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Reactor Internals Materials Changes for the AP1000 Plant

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LIST OF ACRONYMS

3XL	Belgian plants Doel 4 and Tihange 3
COL	Combined Construction and Operating License
CPL	H.B. Robinson
DAP	McGuire 1
DCD	Design Control Document
DCP	Catawba 1
DDP	Catawba 2
GAE	Vogtle 1
GBE	Vogtle 2
IPP	Indian Point 2
RGE	Ginna
SWB	Sizewell B
TGX	South Texas 1

1 INTRODUCTION

1.1 PURPOSE

This technical report addresses AP1000 License Design Control Document on reactor internals materials changes. This report also addresses the COL Application project comments to APP-GW-GLN-015, Rev. A.

1.2 BRIEF DESCRIPTION OF THE CHANGE (WHAT IS BEING CHANGED AND WHY)

The Westinghouse AP1000 Design Control Document (DCD), Paragraph 4.5.2.1, currently specifies Type 304LN stainless materials for construction of the reactor internals. The internals, however, have been designed using Types 304, 304H, and 304L because of the extensive use and experience with these alloys by Westinghouse in previous reactor designs. DCD Paragraph 4.5.2.1 also did not completely list all of the materials used in the internals. This report provides justification for the addition of Types 304 and 304H to the materials of design and it provides the recommended revision to Paragraph 4.5.2.1, which also includes the materials not previously listed.

2 APPLICABILITY DETERMINATION

This evaluation is prepared to document that the change described above is a departure from Tier 2 information of the AP1000 Design Control Document (DCD) that may be included in plant specific FSARs without prior 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.)

☒ The questions above are answered no, therefore the departure from the DCD in a COL application does not require prior NRC review unless review is required by the criteria of 10 CFR Part 52 Appendix D Section VIII B.5.b. or B.5c

3 TECHNICAL DESCRIPTION AND JUSTIFICATION

The technical issue primarily at hand focuses on the use of 'high' carbon stainless steels (Types 304 and 304H) when 'low' carbon stainless steels (Types 304L and 304LN) are available.

The four stainless alloys are distinguished by their chemistries, with carbon and nitrogen being the only differences:

304 requires less than 0.08% carbon and a maximum of 0.10% nitrogen,

304L requires less than 0.03% carbon and a maximum of 0.10% nitrogen,

304H requires 0.04 to 0.10% carbon and a maximum of 0.10% nitrogen,

304LN requires less than 0.03% carbon, but nitrogen between 0.10 and 0.16%.

Type 304L stainless, being a lower strength grade, is unconditionally approved for use where its lower ASME Code properties permit; it should be noted that the use of 304L material is currently specified for [ ][a,c,e] in Westinghouse reactors and therefore is not used for structural components. The remaining three alloys have the same Code minimum yield strength (30 ksi) and tensile strength (75 ksi, 70 ksi for SA182 forgings greater than 5 inches thick). The H-grades of stainless (e.g., 304H, 316H, 321H, etc.) were developed many years ago to ensure better resistance to high temperature creep compared to low carbon grades by maintaining at least 0.04% carbon. Similarly, adding extra nitrogen to low carbon stainless, thus creating the LN-grade, creates materials with room and high temperature properties similar to 304SS.

With respect to the ASME Code allowables, Section II, Part D, properties for 304, 304H, and 304LN have been reviewed with the following conclusions:

For Table 2A, Section III, Class 1 Design Intensity Values; Table U, Tensile Strength Values; and Table Y-1, Yield Strength Values: values are identical for 304, 304H, and 304LN throughout each table for each respective product form (forgings per SA182, plate per SA240, etc.).

Justification for adding 304, 304H, and 304L stainless to DCD Paragraph 4.5.2.1 is that these materials have been used extensively in Westinghouse reactor designs and currently operating nuclear plants and have therefore met the URD requirement for using materials that have extensive operating plant experience (URD Volume III, Chapter 1, Section 5.3.1.8.1). Furthermore, Westinghouse has not experienced SCC issues with these materials. Westinghouse previously designated a change from 304 to 304H for certain components as the reactor design evolved over time. The following table shows a survey of early plants [ ][a,c] in which 304H was not used extensively in the internals, while later plants generally exhibit wide scale use of 304H for internals components. Also evident is the use of 304 material in which carbon was specified as [ ][a,c,e] wt.% (essentially a 304H material with an upper limit on carbon); especially in the later plants of [ ].[a,c] To be consistent with these later plants it is therefore desired to limit carbon in the 304H materials to a maximum of [ ].[a,c,e] In no case to date has Westinghouse experienced a problem with SCC in these components. In addition, the Utilities Requirements Document

(URD) requires Regulatory Guide 1.44 "Control of the Use of Sensitized Stainless Steels" to be implemented. With this as a requirement Westinghouse will require all fabricators to follow Reg. Guide 1.44, which will require the fabricators to establish maximum heat inputs with respect to maximum carbon contents for each welding process when welding stainless steel.

DCD Paragraph 4.5.2.1 has therefore been revised to include Types 304, 304H [ ],[a, c, e] and 304L stainless materials. Also added to the paragraph is the stipulation that fabricators welding on stainless materials will be required to qualify each of their welding procedures with respect to maximum carbon content and maximum heat input in accordance with Reg. Guide 1.44.

With respect to previously unlisted materials; nickel-base alloys and Stellite 6 hardfacing have been added to the paragraph. A statement has also been made that Alloy 600 would not be used in the AP1000 reactor internals.

The current peak reactor internals fluence in the DCD is consistent with the use of a low leakage core design over the entire 60-year life. The peak fluence was revised to a value of [ ][a, c, e] to reflect a more flexible core loading pattern, which would allow higher power fuel assemblies to be located in the peripheral locations. This fluence is comparable to the end-of-life fluence levels that occur in current plants whose operating life has been extended to 60 years.

Table 3-1 Table of Westinghouse Plants With 304H Materials Used in the Internals

a, c

Component												
CB flange												
LRS keys												
LCSP												
Lower support plate												
USP flange												
USP plate												
Upper core plate												
CB outlet nozzle												

Plant designations and years of commercial operation start:

[

]a, c

[ ]a, c, e

[ ]a, c, e

Blank spaces designate components for which the material was not investigated.

4 DCD MARK-UP

Revise Section 4.5.2.1 as follows

4.1 MATERIALS SPECIFICATIONS

The major core support structure material for the reactor internals is SA-182, SA-336, SA-479 or SA-240 Types 304, 304L, 304LN, or 304H stainless steels. Fabricators performing welding of any of these materials are required to qualify the welding procedures for maximum carbon content and heat input for each welding process in accordance with Regulatory Guide 1.44. For threaded structural fasteners the material used is strain hardened Type 316 stainless steel and for the clevis insert-to-vessel bolts either UNS N07718 or N07750. Remaining internals parts not fabricated from Types 304, 304L, 304LN, or 304H stainless steels typically include wear surfaces such as hardfacing on the radial keys, clevis inserts, alignment pins (Stellite™ 6 or 156 or low cobalt hardfaces); dowel pins (Type 316); hold down spring (Type 403 stainless steel (modified)); clevis inserts (UNS N06690); and irradiation specimen springs (UNS N07750~~Type 302 Stainless Steel~~). Core support structure and threaded structural fastener materials are specified in the ASME Code, Section III, Appendix I as supplemented by Code Cases N-60 and N-4. The qualification of cobalt free wear resistant alloys for use in reactor coolant is addressed in subsection 4.5.1.3.

The use of cast austenitic stainless steel (CASS) is minimized in the AP1000 reactor internals. If used, CASS will be limited in carbon (low carbon grade: L grade) and ferrite contents and will be evaluated in terms of thermal aging effects.

The estimated peak neutron fluence for the AP1000 reactor ~~vessel~~ internals has been considered in the design. Susceptibility ~~is acceptable relative to known issues of~~ to irradiation-assisted stress corrosion cracking or void swelling in reactor internals~~.~~ ~~Issues~~ identified in the current pressurized water reactor fleet are being addressed in reactor internals material reliability programs. The selection of materials for the AP1000 reactor internals considers information developed by these programs. ~~The Combined License applicant will address findings from these programs that are applicable to the AP1000 reactor internals design (see subsection 3.9.8.2).~~ Ni-Cr-Fe Alloy 600 is not used in the AP1000 reactor internals.

Note the issues of irradiation-assisted stress corrosion cracking and void swelling are addressed in APP-GW-GLR-035, "Consistency of Reactor Vessel Internals Core Support Structure Materials Relative to Known Issues of Irradiation-Assisted Stress Corrosion Cracking or Void Swelling for the AP1000 Plant."

5 REGULATORY IMPACT

A. FSER IMPACT

The addition of material specification SA-336 and materials 304, 304L, 304H, UNS N07718, N07750, and N06690, and Stellite 6 were not previously considered by the NRC. These materials are all acceptable in accordance with the ASME Code, Section III, Division I, NG-2000, and Section II. In addition, these materials are compatible with reactor coolant in accordance with the ASME Code, Section III, Division I, Subarticles NG-2160 and NG-3120. Note, while Stellite 156 was previously listed in DCD Paragraph 4.5.2.1, neither Stellite 6 nor Stellite 156 are cobalt-free hardfacing.

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 reactor internals materials 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 reactor internals materials 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 reactor internals materials.		
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 reactor internals materials does not adversely affect the stress analysis of the core support structure. The change in reactor internals materials does not adversely affect the safety analyses or design evaluation s of the fuel. The methodology for detailed evaluations of materials, including the effect of neutron fluence on materials, is not included in the DCD.		
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 reactor internals materials		

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 reactor internals materials does not change the frequency of an accident because the reactor internals are not an initiator of any accident. The change in reactor internals materials does not increase the initiation or progression of corrosion in primary pressure boundary materials. The change in reactor internals materials will not increase the frequency of accidents that may result from primary pressure boundary degradation such as pipe or tube ruptures. The change in reactor internals materials 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 reactor internals materials 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 reactor internals materials will not adversely alter heat transfer or flow rates in equipment relied on to cool or transfer reactor coolant. The change in reactor internals materials 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 reactor internals materials does not introduce the possibility of a change in the consequences of an accident. The change in reactor internals materials does not 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 reactor internals materials does not introduce the possibility of a change in the consequences of a malfunction because the change in reactor internals materials will not cause pumps, valves, and heat exchangers to malfunction and result in a larger release to the environment. The change in reactor internals materials 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 reactor internals materials does not introduce the possibility of a new accident because the change in reactor internals materials does not introduce a new failure mode in systems that provide fission product barriers and mitigate postulated accidents. The change in reactor internals materials will not change the manner in which the operator controls the plant or responds to transients or accident conditions. The change in reactor internals materials will not alter the response of the reactor coolant system or engineered safeguards systems to transient conditions. The change in reactor internals materials does not introduce the possibility of a new accident with respect to the fuel because the change in reactor internals materials does 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 reactor internals materials does not introduce the possibility for a malfunction of an SSC with a different result because the change in reactor internals materials does 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 change in reactor internals materials.	
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 reactor internals materials does not result in a change that would cause a system parameter to change. The change in reactor internals materials will have no effect on the stresses in the reactor internals. The fuel performance design evaluation models are not changed by the change in reactor internals materials. Therefore, the change in reactor internals materials 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 change in reactor internals materials do not constitute a departure from a method of evaluation described in the DCD. Detailed evaluations of materials, including the effect of neutron fluence on materials, are not included in the DCD. The methods used to evaluate materials were developed as part of material reliability programs for operating nuclear power plans.	
☒	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 reactor internals materials will not alter barriers or alarms that control access to protected areas of the plant. The change in reactor internals materials will not alter requirements for security personnel. Therefore, the change in reactor internals materials does not have an adverse impact on the security assessment of the AP1000.	