5.0 REACTOR COOLANT AND CONNECTING SYSTEMS

As explained in Chapter 5 of the United States Advanced Pressurized-Water Reactor (US-APWR) Design Control Document (DCD) Tier 2, the reactor coolant system (RCS) is a closed, four-loop system designed to transfer heat generated by the reactor core, located in the reactor vessel (RV) to the secondary side of the steam generators (SGs) to produce steam for power generation.

Chapter 5 of the Comanche Peak Nuclear Power Plant, Units 3 and 4 (CPNPP3&4) Combined License (COL) Final Safety Analysis Report (FSAR) incorporates by reference, Chapter 5, "Reactor Coolant System and Connecting Systems," of the US-APWR DCD. Luminant Generation Company and Comanche Peak Nuclear Power Company LLC (the applicant) provided supplemental information to address COL information items. CPNPP3&4 will be the first plants to incorporate Mitsubishi Heavy Industries' (MHI's) US-APWR certified design; and hence, if issued, CPNPP3&4's COLs will be the reference COLs (RCOLs).

This chapter of the NRC staff's safety evaluation report (SER) for the CPNPP3&4 COL FSAR provides the staff's review of the proposed CPNPP3&4 RCS in four sections: Section 5.1, "Summary Description"; Section 5.2, "Integrity of Reactor Coolant Pressure Boundary"; Section 5.3, "Reactor Vessel"; and Section 5.4, "Reactor Coolant System Component and Subsystem Design."

The staff is reviewing the information in DCD Chapter 5 on Docket Number 52-021. The results of the NRC staff's technical evaluation of the information related to DCD Chapter 5, incorporated by reference in the CPNPP3&4 COL FSAR, will be documented in the staff's SER on the design certification application for the US-APWR design. The SER on the US-APWR is not yet complete, and this is being tracked as part of Open Item [1-1]. The staff will update Chapter 5 of this SER to reflect the final disposition of the DC application.

5.1 Summary Description

This section of the CPNPP3&4 COL SER references a description of the RCS provided in the US-APWR DCD and summarized below. A more detailed description of the content of the COL application (COLA), and the staff's evaluation of that information are provided in Section 5.2, "Integrity of Reactor Coolant Pressure Boundary"; Section 5.3, "Reactor Vessel"; and Section 5.4, "Reactor Coolant System Component and Subsystem Design."

The RCS provides reactor cooling by transferring the heat from the reactor core to the secondary system to produce steam for the turbine. The major components of the RCS are the reactor vessel (RV), steam generators (SGs), reactor coolant pumps (RCPs), pressurizer, pressurizer relief tank, and reactor coolant pipes and valves. The RCS is either connected to or supported by several systems, including component cooling water (CCWS), chemical volume and control (CVCS), RCS high-point vent (RCSHPVS), and residual heat removal (RHRS), but of those, only the RHRS and RCSHPVS are covered in DCD Chapter 5 (Section 5.4).

Section 5.1 of the CPNPP3&4 COL FSAR incorporates by reference, with no departures or supplements, Section 5.1, "Summary Description," of the US-APWR DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[1] The NRC staff's review confirmed that there is no outstanding issue related to this section.

The staff is reviewing the information in DCD Section 5.1 on Docket Number 52-021. The results of the NRC staff's technical evaluation of the information related to the summary information on the RCS incorporated by reference in the FSAR will be documented in the staff SER on the DC application for the US-APWR. The SER on the US-APWR is not yet complete, and this is being tracked as part of Open Item [1-1]. The staff will update Section 5.1 of this SER to reflect the final disposition of the DC application design.

5.2 Integrity of the Reactor Coolant Pressure Boundary

In COL FSAR Section 5.2, "Integrity of the Reactor Coolant Pressure Boundary," (RCPB) the applicant incorporates by reference the US-APWR DCD Tier 2, Section 5.2, "Integrity of the Reactor Coolant Pressure Boundary," with supplemental information provided in order to address COL information items.

5.2.1 Compliance with Codes and Code Cases

5.2.1.1 Compliance with 10 CFR Part 50, Section 50.55a

5.2.1.1.1 Introduction

Section 5.2 of the FSAR includes Section 5.2.1.1, "Compliance with 10 CFR 50, Section 50.55a," Code and Standards." FSAR Section 5.2 incorporates by reference Section 5.2 of the US-APWR DCD, which includes Section 5.2.1.1 of the same title. DCD, Section 5.2.1.1 states that RCPB components are designed and fabricated in accordance with Section 50.55a of Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.55a), which requires compliance with the requirements for Class 1 components of the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (ASME Code). The applicable edition and addenda of the ASME Code, applied in the design of each Class 1 component are as endorsed in 10 CFR 50.55a FSAR Section 5.2.1.1 also addresses the use of ASME Code, Section XI, and the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (ASME OM Code) in accordance with 10 CFR 50.55a for CPNPP3&4 for in-service testing (IST) and inservice inspection (ISI). The classification of structures, systems, and components (SSCs) in accordance with the ASME Code is evaluated in Section 3.2.2 of this SER.

5.2.1.1.2 Summary of Application

Section 5.2.1.1 of the FSAR incorporates by reference, Section 5.2.1.1, "Compliance with 10 CFR 50, Section 50.55a," of the US-APWR DCD with supplemental information to address COL Information Item 5.2(11). The applicant responded to COL information Item 5.2(11) in FSAR Section 5.2.1.1 by replacing the third sentence of the second paragraph of DCD Section

[1] See Section 1.2.2 for a discussion on the staff's review related to verification of the scope of information to be included within a COL application that references a design certification.

5.2.1.1 with the statement that CPNPP3&4 use ASME Code editions and addenda that are the same as those specified in the US-APWR DCD.

5.2.1.1.3 Regulatory Basis

The regulatory basis for the NRC staff review of the CPNPP3&4 COLA is documented in the NRC SER on the certification of the US-APWR Standard Design.

The NRC regulations in Parts 50 and 52 of Title 10 of the *Code of Federal Regulations* (10 CFR Parts 50 and 52) provide the regulatory basis for NRC staff review of the information provided in the CPNPP3&4 COLA. The principal regulations pertaining to this section are:

1. General Design Criterion (GDC) 1, "Quality Standards and Records," of Appendix A to 10 CFR Part 50, which requires that nuclear power plant structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

2. 10 CFR 50.55a, as related to the establishment of the minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of nuclear power plant components, requires conformance with appropriate editions of published industry codes and standards.

Acceptance criteria for meeting the regulations cited above pertinent to this section are given in Section 5.2.1.1, "Compliance with 10 CFR Part 50, Section 50.55a," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR [light-water reactor] Edition," (the SRP) and Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)." They include:

1. RG 1.147, "Inservice Inspection Code Case Acceptability"

2. RG 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code"

3. The editions of and addenda to the ASME Code and the ASME OM Code specified in 10 CFR 50.55a.

5.2.1.1.4 Technical Evaluation

The NRC staff's technical evaluation of the US-APWR DCD related to ASME Code edition and addenda identified in the US-APWR DCD is documented in the NRC SER on the US-APWR Design Certification (DC) application. In that SER, the NRC staff described its evaluation of the ASME Code edition and addenda identified in the US-APWR DCD for use by a COL applicant implementing the US-APWR reactor design. In addition, the staff's review confirmed that all required COL information items have been addressed in this section of the COL application.

COL Information Item

In STD COL 5.2(11), the applicant replaced the third sentence of the second paragraph of DCD Section 5.2.1.1 with the statement that CPNPP3&4 will use ASME Code editions and addenda that are the same as those specified in the US-APWR DCD.

The NRC staff has reviewed STD COL 5.2(11) and concludes that the applicant has satisfactorily specified the ASME Codes and edition and addenda and meets the guidance found in NUREG-0800, Section 5.2.1.1, "Compliance with the Codes and Standards Rule, 10 CFR 50.55a."

The NRC staff reviewed FSAR Section 5.2.1.1 and the referenced DCD to evaluate its compliance with the requirements in 10 CFR Parts 50 and 52. The NRC staff confirmed that the information contained in the application and incorporated by reference in the US-AWPR DCD addresses the relevant information described in 10 CFR 50.55a 'Codes and standards.'

Pending completion of the NRC review of the US-APWR DC application, the NRC staff has determined that FSAR Section 5.2.1.1 appropriately incorporates by reference US-APWR DCD Tier 2, Section 5.2.1.1, in satisfying the NRC regulations for the design, fabrication, erection, testing, and inspection of plant SSCs commensurate with the importance of the safety function to be performed by referencing the use of accepted ASME Code editions and addenda. This satisfies the requirements of GDC 1 and, therefore, is acceptable.

US-APWR DCD Tier 2, Section 5.2.1.1 includes Table 5.2.1-1, "Applicable Code Addenda for RCS Class 1 Components," which indicates the applicability of ASME Code, Sections II, III, V, and XI. In **RAI 2751, Question 05.02.01.01-1**, the NRC staff requested that the COL applicant clarify in the FSAR that preservice and inservice testing of the reactor coolant pressure boundary components will be in accordance with the edition and addenda of the ASME OM Code required by 10 CFR 50.55a as described in applicable DCD sections for pumps, valves, and dynamic restraints. In a letter dated October 19, 2009, the COL applicant indicated that the FSAR would be revised to specify the ASME Code that will be applied at CPNPP3&4. In Revision 1 to the FSAR, the applicant included Section 5.2.1.1, "Compliance with 10 CFR 50, Section 50.55a," stating that CPNPP3&4 use ASME Code editions and addenda that are the same as those specified in the US-APWR DCD. The NRC staff finds that FSAR Section 5.2.1.1, Revision 1, clarifies that CPNPP3&4 will apply the editions and addenda of the ASME Code and ASME OM Code specified in the US-APWR DCD. Therefore, **RAI 2751, Question 05.02.01.01-01**, is resolved and closed.

The staff notes that as specified in US-APWR DCD, Section 5.2.1.1, the COL applicant will need to obtain approval from the NRC for the use of ASME Code editions or addenda other than the code of record specified in the US-APWR DCD.

Interface Requirements

US-APWR DCD Tier 2, Section 1.8, Table 1.8-1, "Significant Site-Specific Interfaces with the Standard US-APWR Design," identifies significant interfaces between the US-APWR standard design and the COL application. This table does not specify any interfaces related to Section 5.2.1.1 of the DCD.

5.2.1.1.5 Post-Combined License Activities

The NRC regulations in 10 CFR 50.55a(f)(4)(i) and (g)(4)(i) state that inservice testing and examinations conducted during the initial 120-month interval must comply with the requirements in the latest edition and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a(b) on the date 12 months before the date scheduled for initial fuel loading. The regulations further state that this requirement applies to a COL issued per 10 CFR Part 52 (or the optional ASME Code cases listed in RGs 1.147 and 1.192), subject to the limitations and modifications listed in 10 CFR 50.55a. NRC inspection of inservice testing and examination operational programs will be conducted by the NRC construction inspection program when these programs are available.

5.2.1.1.6 Conclusions

The NRC staff concludes that the information pertaining to FSAR Section 5.2.1.1 is within the scope of the US-APWR DC. FSAR Section 5.2.1.1 adequately incorporates by reference US-APWR DCD, Section 5.2.1.1. Therefore, the NRC staff concludes that the information provided in FSAR Section 5.2.1.1 satisfies the NRC requirements in 10 CFR Parts 50 and 52, relating to 10 CFR 50.55a compliance, and is acceptable, with the exception that it is contingent upon NRC approval of the US-APWR Design Certification application. Therefore, the staff cannot finalize its conclusion regarding the applicant’s compliance with 10 CFR 50.55a requirements relating to the RCS at this time.

The staff is reviewing the information in DCD Section 5.2.1.1 on Docket Number 52-021. The results of the NRC staff’s technical evaluation of the information related to compliance with Section 50.55a of 10 CFR Part 50, incorporated by reference in the FSAR, will be documented in the staff FSER on the DC application for the US-APWR. The FSER on the US-APWR is not yet complete, and this is being tracked as part of Open Item [1-1]. The staff will update Section 5.2.1.1 of this SER to reflect the final disposition of the DC application.

5.2.1.2 Compliance with Applicable Code Cases

5.2.1.2.1 Introduction

FSAR Section 5.2.1.2, “Compliance with Applicable Code Cases,” addresses Code cases related to the ASME Code and the ASME OM Code used to provide assurance of the integrity of the RCPB at CPNPP3&4. This section also discusses applicable NRC regulatory guides that indicate the acceptance of ASME Code cases with or without conditions.

5.2.1.2.2 Summary of Application

Section 5.2, “Integrity of Reactor Coolant Pressure Boundary,” in Revision 1 to the FSAR incorporates by reference US-APWR DCD Tier 2 (Revision 2), Section 5.2 of the same title. FSAR Section 5.2 includes Section 5.2.1.1, “Compliance with 10 CFR 50, Section 50.55a,” and Section 5.2.1.2, “Compliance with Applicable Code Cases.”

US-APWR DCD Tier 2, Section 5.2.1.2, specifies that the COL applicant will address the addition of ASME Code cases that are approved in RG 1.84, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III,” RG 1.147, “Inservice Inspection Code Case Acceptability,” and RG 1.192, “Operation and Maintenance Code Case Acceptability, ASME OM Code.” US-APWR DCD Tier 2, Table 5.2.1-2, “ASME Code Cases,” lists applicable ASME Code cases for RCPB Class 1 components.

The COL applicant provided supplemental information to address COL Information Items 5.2(1), 5.2(2), and 5.2(3) specified in the US-APWR DCD. This supplemental information is evaluated in Subsection 5.2.1.2.4 of this SER below.

The FSAR also specifies that the use of Code cases including those listed in RG 1.147 is identified in the ISI program (DCD Section 5.2.4, "Inservice Inspection and Testing of the RCPB," and Section 6.6, "Inservice Inspection of Class 2 and 3 Components"). Finally, the FSAR states that the use of Code cases including those listed in RG 1.192 is indicated in the IST program (DCD Section 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs for Pumps, Valves, and Dynamic Restraints," and Section 5.2.4).

5.2.1.2.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed within the FSER related to the DCD.

In addition, the relevant requirements of the compliance with applicable Code Cases, and the associated acceptance criteria, are given in Section 5.2.1.2 of NUREG-0800.

The regulatory basis for acceptance of the supplemental information provided by the COL applicant in response to COL information items 5.2(1), 5.2(2), and 5.2(3) is provided by the NRC regulations in 10 CFR Parts 50 and 52, including the following:

1. GDC 1, which requires that nuclear power plant SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

2. 10 CFR 50.55a, as related to the establishment of the minimum quality standards for the design, fabrication, erection, construction, testing, and inspection of nuclear power plant components, require conformance with appropriate editions of published industry codes and standards.

Acceptance criteria for meeting the applicable NRC regulations as given in SRP Section 5.2.1.2 include:

1. RG 1.84, which lists Code cases related to Section III, "Rules for Construction of Nuclear Facility Components," in the ASME Code, which are acceptable with applicable conditions for the design, fabrication, materials, and testing of components at nuclear power plants.

2. RG 1.147, which lists Code cases related to Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," in the ASME Code, which are acceptable with applicable conditions for use in the inservice inspection of nuclear power plant components and their supports.

3. RG 1.192, which lists Code cases related to the ASME OM Code oriented to operation and maintenance of nuclear power plant components, which are acceptable with applicable conditions for implementation at nuclear power plants.

4. RG 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," as it relates to the content of COL applications regarding ASME Code cases.

5.2.1.2.4 Technical Evaluation

The NRC staff reviewed Section 5.2.1.2 of the CPNPP3&4 COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the information in the COL represent the complete scope of information relating to this review topic.[2] The NRC staff's review confirmed that the information contained in the application and incorporated by reference addresses the required information relating to applicable Code Cases. Section 5.2.1.2 of the US-APWR DCD is being reviewed by the staff under Docket Number 52-021. The NRC staff's technical evaluation of the information incorporated by reference related to applicable Code Cases will be documented in the staff SER on the DC application for the US-APWR design.

The NRC staff's technical evaluation of the US-APWR DCD related to ASME Code cases is documented in the NRC SER on the US-APWR Design Certification application. In that SER, the NRC staff described its evaluation of the ASME Code cases identified in the US-APWR DCD for use by a COL applicant implementing the US-APWR design.

Contingent upon NRC approval of the US-APWR DC application, the NRC staff finds that the ASME Code cases identified in the US-APWR DCD are acceptable as specified in the applicable NRC regulatory guides, or have been reviewed and found acceptable by the staff for use in the US-APWR design. The staff further finds that compliance with the requirements of these Code cases will result in a component quality that is commensurate with the importance of the safety functions of the components and that satisfies the requirements of GDC 1 and 10 CFR 50.55a. The NRC staff stated that a COL applicant may identify within its COL application the planned use of additional Code cases provided they do not alter the staff's safety findings on the US-APWR certified design.

US-APWR DCD Tier 2, Section 5.2.1.2 specifies that the COL applicant will address the addition of ASME Code cases that are approved in RG 1.84, RG 1.147, and RG 1.192. As a replacement for these DCD provisions, FSAR Section 5.2.1.2 states that CPNPP3&4 use no Code cases listed in RG 1.84 beyond those listed in the referenced DCD. The FSAR indicates that the use of Code cases including those listed in RG 1.147 is identified in the ISI program (US-APWR DCD Tier 2, Section 5.2.4 and Section 6.6). The FSAR also states that the use of Code cases including those listed in RG 1.192 is indicated in the IST program (US-APWR DCD Tier 2, Sections 3.9.6 and 5.2.4). In **RAI 2751, Question 05.02.01.01-2**, the NRC staff requested that the COL applicant clarify that the Code cases for ASME Code, Section XI, and the ASME OM Code, currently planned to be applied at CPNPP3&4 are those specifically listed in the US-APWR DCD, or identify any additional Code cases to be used. In a letter dated October 19, 2009, the applicant confirmed that the Code Cases for ASME Code, Section XI, and the ASME OM Code currently planned to be applied at CPNPP3&4 are listed in the US-APWR DCD. The NRC staff finds that the COL applicant has clarified its application of Code cases specified in the US-APWR DCD. Therefore, **RAI 2751, Question 05.02.01.01-2**, is resolved and closed.

US-APWR DCD Tier 2, Section 5.2.1.2 states that any Code case conditionally approved in RG 1.84 for Class 1 components meets the condition established in the RG. US-APWR DCD Tier 2, Table 5.2.1-2, lists ASME Code Case N-71-18, "Additional Material for Subsection NF, Class 1, 2, 3 and MC Supports Fabricated by Welding, Section III, Division 1," for use in the design of

[2] See Section 1.2.2 for a discussion on the staff's review related to verification of the scope of information to be included within a COL application that references a design certification.

supports for specific nuclear power plant components. In **RAI 2795, Question 05.02.01.02-2**, the NRC staff requested that the COL applicant specify the components that will be fabricated using Code Case N-71-18 and the material specifications and grades that will be used. In its October 19, 2009, response to the RAI, the applicant stated that major subassemblies of supports for RCPB Class 1 components and piping in the US-APWR do not use materials listed in Code Case N-71-18. The applicant stated that minor subassemblies or devices which are considered to be part of supports for RCPB Class 1 components and piping may be purchased from vendors incorporating materials listed in Code Case N-71-18. In that no major subassemblies of supports for RCPB Class 1 components and piping will use materials listed in Code Case N-71-18 for the US-APWR, the NRC staff finds the applicant's response to be acceptable. Therefore, **RAI 2795, Question 05.02.01.02-2**, is resolved and closed.

Based on its review, the NRC staff has determined that FSAR Section 5.2.1.2 appropriately incorporates by reference US-APWR DCD Tier 2, Section 5.2.1.2, in satisfying the NRC regulations for the design, fabrication, erection, testing, and inspection of plant SSCs commensurate with the importance of the safety function to be performed by referencing the use of accepted ASME Code cases. As a result, the staff finds that compliance with the provisions of the ASME Code cases accepted in RGs 1.84, 1.147, and 1.192, or individually reviewed and accepted by the NRC staff, will result in component quality that is commensurate with the importance of the safety functions of the components at CPNPP3&4. This satisfies the requirements of GDC 1 and, therefore, is acceptable, contingent upon NRC approval of the US-APWR Design Certification application.

COL Information Items

The COL applicant addresses the COL Information Items listed in US-APWR DCD Tier 2, Section 5.2.6, "Combined License Information," that are related to Section 5.2.1.2 as follows:

COL 5.2(1): The COL applicant addresses the addition of ASME Code cases that are approved in Regulatory Guide 1.84.

> FSAR Subsection 5.2.1.2 states that CPNPP3&4 use no Code Cases listed in RG 1.84 beyond those listed in the referenced DCD.

COL 5.2(2): The COL applicant addresses Code cases invoked in connection with the inservice inspection program that are in compliance with RG 1.147.

> FSAR Subsection 5.2.1.2 states that the use of Code cases including those listed in RG 1.147 is identified in the ISI program (DCD Tier 2, Section 5.2.4 and Section 6.6).

COL 5.2(3): The COL Applicant addresses Code cases invoked in connection with operation and maintenance that are in compliance with RG 1.192.

> FSAR Subsection 5.2.1.2 states that use of Code cases including those listed in RG 1.192 is identified in the IST program (DCD Tier 2, Sections 3.9.6 and 5.2.4).

The NRC staff finds that the response to these COL information items identifies the Code Cases to be applied at CPNPP3&4, and is therefore acceptable.

Interface Requirements

US-APWR DCD Tier 2, Section 1.8, Table 1.8-1, “Significant Site-Specific Interfaces with the Standard US-APWR Design,” identifies significant interfaces between the US-APWR standard design and the COL application. This table does not specify any interfaces related to Section 5.2.1.2 of the DCD.

5.2.1.2.5 Post-Combined License Activities

The NRC regulations in 10 CFR 50.55a(f)(4)(i) and (g)(4)(i) state that inservice testing and examinations conducted during the initial 120-month interval must comply with the requirements in the latest edition and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a(b) on the date 12 months before the date scheduled for initial fuel loading under a COL issued per 10 CFR Part 52 (or the optional ASME Code cases listed in RGs 1.147 and 1.192), subject to the limitations and modifications listed in 10 CFR 50.55a. NRC inspection of inservice testing and examination operational programs will be conducted by the NRC construction inspection program when these programs are available.

5.2.1.2.6 Conclusions

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff’s review confirmed that the applicant addressed the required information relating to applicable Code Cases, and there is no outstanding information expected to be addressed in the CPNPP3&4 COL FSAR related to this section.

The NRC staff concludes that the information pertaining to CPNPP3&4 FSAR Section 5.2.1.2 is within the scope of the US-APWR Design Certification. FSAR Section 5.2.1.2 adequately incorporates by reference US-APWR DCD Tier 2, Section 5.2.1.2, with acceptable supplemental information. Therefore, the NRC staff concludes that the information provided in FSAR Section 5.2.1.2 is acceptable, contingent upon NRC approval of the US-APWR Design Certification application.

The staff is reviewing the information in DCD Section 5.2.1.2 on Docket Number 52-021. The results of the NRC staff’s technical evaluation of the information related to compliance with Section 50.55a of 10 CFR Part 50, incorporated by reference in the FSAR, will be documented in the staff FSER on the DC application for the US-APWR. The FSER on the US-APWR is not yet complete, and this is being tracked as part of Open Item [1-1]. The staff will update Section 5.2.1.2 of this SER to reflect the final disposition of the DC application

5.2.2 Overpressure Protection

Section 5.2.2 of the CPNPP3&4 COL FSAR incorporates by reference, with no departures or supplements, Section 5.2.2, “Overpressure Protection,” of the US-APWR DCD. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review.[3] The NRC staff’s review confirmed that there is no outstanding issue related to this section.

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[3] See Section 1.2.2 for a discussion on the staff’s review related to verification of the scope of information to be included within a COL application that references a design certification.

The staff is reviewing the information in DCD Section 5.2.2 on Docket Number 52-021. The results of the NRC staff's technical evaluation of the information related to the summary information on the RCS incorporated by reference in the FSAR will be documented in the staff SER on the DC application for the US-APWR. The SER on the US-APWR is not yet complete, and this is being tracked as part of Open Item [1-1]. The staff will update Section 5.2.2 of this SER to reflect the final disposition of the DC application design.

5.2.3 Reactor Coolant Pressure Boundary Materials

RCPB Materials are addressed in DCD, Revision 2, Section 5.2.3. DCD Section 5.2, which includes Section 5.2.3, is incorporated by reference in FSAR Section 5.2 in its entirety with certain supplements as noted herein. Within Section 5.2, FSAR Section 5.2.3.2.1 contains supplemental information to address COL Information Item 5.2(12) as in Section 5.2.3.2.1 of this SER below.

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to overpressure protection, and there is no outstanding information expected to be addressed in the CPNPP3&4 COL FSAR related to this section.

The staff is reviewing the information in DCD Section 5.2.3 on Docket Number 52-021. The results of the NRC staff's technical evaluation of the information related to overpressure protection incorporated by reference in the FSAR will be documented in the staff SER on the DC application for the US-APWR design. The SER on the US-APWR is not yet complete, and this is being tracked as part of Open Item [1-1]. The staff will update Section 5.2.3 of this SER to reflect the final disposition of the DC application.

5.2.3.2 Compatibility with Reactor Coolant

5.2.3.2.1 Reactor Coolant Chemistry

FSAR Section 5.2 on Integrity of the RCPB incorporates DCD Section 5.2 by reference with supplemental information. As part of FSAR Section 5.2, FSAR Section 5.2.3.2.1, "Chemistry with reactor Coolant," incorporates the corresponding DCD section by reference, but provides supplemental information which addresses Standard COL Information Item STD COL 5.2(12).

<u>COL Information Item</u>

COL 5.2(12) addresses the COL applicant's use of the Electric Power Research Institute (EPRI) "Primary Water Chemistry Guidelines" and states *" EPRI Primary Water Chemistry Guideline - The COL applicant should specify the applicable version of the EPRI "Primary Water Chemistry Guideline" that will be implemented."*

FSAR Section 5.2.3.2.1, "Chemistry with Reactor Coolant," provides supplemental information in STD COL 5.2(12) to address COL 5.2(12). The applicant replaced the second sentence of the third paragraph of DCD Section 5.2.3.2.1 with the following: "Water chemistry of the US-APWR reactor coolant will meet the latest version of the EPRI Water Chemistry Guidelines in effect at the time of COLA submittal." Subsequently, the staff determined that the applicant did not adequately address the COL Item.

The staff issued **RAI 5683, Question 05.02.03.01-1,** in which the applicant was requested to address COL item 5.2(12) by providing the version of the EPRI “Primary Water Chemistry Guidelines” that the applicant will use. In addition, the staff requested that the applicant modify the CPNPP3&4 FSAR to simply provide supplemental information to address the COL Item and not modify the COL Item as it appears in the US APWR DCD Section 5.2.3.**2.**1. The NRC staff has determined this to be an open item pending staff’s acceptance of the applicant’s RAI response**. RAI 5363, Question 05.02.03.01-1 is being tracked as Open Item 05.02.03.01-1**.

Interface Requirements

US-APWR DCD Tier 2, Section 1.8, Table 1.8-1, “Significant Site-Specific Interfaces with the Standard US-APWR Design,” identifies significant interfaces between the US-APWR standard design and the COL application. This table does not specify any interfaces related to Section 5.2.3 of the DCD.

5.2.3.2.2 Post-Combined License Activities

There are no follow-up actions identified for the COL applicant or NRC staff during the construction stage related to this topic.

5.2.3.2.3 Conclusions

On the basis of its review of the Comanche Peak COL application and the referenced DCD, the NRC staff finds that Section 5.2.3 of the US-APWR DCD pertaining to reactor coolant pressure boundary materials is completely incorporated by reference in the Comanche Peak COL application. The staff is tracking the supplemental information provided by the applicant related to COL Information Item 5.2(12), as **Open item 05.02.03.01-1**.

The NRC staff concludes the requirements of 52.79(d) have been satisfied for this section and it is, thus, acceptable.

5.2.4 Inspection and Testing of the Reactor Coolant Pressure Boundary (RCPB)

5.2.4.1 Introduction

This Section 5.2.4, of the CPNPP3&4 COL SER addresses FSAR Revision 1, Section 5.2.4.1, “Inservice Inspection and Testing Program,” and Section 5.2.4.2, “Pre-Service Inspection and Testing Program.” Components that are part of the RCPB must be designed to permit periodic inspection and testing of important areas and features to assess their structural and leak-tight integrity. Inservice inspection (ISI) programs are based on their requirements of 10 CFR 50.55a, “Codes and Standards,” in that Class 1 components, as defined in Section III of the ASME Code, meet the applicable inspection requirements set forth in Section XI of the ASME Code, “Rules for Inservice Inspection of Nuclear Power Plant Components.”

5.2.4.2 Summary of Application

COLA FSAR Revision 1, Section 5.2.4 incorporates by reference US-APWR DCD FSAR Tier 2 Section 5.2.4 with no departures. However, the applicant provided supplemental information in order to address COL information items as follows:

US-APWR COL Information Items

- STD COL 5.2(4)

The applicant provided additional information in STD COL 5.2(4) to address COL Information Item 5.2(4). The application specifies that the implementation milestones for the ISI program and the inservice testing (IST) programs are provided in Table 13.4-201. Additionally, the application states that the boric acid corrosion control program consists of visual inspection of component surfaces for evidence of leakage, removal of any boric acid residue found, assessment of the corrosion, and inspection follow-up.

- STD COL 5.2(5)

The applicant provided additional information in STD COL 5.2(5) to address COL Information Item 5.2(5). The application specifies that the PSI program complies with the Editions and Addenda of the ASME Code Section XI incorporated by reference in 10 CFR 50.55a(b) as applied to the construction of the component. The implementation milestones for the PSI and preservice testing program are provided in Table 13.4-201.

5.2.4.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed within the FSER related to the US-APWR DCD.

In addition, the relevant requirements of the ISI and testing of the RCPB and the associated acceptance criteria, are given in Section 5.2.4 of NUREG-0800.

The applicable regulatory requirements for the ISI and testing of the RCPB, are as follows:

1. GDC 1, as it relates to requirement that nuclear power plant SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed

2. GDC 32 found in Appendix A to 10 CFR Part 50, as it relates to periodic inspection and testing of the RCPB.

3. 10 CFR 50.55a, as it relates to the requirements for testing and inspecting Code Class 1 components of the RCPB as specified in Section XI of the ASME Code.

4. 10 CFR 52.79(d), which states requirements for COL applications referencing certified designs.

Acceptance criteria adequate to meet the above requirements include:

1. RG 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive Waste-Containing Components of Nuclear Power Plants,” as it relates to the quality group classification of components.

2. RG 1.84, which lists ASME Code Section III Code Cases oriented to design, fabrication,

materials, and testing, which are acceptable to the staff for implementation in the licensing of nuclear power plants.

3. RG 1.147, as it relates to ASME Code Section XI Code cases acceptable for use.

4. NRC Bulletin 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," dated March 17, 1988, as it relates to the establishment of a program to detect and correct potential RCPB corrosion caused by boric acid leaks.

5.2.4.4 Technical Evaluation

The NRC staff reviewed Section 5.2.4 of the CPNPP3&4 COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the information in the COL represent the complete scope of information relating to this review topic.[4] The NRC staff's review confirmed that the information contained in the application and incorporated by reference addresses the required information relating to ISI and testing of the RCPB. Section 5.2.4 of the US-APWR DCD is being reviewed by the staff under Docket Number 52-021. The NRC staff's technical evaluation of the information incorporated by reference related to ISI and testing of the RCPB will be documented in the staff SER on the DC application for the US-APWR design.

US-APWR COL Information Items

- STD COL 5.2(4)

In FSAR, Revision 1, Section 5.2.4.1, "Inservice Inspection and Testing Program," the applicant provided supplemental information in STD COL 5.2(4) to address COL Information Item 5.2(4). The applicant adopted the language of DCD Section 5.2.4.1, "Inservice Inspection and Testing Program," with the exception of replacing the first sentence of the fourth paragraph in DCD Section 5.2.4.1 with the statement that the implementation milestones for the ISI program and the IST programs are provided in Table 13.4-201. Additionally, the applicant added the following text after the first sentence of the fifth paragraph of DCD Section 5.2.4.1: "The boric acid corrosion control program consists of visual inspection of component surfaces for evidence of leakage, removal of any boric acid residue found, assessment of the corrosion, and inspection follow-up."

- STD COL 5.2(13)

In FSAR Revision 1, Section 5.2.4.1.1, "Arrangement and Accessibility," the applicant adopted the language of DCD Section 5.2.4.1.1, "Arrangement and Accessibility," with the exception of replacing the last paragraph of DCD Section 5.2.4.1.1 with the following: "Class 1 component design is the same as the DCD design."

- STD COL 5.2(5)

In FSAR, Revision 1, Section 5.2.4.2, "Pre-Service Inspection and Testing Program," the applicant provided supplemental information in STD COL 5.2(5) to address COL Information Item 5.2(5). The applicant adopted the language of DCD Section 5.2.4.2, "Pre-Service

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[4] See Section 1.2.2 for a discussion on the staff's review related to verification of the scope of information to be included within a COL application that references a design certification.

Inspection and Testing Program," with the exception of replacing the fourth sentence of the first paragraph in DCD Section 5.2.4.2 with a statement that the PSI program complies with the editions and addenda of ASME Code Section XI incorporated by reference in 10 CFR 50.55a(b) as applied to the construction of the component. It further stated that the implementation milestones for the PSI and preservice testing program are provided in FSAR Table 13.4-201.

The staff reviewed conformance of Section 5.2.4 of the FSAR, Revision 1 to the guidance in RG 1.206, Section C.III.1, Chapter 5, C.I.5.2.4, "Inservice Inspection and Testing Program." On the basis of its review of the FSAR, Revision 1, Section 5.2, the staff finds that FSAR Section 5.2 appropriately incorporates by reference Section 5.2 of the U.S. USAPWR DCD Tier 2, with supplemental information that is evaluated in this FSER section. The NRC staff's technical evaluation of the information incorporated by reference related to RCS Pressure Boundary Inservice Inspection and Testing will be documented in the staff's final safety evaluation related to the certification of the US-APWR design.

In Section 5.2.4 of the NRC staff's FSER for the referenced DCD, the staff concluded that the US-APWR ISI program for Code Class 1 components is acceptable and meets the requirements of 10 CFR 50.55a with regard to the preservice and inservice inspectability of these components. The specific version of the ASME Code used as the baseline code in the US-APWR DCD is the 2001 Edition, up to and including the 2003 Addenda. The staff did not identify any portions of the US-APWR ISI program for Class 1, 2 and 3 components that were excluded from the scope of the staff's review of the US-APWR design certification application. Therefore, the staff's conclusions regarding the acceptability of the US-APWR ISI program based on the 2001 Edition of the ASME Code, up to and including the 2003 Addenda, with regard to preservice and inservice inspectability of Class 1 components remains unchanged. Accordingly, the staff's evaluation of this section focused on the acceptability of the COL applicant's supplemental information and responses to US-APWR COL information items and action items. The staff's evaluation in this section also addresses the operational program aspects of the ASME Code Class 1, 2, and 3 PSI and ISI programs.

The staff reviewed STD COL 5.2(4) and 5.2(5) which state that the implementation milestones for the PSI and ISI program and the IST program are provided in FSAR Table 13.4-201. The staff reviewed Table 13.4-201 and noted that the PSI program implementation milestone was listed as prior to initial fuel load, while the ISI program implementation milestone was listed as prior to commercial service. The staff notes that these dates are consistent with the requirements of 10 CFR 50.55a, and are therefore, acceptable. However, at the COL application stage, the PSI/ISI programs are not developed, but will in fact be developed during the construction phase. Though Section 6.6 applies to Class 2 and 3 components, they still comprise portions of the PSI/ISI program which includes Class 1 components

Operational programs are specific programs required by regulations. The COL application should fully describe operational programs as defined in SECY-05-0197, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria," dated October 28, 2005. COL applicants should provide schedules for implementation milestones of these operational programs. The PSI and ISI programs are identified as operational programs in RG 1.206. As discussed in RG 1.206, a fully described PSI and ISI program should address: (1) system boundary subject to inspection; (2) accessibility; (3) examination categories and methods; (4) inspection intervals; (5) evaluation of examination results; (6) system pressure tests; (7) Code exemptions; (8) relief requests; and (9) ASME Code cases. Due to the scope of this operational program, submittal of the schedule for this program development is necessary to plan for and conduct NRC inspections during construction. During construction, the staff must be able to inspect the construction and nondestructive examination of the plant for conformance to the regulations and the ASME Code of record. Therefore, the staff is proposing the following license condition:

- License Condition (5-1) – The licensee shall submit to the Director of NRO, a schedule, no later than 12 months after issuance of the COL, that supports planning and conduct of NRC inspections of the PSI/ISI program (including augmented ISI program). The schedule shall be updated every 6 months until 12 months before scheduled fuel load, and every month thereafter until either the PSI/ISI (including augmented ISI program) have been fully implemented or the plant has been placed in commercial service, whichever comes first.

Finally, as stated in COLA Part 2, FSAR Table 13.4-201, Item 6, and COLA Part 10, the applicant proposes to include the following license condition on the preservice testing program:

- License Condition (5-2) – The licensee shall implement the preservice testing program prior to initial fuel load.

The staff issued **RAI 5677, Question 05.02.04-2**, which proposed a license condition for the applicant to submit a schedule that enables the staff to inspect the PSI/ISI program during the construction phase. Based on the acceptance of this license condition, STD COL 5.2 (4) and 5.2(5) are acceptable to the staff with respect to milestones. This is **Confirmatory Item 05.02.04-2**.

With regard to STD COL 5.2(13), the staff has determined that the applicant's information maintains accessibility described in the US-APWR DCD. Therefore, the staff finds that the application meets the guidance described in NUREG-0800, Section 5.2.4.

STD COL 5.2(4) states that the boric acid control program consists of visual inspection of component surfaces for evidence of leakage, removal, assessment and inspection follow-up. The NUREG-0800, Section 5.2.4 acceptance criterion states that the reviewer verifies that the COL applicant has established a program to detect and correct potential RCPB corrosion caused by boric acid leaks, as described in Generic Letter 88-05. The statement provided by the COL applicant did not clearly and sufficiently describe in terms of scope and level of detail to allow for a reasonable assurance finding of acceptability. Based on the above, in **RAI 2969, Question 05.02.04-1**, the staff requested additional information regarding details of the applicant's boric acid control program.

In its response to **RAI 2969, Question 05.02.04-1**, dated November 5, 2009, the COL applicant committed that its FSAR would supplement the US-APWR DCD, Subsection 5.2.4.1, with more detail in explaining the boric acid operational program. The changes would state that the boric acid corrosion control program (BACCP) for CPNPP3&4 includes procedures for determining the locations where leakage may cause degradation of the primary pressure boundary by boric acid corrosion. The procedures for controlling leakage include provisions to detect and locate small leaks using on-line leakage monitoring systems, sump monitoring, containment air cooler condensate flow rate monitoring, containment airborne particulate radioactivity monitoring, humidity, temperature, and pressure monitoring of the containment atmosphere, and/or visual inspection. If a trend indicates reactor coolant leakage, operators are trained to take action to identify possible leak locations. The changes also discussed the use of visual inspection of accessible and observable areas during system walkdowns early in an outage to detect boric acid buildup due to leakage. The BACCP also would contain methods for conducting examinations, performing engineering evaluations to establish the impact on the reactor coolant pressure boundary if leakage is located, and to establish corrective action to prevent recurrences of this type of corrosion. Based on the commitment to the changes provided by the COL applicant, the staff finds that the COL applicant has provided sufficient detail for the staff to obtain a reasonable assurance of the acceptability of the BACCP. The changes also comply with the guidelines for an acceptable program as outlined in Generic Letter 88-05 and, are therefore, acceptable to the staff. The staff will confirm that the applicant makes the appropriate changes to Section 5.2.4.1 in the next revision of the FSAR. Therefore, **RAI 2969, Question 05.02.04-1**, is being tracked as **Confirmatory Item 05.02.04-1**.

Interface Requirements

US-APWR DCD Tier 2, Section 1.8, Table 1.8-1, “Significant Site-Specific Interfaces with the Standard US-APWR Design,” identifies significant interfaces between the US-APWR standard design and the COL application. This table does not specify any interfaces related to Section 5.2.4 of the DCD.

5.2.4.5 Post-Combined License Activities

For the reasons discussed in the technical evaluation section above, the staff proposes to include the following license condition to address PSI/ISI program details:

- License Condition (5-1) – The licensee shall submit to the Director of NRO, a schedule, no later than 12 months after issuance of the COL, that supports planning and conduct of NRC inspections of the PSI/ISI program (including augmented ISI program). The schedule shall be updated every 6 months until 12 months before scheduled fuel load, and every month thereafter until either the PSI/ISI (including augmented ISI program) have been fully implemented or the plant has been placed in commercial service, whichever comes first.

Finally, as stated in the COLA Part 2, FSAR Table 13.4-201, Item 6, and COLA Part 10, the applicant proposes to include the following license condition on the preservice testing program:

- License Condition (5-2) – The licensee shall implement the preservice testing program prior to initial fuel load.

5.2.4.6 Conclusions

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff’s review confirmed that the applicant addressed the required information relating to ISI and testing of the RCPB, and with the exception of the confirmatory item identified above, there is no other outstanding information expected to be addressed in the CPNPP3&4 COL FSAR related to this section.

The NRC staff concludes that the information pertaining to CPNPP3&4 COL FSAR, Revision 1, Section 5.2.4, is within the scope of the design certification and adequately incorporates by reference USAPWR DCD, Section 5.2.4, “Inservice Inspection and Testing of Class 1 Components.”

The staff is reviewing the information in DCD Section 5.2.4 on Docket Number 52-021. The results of the NRC staff's technical evaluation of the information related to the ISI and testing of the RCPB, incorporated by reference in the CPNPP3&4 COL FSAR will be documented in the staff SER on the DC application for the US-APWR design. The SER on the US-APWR is not yet complete, and this is being tracked as part of Open Item [1-1]. The staff will update Section 5.2.4 of this SER to reflect the final disposition of the DC application.

In addition, the staff concludes that the COL applicant's supplemental information, specified by the USAPWR DCD, Section 5.2.4, conforms to the relevant guidelines in SRP Section 5.2.4 and RG 1.206, Section C.III.1, Chapter 5, and Section C.I.5.2.4, and, is thus, acceptable. Conformance to these guidelines provides an acceptable basis for satisfying in part, the requirements of GDC 32 and 10 CFR 50.55a. The staff further concludes that the CPNPP3&4 PSI and ISI programs and implementation conform to the policy established in SECY-05-0197. Conformance with these guidelines and policy provides an acceptable basis for satisfying in part, the requirements of GDC 32 and 10 CFR 50.55a.

5.2.5 RCPB Leakage Detection

5.2.5.1 Introduction

Section 5.2.5 of the FSAR describes the RCPB leak monitoring system. The RCPB leak monitoring system provides a means of detecting and, to the extent practical, identifying the source of reactor coolant leakage from the reactor coolant and associated systems.

5.2.5.2 Summary of Application

Section 5.2.5 of the FSAR incorporates by reference Section 5.2.5, "Reactor Coolant Pressure Boundary (RCPB) Leakage Detection," of the US-APWR DCD with one item of supplemental information that addresses COL Information Items 5.2(14) and 5.2(15).

5.2.5.3 Regulatory Basis

The regulatory basis of the information incorporated by reference will be addressed within the FSER related to the DCD.

The regulatory basis for review of the RCS leakage detection system is given in SRP Section 5.2.5 "Reactor Coolant Pressure Boundary Leakage Detection," Revision 2, March 2007. Staff acceptance of the leakage detection design is based on its meeting the requirements of the following:

- GDC 2, "Design Basis for Protection Against Natural Phenomena," as it relates to the capability of the design to maintain and perform its safety function following an earthquake

- GDC 30, "Quality of Reactor Coolant Pressure Boundary," as it relates to the detection, identification, and monitoring of the source of reactor coolant leakage

5.2.5.4 Technical Evaluation

The staff reviewed the RCPB leakage detection system in accordance with SRP Section 5.2.5. The NRC staff reviewed the application and checked the referenced DCD to ensure that no issue relating to this section remained for review. The results of this review identified two outstanding issues to be discussed below.

The review of CPNPP COL application is affected by the parallel review of the US-APWR design certification (DC) application. The staff's review for compliance of the RCS leakage detection system with GDC 30 is in accordance with the guidance in RG 1.45, Revision 1, "Guidance on Monitoring and Response to Reactor Coolant System Leakage." The first issue, identified by the NRC staff, is about meeting RG 1.45, Revision 1, Regulatory Position (RP) C.3.3. RG 1.45 states that procedures for converting various indications to a common leakage equivalent should be available to the operators. In a letter, dated February 20, 2009, in response to RAI 165-1967, Question 05.02.05-3**,** relating to US-APWR DCD Section 5.2.5, "Reactor Coolant Pressure Boundary (RCPB) Leakage Detection," the DC applicant (MHI) indicated that the leakage detection procedures and alarm set points should be developed by the COL applicant. Therefore, in **RAI 3457, Question 05.02.05-1**, the NRC staff requested the COL applicant to provide the information below to address RG 1.45, Revision 1, RP C.3.3.

- Provide procedures to convert the instrument indications of various leakage detection instruments (e.g., containment radioactivity monitors, containment sump level monitor, containment air cooler condensate flow rate monitor) into a common leakage rate (gpm).

- Define the alarm set points and demonstrate the set points are sufficiently low to provide an early warning for operator actions prior to approaching Technical Specification (TS) limits.

By letter dated October 26, 2009, in response to **RAI 3457, Question 05.02.05-1**, the applicant agreed to provide the procedures prior to fuel load and revised the CPNPP3&4 FSAR to address the above issue related to COL information item COL 5.2(14). However, the promised procedures were not in the FSAR. To ensure the adequacy of the procedures, the staff determined that the major elements of the procedures needed to be described in CPNPP3&4 FSAR for establishing the licensing bases in meeting GDC 30 via RG 1.45. The applicant was requested in **RAI 4089 Question, 05.02.05-3** to provide such information in the FSAR.

By letter dated February 19, 2010, in response to **RAI 4089, Question 05.02.05-3**, the applicant submitted a marked-up FSAR Section 5.2.5.9 to commit to developing the procedure for leakage detection instrument unit conversion in accordance with RG 1.45, Revision 1, RP C.3, and the applicant stated that the alarm set points will be established as part of the procedure development. Based on the commitment in the FSAR, the schedule, and the major elements of the procedures in the FSAR markup pages, the staff has determined that the RAI response is acceptable. Therefore, this commitment is being tracked as **Confirmatory Item 05.02.05-3 Part (a)**.

The second issue identified by the staff has to do with the procedures for prolonged, low-level leakage detection. In a letter, dated February 20, 2009, MHI responded to RAI 165-1967, Question 05.02.05-4, relating to US-APWR DCD Section 5.2.5. In the response, MHI stated that leakage detection procedures for prolonged, low-level leakage should be developed by the COL applicant. Therefore, in **RAI 3457, Question 05.02.05-2**, the staff requested the COL applicant provide the information below.

The operating experience at Davis Besse indicated that prolonged, low-level, unidentified leakage inside containment could cause material degradation such that it could potentially compromise the integrity of a system leading to the gross rupture of the reactor coolant pressure boundary. The COL applicant was requested to provide operating procedures that specify operator actions in response to prolonged, low-level leakage conditions that exist above normal leakage rates and below the TS limits to provide the operator sufficient time to take actions before the TS limit is reached. The procedures would include identifying, monitoring, trending, and repairing prolonged, low-level leakage. The guidance about developing such procedures for ensuring effective management of leakage, including low-level leakage, is available in RG 1.45, Revision 1, RP C.3.

By letter dated October 26, 2009, in response to **RAI 3457, Question 05.02.05-2**, regarding the procedures for prolonged, low-level leakage, the applicant agreed to provide the procedures prior to fuel load and revised CPNPP3&4 FSAR to address the issue related to COL information item COL 5.2(15). However, the procedures were not in the FSAR. To ensure the adequacy of the procedures, the staff determined that the major elements of the procedures need to be described in the FSAR for establishing the licensing bases in meeting GDC 30 via RG 1.45. The applicant was requested in **RAI 4089 Question 05.02.05-3** to provide such information in the FSAR**.**

By letter dated February 19, 2010, in response to **RAI 4089, Question 05.02.05-3**, the applicant submitted a marked-up FSAR Section 5.2.5.9 to commit to developing the procedure for prolonged, low-level leakage management in accordance with RG 1.45, Revision 1, RP C.3. Based on the commitment for the procedures, the schedule, and the major procedure elements in the FSAR markup pages, the staff has determined that the RAI response is acceptable. Therefore, this commitment is being tracked as C**onfirmatory Item 05.02.05-3 Part (b)**.

FSAR markups for Question 05.02.05-3 Part (a) and Part (b) are both in FSAR Section 5.2.5.9.

Interface Requirements

US-APWR DCD Tier 2, Section 1.8, Table 1.8-1, “Significant Site-Specific Interfaces with the Standard US-APWR Design,” identifies significant interfaces between the US-APWR standard design and the COL application. This table does not specify any interfaces related to Section 5.2.5 of the DCD.

5.2.5.5 Post-Combined License Activities

The applicant committed to have the operating procedures described in COL 5.2(14) and COL 5.2(15) prior to fuel loading. These procedures are subject to inspection prior to fuel loading.

5.2.5.6 Conclusion

The NRC staff’s review confirmed, with the exception of the confirmatory items described above, that the applicant’s information provided in the FSAR is acceptable and meets the requirements of 10 CFR Part 50, Appendix A, General Design Criteria 2 and 30.

The staff is reviewing the information in DCD Section 5.2.5 on Docket Number 52-021. The results of the NRC staff’s technical evaluation of the information related to the RCPB leak monitoring system incorporated by reference in the FSAR will be documented in the staff SER on the DC application for the US-APWR. The SER on the US-APWR is not yet complete, and this is being tracked as part of Open Item [1-1]. The staff will update Section 5.2.5 of this SER to reflect the final disposition of the DC application design.

5.3 Reactor Vessel

5.3.1 Reactor Vessel Materials

5.3.1.1 Introduction

This section of the SER presents the results of the NRC staff review of the material specifications for the CPNPP3&4 reactor vessels (RVs) and applicable attachments and appurtenances such as the shroud support, studs, control rod drive housings, vessel support skirt, stub tubes, and instrumentation housings. The specifications are reviewed for their adequacy for use in the construction of such components on the basis of the mechanical and physical properties of the materials, the effects of irradiation on these materials, their corrosion resistance, and their fabricability. Also addressed are special processes used for manufacture and fabrication of components, special methods for nondestructive examination, and special controls and special processes used for ferritic steels and austenitic stainless steels. Finally, the review covers fracture toughness, material surveillance (which will be referred to as the reactor vessel surveillance capsule program (RVSP) to avoid confusion with material surveillance programs that exist in other parts of a nuclear power plant), monitoring programs and reactor vessel fasteners.

5.3.1.2 Summary of Application

Section 5.3, “Reactor Vessel,” of CPNPP3&4 COL FSAR, Revision 1, which includes Section 5.3.1, “Reactor Vessel Materials,” incorporates by reference Section 5.3 of the US-APWR DCD, Revision 2 with no departures.

In addition, in CPNPP3&4 COL FSAR Section 5.3.1.6, the applicant provided the following supplemental information:

<u>*US-APWR COL Information Items*</u>

- STD COL 5.3(2): Reactor Vessel Material Surveillance Program

COL Information Item 5.3(2) in the DCD states that the COL applicant provides a RV material surveillance program based on information in DCD Section 5.3.1.6. The COL applicant provided supplemental information in STD COL 5.3(2) to address COL Information Item 5.3(2). Under FSAR, Revision 1, Section 5.3.1.6, “Material Surveillance,” STD COL 5.3(2) replaces the second paragraph in DCD Section 5.3.1.6 with the following:

> The reactor vessel material surveillance program is implemented as an operational program. As the reactor vessel materials do not begin to be affected by neutron fluence until the reactor begins critical operation, this program is implemented prior to initial criticality, as identified in Table 13.4-201.

- STD COL 5.3(2): Reactor Vessel Material Surveillance Program

The applicant provided additional supplemental information in STD COL 5.3(2) to address COL Information Item 5.3(2). Under FSAR Section 5.3.1.6.1, “Surveillance Capsules,” STD COL 5.3(2) replaces the last sentence in the sixth paragraph of DCD Section 5.3.1.6.1 of the same title with the following:

> The use of these standby surveillance capsules is incorporated by updating the surveillance program once sufficient data are retrieved to determine the withdrawal schedule for these capsules.

STD COL 5.3(2) also replaces the last paragraph in DCD Section 5.3.1.6.1 with the following:

> Accelerated irradiation capsules as defined in American Society for Testing and Materials (ASTM) E-185 (Ref. 5.3-24) and integrated surveillance program for multiple reactors at a single site, are not applicable at CPNPP Units 3 and 4.

In addition, after the last paragraph in DCD Section 5.3.1.6.3, “Predicted Effects of Radiation on Beltline Region Materials,” STD COL 5.3(2) adds the following text:

> A summary technical report, including test results, is submitted as specified in 10 CFR 50.4, for the contents of each capsule withdrawn, within one year of the date of capsule withdrawal unless an extension is granted by the Director, Office of Nuclear Reactor Regulation.
>
> The report includes the data required by ASTM E-185-82, as specified in paragraph III.B.1 of 10 CFR 50, Appendix H, and includes the results of the fracture toughness tests conducted on the beltline materials in the irradiated and unirradiated conditions.
>
> If the test results indicate a change in the Technical Specifications, either in the pressure-temperature limits or in the operating procedures, the expected date for submittal of the revised Technical Specifications is provided with this report.

- STD COL 5.3(3): Surveillance Capsule Orientation and Lead Factors

COL Information Item 5.3(3) in the DCD states that the COL applicant addresses the orientation and resulting lead factors for the surveillance capsules of a particular US-APWR plant. Under FSAR Section 5.3.1.6.1, “Surveillance Capsules,” the applicant provided additional information in STD COL 5.3(3) to address COL Information Item 5.3(3). STD COL 5.3(3) replaces the last sentence in the fifth paragraph in DCD Section 5.3.1.6.1with the following:.

> These lead factors and the capsule orientation shown in DCD Figure 5.3-1 are applicable for CPNPP Units 3 and 4.

The applicant also provided a license condition in Part 10 of the COL application. See SER Section 5.3.1.5, “Post-Combined License Activities,” below for a discussion of the license condition.

5.3.1.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed within the FSER related to the DCD.

In addition, the associated acceptance criteria are given in Section 5.3.1 of NUREG-0800.

The applicable regulatory requirements for RV materials are as follows:

1. GDC 1 and GDC 30 found in Appendix A to 10 CFR Part 50, as they relate to quality standards for design, fabrication, erection, and testing of structures, systems and components;

2. GDC 4, as it relates to the compatibility of components with environmental conditions;

3. GDC 14, as it relates to prevention of rapidly propagating fractures of the RCPB;

4. GDC 31, as it relates to material fracture toughness;

5. *GDC 32, as it relates to the requirements for a materials surveillance program;

6. 10 CFR 50.55a, as it relates to quality standards for design, and determination and monitoring of fracture toughness;

7. *10 CFR 50.60, “Acceptance criteria for fracture prevention measures for light water nuclear power reactors for normal operation,” as it relates to RCPB fracture toughness and material surveillance requirements of 10 CFR Part 50, Appendix G and Appendix H;

8. 10 CFR Part 50, Appendix B, Criterion XIII, as it relates to onsite material cleaning control;

9. 10 CFR Part 50, Appendix G, as it relates to materials testing and acceptance criteria for fracture toughness;

10. *10 CFR Part 50, Appendix H, as it relates to the determination and monitoring of fracture toughness;

11. 10 CFR 52.79(d), requirements for COLAs that reference a certified design; and

12. *SECY-05-0197 as it relates to fully describing an operational program

*Of the regulatory requirements applicable to RV materials, GDC 32, 10 CFR 50.60, 10 CFR Part 50, Appendix H, and SECY 05-197 are applicable to the RVSP.

Acceptance criteria adequate to meet the above requirements are related to design and construction except for RG 1.190, which is also related to the RVSP. Except for RG 1.190, the others are provided here for information only and include:

1. RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," as it relates to the control of welding in fabricating and joining safety-related austenitic stainless steel components and systems.

2. RG 1.34, "Control of Electroslag Weld Properties," as it relates to acceptable solidification patterns and impact test limits and the criteria for verifying conformance during production welding.

3. RG 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel," as it relates to acceptance criteria for compatibility of austenitic stainless steel with thermal insulation.

4. RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants," as it relates to the quality of water used for final cleaning or flushing of finished surfaces during installation.

5. RG 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components," as it relates to criteria to limit the occurrence of under-clad cracking in low-allow steel safety-related components clad with stainless steel.

6. RG 1.44, "Control of the Use of Sensitized Stainless Steel," as it relates to the compatibility of RCPB materials with the reactor coolant and the avoidance of stress corrosion cracking.

7. RG 1.71, "Welder Qualification for Areas of Limited Accessibility," as it relates to welder requalification.

8. RG 1.99, "Radiation Embrittlement of Reactor Vessel Materials," as it relates to reactor pressure vessel (RPV) fracture toughness.

9. RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," as it relates to the RPV material surveillance program.

5.3.1.4 Technical Evaluation

The NRC staff reviewed Section 5.3.1 of the CPNPP3&4 COL FSAR and checked the referenced DCD to ensure that the information contained therein satisfies the requirements of 10 CFR 52.79(d) and that any supplemental information to be provided by the COL applicant has been addressed in the COL application.

The staff reviewed conformance of Section 5.3 of the FSAR using the guidance in RG 1.206, Section C.III.1, Chapter 5, and Section C.I.5.3.1, "Reactor Vessel Materials." On the basis of its review, the staff finds that Section 5.3 of the FSAR incorporates by reference Section 5.3.1, "Reactor Vessel Materials," of the US-APWR DCD with some supplements to Section 5.3.1.6, "Material Surveillance." The DCD sections related to material specifications, special processes used for manufacturing and fabrication, special methods for nondestructive examination, special controls for ferritic and austenitic stainless steels, fracture toughness and reactor vessel fasteners are incorporated by reference without any departures and/or supplements. These topic areas were previously reviewed and approved in Section 5.3.1 of the US-APWR SER. However, Section 5.3.1.6, "Material Surveillance," has COL information items that must be addressed by the COL applicant.

Operational programs are specific programs required by regulations. The COL application should fully describe operational programs as defined in SECY-05-0197. In addition, COL applicants should provide schedules for implementation milestones for these operational programs. The RVSP is identified as an operational program in RG 1.206. This section of the SER addresses the adequacy of the RVSP description as it relates to meeting the requirements of Appendix H to 10 CFR Part 50.

RG 1.206, Section C.I.5.3.1.6, “Material Surveillance,” recommends that the additional information be included in the COL application. Specifically, this section states that the RVSP and its implementation should be described in sufficient detail to ensure that the program meets the requirements of Appendix H to 10 CFR Part 50. In addition, the application should describe the method for calculating neutron fluence for the reactor vessel beltline and the surveillance capsules. RG 1.206 lists some of the topics that should be addressed in the description of the RVSP:

- Basis for the selection of material in the program
- Number and type of specimens in each capsule
- Number of capsules and proposed withdrawal schedule in compliance with the edition of ASTM E-185 Annual Book of ASTM Standards, Part 30, referenced in Appendix H to 10 CFR Part 50
- Neutron flux and fluence calculations for vessel wall and surveillance specimens and conformance with guidance of RG 1.190
- Expected effects of radiation on vessel wall materials and basis for estimation
- Location of capsules, method of attachment, and provisions to ensure that capsules are retained in position throughout the vessel lifetime

Section 5.3.1.6 of the US-APWR DCD, which is referenced by the COL application, covers the above topics. Section 5.3.1.6.1 of the US-APWR DCD addresses how material is selected for the capsule specimens, the number and type of specimens in each capsule, the number of capsules, and the proposed withdrawal schedule. Sections 5.3.1.6.2 and 4.3.2.8 of the US-APWR DCD address neutron flux and fluence calculations. In addition, the applicant provided the staff with a technical report describing the methodology used to calculate the neutron fluence. Section 5.3.1.6.3 addresses the expected effects of radiation on vessel wall materials and the basis for estimation. Figure 5.3-1 of the US-APWR DCD provides the azimuthal location of the capsules and Section 3.9.5.1.2 states that the capsule guides are fastened to the core barrel using long, socket-head-cap screws.

In the US-APWR design, the RVSP is implemented by enclosing test specimens in six capsules which are attached to the core barrel. The capsule placement is designed so that the test specimens receive a higher rate of fluence than the reactor vessel base material. The test specimens are obtained from reactor vessel base metal, weld metal, and heat affected zone (HAZ). The base metal specimens are obtained from the pre-fabricated reactor vessel material near the locations where fracture toughness test specimens are taken and are oriented in both the longitudinal and transverse direction compared to the principal forging direction of the forging material. The weld test plates for the RVSP specimens have their principal working direction parallel to the weld line so that specimens for the HAZ are normal to the principal working direction. A minimum of 9 tensile-test specimens, 48 Charpy V-notch specimens, and 6 Compact Tension fracture toughness specimens are contained in each capsule. In **RAI 3127, Question 05.03.01-2**, the staff requested that the applicant confirm that the test specimens are obtained from material used for the reactor vessel beltline and that the test specimens are sealed in an inert environment. In its response to **RAI 3127**, **Question 05.03.01-2**, dated October 30, 2009, the applicant confirmed that test specimens are taken from material used for the reactor vessel beltline and that the capsules are sealed in an inert environment. The COL applicant provided a markup copy of its proposed revision to the FSAR that the staff found acceptable. Therefore, the staff will review Section 5.3.1.6 in the next revision of the FSAR to confirm that the COL FSAR has been revised as shown in the markup copy provided in response to **RAI 3127, Question 05.03.01-2**, which is therefore resolved and closed and is being tracked as **Confirmatory Item 05.03.01-2**

Section 5.3.1.6.1 of the US-APWR DCD discusses a recommended withdrawal schedule for general capsules. Section 5.3.1.6.1 of the US-APWR DCD also specifies that the COL applicant is to address the use of such capsules and their withdrawal schedule. Since the COL FSAR incorporates Section 5.3.1.6 by reference and since the withdrawal schedule discussed in the DCD is only a recommendation, staff requested in **RAI 4200, Question 05.03.01-3**, that the COL applicant confirm that it will follow the withdrawal schedule for general capsules recommended in the US-APWR DCD. In its response to **RAI 4200, Question 05.03.01-3**, dated February 19, 2010, the applicant confirmed that it would follow the recommended withdrawal schedule for general capsules found in the US-APWR DCD. The COL applicant provided a markup copy of its proposed revision to the FSAR that the staff found acceptable. Therefore, the staff will review Section 5.3.1.6 in the next revision of the FSAR to confirm that the COL FSAR has been revised as shown in the markup copy provided in response to **RAI 4200, Question 05.03.01-3**, which is being tracked as **Confirmatory Item 05.03.01-3**

Information about the implementation of the applicant's RVSP is provided in Part 10 of the COLA which proposes the following license condition:

- Part 10, License Conditions, Reactor Vessel Material Surveillance

The COL licensee shall implement this operational program prior to initial criticality.

The NRC staff finds that by incorporating by reference Section 5.3.1 of the US-APWR DCD, the COL applicant has appropriately described a major portion of its RVSP program. The remaining portion of the RVSP program description and its implementation should be addressed in the COL application in conjunction with COL information items. Accordingly, the staff reviewed the following COL information items:

<u>STD COL 5.3(2)</u>

The staff reviewed STD COL 5.3(2) related to the COL Information Item included under Section 5.3.4 of the COL FSAR. In Section 5.3.1.6 of the COL FSAR, this Information Item is described as follows:

> The reactor vessel material surveillance program is implemented as an operational program. As the reactor vessel materials do not begin to be affected by neutron fluence until the reactor begins critical operation, this program is implemented prior to initial criticality, as identified in Table 13.4-201.

RG 1.206 and SECY-05-0197 clarify the intent of this COL Information Item. RG 1.206 Section C.III.1, Chapter 5, C.I.5.3.1.6 provides guidelines for addressing an RVSP and states the program should be described in sufficient detail to meet the requirements of Appendix H to 10 CFR Part 50. SECY-05-0197 states the applicant should fully describe the program and identify implementation milestones. SECY-05-0197 further describes the need for COL applications to include license conditions implementing milestones and operational readiness for operational programs including the RVSP. Therefore, in **RAI 3193, Question 05.03.01-1**, the staff requested the applicant revise the COL application to include the following license conditions.

- The licensee shall implement reactor vessel material surveillance prior to initial criticality.

- The licensee will submit to the NRC a schedule, no later than 12 months after the issuance of the COL, that supports the planning for and conduct of NRC inspections of operational programs, including reactor vessel surveillance.

In its response to **RAI 3193, Question 05.03.01-1**, dated October 30, 2009, the COL applicant committed to revising Part 10 of the COL application to specify a license condition for implementation of the RVSP prior to initial criticality. This license condition was included in Revision 1 of the COL application. However, the COL applicant declined to submit a license condition for the operational program schedule. Instead, the applicant committed "to submit a schedule to the NRC that supports the planning and conduct of NRC inspections of operational programs, including the reactor vessel, surveillance program, no later than 12 months after issuance of the COL or at the start of construction as defined in 10 CFR 50.10a, whichever is later." The staff has determined that this commitment satisfactorily addresses the operational program schedule, which SECY-05-0197 recommends to be a license condition. By referencing Section 5.3.1.6 of the US-APWR DCD, as modified by the COL information items, and by specifying the implementation milestones in Table 13.4-201 of the COL FSAR, the applicant has met the guidelines of RG 1.206 and SECY-05-0197 with regard to its RVSP. Therefore, **RAI 3193, Question 05.03.01-1** is resolved and closed.

<u>STD COL 5.3(3)</u>
The staff reviewed STD COL 5.3(3) related to the COL information item included under Section 5.3.4 of the COL FSAR. In Section 5.3.1.6.1 of the COL FSAR, this Information Item is described as follows:

> These lead factors [referring to those in the US-APWR DCD] and the capsule orientation shown in DCD Figure 5.3-1 are applicable for CPNPP Units 3 and 4.

The staff has reviewed the requirements for lead factors and capsule orientation and finds that they meet the requirements of 10 CFR Part 50, Appendix H, and ASTM Standard E 185-82, which is incorporated by reference in Appendix H, and are therefore acceptable.

<u>STD COL 5.3(2)</u>

The staff reviewed STD COL 5.3(2) related to the COL information item included under Section 5.3.4 of the COL FSAR. In Section 5.3.1.6.1 of the COL FSAR, this information item is given as follows:

> The use of these standby surveillance capsules is incorporated by updating the surveillance program once sufficient data are retrieved to determine the withdrawal schedule for these capsules.

The staff has reviewed the requirements for standby surveillance capsules and finds that they meet the requirements of 10 CFR Part 50, Appendix H, and ASTM E 185-82, which is incorporated by reference in Appendix H, and are therefore acceptable.

STD COL 5.3(2)

The staff reviewed STD COL 5.3(2) related to the COL information item included under Section 5.3.4 of the COL FSAR. In Section 5.3.1.6.1 of the COL FSAR, this information item is given as follows:

> Accelerated irradiation capsules as defined in American Society for Testing and Materials (ASTM) E-185 and integrated surveillance program for multiple reactors at a single site, are not applicable at CPNPP Units 3 and 4.

The staff has reviewed the allowances for accelerated irradiation capsules and an integrated surveillance program and finds that the applicant's decision not to use these allowances at CPNPP Units 3 and 4 meets the requirements of 10 CFR Part 50, Appendix H, and ASTM E 185-82, which is incorporated by reference in Appendix H, and is therefore acceptable.

STD COL 5.3(2)

The staff reviewed CP COL 5.3(2) related to the COL information item included under Section 5.3.4 of the COL FSAR. In Section 5.3.1.6.3 of the COL FSAR, this information item is given as follows:

> A summary technical report, including test results, is submitted as specified in 10 CFR 50.4, for the contents of each capsule withdrawn, within one year of the date of capsule withdrawal unless an extension is granted by the Director, Office of Nuclear Reactor Regulation.
>
> The report includes the data required by ASTM E-185-82, as specified in paragraph III.B.1 of 10 CFR 50, Appendix H, and includes the results of the fracture toughness tests conducted on the beltline materials in the irradiated and unirradiated conditions.
>
> If the test results indicate a change in the Technical Specifications, either in the pressure-temperature limits or in the operating procedures, the expected date for submittal of the revised Technical Specifications is provided with the report.

The staff has reviewed the requirements for the summary technical report and finds that they meet the requirements of 10 CFR Part 50, Appendix H, and ASTM E 185-82, which is incorporated by reference in Appendix H, and are therefore acceptable.

NRC Generic Letter (GL) 92-01

GL 92-01, "Reactor Vessel Structural Integrity," addressed NRC concerns regarding compliance with the requirements of Appendices G and H to 10 CFR Part 50, which address fracture toughness requirements and RVSP requirements, respectively. Specifically, the NRC had concerns about (1) Charpy upper-shelf energy (USE) predictions for end of life for the limiting beltline weld and the plate or forging, (2) reactor vessels constructed to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition, and (3) use of RG 1.99, Revision 2, to estimate the embrittlement of the materials in the reactor vessel beltline. In addition, the NRC was concerned about RVSP compliance with ASTM E 185, which requires that the licensee take sample specimens from actual material used in fabricating the beltline of the reactor vessel.

The Comanche Peak FSAR fully incorporates these topics from the US-APWR DCD. Section 5.3.1 of the US-APWR DCD discusses the maximum limits for copper, nickel, and phosphorous content to reduce the sensitivity of the ferritic materials of the reactor vessel beltline to irradiation embrittlement in service. There is a statement in the US-APWR DCD that the RVSP conforms to ASTM E-185.

However, because there is only one opportunity (during vessel fabrication) to take the appropriate sample specimens from the actual material used in fabricating the beltline of the reactor vessel, the NRC staff emphasized this requirement in **RAI 3127, Question 05.03.01-2**, discussed above, in which the staff asked the COL applicant to confirm that the materials selected for the capsules are taken from material used for the reactor vessel beltline. In its response to **RAI 3127**, **Question 05.03.01-2**, dated October 30, 2009, the applicant confirmed that test specimens are taken from material used for the reactor vessel beltline and that the capsules are sealed in an inert environment. The COL applicant provided a markup copy of its proposed revision to the FSAR that the staff found acceptable. Therefore, the staff will review Section 5.3.1.6 in the next revision of the FSAR to confirm that the COL FSAR has been revised as shown in the markup copy provided in response to **RAI 3127, Question 05.03.01-2**, which is being tracked as **Confirmatory Item 05.03.01-2**

The US-APWR DCD also states that end-of-life RT_{NDT} (nil-ductility reference temperature) [where subscript "NDT" stands for nil-ductility transition] and USE projections were estimated using RG 1.99. The construction of the reactor vessel to an ASME Code earlier than the Summer 1972 Addenda of the 1971 Edition is not a concern for new reactors, including CPNPP3&4. In FSAR Section 5.3.1.6.3, the applicant provides additional information in STD COL 5.3(2), stating that when each capsule is withdrawn, a summary technical report of the data required by ASTM E 185-82 and the results of the fracture toughness tests conducted on the beltline materials in the irradiated and un-irradiated conditions will be submitted to the NRC within one year of the date of capsule withdrawal.

On the basis of these commitments stated in the FSAR, the staff finds that the COL applicant has met the intent of GL 92-01. The staff further finds that the COL applicant, which will become the COL licensee, will continue to meet the intent of the GL in the future by providing the summary test reports to the NRC.

The NRC staff's review confirmed that the information contained in the application and incorporated by reference addresses the required information relating to the RV materials. Section 5.3.1 of the US-APWR DCD is being reviewed by the staff under Docket Number 52-021. The NRC staff's technical evaluation of the information incorporated by reference related to RV materials will be documented in the staff SER on the DC application for the US-APWR design.

Interface Requirements

US-APWR DCD Tier 2, Section 1.8, “Interfaces for Standard Design,” identifies significant interfaces between the US-APWR standard plant design and the COL applicant’s proposed plant. However, this section does not specify any interfaces related to Section 5.3.1 of the DCD.

5.3.1.5 Post-Combined License Activities

Part 10, License Conditions, Reactor Vessel Material Surveillance Program:
The COL holder [licensee] shall implement this operational program prior to initial criticality.

- License Condition (5-3) - The licensee shall implement a reactor vessel material surveillance program prior to initial criticality.

For the reasons discussed in the technical evaluation above, the following FSAR commitment is identified as the responsibility of the licensee:

- Submit a schedule to the NRC that supports the planning and conduct of NRC inspections of operational programs, including the reactor vessel surveillance program, no later than 12 months after issuance of the COL or at the start of construction as defined in 10 CFR 50.10a, whichever is later.

In accordance with FSAR STD COL 5.3(2), the COL licensee shall update the surveillance program once sufficient data are retrieved to determine the withdrawal schedule for standby surveillance capsules pursuant to 10 CFR Part 50, Appendix H.

Finally, in accordance with FSAR STD COL 5.3(2), the COL licensee shall submit a summary technical report for the contents of each capsule withdrawn within one year of the date of capsule withdrawal pursuant to 10 CFR Part 50, Appendix H.

5.3.1.6 Conclusions

On the basis of its technical evaluation, the NRC staff concludes that the information pertaining to CPNPP3&4 COL FSAR Section 5.3 is within the scope of the design certification and adequately incorporates by reference Section 5.3.1 of the US-APWR DCD, Revision 2, and is, thus, acceptable.

The staff is reviewing the information in DCD Section 5.3.1 on Docket Number 52-021. The results of the NRC staff’s technical evaluation of the information related to RV materials incorporated by reference in the CPNPP3&4 COL FSAR will be documented in the staff SER on the DC application for the US-APWR design. The SER on the US-APWR is not yet complete, and this is being tracked as part of Open Item [1-1]. The staff will update Section 5.3.1 of this SER to reflect the final disposition of the DC application.

In addition, with the exception of **Confirmatory Items 05.03.01-2 and 05.03.01-3**, the staff concludes that applicant’s proposed resolutions to the COL information items in Section 5.3.1.6 of the COL FSAR meet the relevant guidelines of SRP Section 5.3.1**,** and RG 1.206 Section C.III.1, Chapter 5, C.I.5.3.1.6, and are thus, acceptable. Conformance with these guidelines provides an acceptable basis for satisfying, in part, the requirements of Appendix H to 10 CFR

Part 50. The NRC staff concludes the requirements of 10 CFR 52.79(d) have been satisfied for this Section and it is, thus, acceptable.

5.3.2 Pressure-Temperature Limits, Pressurized Thermal Shock, and Charpy Upper-Shelf Energy Data and Analyses

5.3.2.1 Introduction

The FSAR describes the bases for setting operational limits on pressure and temperature for the RCPB during any condition of normal operation, including anticipated operational occurrences (AOO), and hydrostatic tests. In addition, this discussion should provide detailed assurance that Appendices G and H to 10 CFR Part 50 will be complied with throughout the life of the plant.

Radiation embrittlement causes a reduction in the ductility of the reactor pressure vessel (RPV) beltline materials. This reduction is measured in terms of the adjusted nil-ductility reference temperature (ART). The presence of elements such as copper, nickel and phosphorus is controlled to limit reductions in ductility and fracture toughness in the steel that forms the RPV. Pressure-temperature (P-T) limits, derived using linear-elastic fracture mechanics principles, provide margins of safety to prevent non-ductile fracture during normal operation, heat-up, cooldown, AOOs, system hydrostatic, preservice and inservice leakage tests.

P-T limits are required as a means of protecting the reactor vessel during startup and shut down to minimize the possibility of fast fracture. The methods outlined in Appendix G of Section XI of the ASME Code are employed in the analysis of protection against non-ductile failure. Beltline material properties degrade with radiation exposure, and this degradation is measured in terms of the ART, which includes a nil-ductility reference temperature shift, initial RT_{NDT}, and margin.

5.3.2.2 Summary of Application

Section 5.3.2 of the CPNPP3&4 COL FSAR incorporates by reference Section 5.3.2 of the US-APWR DCD.

In addition, in CPNPP3&4 COL FSAR Section 5.3.2, the applicant provided the following:

US-APWR COL Information Items

- STD COL 5.3(1)

The applicant provided additional information in STD COL 5.3(1) to address COL Information Item 5.3(1). COL Information Item 5.3(1) states that the COL applicant addresses the use of plant-specific P-T limit curves. Generic P-T limits curves for the US-APWR reactor vessel are shown in Figures 5.3-2 and 5.3-3, which are based on the conditions described in Subsection 5.3.2. However, for a specific US-APWR plant, these limit curves are plotted based on actual material composition requirements, and the COL applicant is required to address the use of these plant-specific curves. COLA Part 4, Technical Specification 5.6.4, specifies that plant-specific curves will be developed and included in the pressure and temperature limits reports (PTLRs) for CPNPP3&4.

The applicant also provided additional information in STD COL 5.3(1) to specify that operating procedures will be developed for CPNPP3&4 in accordance with Section 13.5, such that the plant-specific P-T limit curves are not exceeded and Technical Specification requirements are satisfied.

5.3.2.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed within the FSER related to the DCD. In addition, the regulatory basis for acceptance of the resolution to the COL information item (STD COL 5.3(1)) is Appendix G to 10 CFR Part 50 as it relates to fracture toughness requirements.

5.3.2.4 Technical Evaluation

The NRC staff reviewed Section 5.3.2 of the CPNPP3&4 COL FSAR and checked the referenced DCD to ensure that the combination of the DCD and the information in the COL represent the complete scope of information relating to this review topic.[5]

Specifically, the staff reviewed Section 5.3.2 of the DCD to ensure the information contained therein is appropriate for incorporation by reference and that any supplemental information by the COL applicant has been addressed in the COL application. The staff also reviewed the conformance of CPNPP 3 & 4 COL FSAR Section 5.3.2 to the guidance in RG 1.206 Section C.III.5.3.2.1, "Limit Curves," and Section C.III.5.3.2.2, "Operating Procedures." With the exception of STD COL 5.3(1) addressed below, the staff's review confirmed there is no supplemental information to be provided in the COL application related to this section.

The NRC staff's review confirmed that the information contained in the application and incorporated by reference addresses the required information relating to pressure temperature limits. Section 5.3.2 of the US-APWR DCD is being reviewed by the staff under Docket Number 52-021. The NRC staff's technical evaluation of the information incorporated by reference related to pressure temperature limits will be documented in the staff SER on the DC application for the US-APWR design.

In addition, the staff reviewed the supplemental information contained in the CPNPP3&4 COL FSAR to address COL information items as follows:

<u>US-APWR COL Information Items</u>

- STD COL 5.3(1)

COL Information Item No. 5.3(1) states that that the COL applicant addresses the use of plant-specific P-T limit curves. Generic P-T limits curves for the US-APWR reactor vessel are shown in DCD Figures 5.3-2 and 5.3-3, which are based on the conditions described in DCD Section 5.3.2. However, for a specific US-APWR plant, these limit curves are plotted based on actual material composition requirements and the COL applicant should address the use of these plant-specific curves.

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[5] See Section 1.2.2 for a discussion on the staff's review related to verification of the scope of information to be included within a COL application that references a design certification.

In response to COL Information Item 5.3(1), the applicant stated that plant specific curves will be developed and included in the PTLR for CPNPP 3 & 4, as required by COLA Part 4, Technical Specification 5.6.4.

In a letter dated June 8, 2009, MHI, the US-APWR DC applicant, submitted a generic PTLR for US-APWR plants using bounding material properties and projected fluence. On this basis, in **RAI 2353, Question 05.03.02-2,** the staff requested that the COL FSAR be revised to add a commitment that addresses the submittal of plant-specific P-T limits. In its response to **RAI 2353, Question 05.03.02-2,** dated August 7, 2009, the applicant stated that FSAR Section 5.3.2 will be revised to state that the COL Holder will update the P-T limits prior to fuel loading using the PTLR methodologies approved in the US-APWR DCD and the plant-specific material properties and inform the NRC of the updated P-T limits as required by the CCNPP3&4 Technical Specifications. The applicant also provided a markup copy of its proposed revision to the FSAR that the staff found acceptable. The staff finds the applicant's approach acceptable because it is consistent with the approach for all operating reactors as described in GL 96-03, "Relocation of Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," where licensees using PTLRs inform the NRC staff of any subsequent change in P-T limits with no NRC approval necessary when the approved PTLR methodology is followed. In addition, the staff has verified the changes to FSAR Section 5.3.2.1 were included in COLA, Revision 1. Therefore, the staff finds that the applicant has appropriately addressed **RAI 2353, Question 05.03.02-2** is resolved and closed.

The COL applicant also provided additional information in CPNPP 3 & 4 COL FSAR Section 5.3.2.2, "Operating Procedures," to state that operating procedures will be developed for CPNPP Units 3 and 4 in accordance with Section 13.5, such that the plant-specific pressure-temperature limit curves are not exceeded and Technical Specification requirements are satisfied. The staff finds that this commitment conforms to the guidance of RG 1.206 Section C.III.5.3.2.2, "Operating Conditions" and is thus acceptable.

On this basis, the staff finds that the applicant has appropriately addressed the COL Information Item.

Interface Requirements

US-APWR DCD Tier 2, Section 1.8, Table 1.8-1, "Significant Site-Specific Interfaces with the Standard US-APWR Design," identifies significant interfaces between the US-APWR standard design and the COL application. This table does not specify any interfaces related to Section 5.3.2 of the DCD.

5.3.2.5 Post-Combined License Activities

The following items were identified as the responsibility of the COL licensee:

- STD COL 5.3(1), involving the development and inclusion of plant-specific curves in the PTLRs for CPNPP3&4, as required by Technical Specification 5.6.4.

If the generic PTLR submitted by MHI for all US-APWR plants is approved, the COL licensee will update the plant-specific P-T limits using the approved PTLR methodology and inform the NRC of the updated P-T limits. No further review is needed if the PTLR methodology remains unchanged.

5.3.2.6 Conclusions

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to pressure temperature limits and, there is no outstanding information expected to be addressed in the CPNPP3&4 COL FSAR related to this section.

The NRC staff concludes that the information pertaining to CPNPP 3 & 4 COL FSAR Section 5.3 is within the scope of the design certification and adequately incorporates by reference Section 5.3.2 of the US-APWR DCD, Revision 1, and is, thus, acceptable.

The staff is reviewing the information in DCD Section 5.3.2 on Docket Number 52-021. The results of the NRC staff's technical evaluation of the information related to the pressure temperature limits incorporated by reference in the CPNPP3&4 COL FSAR will be documented in the staff SER on the DC application for the US-APWR design. The SER on the US-APWR is not yet complete, and this is being tracked as part of Open Item [1-1]. The staff will update Section 5.3.2 of this SER to reflect the final disposition of the DC application.

In addition, the staff concludes that applicant's proposed resolution to the COL information items in Section 5.3.4 of the COL FSAR meets the relevant guidelines of SRP Section 5.3.2, and RG 1.206 Section C.III.5.3.2 and is thus, acceptable. Conformance with these guidelines provides an acceptable basis for satisfying, in part, the requirements of Appendix G to 10 CFR Part 50.

5.3.3 Reactor Vessel Integrity

5.3.3.1 Introduction

This section describes the specifications related to the integrity of the RV. The RV is the reactor coolant pressure boundary used to support and enclose the reactor core. It provides flow direction with the reactor internals through the core and maintains a volume of coolant around the core. The RV is fabricated by welding together the lower head, the transition ring, the lower shell, and the upper shell. The upper shell contains the penetrations from the inlet and outlet nozzles and direct vessel injection nozzles. Although most of the areas described in this section are reviewed separately under other SRP sections, the integrity of the RV is of such importance that a special summary review of factors relating to the integrity of the RV is warranted. The information in each area is reviewed for completeness and consistency with requirements and information to ensure RV integrity. This section involves the integrated review of Sections 5.2.3, 5.2.4, 5.3.1 and 5.3.2 of NUREG-0800, as they relate to the integrity of the RV.
As part of RV integrity, this section of the SER also addresses the issue of pressurized thermal shock (PTS). PTS events are transients in a pressurized-water RV that can cause severe overcooling of the vessel wall, followed by immediate re-pressurization. The thermal stresses, caused when the inside surface of the RV cools rapidly, combined with high-pressure stresses, will increase the potential for fracture if a flaw is present in a low-toughness material. The materials most susceptible to PTS are those in the RV beltline where neutron radiation causes

embrittlement over time. To protect against PTS events, the applicant must provide plant-specific PTS reference temperature (RT_{PTS}) values as required by 10 CFR 50.61.

5.3.3.2 Summary of Application

In the Comanche Peak Nuclear Power Plant, Units 3 & 4 (CPNPP 3 & 4), combined license application (COLA) Final Safety Analysis Report (FSAR) Section 5.3, the applicant incorporated by reference, without any departures, Section 5.3.2.3, "Pressurized Thermal Shock," Section 5.3.2.4, "Upper Shelf Energy," and Section 5.3.3, "Reactor Vessel Integrity," of the US-APWR DCD, Revision 1.

In addition, the applicant provided the following:

US-APWR COL Information Item

- STD COL 5.3(4)

The applicant provided additional information in CP COL 5.3(4) to address COL Information Item 5.3(4) of the US-APWR DCD, Revision 1. COL Information Item 5.3(4) states that the COL applicant verifies the upper shelf energy (USE) and RT_{NDT} at end-of-life (EOL), including a PTS evaluation based on actual material property requirements of the RV material and the projected neutron fluence for the design-life objective of 60 years.

- STD COL 5.3(5): Preservice and Inservice Inspection

The applicant provided additional information in CP COL 5.3(5) to address COL Information Item 5.3(5). The applicant replaced the fourth and fifth sentences in the first paragraph of DCD Section 5.3.3.7 with the following:

> The detailed list of inservice and preservice inspections for the CPNPP Units 3 and 4 reactor vessel is shown in DCD Tables 5.3-2 and 5.3-3.

5.3.3.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed within the FSER related to the DCD.

In addition, COL applicants of pressurized water reactors are also required to have the reference temperature, RT_{PTS}, evaluated for the end-of-life fluence for each of the RV beltline materials in accordance with requirements of 10 CFR 50.61.

5.3.3.4 Technical Evaluation

The staff reviewed sections 5.3.2 and 5.3.3 of the CPNPP 3 & 4 COL application including the corresponding sections in the referenced DCD. Specifically, the staff reviewed Section 5.3.2, and Section 5.3.3 of the DCD to ensure the information contained therein is appropriate for incorporation by reference and that any supplemental information provided by the COL applicant has been addressed in the COL application. The staff also reviewed the conformance of CPNPP 3 & 4 COL FSAR Section 5.3.2 to the guidance in RG 1.206 Section C.III.5.3.2.3, “Pressurized Thermal Shock,” Section C.III.5.3.2.4, “Upper Shelf Energy,” and Section C.III.5.3.3, “Reactor Vessel Integrity.” With the exception of STD COL 5.3(4) and STD COL 5.3(5) addressed below, the staff’s review confirmed that there is no additional information to be provided in the COL application related to this section.

The staff’s technical evaluation of the information incorporated by reference is discussed in Chapter 5 of the FSER related to the US-APWR DCD. Section 5.3 of the US-APWR DCD is being reviewed by the staff under docket number 52-021. The staff’s review of the information incorporated by reference will be documented in the corresponding SER.

The staff reviewed the information contained in the COL FSAR:

- STD COL 5.3(4)

COL Information Item No. 5.3(4) states that the COL applicant verifies the USE and RT_{NDT} at EOL, including a PTS evaluation using actual material properties of the RV material and the projected neutron fluence for the design-life objective of 60 years. In addition, US-APWR DCD Section 5.3.2.3 states that the RT_{PTS} values will be calculated based on plant-specific material property requirements, which will be verified by the COL applicant. In response to COL Information Item 5.3(4), the applicant stated, in STD COL 5.3(4), that the referenced PTS values and upper shelf energy (USE) at end-of-life (EOL) for CPNPP 3 & 4 are calculated based on the material property requirements detailed in DCD Subsection 5.3.1.5, and the results are shown in DCD Table 5.3-4.

The staff found that the COL applicant had not addressed the evaluation of RT_{PTS} for the CPNPP 3 & 4 RV using plant-specific material properties. To satisfy the requirements of 10 CFR 50.61, the licensee shall have projected values of RT_{PTS} accepted by the NRC for each RV beltline material for the EOL fluence of the material. It is noted that the as-procured RV material properties will be available to the COL holder after acceptance of the RV. Therefore, in order to provide sufficient time for NRC review of the plant-specific PTS evaluation, the staff requested, in **RAI 2317**, **Question 05.03.02-1**, that a license condition be added, related to Section 5.3.2.3 that states, within a reasonable period of time following acceptance of the RV (e.g., 1 year after acceptance of the RV), the COL holder will submit to the NRC staff its plant-specific PTS evaluation. In its May 1, 2009, response, the applicant agreed to provide the plant-specific PTS evaluation within 12 months after acceptance of the RV and stated that this commitment will be included as part of the proposed license condition identified in FSAR Table 13.4-201, “Operational Programs Required by NRC Regulation and Program Implementation,” Item 5, “Reactor Vessel Materials Surveillance Program (RVSP).” The staff found that the applicant’s proposed milestone for submitting the PTS evaluation to the NRC was acceptable. However, the submittal of RT_{PTS} values is not related to the RVSP or any other operational program described in SECY 05-0197. Therefore, the applicant’s proposal to include this commitment as part of a license condition identified in Table 13.4-201 is not fully acceptable. In supplemental

RAI 2353, Question 05.03.02-3 the staff proposed that a license condition be added to COL Part 10, Section 2, “Proposed License Conditions” which states that the plant-specific PTS evaluation will be submitted to the NRC within 12 months after acceptance of the RV and requested that FSAR Section 5.3.2.3 be revised accordingly. The applicant responded on August 7, 2009, that CPNPP3&4 COL Part 10 (ITAAC and Proposed License Conditions) will be revised to provide a proposed license condition stating that the plant specific PTS evaluation of the as-procured reactor vessel will be submitted to the NRC within 12 months following the acceptance of the reactor vessel. The staff finds that the applicant’s response to the RAI is acceptable because it meets the implementation requirements of 10 CFR 50.61.

The staff also confirmed that the proposed license condition has been added to Part 10 of the COLA. On this basis, the staff finds that the actions proposed by the applicant to address the COL information item provide reasonable assurance that the requirements of 10 CFR 50.61 will be met and are therefore acceptable. Therefore, **RAI 2353, Question 05.03.02-3**, is resolved and closed.

- STD COL 5.3(5)

COL Information Item No. 5.3(5) states that the COL applicant provides the information for preservice and inservice inspection described in subsection 5.2.4. In response to COL Information Item 5.3(5), the applicant stated that the detailed list of inservice and preservice inspections for the CPNPP 3 & 4 RV is shown in DCD Tables 5.3-2 and 5.3-3. In addition, the applicant has provided the implementation milestones for the preservice and inservice inspection programs in COL FSAR Table 13.4-201. Therefore, the staff finds that the applicant has appropriately addressed the COL Information Item. The staff’s detailed evaluation of the applicant’s proposed preservice and inservice inspection programs is documented in SER Section 5.2.4.

Interface Requirements

US-APWR DCD Tier 2, Section 1.8, Table 1.8-1, “Significant Site-Specific Interfaces with the Standard US-APWR Design,” identifies significant interfaces between the US-APWR standard design and the COL application. This table does not specify any interfaces related to Section 5.3.3 of the DCD.

5.3.3.5 Post-Combined License Activities

For the reasons discussed in the technical evaluation section above, the staff proposes to include the following license condition:

- License Condition (5-4) – In order to enable timely NRC review of the pressurized thermal shock (PTS) evaluation using the as-procured reactor vessel material properties, it will be provided within 12 months after acceptance of the reactor vessel.

5.3.3.6 Conclusions

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to RV integrity, and with the exception of the confirmatory item identified above, there is no other outstanding information expected to be addressed in the CPNPP3&4 COL FSAR related to this section.

The NRC staff concludes that the information pertaining to CPNPP 3 & 4 COL FSAR Section 5.3 is within the scope of the design certification and adequately incorporates by reference Sections 5.3.2 and 5.3.3 of the US-APWR DCD, and is, thus, acceptable.

The staff is reviewing the information in DCD Section 5.3.3 on Docket Number 52-021. The results of the NRC staff's technical evaluation of the information related to RV integrity incorporated by reference in the CPNPP3&4 COL FSAR will be documented in the staff SER on the DC application for the US-APWR design. The SER on the US-APWR is not yet complete, and this is being tracked as part of Open Item [1-1]. The staff will update Section 5.3.3 of this SER to reflect the final disposition of the DC application.

The staff further concludes that the applicant has appropriately addressed COL Information Item 5.3(4) thereby providing reasonable assurance that the requirements of 10 CFR 50.61 will be met. The staff also concludes that the actions proposed by the applicant to address COL Information Item 5.3(5) are acceptable. The staff's detailed evaluation of the applicant's proposed preservice and inservice inspection programs is documented in SER Section 5.2.4.

5.4 Reactor Coolant System Component and Subsystem Design

5.4.1 Introduction

Section 5.4 of the US-APWR DCD provides information regarding the performance requirements and design features of the reactor coolant pump, steam generator, reactor coolant piping, main steam flow restrictor, residual heat removal system, pressurizer and discharge system, pressurizer relief tank, and RCS high point vent system.

The staff is reviewing the information in DCD Section 5.4 on Docket Number 52-021. The results of the NRC staff's technical evaluation of the information related to the RCS component and subsystem design incorporated by reference in the FSAR will be documented in the staff SER on the DC application for the US-APWR. The SER on the US-APWR is not yet complete, and this is being tracked as part of Open Item [1-1]. The staff will update Section 5.4 of this SER to reflect the final disposition of the DC application design.

5.4.2 Summary of Application

Section 5.4 of the FSAR states that it incorporates by reference, with no departures or supplements, Section 5.4, "Reactor Coolant System Component and Subsystem Design," of the US-APWR DCD. However, as discussed below, the FSAR omitted certain information regarding the steam generator (SG) program that is supposed to be included in accordance with RG 1.206.

5.4.3 Regulatory Basis

The regulatory basis of the information incorporated by reference is addressed within the FSER related to the DCD. In addition, the relevant regulatory requirements for RCS components and the associated acceptance criteria are given in Section 5.4 of NUREG-0800, the SRP. The FSAR is supposed to address supplemental information in addition to the DCD information incorporated by reference, in accordance with RG 1.206, dealing with the SG program.

The applicable regulatory requirements for the SG program are given in SRP Section 5.4.2.2 and are summarized, in part, below. Review interfaces with other SRP sections can be found in Section 5.4.2.2 of NUREG-0800.

1. GDC 32 of Appendix A to 10 CFR Part 50. GDC 32 requires, in part, that the designs of all components that are part of the reactor coolant pressure boundary (RCPB) permit periodic inspection and testing of critical areas and features to assess their structural and leaktight integrity.

2. 10 CFR 50.55a(b)(2)(iii) specifically addresses the inspection of SG tubes and states that if the plant TS include inspection requirements that differ from those in Article IWB-2000 of Section XI of the ASME Code, the TS govern.

3. 10 CFR 50.55a(g) requires that inservice inspection (ISI) programs meet the applicable inspection requirements in Section XI of the ASME Code. The SG program is a portion of the ISI program. In addition, 10 CFR 50.55a(b)(2)(iii) specifically addresses SG tubes and states that if the plant TS include inspection requirements that differ from those in Article IWB-2000 of Section XI of the ASME Code, the TS govern.

4. 10 CFR 50.36 applies to the SG program in the TS.

5. Appendix B to 10 CFR Part 50 applies to the implementation of the SG program. Of particular note are Criteria IX, XI, and XVI.

6. 10 CFR 50.65, as it relates to the SG program.

7. 10 CFR 52.79(d), requirements for COLAs that reference a certified design

8. 10 CFR 52.80(a) The proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will be operated in conformity with the combined license, the provisions of the Atomic Energy Act, and the Commission's regulations.

Acceptance criteria adequate to meet the above requirements include:

1. NEI 97-06, "Steam Generator Program Guidelines."

2. NUREG-1430, NUREG-1431, NUREG-1432, PWR Standard Technical Specifications.

3. RG 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes"

4. RG 1.206, as it relates to including SG program information in FSAR Section 5.4

4. SECY-05-0197 as it relates to fully describing an operational program

5. EPRI PWR Steam Generator Examination Guidelines

5.4.4 Technical Evaluation

The NRC staff reviewed the application and checked the referenced DCD to determine whether any issues relating to this section remained for review. The staff's review revealed that there is one outstanding issue related to this section involving the SG program. Although the applicant had stated that FSAR Section 5.4 incorporated DCD Section 5.4 with no departures or supplements, the staff determined that the applicant did not provide the information regarding the SG program in accordance with RG 1.206. Subsequently, by letter dated March 10, 2010, the staff issued **RAI 4454, Question 13.04-04** in which the staff requested that the applicant clarify that the pre-service and in-service portions of the SG Program are to be included as operational programs that should be fully described within Table 13.4-201, "Operational Programs Required by NRC Regulation and Program Implementation," of FSAR Chapter 13.

In its response to **RAI 4454, Question 13.04-04**, dated April 12, 2010, the applicant informed the staff that the SG program is fully described in COLA Part 4, TS 5.5.9 and the USAPWR DCD, Section 5.4.2.2. In addition, the applicant committed to revise FSAR Table 13.4-201, which lists operational programs, to address the pre-service and in-service inspection program elements applicable to the SGs. The applicant stated that the pre-service inspection is to be performed as required by 10 CFR 50.55a(g) and ASME Code Section XI, and is therefore addressed in FSAR Table 13.4-201. The applicant stated that, as described in the USAPWR DCD, Subsection 5.4.2.2.2, preservice inspection of the entire length of each steam generator tube will be performed in accordance with ASME Section XI and EPRI PWR steam generator examination guidelines. Regarding the in-service inspection of the SG, the applicant stated this examination is performed as required by 10 CFR 50.55a(g) and ASME Code, Section XI, as part of the overall inservice inspection program identified in FSAR Table 13.4-201. The applicant provided a draft of its proposed revision this table to include the SG program. The NRC staff finds this response and the proposed revision acceptable as the applicant has proposed implementation milestones and will include the SG program in Table 13.4-201, consistent with the guidance of SRP Section 5.4.2.2 and RG 1.206. The staff will confirm that the revision to FSAR Table 13.4-201 has been properly incorporated in the next revision of the FSAR. This RAI is now being tracked as **Confirmatory Item 13.04-04.**

5.4.5 Post Combined License Activities

Development and implementation of the SG program by the COL applicant and/or licensee as described herein and in detail in Section 5.4.2.2 of the US-APWR DCD, Tier 2.

5.4.6 Conclusions

The NRC staff reviewed the application and checked the referenced DCD. The NRC staff's review confirmed that the applicant addressed the required information relating to RCS component and subsystem design with the exception of the confirmatory item identified above.

The staff also confirmed that with the exception of the confirmatory item, there is no other outstanding information expected to be addressed in the FSAR related to this section. The staff concludes that the information pertaining to FSAR Section 5.4 is within the scope of the design certification and adequately incorporates by reference Sections 5.4 of the US-APWR DCD, with the exception of the confirmatory cited above, and is, thus, otherwise acceptable.

The staff is reviewing the information in DCD Section 5.3.3 on Docket Number 52-021. The results of the NRC staff's technical evaluation of the information incorporated by reference in the CPNPP3&4 COL FSAR will be documented in the staff SER on the DC application for the US-APWR design. The SER on the US-APWR is not yet complete, and this is being tracked as part of Open Item [1-1]. The staff will update Section 5.4 of this SER to reflect the final disposition of the DC application.