WESTINGHOUSE CLASS 3

AMENDMENT 4 TO RESAR-SP/90 PDA MODULE 5
REACTOR SYSTEM

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AMENDMENT 4 TO RESAR-SP/90 PDA MODULE 5
REACTOR SYSTEM

INSTRUCTION SHEET

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Replace current page 1.6-6 with revised page 1.6-6.

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Replace current page 17.0-1 with revised page 17.0-1.

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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	7/9/70 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 Business Unit Quality Assurance Program Plan	Rev 7	4.2	10/6/88	U
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-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 Energy Systems Business Unit Quality Assurance Plan	Rev 11	17	10/6/88	U 4
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

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

Certain structures, components, and systems of the reactor system are important to safety because they:

a. Assure the integrity of the reactor coolant pressure boundary.

b. Assure the capability to shut down the reactor and maintain it in a safe condition.

c. Assure the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of 10CFR 100.

d. Contain or may contain radioactive material.

The purpose of this section is to classify structures, systems, and components according to the importance of the item in order to provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. Table 3.2-1 of RESAR-SP/90 PDA Module 7, "Structural/Equipment Design", delineates each of the items in the plant which 4
fall under the above-mentioned categories and the respective associated classification that the NRC, ANS and industrial codes committees have developed. Each of the classification categories in Table 3.2-1 is addressed in the following sections.

The classification of specific piping runs and valves in these runs is provided in the system flow diagrams contained in this module. Instrumentation and electrical equipment required to shutdown the plant or mitigate an accident which is associated with the reactor system will be classified as IE (or Safety Class 3 per ANS 51.1) and identified in the appropriate module.

4

3.2.1 Seismic Classification

Seismic classification criteria are set forth in 10CFR 100 and supplemented by Regulatory Guide 1.29.

All components classified as Safety Class 1, 2, or 3 (classifications are as defined by Reference 1), are seismic Category 1.

Seismic Category I structures, components, and systems are designed to withstand the Safe Shutdown Earthquake (SSE) and other applicable load combinations, as discussed in RESAR-SP/90 PDA Module 7, "Structural/Equipment Design". Seismic Category I structures are sufficiently isolated or protected from the other structures to ensure that their integrity is maintained.

3.2.2 System Quality Group Classification

4 The Quality Assurance Program described in Subsection 17.1 is applied to all Safety Class 1, Safety Class 2 and Safety Class 3 structures, systems and components.

The components are classified according to their importance to safety, as dictated by service and functional requirements and by the consequences of their failure. The quality assurance requirements and code requirements for the reactor system meet the intent of Regulatory Guide 1.26.

3.2.3 Safety Classes

Table 3.2-1 lists the safety class assigned to applicable systems and components in accordance with ANS 51.1 (Reference 1). The criteria (of Reference 1) are used in the plant design to provide an added degree of assurance that the plant is designed, constructed, and operated without undue risk to the health and safety of the public.

3.2.4 References

1. "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants", ANS-51.1, November 1983.

TABLE 3.2-1

CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS FOR THE REACTOR SYSTEM

System/Component	Location	Quality Group	Safety Class	Code Classification	Principal Construction Codes & Stds	Seismic Category	Quality Assurance
Reactor Vessel							
Internals							
Integrated Head Package							
Cooling Shroud							
Missile Shield							
Lift Rod Assembly							
Lift Rig Assembly				(See Table 3.2-1 of RESAR-SP/90 PDA Module 7, "Structure/Equipment Design")			
Cable Bridge Assembly							
Cable Assembly							
Seismic Support System							
CRDM Control Rods							
Gray Rods							
Housing							
Displacer Rod Drive Mechanism							
Housing							
Reactor Core							
Fuel assembly							
Water displacer							
Rod assembly							
Gray rod assembly							
Control rod assembly							

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3.9 MECHANICAL SYSTEMS AND COMPONENTS

3.9.2 Dynamic Testing and Analysis

3.9.2.3 Dynamic Response Analysis of Reactor Internals Under Operational Flow Transients and Steady-State Conditions

Flow Induced Vibration (FIV) and resulting wear have been recognized throughout the SP/90 design effort as important issues in reactor internals design. In fact, consideration of potentially damaging effects of FIV have had a significant influence on the basic reactor internals design as evidenced by the selection of an upper calandria configuration. The following sections outline efforts performed in specific areas.

- ACSTIC Evaluations

  ACSTIC is a Westinghouse Proprietary mainframe computer code written in FORTRAN which is used to determine the characteristics of the standing waves and wave propagation in the primary system reactor coolant loop. Given a model of nodes and flow paths, the coolant natural frequencies can be determined. The loop response at the characteristic reactor coolant pump excitation frequencies can also be determined. Fluctuating pressure gradients across structural components can then be estimated. An evaluation of various fluctuating pressure effects on the reactor internals can then be made. 4

  An ACSTIC analysis performed for the SP/90 reactor coolant system indicates the following primary frequencies:

  $f_1 = 7$ Hz
  $f_2 = 10$ Hz
  $f_3 = 13$ Hz
  $f_4 = 18$ Hz
  $f_5 = 23$ Hz

These frequencies are disparate from the expected pump shaft rotational frequencies, and consequently the pump rotational induced response in the reactor coolant loop is relatively small.

o Upper Core Plate Axial Vibration

The upper core plate axial vibratory response to turbulence excitation has been analyzed. A reasonable pressure fluctuation was determined from a survey of available data and the response calculated using structural parameters developed in a finite element analysis. The resulting calculated peak fluctuating displacement has been determined to be acceptable from a fatigue and wear viewpoint.

o Lower Core Plate Axial Vibration

The lower core plate axial vibratory response to turbulence excitation
4 has been analyzed in a similar manner. A reasonable pressure fluctuation was determined from a survey of available data and the response calculated using structural parameters developed in a finite element analysis. The resulting calculated peak fluctuating displacement has been determined to be acceptable from a fatigue and fuel assembly lift-off viewpoint.

o Core Barrel Vibration

The important core barrel vibrations consist of the cantilever beam mode and the lower numbered shell modes. These have been traditionally of interest in the internals design. The SP/90 internals design also incorporates an inner barrel that is in close proximity to the inside of the core barrel for the upper two thirds of the core barrel length. This configuration introduces some important barrel interactions that are not present in past Westinghouse designs.

The core barrel cantilever mode has been investigated using a special simplified configuration of the Reactor Equipment System Model (RESM) finite element model. Normal operation and hot functional test lateral system modes have been calculated for various support conditions at the upper core plate pin interface between the inner barrel and the core barrel and the lower radial support interface between the core barrel and the vessel.

The lower radial support normal operation vibratory loads were determined using the above RESM model and an appropriate downcomer turbulence forcing function. The resulting peak vibratory loads on the lower radial supports are dependent upon the boundary conditions at the upper core plate pin and lower radial support interfaces. Finite element calculations were performed to investigate these effects.

The core barrel shell modes interact with the inner barrel shell modes through the hydraulic coupling in the gap between the barrels. This interaction adds substantial hydrodynamic mass to both the inner barrel and the core barrel. The amount of interaction or coupling for a particular configuration is dependent upon the mode numbers of the two cylinders. For the circumferential modes, only modes with the same number can couple. 4

For the axial modes, specific relationships involving cylinder lengths, axial mode number, and end boundary conditions must be met before complete coupling can occur. For modes that do not meet the specified conditions only partial coupling occurs. When the coupling is nearly zero, the cylinders vibrate independently. For that case, the effective hydrodynamic mass is equivalent to that resulting when the other cylinder is rigid.

For the case of the partial coupling, one of the coupled frequencies is lower than either of the above adjacent rigid wall frequencies, and the other coupled frequency is higher than either of the above adjacent rigid wall frequencies. For the SP/90, the lower mode shell frequencies then fall in the 15 Hz to 25 Hz range.

o Rod Guide Vibration

The reactivity control cluster (RCC) and water displacer rod cluster (WDRC) rod guide designs are arranged differently from existing Westinghouse PWR designs. The SP/90 is a close packed arrangement with a relatively small, 0.24 inch, nominal gap between adjacent rod guide enclosure walls. This arrangement introduces considerable hydrodynamic lateral coupling between the vibratory responses of the rod guides.

An analytical fluid/structure interaction study was performed on a subset of the rod guide region. These studies indicated that the water coupled natural frequencies of the rod guides are considerably lower than the in-air natural frequencies. Also, complex interactions in the horizontal plane exist between the rod guides which result in
4 many system frequencies. The in-air natural frequency of both the RCC and the WDRC rod guide is approximately 36 Hz. The coupled system frequencies are as low as 7.5 Hz.

The fluid/structure interaction studies have been used to develop WDR and RCC rod guide models for use in the RESM system model.

The RCC and WDRC rod guides are bolted to the core plate at the bottom. The upper lateral support for these guides has been an area of extensive investigation. An important parameter affecting the upper support design is the flow induced vibratory reaction load. Calculations have been performed which address the expected FIV reaction loads. Considerable uncertainty exists concerning the exact excitation mechanism of previously measured internal responses. Consequently, relatively conservative assumptions were made in performing the above cited calculations. This resulted in calculated reaction loads that are conservatively high, but which have been shown to be acceptable for the design of the rod guide top end support.

o RCC Vibration and Wear

Significant efforts were expended to demonstrate a RCC wear life of 20 years.

RCC rodlet motions as a result of flow induced vibration were measured in a full scale RCC Axial Flow Test facility. This facility incorporated the capability to adjust the intermediate support locations in the RCC rod guide to achieve the maximum RCC rodlet motion. Separately, impact-fretting wear coefficients were measured under simulated reactor conditions of pressure, temperature, and chemistry. With these data, it was possible to calculate wear as a result of flow induced vibration.

Sliding wear as a result of CRDM stepping was measured in the Westinghouse D-loop facility which contained a complete full-scope RCC (Reactivity Control Cluster) driveline and associated equipment. The driveline included the RCC itself, drive rod and couplings, and the CRDM. Associated equipment included the pressure vessel, simulated lower internals with a prototype fuel assembly, and simulated upper internals with a prototypic RCC rod guide and an upper calandria mock-up. The tests were conducted at reactor conditions of pressure, temperature and chemistry. The CRDM was exercised for 7.8 million steps in a pattern which included simulated baseload operation, load follow, frequency control and rod drops. 4

Combining the flow induced vibration and stepping wear led to the conclusion that the RCC's would be able to operate well in excess of 20 years without clad wear through.

o Scale Model Testing

In addition to the above described analytical activities, a one-fifth scale test of the full SP/90 reactor was performed. This was a low pressure, low temperature test which was executed in 198\ and 1987 at the Takasago Research and Development Center of Mitsubishi Heavy Industries in Japan.

The test incorporated three phases as follows:

(i) Vibration characteristics test (natural frequencies, vibration modes and damping ratios) in air and in water.

(ii) Pressure loss test.

(iii) Flow induced vibration test.

Main conclusions from the latter phase of testing are:

4 (i) The bottom mounted instrumentation, calandria tubes, and flow shrouds were the only components that showed vibration near their natural frequencies; however, the level of vibration was low.

(ii) The response of the other reactor internals showed random vibration in a frequency range of 0-1000 Hz, and neither structural resonance nor unstable vibration (e.g. of the hydroelastic type) was observed.

In general, acceleration and stress levels were significantly below values that would cause fatigue concerns.

3.9.2.4 Preoperational Flow-Induced Vibration Testing of Reactor Internals

Every SP/90 plant to be built will undergo a hot functional testing program. A part of that program is devoted to assuring the structural integrity of the reactor internals by the successful completion of a full flow, high temperature, high pressure test. Additionally, the first SP/90 plant to become operational will also be classified as a "prototype" according to Regulatory Guide 1.20 of the United States Nuclear Regulatory Commission. Thus, that plant will also need to be instrumented for the monitoring of structural vibration responses during preoperational testing.

A scoping study addressing this general issue has been undertaken as a part of the SP/90 reactor internals design process. The main conclusion of the study is that preoperational vibrational assessment testing of the present reactor internals design is feasible with only detailed hardware changes needed for transducer mounting, transducer protection, and transducer lead routing. Consideration was also given to the possible need to perform the preoperational testing with a dummy core in place. The conclusion of the study is that an adequate preoperational vibrational assessment test program could be conducted without the presence of a dummy core. 4

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3.9.2.5 Dynamic System Analysis of the Reactor Internals Under Faulted Conditions

Analyses of the reactor internals for loads resulting from postulated pipe breaks(*) which result in loss-of-coolant accidents (LOCAs) are typically based on the time-history response of the reactor internals to hydraulic forcing functions applied simultaneously. The forcing functions are defined at points in the system where changes in cross-section or direction of flow may occur such that differential loads may be generated as a consequence of the pipe break(s). Because of the complexity of the system and the components, it may be necessary to use finite element stress analysis codes to provide more detailed information at various points.

(*) See RESAR-SP/90 PDA Module 7, "Structural/Equipment Design" for the detailed application of Westinghouse revised pipe break criteria to the WAPWR design.

17.0 QUALITY ASSURANCE

17.1 Quality Assurance During Design and Construction

The Westinghouse Energy Systems Business Unit/Nuclear Fuel Business Unit Quality Assurance Program is described in Reference 1.

4

17.1.1 References

1. "Westinghouse Energy Systems Business Unit/Nuclear Fuel Business Unit Quality Assurance Plan," WCAP-8370, Revision 11, October 1988.

REQUEST FOR ADDITIONAL INFORMATION
WESTINGHOUSE ADVANCED PRESSURIZED WATER REACTOR (RESAR-SP/90)
DOCKET NO. 50-601

The following Questions/Responses were formally transmitted in Addendum 1 to RESAR-SP/90 PDA in Westinghouse letter NS-NRC-88-3304, dated January 7, 1988.

252.1 Verify that the aging and tempering temperatures of heat treatable materials used in the control rod drive mechanisms are specified to eliminate susceptibility to stress corrosion cracking in reactor coolant (4.5.1, Module 5)

Response:

The CRDM heat treatable materials are 410 SST tubing for the drive rod, 403 modified SST bar for the coupling, and Inconel X-750 spring wire. The 410 SST and 403 SST are tempered to provide minimum yield strengths of 80 KSI and 90 KSI respectively. This tempering is well below the threshold where susceptibility to stress corrosion cracking (SCC) becomes a concern - yields greater than 120 KSI. The Inconel X-750 is manufactured to MIL-S-23192 which offers the most favorable conditions for precluding SCC.

252.2 What materials, other than austenitic stainless steels of limited coldwork (maximum yield strength of 90 ksi) are used for reactor internals? (4.5.2, Module 5)

Response:

Other materials used for reactor internals are: 1) Inconel X-750, for guide tube support pins and guide tube flexures (where applicable) and; 2) Stellite Hardfacing, for the radial support keys. Stellite Hardfacing is principally composed of Cobalt (Co) Chromium (Cr) and Tungsten (W).

The following Questions/Responses were formally transmitted in Addendum 5 to RESAR-SP/90 PDA in Westinghouse letter NS-NRC-88-3338, dated May 13, 1988.

440.2 How is the Improved Thermal Design Procedure (ITDP) factored in the 2% power as well as the allowances on pressure and temperature?

RESPONSE:

The Improved Thermal Design Procedure (ITDP) was used for most DNB related transients. Consistent with the methodology presented in WCAP-8567, Reference 3 in Section 4.4 of RESAR-SP/90 PDA Module 5, "Reactor System," allowances for power, pressure, temperature and flow are included. These uncertainties were calculated specifically for the APWR design.

The following Questions/Responses were formally transmitted in Addendum 6 to RESAR-SP/90 PDA in Westinghouse letter NS-NRC-88-3354, dated July 7, 1988.

210.29 The staff's comments in Q210.25 also apply to portions of Section 3.2 and Table 3.2-1 of Module 5. These sections should be revised to agree with the response to Q210.35.

RESPONSE:

Please refer to our original response to Staff Q210.1. Westinghouse believes that the initiative taken to design the SP/90 plant to the latest industry codes and standards, including ANSI/ANS 51.1, provides additional assurance that this plant design will operate more safety and with better reliability than current nuclear power plant designs. If this issue is not settled prior to final design submittal, Westinghouse will reexamine the manner in which Safety classifications are assigned for systems, components, and structures for the SP/90 plant.

210.30 Section 1.5.1.3 of Module 5 briefly discusses proposed tests of the Control Rod Drive Mechanism (CRDM) and the Water Displacer Rod Mechanism. These tests were scheduled to be conducted in 1985 and 1986. Provide a detailed description of the test program and a summary of the results. If applicable, provide a comparison of tests which were conducted on existing Westinghouse CRDM's with the WAPWR CRDMS's.

RESPONSE:

Following is additional information on Control Rod Drive Mechanism (CRDM) and Water Displacer Rod Drive Mechanism (DRDM) testing.

CRDM

The CRDM test program was performed at the D-Loop test facility at the Westinghouse Forest Hills site. Previous CRDM testing has also taken place in this facility.

D-Loop is a high temperature, high flow rate test facility which can test full size components under simulated conditions of chemistry, temperature, pressure and flow. The test section is isothermal; no heat is generated and only depleted nuclear fuel is used. The loop piping is designed for flow rates up to 4500 gpm, a maximum operating pressure of 2000 psig, and a temperature of 600°F. The loop flow rate is measured using a square-edge orifice plate and is adjustable by means of flow control valves in the main loop piping. The flow through the model is measured using four venturi flow meters built into the lower core plate. At 4000 gpm, the canned motor pump is capable of developing a head of 300 feet of water. In actual practice, the maximum loop flow attainable with the APWR model was approximately 2800 gpm.

All piping in the primary loop is Type 304 and Type 316 stainless steel. Loop pressure is automatically controlled by a constantly operating makeup and letdown system. Makeup is maintained by 3 Aldrich positive displacement pumps. Letdown is controlled by Grove Mitymite regulators. A rupture disk, set to relieve at 2400 psia, is provided for overpressure protection.

Loop temperature during steady-state operation is maintained by controlling bypass flow through the loop coolers. A total of 136 KW of heat input is available for startup and for steady-state operation through strip heaters mounted on the loop piping. The heaters are controlled by monitoring loop piping thermocouples and automatically regulating cycle timing of the respective heaters. An additional 235 KW is obtained from the main circulation pump at high flow rates. A Pan-Alarm system provides audible and visual indication of potential system malfunctions.

The D-Loop Facility also includes a cooling air system for the Control Rod Drive Mechanism (CRDM). The CRDM is an electro-mechanically operated device which relies on forced air cooling to maintain the magnet coils at a safe operating temperature. The cooling system consists of a 1000 cfm centrifugal blower, a throttling damper, an airflow measurement section, and a full length cooling baffle. Thermocouples were installed to permit calculation of total heat rejection.

The D-Loop test facility contained a complete full-scale RCC (Reactivity Control Cluster) driveline and associated equipment. The driveline included the RCC itself, drive rod and couplings, and the CRDM. Associated equipment included the pressure vessel, simulated lower internals with a prototype fuel assembly, and simulated upper internals with a prototypic RCC rod guide and an upper calandria mock-up. An additional vessel spool piece was required to accommodate the extra upper internals length caused by the calandria.

The CRDM was exercised for 7.8 million steps in a pattern which included simulated baseload operation, load follow, frequency control and rod drops.

Throughout the test period of approximately 6 months, the CRDM operated without problems. Measured heat rejection was similar to that found on previous D-Loop tests. Post-test inspection revealed that from a wear point of view, the latch arms could probably have operated up to approximately 10 million steps.

As noted previously, CRDM's have been tested in D-Loop in the past. However, maximum stepping duty in previous testing has not exceeded 3.5 to 4.0 million steps, primary because load follow operation and frequency control were not considered.

DRDM

The DRDM prototype hydraulic test program was performed at the M-Loop Test Facility at Westinghouse Electro-Mechanical Division located in Cheswick, PA. This loop has traditionally served as the production test facility for Control Rod Drive Mechanism and was modified to provide the adequate pressure control and make-up water capacity required by the DRDM in order to operate.

The loop pressure source is a 120-gallon pressurizer with a maximum available heat input of 80 KW. Pressure control settings are variable up to 2500 psig. The loop is capable of temperatures up to 650°F, by heat input as provided by electrical resistance strip heaters rated at 103.5 KW. A Westinghouse Model 150-D canned motor pump maintains loop water flow through the heaters. The high pressure makeup pump which maintains liquid level in the pressurizer during DRDM withdrawal, is a variable speed triplex plunger pump capable of flows to 9.2 GPM at 2500 psig. The control for this pump is from the pressurizer liquid level controller.

The facility is designed in accordance with the ASME Boiler & Pressure Vessel Code and is licensed to operate as a Pennsylvania Special Boiler (Pa. Spc. 3020) in the state of Pennsylvania.

The DRDM test unit consisted of five major subassemblies: the pressure housing, the internal cylinder, the drive rod, the weight assembly and the rod position indicator/cooling box.

1) The DRDM performed in an acceptable manner for the duration of the test. No withdrawal or insertion failures occurred.

2) No significant wear was found on the latch and spear; contact marks were visible, but no measurable material removal took place.

3) No significant wear was found on the cylinder with the ID bore unchanged. The ID surface showed light scratch marks which are considered normal for this application.

4) Piston ring wear of the radial wall thickness and the resulting changes in the ring end gap and spring tension were such that the rings are considered to be worn out. This was expected considering the duty imposed on these rings.

5) Addition of crud to the system did not result in noticeable degradation of DRDM operating performance.

6) Heat rejection from the DRDM was higher than calculated.

The DRDM was also tested in the Hot Single Channel Test #2 in Japan, which was funded by the Japanese government. The results of this test have not yet been made public. They are expected to be available at the FDA stage.

The DRDM is a first-of-a-kind component, therefore, no comparison to previous tests can be made.

210.31 In Section 3.9.2.4 of Module 5, it is stated that the recommendations of Regulatory Guide 1.20 will be satisfied by conducting examinations of the reactor internals both before and after confirmatory hot functional testing of the internals.

Based on the staff's understanding of the WAPWR reactor internals design, this is not an acceptable commitment. Section 3.9.5 of Module 5 describes a design which is "significantly different from existing Westinghouse designs." The prototype plant for existing Westinghouse four loop plants is Indian Point Unit 2. The data from the Indian Point 2 reactor internals verification test program has been supplemented by data from tests conducted at the Trojan and Sequoyah plants. This supplemental data was provided at the staff's request to verify that design changes to reactor internals in Westinghouse four loop plants subsequent to the Indian Point 2 design did not result in a significant difference from the Indian Point 2 verification data.

It is not apparent to the staff that the WAPWR reactor internals response to flow induced vibration is enveloped by the above prototype verification data for four loop plants. Therefore, the staff will require a commitment that the first WAPWR plant will be identified as the prototype plant and will meet all of the applicable Regulatory Guide 1.20 guidelines. Revise Section 3.9.2.4 in Module 5 to provide this commitment or provide justification for not doing so.

RESPONSE:

In response to this question, Subsection 3.9.2.4 has been modified.

210.32 As stated in Q210.31, the staff position is that the first WAPWR plant be designated as the prototype as defined in REG Guide 1.20. The information in Section 3.9.2.3 of Module 5 relative to the dynamic response analysis of reactor internals under operational flow transients and steady-state conditions is not acceptable for a prototype plant. Revise this section to be consistent with the guidelines in Standard Review Plan, Sections 3.9.2.I.3, 3.9.2.II.3 and 3.9.2.III.3.

RESPONSE:

In response to this question, Subsection 3.9.2.3 of RESAR-SP/90 PDA Module 5, "Reactor System" has been revised to reflect our commitment to meeting the staff guidelines as outlined in SRP Section 3.9.2.

210.33 Section 3.9.5.1.3.4. of the Module 5 discusses the bottom mounted instrumentation (BMI) thimbles. A problem of unacceptable accelerated wear of Westinghouse designed BMI thimbles in a European 14-foot core plant was identified in 1985. Subsequently, Westinghouse modified the thimble design to reduce flow velocity in the gap between the thimble and the BMI column. However, this modification did not resolve the problem, but instead increased the rate of wear. The same modified design has been incorporated into the South Texas, Unit 2 pressure vessel which also contains a 14-foot core. Since this potential problem could be applicable to the WAPWR, provide a commitment in Section 3.9.5.1.3.4 that the final resolution of this problem for South Texas and the European plants will be incorporated into the WAPWR design, or provide justification for not doing so. A failure of one or more of these thimbles could result in a small break in the reactor coolant pressure boundary which cannot be isolated.

RESPONSE:

The SP/90 and South Texas designs are different in the area of bottom mounted instrumentation (BMI) such that it may not be necessary to incorporate the final resolution of the current BMI problems into the APWR. Prior to FDA submittal the APWR BMI design will be reviewed to determine if design changes are warranted.