

Attachment 3

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Westinghouse Non-Proprietary Class 3

WCAP-17256-NP
Revision 3

May 2012

STP Unit 3 ABWR Prototype Reactor Internals Flow-Induced Vibration Assessment Program



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WCAP-17256-NP
Revision 3

Robert D. Quinn*
ABWR Licensing

May 2012

Reviewer: Subhash Chandra*
ABWR Engineering and Operations

Approved: Nirmal K. Jain*, Manager
ABWR Engineering and Operations
)

*Electronically approved records are authenticated in the electronic document management system.

Westinghouse Electric Company LLC
1000 Westinghouse Drive
Cranberry Township, PA 16066

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LIST OF ACRONYMS

ABWR	Advanced Boiling Water Reactor
APRM	Average Power Range Monitor
ASB	Acoustic Side Branch
ATIP	Automated Traversing In-core Probe
BWR	Boiling Water Reactor
CFD	Computational Fluid Dynamics
CP	Core Plate
CR	Control Rod
CRD	Control Rod Drive
CRDH	Control Rod Drive Housing
CRGT	Control Rod Guide Tube
CVAP	Comprehensive Vibration Assessment Program
DCD	Design Control Document
DP	Differential Pressure
ECCS	Emergency Core Cooling System
EPU	Extended Power Uprate
ERV	Electromatic Relief Valve
FEM	Finite Element Model
FIV	Flow-Induced Vibration
FMCRD	Fine Motion Control Rod Drive
FW	Feedwater
HPCF	High Pressure Core Flooder
ICGT	In-core Guide Tube
ICM	In-core Monitoring
ICMGT	In-core Monitor Guide Tube
ICMH	In-core Monitor Housing
IGSCC	Intergranular Stress Corrosion Cracking
J-ABWR	Japanese ABWR
LPFL	Low Pressure Flooder
LPRM	Local Power Range Monitor
LSR	Lower Support Ring
LTP	Licensed Thermal Power
LVDT	Linear Variable Differential Transformer
MSIV	Main Steam Isolation Valve
MSL	Main Steam Line
NDE	Nondestructive Examination
NRC	U.S. Nuclear Regulatory Commission
OLTP	Original Licensed Thermal Power
PORV	Pilot Operated Relief Valve
PRNM	Power Range Neutron Monitoring
PSD	Power Spectral Density
PT	Pressure Transducer

RFCS	Recirculation Flow Control System
RG	Regulatory Guide
RHR	Residual Heat Removal
RIP	Reactor Internal Pump
RJ-ABWR	Reference Japanese ABWR
RPV	Reactor Pressure Vessel
RMS	Root Mean Square
SCF	Stress Concentration Factor
SG	Strain Gage
SMT	Scale Model Test
SRNM	Startup Range Neutron Monitoring
SRSS	Square Root Sum of the Squares
SRV	Safety Relief Valve
STP	South Texas Project
TSVC	Turbine Stop Valve Closure
USR	Upper Support Ring

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EXECUTIVE SUMMARY

The South Texas Project Unit 3 (STP Unit 3) nuclear power plant is the first Advanced Boiling Water Reactor (ABWR) constructed in the U.S. ABWRs have been successfully operating since 1996 in Japan; there are four similar ABWRs currently operating in Japan and two more units are under construction in Japan. In addition, two other ABWRs are under construction in Taiwan.

U.S. NRC Regulatory Guide (RG) 1.20, Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing, provides guidance for the comprehensive vibration assessment program (CVAP) for nuclear power plants during preoperational and initial startup testing. The program is intended to demonstrate that the reactor internals are adequately designed to withstand flow-induced vibration (FIV) forces at normal and transient plant operating conditions for the design life of the plant. The latest revision (Revision 3, March 2007) of RG 1.20 contains additional requirements based on recent BWR experiences on the steam dryers. STP Unit 3 is designated as a prototype plant. Kashiwazaki-Kariwa Unit 6, herein referred to as the reference Japanese ABWR (RJ-ABWR), commenced successful commercial operation in 1996 and went through extensive testing to show that the reactor internals are adequately designed to withstand FIV loads. Since reactor internals in STP Unit 3 are essentially identical to RJ-ABWR internals, the results of the RJ-ABWR preoperational and startup tests are used as a guide to establish the CVAP for STP Unit 3.

The elements of the CVAP are (a) a vibration and fatigue analysis program, (b) a measurement program, (c) an inspection program, and (d) an evaluation program. This report provides an overview of the STP Unit 3 CVAP.

The vibration and fatigue analysis program is divided into two parts, the non-dryer reactor internals and the steam dryer. The vibration and fatigue analysis program for the non-dryer reactor internals includes the determination of the FIV forcing functions, modal analysis, and assessment of uncertainties and biases associated with the analyses. The forcing functions were developed using various methods, such as Computational Fluid Dynamics (CFD) or simplified analyses. The predictive stress analyses have been performed for the non-dryer components and are documented in WCAP-17371-P, "South Texas Project Units 3 and 4 Reactor Internals Non-Dryer Component Flow-Induced Vibration Assessment."

The vibration and fatigue analysis program for the dryer is documented in WCAP-17385-P, "STP Unit 3 Steam Dryer Flow-Induced Vibration Assessment." The structural adequacy of the dryer for LTP conditions is primarily established by excellent operational experience of plants essentially identical to STP in configuration and operational parameters. These plants have operated for a number of years without any evidence of dryer structural degradation as evidenced by inspection results that found no indications. These dryers have not been repaired for any reason so far. Additional confirmation of the dryer structural adequacy will be provided by extensively instrumenting the STP Unit 3 dryer during initial power ascension tests. Actual STP Unit 3 data will be used to calculate dryer loads at 60% power where dryer loads are expected to be approximately []^{a,c} the full power condition. Dryer stress analyses will be performed at 60% power to show that stresses are below the endurance limit. Limit curves will be developed at higher power levels and it will be shown analytically that stresses are below the endurance limit. An additional measure undertaken by STP to provide further confirmation that dryers stresses will be below the endurance limit, involves design changes []^{a,c} at power levels beyond LTP

condition. This further minimizes the dryer load and helps assure lower dryer stresses. Thus, the approach used to qualify the steam dryer is based mainly on the [

] ^{a,c}

The vibration and fatigue analysis program for the reactor internals components summarized in this report demonstrates that the reactor internals are adequately designed to withstand FIV forces and satisfy the guidelines of RG 1.20, Revision 3.

The STP Unit 3 measurement and inspection programs are described in WCAP-17370-P, "South Texas Project Unit 3 Comprehensive Vibration Assessment Program Measurement, Test, and Inspection Plan." This report describes the vibration and stress measurement program, which includes preoperational and startup testing and an inspection program. The measurement program consists of instrumenting selected internal components with vibration sensors that include strain gages, accelerometers, Linear Variable Differential Transformers (LVDTs) (to measure displacement) and dynamic pressure transducers, and collecting and analyzing the data at selected plant operating conditions during pre-operational testing (prior to fuel load) and during initial plant startup (power ascension after fuel load). Redundancy is accounted for in selecting the instrumentation. The collected data will be compared to allowable vibratory stress limits to determine if the stress levels are acceptable for long term operation of the plant.

The inspection program requires inspection of the reactor vessel internals before and after pre-operational testing to confirm structural integrity of the reactor vessel internals to withstand FIV with no damage to reactor internals or loose parts resulting from the testing.

The evaluation program plan is briefly described in Section 8 of this report. The evaluation will occur after the data is collected during the preoperational and startup test measurement and inspection programs and will be documented in a separate report.

The Comprehensive Vibration Assessment Program (CVAP) of the first U.S. ABWR plant described in this report satisfies the guidelines of RG 1.20, Revision 3 for a prototype plant. The program for plants subsequent to the prototype U.S. ABWR plant will satisfy the guidelines for a Non-prototype, Category I plant.

1 INTRODUCTION

1.1 OVERVIEW

The purpose of this report is to describe the methodologies and results of a FIV evaluation of the reactor vessel internal components for STP Unit 3. A CVAP was developed and will be implemented for STP Unit 3 reactor vessel internals in accordance with RG 1.20 Revision 3 (Reference 1-1). A description of the STP Unit 4 CVAP is contained in Reference 1-5.

1.1.1 Purpose of CVAP

The purpose of the CVAP is to demonstrate that the STP Unit 3 reactor has acceptable FIV characteristics, and that the supporting FIV analysis methodology is accurate for predicting the STP 3 FIV responses. The CVAP described here is in compliance with the guidelines of RG 1.20, Revision 3 (Reference 1-1). The CVAP will demonstrate that the reactor internal components will accommodate the steady state and transient vibratory loads expected throughout the lifetime service of the plant. The STP Unit 3 CVAP consists of four separate programs:

- Vibration and Fatigue Analysis Program,
- Measurement Program,
- Inspection Program, and
- Evaluation Program.

1.1.2 Classification

Under RG 1.20, Revision 3, the STP Unit 3 reactor is categorized as a prototype reactor since it represents a first-of-a-kind design that has not previously been validated by a similar reactor design in the U.S. RG 1.20, Revision 3 requires a several-step process for verifying the FIV characteristics of a prototype reactor. These steps are described in Section 1.2.

1.2 CVAP COMPONENTS

The purpose of the CVAP is to verify the structural integrity of the reactor internals to flow-induced vibration prior to commercial operation. The STP Unit 3 CVAP consists of four separate programs:

- Vibration and Fatigue Analysis Program,
- Measurement Program,
- Inspection Program,
- Evaluation Program.

1.2.1 Vibration and Fatigue Analysis Program Overview

A vibration and fatigue analysis is performed for the non-dryer components prior to testing to demonstrate that the FIV characteristics are acceptable. Analysis should include the pressure fluctuations and vibration expected during steady state and transient conditions that are expected to be encountered during pre-operational, initial start-up, and normal operation. This analysis has been performed for STP Unit 3 for the bounding analysis cases for the reactor internal components. Details of the methods and results of the non-steam dryer reactor internals components are reported in a separate document, WCAP-17371-P (Reference 1-2).

For the steam dryer, the approach to show the structural adequacy of the dryer due to FIV loads is based on plant operating experience for ABWRs operating in Japan and design modifications [

] ^{a,c}

The evaluation of the transient loads will be performed as part of the design calculations that include the evaluation of Service Levels, A, B, C, and D load combinations. For example, the transient condition analysis will include the evaluation of the turbine stop valve closure (TSVC) load on the steam dryer. The transient loads (e.g., TSVC) are short in duration with a limited number of cycles and the resulting loads can produce minimal incremental fatigue usage. These transient loads are not steady state vibratory loads. The TSVC load will be evaluated and documented in the ASME Design Report.

A summary of the vibration and fatigue analysis program is provided in Section 2.

1.2.2 Measurement Program Overview

A Measurement Program is required in which instrumentation is applied to selected reactor internal components which are subjected to higher stresses during operation. The purpose of the Measurement Program is to demonstrate that the reactor internal components have acceptable stress levels during steady state and transient operating conditions, and to show that for the non-dryer components the FIV prediction methods are accurate in estimating stress levels in the components. For the dryer, the measurement program will include instrumentation on the dryer and generation of limit curves at 60% power.

A selected set of internal components will be instrumented with strain gages, accelerometers, pressure transducers, and displacement measurement devices. Measurements on the non-dryer components will be done prior to fuel load and measurements on the dryer will be obtained during the initial plant startup. Details of the Measurement Program for STP Unit 3 are reported in a separate document, WCAP-17370-P (Reference 1-3).

1.2.3 Inspection Program Overview

An Inspection Program is required in which a comprehensive set of inspections of the reactor internal components is carried out prior to and after pre-operational testing. The Inspection Program will examine and document the condition of mating surfaces of the reactor internals at all critical interfaces. Specific locations will be inspected for evidence of contact and wear for effects of vibration. These locations include contact and potential contact surfaces between the major load bearing components, highly stressed locations identified in the analysis program, specific welds, and CVAP instrumentation housing, mounting, and conduits. These inspections will be performed prior to fuel load. The Inspection Program will also cover the inspection of the dryer components after completion of the first cycle of operation. Details of the Inspection Program for STP Unit 3 are reported in a separate document, WCAP-17370-P (Reference 1-3).

1.2.4 Evaluation Program Overview

An Evaluation Program will be performed after completion of the pre-operational testing for the non-dryer components. The evaluation of the steam dryer will occur after completion of the initial startup (power ascension). The evaluation will compare measured strains, displacements, and dynamic pressures against predicted values for the CVAP tests. This comparison will provide an assessment of the accuracies of the FIV prediction methodology. It will also provide a comparison with acceptance criteria for the measured parameters and demonstrate that the FIV characteristics for STP Unit 3 are acceptable. Any significant difference between predicted and measured response will be assessed and reconciled.

1.3 TEST CONDITIONS

Testing will be performed in three phases: preoperational testing, pre-critical testing (after fuel load but prior to achieving reactor criticality), and initial startup testing. Preoperational flow testing will be performed prior to fuel load. Pre-critical testing will be performed after fuel load, but before reactor criticality, and initial startup testing will be performed during power ascension. Justification for test conditions is provided in Reference 1-4. More detailed test conditions will be prepared in the future as the plant startup test procedures become available.

1.3.1 Preoperational Testing

Preoperational testing will be conducted prior to fuel loading with all internal components installed with the exception of fuel assemblies, in-core instrumentation, and the steam dryer. The selected steady state and transient test conditions for preoperational testing are listed in Table 1.3.1-1. [

] ^{a,c}

Table 1.3.1-1 Preoperational Test Conditions

Item	Test Condition	No. of RIPs ON		Notes
		Start	End	
1				
2				
3				
4				
5				
6				
7				
8				

a,c

The planned test duration is a few days. The duration of the test will be such that the lowest natural frequency component will accumulate 10^6 cycles to reach the endurance limit. Therefore, the resulting

vibratory cycles are in excess of 10^6 cycles for vibration of major reactor internal components at their lowest dominant response frequencies.

1.3.2 Pre-Critical Testing with Fuel

Pre-critical testing will be performed with fuel prior to reactor criticality with all components installed and reactor assembly complete. The selected test conditions for Pre-critical testing are listed in Table 1.3.2-1

Table 1.3.2-1 Pre-critical Test Conditions

Item	Test Condition	No. of RIPs ON		Notes*
		Start	End	

1.3.3 Startup Tests

Startup tests will be performed with fuel in place and reactor assembly complete in all respects and after reactor achieved criticality. The first startup testing at power is planned after the reactor reaches normal operating temperature and pressure and rod line between approximately 50% and 75%, and at 100%. The selected test conditions for startup testing are listed in Table 1.3.3-1.

Table 1.3.3-1 Startup Test Plan

Item	Test Condition	No. of RIPs ON		Notes
		Start	End	
1				
2				
3				
4				
5				
6				
7				
8				
9				
10				

a,c

1.4 TEST DURATION

Reference 1-1 requires that the reactor internal components must accumulate a minimum of 10^6 cycles of vibration during the CVAP tests. Based on a bounding lower limit for major internal component fundamental frequency of []^c from Section 5, Table 5.1.2-1, the components will require a minimum of []^c hours of test time to assure all internal components will reach at least 10^6 cycles.

1.5 REFERENCES FOR SECTION 1

- 1-1 U.S. NRC Regulatory Guide 1.20, Rev. 3, "Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing," March 2007.
- 1-2 Westinghouse Report, WCAP-17371-P, Revision 3, "South Texas Project Units 3 and 4 Reactor Internals Non-Dryer Component Flow-Induced Vibration Assessment," May 2012. (Proprietary)
- 1-3 Westinghouse Report, WCAP-17370-P, Revision 3, "South Texas Project Unit 3 Comprehensive Vibration Assessment Program, Measurement, Test and Inspection Plan," May 2012.
- 1-4 XGEN Engineering Document, XGEN-2010-04R1, Rev. 1, "Justification for the Proposed Test and Analysis Plan for STP Unit 3," 11/17/2010. (Proprietary)
- 1-5 Westinghouse Report, WCAP-17257-P, Revision 1, "STP Unit 4 Reactor Internals Flow-Induced Vibration Assessment Program," May 2012. (Proprietary)

2 SUMMARY

2.1 BASIS FOR ASSESSING STRESS MARGIN

2.1.1 Endurance Limit

The ABWR DCD (Reference 2-1, Section 3.9.2.1.1.3) specifies an endurance limit of 9.95 ksi (68.6 MPa) for allowable vibration displacements/deflections. The calculated stress results due to FIV shall be equal to or lower than this established endurance limit. This limit does not need to be adjusted to Young's Modulus at different temperatures (Reference 2-5).

This limit is lower than the "endurance limit" indicated in ASME B&PV Code, Section 3 (Reference 2-2,) fatigue curve. If Curve C for austenitic steels (Reference 2-2, Fig. I-9.2.2) is used, the ASME "endurance limit" is 13.6 ksi. In this report, the term endurance limit, fatigue limit, and allowable stress are used interchangeably and refer to 9.95 ksi (68.6 MPa).

2.2 CALCULATED FATIGUE MARGINS

Dynamic loading conditions or flow-induced vibrations produce cyclical stresses in the components. Reference 2-4 requires that a fatigue analysis be performed to show that the components will not suffer excessive fatigue damage. A measure of the fatigue margin for a given component is provided by the ratio of the peak alternating stress divided by the endurance limit of 9.95 ksi for the component, or:

$$\text{Fatigue margin} = (\text{Peak Alternating Stress}) / (\text{Endurance Limit 9.95 ksi})$$

Table 2.2-1 presents the fatigue margins for the reactor internal components based on bounding operating conditions. Details are provided in Reference 2-3.

For the dryer, the stress ratio, SR, is defined as $SR = (\text{Endurance Limit 9.95 ksi}) / (\text{Peak Alternating Stress})$. The limiting SR based on the structural evaluation of the dryer using RJ-ABWR startup data is $SR = 1.11$. Details are provided in Reference 2-7.

Table 2.2-1 Design Fatigue Margins for STP 3 Non-Dryer Reactor Internals

a,c

2.3 REFERENCES FOR SECTION 2

- 2-1 Toshiba Document, ABWR Design Control Document/Tier 2.
- 2-2 ASME Boiler and Pressure Vessel Code, Division I, Section III, 1989 Edition, with no Addenda.
- 2-3 Westinghouse Report, WCAP-17371-P, Revision 3, "South Texas Project Units 3 and 4 Reactor Internals Non-Dryer Component Flow-Induced Vibration Assessment," May 2012. (Proprietary)
- 2-4 U.S. NRC Regulatory Guide 1.20, Rev. 3, "Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing," March 2007.
- 2-5 Westinghouse Letter, LTR-RIDA-10-294, Rev. 0, "Recommended Damping Ratio, Peak-to-RMS Ratio, and Stress Concentration Factor Application for ABWR Reactor Internals Flow-Induced Vibration Analysis," 10/28/2010. (Proprietary)
- 2-6 XGEN Report, XGEN-2010-03R2, "FIV Evaluation of STP Unit 3/4 Reactor Internals: Top Guide, Core Plate, Vent and Head Spray, Surveillance Specimen Holder and Fuel Support Casting," January 2011. (Proprietary)
- 2-7 Westinghouse Report, WCAP-17385-P, Revision 3, "STP Unit 3 Steam Dryer Flow-Induced Vibration Assessment," May 2012. (Proprietary)

3 DESCRIPTION OF THE STRUCTURAL ASSEMBLIES

The reactor internal components are shown and identified in Figure 3-1. The following sections provide detailed descriptive material for the components taken from References 3-1 through 3-4.

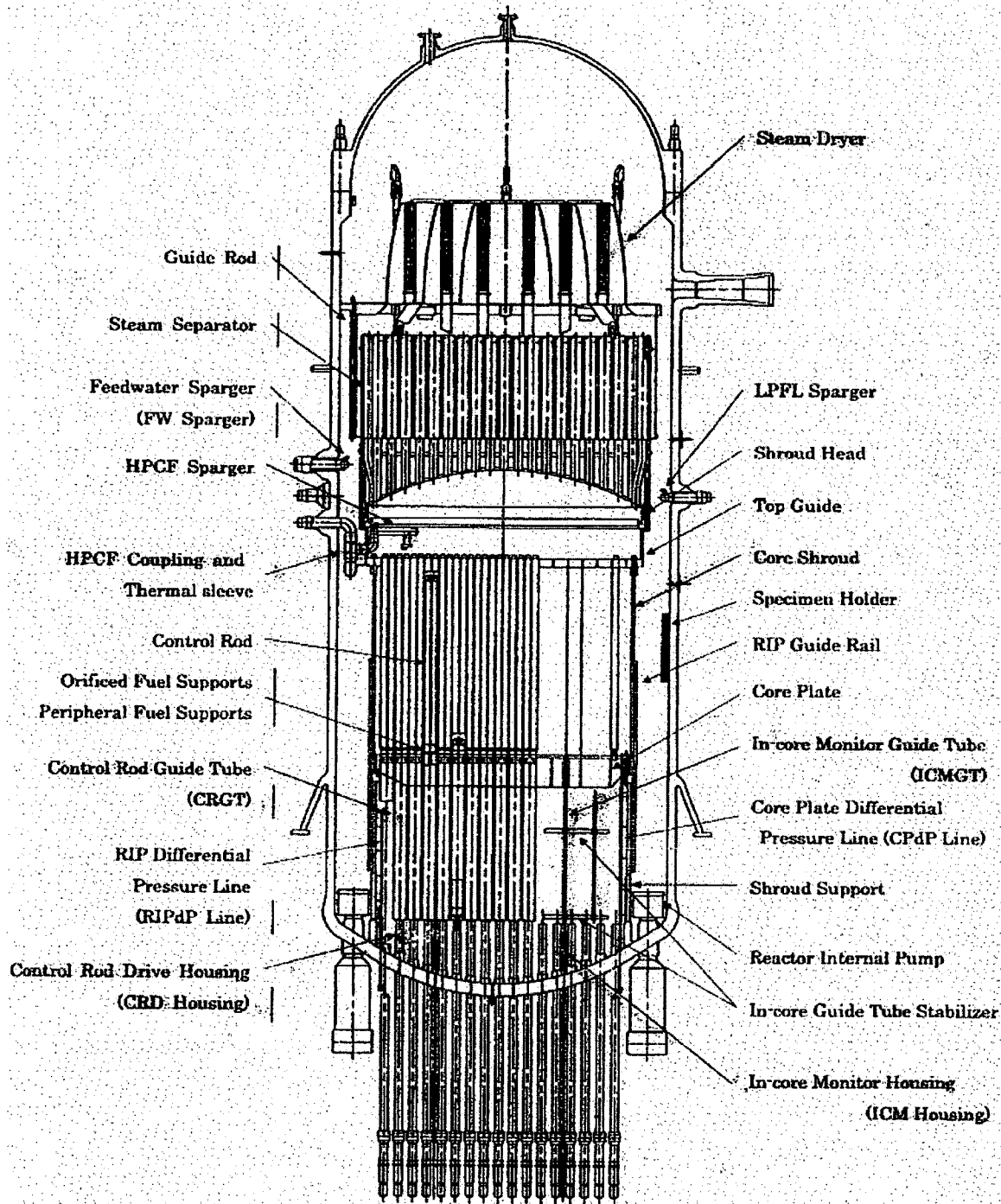
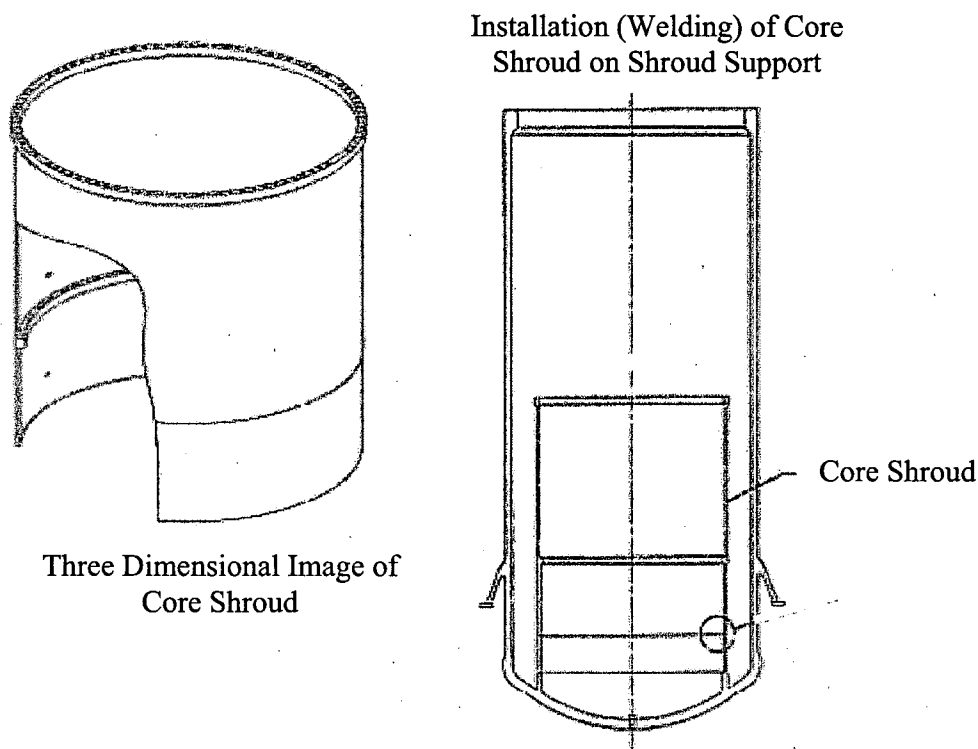


Figure 3-1 ABWR Reactor Internal Component Arrangement
(Taken from Reference 3-1)

3.1 CORE SUPPORT STRUCTURES

3.1.1 Core Shroud

The shroud support, shroud, and top guide make up a stainless steel cylindrical assembly that provides a partition to separate the upward flow of coolant through the core from the downward recirculation flow. This partition separates the core region from the downcomer annulus. The volume enclosed by this assembly is characterized by three regions; the upper, central, and lower regions. The upper portion surrounds the core discharge plenum, which is bounded by the shroud head on top and the top guide plate below. The central portion of the shroud surrounds the active fuel and forms the longest section of the assembly. This section is bounded at the top by the top guide plate and at the bottom by the core plate. The lower portion, surrounding part of the lower plenum, is welded to the RPV shroud support. The shroud provides the horizontal support for the core by supporting the core plate and top guide. Figure 3.1.1-1 shows the installation of the core shroud with the RPV.

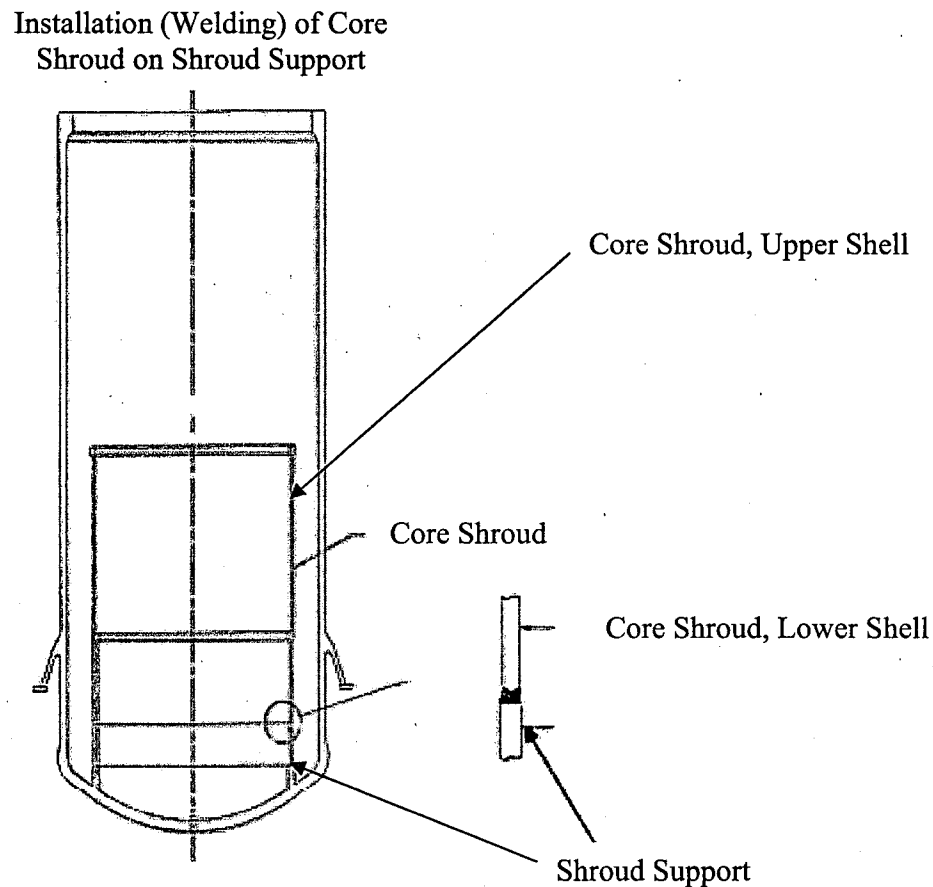


**Figure 3.1.1-1 RPV and Core Shroud Assembly
(Taken from Reference 3-2)**

3.1.2 Shroud Support

The RPV shroud support is designed to support the shroud and includes the internal pump deck that locates and supports the pumps. The pump discharge diffusers penetrate the deck to introduce the coolant to the inlet plenum below the core. The pump deck is a horizontal structure welded to the vessel wall to provide support to the shroud, pump diffusers, and core and pump deck differential pressure (DP) lines. The structure is a ring plate welded to the vessel wall and to a vertical cylinder supported by vertical stilt legs from the bottom head. Figure 3.1.2-1 shows the shroud support assembly.

The shroud support supports the weight of the core shroud, core plate, peripheral fuel assemblies, top guide and shroud head and steam separator assembly.



**Figure 3.1.2-1 RPV and Shroud Support Assembly
(Taken from Reference 3-2)**

3.1.3 Control Rod Guide Tubes (CRGT)

The CRGTs located inside the vessel extend from the top of the control rod drive (CRD) housings up through holes in the core plate. Each guide tube is designed as the guide for the lower end of a control rod and as the support for an orificed fuel support. The fuel support locates the four fuel assemblies surrounding the control rod. The bottom of the guide tube is supported by the CRD housing. The CRGTs also contain holes near the top of the CRGT and below the core plate to allow coolant flow to the orificed fuel supports. Figure 3.1.3-1 shows these components.

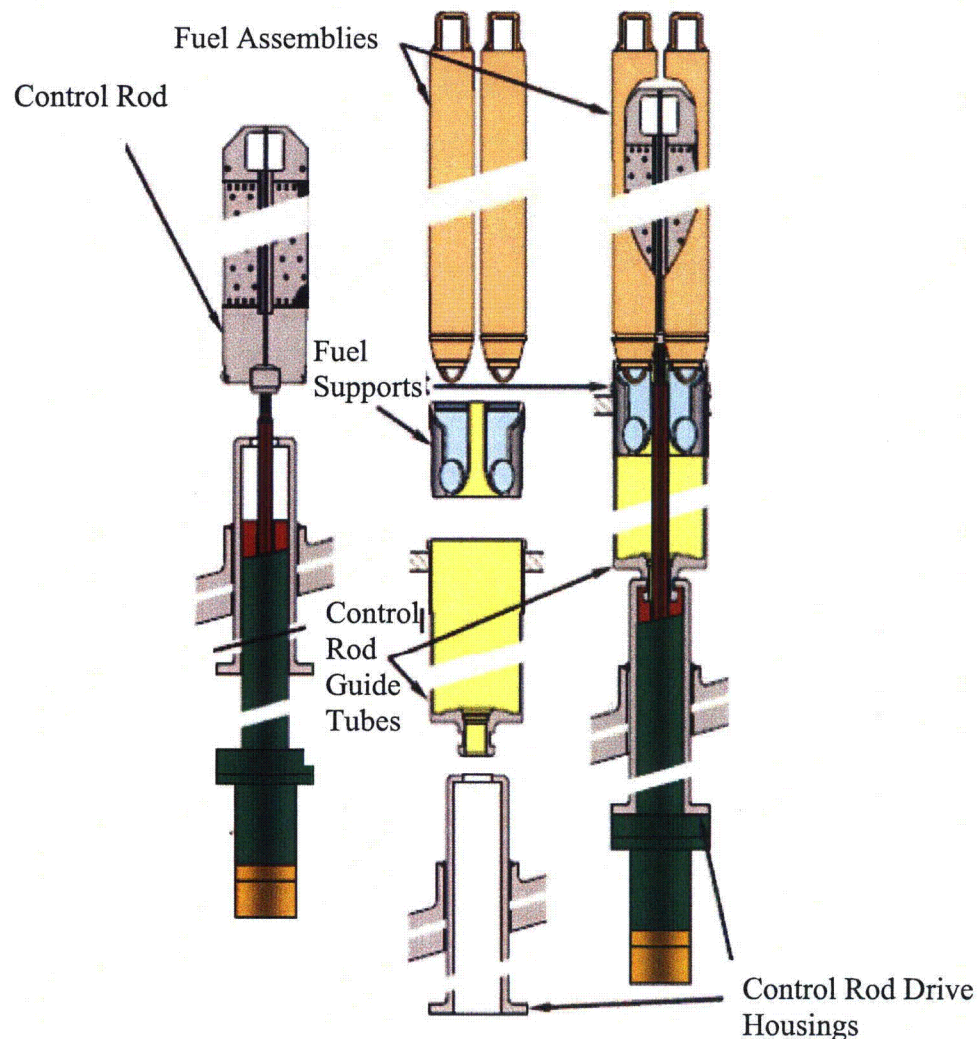


Figure 3.1.3-1 CRGT, and CRD Housing Assembly
(Taken from Reference 3-2)

3.1.4 Core Plate and Fuel Supports

The core plate consists of a circular stainless steel plate with round openings and is stiffened with a rim and beam structure. The core plate provides lateral support and guidance for the CRGT, in-core flux monitor guide tubes, peripheral fuel supports, and startup neutron sources. The peripheral fuel support and startup neutron sources are supported vertically by the core plate. The entire assembly is bolted to a support ledge in the lower portion of the shroud. The core plate is shown in Figure 3.1.4-1.

The fuel supports (Figure 3.1.4-2) are of two basic types: peripheral supports and orificed fuel supports. The peripheral fuel supports are located at the outer edge of the active core and are not adjacent to the control rods. Each peripheral fuel support supports one fuel assembly and has an orifice designed to assure proper coolant flow to the peripheral fuel assembly. Each orificed fuel support holds four fuel assemblies vertically and horizontally and has four orifices to provide proper coolant flow distribution to each rod-controlled fuel assembly. The orificed fuel supports rest on the top of the CRGTs, which are supported laterally by the core plate. The control rods pass through cruciform openings in the center of the orificed fuel support. This locates the four fuel assemblies surrounding a control rod.

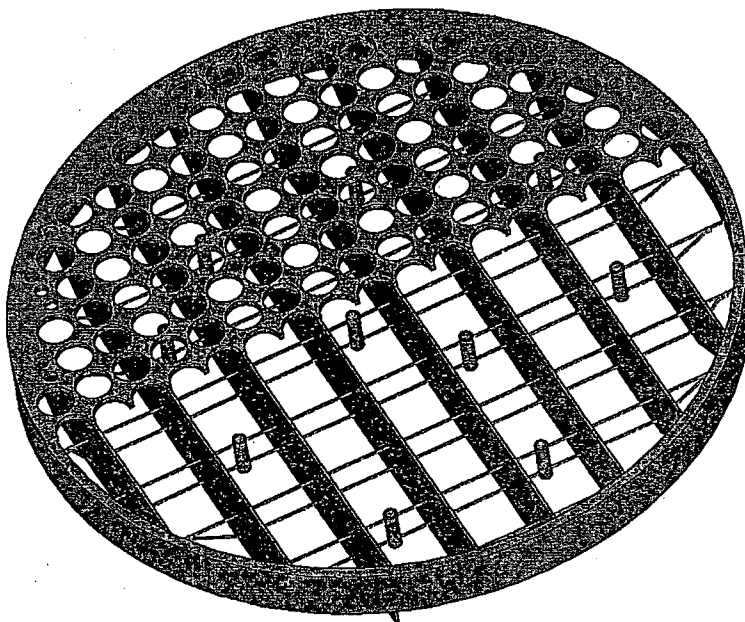


Figure 3.1.4-1 Core Plate

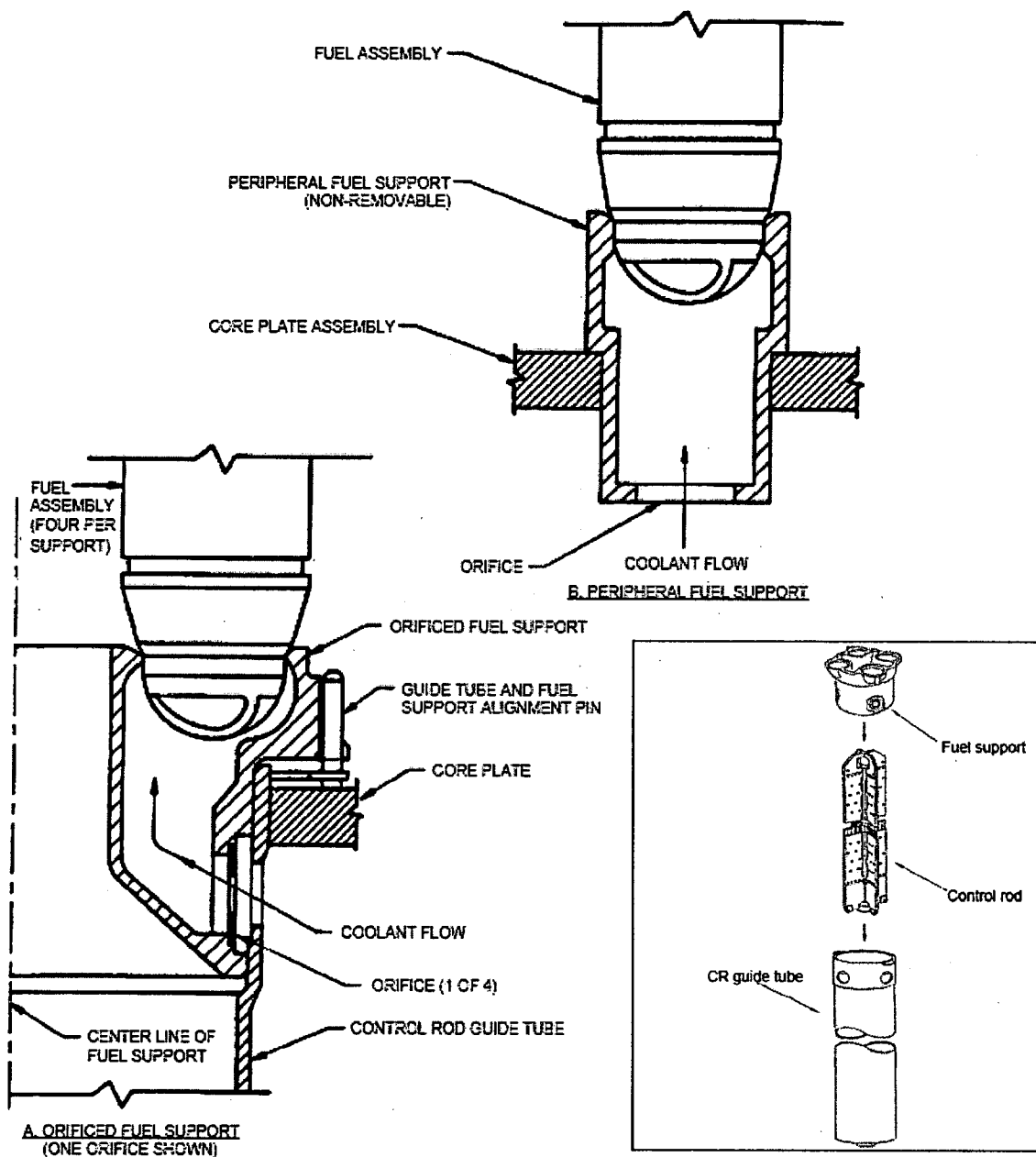


Figure 3.1.4-2 Fuel Supports
(Taken from Reference 3-2)

3.1.5 Top Guide

The top guide consists of a circular plate with square openings for fuel and a cylindrical side forming an upper shroud extension with a top flange for attaching the shroud head. Each opening provides lateral support and guidance for four fuel assemblies or, in the case of peripheral fuel, less than four fuel assemblies. Holes are provided in the bottom of the support intersections to anchor the in-core flux monitors and startup neutron sources. The top guide is mechanically attached to the top of the shroud. Figure 3.1.5-1 and 3.1.5-2 show the top guide and its installation in the RPV.

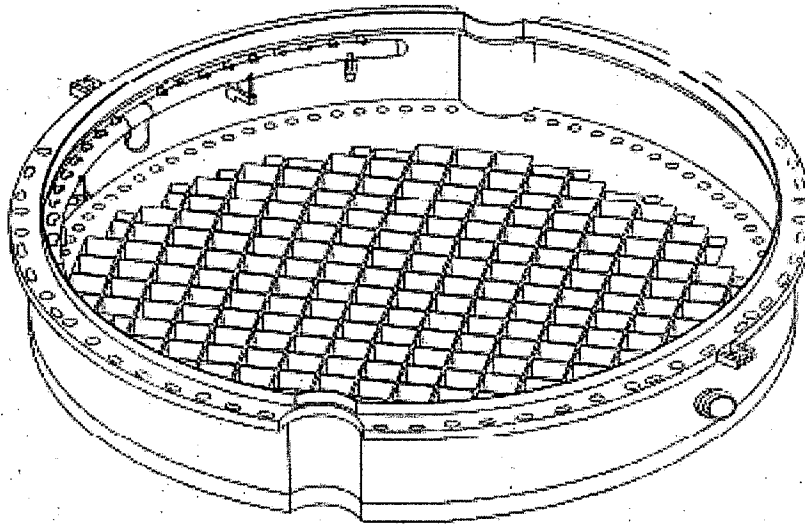
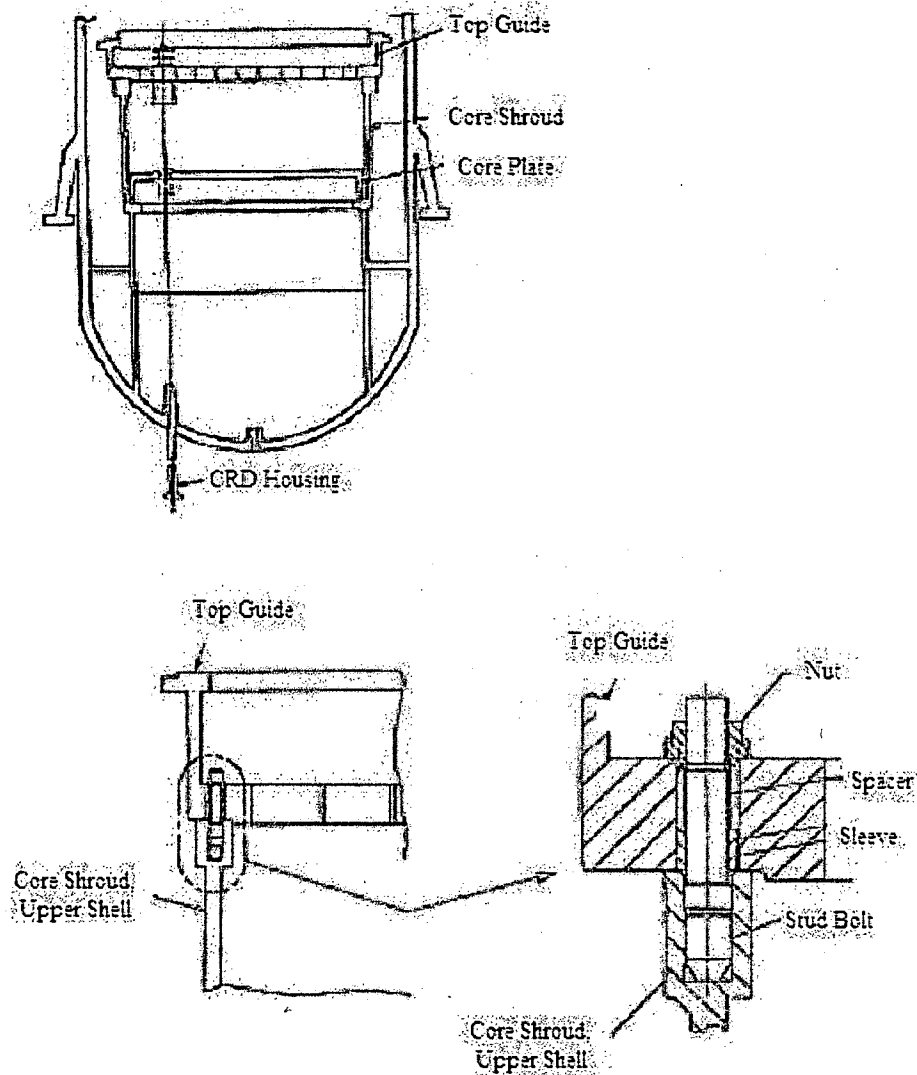


Figure 3.1.5-1 Top Guide
(Taken from Reference 3-2)

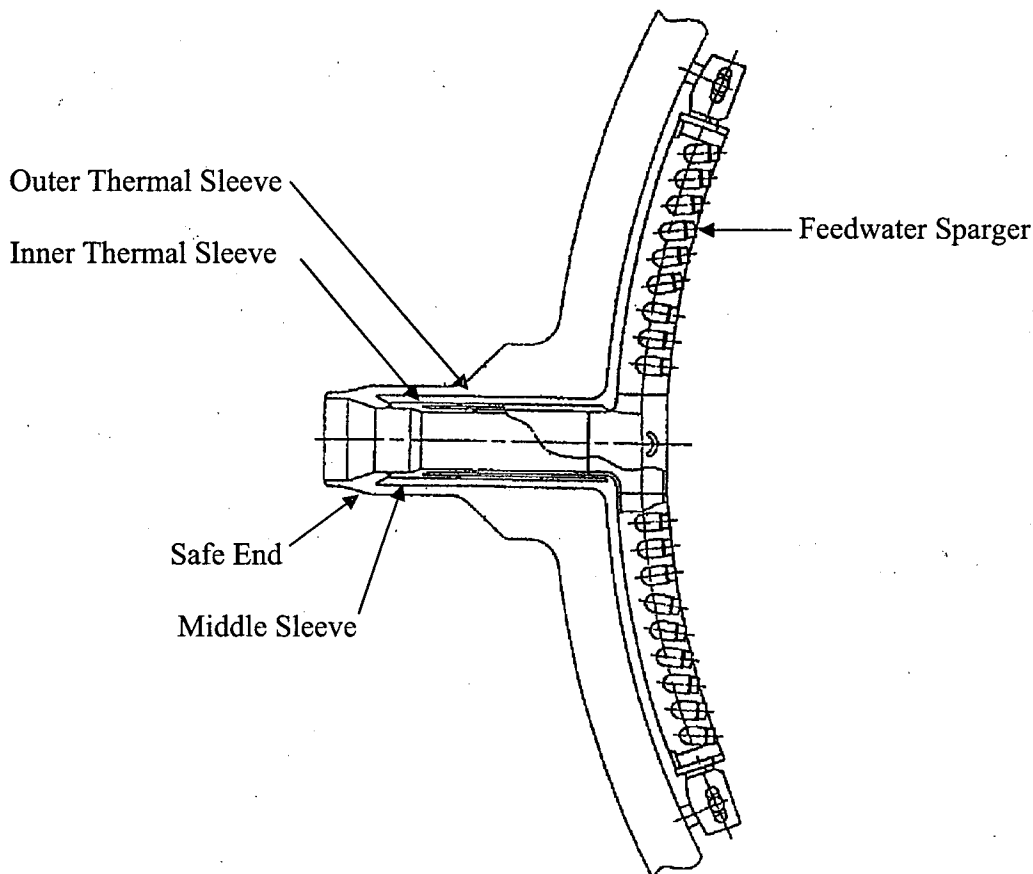


**Figure 3.1.5-2 Installation of Top Guide
(Taken from Reference 3-2)**

3.2 REACTOR INTERNALS

3.2.1 Feedwater Spargers

The feedwater (FW) spargers are stainless steel headers located in the mixing plenum above the downcomer annulus. A separate FW sparger in two halves is fitted to each FW nozzle using a tee, and is shaped to conform to the curve of the vessel wall. The sparger tee inlet is connected to the RPV nozzle safe end by a double thermal sleeve arrangement. Sparger end brackets are pinned to the vessel brackets in order to support the spargers. The FW flow enters the center of the spargers and is discharged radially inward to mix the cooler FW with the downcomer flow from the steam separators and steam dryer before it contacts the vessel wall. The FW spargers also serve to condense steam in the region above the downcomer annulus and to subcool water flowing to the RIPs. Figure 3.2.1-1 shows a typical FW sparger unit.



**Figure 3.2.1-1 FW Sparger
(Taken from Reference 3-2)**

3.2.2 Low Pressure Flooder Spargers

The low pressure flooder (LPFL) spargers are Safety Class 2 components. The design features of these two spargers of the Residual Heat Removal (RHR) shutdown cooling system are similar to those of the six FW spargers, three of which belong to one FW line supporting the same RHR function. During the Emergency Core Cooling System (ECCS) mode, these spargers support low pressure flooding of the vessel. Two lines of the RHR shutdown cooling system enter the reactor vessel through the two diagonally opposite nozzles and connect to the spargers. The sparger tee inlet is connected to the RPV nozzle safe end by a thermal sleeve arrangement with all connections made by full penetration welds. Figure 3.2.2-1 shows a LPFL sparger assembly.

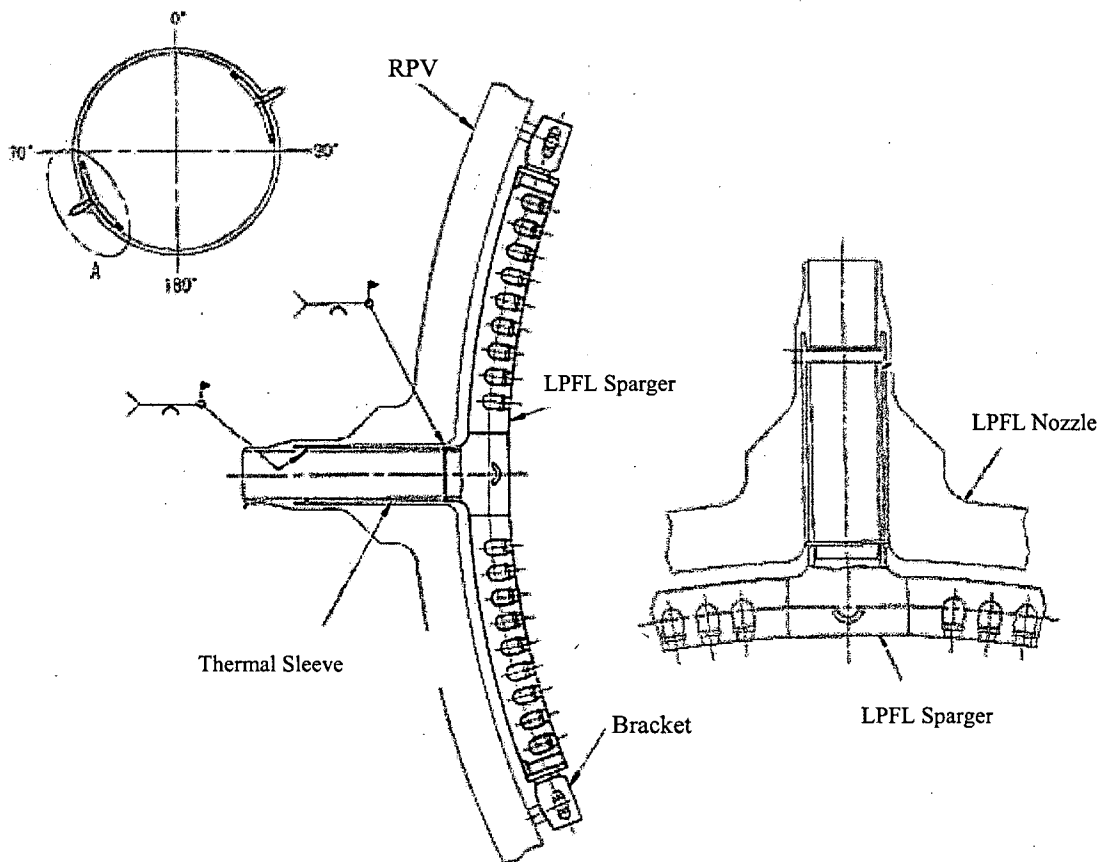


Figure 3.2.2-1 LPFL Sparger Assembly
(Taken from Reference 3-2)

3.2.3 HPCF Sparger and Coupling

The high pressure core flooder (HPCF) spargers and piping are Safety Class 2. The spargers and piping are the means for directing high pressure ECCS flow to the upper end of the core during accident conditions. Each of two HPCF system lines enters the reactor vessel through a diagonally opposite nozzle in the same manner as an RHR low pressure flooder line, except that the curved sparger including the connecting tee is routed around the inside of and is supported by the cylindrical portion of the top guide. A flexible coupling is interposed between the sparger tee inlet and the sleeved inlet connector inside the nozzle. The two spargers are supported to accommodate thermal expansion. Figure 3.2.3-1 shows the HPCF sparger and coupling assembly.

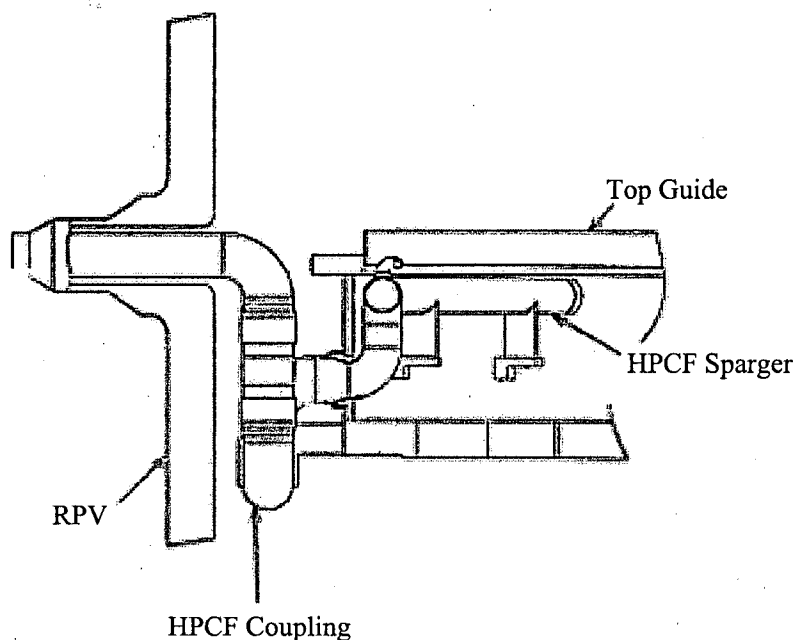


Figure 3.2.3-1 HPCF Sparger and Coupling Assembly
(Taken from Reference 3-2)

3.2.4 Core Plate (CP) and RIP DP Lines

The DP lines comprise the core flow measurement subsystem of the recirculation flow control system (RFCS) and provides two methods of measuring the core flow rates. The CP DP lines (Safety Class 3) and RIP DP lines (non-safety class) enter the reactor vessel separately through reactor bottom head penetrations. Four pairs of the CP DP lines enter the head in four quadrants through four penetrations and terminate immediately above and below the core plate to sense the pressure in the region outside the bottom of the fuel assemblies and below the core plate during normal operation. Figure 3.2.4-1 shows a typical installation for the CP DP lines. Similarly, four pairs of the RIP DP lines terminate above and below the pump deck and are used to sense the pressure rise across the RIPs during normal pump operation. Each pair is routed concentrically through a penetration and upward along a shroud support leg in the lower plenum. Figure 3.2.4-2 shows the typical installation for the RIP DP lines. The CP and RIP DP lines are positioned behind and downstream of the shroud support legs to provide protection from the high velocity jets which pass into the reactor vessel lower plenum from the RIPs.

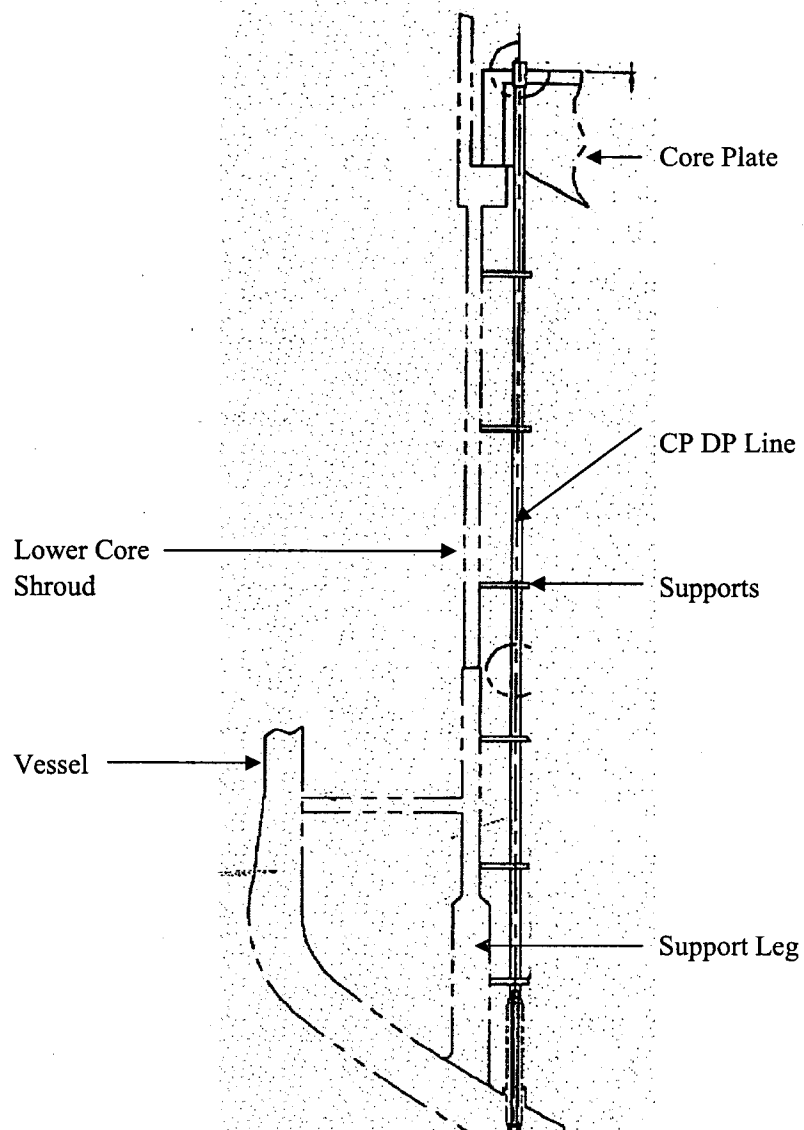
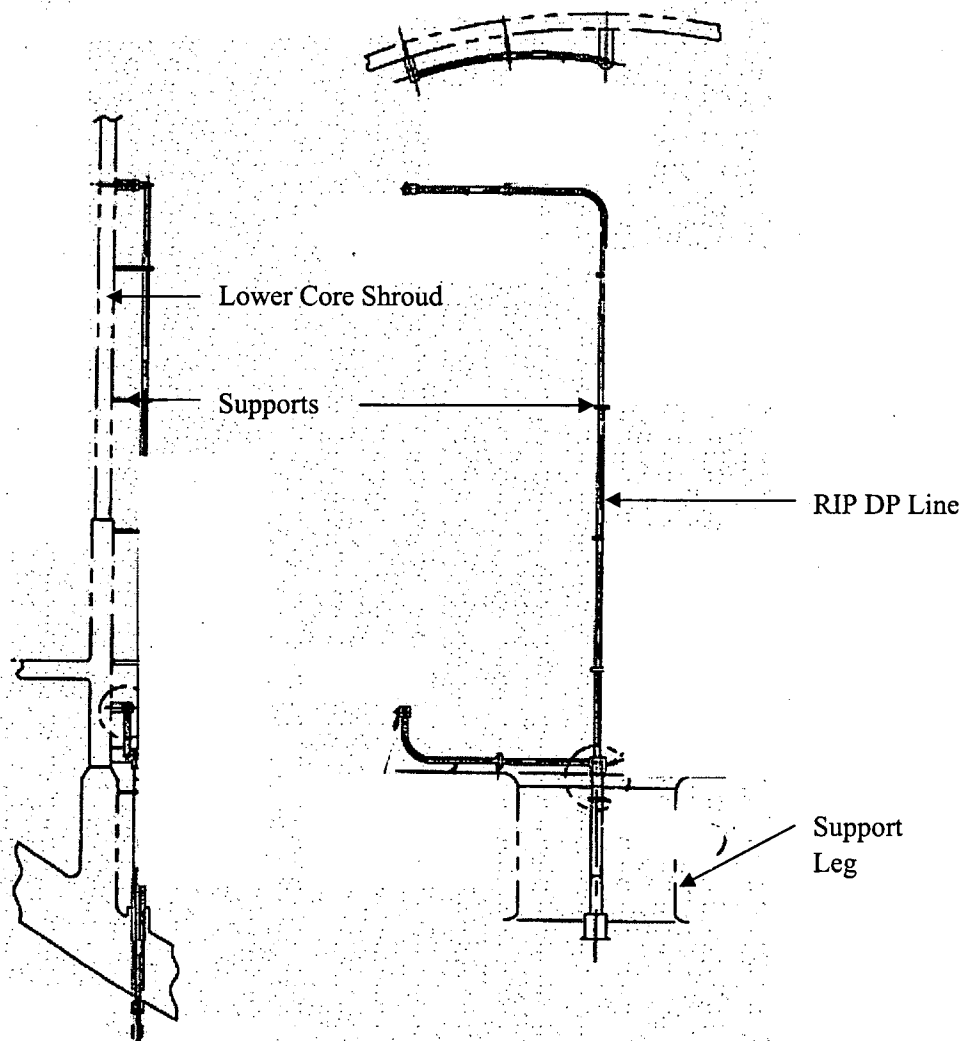


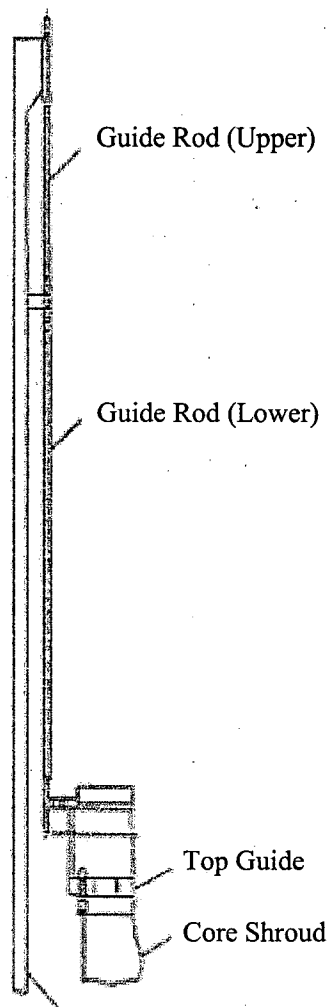
Figure 3.2.4-1 CP DP Line
(Taken from Reference 3-4)



**Figure 3.2.4-2 RIP DP Line
(Taken from Reference 3-4)**

3.2.5 Lower Guide Rods

The lower guide rods are non-safety class components. The lower guide rods have two brackets; the lower and upper. The lower bracket is integrated into the top guide and the upper bracket is welded to the RPV. The lower guide rods and upper guide rods are used to align the shroud head and steam separators during their installation. Once the installation process is complete, the upper guide rods are removed and the lower guide rods remain in the RPV. The lower guide rod is primarily made up of a tube with a solid rod welded to the lower part and a threaded section attached to the upper part. Figure 3.2.5-1 shows the guide rod assembly.



**Figure 3.2.5-1 Guide Rod Assembly
(Taken from Reference 3-4)**

3.2.6 Shroud Head Bolts

The shroud head bolts are used to provide a connection between the shroud head and the top guide. There are 36 shroud head bolts (Reference 3-4) distributed around the circumference of the upper and lower guide rings. The bolts extend vertically from the upper separator ring down to the flange on the shroud head. Figure 3.2.6-1 shows the shroud head bolts.

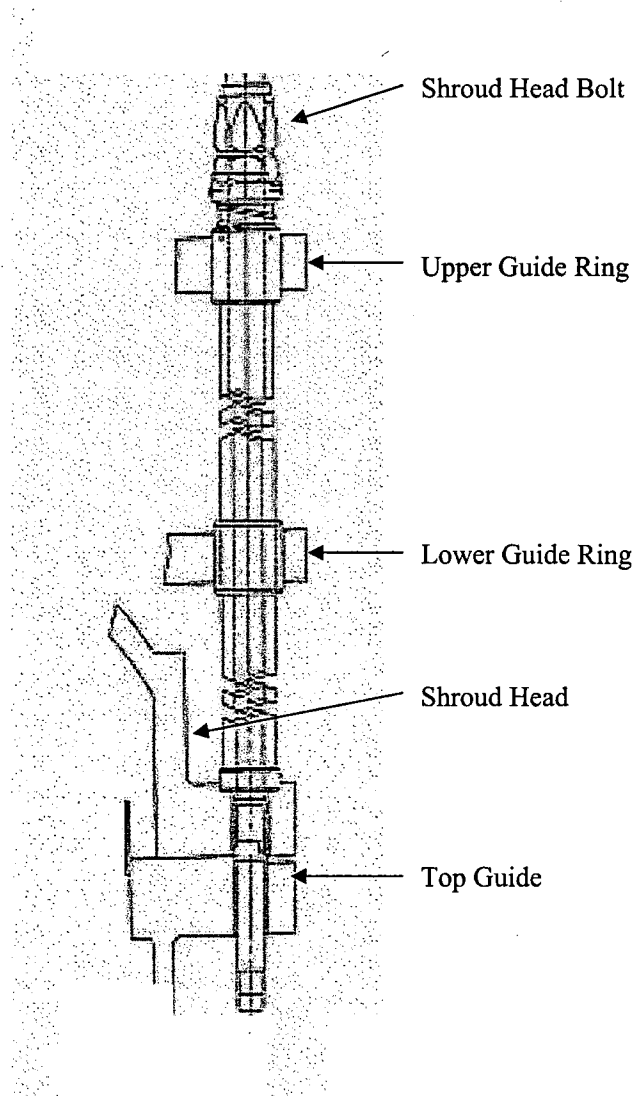


Figure 3.2.6-1 Shroud Head Bolts
(Taken from Reference 3-2)

3.2.7 Shroud Head Lifting Rods/Lugs

The shroud head lifting rods/lugs are non-safety class components. The lifting rods/lugs are used during the installation of the steam separators and shroud head assembly. Their function is to provide the means by which the assembly can be lifted and moved. The lifting rods/lugs remain in the RPV with the shroud head assembly. They consist of four long slender rods with a lug at the top end of each rod and are attached to the shroud head directly, but are also passed through holes in the steam separator guide rings to provide lateral support. Figure 3.2.7-1 shows the shroud head lifting rods/lugs.

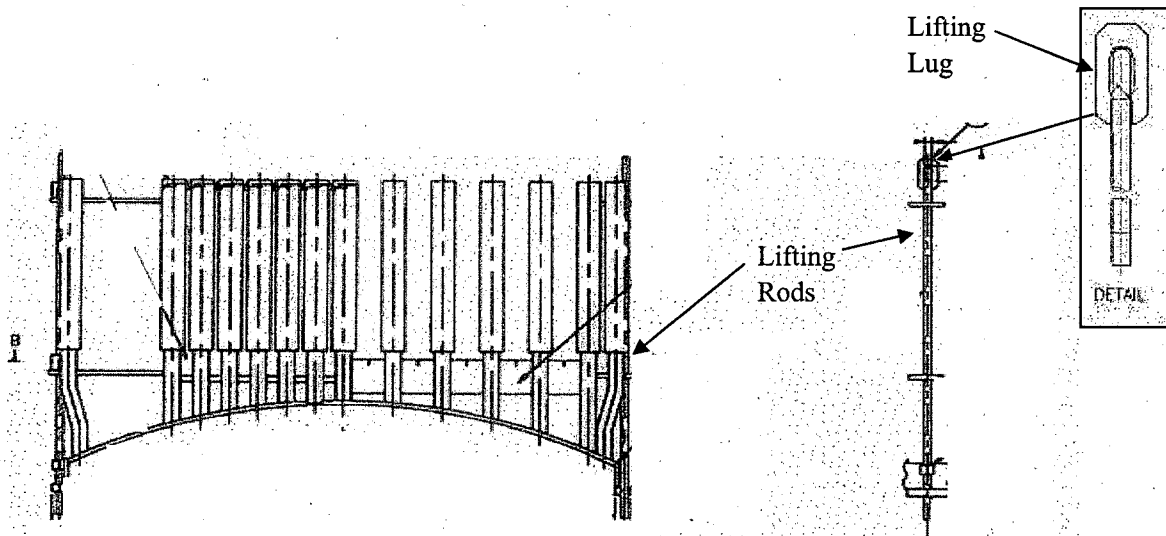


Figure 3.2.7-1 Shroud Head Lifting Rods/Lugs
(Taken from Reference 3-4)

3.2.8 RIP Guide Rails

The RIP guide rails are non-safety class components. They consist of two rails per RIP location. The guide rails are oriented in the vertical direction and extend from approximately mid-core elevation down to an elevation near the pump deck. The guide rails are welded to the core shroud by a series of brackets as shown in Figure 3.2.8-1. The top ends of the guide rails are flared out to assist in positioning the RIP within the guide rails.

The RIP guide rails facilitate the installation and removal of the RIPs by providing guidance to the pump deck elevation. Figure 3.2.8-1 shows the RIP guide rails.

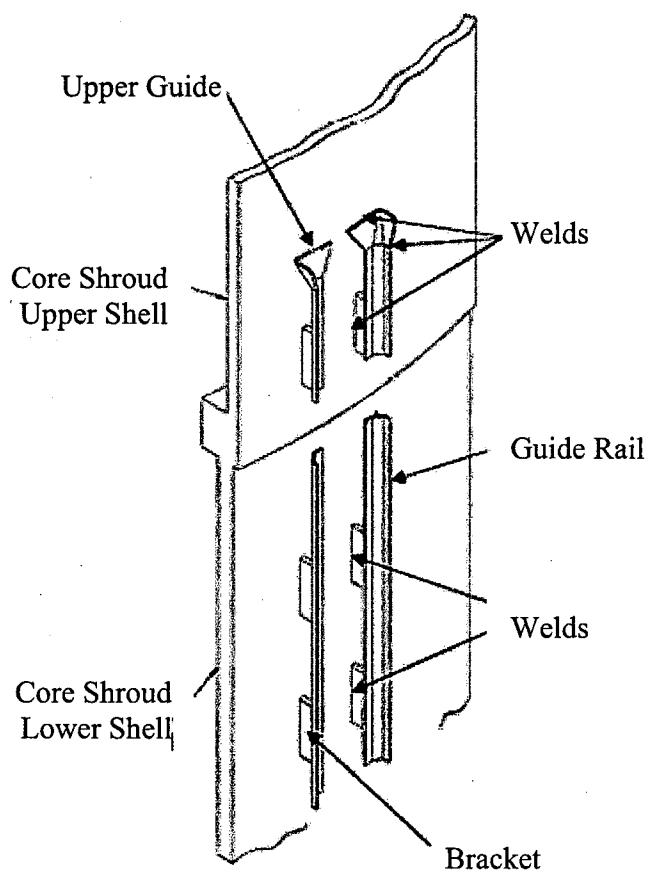
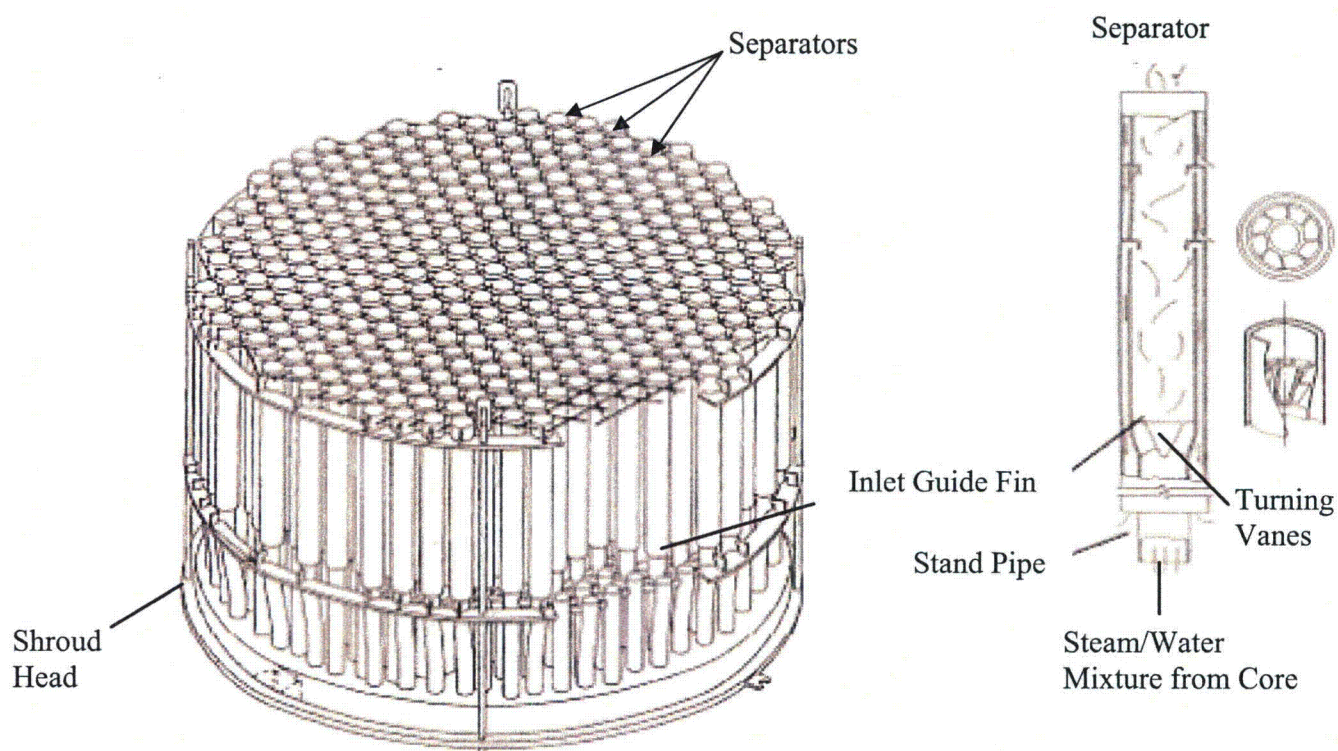


Figure 3.2.8-1 RIP Guide Rails
(Taken from Reference 3-2)

3.2.9 Shroud Head and Separator Assembly

The shroud head and stand pipes/steam separators are non-safety class internal components. The shroud head and steam separator assembly includes the upper flanges and bolts, forming the top of the core discharge mixture plenum, and the upper flanges, separators and their connecting stand pipes, forming the top of the core discharge mixture plenum. The discharge plenum provides a mixing chamber for the steam/water mixture before it enters the steam separators. Individual stainless steel axial flow steam separators are supported on and attached to the top of stand pipes that are welded into the shroud head. The steam separators have no moving parts. In each separator, the steam/water mixture rising through the stand pipe passes vanes that impart a spin to establish a vortex separating the water from the steam. The separated water flows from the lower portion of the steam separator into the downcomer annulus. The assembly is removable from the reactor pressure vessel as a single unit on a routine basis. Figure 3.2.9-1 shows the head and separator assembly.



**Figure 3.2.9-1 Shroud Head and Separator Assembly
(Taken from Reference 3-2)**

3.2.10 Steam Dryer

The steam dryer consists of an upper and lower assembly. The upper assembly consists of the dryer banks, hoods, and troughs. The lower assembly includes the skirt and the drain channels. The upper and lower assemblies are connected by a support ring. The steam dryer is supported by four brackets welded on the vessel shell at the location of the support ring. Figure 3.2.10-1 shows a sketch of the steam dryer.

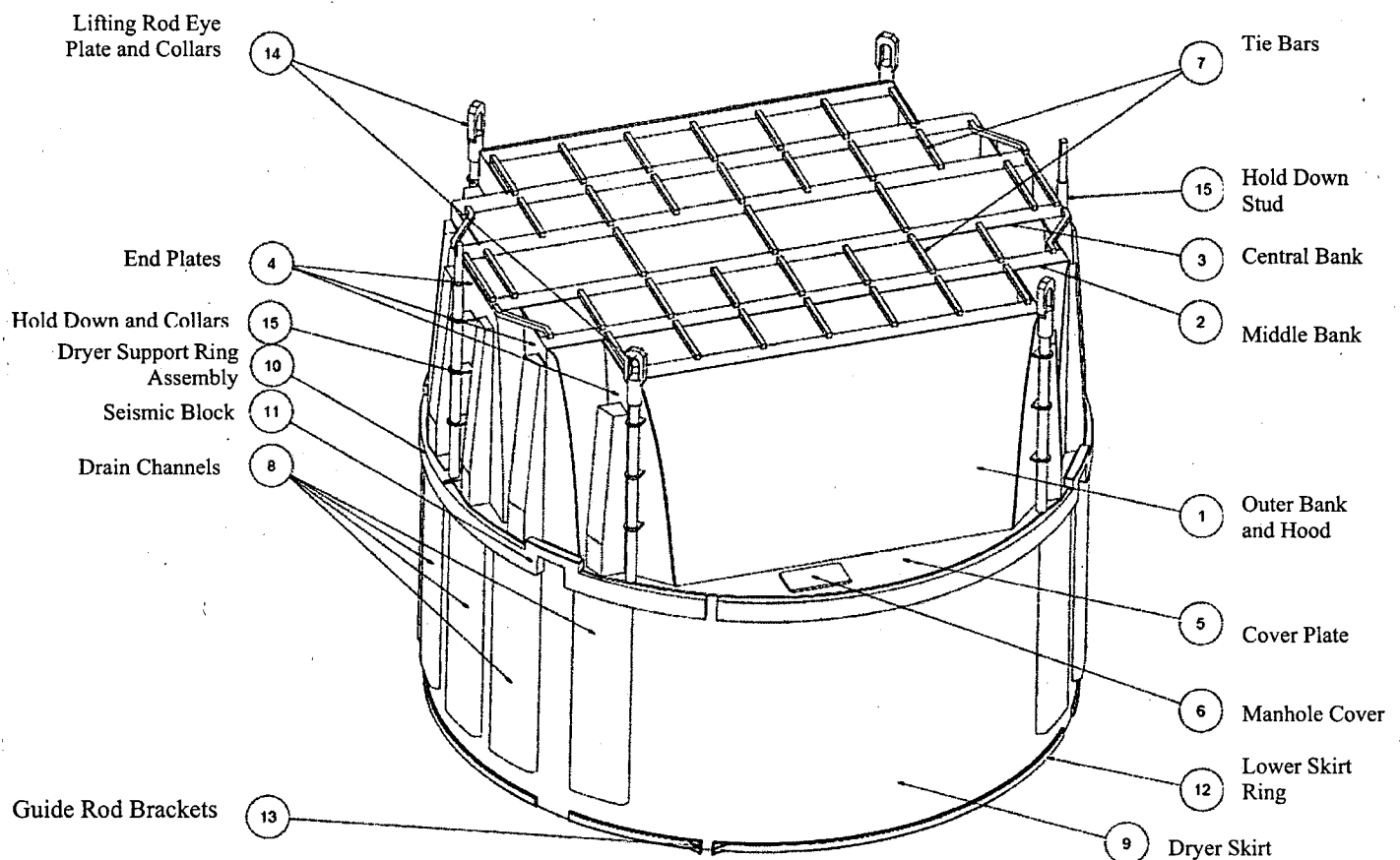


Figure 3.2.10-1 Steam Dryer

3.3 CONTROL ROD DRIVE AND IN-CORE MONITORING HOUSING

3.3.1 Control Rod Housing

The CRD housings extend upwards from the vessel bottom head and connect to the CRGTs; see Figure 3.1.3-1. The CRD housing transmits the weight of the CRGTs, the fuel supports, and the fuel assemblies to the bottom head. The CRD housings also enclose and protect the control rod drive shafts.

3.3.2 In-Core Monitoring Housing System/In-Core Guide Tube Stabilizer (ICMH/ICGT)

The ICMH and ICGT stabilizer are Safety Class 3 components. The guide tubes protect the in-core instrumentation from flow of water in the bottom head plenum and provide a means of positioning fixed detectors in the core as well as a path for insertion and withdrawal of the calibration monitors, Automatic Traversing In-Core Probe (ATIP). The in-core flux monitor guide tubes extend from the top of the in-core flux monitor housing to the top of the core plate. The power range detectors for the power range neutron monitor (PRNM) units, the dry tubes for the startup range neutron monitoring (SRNM) and average power range monitoring (APRM) detectors are inserted through the guide tubes. The local power range monitor (LPRM) detector assemblies are inserted through the guide tubes. Figure 3.3.2-1 shows an individual in-core monitor guide tube.

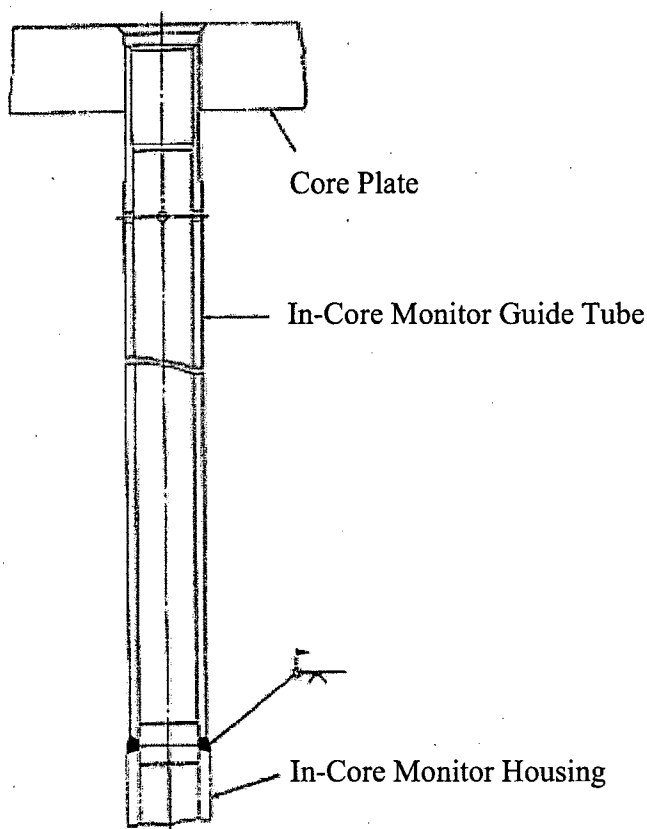


Figure 3.3.2-1 In-Core Monitor Assembly
(Taken from Reference 3-2)

Two levels of stainless steel and nickel alloy stabilizer latticework of clamps, tie bars, and spacers give lateral support and rigidity to the guide tubes. The stabilizers are connected to the core shroud and shroud support. The bolts are tack-welded after assembly to prevent loosening during reactor operation. Figure 3.3.2-2 shows a portion of the latticework arrangement and of the typical clamp assemblies for the ICGT stabilizer.

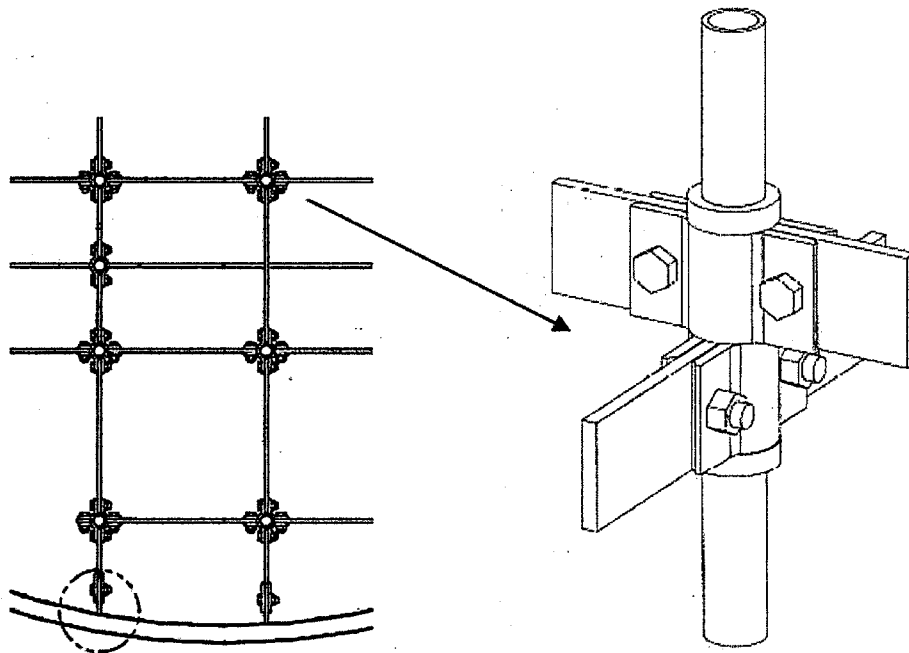
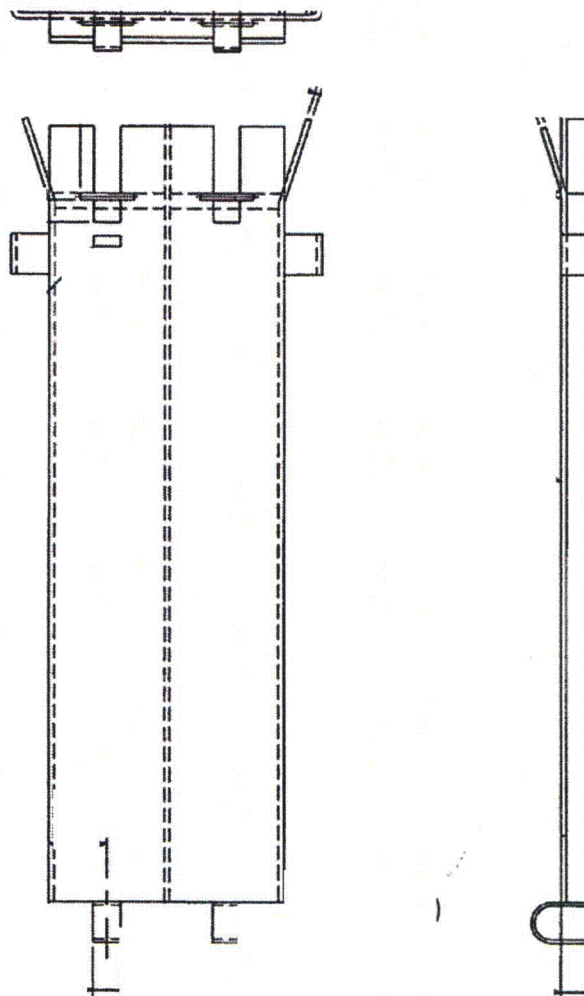


Figure 3.3.2-2 In-Core Guide Tube Stabilizer
(Taken from Reference 3-4)

3.4 OTHER COMPONENTS

3.4.1 Specimen Holder

The specimen holder is a non-safety class component. The surveillance specimen holders are welded baskets containing impact and tensile specimen capsules. The holders have brackets that are attached to the inside of the reactor vessel wall and located in the active core beltline region. The radial and azimuthal positions are chosen to expose the specimens to the same environment and the maximum neutron fluxes experienced by the reactor vessel wall. Figure 3.4.1-1 shows the specimen holder.



**Figure 3.4.1-1 Specimen Holder
(Taken from Reference 3-3)**

3.4.2 Reactor Vessel Upper Head Spray Nozzle

The upper head spray nozzle extends downward from the upper head into head region. The nozzle is located on the vertical centerline of the vessel head and serves both as a head spray and vessel vent.

Figure 3.4.2-1 shows the spray nozzle.

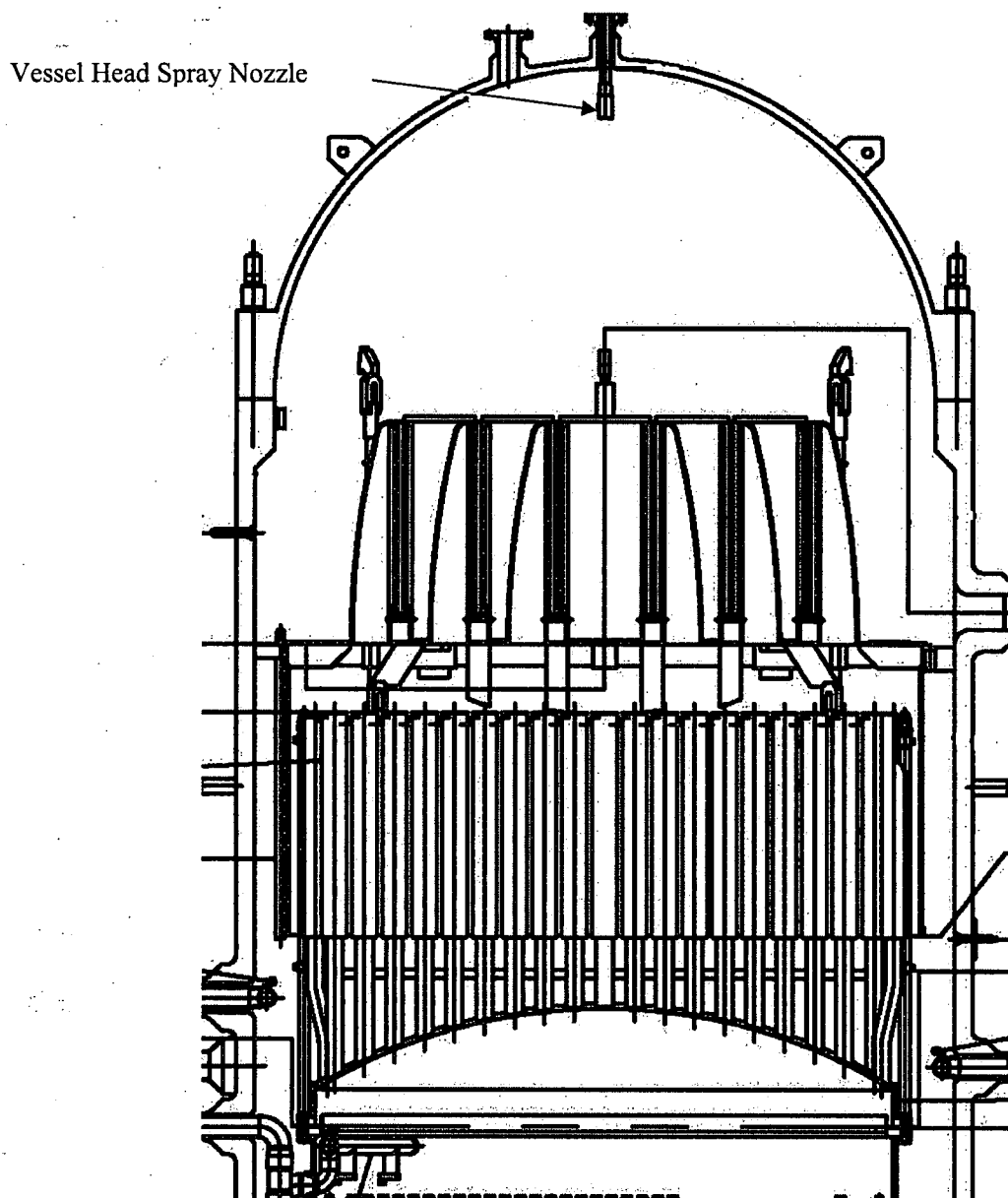


Figure 3.4.2-1 Vessel Head Spray Nozzle and Vent
(Taken from Reference 3-4)

3.5 REFERENCES FOR SECTION 3

- 3-1 Toshiba Document, RS-5126954, Rev. 1, "Prototype ABWR Reactor Internals Flow Induced Vibration Test Report," Figure 3-1. (Proprietary)
- 3-2 Toshiba Document, VCD-2008-000148, Rev. 2, "Outline of ABWR Reactor Components," November 2010.
- 3-3 Toshiba Document, ABWR Design Control Document/Tier 2.
- 3-4 Toshiba Document, 3B11-D001-3005-01, Rev. 3, "Design Specification: Core Support Structures and Reactor Internals," Attachment B, 05/14/2008. (Proprietary)

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4 NON-DRYER FORCING FUNCTIONS

RG 1.20, Revision 3 (Reference 4-4) requires a vibration and fatigue analysis program. Sections 4 and 5 of this report summarize the predictive analyses performed for the non-dryer reactor internals and satisfy the guidance provided in RG 1.20, Revision 3, Section 2.1. The non-dryer forcing functions are determined by developing bounding loads for the analysis cases and are summarized in this section. The modal and structural analysis results are summarized in Section 5 for the non-dryer components.

4.1 ANALYSIS CASES

The non-dryer forcing functions are determined based on the structural analysis cases. Details of the determination of the forcing functions are documented in Reference 4-1 and are summarized in the following sections.

4.1.1 Selection of Analysis Cases for Structural Analysis

The test configuration for the preoperational and initial startup test conditions were examined to select analysis cases for evaluating the design structural responses for the non-dryer reactor internal components. The selected analysis cases are listed in Table 4.1.1-1.

Table 4.1.1-1 Selected Analysis Cases for STP Unit 3

a,c

4.1.2 Basis for Selections of Analysis Cases

Three bounding analysis cases were selected from Table 1.3.1-1 for use in calculating the FIV forcing functions acting on the reactor internal components (Table 4.1.1-1). The basis for the selection of the bounding analysis cases is described below.

Analysis Case 1: Zero Power Operation with Seven Pumps Operating

Analysis Case 1 involves the maximum core flow rate ($[]^{a,c}$ core flow) with seven operational pumps. For conservatism in the analysis, the three non-operating RIPs are assumed to be located adjacent to each other (i.e. RIPs B, C, and D). (In reality, three adjacent non-operating RIPs cannot occur in the plant due to wiring arrangements. The tests will actually be conducted with RIPs C, G, K or B, E, H non-operating). As a result of the assumption in the analysis case, the flow distribution in the lower portion of the downcomer and in the lower plenum regions will be asymmetric in the most adverse manner. In addition, there will be reverse flow through the non-operating pumps, which will cause local flow recirculation regions above and below the pump deck. This case will maximize flow-induced loads on components in the lower portion of the downcomer (RIP guide rails) and in the lower plenum (CP and RIP DP lines). This case may also produce maximum crossflows on the CRGT and possibly on the CRDH.

Analysis Case 1 is an extreme case since the power supply arrangement to the RIPs makes it impossible for three adjacent RIPs to be non-operational during normal operation. This case is bounding for real occurrences, in which either two adjacent RIPs or three equally distributed non-operating RIPs may occur.

Analysis Case 4: Full Power Operation at Maximum Flow Rate

Analysis Case 4 involves the maximum flow rate $[]^{a,c}$ at full power (100%) with ten RIPs running. As such, the flow distributions within the reactor vessel will be mirror symmetric on the vessel $x=0$ and $y=0$ axes. The exception to this statement is the lower plenum region. Here, the use of the in-core monitor assembly produces asymmetrical flow distributions relative to the major x and y axes of the reactor vessel. Maximum coolant dynamic pressures will be present in the reactor core and in regions above the core (top guide and separators).

Analysis Case 4 is bounding because the $[]^{a,c}$ flow rate is the maximum achievable at the 100% power level. The analyses for the internal components, except for the CRDH/CRGTs, ICGT/ICMHs, and stabilizers were done at a more conservative core flow rate of $[]^{a,c}$.

Analysis Case 4': Zero Power Operation at Maximum Flow Rate

Analysis Case 4' is a variant of Analysis Case 4 which involves the maximum flow rate $[]^{a,c}$ at zero power and at relatively low coolant temperatures $[]^c$. Ten RIPs are considered to be operating. This case will produce symmetric flow distributions within the reactor vessel. The combination of high flow rate and low coolant temperatures will produce maximum coolant dynamic pressures in the downcomer and lower plenum regions for the symmetric flow situation.

Analysis Case 4' is bounding because in actual operation, the reactor is limited at zero power to flow rates of approximately []^{a,c} of core flow.

4.1.3 Biases and Uncertainties

The biases and uncertainties that affect the global uncertainties for the forcing functions are the measurement uncertainties associated with the reactor operating parameters and are summarized below. Details of the biases and uncertainties are documented in Reference 4-1.

The estimated measurement uncertainties on key operating parameters for STP Units 3 and 4, from Reference 4-1, are:

1. Pressure: []° (1-sigma)
2. Core inlet temperature: []° (1-sigma)
3. Power level: []° (1-sigma)
4. RIP rotational speed: []° (1-sigma)
5. FW flow rate: []° (1-sigma)
6. Steam flow rate: []° (1-sigma)
7. Core flow rate: []° (1-sigma)

In addition to the measurement uncertainties summarized above, there are also conservative biases used in developing the individual forcing functions. These biases are described for each of the forcing functions.

4.2 LOCAL VELOCITY DETERMINATION

Velocities are determined for the analysis cases due to (1) flow in the downcomer region, (2) flow in the lower plenum, and (3) flow in the top guide, shroud head, and separators. Due to complexity in the flow geometry, the lower plenum region velocities were determined using CFD. The velocities in the other regions (downcomers, top guides, shroud heads, and steam separators) were determined using fluid mechanics equations.

4.2.1 Downcomer Region

The axial coolant velocity in the downcomer region was determined and accounted for the variations in flow area. The downcomer velocities were determined for the analysis cases and details are documented in Reference 4-1.

4.2.2 Lower Plenum Region Velocities

A CFD representation of the lower plenum region is used to calculate the three-dimensional velocity distribution in the lower plenum region. Three analysis cases were evaluated to determine the velocity in the lower plenum. Velocity distributions for the CRGT/CRDH assemblies account for the turbulent fluid force that is non-uniformly distributed over the longitudinal length. Additionally, the flow distributions are determined for the ICGT/ICMH assemblies, stabilizers, CP DP lines, and RIP DP lines. Details are documented in Reference 4-2.

4.2.3 Top Guide, Shroud Head, and Separators

Axial Coolant Velocities

The axial velocities in the top guide, shroud head, and separators are relatively low because of the large cross-sectional flow areas in these components. Details are documented in Reference 4-1.

Crossflow Velocities

Crossflow velocities are significant in the stand pipe region of the separators where the returning flow from the separators crosses the stand pipes as it moves toward the downcomer region. Crossflow velocities are not significant in the top guide and shroud head regions because the flow is generally directed upwards towards the entrance of the separators. Details are documented in Reference 4-1.

4.2.4 Bias Errors and Uncertainties

4.2.4.1 Biases and Uncertainties on Downcomer Velocities

In this section, factors that have bias error or uncertainty for the downcomer velocities are listed and discussions are provided to describe on the effect on the axial flow velocities in the downcomer.

- Bias error

- Flow areas

The downcomer flow areas for Analysis Case 1 (seven operating RIPs) were reduced by []° to increase the calculated downcomer velocities to account for a possible non-uniform velocity distribution for this operating configuration. The resulting increase in velocity is considered conservative, particularly for upper regions of the downcomer which are located far upstream of the effects of reverse flow through the non-operating RIPs.

- Uncertainty

- Volumetric flow rate

The global uncertainty on measurement uncertainty is discussed in detail in Reference 4-1. While there are sizable measurement uncertainties on flow rate measurement, there is significant margin between the flow rates chosen for the three analysis cases to bound the measurement uncertainty on flow rate. This is discussed in more detail in Reference 4-1.

4.2.4.2 Biases and Uncertainties on Lower Plenum Velocities

Factors which may introduce a bias error or uncertainty into the CFD analyses are listed, and discussions are provided to describe the effect of the crossflow velocity on the lower plenum components.

- Bias error

- Numerical diffusion

The effect from numerical diffusion, i.e. discretization error, on the simulated maximum crossflow velocity at the shroud support leg opening is evaluated by a grid-spacing sensitivity study. As a result, the discretization error of the present analysis is estimated at []°, which is considered sufficiently small.

- Uncertainty

- Volumetric core flow rate

Since the uncertainty on volumetric core flow rate is small enough that it will not impact the order of Reynolds numbers, the flow fields can be considered similar within the

uncertainty range of volumetric core flow rate. In this situation, local velocity is almost proportional to volumetric core flow rate and the uncertainty percentage of the crossflow velocity on the lower plenum components is almost the same as the uncertainty percentage of volumetric core flow rate.

- Fluid properties (density and viscosity)

The uncertainties of the density and the viscosity of compressed water are mostly dependent on the uncertainty of temperature assumed. Since the uncertainty of temperature is small enough to make no difference to the order of the Reynolds number, the effect from fluid properties on the simulated crossflow velocity to the lower plenum components can be ignored.

- Turbulence quantities specified as inlet boundary conditions

The effect from turbulence quantities specified as inlet boundary conditions on the simulated maximum crossflow velocity at the shroud support leg opening is evaluated by a sensitivity study. As a result, the uncertainty of analysis results from the uncertainty of turbulence quantities specified at the inlet is evaluated at []%, which is considered sufficiently small.

4.2.4.3 Biases and Uncertainties on Velocities in the Top Guide, Shroud Head, and Separators

Factors which may introduce a bias error or uncertainty into the velocity analyses and the resulting effect on the axial flow velocities for the components located above the core are discussed below.

- Bias error
 - Flow areas

The flow areas for Analysis Case 1 (7 operating RIPs) were reduced by []% to increase the calculated downcomer velocities to account for a possible non-uniform velocity distribution for this operating configuration. The resulting increase in velocity is believed to be conservative, particularly for upper regions of the downcomer located far upstream of the effects of reverse flow through the non-operating RIPs.

- Uncertainty
 - Volumetric flow rate

While there are sizable measurement uncertainties on flow rate measurement, there is significant margin between the flow rates chosen for the three analysis cases to bound the measurement uncertainty on flow rate.

4.3 FORCING FUNCTIONS - METHODOLOGY

The reactor internals components are subjected to several categories of flow-related loads:

- Narrowband vortex shedding (crossflow)
- Broadband turbulence (crossflow or turbulent source)
- Pump-induced acoustic pulsations

The determination of vortex shedding and turbulent loads is summarized below. Pump-induced pulsation loads are defined in Reference 4-1.

4.3.1 Vortex Shedding and Turbulence Buffeting Loads

One of two phenomena occurs for crossflow vibrations: vortex shedding or random turbulence. The phenomena appropriate for a given application depend on the Reynolds number as summarized below.

For vortex shedding or turbulence excitation by crossflow, the alternating lift coefficient, alternating drag coefficient and Strouhal number are functions of the Reynolds number. Vortex shedding occurs when the Reynolds number is in the ranges of $0 - 2 \times 10^5$ and $4 \times 10^6 - 1 \times 10^8$. Turbulence occurs when the Reynolds number is in the range of $2 \times 10^5 - 4 \times 10^6$. Details of the derivation of the vortex shedding and turbulent buffeting loads are documented in Reference 4-1.

4.3.2 Pump-Induced Pulsation Loads

Pump-induced pressure pulsations at the one-per-revolution and vane-passing frequencies have been known to cause significant loads on plant components when they occur at acoustical modes of the system. To provide a continuous internal circulation path for the core coolant flow, STP Unit 3 has ten RIPs circumferentially distributed around the entrance to the vessel inlet plenum. A method for predicting the pump-induced pulsation loads using the ACSTIC2 acoustic harmonic analysis computer program is described in Reference 4-3. ACSTIC2 was used to model the STP Unit 3 fluid systems and to evaluate acoustical loads acting on reactor internal components. Details of the pump-induced pulsation loads are documented in Reference 4-1.

4.4 FORCING FUNCTIONS

Forcing functions are determined for each of the reactor internal components based on size and location within the reactor pressure vessel and for each analysis case. The forcing functions are divided into four main categories:

- Large components in the downcomer region,
- Small components in the downcomer region,
- Components in the lower plenum, and
- Components above the core.

Details of the forcing functions are documented in Reference 4-1.

4.4.1 Large Components in the Downcomer Region

The large components located in the downcomer region include the top guide shell, core shroud, and the shroud support. The forcing functions for these components are turbulent spectra for each analysis case.

4.4.2 Small Components in the Downcomer Region

The small components located in the downcomer region include the specimen holder, RIP guide rails, shroud head lifting rods, shroud head bolts, feedwater spargers, LPFL spargers, and HPCF spargers. The specimen holder is subjected to axial turbulent buffeting loads that are defined as turbulent spectra. The RIP guide rails are subjected to crossflow turbulent buffeting in the downcomer region are defined as turbulent spectra. The lifting rods are subjected to both axial and crossflow turbulent buffeting that are defined as velocities. The shroud head bolts are subjected to both axial flow and crossflow turbulent buffeting that are defined as turbulent spectra. The feedwater, LPFL and HPCF spargers are subjected to axial turbulent buffeting that are defined as turbulent spectra.

4.4.3 Components in the Lower Plenum

The components located in the lower plenum include the CRDH/CRGTs, the ICMH/ICGTs, stabilizers, CP DP lines, and RIP DP lines. The forcing functions for the CRDH/CRGTs, ICMH/ICGTs, and the stabilizers were calculated using the turbulent buffeting relationships for crossflow. The forcing functions for the CP DP and RIP DP lines were calculated using the turbulent buffeting relationships for both axial flow and crossflow.

4.4.4 Components Above the Core

The non-dryer components located above the core are the top guide, shroud head, steam separators, and HPCF sparger. The forcing functions for these components were calculated using turbulent buffeting relationships for axial flow. In addition, turbulent buffeting relationships were used to calculate crossflow forcing functions for the stand pipe region of the separators where returning flow from the separators

cross the stand pipes as flow moves towards the downcomer region. Crossflow is not significant in the top guide and shroud head regions because the flow is generally directed upwards towards the entrance of the separators.

4.4.5 Pump Pulsation Loads

Analysis results were obtained for Analysis Cases 1, 4, and 4' for pump frequencies representing the pump rotational speed, twice the pump rotational speed, and five times the pump rotational speed. These pump pulsation loads were developed in two ways:

1. As a pressure differential amplitude across a component (e.g., core shroud wall and beam mode pressure difference)
2. As a pressure gradient amplitude (pressure difference per unit length)

The pressure difference approach is used for large components for which the stream-wise distance across the component is significant, such as for the two sides of the core shroud wall. The pressure gradient amplitude approach is used for small components, such as the CP/DP line, in which the stream-wise dimension of the component is small compared to the acoustic wavelength. In these cases, the pressure gradient amplitude must be multiplied by the dimension of the component (e.g., diameter) to obtain the pressure difference amplitude.

Pump-induced acoustic loads acting on the reactor internals were calculated for STP Unit 3 and are documented in detail in Reference 4-1. These loads were calculated for three operating conditions and three pump frequencies (the rotational speed, twice the rotational speed, and five times the rotational speed). The following observations can be drawn from the analysis are:

1. Pump flow-induced vibration analyses, forcing function and analysis results documented in detail in Reference 4-1. The results are consistent with previous reactor internals FIV analyses performed by Westinghouse.
2. The maximum pressure differential is approximately []° across the core shroud wall at the upper middle of the core shroud. The maximum differential pressure in the annular downcomer section across the shroud is []°.

4.4.5.1 Biases and Uncertainties

This section discusses biases and uncertainties associated with the pump pulsation analysis.

a,c

a,c

4.5 REFERENCES FOR SECTION 4

- 4-1 Westinghouse Report, WCAP-17371-P, Revision 3, "South Texas Project Units 3 and 4 Reactor Internals Non-Dryer Component Flow-Induced Vibration Assessment," May 2012. (Proprietary)
- 4-2 Toshiba Document, 7B11-D001-3809-02, Rev. 2, "Forcing Function Analysis Report for Lower Plenum," January 2011. (Proprietary)
- 4-3 "A Method for Predicting Pump-Induced Acoustic Pressures in Fluid-Handling Systems," American Society of Mechanical Engineers, PVP-Vol. 63 (1982): pg. 167-184.
- 4-4 U.S. NRC Regulatory Guide 1.20, Revision 3, "Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing," March 2007.

5 STRUCTURAL RESPONSES – NON-DRYER COMPONENTS

FEMs were generated using general purpose finite element programs, ANSYS® (Reference 5-1) and ABAQUS (Reference 5-2). Modal and stress analyses were performed for the non-dryer components and are described in Sections 5.1 and 5.2, respectively.

5.1 MODAL ANALYSIS

5.1.1 Methods

FEMs were built for each of the reactor internal components. ABAQUS models were generated for the following reactor internal components. The Block Lanczos method is used to determine the natural frequencies and corresponding mode shapes.

- CRDH and CRGT
- ICGT and ICMH
- HPCF sparger and coupling
- Core shroud

ANSYS® models were generated for the following reactor internal components. The Block Lanczos method is used to determine the natural frequencies and corresponding mode shapes.

- Shroud head bolts
- Shroud head lifting rods
- CP DP lines
- RIP DP lines
- FW spargers
- LPFL spargers
- RIP guide rails
- Shroud head and steam separators
- Lower guide rods

For the remaining internal components listed below, assessments are made to show that the vibration characteristics of these components are acceptable. This includes an assessment of the separation of the

component's natural frequencies versus the forcing frequencies, and the magnitude of the forcing functions.

- Top guide
- Core plate steam vent and spray
- Specimen holders

In addition, the specimen holders have been evaluated for the forcing function frequencies below the first mode, which demonstrates that all stresses due to FIV are negligible. Based on current plant experience, the orificed fuel supports do not need to be evaluated, because FIV will not occur on this component.

In order to validate the FEM developed for the analysis, the results were compared with the RJ-ABWR data (when available). The modal analyses of the following components were compared to RJ-ABWR frequency data (hammer test):

- CRDH and CRGT
- ICGT and ICMH
- HPCF sparger and coupling

5.1.2 Modal Analysis Results

Natural frequencies and mode shapes were calculated and the fundamental frequency results are summarized below for the non-dryer components. Details are documented in Reference 5-3.

Table 5.1.2-1 contains a list of the components analyzed and their lowest predicted natural frequencies.

Table 5.1.2-1 Component Lowest Natural Frequency

c

5.1.3 Biases and Uncertainties

The uncertainties in this analysis are small and in some cases boundary conditions are chosen to give conservative results related to the loads. Details for the biases and uncertainties are documented in Reference 5-3.

5.2 STRESS ANALYSIS

5.2.1 Load Application and Other Inputs

The power spectral density (PSD) spectrum solution is based on model data as well as the modal solution data. The PSD mode combination method is used for the random vibration analysis. This method results in a root mean square (RMS) (one-sigma) stress in the structure. This is the standard deviation of the response, i.e. the expectation is that this stress value will not be exceeded 68.3% of the time.

The ratio of 3.0 between 0-peak values and the achieved RMS values is used for turbulent buffeting based on Reference 5-4.

RG 1.20, Revision 3 (Reference 5-7) suggests that any damping ratio greater than 1% be justified. The damping ratios derived from RJ-ABWR for a CRGT and an ICGT are []%, respectively. Therefore, a damping ratio of []% is used for the analyses of most components (Reference 5-3).

For the HPCF sparger assembly, a damping ratio between []% Therefore, for the FW, LPFL, and HPCF sparger assemblies, a damping ratio of []% is used for the analyses.

For details of the load application for each of the non-dryer reactor internal components, refer to Reference 5-3.

5.2.2 Stress Evaluation

The non-dryer reactor internals were evaluated for vortex shedding, turbulent buffeting and pump induced pulsations loads, as appropriate, as described in Section 4. The stresses calculated from different random vibration loads are combined using the square root of the sum of the squares method (SRSS).

SRSS can be used to combine the responses from different load sources, i.e., vortex shedding, pump pulsation and random turbulences. For certain low stress components, the absolute sum may be conservatively used to reduce the number of iterations.

Details of the stress evaluation are documented in Reference 5-3.

5.2.3 Acceptance Criteria

The requirement for fatigue evaluation is that normal operating conditions shall meet a fatigue limit of 9.95 ksi (68.6 MPa) per Reference 5-5. The fatigue factor is $f = 4$ for welded structures per References 5-3 and 5-6.

5.2.4 Stress Results

Dynamic loading conditions, or flow-induced vibrations, produce cyclical stresses in the components. Reference 5-5 requires that a fatigue analysis be performed to show that the components will not suffer any fatigue damage. A measure of the fatigue margin for a given component is provided by the ratio of the peak alternating stress of 9.95 ksi divided by the endurance limit for the component, or:

$$\text{Fatigue margin} = (\text{Peak Alternating Stress}) / (\text{Endurance Limit of 9.95 ksi})$$

Table 5.2.4-1 presents the fatigue margins for the non-dryer reactor internal components based on bounding operating conditions. Details are provided in Reference 5-3.

Table 5.2.4-1 Design Fatigue Margins for STP 3 Non-Dryer Reactor Internals

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5.2.5 Biases and Uncertainties

The uncertainties in this analysis are small and in some cases boundary conditions are chosen to give conservative results for the loads. Details for the biases and uncertainties are documented in Reference 5-3.

5.3 REFERENCES FOR SECTION 5

- 5-1 ANSYS® Release 10.0 UP20050718.
- 5-2 ABAQUS, Version 6.6-3.
- 5-3 Westinghouse Report, WCAP-17371-P, Revision 3, "South Texas Units 3 and 4 Reactor Internals Non-Dryer Component Flow-Induced Vibration Assessment," May 2012. (Proprietary)
- 5-4 Westinghouse Letter, LTR-RIDA-10-294, Rev. 0, "Recommended Damping Ratio, Peak-to-RMS Ratio, and Stress Concentration Factor Application for ABWR Reactor Internals Flow-Induced Vibration Analysis," 10/28/2010. (Proprietary)
- 5-5 ABWR Design Control Document, Rev. 4.
- 5-6 ASME Boiler and Pressure Vessel Code, Division I, Section III, 1989 Edition, with no Addenda.
- 5-7 U.S. NRC Regulatory Guide 1.20, Rev. 3, "Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing," March 2007.
- 5-8 XGEN Report, XGEN-2010-03R2, "FIV Evaluation of STP Unit 3/4 Reactor Internals: Top Guide, Core Plate, Vent and Head Spray, Surveillance Specimen Holder and Fuel Support Casting," January 2011. (Proprietary)

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6 STEAM DRYER QUALIFICATION

The main function of the steam dryer in a Boiling Water Reactor (BWR) is to remove moisture from the steam in order to minimize erosion of piping and the turbine. The dryer is not safety-related and is not an ASME Code component. Although there have been some differences in the design configuration of the steam dryer with evolution of BWR types (BWR/2-6) and the ABWR, the basic features of the dryer have remained essentially the same.

The steam dryer assembly is mounted in the reactor vessel above the separator assembly and forms the top and sides of the wet steam plenum. During refueling operations, the dryer is removed to allow access to the reactor core. Vertical guide rods on the inside of the vessel provide alignment for the dryer assembly during re-installation. The dryer assembly is supported by brackets extending inward from the vessel wall and is held down in position during operation by the vessel head hold down brackets. Steam from the separators flows upward and outward through the dryer's drying vanes. These vanes are attached to a top and bottom supporting member forming a rigid integral unit. Moisture is removed and carried by a system of troughs and drains to the pool surrounding the separators and then into the recirculation downcomer annulus between the core shroud and reactor vessel wall.

RG 1.20 provides guidance for the CVAP for nuclear power plants during preoperational and initial startup testing. The program is intended to demonstrate that the reactor internals are adequately designed to withstand FIV forces at normal and transient plant operating conditions for the design life of the plant. The ABWR was designed and certified under RG 1.20 Revision 2. This same design is employed in multiple ABWR plants in Japan. One of those Japanese ABWRs, as described in WCAP-17269-P and referred to as the reference Japanese ABWR (RJ-ABWR), commenced commercial operation in 1996 after going through extensive preoperational and start-up testing to confirm that the reactor internals are adequately designed to withstand FIV loads. The RJ-ABWR steam dryer has an excellent operating history as demonstrated by tests and inspections. The latest revision (Revision 3, March 2007) of RG 1.20 contains additional requirements based on recent BWR experiences on steam dryers. Based on the need to address the current guidance, STPNOC (now NINA) decided that STP Unit 3 reactor internals will be designated as the U.S. ABWR prototype.

6.1 APPROACH - OVERVIEW

As stated above, the steam dryer design in the ABWR certified design was developed to satisfy the guidance of RG 1.20 Revision 2. The Final Safety Evaluation Report (FSER) for the ABWR (NUREG-1503) states that the response of the dryer to FIV must be predicted before final design approval, discusses the analyses that were done by the Design Certification (DC) applicant which were reviewed by the NRC staff, and concludes that the combination of predictive analysis, pre-test inspections, tests, and post-test inspections assures that the reactor internals will perform without loss of structural integrity. The DCD specifies a peak stress amplitude limit that is significantly lower than the ASME Code allowable value (the ASME allowable is approximately 37% higher than the DCD allowable value). Thus the certified dryer design has a predictive analysis that is adequate for this lower allowable stress. This has been confirmed by pre-operational testing at the RJ-ABWR, which confirmed the peak and alternating stresses in the steam dryer meet the allowable limit, and that the maximum stresses are less than this conservative allowable limit.

RG 1.20 Revision 3 Part D – Implementation, states, “Except in those cases in which a licensee proposes or has previously established an acceptable alternative method for complying with specified portions of the NRC’s regulations, the NRC staff will use the methods described in this guide...”. Because the certified ABWR steam dryer design was not designed specifically to meet the guidance of RG 1.20 Revision 3 (as portions of this guidance were not available at the time), but was designed to satisfy the earlier guidance of RG 1.20 Revision 2, the approach to show the structural acceptability of the dryer is to use a combination of the previously established qualification of the dryer along with a proposed alternative to provide a means to demonstrate that the reactor internals will perform without loss of structural integrity, as discussed in the FSER. The alternative approach consists of the activities as described in the following paragraphs:

Operating Experience

The STP dryers are essentially identical to the RJ-ABWR and []^{a,c}, herein referred to as the J-ABWR. The RJ-ABWR has operated for over twelve years and the J-ABWR has operated since 2005. The operating ABWRs have dryers that are essentially identical in both configuration and plant operating conditions. The dryers for both operating plants have been inspected and no indications have been found. In addition, the ABWR dryer incorporates improvements in the dryer design that results in greater structural capability and better performance than earlier dryer designs.

Design Modifications to Avoid Acoustic Resonance

[]^{a,c} was performed to determine if an acoustic resonance would occur for power levels up to 100%. []

[]^{a,c} eliminated the acoustic resonance.

Dryer Instrumentation During Startup

The STP Unit 3 dryer instrumentation uses an []

[]^{a,c} are used to determine the sensor types and locations.

Startup Test – Power Ascension Plan

Additionally, the startup testing will include an initial hold point at 60% power to collect steam dryer data and generate limit curves to ensure that the steam dryer stresses are below the fatigue limit. Thereafter,

for every []^{a,c} up to 100% power, the plant will hold power to collect steam dryer data and generate limit curves.

Thus, the[]

the structural adequacy of the STP steam dryer due to FIV loads.]^{a,c} will ensure

6.2 OPERATING EXPERIENCE

The ABWR is an evolutionary design that builds on operating BWR experience and includes significant design improvements in the dryer design.

Historically, there have been instances of minor cracking in BWR steam dryers, with the exception of the more significant Quad Cities EPU-related dryer cracking. A variety of options, (e.g., hole drilling to stop crack growth, increasing weld size and replacement of plates with thicker plates), have been implemented successfully as design fixes. Cracking in the dryer would have to be very significant to cause structural integrity concerns. Because dryers are visually inspected after removal in outages, significant cracking and associated loose parts due to cracking during the subsequent cycle are unlikely (the experience at the two Quad Cities units was an exception because of the Electromatic Relief Valve (ERV) stand pipe configuration which caused acoustic resonance and subsequent dryer cracks when EPU conditions were applied). When cracking has occurred as a result of plant operation, design fixes were implemented successfully and subsequent designs (such as the RJ-ABWR) incorporated the design fix ideas applied to the previous dryer designs. The successful operating experience with the RJ-ABWR and J-ABWR dryers (which are essentially identical to the STP dryers) shows that the design improvements based on the fixes for previous steam dryer problems have been effective. Reference 6-1 documents the successful operating experience and results of inspections of two operating ABWRs. Table 6.2-1 shows a summary of the improvements to the ABWR dryers.

The ABWR dryer operating experience is documented in Reference 6-3 and describes the operating experience as well as the inspection results for two operating ABWRs. The STP dryers are essentially identical in geometry (configuration) and plant operating parameters as the RJ-ABWR and J-ABWR. RJ-ABWR commenced commercial operation in 1996 and J-ABWR commenced commercial operation in January 2005. Table 6.2-2 tabulates the significant dimensions of the STP dryers as compared to RJ-ABWR and J-ABWR. Table 6.2-3 compares the plant operating parameters between STP, RJ-ABWR, and J-ABWR and shows that the operating conditions for STP are essentially identical to those ABWRs with successful operation.

In 2007, RJ-ABWR experienced an earthquake that exceeded the original seismic design and the plant was shutdown for inspections and evaluations. The equivalent of an ASME VT-3 inspection was performed on the RJ-ABWR dryer and no indications were found. In addition, the equivalent of an ASME VT-1 inspection was performed on the J-ABWR dryer and no indications were found. Details of the inspections are contained in Reference 6-3.

Since the design and operation of the STP dryers are essentially identical to the RJ-ABWR and J-ABWR, the performance of the STP dryers are expected to be the same.

Table 6.2-1 [

] ^{a,c}

Cracking Attribute				a,b,c
Component	Support Ring			
	Drain Channel			
	Dryer Hood			
	Tie Bar			
	Skirt			
	Lifting Rod			
Cracking Mechanism	IGSCC			
	Fatigue			
	Crevice cracking			
	Acoustic Resonance			
Regulatory Codes	ASME			
	BWRVIP			

Table 6.2-2 [

] ^{a,c}

a,b,c

Table 6.2-3 [

]^{a,c}

^{a,b,c}

6.3 ELIMINATION OF ACOUSTIC RESONANCE

Startup data for the RJ-ABWR shows [

] ^{a,c}

a,c

6.3.1 [] ^{a,c}

a,c

Figure 6.3-1 [] ^{a,c}

6.3.2 []^{a,c}

a,c

Conclusion

The modification to []^{a,c} will eliminate acoustic resonance for STP Units 3 and 4.

a,c

Figure 6.3.2-1 []^{a,c}

6.4 DRYER STARTUP INSTRUMENTATION

In order to determine the[

] ^{a,c}

a,c

6.4.1 Finite Element Model

Figure 6.4.1-1 Steam Dryer FEM

6.4.1.1 Modal Analysis

a,c

Table 6.4.1-1 Comparison of RJ-ABWR Hammer Test and STP Modal Analysis Results

a,c

6.4.2 Instrumentation Locations

6.4.2.1 Pressure Transducers

a,c

a,c

Table 6.4.2.1-1 PT Locations

a,c

a,c

Figure 6.4.2.1-1 PT Locations []^{a,c}

a,c



a,c

Figure 6.4.2.1-2 PT Locations [

]^{a,c}



a,c

6.4.2.2 Strain Gages

The strain gage locations and orientations are selected by [

]^{a,c}

[

]^{a,c}

Table 6.4.2.2-1 Strain Gage Locations

a,c



Figure 6.4.2.2-1 Strain Gage Locations []^{a,c}

[]^{a,c}



Figure 6.4.2.2-2 Strain Gage Locations []^{a,c}

[]^{a,c}

6.4.2.3 Accelerometer**Table 6.4.2.3-1 Accelerometer Locations****Figure 6.4.2.3-1 Accelerometer Locations**

6.5 STRUCTURAL EVALUATION USING RJ-ABWR STARTUP DATA

While the primary basis for qualification of the STP steam dryer is operating experience, a structural evaluation of the STP steam dryer was performed using RJ-ABWR startup data. The results of this structural evaluation showed that the dryer stresses were below the endurance limit of 9.95 ksi (68.6 MPa). The calculated stress was 8948 psi which corresponds to a stress ratio, $SR = 1.11$. Details of the evaluation are contained in Reference 6-2, Section 5.5.

6.6 STEAM DRYER POWER ASCENSION PLAN

The power ascension plan for the steam dryer will include an initial hold point at 60% power to collect steam dryer data and generate limit curves to ensure that the steam dryer stresses are below the fatigue limit. At 60% power, the flow is [

^{a,c} Thereafter, for every [^{a,c}

To ensure that the dryer stresses are within acceptable limits, the following process will be used during initial startup power ascension:

a,c

a,c

6.6.1 Biases and Uncertainties

a,c

Table 6.5.1-1 Bias and Uncertainty for |**|^{a,c}**

a,c

a,c

6.7 CONCLUSIONS

The structural adequacy of the STP Unit 3 dryer is primarily based on the successful operating experience of ABWRs that have been operating since 1996 in Japan. The approach to qualify the STP Unit 3 dryer using the guidance of RG 1.20, Revision 3 (Reference 6-1) is based on the following.

- Operating Experience

- [

]^{a,c}

- Design Modifications []^{a,c}
[

]^{a,c}

- Instrumentation During Startup
[

]^{a,c}

- Structural Evaluation Using RJ-ABWR Startup Data

A structural evaluation of the STP Unit 3 steam dryer was performed using RJ-ABWR startup data and the calculated stresses were below the endurance limit.

In summary it can be stated that:

- The ABWR design is an evolutionary design that has incorporated design changes and improvements based upon operational experience of previous generations of BWR designs.

- RJ-ABWR has gone through an extensive preoperational and power testing and a thorough inspection program during startup without showing any damage to the dryer.
- RJ-ABWR and J-ABWR have accumulated over 12 years of operation without any indication of damage to the dryers as evidenced by the inspections performed the details (described in Reference 6-3).
- RJ-ABWR and J-ABWR have never undergone any repairs to the dryers.
- STP Units 3 and 4 dryers are essentially identical to RJ-ABWR and J-ABWR in both structural configuration as well as operational parameters.
- Acoustic resonance has been eliminated due []^{a,c}
- The STP Unit 3 dryer will undergo a CVAP that involves a power ascension plan to hold the power level at 60% to collect []^{a,c} to allow development of limit curves that would be utilized to go up to 100% power level in discrete steps.
- A structural evaluation of the STP Unit 3 steam dryer was performed using RJ-ABWR startup data and the calculated stresses were below the endurance limit.

This report concludes that based upon discussion above, STP Unit 3 dryer would meet the cyclic stress requirements and perform without degradation during power operation.

6.8 REFERENCES FOR SECTION 6

- 6-1 US NRC Regulatory Guide 1.20, Revision 3, "Comprehensive Vibration Assessment Program for Reactor Internals during Preoperational and Initial Startup Testing," March 2007.
- 6-2 Westinghouse Report, WCAP-17385-P, Revision 3, "STP Unit 3 Steam Dryer Flow-Induced Vibration Assessment," May 2012. (Proprietary)
- 6-3 Westinghouse Report, WCAP-17369-P, "ABWR Dryer Operating Experience for STP Units 3 and 4," February 2011. (Proprietary)

7 ACCEPTANCE CRITERIA FOR CVAP TEST DATA

7.1 BASES FOR ACCEPTANCE CRITERIA

Details of the sensor types and locations are provided in Reference 7-1. Acceptance criteria will be developed for each of the sensors to be used during the CVAP. The categories of instrumentation include:

- Strain gages to measure local strain values for eventual use in determining stress values,
- Pressure transducers to measure deterministic pump pulsations and turbulence levels for comparison with predicted values,
- Accelerometers to measure local component accelerations for eventual use in determining component displacements,
- Displacement transducers for direct measurement of relative motion of components.

The acceptance criteria will focus exclusively on the dynamic responses of the instrumentation related to flow-induced vibration. Acceptance criteria will account for transducer locations, operating conditions for each test, uncertainties and biases, and margins to be added for conservatism to ensure that the allowable fatigue stress (9.95 ksi) will not be exceeded. Redundancy of the sensors is considered in the location and number of instruments.

Tables of maximum allowable test values will be generated for the sensors for use in the detailed test procedures. These maximum allowable test values will provide guidance for the test operators when they are conducting the CVAP tests. This information will allow the operators to determine the margins between the sensor values being measured and the maximum allowable test values as the tests are progressing.

7.2 REFERENCES FOR SECTION 7

- 7-1 Westinghouse Report, WCAP-17370-P, Revision 3, "South Texas Project Unit 3 Comprehensive Vibration Assessment Program, Measurement, Test and Inspection Plan," May 2012.
(Proprietary)

8 EVALUATION PLAN

To support the CVAP tests, a best estimate predictive analysis will be performed. In this analysis, the best estimate values of component stresses will be determined using best estimate values of the expected CVAP operating conditions and best estimate forcing functions. The best estimate stress values will be used as a basis for comparison with the measured values for evaluating the accuracy of the FIV predictive methods.

8.1 ADJUSTMENTS BETWEEN MEASURED AND PEAK STRESS LOCATIONS

Peak calculated stresses are determined for each component, as well as the distribution of stresses throughout the component. Due to practical considerations, such as situations where peak stresses occur at welds or at threaded joints where strain gages cannot be positioned, difficulties in accessing a desired peak stress location, or difficulties in routing of instrument cables, CVAP instrumentation may have to be located in non-peak stress positions. In these cases, adjustments will be made to scale the measured strains and resulting stresses to the peak stresses for the component. These adjustments will be factored into the determination of the acceptance criteria for the CVAP instruments.

8.2 ADJUSTMENTS BETWEEN MEASURED AND CALCULATED OPERATING CONDITIONS

Small differences may exist between the actual measured CVAP test conditions and the calculated test conditions for the CVAP test point that were used to define the analytical forcing functions and resulting calculated stresses. An allowance will be determined which will cover possible variations between measured and calculated operating conditions. The effect of this allowance will be taken into account when establishing the acceptance criteria for the CVAP instruments.

8.3 ADJUSTMENTS DUE TO EFFECTS OF BIASES AND UNCERTAINTIES

There will be biases and measurement uncertainties associated with determining the CVAP operating conditions and the measured vibration parameter values during the CVAP tests. These biases and uncertainties will be taken into consideration when establishing the acceptance criteria for the CVAP instruments.

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9 CONCLUSIONS

Westinghouse/Toshiba has analyzed the STP Units 3 and 4 reactor internal components.

Non dryer components are analyzed under bounding conditions to evaluate the expected flow-induced vibration levels during test and normal operating conditions.

The structural acceptability of the dryer is primarily based on the successful operating experience of ABWRs with the elimination of acoustic resonance and structural evaluation of the dryer using RJ-ABWR startup data. The evaluations and assessment show that the vibration levels of the reactor internal components are acceptable and that the STP Units 3 and 4 internals design is adequate.

The recommendations of NRC Regulatory Guide 1.20, Revision 3 will be satisfied by conducting confirmatory preoperational and startup measurement tests as well as surface inspections of the internal components before and after operation. Such a program constitutes the CVAP. CVAP tests and inspections are planned for STP Unit 3 and are outlined in Sections 1.2 and 1.3.

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