. By applying the procedures of Appendix G to ASME Section III, the stress limits including appropriate Code safety margin can be met.

TABLE 1.8-2 (Continued)

4. Recent surveillance capsule data indicate a steady-state condition of radiation damage well below that predicted by current trend curves.[a] This effect is believed to be due to the annealing of the vessels at the operating temperature. As an alternative to Regulatory Guide 1.99, operating limits will be determined using the current radiation damage curves developed by Westinghouse.[b] It is expected that as future surveillance specimens are evaluated it will become increasingly evident that both the Regulatory Guide 1.99 and Westinghouse trend curves are very conservative.

REGULATORY GUIDE 1.126, REVISION 1, MARCH 1978, AN ACCEPTABLE MODEL AND RELATED STATISTICAL METHODS FOR THE ANALYSIS OF FUEL DENSIFICATION

This guide states that the model presented in this guide is not intended to supersede NRC approved vendor models. The WAPWR uses the Westinghouse model which has been approved by the NRC. Refer to Subsection 4.2.3.2 for further discussion.

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a. Hawthorne, J. R., "Radiation Effects Information Generated on the ASTM Reference. Correlation-Monitor Steels," ASTM, Philadelphia, 1974.