ENCLOSURE 1

APP-GW-GLN-012, Revision 0

"AP1000 Reactor Internals Design Changes"

Technical Report 29

Westinghouse Non-Proprietary Class 3

WCAP-16716-NP
Revision 0

February 2007

AP1000 Reactor Internals Design Changes

Westinghouse

F-AP-3.2-1 Rev. 0

AP1000 DOCUMENT COVER SHEET

TDC: ______ Permanent File: ______ APY: ______

RFS#: ______ RFS ITEM #: ______

AP1000 DOCUMENT NO.	REVISION NO.		ASSIGNED TO
APP-GW-GLN-012	0	Page 1 of ~~24~~ 23 (RA 2-13-07)	W-Quinn

ALTERNATE DOCUMENT NUMBER: WCAP-16716-NP   WORK BREAKDOWN #:

ORIGINATING ORGANIZATION: ARIDA

TITLE: **AP1000 Reactor Internals Design Changes**

ATTACHMENTS:	DCP #/REV. INCORPORATED IN THIS DOCUMENT REVISION:
CALCULATION/ANALYSIS REFERENCE:	APP-GW-GEE-119, Rev. 0 APP-GW-GEE-111, Rev. 1 APP-GW-GEE-056, Rev. 0

ELECTRONIC FILENAME	ELECTRONIC FILE FORMAT	ELECTRONIC FILE DESCRIPTION
WCAP-16716-NP	MS Word, PDF	

☒ WESTINGHOUSE CLASS 3 (NON PROPRIETARY)

Class 3 Documents being transmitted to the NRC require the following two review signatures in lieu of a Form 36.

LEGAL REVIEW T. J. White	SIGNATURE/DATE J. C. Valentine 2/12/2007
PATENT REVIEW M. M. Corletti	SIGNATURE/DATE M M Corletti 2/9/07

☐ WESTINGHOUSE PROPRIETARY CLASS 2

This document is the property of and contains Proprietary Information owned by Westinghouse Electric Company LLC and/or its subcontractors and suppliers. It is transmitted to you in confidence and trust, and you agree to treat this document in strict accordance with the terms and conditions of the agreement under which it was provided to you.

ORIGINATOR David R. Forsyth	SIGNATURE/DATE David R. Forsyth 2/9/07	
REVIEWERS Dale A. Wiseman	SIGNATURE/DATE Dale A. Wiseman 2/9/07	
Donald A. Lindgren	D. A. Lindgren 2/9/2007	
VERIFIER David A. Altman	SIGNATURE/DATE 2-12-07 David A. Altman	VERIFICATION METHOD REVIEWED TEXT + REFERENCES.
AP1000 RESPONSIBLE MANAGER David R. Forsyth	SIGNATURE* David R. Forsyth	APPROVAL DATE 2/12/07 See EDMS

* Approval of the responsible manager signifies that document is complete, all required reviews are complete, electronic file is attached and document is released for use.

WCAP-16716-NP
Revision 0

AP1000 Reactor Internals Design Changes

Authors

N. R. Singleton

D. R. Forsyth

M. W. Ryan

February 2007

Verified: D. A. Altman*, Principal Engineer
Advanced Reactor Internals Design and Analysis

Approved: D. R. Forsyth*, Manager
Advanced Reactor 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

RECORD OF REVISIONS

Revision	Section	Description of Change
0	All	Initial Issue

Trademark Note:

AP1000 is a trademark of Westinghouse Electric Company LLC.

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

LIST OF ACRONYMS

3XL	3 Loop Extended Length (Belgian plants Doel 4 and Tihange 3)
CFD	Computational Fluid Dynamics
COL	Combined Construction and Operating License
DCD	Design Control Document
FSAR	Final Safety Analysis Report
FSER	Final Safety Evaluation Report
LCSP	Lower Core Support Plate
LOCA	Loss-of-Coolant Accident
TR	Technical Report
USNRC	United States Nuclear Regulatory Commission

1 INTRODUCTION

1.1 PURPOSE

The reactor internals are part of the reactor system as defined in the AP1000™ reactor system specification document. The internals consist of two basic assemblies: an upper internals assembly that is removed during each refueling operation to obtain access to the reactor core, and a lower internals assembly that can be removed, if desired, following a complete core unload. The purpose of the reactor internal components is to:

- Support, orient, and guide the core components, namely the fuel assemblies and control rod assemblies.
- Direct the main coolant flow to and from the fuel assemblies.
- Absorb control rod dynamic loads, fuel assembly loads, and other loads, and transmit these loads to the reactor vessel.
- Support in-core instrumentation within the reactor vessel.
- Convey cooling water to the core for a postulated loss-of-coolant accident (LOCA).
- Provide protection for the reactor vessel against excessive irradiation exposure from the core.
- Position and support reactor vessel irradiation surveillance specimens.

This technical report describes the major AP1000 reactor internals design changes relative to the descriptions and figures found in Revision 15 of the AP1000 Design Control Document (DCD).

1.2 BRIEF DESCRIPTION OF THE CHANGE

In order to meet the requirements of the reactor internals design specification, functional specification and the ASME B&PV Code Section III, Subsection NG, the major AP1000 reactor internals design changes relative to the descriptions and figures found in Revision 15 of the AP1000 DCD are as follows:

- Relocation of Radial Support Keys and Tapered Periphery on Lower Core Support Plate (LCSP)
- Addition of Flow Skirt to the Reactor Vessel Lower Head
- Addition of the Neutron Panels

2 APPLICABILITY DETERMINATION

This evaluation is prepared to document that the changes described above is a departure from Tier 2 information of the AP1000 Design Control Document (DCD) that may be included in plant-specific final safety analysis reports (FSARs) without prior United States Nuclear Regulatory Commission (NRC) approval.

A.	Does the proposed change include a change to:		
	1. Tier 1 of the AP1000 Design Control Document APP-GW-GL-700	☒ NO ☐ YES	(If YES prepare a report for NRC review of the changes)
	2. Tier 2 of the AP1000 Design Control Document, APP-GW-GL-700	☒ NO ☐ YES	(If YES prepare a report for NRC review of the changes)
	3. Technical Specification in Chapter 16 of the AP1000 Design Control Document, APP-GW-GL-700	☒ NO ☐ YES	(If YES prepare a report for NRC review of the changes)
B.	Does the proposed change involve:		
	1. Closure of a Combined License Information Item identified in the AP1000 Design Control Document, APP-GW-GL-700	☒ NO ☐ YES	(If YES prepare a COL item closure report for NRC review.)
	2. Completion of an ITAAC item identified in Tier 1 of the AP1000 Design Control Document, APP-GW-GL-700	☒ NO ☐ YES	(If YES prepare an ITAAC completion report for NRC review.)

3 TECHNICAL DESCRIPTION AND JUSTIFICATION

3.1 DESIGN CHANGES

3.1.1 Relocation of Radial Support Keys and Tapered Peripheral on the LCSP

The four lower radial support keys for the core barrel are currently located 45 degrees from the cardinal axes. There is also a spherical radius on the outer diameter of the LCSP. Core inlet flow distribution and reactor vessel pressure drop results from computational fluid dynamics (CFD) computer analysis showed that the core inlet flow distribution and the reactor vessel pressure drop were acceptable with a 6-degree slope on the outer diameter of the lower core support plate. Having the slope instead of the spherical radius on the outer diameter of the lower core support plate results in sufficient room for the radial support keys to be relocated to the cardinal axes, which is the preferred location. This relocation of the radial support keys eliminates the potential for interference with the core shroud attachment studs and nuts at the 45-, 135-, 225-, and 315-degree locations.

3.1.2 Addition of Flow Skirt to the Reactor Vessel Lower Head

The results of the CFD calculations using the existing structures in the lower plenum along with the LCSP flow hole geometry indicated that the core inlet flow distribution needed to be adjusted to create a more uniform core inlet flow distribution. The core inlet flow distribution was improved by the addition of a flow skirt to the lower plenum of the reactor vessel.

CFD analyses of numerous configurations of the hardware in the lower reactor vessel have been made with the objective of obtaining a core inlet flow distribution that meets specifications established by the Westinghouse fuel group. It has been determined that flow distributions that meet the requirements are obtained with a flow skirt. A flow skirt is a perforated cylinder in the lower reactor vessel head that is attached to the reactor vessel bottom head (See Figure 3-1). The flow skirt is attached to the lower head of the reactor vessel at the plant site after measurements for machining of the core barrel clevises have been completed. The attachment consists of welds across eight tabs that rest on support lugs provided on the reactor vessel lower head.

There is a circumferential weld between the spherical bottom vessel head and the conical transition to the cylindrical portion of the reactor vessel. The weld is just above the top surface of the flow skirt support lugs. There is some radial clearance between the outside of the flow skirt and the inside surface of the reactor vessel at the circumferential weld location. Examination Category B-N-2 of Section XI, Subsection IWB-2500, provides requirements for the visual (VT-3) examination of "interior attachments beyond the beltline region" of the reactor vessel. Vertical access for a pole-mounted camera is possible around the full circumference of the flow skirt with partial blockage at the four lower radial support keys located on the cardinal axes. It has been judged that the flow skirt and attachment welds could be inspected using VT-3 examinations. If any relevant condition is detected, IWB-3122 (prior to service) or IWB-3142 (in-service) provides options for correcting the condition.

3.1.3 Addition of Neutron Panels

To provide flexibility in the core design over the life of the plant, end-of-life reactor vessel fluence calculations were made assuming a radial core power distribution of higher power fuel assemblies in the outmost peripheral locations than in a normal low leakage core. To maintain the end-of-life reactor vessel fluence values at less than the maximum allowed in Regulatory Guide 1.99, neutron panels were attached to the outside diameter of the core barrel. The resulting reactor vessel fluence is 8.9E19 n/cm^2 ($E \geq 1.0$ MeV) at the end of the 60-year life. Neutron panels have been used on the recent Westinghouse reactor internals designs. They reduce the reactor vessel fluence at the circumferential locations that have the highest fluence values and provide a relatively rigid structure that has a smaller downcomer cross sectional area than a full cylinder.

The neutron panels are located at four circumferential locations where fuel assemblies are closest to the reactor vessel (0, 90, 180, and 270 degrees) as shown in Figure 4-2. Each pad covers ~30 degrees circumferentially and extends over the entire length of the active core region (14 feet). The pads are contoured to minimize the impact on the downcomer annulus flow area and to reduce the probability of vortex generation in the downcomer.

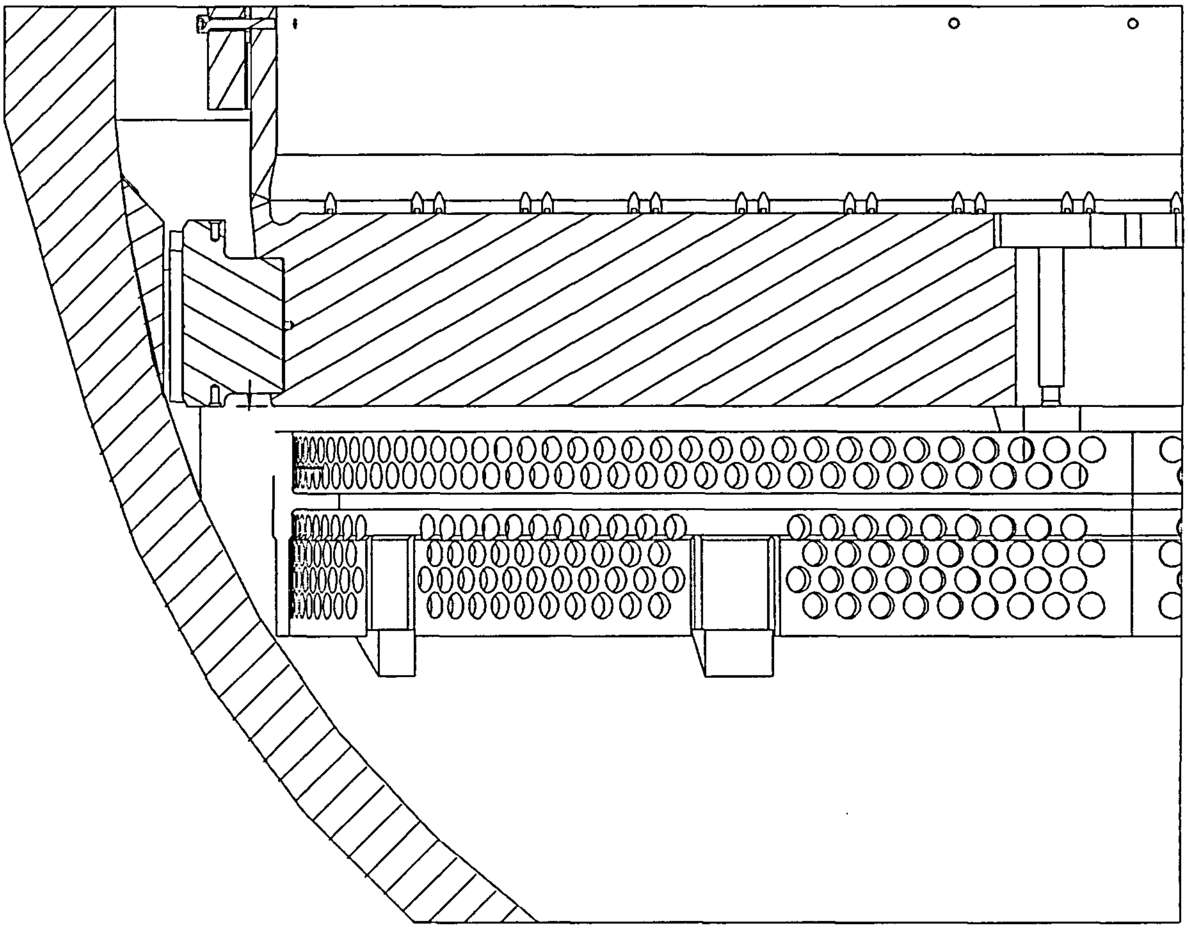

Figure 3-1 Elevation View of Reactor Bottom Vessel Head, Flow Skirt, and Lower Core Support Plate

4 DCD MARK-UP

(Note: Insertions are shown as underlined and deleted items are indicated by strikethrough text.)

Revise the ninth paragraph in subsection 3.9.2.3 as follows:

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

Successive design changes that have been incorporated into the AP1000 design since the reference plant tests have also been tested in preoperational plant vibration measurement programs, including the following:

- Inverted hat upper internals and 17x17 guide tubes at DOEL 3 and Sequoyah 1
- XL lower core support structure at DOEL 4
- ~~Elimination of reactor vessel shielding outside the core barrel at PALUEL 1~~
- Core shroud at Yonggwang 4
- Neutron panels at Trojan 1

Revise the 10th and 11th paragraphs of subsection 3.9.2.3 as follows:

These tests confirmed that the internals behaved as expected and that the vibration levels were within allowable values. The vibration testing for 17x17 fuel internals and inverted hat upper internals is reported in WCAP-8766 (Reference 4) and WCAP-8516-P (Reference 5). The vibration testing of three-loop XL type lower core support structure in DOEL 4 is reported in WCAP 10846 (Reference 6). The vibration evaluations of upper and lower internals assemblies for a four-loop XL plant~~, including reference to the test results in Paluel 1 (four-loop XL type without neutron pads),~~ are reported in WCAP-10865 (Reference 7). The vibration testing of the core shroud lower internals design is reported in Reference 13.

The results of the Doel 3~~,~~ and Doel 4~~, and Paluel 1~~ reactor internals vibration test programs will be utilized to perform the vibration assessment of the AP1000 reactor internals. The measured responses from Doel 3 and Doel 4 are adjusted to the higher AP1000 flow rate to support the determination of the expected upper internals and lower internals vibration levels respectively. The velocity through the core is approximately the same as that of Doel 4.

The results of the Trojan 1 tests showed that the lower internals vibrations are lower with neutron panels than with a circular thermal shield as reported in WCAP-8766 (Reference 4).

Revise the 18th and 19th paragraphs of 3.9.2.3 as follows.

The core barrel outside diameter and inside diameter and the reactor vessel inside diameter are the same as the tested three-loop plants. The core barrel length is 11 inches longer (~6%). Although the AP1000 coolant velocity at the inlet nozzle is higher, the coolant velocity at the elevation of the lower radial support keys is approximately the same compared to previous three-loop plants. The coolant velocity in the downcomer annulus between the core barrel and the reactor vessel wall is ~~lower~~ slightly higher in the AP1000 design than in previous three-loop plants ~~because the AP1000 has no thermal shield or neutron pads in the annulus to restrict this flow~~.

The vibrational response of the core barrel was measured during the Doel 4 reactor internals vibration measurement program. The diameter, length and thickness are nearly identical to the AP1000 core barrel and both utilize the single combined lower core support plate and neutron panels. ~~Comparison of the 4XL scale model to the Paluel plant test results indicate that the removal of the neutron panels has little effect on core barrel vibration.~~

Revise the first paragraph of 3.9.5.1.1 as follows:

3.9.5.1.1 Lower Core Support Assembly

The major containment and support member of the reactor internals is the lower core support assembly, shown in Figure 3.9-5. This assembly consists of the core barrel, lower core support plate, secondary core support, vortex suppression plate, core shroud, neutron panels, radial supports, and related attachment hardware. The major material for this structure is 300 series austenitic stainless steel. The lower core support assembly is supported at its upper flange from a ledge in the reactor vessel flange. Its lower end is restrained in its transverse movement by a radial support system attached to the vessel wall. The radial support system consists of keys attached to the lower end of the core barrel subassembly. These keys engage clevis inserts in the reactor vessel. This system restricts the lower end of the core barrel from rotational and/or translational movement, but allows for radial thermal growth and axial displacement.

Revise Figures 3.9-5, 3.9-8 and 5.3-4 as follows:

Figure 3.9-5, page 3.9-174 of DCD Revision 15: This figure is modified to show:

- A "rotated into position for clarity" statement is added for specimen basket.
- Neutron panels
- Radial keys on cardinal axes, not at 45 degrees

Figure 3.9-8, page 3.9-177 of DCD Revision 15: This figure is modified to show:

- Tapered outer diameter of the lower core support plate
- Neutron panels
- Flow skirt

Proposed changes to the Design Control Document – Tier II Section 3.9.5

3.9.5.1.4 Flow Skirt[1]

The flow skirt is a perforated cylindrical ring that is an attachment to the reactor vessel bottom head. However since this structure is located entirely within the pressure boundary, it will be described in this reactor internals section. The flow skirt is welded to support lugs on the inside surface of the reactor vessel bottom head. A vertical clearance is provided between the top of the flow skirt and the bottom surface of the lower core support plate to prevent contact during operation and postulated core drop accident conditions. The flow skirt provides a more uniform core inlet flow distribution.

3.9.5.1.~~4~~ 5 Reactor Internals Interface Arrangement[2]

Proposed change to the Design Control Document – Tier II Section 5.3

5.3.1.1 Safety Design Basis

The reactor vessel, as an integral part of the reactor coolant pressure boundary will be designed, fabricated, erected and tested to quality standards commensurate with the requirements set forth in 10 CFR 50, 50.55a and General Design Criterion 1. Design and fabrication of the reactor vessel is carried out in accordance with ASME Code, Section III, Class 1 requirements. Subsections 5.2.3 and 5.3.2 provide further details.

- The performance and safety design bases of the reactor vessel follow:
- The reactor vessel provides a high integrity pressure boundary to contain the reactor coolant, heat generating reactor core, and fuel fission products. The reactor vessel is the primary pressure boundary for the reactor coolant and the secondary barrier against the release of radioactive fission products.
- The reactor vessel provides support for the reactor internals, flow skirt, and core to ensure that the core remains in a coolable configuration.
- The reactor vessel directs main coolant flow through the core by close interface with the reactor internals and flow skirt.
- The reactor vessel provides for core internals location and alignment.
- The reactor vessel provides support and alignment for the control rod drive mechanisms and in-core instrumentation assemblies.

[1] New paragraph.
[2] Renumbered from 3.9.5.1.4.

- The reactor vessel provides support and alignment for the integrated head assembly.
- The reactor vessel provides an effective seal between the refueling cavity and sump during refueling operations.
- The reactor vessel supports and locates the main coolant loop piping.
- The reactor vessel provides support for safety injection flow paths.
- The reactor vessel serves as a heat exchanger during core meltdown scenario with water on the outside surface of the vessel.

5.3.1.2 Safety Description

4th paragraph:

> The interfaces between the reactor vessel and the lower internals core barrel are such that the main coolant flow enters through the inlet nozzle and is directed down through the annulus between the reactor vessel and core barrel, <u>and through the flow skirt</u> and flows up through the core. The annulus is designed such that the core remains in a coolable configuration for all design conditions.

5.3.2.2 Special Processes Used for Manufacturing and Fabrication

Paragraph 9[3]:

> <u>The flow skirt is also welded to support lugs in the field after the reactor vessel/internals system is set.</u>

[3] New paragraph.

Table 5.3-2 Reactor Vessel Quality Assurance Program				
	RT[a]	UT[a]	PT[a]	MT[a]
Forgings				
Flanges		Yes		Yes
Studs and nuts		Yes		Yes
CRDM head adapter tube		Yes	Yes	
Instrumentation tube		Yes	Yes	
Main nozzles		Yes		Yes
Nozzle safe ends		Yes	Yes	
Shell sections		Yes		Yes
Heads		Yes		Yes
Plates		Yes		Yes
Weldments				
Head and shell	Yes	Yes		Yes
CRDM head adapter to closure head connection			Yes	
Instrumentation tube to closure head connection			Yes	
Main nozzle	Yes	Yes		Yes
Cladding		Yes	Yes	
Nozzle to safe ends	Yes	Yes	Yes	
CRDM head adapter flange to CRDM head adapter tube	Yes		Yes	
All full-penetration ferritic pressure boundary welds accessible after hydrotest		Yes		Yes
Full-penetration nonferritic pressure boundary welds accessible after hydrotest a. Nozzle to safe ends		Yes	Yes	
Seal ledge				Yes
Head lift lugs				Yes
Core pad welds			Yes	
<u>Flow skirt support lugs weld buildup</u>		<u>Yes</u>	<u>Yes</u>	

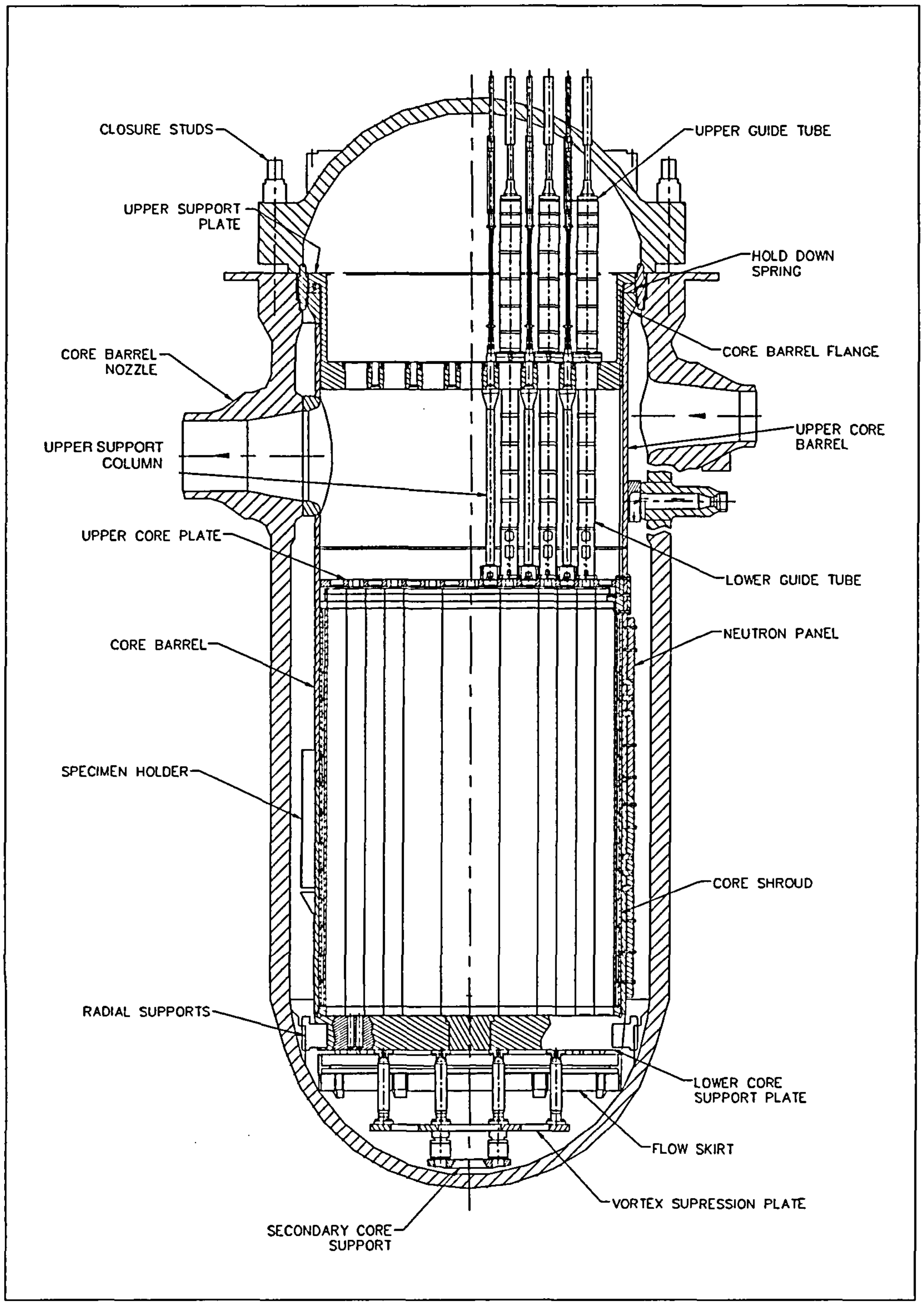

Figure 4-1 Changes to Reactor Internals Interface Arrangement (DCD, Figure 3.9-8)

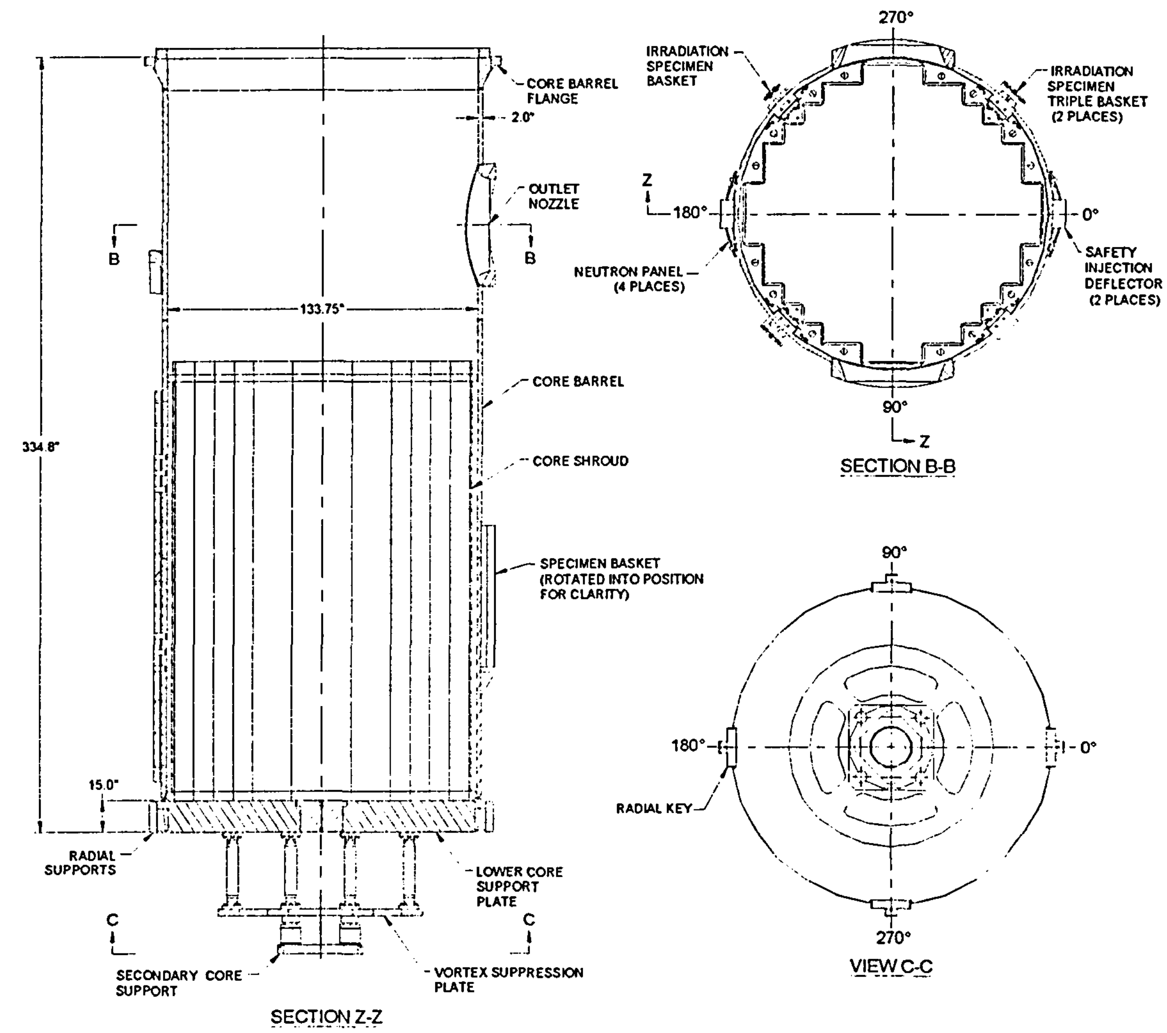

Figure 4-2 Changes to Reactor Internals Interface Arrangement (DCD, Figure 3.9-5)

5 REGULATORY IMPACT

A. FSER IMPACT

These changes are required in order to meet the design and functional requirements as prescribed in the specifications and those in the ASME Code Section III, Subsection NG.

Changes to meet design and functional requirements of the reactor internals do not impact the conclusions outlined in the NRC Final Safety Evaluation Report (FSER) or change conformances to applicable Regulatory Guides and the ASME B&PV Code Section III, Subsection NG.

Subsection 3.9.2.3 of the FSER describes the AP100 reactor vessel internal conformance with RG 1.20. The first AP1000 reactor internals design is classified as a prototype, as defined in RG 1.20. However, the applicant states that it does not consider the AP1000 reactor vessel internals a first-of-a-kind or unique design. Several units that have operating experience collectively have similar reactor vessel internals design features and are referenced in support of the AP1000 reactor vessel internals design. With the addition of neutron panels to the reactor vessel internals design the applicable referenced plant test has changed from PALUEL 1 (no reactor shielding) to Trojan 1 (similar to current neutron panel AP1000 configuration). The change in referenced plant tests will not impact the conclusion that WCAP-15949 has adequate predictive analysis of the effects of flow-induced vibration on the AP1000 reactor internals and provides adequate justification, for purposes of design certification, of the structural integrity of the conceptual design of the AP1000 reactor internals when subjected to operational flow transients.

The safety description of the reactor vessel described in subsection 5.3 of the FSER should be updated to describe the addition of the flow skirt to the core barrel-vessel wall annulus. The flow skirt design, fabrication and inspection shall conform to all of the requirements described in subsection 5.3 and therefore will not impact the conclusions drawn from the FSER.

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 change in the reactor internals design does not impact the reactor internals design functions including providing support for and maintaining the alignment of the fuel assemblies. The design function of directing reactor coolant flow though the core is not impacted.		
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 change in the reactor internals design will not affect the manner in which the plant is operated and will not require changing the normal operation of the reactor coolant system or supporting systems. The operating procedures used to startup and shutdown the plant and to respond to operational transients and postulated accident conditions are not adversely affected by the change in design of the reactor internals.		
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 change in design of the reactor internals does not adversely affect the stress analysis of the core support structures. The change in design of the reactor internals does not adversely affect the safety analyses or design evaluations of the fuel..		
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 plant, including the RCS, will not be utilized or controlled in a manner that is outside the reference bounds of the design for the plant due to the change in design of the reactor internals.		

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 change in design of the reactor internals does not change the frequency of an accident because the reactor internals are not an initiator of any accident. The change in the design of the reactor internals does not increase the initiation or progression of corrosion in primary pressure boundary materials. The change in the design of the reactor internals will not increase the frequency of accidents that may result from primary pressure boundary degradation such as pipe or tube ruptures. The change in the design of the reactor internals does not introduce a new failure mode in components that would result in an accident previously evaluated.	
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
The change in design of the reactor internals does not introduce the possibility of a change in the likelihood of a malfunction because reactor internals are not an initiator of any malfunctions. The change in the design of the reactor internals will not adversely alter heat transfer or flow rates in equipment relied on to cool or transfer reactor coolant. The change in the design of the reactor internals does not introduce a new failure mode in equipment relied upon to prevent or mitigate design basis accidents.	
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 change in the design of the reactor internals does not introduce the possibility of a change in the consequences of an accident. The change in the design of the reactor internals does not adversely change the response of the reactor coolant system and engineered safeguard systems to postulated accident conditions.	
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 change in the design of the reactor internals does not introduce the possibility of a change in the consequences of a malfunction because the change in the reactor internals will not cause pumps, valves, and heat exchangers to malfunction and result in a larger release to the environment. The change in the design of the reactor internals has no effect on systems and components used to mitigate the consequences of postulated accidents.	
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 change in the design of the reactor internals does not introduce the possibility of a new accident because the changes do not introduce a new failure mode in systems that provide fission product barriers and mitigate postulated accidents. The change in the design of the reactor internals will not change the manner in which the operator controls the plant or responds to transients or accident conditions. The change in the design of the reactor internals will not alter the response of the reactor coolant system or engineered safeguards systems to transient conditions. The change in the design of the reactor internals does not introduce the possibility of a new accident with respect to the fuel because the changes do not introduce a new failure mode in the fuel.	

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 change in the design of the reactor internals does not introduce the possibility for a malfunction of an SSC with a different result because the changes do not change the operation or function of systems and components and does not introduce a new failure mode in systems and components. Clearances and dimensions in the core are not altered by the changes in the design of the reactor internals.	
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 change in the design of the reactor internals does not result in a change that would cause a system parameter to change. The fuel performance design evaluation models are not changed by the changes in the design of the reactor internals. Therefore, the change in the design of the reactor internals does not result in a design basis limit for a fission product barrier as described in the DCD being exceeded or altered.	
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 methods used to evaluate the changes in the design of the reactor internals do not constitute a departure from a method of evaluation described in the DCD.	
☒ 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 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 systems and components identified in the DCD Subsection1.9.5 and Appendix 19 B that mitigate severe accidents are not impacted by a change in reactor internals materials.	
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 he 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 change in the design of the reactor internals will not alter barriers or alarms that control access to protected areas of the plant. The change in the design of the reactor internals will not alter requirements for security personnel. Therefore, the changes in the design of the reactor internals does not have an adverse impact on the security assessment of the AP1000.	