Westinghouse Non-Proprietary Class 3

WCAP-16687-NP
Revision 1

March 2007

AP1000 Reactor Internals Expected and Acceptable Responses During Preoperational Vibration Measurement Program

Westinghouse

WCAP-16687-NP
Revision 1

AP1000 Reactor Internals Expected and Acceptable Responses During Preoperational Vibration Measurement Program

Authors:

D. R. Bhandari
B. F. Good
G. M. Imbrogno
J. E. Kubes
Y. K. Law
R. E. Schwirian
N. R. Singleton*
M. D. Snyder
J. Yang
C. Yu

Verifier:

Y. K. Law*
Advanced Reactors Internals Design and Analysis

March 2007

Reviewer: David A. Altman*
Advanced Reactors Internals Design and Analysis

Approved: D. R. Forsyth*, Manager
Advanced Reactors Internals Design and Analysis

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

Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

© 2007 Westinghouse Electric Company LLC
All Rights Reserved

RECORD OF REVISIONS

Revision	Section	Description of Change
0	All	Initial Issue
1	All	Revisions made to reflect NuStart Energy and Westinghouse reviewers' comments.

Trademark Note:

AP1000 is a trademark of Westinghouse Electric Company LLC.

TABLE OF CONTENTS

LIST OF TABLES iv

LIST OF FIGURES iv

LIST OF ACRONYMS v

1 INTRODUCTION 1-1
- 1.1 BACKGROUND 1-1
- 1.2 DCD MARKUP 1-1

2 CALCULATIONS AND BASES FOR DEVELOPMENT OF EXPECTED AND ACCEPTABLE RESPONSES 2-1
- 2.1 CALCULATIONS FOR EXPECTED RESPONSES FOR THE INTERNALS 2-1
  - 2.1.1 Hot Functional Test Flow Rate with Four Reactor Coolant Pumps in Operation 2-1
  - 2.1.2 Effect of Absence of the Core on Downcomer Modal Properties 2-1
  - 2.1.3 Natural Frequencies of Guide Tube, Upper Mounted Instrumentation, and Lower Plenum Structures 2-2
- 2.2 EXPECTED RESPONSES AND ACCEPTABLE RESPONSES 2-10

3 PRE- AND POST-HOT FUNCTIONAL TEST INSPECTION PROGRAM 3-1

4 REGULATORY IMPACT 4-1

5 REFERENCES 5-1

LIST OF TABLES

Table 2.1.2-1 System Model with Core, with Water and No Contact at the Radial Keys 2-3

Table 2.1.2-2 System Model without Core, with Water and No Contact at the Radial Keys..... 2-5

Table 2.1.2-3 Comparison of Common HFP Condition and HFT Condition Modes 2-7

Table 2.1.4-1 Upper Mounted Instrumentation Assembly Natural Frequencies Upper Guide Tube – UMIA Natural Frequencies 2-8

Table 2.1.4-2 Upper (With No UMIA Attached) and Lower Guide Tube Natural Frequencies..... 2-8

Table 2.1.4-3 Upper Support Column Natural Frequencies..... 2-8

Table 2.1.4-4 Secondary Core Support and Vortex Suppression Plate Assembly Natural Frequencies 2-9

Table 2.1.4-5 Core Shroud Natural Frequencies 2-9

Table 2.2-1 Upper Support Skirt Strain Gage Responses at Full HFT Temperature 2-10

Table 2.2-2 Lower Guide Tube Strain Gage Responses at Full HFT Temperature 2-10

Table 2.2-3 Upper Guide Tube Strain Gage Responses at Full HFT Temperature..... 2-11

Table 2.2-4 Upper Support Column Strain Gage Responses at Full HFT Temperature 2-11

Table 2.2-5 UMIA Guide Column Strain Gage Responses at Full HFT Temperature..... 2-11

Table 2.2-6 Core Barrel Flange Strain Gage Responses at Full HFT Temperature..... 2-12

Table 2.2-7 Core Barrel Accelerometer Responses at Full HFT Temperature..... 2-12

Table 2.2-8 Lower Core Support Plate Accelerometer Responses at Full HFT Temperature..... 2-13

Table 2.2-9 Vortex Suppression Plate Strain Gage Responses at Full HFT Temperature..... 2-13

Table 2.2-10 Core Shroud Accelerometer Responses at Full HFT Temperature..... 2-14

Table 2.2-11 Core Shroud Responses Relative to Core Barrel at Full HFT Temperature 2-14

Table 3-1 Pre- and Post-Hot Functional Test Examination 3-1

LIST OF FIGURES

Figure 3-1 Pre- and Post-Hot Functional Test Examination Locations 3-3

LIST OF ACRONYMS

CB	Core Barrel
CS	Core Shroud
COL	Combined Operation License
CRDM	Control Rod Drive Mechanism
DCD	Design Control Document
FA	Fuel Assembly
FSER	Final Safety Evaluation Report
GT	Guide Tube
HFP	Hot Full Power
HFT	Hot Functional Test
Hz	Hertz (cycles per second)
LCSP	Lower Core Support Plate
LI	Lower Internals
MDF	Mechanical Design Flow
RCP	Reactor Coolant Pump
RMS	Root Mean Squared
RV	Reactor Vessel
UI	Upper Internals
UMIA	Upper Mounted Instrumentation Assembly
USNRC	United States Nuclear Regulatory Commission
USC	Upper Support Column
USP	Upper Support Plate

1 INTRODUCTION

1.1 BACKGROUND

The first unit of the AP1000™ reactor design has been designated to be a prototype for the purposes of internals vibrations as defined in United States Nuclear Regulatory Commission (USNRC) Regulatory Guide 1.20, Revision 2 (Reference 1). This report describes the expected and acceptable responses at the transducer locations for the internals vibration preoperational measurement program. This preoperational measurement program will be completed using the first AP1000 reactor internals.

The AP1000 reactor internals flow-induced vibration program report was issued in July 2003 (Reference 2). Since Reference 2 was written, there were several reactor internals design changes, as described in WCAP-16716-NP/APP-GW-GLN-012 (Reference 3). These design changes were considered in determining the values contained in this report.

The regulatory guide (Reference 1) also calls for a pre- and post-hot functional test inspection program. This program is described in Section 3 of this report.

With the completion of the expected and acceptable reactor internals vibration responses documented in this report, the USNRC should consider the Combined Operation License (COL) item requiring this vibration related information to be closed and applicable to all COL applications referencing the AP1000 design certification.

1.2 DCD MARKUP

The information in this report is required by COL Item 3.9.1 in paragraph 3.9.8.1 of the Design Control Document (DCD) as follows:

3.9.8.1 Reactor Internals Vibration Assessment and Predicted Response

Information, including predicted vibration response and ~~allowable~~ acceptable response, ~~will be provided prior~~ to support the preoperational vibration testing of the first AP1000 consistent with the guidance of Regulatory Guide 1.20, Rev. 2 is provided in WCAP-16687 (Reference 27).

Add Reference 27 to Subsection 3.9.9 as follows:

27. "AP1000 Reactor Internals Expected and Acceptable Responses During Preoperational Vibration Measurement Program" WCAP-16687-P, Rev. 1(Proprietary), March 2007.

The DCD, Rev. 15, and the revised DCD vibration measurement program's transducer locations are presented below.

DCD, Rev. 15 Table 3.9-4

Table 3.9-4

FIRST PLANT AP1000 REACTOR INTERNALS VIBRATION MEASUREMENT PROGRAM TRANSDUCER LOCATIONS

Instrumented Component	Number and Type of Transducers	Approximate Transducer Locations	Direction of Sensitivity
Core Shroud (Inner Wall)	4 accelerometers	0°, 180°, 225°, 270°	Radial
Core Shroud to Core Barrel	2 relative displacement transducers	0°	Radial
Core Barrel Flange (Outer Wall)	4 strain gages	0°, 90, 180°, 270°	Axial
Core Barrel Flange (Inner Wall)	2 strain gages	180°, 270°	Axial
Core Barrel Mid-elevation	3 accelerometers	0°, 180°, 225°	Radial
Core Barrel Mid-elevation	1 pressure transducer	0°	Radial
Upper Support Skirt (Inside and Outside)	3 strain gages	180°, 90° inside	Axial
Lower Core Support Plate Weld (Outside)	2 strain gages	0°, 90°	Vertical
Vortex Suppression Plate Support Columns (2)	4 strain gages or 4 accelerometers	On column near lower core support plate or On vortex suppression ring	Axial Horizontal
Reactor Vessel (Head Studs)	4 accelerometers	0°, 90°, 180°, 270°	Vertical
	3 accelerometers	0°, 90°, 180°	Horizontal
Support Column Extension	2 strain gages	0°, 90°	Axial
Guide Tube	4 strain gages	0°, 90°, 180°, 270°	Axial
Upper Support Column	4 strain gages	0°, 90°, 180°, 270°	Axial

Revised DCD Table 3.9-4

Table 3.9-4

FIRST PLANT AP1000 REACTOR INTERNALS VIBRATION MEASUREMENT PROGRAM TRANSDUCER LOCATIONS

Instrumented Component	Number and Type of Transducers	Approximate Transducer Locations	Direction of Sensitivity
Core Shroud (Inner Wall)	4 accelerometers	0°, 45°, 180°, 315°	Radial
Core Shroud to Core Barrel	4 relative displacement transducers	225°, 315°	2 Radial, 2 Tangential
Core Barrel Flange (Outer Wall)	4 strain gages	0°, 90°, 180°, 270°	Axial
Core Barrel Flange (Inner Wall)	1 strain gage	180°	Axial
Core Barrel Mid-elevation (Outer Wall)	3 accelerometers	0°, 45°, 180°	Radial
Core Barrel Mid-elevation	1 pressure transducer	0°	Radial
Upper Support Skirt (Inner Wall)	3 strain gages	0°, 90°, 180°	Axial
Upper Support Skirt (Outer Wall)	1 strain gage	90°	Axial
Lower Core Support Plate	1 accelerometer	Near the center of the plate	Vertical
Vortex Suppression Plate Support Columns (2)	4 strain gages	On outside columns at an elevation near LCSP with 3 gages on one column and 1 gage on another column, these two columns are 180° apart	Axial
Reactor Vessel (Head Studs)	4 accelerometers	Studs at 0°, 90°, 180°, 270°	Vertical
	3 accelerometers	Studs at 0°, and 180° (x-direction), Stud at 180° (y-direction)	Horizontal
UMIA Column on G-8 above the Bracket	2 strain gages	0°, 90°	Axial
UMIA Column on B-7 below the Bracket	3 strain gages	0°, 90°, 180°	Axial
Lower Guide Tube on B-6	4 strain gages	0°, 90°, 180°, 270°	Axial
Upper Guide Tube on B-6	2 strain gages	0°, 90°	Axial
Upper Support Column on B-7	4 strain gages	0°, 90°, 180°, 270°	Axial

2 CALCULATIONS AND BASES FOR DEVELOPMENT OF EXPECTED AND ACCEPTABLE RESPONSES

2.1 CALCULATIONS FOR EXPECTED RESPONSES FOR THE INTERNALS

2.1.1 Hot Functional Test Flow Rate with Four Reactor Coolant Pumps in Operation

For the hot functional test (HFT), it is planned that the core will not be installed and that debris screens will be installed in the lower core support plate (LCSP) flow holes. The estimated total flow with all pumps running at full speed for HFT conditions is [ ]^{a,c} for best estimate flow and [ ]^{a,c} for mechanical design flow, Reference 4. These two flow rates are [ ]^{a,c} greater than the HFP best estimate flow [ ]^{a,c} and the mechanical design flow (MDF) [ ],^{a,c} respectively.

2.1.2 Effect of Absence of the Core on Downcomer Modal Properties

The system model modes and modal stiffnesses for hot full-power (HFP) with the core installed and for HFT conditions, 557°F, no core (or control rods) are listed in Table 2.1.2-1 and Table 2.1.2-2. Fuel assembly modes are not included in the HFT list since the core will not be installed for HFT.

Common with-core and without-core modes are compared in Table 2.1.2-3. It is noted that the modes are generally similar with the exception that fuel assembly modes are not present for the HFT case.

The typical core barrel beam modes and core shroud modes have similar generalized stiffnesses and natural frequencies as shown in Table 2.1.2-3. Most other modes such as those dominated by reactor vessel motion or control rod drive mechanism motion (CRDM) are also similar.

The effects of the absence of the core on the core barrel beam mode were estimated by:

- Selecting a with-core beam mode having a mode shape similar to that of the no-core beam mode and determining the ratio of the generalized stiffnesses.
- Determining the HFT to HFP test inlet ρV^2 .
- Combining these ratios.

The ratios used to obtain HFT core barrel displacement results from HFP displacements are:

- Flow rate squared ratio = [ ]^{a,c}
- Density ratio = [ ]^{a,c}
- Generalized stiffness ratio = [ ]^{a,c}

Tables 2.1.2-1 and 2.1.2-2 present the core barrel beam mode and core shroud beam mode both with-core and without-core, respectively.

2.1.3 Natural Frequencies of Guide Tube, Upper Mounted Instrumentation, and Lower Plenum Structures

Upper mounted instrumentation, guide tube, secondary core support, and vortex suppression structure natural frequencies are not significantly affected by the presence or absence of the core. Expected natural frequencies for these components are listed in Table 2.1.4-1 through Table 2.1.4-5.

Table 2.1.2-1 System Model with Core, with Water and No Contact at the Radial Keys

Mode	Frequency (Hz)	Generalized Mass (lb sec^2/in)	Generalized Stiffness (lb/in)	Description
1				
2				
3				
4				
5				
6				
7				
8				
9				
10				
11				
12				
13				
14				
15				
16				
17				
18				
19				
20				
21				
22				
23				
24				
25				
26				
27				
28				
29				
30				
31				
32				
33				
34				
35				
36				
37				
38				

a, c

Table 2.1.2-1 System Model with Core, with Water and No Contact at the Radial Keys (Cont.)

a, c

Mode	Frequency (Hz)	Generalized Mass (lb sec^2/in)	Generalized Stiffness (lb/in)	Description
39				
40				
41				
42				
43				
44				
45				
46				
47				
48				
49				
50				
51				
52				
53				
54				
55				
56				
57				
58				
59				
60				
61				
62				
63				
64				
65				
66				
67				
68				
69				
70				

Table 2.1.2-2 System Model without Core, with Water and No Contact at the Radial Keys

Mode	Frequency (Hz)	Generalized Mass (lb sec^2/in)	Generalized Stiffness (lb/in)	Description
1				
2				
3				
4				
5				
6				
7				
8				
9				
10				
11				
12				
13				
14				
15				
16				
17				
18				
19				
20				
21				
22				
23				
24				
25				
26				
27				
28				
29				
30				
31				
32				
33				
34				
35				
36				
37				
38				
39				
40				

a, c

Table 2.1.2-2 System Model without Core, with Water and No Contact at the Radial Keys (Cont.)

Mode	Frequency (Hz)	Generalized Mass (lb sec^2/in)	Generalized Stiffness (lb/in)	Description
41				
42				
43				
44				
45				
46				
47				
48				
49				
50				

a, c

Table 2.1.2-3 Comparison of Common HFP Condition and HFT Condition Modes

Mode (Shape) Description	HFP				HFT			
	Mode Number	Natural Frequency (Hz)	Generalized Mass (lb sec^2/in)	Generalized Stiffness (lb/in)	Mode Number[1]	Natural Frequency (Hz)	Generalized Mass (lb sec^2/in)	Generalized Stiffness (lb/in)

a, c

Table 2.1.4-1 Upper Mounted Instrumentation Assembly Natural Frequencies Upper Guide Tube – UMIA Natural Frequencies

Mode Number	G-8 (Hz)	B-7 (Hz)
1		
2		
3		
4		
5		
6		
7		
8		
9		
10		
11		
12		

a, c

Table 2.1.4-2 Upper (With No UMIA Attached) and Lower Guide Tube Natural Frequencies

Mode Number	Upper	Lower
1		
2		
3		
4		

a, c

Table 2.1.4-3 Upper Support Column Natural Frequencies

Mode Shape	Natural Frequency (Hz)
First Beam Mode	
Second Beam Mode	

a, c

Table 2.1.4-4 Secondary Core Support and Vortex Suppression Plate Assembly Natural Frequencies

Mode Number	Mode Shape	Natural Frequency (Hz)	
		3D Model	Beam Model
1	Translation		
2	Translation		
3	Torsion		

a, c

Table 2.1.4-5 Core Shroud Natural Frequencies

Mode Shape	Natural Frequency (Hz)	
	Modified Braces	
	Mode No. (1)	Fn (Hz)
In water, no core, no radial support contact, n = 2, m = 1		
In water, no core, no radial support contact, n =1, m = 1		

Note:

1. See Table 2.1.2-2.

a, c

2.2 EXPECTED RESPONSES AND ACCEPTABLE RESPONSES

The expected and acceptable responses at internals transducer locations are summarized in Table 2.2-1 through Table 2.2-11.

Lower internals responses will be compared and adjusted, if necessary, based on scale model flow test results.

Table 2.2-1 Upper Support Skirt Strain Gage Responses at Full HFT Temperature

Location	Direction of Sensitivity	0.25 to100 Hz RMS Strain Level (με)	
		Expected	Acceptable
Inner wall, 0°	Vertical		
Inner wall, 90°	Vertical		
Inner wall, 180°	Vertical		
Outer wall, 90°	Vertical		

a, c

Table 2.2-2 Lower Guide Tube Strain Gage Responses at Full HFT Temperature

Location	Direction of Sensitivity	Zero to Peak Strain Level (με)	
		Expected	Acceptable
Below lower guide tube B-6 flange, 0°	Vertical		
Below lower guide tube B-6 flange, 90°	Vertical		
Below lower guide tube B-6 flange, 180°	Vertical		
Below lower guide tube B-6 flange, 270°	Vertical		

a, c

Table 2.2-3 Upper Guide Tube Strain Gage Responses at Full HFT Temperature

Location	Direction of Sensitivity	Zero to Peak Strain Level (με)	
		Expected	Acceptable
Above upper guide tube B-6 flange, 0°	Vertical		
Above upper guide tube B-6 flange, 90°	Vertical		

a, c

Table 2.2-4 Upper Support Column Strain Gage Responses at Full HFT Temperature

Location	Direction of Sensitivity	Zero to Peak Strain Level (με)	
		Expected	Acceptable
B-7, 12.24 inches below USP, 0°	Vertical		
B-7, 12.24 inches below USP, 90°	Vertical		
B-7, 12.24 inches below USP, 180°	Vertical		
B-7, 12.24 inches below USP, 270°	Vertical		

a, c

Table 2.2-5 UMIA Guide Column Strain Gage Responses at Full HFT Temperature

Location	Direction of Sensitivity	Zero to Peak Strain Level (με)	
		Expected	Acceptable
Core location B-7 below bracket, 0°	Vertical		
Core location B-7 below bracket, 90°	Vertical		
Core location G-8 above bracket, 0°	Vertical		
Core location G-8 above bracket, 90°	Vertical		
Core location B-7 below bracket, 180°	Vertical		

a, c

Table 2.2-6 Core Barrel Flange Strain Gage Responses at Full HFT Temperature

Location	Direction of Sensitivity	RMS Strain Level (με)	
		Expected	Acceptable
Below weld, outer wall, 0°	Vertical		
Below weld, outer wall, 90°	Vertical		
Below weld, outer wall, 180°	Vertical		
Below weld, outer wall, 270°	Vertical		
Below weld, inner wall, 180°	Vertical		

a, c

Table 2.2-7 Core Barrel Accelerometer Responses at Full HFT Temperature

Location	Direction of Sensitivity	RMS Displacement Level (mils)	
		Expected	Acceptable
Core barrel outer wall, mid-elevation, 0°	Radial		
Core barrel outer wall, mid-elevation, 45°	Radial		
Core barrel outer wall, mid-elevation, 180°	Radial		

a, c

Table 2.2-8 Lower Core Support Plate Accelerometer Responses at Full HFT Temperature

Location	Direction of Sensitivity	Zero to Peak Displacement Level	
		Expected	Acceptable
Lower surface of LCSP, near center	Vertical		

a, c

Table 2.2-9 Vortex Suppression Plate Strain Gage Responses at Full HFT Temperature

Location	Direction of Sensitivity	Zero to Peak Strain Level (με)	
		Expected	Acceptable
Outer radial surface on first column	Vertical		
Inner radial surface on first column	Vertical		
Tangential surface on first column	Vertical		
Tangential surface on second column[1]	Vertical		

Note:

1. Second column is located 180° from the first.

a, c

Table 2.2-10 Core Shroud Accelerometer Responses at Full HFT Temperature

Location	Direction of Sensitivity	RMS Displacement Level (mils)		Zero to Peak Displacement Level (mils)	
		Expected	Acceptable	Expected	Acceptable
Near top plate, inner wall, 0°	Radial				
Near top plate, inner wall, 45°	Radial				
Near top plate, inner wall 180°	Radial				
Near top plate, inner wall 315°	Radial				

a, c

Table 2.2-11 Core Shroud Responses Relative to Core Barrel at Full HFT Temperature

Location	Direction of Sensitivity	RMS Displacement Level (mils)	Zero to Peak Displacement Level (mils)	Acceptable Level
		Expected	Expected	
Inner wall, near top plate, 225°	Radial			
Inner wall, near top plate, 225°	Tangential			
Inner wall near top plate, 315°	Radial			
Inner wall, near top plate, 315°	Tangential			

a, c

3 PRE- AND POST-HOT FUNCTIONAL TEST INSPECTION PROGRAM

The inspection methods and the locations to be inspected are listed in Table 3-1 and shown in Figure 3-1.

Table 3-1 Pre- and Post-Hot Functional Test Examination

SEE NOTES	LOCATION	FEATURES TO BE EXAMINED	COMMENTS AND OBSERVATIONS BEFORE FUNCTIONAL TEST	COMMENTS AND OBSERVATIONS AFTER FUNCTIONAL TEST
4	1	Inside Surfaces of UMIA Funnels		
1,7	2	Column Extension Attachments and Locking Devices		
1,4	3	Accessible Surfaces of Upper Guide Tube to UMIA Top Plate		
1	4	Flow Skirt to Reactor Vessel Lower Head Welds		
1	5	Upper Support Column (USC) Nut to Upper Support Plate (USP) Lock Welds, and UMIA Column to USC Attachment		
10	6	Core Shroud Studs and Nuts		
6	7	Hold Down Spring Interface Surface Condition		
1	8	Accessible Welds on Support Column Lower Nozzles		
3,6	9	Upper Core Plate Inserts		
2	10	Guide Tube Flange Bolts and Locking Devices		
2	11	Accessible Support Columns, and Core Plate Insert Screw Locking Devices		
1	12	Upper Support Skirt to Plate Girth Weld		
1	13	Upper Support Skirt to Flange Girth Weld		
1	14	Accessible Guide Tube Welds		
1,2	15	Upper Core Plate Insert Locking Devices		
1	16	Upper Barrel to Flange Girth Weld		
1	17	Upper Barrel to Lower Barrel Girth Weld		
1	18	Lower Barrel to Lower Core Support Plate Girth Weld		
1,4	19	Alignment Plate Bearing Surfaces		
6	20	Outlet Nozzle Interface Surface Condition		
1	21	Neutron Shield Panel Dowel Pin Welds; Head Cooling Spray Nozzle Welds		
1,2	22	Neutron Shield Panel Screw Locking Devices		
7	23	Interface Surfaces at the Spacer Pads Along the Top and Bottom Ends of the Neutron Panels		
1	24	Core Shroud C Panel to W Panel Welds		
1	25	Fuel Pin to Lower Core Support Plate Locking Devices		
1	26	Secondary Core Support Housing to Base Plate Weld		
1,2,8,9	27	Locking Devices and Contact of the Butt Columns where Attached to the Core Support and Vortex Suppression Plate		
1,2,8,9	28	Locking Devices and Contact of the Secondary Core Support Columns at the Lower Core Support Plate and at the Vortex Suppression Plate		

Table 3-1 Pre- and Post-Hot Functional Test Examination (Cont.)

SEE NOTES	LOCATION	FEATURES TO BE EXAMINED	COMMENTS AND OBSERVATIONS BEFORE FUNCTIONAL TEST	COMMENTS AND OBSERVATIONS AFTER FUNCTIONAL TEST
1	29	Radial Support Key Welds		
1,4	30	Radial Support Key Bearing Surfaces		
1,4	31	Head and Vessel Alignment Pins Locking Devices and Bearing Surfaces		
1,2	32	Irradiation Specimen Guide Locking Devices and Dowel Pins		
6	33	Vessel Nozzle Interface Surface Condition		
1,3,4	34	Vessel Clevis Locking Arrangements and Bearing Surfaces		
1	35	Reinforcement Plate Welds		
1	36	Longitudinal Welds that are Accessible (Core Barrel, USP Skirt)		
11	37	Inner Surfaces of Flow Skirt		
1,6	38	Direct Injection Deflector Welds		
6	39	Core Shroud Top Plate Inserts		
2	40	Core Shroud Top Plate Inserts Locking Devices		
4	41	Top Surface of Flow Skirt (Absence of Contact)		
2	42	Alignment Plate Screws and Locking Devices		

NOTES:

1. VISUALLY EXAMINE WELDS USING 5–10X MAGNIFICATION. NO CRACKS ALLOWED.
2. VERIFY THAT LOCKING DEVICES ARE CRIMPED AND UNDAMAGED.
3. VERIFY THAT INSERTS ARE SEATED (.0015 [0.038] FEELER MUST NOT PASS THRU INTERFACE).
4. VISUALLY EXAMINE FACES FOR DAMAGE USING 5–10X MAGNIFICATION.
5. VERIFY THAT FITTINGS ARE TIGHT.
6. VISUALLY EXAMINE INTERFACE SURFACES FOR ANY EVIDENCE OF DAMAGE.
7. VERIFY SEATING USING A .0015 [0.038] FEELER GAGE. FEELER MUST NOT PASS THRU 90°.
8. VERIFY SEATING USING A .0015 [0.038] FEELER GAGE NO GAPS ALLOWED.
9. U.T. INSPECTION PERFORMED ON E.B. WELDS.
10. U.T. INSPECTION TO VERIFY PRELOAD.
11. VISUALLY EXAMINE INNER SURFACES FOR DAMAGE USING 5–10X MAGNIFICATION.

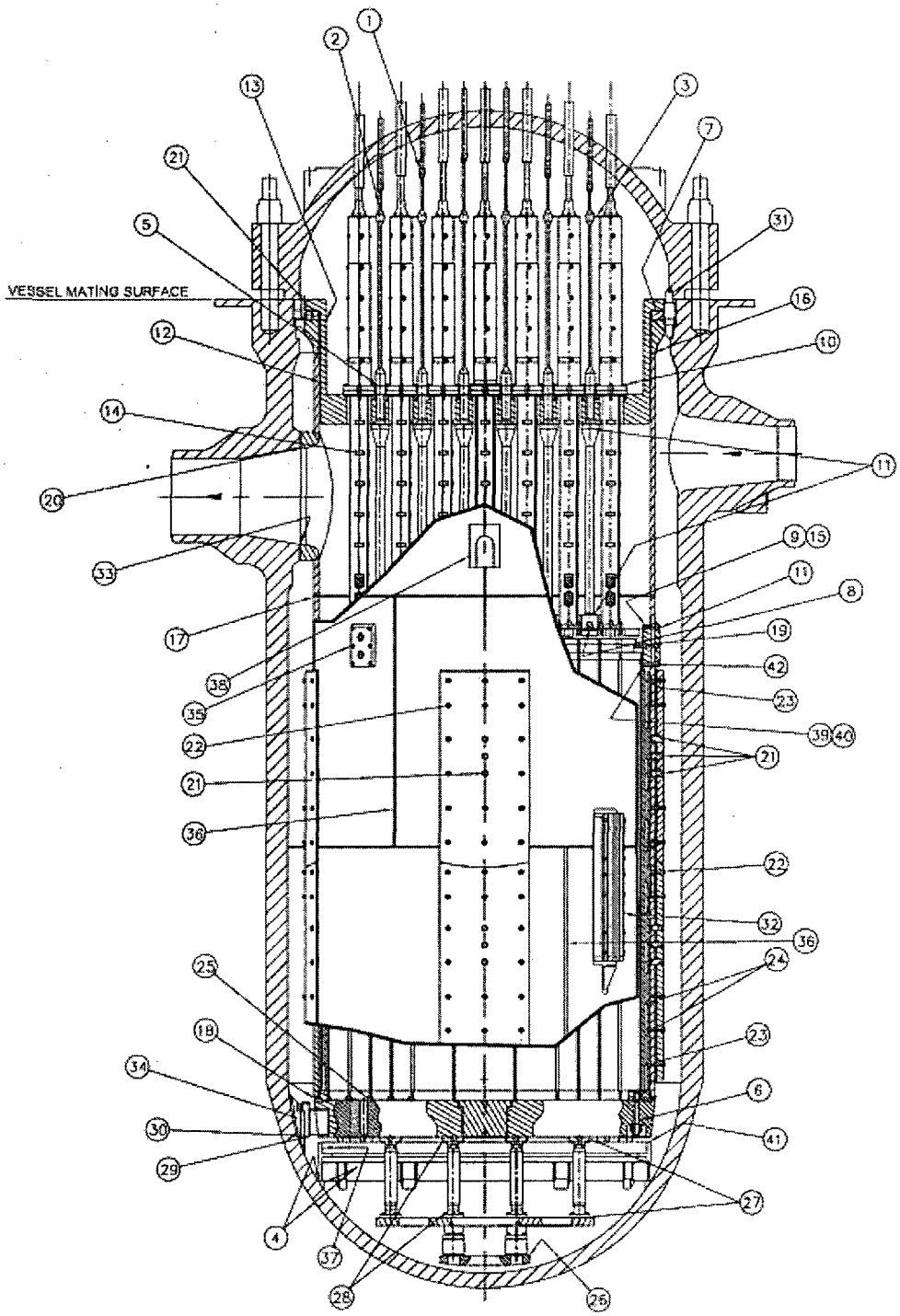

Figure 3-1 Pre- and Post-Hot Functional Test Examination Locations

4 REGULATORY IMPACT

FSER IMPACT

The preoperational flow-induced vibration analysis and testing of reactor internals is addressed in Subsection 3.9.2.3 of the USNRC Final Safety Evaluation Report (FSER, Reference 5) write-up. The discussion of the testing is generally consistent with the testing outlined in this report.

The conclusions of the FSER are not altered by the changes outlined in this report.

B. SCREENING QUESTIONS (Check correct response and provide justification for that determination under each response)

1. Does the proposed change involve a change to an SSC that adversely affects a DCD described design function? ☐ YES ☒ NO

   The changes outlined in the testing program do not alter DCD described design functions of the reactor internals.

2. Does the proposed change involve a change to a procedure that adversely affects how DCD described SSC design functions are performed or controlled? ☐ YES ☒ NO

   The changes outlined in the testing program do not alter the response of the reactor internals to anticipated transient and postulated accidents and do not affect how DCD described SSC design functions are performed or controlled.

3. Does the proposed activity involve revising or replacing a DCD described evaluation methodology that is used in establishing the design bases or used in the safety analyses? ☐ YES ☒ NO

   The changes outlined in the testing program will not affect any evaluation methodology that is used in establishing the design bases or used in the safety analysis. The methodology used to predict vibration response in the reactor internals is not altered. The methods and assumptions used for safety analyses are not altered.

4. Does the proposed activity involve a test or experiment not described in the DCD, where an SSC is utilized or controlled in a manner that is outside the reference bounds of the design for that SSC or is inconsistent with analyses or descriptions in the DCD? ☐ YES ☒ NO

   The changes outlined in the testing program will not require an additional test, experiment, or changes to complete testing. The vibration testing is completed prior to fuel load.

C. EVALUATION OF DEPARTURE FROM TIER 2 INFORMATION (Check correct response and provide justification for that determination under each response)

10 CFR Part 52, Appendix D, Section VIII. B.5.a. provides that an applicant for a combined licensee who references the AP1000 design certification may depart from Tier 2 information, without prior NRC approval, if it does not require a license amendment under paragraph B.5.b. The questions below address the criteria of B.5.b.

1. Does the proposed departure result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the plant-specific DCD?	☐ YES ☒ NO
The changes outlined in the testing program will not adversely affect capability of the reactor internals to support and align fuel assemblies or alter the capability of the reactor internals to direct reactor coolant flow through the core. Therefore these changes will not increase the frequency of occurrence of an accident previously evaluated in the plant specific DCD.	
2. Does the proposed departure result in more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system, or component (SSC) important to safety and previously evaluated in the plant-specific DCD?	☐ YES ☒ NO
There are no changes which will cause an increase in the probability of an occurrence of a malfunction of any SSC important to safety and previously evaluated in the plant specific DCD. The changes outlined in the testing program will not alter the operating parameters for the reactor coolant system or systems and components used to mitigate postulated accident conditions. The response of the reactor internals to transient conditions and postulated accident conditions is not altered by the changes.	
3. Does the proposed departure result in more than a minimal increase in the consequences of an accident previously evaluated in the plant-specific DCD?	☐ YES ☒ NO
The changes outlined in the testing program will have no impact on the consequences of an accident. The changes outlined in the testing program will not alter the calculated radioactive releases from postulated accidents. The changes outlined in the testing program will not alter the response of the containment systems to postulated accident conditions. The changes outlined in the testing program will not alter the integrity of the containment pressure boundary or the primary pressure boundary provided by the reactor vessel.	
4. Does the proposed departure result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the plant-specific DCD?	☐ YES ☒ NO
The changes outlined in the testing program will not alter the operating parameters for the reactor coolant system or systems and components used to mitigate postulated accident conditions. The changes outlined in the testing program will not alter the calculated radioactive releases from postulated accidents. The changes outlined in the testing program will not result in an increase in the consequences of a malfunction of any SSC important to safety previously evaluated in the plant specific DCD.	
5. Does the proposed departure create a possibility for an accident of a different type than any evaluated previously in the plant-specific DCD?	☐ YES ☒ NO
The changes outlined in the testing program will not introduce any additional failure modes. The changes outlined in the testing program will not alter the operating parameters for the reactor coolant system or systems and components used to mitigate postulated accident conditions. Therefore, these changes will not result in an accident of a type different than what has already been evaluated in the DCD.	

6. Does the proposed departure create a possibility for a malfunction of an SSC important to safety with a different result than any evaluated previously in the plant-specific DCD?	☐ YES ☒ NO
The changes outlined in the testing program will not alter the operating parameters for the reactor coolant system or systems and components used to mitigate postulated accident conditions. Therefore, these changes will not create a possibility for a malfunction of any SSC important to safety.	
7. Does the proposed departure result in a design basis limit for a fission product barrier as described in the plant-specific DCD being exceeded or altered?	☐ YES ☒ NO
The changes outlined in the testing program will not adversely affect the support of the fuel assemblies provided by the reactor internals. These changes will have no impact on the fuel rod cladding. The changes will not result in any impact to the fission product barrier as described in the plant specific DCD.	
8. Does the proposed departure result in a departure from a method of evaluation described in the plant-specific DCD used in establishing the design bases or in the safety analyses?	☐ YES ☒ NO
The changes outlined in the testing program will not alter the methodology or acceptance criteria used to evaluate the vibration response of the reactor internals. The changes outlined in the testing program will not alter the methodology used for safety analyses. The changes will not result in a departure from a method of evaluation described in the plant-specific DCD used in establishing the design bases or in the safety analyses.	
☒ The answers to the evaluation questions above are "NO" and the proposed departure from Tier 2 does not require prior NRC review to be included in plant specific FSARs as provided in 10 CFR Part 52, Appendix D, Section VIII. B.5.b ☐ One or more of the answers to the evaluation questions above are "YES" and the proposed change requires NRC review.	

D. IMPACT ON RESOLUTION OF A SEVERE ACCIDENT ISSUE

10 CFR Part 52, Appendix D, Section VIII. B.5.a. provides that an applicant for a combined licensee who references the AP1000 design certification may depart from Tier 2 information, without prior NRC approval, if it does not require a license amendment under paragraph B.5.c. The questions below address the criteria of B.5.c.

1. Does the proposed activity result in an impact features that mitigate severe accidents. If the answer is Yes answer Questions 2 and 3 below.	☐YES ☒ NO
The changes will not impact any features that mitigate severe accidents.	
2. Is there is a substantial increase in the probability of a severe accident such that a particular severe accident previously reviewed and determined to be not credible could become credible?	☐YES ☐ NO ☒ N/A

3. Is there is a substantial increase in the consequences to the public of a particular severe accident previously reviewed?	☐ YES ☐ NO ☒ N/A
☒ The answers to the evaluation questions above are "NO" or are not applicable and the proposed departure from Tier 2 does not require prior NRC review to be included in plant specific FSARs as provided in 10 CFR Part 52, Appendix D, Section VIII. B.5.c ☐ One or more of the answers to the evaluation questions above are "YES" and the proposed change requires NRC review.	

E. SECURITY ASSESSMENT

1. Does the proposed change have an adverse impact on the security assessment of the AP1000. ☐ YES ☒ NO

   The changes outlined in the testing program will not alter barriers or alarms that control access to protected areas of the plant. The changes outlined in the testing program will not alter requirements for security personnel.

5 REFERENCES

1. Regulatory Guide 1.20, Revision 2, "Comprehensive Vibratory Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing."

2. WCAP-15949-P, Revision 1, "AP1000 Reactor Internals Flow-Induced Vibration Assessment Program," M.D. Snyder, et al, July 2003, Westinghouse Proprietary.

3. WCAP-16716-NP (APP-GW-GLN-012), Revision 0, Technical Report 29, "AP1000 Reactor Internals Design Changes," January 2007.

4. DCP/WNS0001, "RCP Operating Conditions for AP1000 Reactor Vessel Internals Design," April 17, 2006, Westinghouse Proprietary.

5. NUREG-1793, NRC "Final Safety Evaluation Report for AP1000 Design," September 2004.