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Your ref: Docket No. 52-006
Our ref: DCP/NRC2163

June 20, 2008

Subject: AP1000 Response to Requests for Additional Information (SRP3.9.2)

Westinghouse is submitting a response to the NRC request for additional information (RAI) on SRP Section 3.9.2. This RAI response is submitted in support of the AP1000 Design Certification Amendment Application (Docket No. 52-006). The information included in the response is generic and is expected to apply to all COL applications referencing the AP1000 Design Certification and the AP1000 Design Certification Amendment Application.

A response is provided for RAI-SRP3.9.2-EMB1-01 through -11, as sent in an email from Mike Miernicki to Sam Adams dated April 29, 2008. This response completes all requests received to date for SRP Section 3.9.2.

Questions or requests for additional information related to the content and preparation of this response should be directed to Westinghouse. Please send copies of such questions or requests to the prospective applicants for combined licenses referencing the AP1000 Design Certification. A representative for each applicant is included on the cc: list of this letter.

Very truly yours,

Robert Sisk, Manager
Licensing and Customer Interface
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/Enclosure

1. Response to Requests for Additional Information on SRP Section 3.9.2

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NRO

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ENCLOSURE 1

Response to Requests for Additional Information on SRP Section 3.9.2

AP1000 TECHNICAL REPORT REVIEW

Response to Request For Additional Information (RAI)

RAI Response Number: RAI-SRP3.9.2-EMB1-01
Revision: 0

Question:

Describe the design and modeling of the core barrel/upper core plate as they relate to flow-induced vibration (FIV) structural dynamic analysis. What is the uncertainty that the model of the interface employed in the modal analysis is representative of the physical support?

Westinghouse Response:

The upper core plate is modeled as a part of the upper internals in the system model. The gaps between the upper core plate (and core shroud) slots and the alignment plates mounted on the core barrel are also modeled (see Figures 1 and 2).

To ensure that the entire range of possible gaps between the upper core plate and the core barrel alignment plates is evaluated, time-history analyses were performed with various sets of gaps (upper core plate, top core shroud plate, and core barrel lower supports). Table 6-9 in WCAP-15949-P, Rev 2 (Reference 1), shows the gaps modeled and the resulting loads. The resulting highest load was used in the structural analysis.

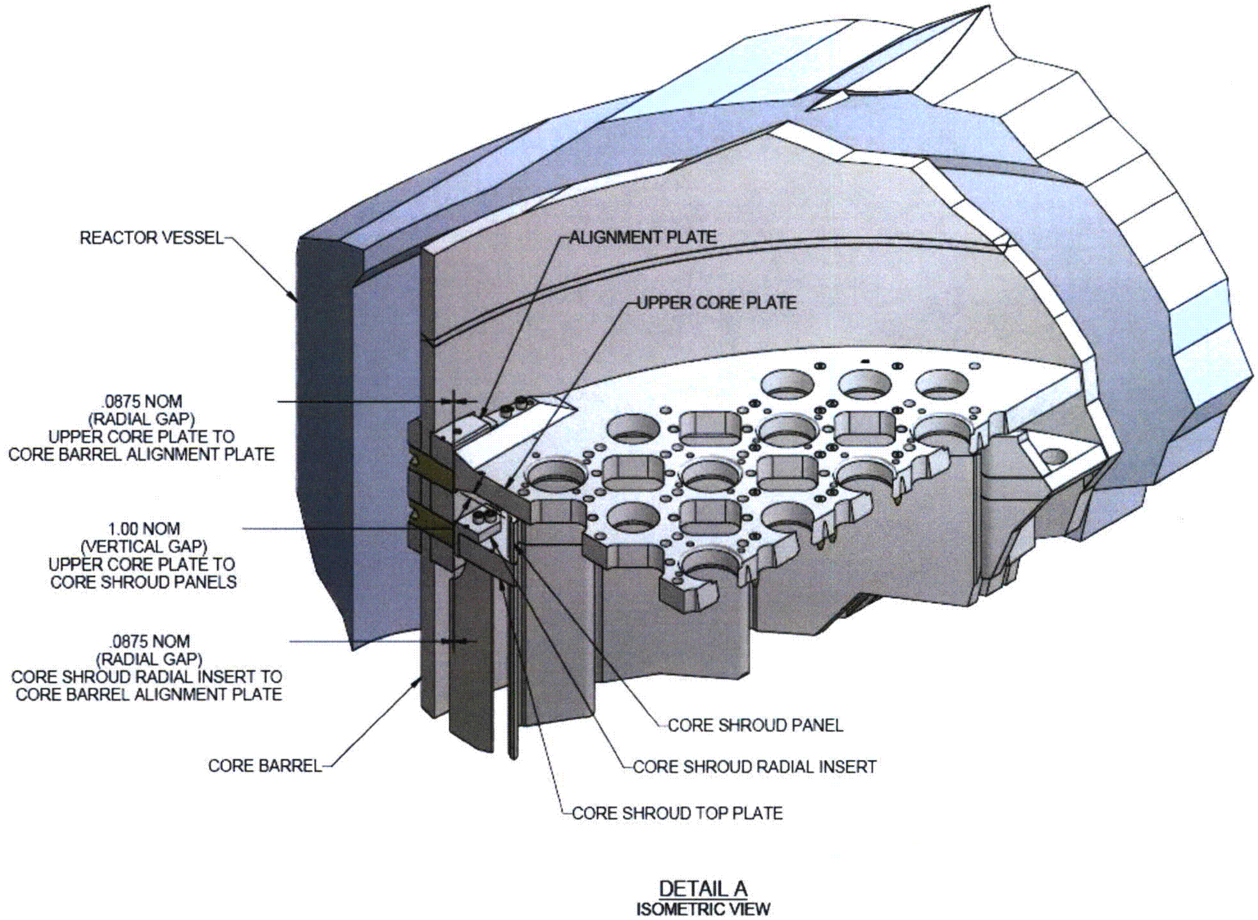
References:

1. Westinghouse Technical Report, WCAP-15949-P, Rev. 2, "AP1000 Reactor Internals Flow-Induced Vibration Assessment Program," June 22, 2007.

Design Control Document (DCD) Revision: None.

PRA Revision: None.

Technical Report (TR) Revision: None.



REACTOR VESSEL
ALIGNMENT PLATE
UPPER CORE PLATE
.0875 NOM
(RADIAL GAP)
UPPER CORE PLATE TO
CORE BARREL ALIGNMENT PLATE
1.00 NOM
(VERTICAL GAP)
UPPER CORE PLATE TO
CORE SHROUD PANELS
.0875 NOM
(RADIAL GAP)
CORE SHROUD RADIAL INSERT TO
CORE BARREL ALIGNMENT PLATE
CORE BARREL
CORE SHROUD PANEL
CORE SHROUD RADIAL INSERT
CORE SHROUD TOP PLATE
DETAIL A
ISOMETRIC VIEW

Figure 1

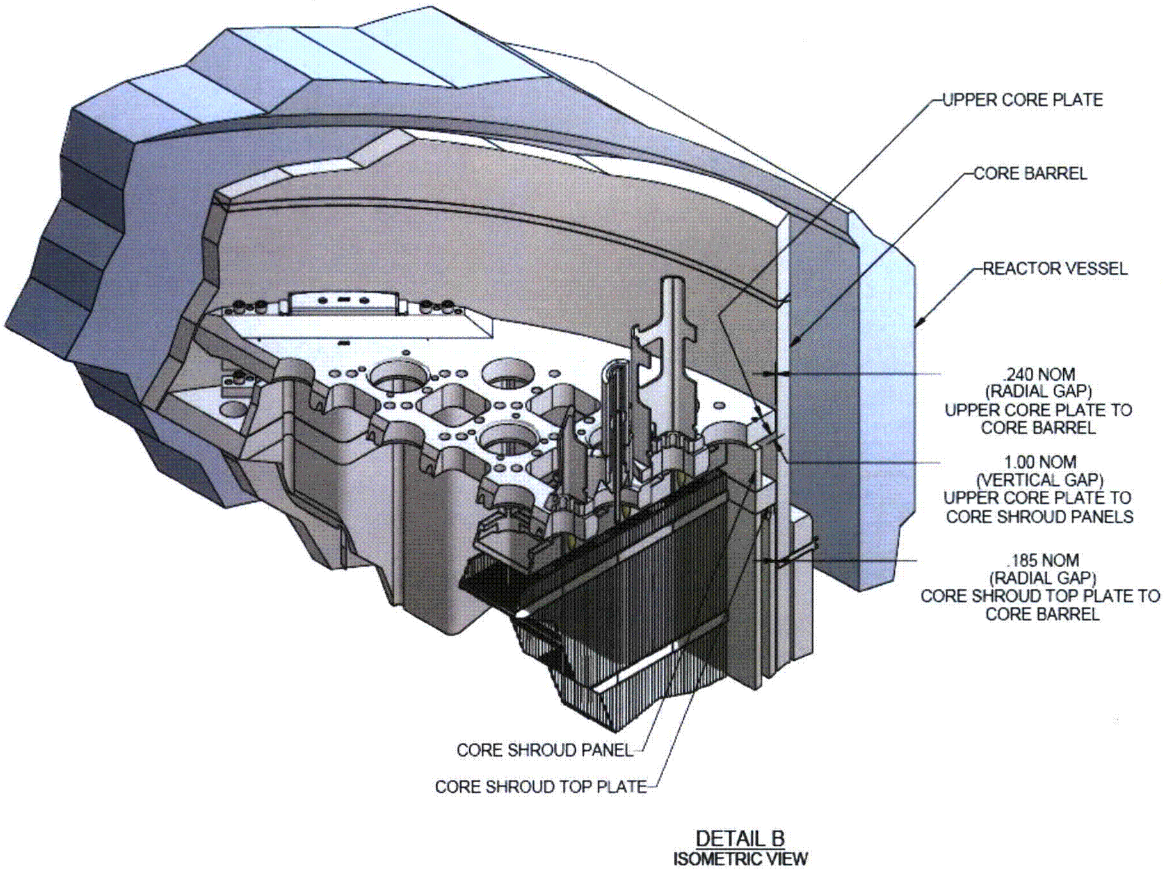

UPPER CORE PLATE
CORE BARREL
REACTOR VESSEL
.240 NOM (RADIAL GAP) UPPER CORE PLATE TO CORE BARREL
1.00 NOM (VERTICAL GAP) UPPER CORE PLATE TO CORE SHROUD PANELS
.185 NOM (RADIAL GAP) CORE SHROUD TOP PLATE TO CORE BARREL
CORE SHROUD PANEL
CORE SHROUD TOP PLATE
DETAIL B
ISOMETRIC VIEW

Figure 2

AP1000 TECHNICAL REPORT REVIEW

Response to Request For Additional Information (RAI)

RAI Response Number: RAI-SRP3.9.2-EMB1-02
Revision: 0

Question:

Discuss the potential fluid forces created by the redesigned neutron panels and their potential effects on the FIV excitation of the core barrel/core shroud.

Westinghouse Response:

The circumferential extent of the neutron panels was limited to correspond to the high vessel fluence levels, and thus minimize the flow blockage in the downcomer. The neutron panels are tapered circumferentially (following the reduction in fluence level) to minimize the flow area reduction. In addition, the reactor vessel inside diameter was increased by two inches over the core elevations when the panels were added. This results in a net flow area increase of 4% relative to the vessel-core barrel downcomer flow area before the panels were added. The lower average downcomer velocity is expected to offset the effects of the turbulence added by the neutron panels.

Design Control Document (DCD) Revision: None.

PRA Revision: None.

Technical Report (TR) Revision: None.

AP1000 TECHNICAL REPORT REVIEW

Response to Request For Additional Information (RAI)

RAI Response Number: RAI-SRP3.9.2-EMB1-03
Revision: 0

Question:

Discuss the rationale for and the location of instrumentation to provide predicted stresses. There is no instrumentation between the upper end of the core shroud and the lower core support plate. If these are not the maximum stresses, provide the value and location of the maximum stresses for the core barrel/core shroud assembly.

Westinghouse Response:

A detailed description of the internals model is provided in WCAP-15949, Revision 2. The instrumentation is designed to provide adequate information to describe the vibration time histories and modal content.

In the case of the core barrel, the beam modes can be inferred from the core barrel flange strain gages. The fundamental shell modes of the core barrel cover the entire length, the approximate mid point being at the top of the core shroud where three radially sensitive accelerometers are mounted.

For maximum stresses, the motions are defined by an assembly model. Where needed, sub-models are made to accurately define local, maximum stresses. Detailed core shroud models and sub-models are used to define maximum vibratory stress levels in the core shroud. Similarly, for the core barrel, models are used to define stresses at key locations such as core barrel flange (dominantly beam mode-induced stresses), and shell mode stresses) and barrel shell lower core support plate stresses (includes vertical motion-induced stresses).

The strain gages and other transducers are located such that they are not in an extremely high gradient area and so that, with the analytical models they can adequately define the vibration so that maximum stresses can be determined from the analytical models.

Thus, the instrumentation supported by the structural model (which is supported by the calculated versus measured mode shapes and natural frequencies) is considered adequate to define the maximum stresses due to flow and RCP-induced vibration.

The maximum stresses for the core barrel/core shroud are provided in Table 2-1 of WCAP-15949 (Reference 1). The maximum core barrel stress is at the core barrel wall to core barrel flange interface. The maximum core shroud stress is at the corner of the panel.

Reference:
1. Westinghouse Technical Report, WCAP-15949-P, Rev. 2, "AP1000 Reactor Internals Flow-Induced Vibration Assessment Program," June 22, 2007.

Design Control Document (DCD) Revision: None.

PRA Revision: None.

Technical Report (TR) Revision: None.

AP1000 TECHNICAL REPORT REVIEW

Response to Request For Additional Information (RAI)

RAI Response Number: RAI-SRP3.9.2-EMB1-04
Revision: 0

Question:

Discuss how the vibration content affects the strain gage data, how associated conversion factors from 3XL to AP1000 are affected, and the uncertainties in the conversion factors. Table 5.3, "Comparison of calculated and measured 3XL responses," in WCAP 15949 states that the accelerations are considered to be influenced by accelerometer pressure sensitivity and that vertical vibration content in the core barrel strain gages is difficult to ascertain because of masking by other contributors.

Westinghouse Response:

The strain gages are used to measure mean and oscillatory reactor internal responses. For example, in the core barrel flange strain gages, the oscillatory content includes contributions from core barrel beam modes, the vertical modes of the core barrel, and the shell modes of the core barrel. Supported by the core barrel analytical model and data from other transducers, the contribution of the various modes can be determined. This information is used to support the determination of the maximum stress in the core barrel flange.

During the 3XL hot functional vibration testing it was observed that the accelerometer data included an unexpected magnitude of response at a particular frequency that was postulated to be due to system pressure pulsations. The accelerometer pressure sensitivity was confirmed by the accelerometer vendor. It is considered that this was adequately recognized in the interpretation of the 3XL data.

The 3XL test data are used only to bench mark the analytical methods used to predict AP1000 responses, primarily the CFD based prediction of core barrel vibration. There are no conversion factors used in developing the AP1000 responses, since all of the AP1000 predictions are from analytical models.

Design Control Document (DCD) Revision: None.

PRA Revision: None.

Technical Report (TR) Revision: None.

RAI Response Number: RAI-SRP3.9.2-EMB1-05
Revision: 0

Question:

Describe the methodology for determining bias errors and uncertainties associated with data obtained from various sources for evaluating AP 1000 reactor internals responses. The overall methodology for estimating the vibration forces and using these forces to predict the response of the reactor internals is outlined in Figure 5-1 of WCAP-15949.

Westinghouse Response:

The transducers are calibrated prior to use. From this calibration, the voltage conversions at the temperature that the data were acquired are applied. Any uncertainty in the factors that convert voltages to physical units will also be recognized.

It is also noted that expected and measured responses were similar in past tests. In view of these factors, it is considered that bias errors and uncertainties are less than the minimum margin to allowable values-presently 0.2 for AP1000 (per WCAP-15949, Rev. 2, Table 2-1)

Reference:

1. Westinghouse Technical Report, WCAP-15949-P, Rev. 2, "AP1000 Reactor Internals Flow-Induced Vibration Assessment Program," June 22, 2007.

Design Control Document (DCD) Revision: None.

PRA Revision: None.

Technical Report (TR) Revision: None.

AP1000 TECHNICAL REPORT REVIEW

Response to Request For Additional Information (RAI)

RAI Response Number: RAI-SRP3.9.2-EMB1-06
Revision: 0

Question:
Discuss and summarize the significant additional information/items provided in WCAP 15949- Rev. 2; dated June 2007. WCAP 15949-P Rev 1 "AP 1000 Reactor Internals Flow Induced Vibration Assessment Program" was reviewed previously by the staff during the review of the AP 1000 standard design in September 2004 (Ref. NUREG - 1793 Vol. 1).

Westinghouse Response:

The most significant changes between Revision 1 and Revision 2 of WCAP 15949 are the addition of the neutron panels, the reactor vessel diameter increase in the core region, the revised specimen basket arrangement, and the addition of a flow skirt to the reactor vessel. The overall conclusion that the vibration amplitudes are sufficiently low for structural adequacy of the AP1000 reactor internals has not changed.

The itemized changes between WCAP-15949-P, Rev. 1 and Rev. 2 are noted below.

Record of Revisions
page iii The bases for information determined to be proprietary is on this page (was on Page 1-3) and Trademark Notes have been added.

Section 1 "Introduction"
page 1-2 Values for relative velocities (in Table 1-1 on page 1-3 have been adjusted.
page 1-2 A statement stating that a vortex suppression plate is now included has been added.
page 1-2 A section describing the evolution of Westinghouse internals has been added.
page 1-3 The flow ratios have been re-evaluated.
page 1-3 The bases for information determined to be proprietary has been moved to page iii.

Section 2 "Summary"
page 2-1 The Summary has been re-worded. The development of the AP1000 design has been made more general and a list of the sources of internals vibration, a summary of the report content and a summary of peak stresses and factors of safety have been added.
page 2-2 a summary of stresses and factors of safety has been added.

Section 3 "Design Differences and Relationships to Flow-Induced Vibrations"
page 3-1 Reference to neutron panels has been added.
page 3-1 The flow rate ratio to 3 XL has been adjusted.
page 3-1 A value for the change in vessel length has been added.
page 3-1 The relative vessel diameters have been updated.
page 3-1/3-2 The statement on outlet nozzles has been refined.
page 3-2 A statement regarding the addition of a flow skirt has been added.

Section 3.2 Lower Internals Assembly

page 3-2 The change in core barrel length relative to that of the 3XL has changed slightly.

page 3-2 Variable thickness neutron pads are introduced.

page 3-2 The statement regarding no neutron pads has been replaced with a statement regarding thermal shield Vs neutron panel internals.

page 3-2/3 Discussion of slightly higher core barrel vibration and removal of the baffle formers are changed from Section 5.3.2 to Section 6.

Section 3.2 Upper Internals Assembly

page 3-3 "guide plate to enclosure by pins instead of weld tabs" replaced by "guide plate to enclosure attachments by welds."

page 3-3 Upper assembly skirt length increase changed from 10 inches to 11 inches.

page 3-4 "top" mounted in-core instrumentation assembly changed to "upper" mounted in-core instrumentation assembly.

page 3-4 "[1.5 %] core bypass flow is expected to result in high cycle fatigue stresses within code allowable limits." has been changed to "[1.5 percent] upper head spray nozzle flow is expected to result in negligible vibrations."

page 3-4 The statement that "Motion of the upper support plate results high cycle fatigue stresses within allowable limits" has been added to Rev. 2.

page 3-5 The figures have been updated to show the addition of neutron panels and a flow skirt.

Scale Model and Plant Test Experience

4.1 General

page 4-1 "Westinghouse Experience:" deleted.

page 4-1 "Many years of operating experience" added to list of program bases.

page 4-1 A paragraph introducing the use of CFD and structural models has been removed from this Section.

4.2 Test Results Applicable to Lower Internals

page 4-1 "with no lower contact at the restraints" has been deleted.

page 4-2 Two paragraphs introducing vibration amplitudes and natural frequencies have been removed.

4.3 Test Results Applicable To Upper Internals

page 4-2 AP1000 outlet nozzle velocity changed from "4 percent higher that the velocity of the previously measured three-loop reactor" to "13 percent higher than the velocity of the 3XL reactor."

page 4-3 Table 4-1 removed the following test: Paluel plant test (French XL prototype).

page 4-4 Note No. 3 added.

page 4-5 The values for the Upper Guide Tube in Table 4-5 have changed in the Doel 3 Plant Test column as well as note "a" has been deleted and No. 1 has been added.

5.1 Vibration Response Calculations

section 5.1.3 "Effects of Internals Changes Made Since Basic Calculations Were Completed" was added.
section 5.1.3.1 "Downcomer Excitations and Related Responses" was added.
section 5.1.3.2 "Reduction of Core Shroud Brace Thickness" was added.
section 5.1.3.3 "RCP-Induced Loads" was added.

5.2 Development of Forcing Functions

page 5-4 the first two sentences before 5.2.1 were added.
page 5-6 table numbers changed.
page 5-6 Doel 3 changed to Doel 4 in the sixth section.
page 5-7 table number changed.
page 5-7 added superscripted note 1.
page 5-8 table numbers changed.
page 5-8 "A tentative pump" was replaced with "analyzed AP1000 RCP."
page 5-8 added "(current at the time the original calculations were carried out)."
page 5-9 the value of "f" was changed from 1750 rpm to 1780 rpm.
page 5-9 table number changed.
page 5-9 changed sentence from "Evaluation of the RCP loads during heat up are not expected to cause unacceptably high stress. These loads are considered in the design of the AP1000 reactor internals" to "Evaluation of the RCO loads during heat up is not expected to cause unacceptably high stresses because the pressure differences are small and exist for relatively short times.
page 5-9 added sentence " The [8.25 Hz][b,c] responses at Doel 4 are considered to be the system fundamentals acoustic mode."
page 5-9 added word "corresponding" to last sentence in first paragraph of 5.2.3
page 5-10 changed "(Reference 5-8)" to "[1-10]."
page 5-11 added Table 5-1 Comparison of Core Shroud Natural Frequencies due to Reduction of Brace Thickness.
page 5-11 table number changed.
page 5-12 table number changed.
page 5-12 value was added for Ratio of Calculated to Test for the Reactor Vessel.
page 5-13 table number changed.
page 5-14 table number and title changed.
page 5-14 Frequency (Hz) changed in table.
page 5-14 added note No. 1.
page 5-15 table number and title changed.
page 5-15 Frequency (Hz) changed in table.
page 5-15 added note No. 1.
page 5-16 table number and title changed.
page 5-16 Frequency (Hz) changed in table.
page 5-16 added note No. 1.
page 5-17 table number and title changed.
page 5-18 table number changed.
page 5-18 added Table 5-9b Updated Operating Conditions for AP1000 Plant/Reactor Coolant Pump Startup from Cold Conditions Using Variable Frequency Drives-All Four Pumps Running.[1]

page 5-19 table number changed.
page 5-24 "Doel 4" added to title.
page 5-25 "Doel 4" added to title.
page 5-26 "Doel 4" added to titles.
page 5-27 "Doel 4" added to titles.
page 5-27 graph 5-8a and 5-8 b switched places.

6.2 RMS Displacements
page 6.2 re-worded last paragraph in section 6.2.

6.3 High-Cycle Fatigue of Critical AP1000 Reactor Internals Components
page 6.2 added all information before 6.3.1.
page 6.2 re-worded paragraph in 6.3.1.1.
page 6.3 re-worded paragraph in 6.3.1.2.
page 6.3 added 6.3.1.3 Key/Clevis.
page 6-3 re-worded last two sentences in 6.3.1.4.
page 6-3 added 6.3.2.1 Upper Support Assembly.
page 6-3 Upper Support Skirt and Flange omitted.
page 6-4 "3XL" changed to "Doel 3."
page 6-4 added sentence, "However, since the AP1000 guide tube frequencies are not near the pump frequencies, the RCP-induced vibrations are negligible" to "The RCP-induced excitations were added algebraically at each location."
page 6-4 re-worded sentence after the previously aforementioned change.
page 6-5 "3XL" changed to "Doel 3."
page 6-5 "Guide Tube High-Cycle Fatique Analysis" section completely re-worked with new information.
page 6-5 "Support Column High-Cycle Fatigue Analysis" section, "fatigue margins" was changed to "fatigue factor of safety."
page 6-5 minimum fatigue factor of safety was 2.1, now is [3.1][b,c]
page 6-5 added table reference in section 6.3.2.3.
page 6-7 added section 6.3.2.4 Upper Core Plate/Core Barrel Interface.
page 6-7 added "Functional Requirements" to "6.4 Interface Loads" section title.
page 6-7 changed sixteen bolts to twelve bolts in section 6.4.2.
page 6-7 added sentence "The same total load derived with sixteen connections will be modified for the twelve bolt design" in section 6.4.2.
page 6-7 added last paragraph in section 6.4.2.
page 6-7 table reference was added in section 6.4.3.1.
pages 6-8 & 6-9 section 6.4.4 "Core Barrel Interface Loads" was added.
page 6-10 title and values changed in Table 6.2.
page 6-10 values changed in Table 6.3.
page 6-10 values changed in Table 6.4.
page 6-11 title and values changed in Table 6-5.
page 6-11 title and values changed in Table 6-6.
page 6-12 added notes No. 2, 3 & 4 in Table 6-7.
page 6-12 changed value in "Sum of forces on all bolts" row in Table 6-7.
Page 6-13 deleted table "Net Preload Acting on Lower Flange of Core Shroud per Bolt for Hot, Full Power."

7 Preoperational Internals Vibration Measurement Program
page 7-1 added "Revision 2" to "NRC Regulatory Guide 1.20."
page 7-1 changed "top-mounted instrumentation guide tube conduits" to "UMIA conduits."

7.1 Locations of Transducers
pages 7-1 & 7-2 several changes to most of the location and types of transducers.

7.2 Transducers and Data Acquisition Equipment
page 7-3 added "ceramic insulated" in second paragraph.

7.3 Equipment Calibration
page 7-4 reformatted section.

7.5 Hot Functional Test Conditions
page 7-4 changed temperature from 529°F to 556°F.

7.6 Predicted Responses
page 7-5 added reference information.

7.8 Test Program Summary
page 7-6 re-worded last sentence in second paragraph.
page 7-7 several changes within Table 7-1 AP1000 Transducer Locations.

8 Pre-and Post-Hot Functional Test Inspection
page 8-1 re-worded parts of this section.
page 8-1, 8-2 & 8-3 added Table 8-1 Pre- and Post-Hot Functional Test Examination and Figure 8-1 Pre- and Post-Hot Functional Test Examination Locations.

9 Conclusions
page 9-1 added "Revision 2."

10 References
pages 10-1 & 10-2 deleted Ref. 1-11, 5-8 & 6-2.
pages 10-1, 10-2 & 10-3 added and/or changed Ref. 1-11, 4-1, 4-3, 6-4 & 7-2.

Design Control Document (DCD) Revision: None.

PRA Revision: None.

Technical Report (TR) Revision: None.

AP1000 TECHNICAL REPORT REVIEW

Response to Request For Additional Information (RAI)

RAI Response Number: RAI-SRP3.9.2-EMB1-07
Revision: 0

Question:

Discuss the potential for generation of vortices in the region of the flow skirt due to the presence of these welded joints as well as the flow skirt itself and the potential adverse effects on the response of other internals components. Also discuss any tests related to the evaluation of the flow skirt performance.

Westinghouse Response:

Any vortices in this region would be proportional in size to the minimum open dimension between the vessel and the flow skirt. This will be on the order of 0.376 inch. Any vortices generated will therefore be too small and of too high a frequency (frequency is proportional to velocity divided by vortex dimension) to be of concern. If anything, the flow skirt will tend to dissipate any larger vortices that may be produced by the flow around the radial keys.

The fact that the flow skirt makes the lower plenum flow field more uniform is an additional benefit. Because of this, there is a diminished possibility of large velocity gradients entering the lower plenum from the vessel down comer. Lower velocity gradients (greater flow uniformity) also diminish the probability of large vortex-formation.

Flow skirts of similar design have been successfully used in operating System-80 plants.

A scale model flow test, which includes the flow skirt and its connections to the reactor vessel is planned as a confirmatory test.

Design Control Document (DCD) Revision: None.

PRA Revision: None.

Technical Report (TR) Revision: None.

AP1000 TECHNICAL REPORT REVIEW

Response to Request For Additional Information (RAI)

RAI Response Number: RAI-SRP3.9.2-EMB1-08
Revision: 0

Question:

Discuss the redundancy in the instrumentation proposed for the AP1000 reactor internals preoperational test program. Based on past experience related to testing of reactor internals, instrument failures do occur during testing. Therefore, it is prudent to provide redundancy in the data acquisition process.

Westinghouse Response:

Some redundancy is included in the number, location, and types of transducers installed during the Hot Functional Test program. For example both accelerometers and strain gages are installed on the core barrel, which provides some redundancy in the event that an individual transducer would fail. In previous prototype tests conducted by Westinghouse the instrument failures were not of sufficient quantity to preclude drawing the needed conclusions.

The transducers are installed on the reactor internals and subjected to known static and dynamic inputs prior to the Hot Functional Test. These calibration tests relate displacements to measured strains and accelerations and this data is used to interpret the mean flow loads and flow-induced vibration amplitudes. The operability of these transducers is also verified during these static and dynamic calibration tests.

In addition, some redundancy is included in the interpretation of the results in that a narrow band response centered on a particular frequency can be associated with a particular mode and the damping of that mode. This enables the stress distribution associated with this mode to be used to completely describe the stresses related to this mode.

Design Control Document (DCD) Revision: None.

PRA Revision: None.

Technical Report (TR) Revision: None.

AP1000 TECHNICAL REPORT REVIEW

Response to Request For Additional Information (RAI)

RAI Response Number: RAI-SRP3.9.2-EMB1-09
Revision: 0

Question:

Provide the following topical reports, which relate to preoperational test programs for the Trojan 1 and Doel 4 plants that are referenced in the AP1000 DCD Revision 16:

> WCAP-8766, Verification of Neutron Pads and 17x17 Guide Tube Designs by Preoperational Tests on the Trojan 1 Power Plant
>
> WCAP-10846, Doel 4 Reactor Internals Flow-induced Vibration Measurement Program

Also, provide test data from the core shroud at the Yonggwang 4 plant, which is relevant to the evaluation of the AP1000 reactor internals.

Westinghouse Response:

The two WCAPs and the Yonggwang core shroud test report will be made available for review at the Westinghouse Rockville office.

Design Control Document (DCD) Revision: None.

PRA Revision: None.

Technical Report (TR) Revision: None.

RAI Response Number: RAI-SRP3.9.2-EMB1-10
Revision: 0

Question:

Provide analytical or test data to quantitatively validate the following conclusion in Section 5.1.3.1 of WCAP-15949:

"The flow area including the addition of the neutron panels, increased vessel diameter, and different specimen basket design is increased by approximately 4 percent. This is expected to offset the turbulence and increase in local velocities generated by the presence of the neutron panels. Due to the addition of a flow skirt to the lower head of the reactor vessel, the excitations of the structures in the lower vessel head plenum are likely to be lower, also contributing to a lower core barrel vibration."

Westinghouse Response:

All previous test data show that, for a given geometry and inlet flow pattern, the turbulence excitation decreases-usually by an exponent greater than 2-with decreased flow rate.

Design Control Document (DCD) Revision: None.

PRA Revision: None.

Technical Report (TR) Revision: None.

RAI Response Number: RAI-SRP3.9.2-EMB1-11
Revision: 0

Question:

Discuss how the uncertainties associated with acoustic analysis have been factored in the results of the updated calculations. Section 5.1.3.1 of WCAP 15949 states:

"The impact of the results of the updated calculations has been addressed in the individual component analyses for the guide tube, upper support column, core barrel, and core shroud. The reactor internals were evaluated for the RCP startup conditions shown in Table 5-9a. The updated reactor conditions are shown in Table 5-9b. Note that the updated conditions are less severe since the time to reach hot standby is the same for the new and old conditions but the flow rates during heat-up are lower for the new conditions. Therefore, fluid velocities are lower for the updated startup conditions than for the evaluated startup conditions. Lower flow rates would result in lower flow turbulence loads."

In order to evaluate the impact on predicted pressure differences due to the design changes, an updated ACSTIC calculation was performed. However, simplifying assumptions were made in the acoustic modeling. Therefore the above conclusions are not necessarily valid unless adequate justification is provided that the uncertainties associated with the ACSTIC calculation have been taken into consideration.

Westinghouse Response:

The uncertainties associated with the ACSTIC calculation were considered by employing a general design basis in which the RCP-related responses are taken to be coincident with natural frequency if the natural frequency is within ±10 % of the RCP excitation frequency. The calculated maximum forces from this resonance condition were then utilized in the reactor internals component structural evaluation.

Design Control Document (DCD) Revision: None.

PRA Revision: None.

Technical Report (TR) Revision: None.