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Consistency of Reactor Vessel Internals Core Support Structure Materials Relative to Known Issues of Irradiation-Assisted Stress Corrosion Cracking (IASCC) and Void Swelling for the AP1000 Plant

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LIST OF ACRONYMS

COL	Combined Construction and Operating License
CSS	Core Support Structure
DCD	Design Control Document
dpa	Displacements per Atom
EFPY	Effective Full Power Years
EPRI	Electronic Power Research Institute
IASCC	Irradiation-Assisted Stress Corrosion Cracking
IS	Internals Structure
LCSP	Lower Core Support Plate
RV	Reactor Vessel
RVI	Reactor Vessel Internals
SCC	Stress Corrosion Cracking
UCP	Upper Core Plate
VS	Void Swelling

1 INTRODUCTION

1.1 AP1000 COMBINED LICENSE INFORMATION ITEM 3.9-2

This technical report addresses AP1000 Combined License Information Item 3.9-2 (FSER Action Item 3.9.2.4-1) on void swelling and irradiated-assisted stress corrosion cracking (IASCC) in reactor vessel internals core support structure materials. Internals Structures, such as the core shroud, are not considered in this report. The information item is as follows:

3.9.8.2 Design Specifications and Reports

> Combined License applicants referencing the AP1000 design will have available for NRC audit the design specifications and design reports prepared for ASME Section III components. Combined License applicants will address consistency of the core support materials relative to known issues of irradiation-assisted stress corrosion cracking or void swelling (see subsection 4.5.2.1). [*The design report for the ASME Class 1, 2, and 3 piping will include the reconciliation of the as-built piping as outlined in subsection 3.9.3. This reconciliation includes verification of the thermal cycling and stratification loadings considered in the stress analysis discussed in subsection 3.9.3.1.2.*]

This report addresses the second sentence of the COL information item. The other portions of the information item are addressed in separate AP1000 Standard COLA Technical Reports.

1.2 FIELD EXPERIENCE

The materials of the AP1000 reactor vessel internals are the same as those used in operating Westinghouse PWR reactors, [ ][a,c,e] (modified) stainless steels, Stellite 6 and Stellite 156, and nickel-base alloys UNS N07750 and N07718. While irradiation-assisted stress corrosion cracking (IASCC) is a prominent issue with BWRs, the experience in PWRs has been more favorable. In Westinghouse reactors these materials have not exhibited IASCC or void swelling (VS) problems in operation to date based on in-service inspections, however, other reactor vendors have experienced limited IASCC in bolting applications. Results from inspections of cold worked Type 316 baffle bolts from the Farley Unit 1 plant that has operated for 17 EFPY have shown no cracking (Reference 1-1). The estimated exposure of these baffle bolt heads is 20 dpa.

The evaluations below include components classified as core support structures (CSS), as identified in Reference 1-2.

2 STRESS AND dpa LEVELS

Component stresses, dpa levels and temperatures are needed for the evaluations of IASCC and void swelling. The maximum levels of these parameters for the core support structures are listed in Tables 2-1, 2-2, and 2-3.

With respect to IASCC, screening is based upon peak stress to which a component is subjected at full hot power. The peak stress would be comprised of the membrane stress intensity with additions due to bending and stress concentrations, steady state thermal stress additions and high-cycle fatigue components. Transients do not need to be considered.

For bolts whose stresses are determined by preload stresses, the stress levels after relaxation due to 3 dpa fluence were comparable to the allowable screening stress at 3 to 10 dpa. Three dpa is selected for the relaxation stress since dpa levels below 3 dpa are not considered applicable for IASCC screening.

Table 2-1 Structural Stress Levels During Normal Operation

a, c, e

Component	Location	Stress (psi)
Alignment plates	Lowest elevations	
Clevis insert (radial support key)	Bearing stress on interface	
Clevis insert (radial support key) bolts and dowel pins	Bolt shank	
Core barrel flange wall	At weld to CB	
Core barrel outlet nozzles	(Maximum A & B level)	
Core barrel to LCSP weld	Circumferential location having highest value	
Core barrel outer wall	Mid-core elevation	
Core barrel inner wall	Mid-core elevation	
LCSP fuel assembly alignment pins shank shoulder	Shank near shoulder	
LCSP fuel assembly alignment pins threads	At threads	
LCSP access port plug bolts shank shoulder	At shoulder	
LCSP access port plug bolts threads	At threads	
UCP fuel assembly alignment pins	Lowest elevations	
LCSP center region	Near center (access plug)	
LCSP periphery (rim) region	Near periphery	
Radial support keys	Weld	
UCP center region	--	
UCP periphery (outer) region, alignment plate cutout	Alignment plate cutout	
Upper support column bolts	Body to UCP	
Upper support column leg	Leg	
Upper support assembly, flange	flange (alignment pin slot cutout corner)	
Upper support assembly, skirt	skirt (skirt - USP juncture corner)	
Upper support assembly, upper support plate	upper support plate (USP) (near outmost hole)	

Notes:
1. Compressive stress.
2. Conservative; stress includes thermal transient stresses. Stresses under steady-state conditions are not presently available.
3. Adjusted for loss of preload at 3 dpa threshold

Table 2-2 dpa Levels at End-of-Life

a, c, e

Component	Location	Fluence (dpa)
Alignment plates	Lowest elevations	
Clevis insert (radial support key)	Bearing stress on interface	
Clevis insert (radial support key) bolts and dowel pins	Bolt shank	
Core barrel flange wall	At weld to CB	
Core barrel outlet nozzles	(Maximum A & B level)	
Core barrel to LCSP weld	Circumferential location having highest value	
Core barrel outer wall	Mid-core elevation	
Core barrel inner wall	Mid-core elevation	
LCSP fuel assembly alignment pins shank shoulder	Shank near shoulder	
LCSP fuel assembly alignment pins threads	At threads	
LCSP access port plug bolts shank shoulder	At shoulder	
LCSP access port plug bolts threads	At threads	
UCP fuel assembly alignment pins	Lowest elevations	
LCSP center region	Near center (access plug)	
LCSP periphery (rim) region	Near periphery	
Radial support keys	Weld	
UCP center region	--	
UCP periphery (outer) region, alignment plate cutout	Alignment plate cutout	
Upper support column bolts	Body to UCP	
Upper support column leg	Leg	
Upper support assembly, flange	flange (alignment pin slot cutout corner)	
Upper support assembly, skirt	skirt (skirt - USP juncture corner)	
Upper support assembly, upper support plate	upper support plate (USP) (near outmost hole)	

Table 2-3 Estimated Structural Temperatures During Normal Reactor Operation

a, c, e

Component	Location	Temp. (°F)
Alignment plates	Lowest elevations	
Clevis insert (radial support key)	Bearing stress on interface	
Clevis insert (radial support key) bolts and dowel pins	Bolt shank	
Core barrel flange wall	At weld to CB	
Core barrel outlet nozzles	(Maximum A & B level)	
Core barrel to LCSP weld	Circumferential location having highest value	
Core barrel outer wall	Mid-core elevation	
Core barrel inner wall	Mid-core elevation	
LCSP fuel assembly alignment pins shank shoulder	Shank near shoulder	
LCSP fuel assembly alignment pins threads	At threads	
LCSP access port plug bolts shank shoulder	At shoulder	
LCSP access port plug bolts threads	At threads	
UCP fuel assembly alignment pins	Lowest elevations	
LCSP center region	Near center (access plug)	
LCSP periphery (rim) region	Near periphery	
Radial support keys	Weld	
UCP center region	--	
UCP periphery (outer) region, alignment plate cutout	Alignment plate cutout	
Upper support column bolts	Body to UCP	
Upper support column leg	Leg	
Upper support assembly, flange	flange (alignment pin slot cutout corner)	
Upper support assembly, skirt	skirt (skirt - USP juncture corner)	
Upper support assembly, upper support plate	upper support plate (USP) (near outmost hole)	

3 SCREENING OF CORE SUPPORT STRUCTURE MATERIALS FOR IRRADIATION-ASSISTED STRESS CORROSION CRACKING

3.1 METHODOLOGY

The core support structures are evaluated for irradiation-assisted stress corrosion cracking. EPRI sponsored industry programs are addressing long term aging mechanisms and their affects on current nuclear power plants considering life extension. IASCC is a fluence dependent stress corrosion cracking phenomenon capable of affecting austenitic stainless steels. A PWR-specific screening criteria based on stress state and fluence (Reference 2-1) was considered in this evaluation. Satisfaction of the criteria (i.e., exceeding the stress and fluence values) does not imply that IASCC will absolutely occur, rather that it should be considered.

The IASCC screening criteria used in this report are summarized below. The levels of stress and fluence are consistent with the levels in Reference 2-1.

For fluence < 3 dpa IASCC is not considered applicable
Stress ≥ 62 ksi and 3 dpa ≤ fluence ≤ 10 dpa
Stress ≥ 46 ksi and 10 dpa < fluence ≤ 20 dpa
Stress ≥ 30 ksi and 20 dpa < fluence ≤ 40 dpa

A conservative approach to the criteria has been used as the following examples exhibit: if the fluence is 5 dpa, the corresponding stress threshold considered would be 62 ksi; if fluence is 15 dpa, the corresponding stress threshold would be 46 ksi.

3.2 RESULTS/CONCLUSIONS

As shown in Table 3-1, various core support structures are listed with respect to their expected maximum fluence (from Table 2-2) and maximum or peak stress that a component is subject to at hot, full-power condition (from Table 2-1). The peak stress is comprised of the membrane, bending, and thermal stress augmented by effects of the stress concentrations and stress relaxation, where applicable. For example, the maximum fluence listed for the Core Barrel Outer Wall is listed as [ ] a, c, e; the threshold stress from the next higher fluence level (20 dpa) is 46 ksi. Utilizing the IASCC screening criteria (see page 6) and performing evaluations and assessments on the AP1000 core support structures, it was concluded that IASCC is not a potential degradation concern for the core support structures for the AP1000 plant.

Table 3-1 Comparison of IASCC Threshold Stresses to Operating Stresses

a, c, e

Structure	Maximum Fluence (dpa)	Maximum Stress (psi)	Threshold Stress (psi)	Above Threshold?
Alignment plates			N/A(1)	No
Clevis inserts (radial support key)			N/A(1)	No
Clevis insert (radial support key) bolts and dowel pins			N/A(1)	No
Core barrel flange wall			N/A(1)	No
Core barrel outlet nozzles			N/A(1)	No
Core barrel to LCSP weld			N/A(1)	No
Core barrel outer wall			46,000	No
Core barrel inner wall			30,000	No
LCSP fuel assembly alignment pins shank shoulder			62,000	No
LCSP fuel assembly alignment pins threads			62,000	No
LCSP access port plug bolts shank shoulder			62,000	No
LCSP access port plug bolts threads			N/A(1)	No
UCP fuel assembly alignment pins			N/A(1)	No
LCSP center region			62,000	No
LCSP periphery (rim) region			62,000	No
Radial support keys			N/A(1)	No
UCP center region			N/A(1)	No
UCP periphery (outer) region, alignment plate cutout			N/A(1)	No
Upper support column bolts			N/A(1)	No
Upper support column leg			N/A(1)	No
Upper support assembly, flange			N/A(1)	No
Upper support assembly, skirt			N/A(1)	No
Upper support assembly, upper support plate			N/A(1)	No

Notes:
1. For fluence levels less than 3 dpa IASCC is not considered applicable, therefore no threshold stress is applicable
2. Adjusted for loss of preload at 3 dpa threshold
3. Conservative; stress includes thermal transient stresses. Stresses under steady-state conditions are not presently available.
4. Compressive stress.

4 EVALUATION OF CORE SUPPORT STRUCTURE MATERIALS FOR IRRADIATION-INDUCED VOID SWELLING

4.1 SCREENING METHODOLOGY

The core support structures are evaluated for irradiation-induced void swelling through a criteria developed by EPRI (Reference 2-1). For stainless steels and welds EPRI has established one criteria based on temperature and fluence. Simply put, if the temperature equals or exceeds 608°F during normal reactor operation and fluence equals or exceeds 20 dpa then void swelling has a potential to occur.

4.2 RESULTS

As shown in Table 2-3 a number of components exceed the temperature criteria of 608°F. For these components review of their respective fluence levels in Table 2-2 shows that no fluence levels meet the 20 dpa criteria.

4.3 CONCLUSIONS

From the above evaluations it is concluded that void swelling is not significant for core support structures.

5 REGULATORY IMPACT

5.1 FSER IMPACT

FSER Subsection 3.9.5 discusses the design of the AP1000 reactor internals. The write-up in Subsection 3.9.5 of the FSER is not impacted. The conclusions in Subsection 3.9.5 of the FSER are not altered. FSER Subsection 4.5.2 discusses the AP1000 reactor internals materials. The write-up in Subsection 4.5.2 of the FSER is not impacted. AP1000 COLA Technical Report APP-GW-GLN-015 incorporated additional material specifications that are not included in the FSER write-up. The conclusions in Subsection 4.5.2 of the FSER are not altered.

The information provided in this document and change to the DCD has no effect on design function. This change has no effect on analysis or analysis method. This change has no effect on procedures that control how DCD described SSC design functions are performed or controlled. This change has no effect on Tier 1 information.

The information contained in this document does not require changes to the evaluation of the response to postulated accident conditions. The information does not require changes to the structural or safety analysis of any safety related equipment.

The information contained in this document does not require an additional test or experiment or changes to testing.

5.2 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 license 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 information contained in this document does not affect resolution of a severe accident issue and does not require a license amendment based on the criteria of VIII. B. 5.c of Appendix D to 10 CFR Part 52.

5.3 SECURITY ASSESSMENT

The information and subject changes will not alter barriers or alarms that control access to protected areas of the plant. The information and subject changes will not alter requirements for security personnel. Therefore, the information and proposed change do not have an adverse impact on the security assessment of the AP1000.

6 DCD MARKUP

The following DCD markups identify how COL application FSARs should be prepared to incorporate the subject change.

Revise Subsection 3.9.8.2 as follows:

3.9.8.2 Design Specifications and Reports

Combined License applicants referencing the AP1000 design will have available for NRC audit the design specifications and design reports prepared for ASME Section III components. The consistency of the core support structure materials relative to known issues of irradiation-assisted stress corrosion cracking or void swelling has been evaluated and addressed in APP-GW-GLR-035, (Reference 21) [*The design report for the ASME Class 1, 2, and 3 piping will include the reconciliation of the as-built piping as outlined in subsection 3.9.3. This reconciliation includes verification of the thermal cycling and stratification loadings considered in the stress analysis discussed in subsection 3.9.3.1.2.*]*

Note: In COL Technical Report APP-GW-GLR-021, "AP1000 As-Built COL Information Items" the following paragraph was added to Subsection 3.9.8.2:

The final design reports including the reconciliation of the as-built piping are completed by the COL holder after the construction of the piping systems and prior to fuel load.

Add a Reference 21 to Subsection 3.9.9 as follows:

3.9.9 References

21 APP-GW-GLR-035, "Consistency of Reactor Vessel Internal Core Support Structure Materials Relative to Known Issues of Irradiation-Assisted Stress Corrosion Cracking or Void Swelling for the AP1000 Plant," July 2006.

Additional information on changes to reactor internals materials is provided in a separate technical report.

7 REFERENCES

1-1 Attachment to LTR-MSI-01-92, "Hot Cell Testing of Baffle/Former Bolts Removed from Two Lead Plants," Final Report, September 2001.

1-2 APP-MI01-Z0-101, Rev. 0, "AP1000 Reactor Internals Design Specification," July 2006.

2-1 EPRI Letter from H. T. Tang to Ted Meyers, Cheryl Boggess, Bill Gray, and Steve Fyfitch, on "Screening Criteria for PWR Internals Components Ranking and Categorization," dated November 1, 2005.