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3 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

This chapter describes the U.S. Nuclear Regulatory Commission (NRC) staff review of the design of U.S. EPR structures, systems, and components (SSCs) important to safety for compliance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, “General Design Criteria (GDC) for Nuclear Power Plants.” It also describes the staff review of the U.S. EPR design for compliance with additional criteria, such as those contained in industry codes and standards, to provide protection for structures, systems, and components important to safety from external and internal events, including, for example, high wind and tornados, seismic events, dynamic effects of pipe break, and internally generated missiles.

3.1 Compliance with U.S. Nuclear Regulatory Commission General Design Criteria

This section of the U.S. EPR Final Safety Analysis Report (FSAR) describes the U.S. EPR compliance with the GDC. Each of the 64 GDC is restated in the application followed by a summary of how the applicant states the U.S. EPR design complies with it. The staff’s evaluation of the U.S. EPR design compliance with each of the GDC is provided in the more detailed sections of this report. This section of the application also contains a combined license (COL) information item that states, “A COL applicant that references the U.S. EPR design certification will identify the site-specific QA Program Plan that demonstrates compliance with GDC 1, ‘Quality Standards and Records’.” This COL information item will be addressed in Section 3.2.1 of this report.

3.2 Classification of Structures, Systems, and Components

3.2.1 Seismic Classification

3.2.1.1 *Introduction*

NRC regulations require that SSCs important to safety be designed to withstand the effects of natural phenomena, such as earthquakes, without loss of capability to perform their safety functions. As defined in 10 CFR Part 50, Appendix A, important to safety SSCs are those that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. In addition, NRC regulations require that the safety SSCs required to withstand the effects of the safe-shutdown earthquake (SSE) ground motion are those necessary to assure:

- The integrity of the reactor coolant pressure boundary
- The capability to shut down the reactor and maintain it in a safe-shutdown condition
- The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 50.34(a)(1), “Contents of applications; technical information.”

The SSE is based upon an evaluation of the maximum earthquake potential and is the earthquake which produces the maximum vibratory ground motion for which safety-related SSCs are designed to remain functional. Those plant features that are designed to remain functional if an SSE occurs are designated Seismic Category I in accordance with Regulatory Guide (RG) 1.29, "Seismic Design Classification."

The objective of the staff review is to determine that SSCs important to safety have been appropriately classified and designed to withstand the effects of earthquakes without loss of capability to perform their intended functions.

3.2.1.2 *Summary of Application*

FSAR Tier 1/ITACC: Seismic Classification is addressed in FSAR Tier 1, Chapter 2 of Tier 1.

FSAR Tier 2: To meet the NRC seismic requirements with regard to the design for earthquakes, the FSAR indicates that SSCs are seismically classified in accordance with RG 1.29. The FSAR states that SSCs of radioactive waste management systems meet the seismic design recommendations specified in RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," and the regulatory requirements in GDC 61, "Fuel Storage and Handling and Radioactivity Control," as it relates to the design of radioactive waste systems and other systems that may contain radioactivity. The FSAR also states that safety-related instrument sensing lines meet the seismic design recommendations contained in RG 1.151, "Instrument Sensing Lines." Further, FSAR Tier 2, Section 3.1.1.2.1, "U.S. EPR Compliance," states that, in regard to compliance with GDC 2, "Design Basis for Protection Against Natural Phenomena," the safety-related SSCs are designed to either withstand the effects of natural phenomenon without loss of the capability to perform their safety functions, or to fail in a safe condition. FSAR Tier 2, Table 1.9-2, "U.S. EPR Conformance with Regulatory Guides," states that the U.S. EPR conforms to RG 1.29, Revision 4.

Five categories of seismic classification are used in the U.S. EPR. These are Seismic Category I, Seismic Category II, Radwaste Seismic (RS), Conventional Seismic (CS), and Non-Seismic (NSC).

Seismic Category I SSCs are designed to perform their safety function during and after an SSE. SSCs that are classified as safety-related are subject to the quality assurance program requirements of 10 CFR Part 50, Appendix B.

Certain SSCs that perform no safety-related function could, if they failed under seismic loading, prevent or reduce the functional capability of a Seismic Category I SSC, or cause incapacitating injury to main control room occupants during or following an SSE. These non-safety-related SSCs are classified as Seismic Category II. Seismic Category II SSCs, as stated in the FSAR, are to be designed to withstand SSE seismic loads without structural failure. Seismic Category II SSCs are subject to the pertinent quality assurance program requirements of 10 CFR Part 50, Appendix B.

Certain SSCs required for the management of radioactive waste that are classified as RW-IIa per RG 1.143 are designed to withstand seismic loads up to one-half SSE and are seismically categorized as RS.

Some non-safety-related SSCs that do not fall within the criteria for classification as Seismic Category I or II, may still be subject to seismic design criteria that are contained in applicable commercial or industry standards. These SSCs are classified as CS.

SSCs that do not fall within the RG 1.29 criteria for classification as Seismic Category I or II; do not fall within the RG 1.143 criteria for RW-IIa, RW-IIb, or RW-IIc, and are not subject to any other seismic design criteria are stated in the FSAR to be classified as (NSC). Non-safety-related SSCs designated as supplemented Grade NS-AQ are included in the 10 CFR Part 50, Appendix B program, if a significant licensing requirement or commitment is invoked.

The U.S. EPR important to safety SSCs that are classified as Seismic Category I, Seismic Category II, RS or CS are identified in FSAR Tier 2, Table 3.2.2-1, "Classification Summary." FSAR Tier 2, Table 3.2.2-1 includes both pressure boundary components of fluid systems and non-pressure boundary items such as structures and reactor pressure vessel (RPV) internals. The descriptions of the various system safety functions and applicable simplified piping and instrumentation drawings (P&IDs) given in other sections of FSAR Tier 2 also include seismic classifications for fluid systems.

FSAR Tier 2, Table 3.2.2-1 also identifies the safety classification as "S" for safety-related SSCs, "NS" for non-safety-related components, or "NS-AQ" for Supplemented Grade. The designation as safety-related indicates the quality assurance (QA) requirements of 10 CFR Part 50, Appendix B are applied.

3.2.1.3 ***Regulatory Basis***

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (hereafter referred to as NUREG-0800 or the SRP) Section 3.2.1 and are summarized below. Review interfaces with other SRP sections are also found in NUREG-0800, Section 3.2.1.

The applicable regulatory requirements are as follows:

1. GDC 1 and the pertinent QA requirements of 10 CFR Part 50, Appendix B, as they relate to applying QA requirements to activities affecting the safety-related functions of SSCs designated as Seismic Category I, commensurate with their importance to safety.

2. GDC 2, as it relates to the requirements that SSCs important to safety shall be designed to withstand the effects of earthquakes without loss of capability to perform necessary safety functions.

3. GDC 61, as it relates to the design of radioactive waste systems, and other systems that may contain radioactivity, to assure adequate safety under normal and postulated accident conditions.

4. 10 CFR Part 100, Appendix A, "Seismic and Geological Setting Criteria for Nuclear Power Plants" and 10 CFR Part 50, Appendix S, "Earthquake Engineering Criteria for Nuclear Power Plants," as it relates to certain SSCs being designed to withstand the SSE and/or operating-basis earthquake (OBE) and remain functional.

.Acceptance criteria adequate to meet the above requirements include:

1. RG 1.29 as it relates to an acceptable method of identifying and classifying those plant features that should be designed to withstand the effects of the SSE. RG 1.29, Regulatory Position C.1 states that the pertinent QA requirements of 10 CFR Part 50, Appendix B should be applied to all activities affecting the safety-related functions of Seismic Category I SSCs. RG 1.29, Regulatory Position C.2 states that those portions of SSCs whose continued function are not required, but whose failure could reduce the functioning of any Seismic Category I SSC to an unacceptable level, or could result in an incapacitating injury to occupants of the control room, should be designed and constructed so that an SSE could not cause such failure. RG 1.29, Regulatory Position C.3 provides guidelines for designing interfaces between Seismic Category I and non-seismic SSCs. RG 1.29, Regulatory Position C.4 states that the pertinent QA requirements of 10 CFR Part 50, Appendix B should be applied to all activities affecting the safety-related functions of SSCs discussed in RG 1.29, Regulatory Positions C.2 and C.3.

2. RG 1.143 as it relates to the establishment of the seismic design requirements of radioactive waste management SSCs to meet the requirements of GDC 2 and GDC 61 as they relate to designing these SSCs to withstand earthquakes. RG 1.143 identifies radioactive waste SSCs requiring some level of seismic design consideration.

3. RG 1.151 as it relates to seismic design requirements and classification of safety-related instrumentation sensing lines.

4. RG 1.155, "Station Blackout," as it relates to the seismic classification of non-safety-related risk-significant components.

5. RG 1.189, "Fire Protection for Nuclear Power Plants," as it relates to establishment of the design requirements for fire protection to meet the requirements of GDC 2, to design these SSCs to withstand earthquakes. RG 1.189 identifies portions of fire protection SSCs requiring some level of seismic design consideration.

3.2.1.4 *Technical Evaluation*

The staff reviewed FSAR Tier 2, Section 3.2.1, in accordance with SRP Section 3.2.1 and the guidance contained in RG 1.29. The staff's review included evaluation of the criteria used to establish the seismic classification and the application of those criteria to the classification of principal components included in FSAR Tier 2, Table 3.2.2-1. The staff reviewed the following areas and determined that additional information is needed to complete the review.

Classification Criteria

The staff reviewed the criteria identified in FSAR Tier 2, Section 3.2.1 used to select the appropriate seismic classification in FSAR Tier 2, Table 3.2.2-1 for principal components. The staff determined that the classification criteria for Seismic Category I, Seismic Category II and non-seismic design categories are similar to those in RG 1.29, Revision 4 and SRP Section 3.2.1 for seismic classification. One difference in terminology is that RG 1.29 does not use the term Seismic Category II, but the applicant's basic methodology, that SSCs are to be seismically analyzed if their failure could adversely affect Seismic Category I SSCs, is consistent. Also, RG 1.29 does not use the term RS, but the substance of the RS classification conforms to RG 1.143 for seismic requirements for radioactive waste management SSCs. Another difference is that the applicant has classified certain components that may be important to safety but are not safety-related as conventional seismic. The term CS is not used in regulatory guidance.

Systems that provide post 72-hour cooling and post-accident monitoring are examples of important to safety systems. Fire protection is another example of a non-safety--related system that may be important to safety. FSAR Tier 2, Table 1.9-2 identifies conformance to RG 1.189, Revision 1 for fire protection systems, and there are no exceptions identified to RG 1.189 or SRP Section 9.5.1 seismic requirements in FSAR Tier 2, Section 9.5, "Other Auxiliary Systems." SRP Section 9.5.1, Appendix B, "Supplemental Fire Protection Review Criteria for Advanced Reactors," identifies the enhanced fire protection criteria for advanced reactors, and the acceptability of non-safety grade systems is predicated on the acceptable resolution of the "regulatory treatment of non-safety systems" (RTNSS) issue. FSAR Tier 2, Table 1.9-2 also recognizes conformance to RG 1.151 concerning instrument sensing lines.

Compliance with GDC 2

FSAR Tier 2, Section 3.2.1.1, "Seismic Category I," in combination with FSAR Tier 2, Section 3.1.1.2.1, identifies that safety-related SSCs are designed either to withstand the effects of natural phenomena, including the SSE, without loss of capability to perform their safety functions, or to fail in a safe condition. GDC 2 applies to all important to safety SSCs and not only to SSCs that are considered safety-related. In 10 CFR 50.2, "Definitions," safety-related SSCs are defined as those structures, systems, and components that are relied upon to remain functional during and following design-basis events to assure: (1) The integrity of the reactor coolant pressure boundary; (2) the capability to shut down the reactor and maintain it in a safe-shutdown condition; or (3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures. In 10 CFR Part 50, Appendix A, SSCs important to safety are defined as structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.) In Request for Additional Information (RAI) 71, Question 03.02.01-1, the staff requested that the applicant expand FSAR Tier 2, Sections 3.2.1.1 and 3.1.1.2.1 to clarify how GDC 2 is satisfied relative to SSCs that are not identified as safety-related, but are considered important to safety and have augmented seismic requirements, such as the non-safety-related fire protection system or any SSC that is classified as Seismic Category II.

In a November 14, 2008, response to RAI 71, Question 03.02.01-1, the applicant provided several examples from earlier NRC documents where the terms safety-related and important to safety are used interchangeably. The applicant also stated that the U.S. EPR conforms to RGs 1.29, 1.143, 1.151, and 1.189 given in the SRP Section 3.2.1 and, that therefore, the U.S. EPR SSCs are designed to the requirements of GDC 2.

Based on the applicant's RAI response, the applicant's process to apply the terms safety-related and important to safety to the classification of SSCs is considered unclear and unresolved such that additional information is needed to clarify how these terms are applied and to explain the process to develop supplemental seismic and quality requirements (special treatment) for non-safety-related risk-significant SSCs considered important to safety to satisfy GDC 2. The staff closed RAI 71, Question 03.02.01-1 and, to comply with GDC 2 for seismic classification, the applicant was requested in follow-up RAI 420, Question 03.02.01-12 to further clarify how these terms are applied to satisfy GDC 2, revise the FSAR Subsection 3.1.1.2.1 stated conformance to GDC 2 to replace the term “safety related” with the more comprehensive term “important to safety,” and factor risk significance into seismic and quality group classification, based on the definition of the term important to safety in 10 CFR 50.

In regard to risk significance, the staff is concerned that the applicant has not adequately identified risk-significant SSCs in the FSAR that may be important to safety or defined supplemental design and seismic requirements to ensure their availability after an earthquake and the reliability assumed in the PRA “Probabilistic Risk Assessment”. Portions of non-safety-related systems that are risk-significant may be important to safety and require special treatment and appropriate seismic classification so that they are designed to withstand earthquakes consistent with assumptions in the PRA. The complete list of risk-significant SSCs is developed in phases and SRP Section 17.4 “Reliability Assurance Program” indicates that, during the first phase, SSCs are identified for inclusion in the program. The June 3, 2008, response to RAI 5, Question 17.4-1 includes a component list as PRA input to the RAP “Reliability Assurance Plan” component identification process and the August 8, 2008, response to RAI 21, Question 17.04-2 identifies that the full scope RAP will include passive components and the COL applicant is to provide the final list. The July 24, 2009, response to RAI 226, Question 17.04-16 addressed below further identified that the FSAR will be revised to include a list of risk-significant SSCs. Since risk-significant SSCs are to be included in Section 17.4 of the DCD, the scope of risk-significant SSCs is to be evaluated in that subsection in combination with the Chapter 19 “Probabilistic Risk Assessment and Sever Accident Evaluation” evaluation. However, the supplemental seismic requirements needed to satisfy GDC 2 for risk-significant SSCs are unclear and the applicant should either identify these specific requirements or explain the process, such as D-RAP “Design Reliability Assurance Program” and the NS-AQ classification process, used to develop and apply these requirements. For example, the basis for concluding that all risk-significant SSCs important to safety are designed to withstand earthquakes should be identified. Until the applicant submits a response to resolve this issue, **RAI 420, Question 03.02.01-12 is being tracked as an open item.**

Scope

The staff reviewed the completeness of the SSCs in FSAR Tier 2, Table 3.2.2-1 and determined that the table includes major SSCs such as piping, structures, electrical equipment, instrumentation and control (I&C) equipment, and cranes. In RAI 71, Question 03.02.01-2, the staff requested that the applicant clarify that FSAR Tier 2, Table 3.2.2-1 includes all SSCs that are within the scope of the FSAR and not site-specific SSCs. The staff also recommended to the applicant that FSAR Tier 2, Table 3.2.2-1 be revised to clarify that the table includes structures by changing the second column heading to read, “Structures, Systems and Components,” rather than, “System or Component Description.”

In a November 14, 2008, response to RAI 71, Question 03.02.01-2, the applicant clarified that FSAR Tier 2, Table 3.2.2-1 includes all SSCs that are within the scope of the FSAR, such as the reactor coolant system (RCS) insulation. The reactor vessel is part of the RCS, and the RCS insulation includes the reactor vessel insulation. As stated in FSAR Tier 2, Section 9.2.5.2, “System Description,” the ultimate heat sink makeup water system is the responsibility of the COL applicant; the classification of site-specific SSCs such as the ultimate heat sink screens will be in the combined license application (COLA). The applicant also stated that there are no tunnels in the U.S. EPR design. The applicant further clarified that FSAR Tier 2, Section 1.8, “Interfaces with Standard Designs and Early Site Permits,” identifies those conceptual designs that are outside the scope of the U.S. EPR standard design. This includes buried conduit duct banks, pipe ducts, and piping and portions of the circulating water supply system outside the Turbine Building (including the circulating water makeup intake structure and screens). Additionally, as noted in FSAR Tier 2, Section 3.2.1, COL Information Item 3.2-1, a COL applicant that references the U.S. EPR design will identify the seismic classification of applicable site-specific SSCs that are not identified in FSAR Tier 2, Table 3.2.2-1. The staff finds COL Information Item 3.2-1, which identifies the COL applicant’s responsibility for classifying the seismic category of site-specific SSCs acceptable. The staff also finds COL Information Item 3.1-1 that identifies the COL applicant’s responsibility for identifying the site-specific QA Program Plan that demonstrates compliance with GDC 1, is acceptable.

The staff has confirmed that Revision 1 of the FSAR, dated May 29, 2009, was revised to change the second column to, “SSC Description,” as requested. Accordingly, the staff finds the applicant has adequately addressed this issue and, therefore, the staff considers RAI 71, Question 03.02.01-2 resolved.

SSCs Classified as Non-Safety-Related

The seismic classification of each SSC depends on the safety function and classification as safety-related or non-safety-related. FSAR Tier 2, Section 3.2, “Classification of Structures, Systems, and Components,” does not clearly define the safety function of SSCs that are important to safety, but are classified as non-safety-related. For example, certain components considered non-safety–related that are internal to the reactor vessel, or part of the control rod drive system, accident monitoring functions, severe accident instrumentation and control, and core melt stabilization system appear to be important to safety, but are not specifically identified as safety-related and Seismic Category I. Based on the classification it is not clear if these SSCs are required to be or are credited to be functional during or following a seismic event.

For the SSCs that are important to safety and are classified as non-safety-related in FSAR Tier 2, Table 3.2.2-1, in RAI 71, Question 03.02.01-3, the staff requested that the applicant clarify the technical basis for each non-safety-related classification and identify if the seismic classification as Seismic Category II or other seismic classification conforms to the applicant’s PRA assumptions.

In a November 14, 2008, response to RAI 71, Question 03.02.01-3, the applicant references the November 14, 2008, response to RAI 71, Question 03.02.01-1 regarding examples where the terms safety-related and important to safety are used interchangeably. The staff does not consider these terms as synonymous and this concern is further addressed in the follow-up question to RAI 71, Question 03.02.01-1. The response also refers to Chapter 19, “Probabilistic Risk Assessment and Severe Accident Evaluation,” (RAI 8, Question 19.01-6) that provides a list of SSCs modeled in the PRA-based seismic margin assessment.

The staff closed RAI 71, Question 03.02.01-3 and in follow-up RAI 420, Question 03.02.01-13 staff is concerned that Table 19.1-107 provided in the response does not list specific equipment with component numbers, and it is still not clear if the specific SSCs discussed in RAI 71, Question 03.02.01-03 are credited to be functional during or following a seismic event. As indicated in the July 31, 2008, Chapter 19 RAI 8, Question 19.01-6 response, the seismic margin assessment does not credit SSCs that are not seismically qualified, but the applicant should establish the basis for post earthquake functionality of any important to safety SSCs that are currently classified as nonseismic. If there are no important to safety SSCs that are classified as non-seismic, the applicant was requested to so clarify.

The April 13, 2011, response to RAI 420, Question 03.02.01-13 clarifies that AREVA does not consider the terms “important to safety” and “safety-related” to be synonymous and states that the SSCs that are required to be functional during or following a seismic event are classified as Seismic Category I in FSAR Tier 2, Table 3.2.2-1 in compliance with the guidance of RG 1.29. The response further explains that in accordance with RG 1.29, the reactor coolant pressure boundary, the reactor core and reactor vessel internals, such as the control rod drive mechanism (CRDM), latch mechanism, the pressure boundary portions of the CRDMs, the heavy reflector and its associated components, are Seismic Category I.

As shown in FSAR Tier 2, Table 3.2.2-1, since the non-pressure boundary components of the reactor pressure vessel are classified as NS-AQ and Seismic Category II, in accordance with RG 1.29 and FSAR Tier 2, Section 3.2.1.2, they are designed to withstand SSE seismic loads without incurring a failure that permits deleterious interaction with any Seismic Category I SSC, or that could result in injury to main control room occupants. Components used for accident monitoring functions are also classified as Seismic Category I (see FSAR Tier 2, Table 3.2.2-1, KKS Code JR) in accordance with RG 1.29. The April 13, 2011, response to RAI 420, Question 03.02.01-13 also states there are no regulatory requirements for the severe accident instrumentation and control and the core melt stabilization system to be classified as safety-related and Seismic Category I. As noted in NRC-approved AREVA NP Topical Report ANP-10268P-A, and in FSAR Tier 2, Section 19.2, these SSC are relied upon to mitigate the consequences of a severe accident which is a beyond design basis accident.

The April 13, 2011, response to RAI 420, Question 03.02.01-13 also refers to the response to RAI 234, Question 19-304 that revised the FSAR to include a list of specific SSC credited in the PRA-based seismic margin assessment. The list is shown in FSAR Tier 2, Table 19.1-106. The U.S. EPR PRA-based seismic margin assessment does not credit any non-seismic equipment to meet the commitment for a high confidence, low probability of failure plant-level capacity of 1.67 times the SSE. Therefore no SSCs currently classified as non-seismic are required to meet GDC 2 or any other licensing commitments related to seismic design.

Since the April 13, 2011, response to RAI 420, Question 03.02.01-13 clarified that no SSCs currently classified as non-seismic are required to meet GDC 2 or any other licensing commitments related to seismic design; the staff concludes that the applicant has established an appropriate basis for the nonseismic classification. The staff will have the opportunity to audit design basis documents associated with the seismic classifications for risk-significant components and staff concerns associated with application of the terms “important to safety” and “safety-related” to satisfy GDC 1 and 2 is further addressed in other RAIs. Therefore, based on the RAI response and changes to the FSAR made in response to other RAIs, with the exception of verification, the staff considers the applicant’s response to the concerns identified in RAI 420, Question 03.02.01-13 associated with the non-seismic classification of specific SSCs acceptable. **RAI 420, Question 03.02.01-13 is being tracked as a confirmatory item** until the audit is complete to validate that the applicant has a process to verify the design basis for seismic classification of non-safety-related risk-significant SSCs that are important to safety.

Reactor Internals

FSAR Tier 2, Table 1.9-2 indicates that the U.S. EPR conforms to RG 1.29. RG 1.29, Regulatory Position C.1.b states that reactor vessel internals are designated as Seismic Category I and must be designed to withstand the effects of the SSE and remain functional. However, in FSAR Tier 2, Table 3.2.2-1, some reactor internal components (e.g., dome spray nozzle, flow distribution device, control rod drive mechanism adaptor thermal sleeves, etc.) are designed as Seismic Category II.

In RAI 201, Question 03.02.01-08, the staff requested that the applicant clarify whether the Seismic Category II classification for reactor internals is an exception to RG 1.29 and, if so, to identify this as an exception in FSAR Tier 2, Table 1.9-2, with the technical justification for the exception. Also, if the reactor internal components that are designed as Seismic Category II are not required to be functional during or after an SSE, the applicant was requested to state the basis.

In a May 6, 2009, response to RAI 201, Question 03.02.01-08, the applicant stated that the classification of some reactor internal structures as Seismic Category II conforms to previously certificated designs. American Society of Mechanical Engineers, (ASME) Code, Section III, Division 1, Subsection NG-1122 defines internal structures as all structures within the reactor pressure vessel other than core support structures, fuel and blanket assemblies, control assemblies, and instrumentation, and requires that construction of all internal structures is such as not to affect adversely the integrity of the core support structure. Core support structures are those structures or parts of structures which are designed to provide direct support or restraint of the core (fuel and blanket assemblies), and are classified as Seismic Category I in the FSAR. Classifying the internal structures as Seismic Category II will ensure that they will not adversely affect the integrity of the core support structures, and satisfy the ASME requirements and RG 1.29, Regulatory Position C.2.

Further, the response stated that the Seismic Category II classification for RPV internal components designated as internal structures is consistent with regulatory guidance and NRC-approved precedent. Although the applicant did not revise the FSAR to identify this as an exception to RG 1.29, the staff finds concurs that non-safety-related reactor internals need not be classified as Seismic Category I and considers the concerns associated with RAI 201, Question 03.02.01-08 resolved.

Passive Recombiners

FSAR Tier 2, Revision 2, Section 3.2.1 states that to meet the requirements of both GDC 2 and 10 CFR Part 50, Appendix S with regard to the design for earthquakes, U.S. EPR SSC are seismically classified in accordance with RG 1.29. Also, FSAR Tier 2, Table 1.9-2 states that that the U.S. EPR complies with RG 1.29, Revision 4. RG 1.29, Regulatory Position C.1.c. identifies that the systems that are required for post accident containment atmosphere cleanup (e.g., hydrogen removal system) are designated as Seismic Category I and must be designed to withstand the effects of the SSE and remain functional. However, FSAR Tier 2, Table 3.2.2-1 identifies that the Passive Autocatalytic Recombiners in the Combustible Gas Control System are classified as Seismic Category II rather than Seismic Category I. Therefore, in RAI 510, Question 03.02.01-19, the staff requested that the applicant describe the basis for the seismic classification of the Passive Autocatalytic Recombiners as Seismic Category II and clarify why this classification is consistent with RG 1.29 or revise the FSAR to show this as an exception to

RG 1.29 and justify the acceptability of this exception. **RAI 510, Question 03.02.01-19 is being tracked as an open item**.

Risk Insights

Based on FSAR Tier 2, Table 1.9-4, "U.S. EPR Conformance with Advanced and Evolutionary Light-Water Reactor Design Issues (SECY-93-087, 'Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced LWR Designs')," and FSAR Tier 2, Section 19.1.7.5, "PRA Input to the Regulatory Treatment of Non-Safety-Related Systems Program," the staff understands that the RTNSS process is not applicable to the U.S. EPR design. However, risk insights can provide useful information in determining the safety significance and seismic classification of important to safety SSCs that are either considered safety-related or non-safety-related. The staff could not locate the list of risk-significant SSCs that are part of the reliability assurance program could not be located in FSAR Tier 2, Section 17.4, "Reliability Assurance Program," or in FSAR Tier 2, Chapter 19, "PRA and Severe Accident Evaluation."

In RAI 71, Question 03.02.01-4, the staff requested that the applicant advise if the PRA or other design documents identify the safety significance of each important to safety SSC when subjected to an SSE so that the staff can evaluate the applicant's seismic classification based on the specific safety function. If this design information and list of risk-significant SSCs is in a Topical Report (TR) or other auditable form, the staff requested that the applicant reference the appropriate document(s).

In a November 14, 2008, response to RAI 71, Question 03.02.01-4, the applicant referred to FSAR Tier 2, Chapter 17, "Quality Assurance, and Reliability Assurance," RAI responses that provided several lists of important SSCs based on the Fussell-Vesely value, risk achievement value, or common cause. Based on the RAI response and future expected revision to the FSAR Chapter 17, the staff considers RAI 71, Question 03.02.01-4 resolved, but a follow-up RAI is needed to clarify the basis for the non-seismic classification of any risk-significant SSC. The response to Chapter 17 RAI provided a list of systems that were added to the reliability assurance program. In Table 17.04-1-1 attached to the response to RAI 5, Question 17.04-1, components such as the station blackout (SBO) Diesel Generators XKA50 and XKA80 are identified as risk significant components; however, in Table 3.2.2-1 of the FSAR, the SBO generators are designated as non-seismic. As stated previously, the non-seismic SSCs are not credited in the seismic margin assessment, therefore in RAI 420, Question 03.02.01-14; the staff requested that the applicant clarify the basis for the non-seismic classification of any risk-significant SSC such as the SBO diesel generators.

In an April 13, 2011, response to RAI 420, Question 03.02.01-14, the applicant states that the guidance for determining whether a component is seismic or non-seismic is provided in RG 1.29 in accordance with SRP Section 3.2.1. RG 1.29 does not identify the SBO diesel generators in the list of SSCs that should be designated as Seismic Category I; therefore they are classified as non-seismic. The response clarifies that this is consistent with RG 1.155, Appendix B which states that seismic qualification is not required for SBO equipment and that they are not required to be safety–related. The response clarified that for seismic risk significance determination, the U.S. EPR design uses a PRA-based seismic margin assessment to determine seismic-related risk significance as part of an overall input to the reliability assurance program. The U.S. EPR PRA-based seismic margin assessment does not credit any non-seismic equipment to meet the commitment for a high confidence low probability of failure plant-level capacity of 1.67 times the SSE. Therefore, the response stated that no additional risk significant SSCs currently classified as non-seismic are required to meet GDC 2 or any other licensing commitments related to seismic design.

FSAR Tier 2, Section 3.2.1 was not revised to reference RG 1.155 or other basis. However, as a result of the clarifications included in the response, the staff considers RAI 420, Question 03.02.01-14 resolved.

Auditable Information

10 CFR Part 52.47, "Content of Applications; Technical Information," identifies that the NRC will require prior to design certification, that information normally contained in certain procurement specifications and construction and installation specifications be completed and available for audit. FSAR Tier 1, Chapter 2, "System Based Design Descriptions and ITAAC," includes system-based design descriptions including structures. This chapter identifies that specifications exist for components, piping, and supports shown as ASME Code, Section III. The staff understood that this information is based on the information included in FSAR Tier 2, and design specifications are required for ASME Code, Section III systems and components, but it is unclear to the staff if specifications exist for structures and non-ASME systems and components. Therefore, In RAI 71, Question 03.02.01-5, the staff requested that the applicant clarify if the design information on seismic classification for all important to safety SSCs within the scope of the certified design, including structures, is included in specifications and if this information is now available for audit.

In a November 14, 2008, response to RAI 71, Question 03.02.01-5, the applicant stated that the design information contained in the FSAR Tier 2 portion of the design certification application is provided in system design requirements documents, system description documents, and P&IDs. The design information on the seismic classification for SSCs within the scope of the design, including structures, is included in these design documents which are available for NRC inspection. The applicant also clarified that the statements in FSAR Tier 1 are written in the present tense as they would exist at the time that a closeout letter is submitted. The FSAR Tier 1 statement that specifications exist does not imply that they currently exist. The applicant stated that the design information contained in the FSAR Tier 2 portion of the design certification application is provided in system design requirements documents, system description documents and P&IDs. The design information on the seismic classification for SSCs within the scope of the certified design, including structures, is included in these design documents which are available for NRC inspection. The applicant also clarified that the statements in FSAR Tier 1 are written in the present tense as they would exist at the time that a closeout letter is submitted. The FSAR Tier 1 statement that specifications exist does not imply that they currently exist. RAI 71, Question 03.02.01-5 is closed and the staff will schedule the audit when the design information is available. In follow-up RAI 420 Question 03.02.01-15 the applicant was requested to identify when such design information will be available. The RAI response noted that the statements in the U.S. EPR Tier 1 sections referring to the existence of specifications were deleted in response to other RAIs. The April 13, 2011, RAI 420 Question 03.02.01-15 response referenced RAI 107, Question 3.9.3-4 and RAI 404, Question 03.09.03-24 to address the availability of design specifications. The issue of availability of design documents for audit is pending an acceptable response until RAI 107, Question 3.9.3-4 and RAI 404, Question 03.09.03-24 are resolved; therefore **RAI 420, Question 03.02.01-15 is being tracked as an confirmatory item** until the audit is completed.

ITAAC

FSAR Tier 1, Chapter 2 and FSAR Tier 2, Section 14.3, “Inspection, Test, Analysis, and Acceptance Criteria,” describe various ITAAC to confirm that systems designated as ASME Code, Section III have been designed and tested in accordance with ASME Code requirements. It is unclear to the staff if there is a proposed ITAAC to address the design and testing of other systems that may be important to safety that are not constructed to ASME Code, Section III. Therefore, in RAI 71, Question 03.02.01-6, the staff requested that the applicant identify if there is an ITAAC to address the design and analysis of other important to safety systems that are not designated as ASME Code, Section III, or explain why an ITAAC is not required.

In a November 14, 2008, response to RAI 71, Question 03.02.01-6, the applicant clarified that ITAAC are also provided in FSAR Tier 1 for safety-significant systems that are not specified as ASME Code, Section III. For example, ITAAC are provided in FSAR Tier 1, Section 2.3.3, “Severe Accident Heat Removal System,” for portions of the severe accident heat removal system (SAHRS), such as the SAHRS pump, SAHRS heat exchanger, and spray header that are not specified as ASME Code, Section III. The response did not completely address staff’s concern since ITAAC for important to safety SSCs such as Seismic Category II SSCs was not addressed and additional information is needed to resolve the concern. The staff closed RAI 71, Question 03.02.01-6 and issued follow-up RAI 201, Question 03.02.01-10 to address this remaining staff concern.

FSAR Tier 1, Table 2.2.8-2, “FHS ITAAC,” lists ITAAC for Seismic Category II equipments to ensure that they can withstand design basis seismic event without losing their structural stability. However, in other sections of FSAR Tier 1, there are no ITAAC for Seismic Category II SSCs (e.g., reactor coolant system, liquid radwaste system, etc.) In RAI 201, Question 03.02.01-10, the staff requested that the applicant review all ITAAC tables to include Seismic Category II SSCs and that the applicant provide a basis for any Seismic Category II SSCs that are not covered by an ITAAC.

In a May 06, 2009, response to RAI 201, Question 03.02.01-10, the applicant stated that safety-significant design features are included in FSAR Tier 1, and the associated Seismic Category II entries in FSAR Tier 1 tables will be deleted. RG 1.29, Regulatory Position C.2 states that non-safety-related SSCs that can reduce the function of safety-related SSCs should be designed and constructed to withstand the effect of an SSE. Seismic Category II SSCs can have safety significance if their failure will impact the function of safety-related SSCs. The applicant has not explained the basis for finding that Seismic Category II SSCs are not safety-significant such that the requirement could be removed from FSAR Tier I.

If the applicant decides not to have Seismic Category II ITAAC on an SSC level, at a minimum, there should be a generic ITAAC to ensure that the as-built non-safety-related SSCs in the plant will not reduce the function of safety-related SSCs during and after an SSE. Therefore, in RAI 420, Question 03.02.01-16, the staff requested that the applicant clarify if a generic ITAAC exists to verify classifications.

In an April 13, 2011, response to RAI 420, Question 03.02.02-16, the applicant referenced the September 2, 2010, response to RAI 370, Question 03.07.03-38 that provided a new generic ITAAC in FSAR Tier 1, Section 3.9 to verify that the as-built non-safety-related SSCs in the plant will not reduce the function of safety-related SSCs during and after an SSE. Since the applicant provided an ITAAC in Section 3.9 for non-safety-related SSCs the staff concern associated with RAI 420, Question 03.02.01-16 is resolved, pending revision to the FSAR tracked under RAI 370, Question 03.07.03-38.

Risk-Significant Electrical Systems Classified as NSC

GDC 2 requires that SSCs that are important to safety be designed to withstand the effects of earthquakes. Certain electrical systems that are considered risk-significant are identified in FSAR Tier 2, Table 3.2.2-1 as non-safety-related and NSC. For example, portions of the process automation system (PAS), protection system (PS), normal power supply system (NPSS), 12-hour uninterruptible power supply system (12UPS), and alternate alternating current (AAC) source electrical system are identified as having a high review level in the NRC risk insights document that is based on the applicant's Chapter 19 information, but these systems are identified as non-safety-related and are classified as NSC. In RAI 71, Question 03.02.01-7, the staff requested that the applicant identify the basis for the NSC classification for these potentially risk-significant and important to safety electrical systems.

In a November 14, 2008, response to RAI 71, Question 03.02.01-7, the applicant referred to its November 14, 2008, response to RAI 71, Question 03.02.01-3 which referenced GL 84-01 for application of the term important to safety. The response also referred to Chapter 19 RAI responses that provided a list of SSCs modeled in the PRA-based seismic margin assessment. However, Table 19.1-107 provided in the responses does not list specific equipment with component numbers, and the basis for classifying the PAS, PS, NPSS, 12UPS and AAC electrical systems as NSC stills needs to be clarified. The staff closed RAI 71, Question 03.02.01-7 and in RAI 420, Question 03.02.01-17, the staff again requested that the applicant justify the seismic classification of risk-significant electrical systems that may be important to safety. Alternatively, if the seismic classification of electrical systems is addressed in FSAR Tier 2, Chapter 8, the applicant should so indicate.

In an April 13, 2011, response to RAI 420, Question 03.02.01-17, the applicant references the response to RAI 234, Question19-304 that revised the FSAR to include a list of specific SSCs credited in the PRA-based seismic margin assessment. The list is shown in FSAR Tier 2, Table 19.1-106. The U.S. EPR PRA-based seismic margin assessment does not credit any non-seismic equipment to meet the commitment for a high confidence, low probability of failure plant-level capacity of 1.67 times the SSE. Therefore, no SSC currently classified as non-seismic, such as process automation system (PAS), preferred power supply (PPS), Normal Power Supply System (NPSS), 12-hour uninterruptible power supply (12UPS) or AAC electrical systems are required to meet GDC 2 or any other licensing commitments related to seismic design. Since the response references a list of specific SSCs credited in the PRA for a seismic event, the response is adequate to resolve the staff concern. Seismic classifications of electrical systems are to be further evaluated in Chapter 8 of this report. The staff considers all issues associated with RAI 420, Question 03.02.01-17 resolved.

List of SSCs Required for Continued Operation

10 CFR Part 50, Appendix S, IV(a)(2)(I) states that SSCs necessary for continued operation without undue risk to the health and safety of the public must remain functional and within applicable stress, strain, and deformation limits when subject to the effects of the OBE ground motion. SRP Section 3.2.1 states that, if the applicant has set the OBE ground motion to the value one-third of the SSE ground motion, then the applicant should also provide a list of SSCs

necessary for continued operation that must remain functional without undue risk to the health and safety of the public and within applicable stress, strain, and deformation, during and following the OBE. FSAR Tier 2, Section 3.7, "Seismic Design," states that the U.S. EPR standard plant design is defined as one-third of the standard plant SSE. In RAI 201, Question 03.02.01-9 and in follow-up RAI 291, Question 03.02.01-11, the staff requested that the applicant provide this list of SSCs necessary for continued operation. If FSAR Tier 2, Table 3.2.2-1 serves this purpose, the applicant the staff requested that the applicant clearly state in the FSAR that the table contains the list of SSCs necessary for continued operation.

In a May 06, 2009, response to RAI 201, Question 03.02.01-9 and in a September 10, 2009, response to RAI 291, Question 03.02.01.11, the applicant stated that those SSC that are designed to withstand an SSE are classified as Seismic Category I and are listed in FSAR Tier 2, Table 3.2.2-1. This classification is in accordance with SRP Section 3.2.2-1. Based on the applicants statement that the list is addressed through FSAR Tier 2, Table 3.2.2-1 and the staff finding the Table acceptable, the staff considers RAI 201, Question 03.02.01-9 and RAI 291, Question 03.02.01-11 resolved.

3.2.1.5 *Combined License Information Items*

Table 3.2.1-1 provides a list of seismic classification related COL information item numbers and descriptions from FSAR Tier 2, Table 1.8-2:

Table 3.2.1-1 U.S. EPR Combined License Information Items

Item No.	Description	FSAR Tier 2 Section
3.1-1	A COL applicant that references the U.S. EPR design certification will identify the site-specific QA Program Plan that demonstrates compliance with GDC 1.	3.1.1.1.1
3.2-1	A COL applicant that references the U.S. EPR design certification will identify the seismic classification of applicable site-specific SSCs that are not identified in Table 3.2.2-1.	3.2.1

The staff determined the above listing to be complete. Also, the list adequately describes actions necessary for the COL applicant. No additional COL information items need to be included in FSAR Tier 2, Table 1.8-2 for seismic classification consideration.

3.2.1.6 *Conclusions*

On the basis of its review of FSAR Tier 2, Section 3.2.1 and Table 3.2.2-1, the applicable simplified P&IDs, and other supporting information in FSAR Tier 2, the staff concludes that the U.S. EPR safety-related SSCs, including their supports, are properly classified as Seismic Category I, in accordance with RG 1.29, Regulatory Position C.1. In addition, the staff concludes that FSAR Tier 2 includes acceptable consistency with RG 1.29, Regulatory

Positions C.2, C.3, and C.4, and that, except for open items associated with resolution of seismic and QA requirements for certain non-safety-related SSCs and risk-significant candidates, the necessary SSCs are properly classified as Seismic Category II. This constitutes an acceptable basis for satisfying, in part, the portion of GDC 2 which requires that all SSCs important to safety be designed to withstand the effects earthquakes.

This conclusion is based on:

- The applicant having met the requirements of GDC 1 by providing information in the FSAR that, except for certain non-safety-related SSCs, Seismic Category I SSCs will be designed, constructed, and operated under a quality assurance program, in compliance with the requirements of 10 CFR Part 50, Appendix B.

- Pending resolution of conventional seismic classification requirements for risk-significant candidates, the applicant having met the requirements of GDC 2 and 10 CFR Part 50, Appendix S, by having properly classified safety-related SSCs as Seismic Category I in accordance with the positions of RG 1.29.

- Pending resolution of seismic requirements for risk-significant candidates, those SSCs not identified as Seismic Category I, but whose failure could reduce the functioning of any Seismic Category I feature to an unacceptable safety level or result in incapacitating injury to control room personnel, are identified for analysis to assure that they will not fail during an SSE in accordance with RG 1.29.

- The applicant having met the guidance in RG 1.143, RG 1.151, and RG 1.189, with regard to the establishment of seismic design requirements for radioactive waste systems, instrument sensing lines, and fire protection SSCs, respectively.

3.2.2 System Quality Group Classification

3.2.2.1 *Introduction*

NRC regulations require 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. This requirement is applicable to both pressure-retaining and nonpressure-retaining SSCs that are part of the reactor coolant pressure boundary (RCPB) and other systems important to safety. As defined in 10 CFR Part 50, Appendix A, important to safety SSCs are those that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. The plant will rely on safety-related SSCs for the following functions:

- prevent or mitigate the consequences of accidents and malfunctions originating within the RCPB

- permit shutdown of the reactor and maintain it in a safe-shutdown condition

- retain radioactive material

Further components that are part of the RCPB must meet the requirements for ASME Code Class 1 components in ASME Boiler and Pressure Vessel (B&PV) Code, Section III. Quality

Group A standards that are required for pressure-containing components of the RCPB comply with ASME B&PV Code, Section III, Class 1. Quality Group B and Quality Group C must meet the requirements for ASME Code Class 2 and Class 3, respectively. RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components Of Nuclear Power Plants Quality Assurance Programs," Revision 4, identifies those fluid systems or portions of systems and system functions classified as Quality Group B, C, and D and their applicable quality standards.

3.2.2.2 *Summary of Application*

In FSAR Tier 2, Section 3.2.2, "System Quality Group Classification," and FSAR Tier 2, Table 3.2.2-1, the U.S. EPR safety-related fluid systems and components are classified as Quality Group (QG) A, B, or C. Non-safety-related fluid systems and components that do not fall within QG A, B, or C also appear in these tables as QG D and E. FSAR Tier 2, Table 3.2.2-1 identifies the safety classification as "S" for safety-related, "NS" for non-safety-related, and "NS-AQ" for supplemented grade. FSAR Tier 2, Table 3.2.2-1 also includes the basic commercial codes and standards applicable to major SSCs and the SSCs to which 10 CFR Part 50, Appendix B applies. The applicable chapters on various systems, together with simplified P&IDs in other sections of FSAR Tier 2, also identify applicable codes and industry standards, as well as quality and safety classifications for fluid systems.

FSAR Tier 2, Section 3.1.1.1, "Criterion 1 – Quality Standards and Records," identifies that, in regard to GDC 1, the Quality Assurance Plan described in FSAR Tier 2, Section 17.5, "Quality Assurance Program Description," provides confidence that safety-related SSCs are designed to quality standards commensurate with the safety-functions to be performed.

FSAR Tier 2, Section 3.2.2 states that, to meet the requirements of 10 CFR 50.55a(a)(1) and GDC 1, the U.S. EPR complies with the requirements of 10 CFR 50.55a(c) for the reactor coolant pressure boundary, and conforms to the guidance of RG 1.26 for "other safety-related components containing water, steam, or radioactive material." FSAR Tier 2, Table 1.9-2, identifies the U.S. EPR conforms to RG 1.26, Revision 4; RG 1.143, Revision 2; and RG 1.151, Revision 7.

3.2.2.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 3.2.2 and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Section 3.2.2.

- 10 CFR Part 50, Appendix A, GDC 1 and 10 CFR Part 50.55a, as they relate to structures, systems, and components important to safety being designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety-function to be performed.

Acceptance criteria adequate to meet the above requirements include:

1. RG 1.26 is an acceptable method of meeting the requirements of GDC 1 and 10 CFR 50.55a. This guide describes an acceptable method for determining quality standards for portions of systems that are not part of the reactor coolant pressure

boundary and defined as Quality Group B, C, and D water and steam containing components important to safety in light-water-cooled nuclear power plants.

2. RG 1.143, as it relates to the classification and application of quality standards for radwaste management systems.

3. RG 1.143, as it relates to the classification and application of quality standards for instrument sensing lines

3.2.2.4 *Technical Evaluation*

The staff reviewed the FSAR in accordance with SRP Section 3.2.2 and the guidance contained in RG 1.26, Revision 4. The review included evaluation of the criteria used to establish the Quality Group classifications and the application of the criteria to the classification of principal components included in FSAR Tier 2, Table 3.2.2-1. Additional information beyond the information contained in the original application was required to complete this review, and the applicant was requested to respond to RAIs discussed below under each review topic.

3.2.2.4.1 Classification Criteria

The staff reviewed the criteria and methodology identified in FSAR Tier 2, Section 3.2.2 used to select the appropriate quality classification in FSAR Tier 2, Table 3.2.2-1 for principal components. The staff determined that the classification criteria comply with 10 CFR 50.55a and conform to RG 1.26 for QG classification, except the staff is concerned that certain non-safety-related components that may be important to safety appear to have no supplemental quality requirements. In FSAR Tier 2, Table 1.9-2, the applicant committed to conform to RG 1.143 for radwaste systems. The classification of radwaste systems relative to RG 1.143 guidance is also addressed in FSAR Tier 2, Chapter 11, “Radioactive Waste Management,” and is evaluated in Chapter 11 of this report. In FSAR Tier 2, Table 1.9-2, the applicant committed to conform to RG 1.151 for instrument sensing lines. The staff evaluation of the applicant’s classification criteria are further addressed below.

Compliance with GDC 1

GDC 1 requires, in part, that SSCs important to safety shall be designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety-functions to be performed. Where generally recognized codes and standards are used, they shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety-function. FSAR Tier 2, Section 3.2, in combination with FSAR Tier 2, Section 3.1.1.1.1, identifies that safety-related SSCs are designed to quality standards commensurate with their safety-functions or to fail in a safe condition. GDC 2 applies to all important to safety SSCs and not only SSCs that are considered safety-related. As defined in 10 CFR Part 50, Appendix A, important to safety SSCs are those that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public.

In RAI 72, Question 03.02.02-1, the staff requested that the applicant clarify in FSAR Tier 2, Sections 3.2 and 3.1.1.1.1 how GDC 1 is satisfied relative to SSCs that are not identified as safety-related, but are considered important to safety and have augmented quality requirements (e.g. NS-AQ), such as the non-safety-related fire protection system or any SSC that is classified as Seismic Category II.

In a November 14, 2008, response to RAI 72, Question 03.02.02-1, the applicant referenced the November 14, 2008, response to RAI 71, Question 03.02.01-1 concerning GDC 2 and seismic classification of important to safety SSCs that are not considered safety-related. The November 14, 2008, response to RAI 71, Question 03.02.01-1 stated that, while the term "safety-related" is well defined and codified, this is not true of the term, "important to safety." The response refers to supplemented grade as non-safety-related, but to which a significant licensing requirements or commitment applies. However, the supplemental requirements (special treatment) are not clearly defined for these SSCs. The response concludes that the safety and seismic classifications of the U.S. EPR SSCs conform to NRC regulations, guidance, industry standards, and NRC-accepted regulatory precedent.

Based on the applicant's November 14, 2008, response to RAI 71, Question 03.02.01-1, the applicant's process to apply the terms, "safety-related" and "important to safety" to the classification of SSCs is considered unclear and unresolved such that additional information is needed to clarify how these terms are applied and to explain the process to develop supplemental quality requirements (special treatment) for non-safety-related risk-significant SSCs considered important to safety to satisfy GDC 1.

As stated in RAI 71, Question 03.02.01-1, the term, "important to safety," used in many of the GDC applies to SSCs that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. This definition of this term is included in the introduction to 10 CFR Part 50, Appendix A. The term, "risk" is further considered in "risk-informed guidance," and is evaluated in the probabilistic risk assessment in combination with the regulatory treatment of non-safety systems processes. Safety-related SSCs are defined in 10 CFR 50.2, as those SSCs that are relied upon to remain functional during and following design-basis events to ensure one of three important safety-functions.

An important NRC guidance document concerning this topic is the November 20, 1981, memorandum from Harold Denton, Director of the Office of Nuclear Reactor Regulation (NRR), to all NRR personnel. The current 10 CFR Part 50 definitions were endorsed and safety-related SSCs were considered to represent a subset of important to safety SSCs. This document was reviewed by the Utility Safety Classification Group, a group representing 30 electric utility owners of nuclear power plants, and their August 26, 1983, letter to the NRC identified this as an issue of major importance with increasing prominence.

Also in regard to the use of these terms relative to quality assurance, NRC Generic Letter (GL) 84-01, "NRC Use Of The Terms 'Important To Safety' and 'Safety-Related'," stated "….where we have found that quality assurance requirements beyond normal industry practice were needed for equipment "important to safety," we have not hesitated in imposing additional requirements commensurate with the importance to safety of the equipment involved."

This concern was addressed in various rulemaking efforts including consideration of a risk-informed approach described in Commission Paper SECY-98-300, "Options for Risk-Informed Revisions to 10 CFR Part 50 – 'Domestic Licensing of Production and Utilization Facilities'," Part of this effort was the optional risk-informed classification approach of 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors," and considered in RG 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance," concerning categorization for special treatment. Another option considered was to expand a 10 CFR 50.2 definition or define a currently used 10 CFR Part 50 term. This alternative may have better defined important to safety, but this alternative was not selected.

FSAR Tier 2, Section 19.1.7.5 identifies that the RTNSS process is not applicable to the U.S. EPR, but risk-significant SSCs, including those that are non-safety-related, are required to be identified.

The Electric Power Research Institute (EPRI) Utility Requirements Document, April 1992, for advanced light-water reactors (ALWR) identifies that essential equipment includes all safety-related equipment and may include some non-safety-related equipment, based on PRA insights. The EPRI Utility Requirements Document also includes the basis for RTNSS systems and identifies that the plant designer shall identify and document risk-significant non-safety SSC functional reliability/availability (R/A) missions from the focused PRA. All non-safety SSCs are subject to assessment regarding their risk-significant functions.

Therefore, the response does not adequately address the request in regard to compliance with GDC 1 for non-safety-related SSCs that are important to safety. Specifically, there should be a process in place to assure that those risk-significant non-safety-related SSCs have appropriate special treatment, such as a QA program and appropriate design considerations, to ensure reliability consistent with their safety function, the D-RAP and reliability assumed in the PRA. For example, the process to apply the NS-AQ supplemented safety classification to certain SSCs should be explained so that GDC 1 is satisfied for all important to safety SSCs and not just those specifically designated as safety-related. The staff closed RAI 72, Question-03.02.02-1 and in RAI 435, Question 03.02.02-12, the staff requested that the applicant clarify in the FSAR how this consideration is accomplished or provide a pointer in the FSAR to the sections where this methodology is considered if the process is already described. Until the applicant adequately responds to resolve this concern, **RAI 435, Question 03.02.02-12 is tracked as an open item**.

Provided all important to safety SSCs are classified such that they are designed to quality standards commensurate with their safety significance, compliance with GDC 1 will be satisfied. Except for the open item discussed above, the staff agrees that, QG classification of U.S. EPR pressure retaining components and their supports meets the acceptance criteria in SRP Section 3.2.2 and conforms to the guidance in RGs 1.26, 1.143, 1.151, and 1.189. RG 1.26 provides general guidance on QG classifications for pressure retaining components and their supports that perform important to safety-functions. Other RGs such as RG 1.143, RG 1.151, and RG 1.189 provide guidance on specific important to safety-systems. By conforming to the guidance in the SRPs and RGs, the applicant will provide reasonable assurance that the important to safety-pressure retaining systems and their supports will be designed to quality standards commensurate with their safety significance and, therefore, would satisfy, the requirements of GDC 1 in part.

3.2.2.4.2 Application of Quality Group Classification Criteria

The staff reviewed the application of the applicant’s quality classification criteria to safety-related fluid systems identified in RG 1.26 and non-safety-related systems that may be risk-significant. Based on the examples reviewed, the staff identified that additional information was needed to confirm that the criteria were appropriately applied to the classification of specific fluid system components.

Scope of SSCs within the Quality Assurance Program

GDC 1 requires, in part, that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety-functions to be performed. Where generally recognized codes and standards are used, they shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety-function. The QA Plan described in TR ANP-10266A, Revision 1, "AREVA NP Inc., Quality Assurance Plan (QAP) for Design Certification of the U.S. EPR Topical Report," applies to both safety-related and non-safety-related SSCs, but this report does not identify a specific list of important to safety SSCs that require application of the 10 CFR Part 50, Appendix B QA program or the list of non-safety-related SSCs that come under to the QA program that do not comply with 10 CFR Part 50, Appendix B. FSAR Tier 2, Table 3.2.2-1 does include a list of safety-related and non-safety-related SSCs defined as NS-AQ that require the application of a 10 CFR Part 50, Appendix B program, but the list of specific non-safety-related SSCs that come under the QA program that do not comply with 10 CFR Part 50, Appendix B is not clearly defined.

In RAI 72, Question 03.02.02-2, the staff requested that the applicant clarify which non-safety-related SSCs apply to the QA program for non-safety-related SSCs and identify if these SSCs have a unique quality classification.

In a November 14, 2008, response to RAI 72, Question 03.02.02-2, the applicant identifies that SSCs classified as supplemental grade (NS-AQ) are included in the 10 CFR 50 Appendix B QA Program, if inclusion is explicitly invoked by the relevant significant licensing requirement or commitment. The response references the November 14, 2008, response to RAI 71, Question 03.02.01-1 for further discussion of the NS-AQ classification. The staff is concerned that SSCs with a safety classification of NS-AQ that may be important to safety do not consistently invoke the 10 CFR Part 50, Appendix B program or elements of a similar program. For example, the Station Blackout diesel generators considered risk-significant and is classified as NS-AQ, but there is no 10 CFR Part 50, Appendix B program or similar special treatment identified in the Classification Table 3.2.2-1. The staff closed RAI 72, Question 03.02.02-2 and in RAI 420, Question 03.02.02-7, the staff requested that the applicant review classification FSAR Tier 2, Table 3.2.2-1 and identify those additional risk-significant SSCs that should apply the 10 CFR Part 50, Appendix B program or similar special treatment provisions. **RAI 420, Question 03.02.02-7 is being tracked as an open item.**

Additionally, FSAR Tier 2, Section 3.2.2, COL Information Item 3.2-2 states that a COL applicant that references the U.S. EPR design certification will identify the QG classification of applicable site-specific SSCs. The staff finds the COL information item which identifies the COL applicant's responsibility for classifying the quality group of site-specific SSCs, acceptable.

Supplemental Requirements for NS-AQ SSCs

FSAR Tier 2, Section 3.2 describes Supplemented Grade as those SSCs deemed to be important by the staff. Important to safety SSCs are not deemed important by the staff, but are identified as important to safety on the basis of the safety function as determined by the applicant's evaluation process such as the PRA, expert panel, or other RTNSS process. FSAR Tier 2, Table 3.2.2-1 identifies those SSCs that are defined as NS-AQ. In RAI 72, Question 03.02.02-3, the staff requested that the applicant revise FSAR Tier 2, Section 3.2

wording to clarify the applicant's process to determine SSCs that are important to safety and, for those SSCs classified as NS-AQ, identify the supplemental design and quality requirements to ensure the reliability assumed in the PRA.

In a November 14, 2008, response to RAI 72, Question 03.02.02-3, the applicant referred to its November 14, 2008, response to RAI 71, Questions 03.02.01-1 and 03.02.01-4, and FSAR Tier 2, Section 17.4 for a description of the reliability assurance program. The responses to the referenced RAIs and the description of the reliability assurance program in FSAR Tier 2, Section 17.4 do not adequately identify the list of risk-significant SSCs or define the supplemental design and quality requirements for each non-safety-related SSC classified in FSAR Tier 2, Table 3.2.2-1 as NS-AQ that may be important to safety. For example, passive components such as piping are not included in the list of risk-significant SSCs included in the applicant's June 03, 2008, response to RAI 5, Question 17.04-1. The staff closed RAI 72, Question 03.02.02-3 and in RAI 420, Question 03.02.02-8, the staff requested that the applicant identify or reference the list of non-safety-related SSCs that require special treatment in the FSAR and confirm that all non-safety-related SSC are or will be included in FSAR Tier 2, Table 3.2.2-1 with an appropriate classification based on it safety significance. The staff also requested that the applicant identify the special treatment, or if not yet developed, revise FSAR Tier 2, Section 3.2.2 to reference the D-RAP or other process to ensure the integrity and reliability assumed in the PRA and identify when special treatment requirements are to be identified. **RAI 420, Question 03.02.02-8 is being tracked as an open item.**

Refueling Seal

FSAR Tier 2, Revision 2, Table 3.2.2-1 revised in the applicant's August 5, 2011, response to RAI 337, Question 09.01.04-14 shows the RPV refueling cavity seal as NS-AQ, Quality Group D, and Seismic Category I with a 10 CFR Part 50, Appendix B program applied. The staff finds that seismic classification as Seismic Category I with QA to comply with 10 CFR Part 50, Appendix B is consistent with a safety-related SSC and RG 1.29 and RG 1.13. However, the QG D classification for a non-safety-related component is inconsistent with the Seismic Category I classification. Therefore, in RAI 481, Question 03.02.02-13, the staff requested that the applicant address the following basic safety function and related classification concerns:

- Clarify if the refueling cavity seal is a mechanical or a structural component and describe the extent that codes and standards are applied. If considered a structural component, Quality Group should not apply and the structural branch is to review. If considered a mechanical component, describe the extent of certification and stamping and explain why this component is classified as QG D rather than QG C. A component designed to ASME Section III Subsection ND is normally designated as QG C.

- The basis for the classification as QG D has not been justified. In particular the following information is needed to evaluate the classification:

    - Establish if the seal is defined as safety-related or important to safety using the definitions in 10 CFR 50 and Appendix A

    - Describe the safety function and the basis for the designation as safety-related, important to safety or non-safety-related.

- Since the seal is classified as Seismic Category I that is normally used for safety-related SSCs, explain why the seal classified as QG D is not also considered safety-related.

- If the seal is considered safety-related, the basis for the classification as QG D should be described.

- If the seal is defined as non-safety-related, but is important to safety concerning the risk to health and safety of the public, describe the evaluation of risk-significance.

- If the seal is not postulated to fail, justify why a single failure (rupture or crack) is not postulated to occur with consideration of specified quality requirements.

- If the seal could fail or leak as a postulated passive failure during refueling operations explain why the seal failure will not result in excess off-site doses (ref. RG 1.26 Regulatory Position C.2(d).

The FSAR was revised in the August 5, 2011, response to RAI 337, Question 09.01.04-14 and, in a follow-up RAI 504, Question 03.02.02-14, the staff requested that the applicant provide additional information on the classification of the cavity seal. In a May 11, 2011, response to RAI 481, Question 03.02.02-13, the applicant identified that the concerns addressed in RAI 481, Question 03.02.02-13 regarding classification of the refueling cavity seal will be addressed in the response to RAI 504, Question 03.02.02-14. Therefore, the staff considers RAI 481, Question 03.02.02-13 closed.

FSAR Tier 2,Revision 2, Table 3.2.2-1 adds the RPV Refueling Cavity Seal and classifies this component as non-safety-related Safety Class NS-AQ, Quality Group (QG) D and Seismic Category I. The applicant's August 5, 2011, response to RAI 337, Question 09.01.04-14 changes the name of the seal to "ring" and revises the classification from QG D to N/A. The response also clarifies that the cavity ring is a mechanical component designed in accordance with ASME Code, Section III, Subsection ND and quality group does not apply to the cavity ring since it is not a pressure-retaining component. The seismic classification as Seismic Category I pursuant to 10 CFR Part 50, Appendix B is consistent with a component that is a safety-related SSC and with RG 1.29 and RG 1.13. However, the QG classification for a non-safety-related mechanical component appears to be inconsistent with RG 1.26, the Seismic Category I classification and the ASME Code Class such that additional information is needed regarding the basis for the QG classification.

Therefore, in RAI 504, Question, 03.02.02-14, the staff requested that the applicant clarify why the refueling cavity seal/ring is not considered a structural or pressure-retaining component and describe the extent that codes and standards are applied including the following:

- Explain the specific function of the cavity seal/ring, such as precluding leakage of radioactive fluids and the differential design pressure it can withstand.

- If the ring serves the same purpose as the pool liner structure and is not considered pressure-retaining, explain why this item is not considered a structural component.

- Specifically describe the extent of certification and stamping and explain why this component is classified as QG N/A rather than QG C. A component designed to ASME Section III Subsection ND is normally designated as QG C.

Mechanical components that contain radioactive materials are normally QG C or QG D. The basis for the classification as QG N/A and NS-AQ has not been justified. Therefore, the staff also requested that the applicant provide the following information is needed to evaluate the classification.

- Establish if the cavity seal/ring is defined as safety-related or important to safety using the definitions in 10 CFR Part 50 and Appendix A and clarify if the seal is on the QA list required by 10 CFR 50.34 and 10 CFR Part 50, Appendix B.

- Describe the specific safety function and the basis for the designation as safety-related, important to safety or non-safety-related.

- Clarify if the ring/seal contains radioactive fluids and if this QG classification as N/A is an exception to RG 1.26 and, if so, include the technical justification in the FSAR.

- Since the seal/ring is classified as Seismic Category I with a 10 CFR Part 50, Appendix B QA Program that is normally used for safety-related SSCs, clarify why the seal/ring classified as QG N/A is not also considered safety-related.

- If the seal/ring is considered safety-related, describe the basis for the classification as QG N/A.

- If the seal/ring is defined as non-safety-related, but is important to safety concerning the risk to health and safety of the public, describe the evaluation of risk-significance.

- If the seal/ring is considered non-safety-related, clarify why this is a nonessential component and clarify if this is an exception to SRP Sections 9.1.2 or 9.1.3.

RAI 504, Question 03.02.02-14 is being tracked as open item.

QA Program for Certain SSCs Classified as Seismic Categories I and II

FSAR Tier 2, Sections 3.2.1.1 and 3.2.1.2, "Seismic Category II," states that Seismic Category I and II SSCs are subject to the QA program requirements of 10 CFR Part 50, Appendix B. FSAR Tier 2, Table 3.2.2-1 typically identifies that the 10 CFR Part 50, Appendix B QA program applies to SSCs classified as Seismic Category I or II. However, in FSAR Tier 2, Table 3.2.2-1, a limited number of non-safety-related SSCs classified as Seismic Category I and II are not listed under the 10 CFR Part 50, Appendix B program. For example, certain non-safety-related monitors supporting the leak detection system are identified as Seismic Category I with no 10 CFR Part 50, Appendix B program applied.

Therefore, in RAI 72, Question 03.02.02-4, the staff requested that the applicant correct this apparent discrepancy or justify the basis for not applying pertinent requirements of the 10 CFR Part 50, Appendix B program to SSCs that are classified as Seismic Category I or II.

- In a January 27, 2009, response to RAI 72, Question 03.02.02-4, the applicant stated that a review of FSAR Tier 2, Table 3.2.2-1 determined that the 10 CFR Part 50, Appendix B program was not applied to certain Seismic Category I items including the sampling activity monitoring system's (designated KLS in the Kraftwerks Kennzeichen System (KKS) for coding systems and components) mechanical components and radioactivity monitors used to support the leak detection system. The applicant indicated that FSAR Tier 2, Table 3.2.2-1 will be revised to apply the 10 CFR Part 50, Appendix B program to these mechanical components. However, the response did not address Seismic Category II SSCs, such as the Safeguard Building Controlled Area Ventilation System (KLC) Fire Dampers. The staff closed RAI 72, Question 03.02.02-4, and in RAI 420, Question 03.02.02-9, requested that the applicant clarify if all Category II SSCs, such as the Safeguard Building Controlled Area Ventilation System (KLC) Fire Dampers apply pertinent requirements of the 10 CFR Part 50, Appendix B program and update the FSAR Tier 2, Table 3.2.2-1 to be consistent.

- In an April 13, 2011, response to RAI 420 Question 03.02.02-9, the applicant states that the safeguard building controlled area ventilation system (KLC) fire dampers were changed to safety-related and Seismic Category I and applied the 10 CFR Part 50, Appendix B program (see the markups to the FSAR associated with RAI 277, Question 09.04.05-2). The applicant has determined that 10 CFR Part 50, Appendix B should be applied to the other components which are also Seismic Category II and the FSAR Tier 2, Table 3.2.2-1 will be revised.

Therefore staff's concerns pertaining to RAI 72 Question 03.02.02-4 and RAI 420 Question 03.02.02-9 are resolved. **RAI 420, Question 03.02.02-9 being tracked as a confirmatory item** to confirm that the FSAR has been revised.

Codes and Standards (GDC 1)

As acknowledged in FSAR Tier 2, Table 1.9-4, "U.S. EPR Conformance with Advanced and Evolutionary Light-Water Reactor Design Issues (SECY-93-087)," pertaining to SECY-93-087, Issue II.A, the NRC Staff Requirements Memorandum (SRM), that the staff will review both evolutionary and passive plant design applications using the newest codes and standards endorsed by the NRC, and unapproved revisions to the codes will be reviewed on a case-by-case basis. Editions of various industrial codes and standards referenced in FSAR Tier 2, Section 3.2.3, "References," and FSAR Tier 2, Table 3.2.2-1 notes are not current, and industrial codes and standards for certain SSCs, such as structures and cooling tower fans, are not given in FSAR Tier 2, Section 1.9, "Conformance with Regulatory Criteria," or FSAR Tier 2, Table 3.2.2-1. To comply with GDC 1, and conform to RG 1.26 and SRP Section 3.2.2, codes and standards for important to safety-fluid systems and their supports should be identified and supplemented as necessary to achieve a quality product.

In RAI 72, Question 03.02.02-5, the staff requested that the applicant include missing codes and standards and clarify which code editions applied to the U.S. EPR design are currently endorsed by the NRC. Additionally, the staff requested that the applicant clarify if current editions of industrial codes and standards will be applied to the detailed design and procurement of U.S. EPR SSCs so that these editions may be reviewed on a case-by-case basis.

In a January 27, 2009, response to RAI 72, Question 03.02.02-5, the applicant stated that FSAR Tier 2, Table 3.2.2-1 will be revised to include missing codes and standards applicable to certain mechanical systems with a QG B, C or D classification. Additional codes and standards applicable to various mechanical components should be acceptable, provided the specific editions are reviewed and endorsed or approved by the NRC and included in the FSAR. **RAI 72, Question 03.02.02-5 is being tracked as a confirmatory item**.

3.2.2.4.3 Onsite Reviews

The detailed design includes additional information that should be reviewed by the staff on a sampling basis to establish that the design is essentially complete and that there are appropriate design processes in place to meet regulations pertaining to QG classifications. A combination of onsite reviews of the classification design-basis information and ITAAC are intended to ensure that QG classifications are properly translated into design and procurement documents and the as-built QG classifications conform to the design.

Auditable Information

10 CFR Part 52.47 identifies that prior to design certification, the NRC will require that information normally contained in certain procurement specifications and construction and installation specifications be completed and available for audit. FSAR Tier 1, Chapter 2, "Site Characteristics," includes system-based design descriptions. This chapter identifies that specifications exist for components, piping, and supports shown as ASME Code, Section III. In RAI 72, Question 03.02.02-6, the staff requested that the applicant clarify if the design information on QG classification for all important to safety systems and components within the scope of the FSAR is included in Design Specifications and if this information is now available for audit. In a November 14, 2008, response to RAI 72, Question 03.02.02-6, the applicant referenced its November 14, 2008, response to RAI 71, Question 03.02.01-5, which states the specifications will exist when a closeout letter is submitted and will be available for NRC inspection. The staff plans to audit this information to determine if the design is essentially complete in scope regarding quality group classifications of important to safety SSCs. Therefore, with the exception of verification, the staff considers RAI 72, Question 03.02.02-6 resolved. **RAI 72, Question 03.02.02-6 is being tracked as an open item** until an audit is conducted to complete verification of the design documents.

ITAAC

FSAR Tier 1, Chapter 2 and FSAR Tier 2, Section 14.3 describe various ITAAC to confirm that fluid systems designated as ASME Code, Section III have been designed and tested in accordance with Code requirements. It is unclear to the staff whether there is a proposed ITAAC to address the design and testing of any potential other systems that may be important to safety that are not constructed to ASME Code, Section III. During the planned audit, the applicant will be requested to identify ITAAC to address the design and analysis of other important to safety systems that are not designated as ASME Code, Section III or explain why an ITAAC is not required.

FSAR Tier 1 Subsection 1.0 identifies that Tier 1 information is derived from Tier 2 and SRP Section 14.3 states that safety findings are based on Tier 2, not Tier 1, information because Tier 1 information is derived from Tier 2. SRP Section 14.3 further identifies that Tier 1 is to be clear and consistent with Tier 2 information. In regard to the FSAR Tier 1 ASME Code Class information included in the Chapter 2 system based design descriptions and ITAAC, the applicant was requested in RAI 420, Question 03.02.02-11 to update the figures included in Tier 1 to be consistent with Tier 2 figures in terms of level of detail for ASME classifications. Until the applicant adequately responds to resolve this concern, **RAI 420, Question 03.02.02-11 is being tracked as an open item.**

Buried Piping

Based on the KKS designator applied to FSAR Tier 2, Table 3.2.2-1, the staff identified that there is the potential for use of buried piping in the U.S. EPR design. Although buried piping is not specifically identified for any important to safety-piping system within the design certification scope, the COL applicant may have buried piping in systems such as the essential service water system (ESWS). FSAR Tier 2, Section 9.2.1.3.5, “Piping, Valves, and Fittings,” states that a COL applicant that references the U.S. EPR design certification will provide a description of the materials that will be used for the essential service water system at their site location, including the basis for determining that the materials to be used are appropriate for the site location and for the fluid properties that apply (COL Information Item 9.2-4). Therefore, during the audit of detailed design documents, the applicant will be requested to confirm that appropriate quality standards are specified for any buried or non-metallic piping.

3.2.2.5 *Combined License Information Items*

Table 3.2.2-1 provides a list of system quality group classification related COL information item numbers and descriptions from FSAR Tier 2, Table 1.8-2:

Table 3.2.2-1 U.S. EPR Combined License Information Items

Item No.	Description	FSAR Tier 2 Section
3.2-2	A COL applicant that references the U.S. EPR design certification will identify the quality group classification of applicable site-specific SSCs important to safety that are not identified in Table 3.2.2-1.	3.2.2

3.2.2.6 *Conclusions*

On the basis of its review of the applicable information in the FSAR, and the above discussion, the staff concludes that, except for the identified open items, the Quality Group classifications of the pressure-retaining fluid systems and their supports important to safety, as identified in

FSAR Tier 2, Table 3.2.2-1, and related P&IDs in the FSAR, conform to RG 1.26 as supplemented by SRP Section 3.2.2, and are therefore acceptable. FSAR Tier 2, Table 3.2.2-1 and simplified P&IDs identify principal components in fluid systems (i.e., pressure vessels, heat exchangers, storage tanks, piping, pumps, valves, and applicable supports) and in mechanical systems (e.g., cranes, fuel handling machines, and other miscellaneous handling equipment). In addition, the simplified P&IDs in the FSAR identify the classification boundaries of interconnecting piping and valves. All of the above fluid systems and their supports will be constructed to conform to applicable ASME Code and industry standards. Conformance to NRC guidance including RG 1.26, RG 1.143, RG 1.151, and applicable ASME Codes and industry standards provide reasonable assurance that component quality will be commensurate with the importance of the safety-functions of these systems. The staff finds this provides the basis for fluid systems and their supports satisfying GDC 1, and is therefore acceptable.

The staff's conclusion is based on the following:

- The applicant having met the requirements of GDC 1 by having properly classified these pressure-retaining components important to safety as QG A, B, C, or D in accordance with the regulatory guidance positions of RG 1.26, March 2007; RG 1.143, and RG 1.151 with the exception of the open and confirmatory items.

- The identified pressure-retaining components include those that are necessary (1) to prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary, (2) to permit shutdown of the reactor and maintain it in a shutdown condition, and (3) to contain radioactive materials which are all required in order to meet the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the applicable exposures set forth in 10 CFR 50.34(a)(1) or 10 CFR 100.11, as applicable.

3.6 Protection Against Dynamic Effects of Postulated Rupture of Piping

3.6.3 Leak-Before-Break Evaluation Procedures

3.6.3.1 *Introduction*

10 CFR Part 50, Appendix A, GDC 4, "Environmental and Dynamic Effects Design Bases," allows the use of analyses reviewed and approved by the Commission to eliminate from the design basis the dynamic effects of postulated pipe ruptures when the analyses demonstrates that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping. A staff-approved leak-before-break (LBB) analysis permits applicants to eliminate the need to install protective hardware such as pipe whip restraints and jet impingement barriers; to mitigate the consequences of pipe breaks. The staff's review ensures that adequate consideration has been given to direct and indirect pipe failure mechanisms and other degradation sources that could challenge the integrity of piping.

3.6.3.2 *Summary of Application*

FSAR Tier 1: In FSAR Tier 1, Section 2.2.1, "Reactor Coolant System," the applicant states that the applicable piping and interconnected component nozzles given in Table 2.2.1-1, "RCS Equipment Mechanical Design," are evaluated for LBB.

FSAR Tier 2: The applicant has provided an FSAR Tier 2 description of the U.S. EPR LBB analysis in Section 3.6.3, "Leak-Before-Break Evaluation Procedures," summarized here, in part, as follows.

The applicant described the analyses it used to eliminate from the design basis the dynamic effects of certain pipe ruptures for high-energy piping systems and to demonstrate that the probability of pipe rupture is extremely low under conditions in conformance with the design basis for the piping. The application of LBB to the U.S. EPR is limited to the following high-energy piping systems: (1) Main coolant loop (MCL) piping (i.e., hot legs, crossover legs, and cold legs); (2) pressurizer surge line (SL); and (3) main steam line (MSL) piping inside the containment (i.e., from the steam generators to the first anchor point location at the Containment Building penetration). For each of these piping systems, the analyses consider various potential piping failure mechanisms (e.g., water hammer, creep, corrosion and erosion/corrosion, stress corrosion cracking, thermal aging thermal stratification, and indirect causes). The analyses also considered failure prevention and detection. Inputs for the LBB analysis included geometry and operating condition, materials, and material properties.

The U.S. EPR LBB analyses used the load combination method. For the MCL and the SL piping, the leak rate calculations, performed considering air fatigue crack morphology, were determined using AREVA NP (AREVA) computer code KRAKFLO (see FSAR Tier 2, Section 3.6.3.5.2, "Leak Rate Determination Method for Main Coolant Loop and Surge Line"). For the MSL LBB analysis, computer code SQUIRT Version 1.1 (see FSAR Tier 2, Section 3.6.3.5.3, "Leak Rate Determination Method for Main Steam Line") was used. Since the MCL and SL piping materials are highly ductile austenitic stainless steels and the welds are higher toughness gas tungsten arc welds (GTAW) rather than flux welds, both the limit load analysis and the flaw stability analysis methodologies were considered appropriate. Since the MSL is made of ferritic steel, the flaw stability methodology was considered appropriate to use for this piping system. The flaw stability analysis considered a circumferential through-wall crack in straight pipe, an axial through-wall crack in straight pipe, and a circumferential through-wall extrados crack in an elbow. The applicant performed a J-Tearing (J-T) stability analysis to determine at what applied load the crack becomes unstable. The applicant then determined the maximum allowable piping moment and identified locations for flaw stability analysis to develop Allowable Load-Limit (ALL) diagrams. The results for each of the three LBB piping systems for each of the cracked pipe geometries were described in FSAR Tier 2, Section 3.6.3.6, "Results." In order to provide a factor of safety of 10 to the actual plant leakage detection system capabilities, leak rates of 18.9 liters per minute (lpm) (5.0 gallons per minute (gpm)) for the MCL and SL and 3.79 lpm (1.0 gpm) for MSL were used for determining the leakage flaw sizes. FSAR Tier 2, Section 5.2.5, "RCPB Leakage Detection," describes the leakage detection systems for the primary coolant inside containment.

ITAAC: Item 3.7 in FSAR Tier 1, Table 2.2.1-5, "RCS Inspections, Tests, Analyses, and Acceptance Criteria," states that an analysis will be performed that assesses the LBB capability of the piping, interconnected component nozzles, and equipment given in FSAR Tier 1, Table 2.2.1-1.

Technical Specifications: There are no Technical Specifications applicable to the leak-before-break analysis; however, related Technical Specification information can be found in FSAR Tier 2, Chapter 16, Section 3.4.12, Reactor Coolant System “RCS Operational Leakage,” and Section B 3.4.12, “RCS Operational Leakage.”

3.6.3.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 3.6.3, “Leak-Before-Break Evaluation Procedures,” and are summarized below. Review interfaces with other SRP sections can be found in NUREG-0800, Section 3.6.3.

- GDC 4 as it relates to the exclusion of dynamic effects of the pipe ruptures that are postulated in SRP Section 3.6.2. The design basis for the piping means those conditions specified in the SAR, as amended, and may include regulations in 10 CFR Part 50, applicable sections of the SRP, Regulatory Guides, and industry standards such as the American Society of Mechanical Engineers (ASME) Code.

3.6.3.4 *Technical Evaluation*

This section describes the technical evaluation of the applicant’s FSAR Tier 2, Section 3.6.3 in the order in which it is presented. The section-by-section evaluation of the FSAR is presented below. The staff’s review of the applicant’s LBB evaluation procedures is closely related to the staff’s review of the reactor coolant pressure boundary leakage detection system in Section 5.2.5 of this report.

GDC 4 states that the dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the NRC demonstrate that the probability of fluid system piping rupture is “extremely low” − defined as 1×10^{-6} per reactor year in the final rule on the modification to GDC 4 requirements (52 FR 41288), October 27, 1987 − under conditions consistent with the design basis for the piping. Alternatively, a deterministic evaluation with verified design and fabrication, in addition to adequate inservice inspection, can meet the extremely low probability criterion. The deterministic evaluation is based on the requirement that structures and components are correctly engineered to meet the applicable regulations and NRC-endorsed industry codes. This section presents the review of FSAR Tier 2, Section 3.6.3. The staff followed the review guidelines of NUREG-0800, Section 3.6.3.

3.6.3.4.1 Potential Piping Failure Mechanisms

The staff reviewed the potential for failure from various degradation mechanisms that could occur over the service life of the candidate LBB piping systems. NUREG-1061, “Evaluation of Potential for Pipe Breaks,” Volume 3 identifies limitations applicable for LBB application to piping systems. In addition, the staff assessed failure mechanisms relating to LBB application including water hammer, creep damage, erosion, corrosion, fatigue, and deleterious effects of environmental conditions. Piping systems subject to failure from these mechanisms are not candidates for LBB, because the basic assumptions for LBB may be invalidated. As an example, corrosion and fatigue could result in flaws whose crack morphology may not be bounded by the postulated LBB through-wall flaw, and water hammer may result in excessive dynamic loads that are not considered in the LBB analysis. The staff also evaluated indirect failure mechanisms for the certified design that could lead to pipe rupture. These include seismic events and system over-pressurizations due to accidents resulting from human error, fires, or flooding which cause electrical and mechanical control systems to malfunction. Missiles from equipment, damage from moving equipment, and failures of structures, systems, or components in close proximity to the piping are evaluated as well.

FSAR Tier 2, Section 3.6.3.3, "Potential Piping Failure Mechanisms," addresses all the degradation mechanisms identified above. Certain additional degradation mechanisms are also addressed by the applicant in FSAR Tier 2, Section 3.6.3.3. These degradation mechanisms include stress corrosion cracking (SCC), thermal aging, and thermal stratification.

Water Hammer

For the MCL and SL, the applicant cites operating experience with existing pressurized water reactors (PWRs) and various NRC publications (i.e., NUREGS and Information Notices) as demonstrating that water/steam events as described in these documents have only resulted in support damage. These portions of the reactor coolant system are designed to preclude void formation, and since safety valve discharge loads associated with the pressurizer have been included in the component design basis, the NRC Staff concludes that MCL and SL piping will have an extremely low probability of failure from water hammer loadings.

The U.S. EPR MSL piping and its supports will be designed to accommodate dynamic loads resulting from inadvertent closure of the main steam isolation valve. The numbers of elbows and miters will be minimized to reduce the effects of steam and water hammer. Steam propelled water slugs will be prevented by features in the main steam system design and layout. Due to the low severity of steam and water hammer events described in NUREG/CR-2781, "Evaluation of Water Hammer Events in Light Water Reactor Plants," and the design of the main steam supply system, the LBB portion of the MSL piping has an extremely low probability of failure due to steam and water hammer events. Based on the information provided above, the staff finds the approach to control failure of the MSL, MCL and SL piping due to steam and water hammer events acceptable.

Creep

Creep and creep fatigue are not concerns for ferritic piping operated below 371.1 °C (700 °F) and for austenitic piping operating below 426.7 °C (800 °F). Since the MCL, SL, and MSL operate below these limits, creep and creep fatigue are not concerns for these piping systems.

Corrosion and Erosion/Corrosion

The austenitic stainless steel material used to fabricate the MCL and SL piping is resistant to corrosion, and the applicant has affirmed that the EPRI guidelines for water chemistry will be implemented. Therefore, this piping will have a very low likelihood of failure from corrosion and erosion-corrosion.

Flow-accelerated corrosion (FAC), also called erosion-corrosion, has occurred in the secondary side of PWR water-steam systems. Operating conditions and applicable secondary side-water-chemistry regimes are among the factors to be evaluated to minimize the potential for FAC in the MSL. The applicant describes its FAC monitoring program to identify piping degradation that conforms to current industry practice. With these measures, the probability of failure due to FAC will be extremely low for the MSL.

Stress Corrosion Cracking

For SCC to occur, material susceptibility, corrosive environment, and tensile stress conditions must occur simultaneously.

Material susceptibility for the MCL and SL is reduced by conformance to the requirements of the ASME Code, Section III as supplemented by the guidelines of RG 1.44, "Control of the Use of Sensitized Stainless Steel," and ASME NQA-1-1994, "Quality Assurance Requirements for Nuclear Facility Applications." The piping and welds are "L" grade which reduces the potential for sensitization. The dissimilar metal welds are Alloy 52 which is more resistant to SCC than Alloy 82/182. However, Alloy 52 is not completely immune to pressurized-water, stress-corrosion cracking (PWSCC). Contributing factors which could increase susceptibility to PWSCC include welding processes, control of welding parameters (i.e., heat input), dilution effects on dissimilar-metal welds (DMW), and chromium content in DMW. Controls to minimize dilution effects and maximize chromium content were not originally addressed by the applicant. The reference cited by the applicant (EPRI Report 1009801) confirmed the importance of this issue, since it acknowledged a gap in test data for weld dilution as it affects chromium content near the Alloy 152/52 weld interface with carbon steel and stainless steel.

The welding and welding control issues stated above were originally raised by staff in RAI 48, Questions 03.06.03-1 and 03.06.03-13 with follow up RAI 265, Question 03.06.03-23.

In a November 7, 2008, response to RAI 48, Question 03.06.03-13, the applicant provided proprietary information on experimental test results from Reverse U-Bend and Constant Elongation Rate Tensile tests which substantiated the claim that Alloy 52/152 welds in U.S. EPR LBB piping are not susceptible to PWSCC. Also, it was indicated that weld repairs that would be in contact with the fluid would be made such that there would be compressive stress conditions on the wetted surface.

Furthermore, in a December 18, 2009, response to RAI 265, Question 03.06.03-23 (follow-up to RAI 48, Question 03.06.03-13), the applicant discussed specific proprietary weld practices to demonstrate that PWSCC is not a concern due to chromium content, dilution effects, cleaning methods, weld qualifications, and environmental effects on crack growth in Alloy 690. The applicant proposed specific revisions to the FSAR consistent with this response. The staff finds these proposed revisions acceptable. **RAI 265, Question 03.06.03-23 is being tracked as a confirmatory item** for COL applicant action.

Corrosive environment for the MCL and SL relies on reactor coolant chemistry controls to prevent SCC. Non-metallic insulation for the MCL and SL piping conforms to the guidelines of RG 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel," which restricts the use of chlorides and fluorides in the insulation to prevent SCC. Since the applicant stated that the EPRI guidelines for water chemistry will be implemented, corrosive environment should not exist, thereby satisfying an extremely low probability for the potential for SCC.

Tensile stress close to the material yield stress is required in a light-water environment to initiate SCC. Since the MCL and SL are constructed to the ASME Code, Section III, there are specified margins to yield stress during normal operation for applied loads. ASME Code, Section III does not consider weld residual stress levels. These stresses can exceed yield, but with control of material susceptibility and corrosive environment, SCC potential is minimized. Therefore, in RAI 48, Question 03.06.03-1, the staff requested that the applicant clarify whether welding procedures (including repair) will be adopted that will minimize tensile stresses on the internal diameter (ID). In a September 18, 2008, response to RAI 48, Question 03.06.03-1, the applicant stated that such procedures will be used and will decrease the probability that SCC will occur. Actions to avoid intergranular attack; intergranular stress, corrosion, and cracking (IGSCC); and transgranular stress, corrosion and cracking (TGSCC) are presented in FSAR Tier 2, Section 5.2.3, "Reactor Coolant Pressure Boundary Materials," and encompass the methods discussed in this section of the staff's SER. Again, the applicant's affirmation that the EPRI guidelines for water chemistry will be implemented means that the occurrence of SCC in the MCL or SL and any resulting failures would have an extremely low probability. However, as discussed under material susceptibility above, the applicant addressed and resolved the issue of weld dilution and chromium content in a December 18, 2009, response to RAI 265, Question 03.06.03-23. However, it will be tracked as a confirmatory item as stated above.

Stress corrosion cracking of the ferritic piping in the main steam lines of any nuclear power plants has not occurred. Control of other corrosion effects stemming from general corrosion and flow-assisted corrosion relies on volatile chemistry treatment to increase cycle pH and provide a reducing environment. This water-chemistry treatment results in the lowest possible general corrosion rate such that the probability of pipe rupture due to corrosion is extremely low.

Fatigue

The MCL and SL piping are designed and constructed in accordance with the rules of ASME Code, Section III which require a fatigue analysis for Normal and Upset Condition loadings. The load combinations and a commitment to account for the effects of the reactor coolant environment on fatigue are specified in Section 3.4.1 of ANP-10264NP-A, Revision 0, "U.S. EPR Piping Analysis and Pipe Support Design Topical Report," AREVA NP, Inc., November 2008, in FSAR Tier 2. The potential for high cycle fatigue due to excessive pump vibrations is controlled via alarms in the main control room. Also, fatigue monitoring will be employed to provide an accurate assessment of fatigue over the plant lifetime. If thermal stratification occurs in the SL, it will affect the fatigue evaluation. This is addressed below in the Thermal Stratification section in this report.

Fatigue is evaluated for ASME Code Class 2 piping, which includes the MSL, following the requirements of ASME Code, Section III, paragraph NC-3611.2. The allowable stress for thermal expansion is reduced for cyclic conditions by a factor f, based on the number of equivalent full temperature cycles. The number of equivalent full temperature cycles for the MSL is less than 7,000 cycles. Therefore, in accordance with the applicable ASME Code rules, the stress range reduction factor f is equal to 1.0. In addition, the applicant states that there are no normal or upset temperatures or pressure variations that would result in significant local or through-wall stresses. ASME Code rules do not require calculation of a cumulative usage factor for ASME Code Class 2 piping. Therefore, based on the rules of Section III of the ASME Code, the NRC staff concludes that the probability of failure of the MSL due to fatigue is extremely low.

Thermal Aging

The base material for the MCL and SL is forged austenitic stainless steel and the base material for the MSL is SA 106 Grade C carbon steel. The austenitic stainless steel base metal has a very high initial toughness, so that a small amount of thermal aging has no significant effect. The A106 Grade C base metal, its weld metal, and the stainless steel weld metal may have a much lower un-aged toughness than the austenitic steel. Therefore, the staff raised concerns about thermal aging on those materials. As an example, the welds in the MCL are fabricated using the GTAW weld process and meet requirements of ASME Code, Section III and the guidance of RG 1.31, “Control of Ferrite Content in Stainless Steel Weld Metal,” − are subject to thermal aging as shown in test results from Argonne National Laboratory (NUREG/CR-6428 ANL95/47, “Effects of Thermal Aging on Fracture Toughness and Charpy-Impact Strength of Stainless Steel Pipe Welds”). Also, the reactor coolant pump casing is statically cast stainless steel and is predicted to experience the greatest reduction in toughness due to thermal aging particularly at the pump nozzle weld.

In RAI 48, Questions 03.06.03-2 and 03.06.03-6, the staff requested that the applicant address the potential for thermal aging of the MSL carbon steel welds, A106C base metal, SL stainless steel weld, and nozzle weld geometry for the reactor coolant pump housing. In a September 18, 2008, response to RAI 48, Questions 03.06.03-2 and 03.06.03-6, the applicant provided a list of applicable references used to determine whether thermal aging embrittlement is a concern for RCPB materials and welds, including the J-R curves for the RCPB materials. The MSL piping welds will be subject to some degree of thermal aging. The applicant does not believe that thermal aging for the MSL welds is likely and refers to the J-R curve toughness values used for SA-106 Grade C material for justification. However, the J-R curve does not substantiate that thermal aging will not occur. In a June 26, 2008, audit, the applicant provided the specific proprietary J-R curves for the materials along with curve fit values. In a June 3, 2009, response to follow-up RAI 265, Question 03.06.03-21, on this issue, the applicant provided a satisfactory explanation consistent with data published in NUREG/CR-6765 for the 30 percent reduction as a result of thermal aging in the J-R value for stainless steel weld material based on experimental data to account for heat-to-heat variation. The use of the lower J-R curve values in the subsequent LBB analyses therefore is conservative, and the staff concludes that the probability likelihood of failure due to thermal aging is extremely low.

Thermal Stratification

Stagnant flow conditions do not exist in the MSLs, and, therefore, the MSLs are not subject to thermal stratification and thermal striping. The MSL operates in a saturated steam environment; therefore, thermal stratification and thermal striping will not occur in the MSL. In U.S. operating reactors, the SL piping has been subject to thermal stratification and striping, because it contains essentially horizontal pipe segments that experience fluid at low flow velocities at a significantly different temperature than the fluid in the piping. The applicant states that thermal stratification is not a concern for the SL line in the U.S. EPR plant because of factors such as layout of the SL geometry and the continuous bypass spray flow. In RAI 48, Question 03.06.03-4, the staff requested that the applicant clarify whether the values of differential temperature (ΔT) and SL stratification for a French PWR will be typical of the U.S. EPR SL and how such effects will be evaluated. The staff also requested that the applicant describe in detail the “improved system operation” to minimize the ΔT, between the hot leg and pressurizer. In a September 18, 2008, response to RAI 48, Question 03.06.03-4, the applicant provided further information designated as proprietary that indicated that the stratification ΔT values for the U.S. EPR will be lower than those for existing Westinghouse and Babcock & Wilcox (B&W) plants and will be similar to the values supplied by the applicant for existing French PWRs. Stratification effects in susceptible ASME Code Class 1 piping systems such as the pressurizer surge line will be evaluated in accordance with the rules of ASME Code, Section III, NB-3600. System operation which includes constant bypass flow during normal operation and system layout that includes a vertical takeoff from the hot leg which minimizes turbulent penetration of fluid of the hot leg are effective measures that will be employed to further reduce the magnitude of thermal stratification. With improved system operation and SL layout as specified by the applicant, plus the relatively low stratification delta T values identified, the probability of significant thermal stratification and failure from its effects is extremely low. Therefore, the staff considers this issue and RAI 48, Question 03.06.03-4 resolved.

Failure from Indirect Causes

The potential for pipe degradation or failure due to the following indirect causes is described in the FSAR as follows:

Missile prevention and protection are described in FSAR Tier 2, Section 3.5, "Missile Protection." For missile prevention and protection inside containment, the effects of potential internally generated missiles will be minimized by separation and redundancy throughout containment. Missile barriers will be provided between equipment housed adjacent to one another. The U.S. EPR will be designed such that a postulated missile from the reactor coolant system does not cause a loss of integrity of the primary containment, main steam, feedwater, or any other loop of the RCS. In addition, a postulated missile from any other system will not cause a loss of integrity of the primary containment or RCS pressure boundary. Based on these measures as described in the FSAR, the NRC staff concludes that the potential for MCL, SL, and MSL failure due to missiles is negligible.

Flood protection and analysis are provided in FSAR Tier 2, Section 3.4, "Water Level (Flood) Design." Inside of containment, safety-related systems and components are located above the maximum water level, protecting them from the effects of flooding. Water level instrumentation and other leak-detection systems detect pipe ruptures that could result in internal flooding. These leak-detection systems provide a signal to automatically isolate the affected system or to provide indication to the main control room to initiate operator action. The Nuclear Island drain and vent system prevents backflow of water from affected areas of the plant that contain safety-related equipment. As a result, the NRC staff concludes that the potential for MCL, SL, and MSL pipe rupture due to flooding is negligible.

Fire prevention and protection is described in FSAR Tier 2, Section 9.5.1, "Fire Protection System." The fire protection system (FPS) and its design meet all applicable codes and standards. The FPS is designed in accordance with 10 CFR 50.48, "Fire Protection," 10 CFR Part 50, Appendix A, GDC 3, " Fire Protection," GDC 5, "Sharing of Structures, Systems, and Components," GDC 19, "Control Room," GDC 23, "Protection System Failure Modes," GDC 56, "Primary Containment Isolation," SRP Section 9.5.1, "Fire Protection Program," and RG 1.189, "Fire Protection for Nuclear Power Plants." Because the FPS analysis (FSAR Tier 2, Appendix 9A) evaluated and rated the fire protection that will be provided for systems and plant areas as very high, the NRC staff concludes that the probability of MCL, SL, and MSL pipe rupture due to fire is extremely low.

Protection from over-pressurization of the RCS is described in FSAR Tier 2, Section 5.2.2, “Overpressure Protection,” and overpressure protection for the MSL is described in FSAR Tier 2, Section 10.1, “Summary Description.” Overpressure protection for both the RCS and MSL complies with the rules of ASME Code, Section III. In the case of the RCS, protection is provided by the pressurizer safety relief valves. Each MSL is protected by two safety valves and, in the event of loss of load and/or turbine trip, it is also protected by the main steam relief trains. Therefore, the staff concludes that pipe rupture due to over-pressurization is negligible for the MCL, SL, and MSL.

Damage from moving equipment is referred to in FSAR Tier 2, Chapter 15, “Transient and Accident Analysis.” The probability of load drops is very low inside of containment because of the imposition of administrative controls, the design of handling devices, and a prohibition against moving heavy loads inside containment while at power. Therefore, the staff concludes that MCL, SL, and MSL piping failures due to load drops are negligible.

Seismic classification and associated design bases are addressed in FSAR Tier 2, Section 3.2, “Classification of Structures, Systems, and Components.” The MCL, SL, and MSL are classified as Seismic Category I and, as such, are designed to be capable of performing their safety functions during and following a safe-shutdown earthquake. To perform their safety function requires that pressure-boundary integrity must be maintained. These systems are required to meet all applicable codes, standards, and regulatory requirements related to seismic design. The staff concludes that failure of the MCL, SL, and MSL piping due to a design-basis seismic event is highly unlikely because of their meeting these codes, standards and requirements.

Cleavage Type Failures

Regarding cleavage fracture for the MSL materials, the applicant provided references that showed that the J-R curves for SA 106 Grade C material at room temperature are higher than those at operating temperatures; therefore, there is no potential for cleavage fracture for that material. Also, Charpy impact energy values for SA 106 Grade C support this determination. In RAI 48, Question 03.06.03-6, the staff requested that the applicant address the effect of thermal aging on the cleavage behavior of the pump housing nozzle connection. As detailed above under the topic of Thermal Aging in a September 18, 2008, response to RAI 48, Question 03.06.03-6, the applicant addressed and resolved this issue. Therefore, the staff considers RAI 48, Question 03.06.03-6 resolved.

3.6.3.4.2 Failure Prevention and Detection

Snubber Reliability

The staff reviewed the reliability of snubbers using the guidelines in SRP 3.6.3 to ensure that the likelihood of a snubber failure will not invalidate the stresses in the piping that are used in the fracture mechanics analyses.

The design bases for snubbers are specified in FSAR Tier 2, Section 5.4.14.1, “Design Bases.” They are designed in accordance with the ASME Code, Section III, Subsection NF and in accordance with the guidance of RG 1.124, “Service Limits and Loading Combinations for Class 1 Linear-Type Supports.” Snubbers also meet the requirements of the ASME Operating and Maintenance (OM) Code, 2004 Edition. The U.S. EPR design incorporates provisions that allow ready access for maintenance, inspection, and testing of components. Snubber preservice testing (PST) and inservice testing (IST) are performed in accordance with the ASME OM Code, Subsection ISTD. The overall PST and IST intervals are as defined in Subsection ISTA of the ASME OM Code. The service life of snubbers is evaluated at least once per fuel cycle and increased or decreased as warranted. The requirements for PST and IST as stated in Subsection ISTA of the ASMEOM Code provide reasonable assurance that snubber failure rates are kept acceptably low during plant operation.

Inservice Inspection

ASME Code, Section III and ASME Code, Section XI preservice and inservice inspection requirements provide for the integrity of ASME Code Class 1 systems including the MCL and SL piping, as well as ASME Code Class 2 systems which include the MSL piping. In Sections 5.2.4 ad 6.6 of this report, the staff discusses that the design meets the accessibility requirements for inspection in accordance with ASME Code, Section XI, Subarticle IWA-1500 and 10 CFR 50.55a(g)(3)(i). In addition to provide continued assurance of structural integrity, the welds in the MSL will be subject to augmented inservice inspection in accordance with the requirements of Article IWC-2000 for Examination Category C-F and as specified in SRP Section 6.6, 7B, and D. Therefore, the staff concludes that the design of the welds in the MCL, SL and MSL enable the inservice inspection requirements are met.

3.6.3.4.3 Inputs for Leak-Before-Break Analyses (FSAR Tier 2, Section 3.6.3.4)

Geometry and Operating Condition (FSAR Tier 2, Section 3.6.3.4.1)

FSAR Tier 2, Section 3.6.3.4.1, "Geometry and Operating Condition," details the geometry and operating conditions for each of the three LBB systems considered. Plan, elevation and isometric views of the MCL and SL and an isometric view of the MSL are provided that identify both shop and field welds in each of these lines. FSAR Tier 2, Tables 3.6.3-1, "Main Coolant System Piping Dimensions and Operating Condition," 3.6.3-2, "Surge Line Piping Dimensions and Operating Condition," and 3.6.3-3, "Main Steam Line Dimensions and Operating Condition," also provide the operating conditions for the MCL, SL, and MSL, respectively. For the LBB evaluation, the plant is assumed to be operating under normal full power conditions with a postulated flaw size that produces 10 times the overall leak detection capability of a given piping system. FSAR Tier 2, Figures 3.6.3-1, "Plain View of U.S. EPR RCS Primary Piping," through 3.6.3-4, "Isometric View of the Main Steam Line," show the plan, elevation, and isometric views of the MCL, SL, and the MSL, respectively. In a September 18, 2008, response to RAI 48, Question 03.06.03-9, the applicant proposed to revise FSAR Tier 2, Figure 3.6.3-3, "Plan, Elevation, and Isometric View of the U.S. EPR Surge Line," to show that the surge line dissimilar metal weld is a shop weld and not a field weld. The above approach adequately addresses the review guidance provided in SRP Section 3.6.3. **RAI 48, Question 03.06.03-9 is being tracked as a confirmatory item**.

Materials (FSAR Tier 2, Section 3.6.3.4.2)

Main Coolant Loop and Surge Line: The MCL and SL piping consist of SA-336 F304LN or SA-182 F304LN austenitic stainless steel. The RCP casings are the only cast stainless product form within the MCL and are made of SA-351 CF-3. The stainless steel pipe welds are fabricated with dual-certified ER308/308L using the narrow-groove GTAW welding process. The safe end forging material is SA-182 F316LN or SA-336 F316LN. The dissimilar metal weld joints between the safe ends and the respective component nozzles of the pressurizer surge nozzle, the steam generator (SG) nozzles, and RPV nozzles are fabricated using Ni-Cr-Fe alloy filler metal Alloy 52/52M (ERNiCrFe-7/ERNiCrFe-7A, respectively). The pressurizer surge nozzle (forging) material and the steam generator inlet and outlet nozzle (forging) material are SA-508 Grade 3 ASME Code Class 2, and the RPV inlet and outlet nozzle material is SA-508 Grade 3 ASME Code Class 1.

Main Steam Line: The MSL piping is made of SA 106 Grade C carbon steel material.

The above approach as previously discussed in 3.6.3.4.1 of the SER adequately addresses the material specification guidance of III.11 in SRP Section 3.6.3.

Material Properties (FSAR Tier 2, Section 3.6.3.4.3)

FSAR Tier 2, Section 3.6.3.4.3 presents material properties for base and weld metals used in the MCL, SL, and MSL lines including: Dissimilar welds; primary component nozzles; RCP casing nozzles; and surge nozzles.

The analytical model used for tensile properties is based on a Ramberg-Osgood approach, and the fracture resistance properties and J-R curves are fit to a standard power law equation. The staff finds both these analytical models acceptable because they are generally employed in such analyses and are technically defensible.

FSAR Tier 2, Tables 3.6.3-4, “Tensile Properties for the Main Coolant Loop Piping,” 3.6.3-5, “Tensile Properties for the Surge Line Piping,” and 3.6.3-7, “Tensile Properties for the Main Steam Line Piping,” give the tensile properties for the MCL, SL, and MSL lines, respectively. In these tables, the yield and ultimate strength, flow stress, Young’s modulus, and Ramberg-Osgood parameters are presented. However, only one value is shown for the variety of materials in each line. It was not known if the material properties shown are properties obtained from ASME Code or from some other source. Therefore, in RAI 48, Question 03.06.03-5, the staff requested that the applicant identify the source of the material properties and asked if they were the lower bound or average tensile properties used for stability/leak-rate analyses. In a September 18, 2008, response to RAI 48, Question 03.06.03-5, the applicant resolved this issue by clarifying that for the MCL and SL the ASME Code minimum properties were used for modulus, yield and ultimate strength, while the Ramberg-Osgood parameters were obtained by an appropriate fit of the true-stress true-strain curves.

A description of the toughness properties for each material is given in the appropriate section of the FSAR.

Main Coolant Loop: For the main coolant loop piping weld and base metal toughness, the FSAR states that lower-bound toughness properties are used and that thermal aging of the base metal is negligible, with which statement the staff agrees. For the dissimilar-metal weld between the component nozzle and the MCL piping, which is an Alloy 52 weld, the FSAR states the toughness properties are developed from test data taken from pre-cracked, fatigue specimens at the weld fusion line. The use of test data is acceptable per the guidance in SRP Section 3.6.3, but the details of the toughness data were not included. During a June 26, 2008, audit at the applicant’s Rockville, MD offices, the applicant provided additional details of the toughness properties (reference letters from AREVA July 2, 2008, and December 18, 2008).

For the primary component nozzle of the MCL, lower-bound J-R curves are used that include the effect of thermal aging. These properties were taken from available public information. For the SA-508 Grade 3 Class 2 materials, a correlation between Charpy energy and elastic-plastic fracture toughness J_{Ic} was used. In a September 18, 2008, response to RAI 48, Question 03.06.03-6, the applicant provided additional details about this correlation.

The RCP casings are fabricated from cast stainless steel. The lower-bound toughness properties, which include aging effects, are taken from NUREG/CR-6177, "Assessment of Embrittlement of Cast Stainless Steels," and are acceptable to the staff for this purpose.

Surge Line: The toughness used in the surge line analyses was taken from a test program described in the FSAR. The lower-bound J-R curves were used in all cases. Thermal aging was taken into account for the weld metal used. However, the details of the J-R curves used were not included in the FSAR, but were provided by the applicant as proprietary data during the June 26, 2008, audit and also in a September 13, 2008, response to RAI 48, Question 03.06.03-6.

For the surge nozzle, SA-508 Grade 3, ASME Code Class 2 lower-bound material properties were used. Effects of thermal aging were taken into account in the J-R curve. However, the J-R curve used in the analyses was not included in the FSAR, but the applicant provided this as described above. The FSAR states that SA106 Grade C properties were used for the surge nozzle tensile properties, since these were not readily available for SA-508. This is an acceptable replacement. However, the effects of dynamic strain aging (DSA) for these carbon-steel materials at high loading rates needed to be addressed by the applicant. Therefore, in RAI 48, Question 03.06.03-7, the staff requested that the applicant address the effects of DSA on the material toughness for this material and explain how the negative effects of DSA have been accounted for in the MSL design.

In a September 18, 2008, response to RAI 48, Question 03.06.03-7, the applicant cited experimental results from work done in Korea to support the fact that DSA of carbon steels is not an issue of concern. At a June 9, 2009, audit, the staff reviewed information and references related to DSA and provided recommendations regarding metallurgical and heat treatment specifications and improvements to production weld procedures to minimize DSA concerns. In a June 3, 2009, response to RAI 265, Question 03.06.03-20, which was follow up to RAI 48, Question 03.06.03-7, the applicant adopted the staff's recommendations by confirming that (i) the residual stresses will be removed during the post fabrication stress relief heat treatment or post-weld heat treatment, (ii) the composition of SA106 Grade C will be selected to minimize susceptibility to DSA, and (iii) radiography will be performed in accordance with ASME Code, Section III to detect and eliminate sharp notches and discontinuities so as to minimize the potential for DSA.

Main Steam Line: For the material of the main steam line, the A106 Grade C data was developed from a test program described in the FSAR. The test program included base metals as well as shielded metal arc welding (SMAW) and submerged arc welding (SAW) welds. In all cases, lower-bound material properties were used in the analyses. However, it is unclear whether the effects of dynamic strain aging were accounted for in both the tensile and fracture properties. For instance, ferritic steels that are susceptible to dynamic strain aging may have toughness values at seismic loading rates that are half the value at quasi-static loading rates. In RAI 48, Question 03.06.03-7, the staff requested that the applicant address the effect of dynamic strain aging on fracture properties. The applicant's response identified that the testing methodology was the same for the MSL as for the SL which was discussed above. The staff considers the applicant's approach acceptable.

General Methodology

In FSAR Tier 2, Section 3.6.3.5, "General Methodology," the applicant provided details of both the leak-rate computation methods as well as its fracture mechanics analysis for assessing flaw stability as outlined in SRP Section 3.6.3. The premise of the LBB concept in piping is that a flaw will be detected via loss of fluid prior to the failure of the pipe. Based on the leak rate and crack morphology assumptions, a through-wall flaw is sized using specialized fluid-mechanics-based software. In SRP 3.6.3 two types of analyses are required; one in which the minimum load at normal operating conditions that leads to a detectable leak rate is calculated (with a safety factor on the leak rate), and another which calculates the maximum allowable load in the flawed pipe (with a safety factor on the leakage crack length).

In FSAR Tier 2, Section 3.6.3, the applicant states that air-fatigue crack morphology will be used in all leak-rate calculations using the KRAKFLO computer code for the two-phase flow problems and SQUIRT 1.1 for the steam space problem. The use of these codes is acceptable for calculating the leak flow rate, but using an air-fatigue crack morphology parameters with no turns in the flow path will be nonconservative for any crack. The basis for using air-fatigue morphology with no turns is needed. For instance, NUREG/CR-6765, "Development of Technical Basis for Leak-Before-Break Evaluation Procedures," recommends using a corrosion-fatigue crack morphology (SCC is eliminated in the screening criteria) in the leak-rate calculations, since cracks would more realistically initiate on the wetted surface (inside diameter (ID)) of the pipe. Also, analyses of air-fatigue cracks used in leak-rate tests had to use some number of turns in the flow path to get reasonable agreement between the test data and predicted leak-rates. The additional information provided by the applicant in their January 23, 2009, letter and the subsequent confirmatory calculations have determined that the number of turns used in the leak rate calculations are most critical in the prediction of the moment versus crack length curves. In RAI 48, Question 03.06.03-10, the staff requested that the applicant provide the crack morphology parameters used in its leak rate calculations.

In a November 7, 2008, response to RAI 48, Question 03.06.03-10, the applicant provided justification for the air-fatigue crack morphology parameters used in LBB evaluation procedures. The air-fatigue crack morphology parameters for LBB evaluation are consistent with those used in past LBB submittals that were approved by the staff. The staff finds that these air-fatigue crack morphology parameters are generally applicable to the U.S. EPR plant, and are therefore, acceptable. The staff and industry are continuing to verify the applicability of air-fatigue crack morphology parameters to newer weld and base materials in certain types of bi-metallic weld joints.

In a December 18, 2009, response to the follow-up RAI 265, Question 03.06.03-22, the applicant confirmed that both KRACKFLO and the NRC code SQUIRT yielded consistent leak-rate vs. moment curves for the same input variables and provided an adequate justification for the penalty factor of 26 used in the KRAKFLO algorithm.

Also, in a December 18, 2009, response to RAI 265, Questions 03.06.03-22 and 03.06.03-23, the applicant confirmed that the following actions would be taken to demonstrate that the piping is not susceptible to PWSCC where LBB is being applied:

- The weld material for the dissimilar metal welds will conform to an ASME specification for Alloy 52, 52M or 152 equivalents,
- For dissimilar metal welds the filler material and weld procedure will be selected such that there is at least 24 percent Cr in the initial layers,
- ASME Code Section IX weld qualification tests and procedures will be followed.

The staff finds these actions acceptable and, therefore, considers RAI 265, Questions 03.06.03-22 and 03.06.03-23 resolved.

Load Combination Methods

FSAR Tier 2, Section 3.6.3.5.1, "Load Combination Methods," presents the absolute Load Combination Methods per SRP Section 3.6.3 for LBB analysis including a factor of 1.4 times the normal plus SSE loads as specified.

The minimum moment for normal operating conditions corresponds to deadweight, steady-state pressure, and thermal expansion loading under normal operation. The maximum moment combines the minimum moment with the seismic and seismic anchor motions (or any other large transient loads such as start-up and shut-down thermal stresses in surge lines). The maximum allowable load must exceed the minimum load evaluated for leakage crack size, with applicable margins of safety on both flaw size and load.

For these cases, a bounding analysis in the form of an LBB allowable load window approach is used. This window is developed by use of the minimum and maximum moments described above. Once the window is generated, the actual piping loads can be plotted on the window to determine acceptability. The LBB safety factors, as described in SRP Section 3.6.3 are embedded in these windows. Even though this analysis method is not explicitly used in the SRP Section 3.6.3 procedure, it is consistent with the analyses methodology and is, therefore, considered acceptable.

Leak Rate Determination Method for MCL and SL

FSAR Tier 2, Section 3.6.3.5.2, "Leak Rate Determination Method for Main Coolant Loop and Surge Line," presents the method using the applicant's software code KRAKFLO. The KRAKFLO code was stated to be similar to the NRC code LKRATE, has been benchmarked against the leak-rate experiments conducted in EPRI Report NP-3395, "Calculation of Leak Rates Through Crack in Pipes and Tubes," December 1983, and is in good agreement with the experimental data. However, the calculation of crack-opening displacement (COD) is vital to proper leak-rate predictions. The FSAR did not provide details on how COD was calculated or if crack-face pressure was included in the analyses. The applicant provided additional information on the leak rate methodology and screen shots from their PICEP code calculations that were used to validate the KRAKFLO code in their January 23, 2009, letter. As mentioned above, in RAI 48, Question 03.06.03-10, the staff requested that the applicant provide the crack morphology parameters used in its leak rate calculations. As detailed in the General Methodology section above, the issue of crack morphology was adequately addressed by the applicant in a December 18, 2009, response to RAI 265, Question 03.06.03-22.

For MCL, the locations evaluated are as follows:

- Reactor pressure vessel outlet nozzle region at the hot leg
- SG inlet nozzle region at the hot leg
- SG outlet nozzle region
- Crossover leg, RCP outlet nozzle region, cold-leg pipe, and RPV inlet
- RCP inlet nozzle region

For SL, the locations evaluated were:

- Pressurizer surge nozzle end of the SL
- Hot-leg nozzle end of the SL

A leak rate of 18.9 lpm (5 gpm) (including the safety factor of 10 on the documented leak-detection system capability) was used in the calculations for the MCL and SL piping locations. Both axial and circumferential through-wall cracks are analyzed in straight piping. For the axial cracks, only pressure loading was assumed, while for the circumferential cracks, pressure and bending were considered. The staff found this acceptable since the applied loadings would be perpendicular to the crack and enables crack opening displacement.

The leakage crack size as a function of minimum moment for the MCL and SL is given in FSAR Tier 2, Figures 3.6.3-5, “Minimum Moment versus Circumferential Crack Leakage Sizes for 5 gpm at Various Main Coolant Loop Locations,” and 3.6.3-6, “Minimum Moment versus Circumferential Crack Leakage Sizes for 5 gpm at Two Main Coolant Loop Locations,” respectively. The staff finds the calculated values of crack length versus minimum moment for the MCL and SL acceptable as discussed above.

Leak Rate Determination Method for MSL

FSAR Tier 2, Section 3.6.3.5.3 presents the use of SQUIRT Version 1.0 instead of KRAKFLO for calculating leak rates for the MSL. The SQUIRT code has been developed by the NRC and has been rigorously validated with experiments. Therefore, the use of this code for calculating the leak rate is acceptable. For these analyses, a leak rate of 3.8 lpm (1 gpm) (10 times the documented leak detection system capability) was used in the calculations for MSL. Both axial and circumferential through-wall cracks are analyzed in straight piping. As with the MCL, pressure only was assumed for the axial crack cases, while bending and pressure were assumed for the circumferential crack cases. As illustrated in FSAR Tier 2, Figure 3.6.3-7, “Pressure Only Leakage Rate versus Crack Length for both Axial and Circumferential Crack Morphologies,” when only pressure loads are assumed the circumferential cracks had larger crack sizes for the same leakage rate. Therefore, the circumferential crack with pressure was conservatively used to analyze the axial crack leakage, and the staff finds this acceptable. For circumferential cracks, the external axial loads were conservatively set to zero for the calculation of leakage crack size, because external axial loads will decrease the crack size for the same leak rate.

The FSAR uses leakage crack sizes from circumferential through-wall-cracked straight pipe for the analysis of elbows with an extrados crack. The justification for this assumption had not been adequately addressed by the applicant. Therefore, in RAI 48, Question 03.06.03-18, the staff requested that the applicant justify this assumption. In a September 18, 2008, response to RAI 48, Question 03.06.03-18, the applicant provided a reference to NUREG/CR-6837, “The Battelle Integrity of Nuclear Piping (BINP) Program Final Report,” that provides a basis for using straight crack solutions for elbows since the crack driving force for the two cases have similar values. Therefore, the staff considers RAI 48, Question 03.06.03-18 resolved since the applicant appropriately applied the NUREG.

Flaw Stability and J-T Stability Analysis Methods

FSAR Tier 2, Sections 3.6.3.5.4, “Flaw Stability Analysis Method,” and 3.6.3.5.5, “J-T Stability Analysis Procedure,” summarize the flaw stability analysis method and the J-T analysis based on tearing instability theory for flawed piping. The J-T methodology for predicting critical crack size and crack stability has been shown to be accurate as long as the correct crack-driving force and material resistance are supplied.

The following cases are analyzed for each of the three LBB systems:

- Circumferential through-wall crack in a straight pipe
- Axial through-wall crack in straight pipe
- Circumferential through-wall extrados crack in an elbow

For the circumferential through-wall crack in a straight pipe, the EPRI/GE solution in EPRI NP-5596, “Elastic-plastic Fracture Analysis of Through-wall and Surface Flaws in Cylinders,” is used with modifications. For an axial through-wall crack, the ln-sec solution in the Ductile Fracture Handbook is used. For an extrados crack, the criteria in NUREG/CR-6837, “The Battelle Integrity of Nuclear Piping (BINP) Program Final Report,” are used. The staff reviewed the modifications to EPRI NP-5596 and found them to be acceptable.

Circumferential Through-Wall Crack in Straight Pipe

Main Coolant Loop and Main Steam Line Piping

For these lines, the GE/EPRI pressure and bend solution (with plastic-zone correction) was used to calculate the crack-driving force. This solution has been published in many references and was determined through NRC research (NUREG/CR-6540, BMI-2196, "Comparison of crack-opening displacement predictions for LBB applications," February 1998) to be conservative in predicting the maximum-load-carrying capacity of circumferential-cracked straight pipe. In the analyses presented, the solution for an R/t=10 was used conservatively, since the h-functions were only valid for R/t between 10 and 20. Consequently, the staff finds this procedure acceptable for calculating the crack-driving force.

Surge-Line Piping

For the surge-line piping, a bending only GE/EPRI solution was used. Even though the solution presented above would have been adequate for the surge line, a different solution was used. The rationale was that the surge line piping had an R/t=5 and the solutions for both the MCL and MSL lines were not applicable. However, as stated above, the solution for R/t=10 can be used conservatively. Therefore, in RAI 48, Question 03.06.03-12, the staff requested that the applicant provide additional information regarding its flaw stability for the SL. In a November 7, 2008, response to RAI 48, Question 03.06.03-12, the applicant proposed revisions to FSAR Tier 2, Section 3.6.3.5.4. The proposed revisions which identified the Ramberg-Osgood constant "n" values used by the applicant for the R/t solution are acceptable to the staff. **RAI 48, Question 03.06.03-12 is being tracked as a confirmatory item**.

Since the surge line is realistically a pressure and bending case and the solution chosen is a bending-only solution, the pressure term was handled as an equivalent bending moment. This substitution was achieved by equating the axial and bending stress intensity factors, and backing out an equivalent bending moment for the applied pressure. This equivalent bending moment was then added to the applied bending moment and used in the bending-only solution. However, the limit-load solution and the h-functions were calibrated for bending loads only. For the limit condition, the addition of the axial loads will shift the neutral axis and lower the collapse moment. Whether the equivalent moment calculated by equating the stress intensity factors is fully capturing the shift of the neutral axis due to the axial loads is unknown.

Therefore, extensive confirmatory analysis was conducted by the staff and its consultants (1) using leak rate computations to obtain the moment versus leak rate curves, (2) finite-element analysis to verify the crack driving force, and (3) J-estimation scheme analysis for flaw stability evaluation and development of the Allowable Load Limit (ALL) diagrams as presented by the application. In a December 18, 2009, response to RAI 265, Question 03.06.03-26, the applicant re-did their flaw stability analysis and showed that their results matched those conducted by the staff's confirmatory LBB analysis using an air-fatigue morphology for the crack which the staff finds acceptable as detailed below.

Even though the design conditions for the SL falls in-between the air-fatigue and corrosion/thermal fatigue crack morphology curves, it is important to note that there are inherent safety margins in all LBB analyses.

LBB is a multi-step process in the performance of a deterministic flaw tolerance evaluation. This process is designed to be conservative via the use of specified margins and input assumptions meant to cover numerous uncertainties. For example, there is a safety margin of 10 on leak rate and an additional safety margin of 2 on crack size. The current LBB process has been demonstrated to be conservative by results from comprehensive research programs such as IPIRG (International Piping Integrity Research Group). Other related research programs confirm and extend the IPIRG results. If any one LBB margin is not met or if any input assumption or value is deemed to be non-conservative it does not necessarily imply that the overall results from the LBB evaluation will be significantly affected. It does require, however, that the net effect on those results be evaluated.

In a December 18, 2009, response to RAI 265, Question 03.06.03-26, the applicant provided the Finite element analysis between the two materials that would be used in the surge line in the EPR design. The staff analyzed the ALL diagram for the case of a Pressurizer Surge Nozzle crack which showed that the design conditions fall between the air-fatigue crack morphology and corrosion fatigue crack morphology curves. The ratio of the maximum moment for the design conditions to that for the air-fatigue crack morphology curve in the ALL diagram yields a safety margin of approximately 1.08, and approximately 0.89 for the corrosion fatigue assumption. Given the above considerations and overall safety margins in LBB analyses the staff concludes that the approximate 11 percent reduction in margin for the corrosion fatigue assumption is acceptable since the evaluation procedure also has an assumption of a safety margin of 10 (or 900 percent) on leak rate and 2 (or 100 percent) on the crack size.

The staff performed a confirmatory analysis using both the AREVA codes and the NRC codes and were able to validate the solutions obtained. Based on the analysis, the staff considers **RAI 265, Question 03.06.03-26 resolved.**

Based on the analysis, the staff requested that the applicant revise the remaining ALL diagrams for Main Steam Line (MSL),Main Coolant Line (MCL) and Surge Line (SL) cases and provide the staff with the revised versions of the FSAR Tier 2, Figures 3.6.3-12, 3.6.3-13, 3.6.3-14, 3.6.3-15, 3.6.3-16, 3.6.3-17, 3.6.3-21, 3.6.3-22, and 3.6.3-23. **RAI 467, Question 03.06.03-28 is being tracked as an open item.**

Axial Through-Wall Crack in Straight Pipe:

For all cases, the In-sec equation presented in the Ductile Fracture Handbook was used to predict the crack driving force for an axial crack in straight pipe. A higher order bulging factor was used. This method is accurate and is therefore acceptable.

Circumferential Through-Wall Extrados Crack in an Elbow:

The crack-driving force solution developed in the NRC BINP program (NUREG/CR-6837) was used for a circumferential through-wall extrados crack in an elbow. This solution is only valid for cracks with half angles of 45 and 90 degrees. These analyses will determine if the straight-pipe analysis is a conservative representation of the elbow solution. The resolution of this issue is discussed above under Leak rate Determination Method for the MSL.

As noted earlier, the J-T methodology for predicting the stability of cracked pipe has been shown to be accurate as long as the crack-driving force is accurately captured. The staff finds the procedures presented in the FSAR acceptable as long as the responses to the RAIs given in this section are satisfactory. However, in some analyses, applicants may not take full credit for displacement-controlled loading in critical crack size predictions. The claim is relaxation of a portion of the displacement-controlled loads may occur depending on the system compliance and the crack length. Acceptable, conservative analyses use 100 percent of the displacement-controlled loading in the stability calculations. As discussed above for the surge line case, the applicant will revise all ALL diagrams per the modifications to the flaw stability analysis approach as recommended by the staff.

As discussed above for the remaining ALL diagrams for Main Steam Line (MSL), Main Coolant Line (MCL) and Surge Line (SL) cases, the applicant will revise FSAR Tier 2, Figures 3.6.3-12, 3.6.3-13, 3.6.3-14, 3.6.3-15, 3.6.3-16, 3.6.3-17, 3.6.3-21, 3.6.3-22, and 3.6.3-23. **RAI 467, Question 03.06.03-28 is being tracked as an open item.**

Maximum Allowable Piping Moment

FSAR Tier 2, Section 3.6.3.5.6, "Determination of Maximum Allowable Piping Moment," describes the determination of maximum allowable piping moment with a factor of safety of 2 on the leakage crack size as recommended by SRP Section 3.6.3. Tables for critical locations for which flaw stability analyses were carried out are provided in FSAR Tier 2, Section 3.6.3.5.7, "Identification of Locations for Flaw Stability Analysis," for all three LBB systems.

For all three LBB systems, the ALL diagrams were developed using flaw stability analyses. This ALL diagram is a plot of the minimum versus the maximum moment for a given axial load and defines both the "No Leak Zone" and the "ALL LBB Zone." The presence of a 45 degree line on these plots indicates that the maximum moment will never be below the minimum moment. The underlying deterministic fracture-mechanics-based approach used to develop the ALL diagram is consistent with the recommendations of the SRP methodology and is, therefore, acceptable to the staff.

3.6.3.4.4 LBB Results (FSAR Tier 2, Section 3.6.3.6)

As indicated above, the results of the LBB evaluation are presented as ALL diagrams that graph the maximum moment versus minimum moment. Each of these curves is developed for a specific value of axial loads and separates the ALL diagram into "No Leak Zone" and the "ALL LBB Zone." If a specific analysis case for a given size flaw and leak-rate falls *above* the ALL LBB Zone on the diagram, then that case is deemed nonconservative and further evaluation may be necessary to validate LBB.

MCL Piping (FSAR Tier 2, Section 3.6.3.6.1)

Circumferential Through-Wall Crack in Straight Pipe:

For this flaw type, ALL diagrams were developed for the following locations:

- Steam generator inlet nozzle at the Alloy 52 weld
- Steam generator outlet nozzle at the Alloy 52 weld
- RCP outlet nozzle region (includes cold-leg pipe and RV inlet nozzle)

- RCP inlet nozzle region

- The hot leg/crossover leg piping region

In each of the cases above that contain Alloy 52 welds, the lower-bound fusion-line toughness and the stainless steel base metal tensile properties were used in the analyses. For the RCP outlet (cold-leg pipe, reactor vessel (RV) inlet nozzle) and inlet nozzle region, the cast austenitic stainless steel (CASS) RCP casing lower-bound toughness properties were conservatively used. Finally, for the hot-leg/cross-over piping, the tensile and toughness properties of the base metal were used.

Circumferential Through-Wall Extrados Crack in an Elbow:

An ALL diagram analysis was performed to demonstrate that the case above for a straight pipe provides a "bounding" analysis for a through-wall crack in the extrados of an elbow. A steam generator inlet elbow was analyzed and compared to the adjoining straight pipe. It was shown that the load-carrying capacity for the elbow was 10 percent higher than the straight pipe with the same crack size. Therefore, the applicant's use of a straight-pipe solution for bounding the elbow-crack case needed justification. The justification and resolution of this issue is discussed above under Leak rate Determination Method for MSL.

Axial Through-Wall in Straight Pipe:

The same locations as those for the circumferential crack above were also considered. Using the ALL analysis method the applicant showed that the minimum critical crack size was greater than 0.432 m (17 in.) which leads to a margin of 6.25 on the leakage crack size. The staff finds this margin meets the guidelines of SRP Section 3.6.3, and is therefore acceptable.

Surge Line Piping (FSAR Tier 2, Section 3.6.3.6.2)

Circumferential Through-Wall Crack in Straight Pipe:

For this flaw type, the following locations were analyzed:

- Pressurizer surge nozzle at the Alloy 52 weld fusion line

- Surge-line piping

- Hot-leg nozzle

For the locations where Alloy 52 weld metal exists, the lower-bound fusion-line toughness was used in the analyses. For the piping, the base-metal tensile and toughness material properties were used and are found to be acceptable, as these are consistent with the recommendations in SRP Section 3.6.3.

Circumferential Through-Wall Extrados Crack in an Elbow:

As with the MCL piping, a sample problem was analyzed to justify the use of the straight pipe ALL diagram as a “bounding” case for the crack in the elbow extrados. In this case, the maximum moment for the elbow (which had a smaller wall thickness) was similar to the moment in a straight pipe with the same crack size. However, NUREG/CR-6444, “Methodology for Analyzing Precursors to Earthquake-Initiated and Fire-Initiated Accident Sequences,” showed that a multiplier on the moment was needed to use the straight pipe solution, where that multiplier was a function of the elbow B2 stress indices. Therefore, the appropriateness of using a straight-pipe solution to characterize the elbow problem needed to be justified. The resolution of this issue is discussed above under Leak Rate Determination Method for MSL.

Axial Through-Wall Crack in Straight Pipe:

The minimum safety factor for the SL locations was 4.2, which again satisfies the SRP Section 3.6.3 recommendations, and is therefore acceptable to the staff.

Main Steam Line Piping (FSAR Tier 2, Section 3.6.3.6.3)

Circumferential Through-Wall Crack in Straight Pipe:

For these analyses, both the weld and base metal material properties were used to demonstrate that the base metal was the most limiting material for the MSL piping. A large percentage of carbon steels is susceptible to dynamic strain aging, which causes a reduction in toughness at seismic loading rates. The applicant needed to address the effect of dynamic strain aging on material toughness properties. The applicant’s response to this issue is covered above in the above topic involving Surge Line.

For these analyses, the ALL diagrams were developed with a safety factor of 2 and 1.7 on leakage flaw size, even though SRP Section 3.6.3 explicitly recommends a safety factor of 2. In RAI 48, Question 03.06.03-14, the staff requested that the applicant justify its use of a safety factor of 1.7.

During both the June 26, 2008, audit meeting and in a September 19, 2008, response to RAI 48, Question 03.06.03-14, the applicant indicated that 1.7 was the highest safety factor attainable considering a generic 0.3g zero period acceleration (ZPA) safe-shutdown earthquake for the U.S. EPR. Also, the loading analyses for the MSL bounds 12 different soil conditions ranging from the softest to the hardest profile. In follow-up RAI 265, Question 03.06.03-24, the staff requested the applicant check the stress-strain properties used in their analysis to insure that the crack driving force J was modeled correctly.

In an October 16, 2009, response to RAI 265, Question 03.06.03-24, the applicant indicated that using the modal combinations recommended in RG 1.92, Revision 2 was more accurate as compared to RG 1.92, Revision 1. The analysis using the modal combinations identified in Revision 2 to RG 1.92 is more accurate and provides a higher safety factor of 2.0. Therefore, the analysis using Revision 2 to RG 1.92 is more conservative than Revision 1 to RG 1.92. The technical staff considers this response to be acceptable. Furthermore, in a March 2, 2010, response to follow-up RAI 338, Question 03.06.03-27, the applicant clarified that the support configuration has also been modified to facilitate load reduction and stress qualification. The staff acknowledges that the modal combination methods provided in RG 1.92, Revision 2 reduce excessive conservatism from the modal combination methods provided in RG 1.92, Revision 1. In addition to using RG 1.92, Revision 2 to perform piping load reduction, the applicant's modified support configuration also provides further reduction in piping loads. On the basis that the piping load is reduced by using RG 1.92, Revision 2 and the modified support configuration, the staff concludes that it is reasonable that the piping SSE loads will be reduced to ensure a margin of 2.0 between the leakage crack size and critical crack size. Therefore, the staff considers RAI 338, Question 03.06.03-27 resolved.

Circumferential Through-Wall Extrados Crack in an Elbow:

Similar to the analyses for the MCL and SL, the case of a straight pipe with a circumferential through-wall crack was used to demonstrate that it bounds the results of a similar crack in the extrados of elbows. As illustrated in FSAR Tier 2, Figure 3.6.3-22, "ALL for Main Steam Line Piping with Safety Factor of 2 on Flaw Size (Base Metal)," the elbow ALL curve is above the curves for the straight pipe cases, indicating that there is a greater safety margin for the elbow than for the straight pipe. The applicant needed to justify its use of the straight-pipe solution for bounding the elbow results. The resolution of this issue is discussed above under Leak rate Determination Method for MSLs.

Axial Through-Wall Crack in Straight Pipe

The hoop stresses due to operating pressure are the main crack-driving force on the axial through-wall crack in a straight pipe, while the effects of external bending loads are not considered significant for this crack orientation. Therefore, no allowable load-limit diagram is generated for the axial through-wall crack in a straight pipe. However, the margin was calculated by the applicant as 2.8 on crack size using the lower-bound base metal material properties; the staff finds that this complies with SRP Section 3.6.3.

3.6.3.4.5 Leak Detection (FSAR Tier 2, Section 3.6.3.7)

FSAR Tier 2, Section 3.6.3.7 provides a brief summary of leak-detection systems that should adhere to the recommendations of RG 1.45, Revision 0, "Guidance on Monitoring and Responding to Reactor Coolant System Leakage." For MCL and SL, the leak rates of 18.9 lpm (5 gpm) require a leak-detection system of 1.89 lpm (0.5 gpm). For MSL, a leak-rate assumption of 3.8 lpm (1 gpm) requires a detection system of 0.38 lpm (0.1 gpm). More details about leak-detection systems are covered under SRP Section 5.2.5. Since leak detection is closely related to LBB evaluation procedures, in RAI 48, Question 03.06.03-16, the staff requested that the applicant provide more information about the adequacy of the leak detection systems especially under SSE loading. In a September 18, 2008, response to RAI 48, Question 03.06.03-16, the applicant confirmed that the leak detection systems will be in accordance with RG 1.45, Revision 1 with two independent diverse measurements. In a January 26, 2009, response to follow-up RAI 168, Question 03.06.03-19, the applicant provided information about a leak detection system for MSL capable of performing following a seismic event and has proposed changes to FSAR Tier 2, Section 3.6.3.7 and associated Tables reflecting this. The staff finds that the proposed changes to the FSAR adequately resolve this issue. **RAI 168, Question 03.06.03-19 is being tracked as a confirmatory item**.

3.6.3.4.6 Inspection, Tests, Analyses, and Acceptance Criteria

As stated in FSAR Tier 2, Section 3.6.3.2, "Methods and Criteria," the following ITAAC was presented in the FSAR Tier 1 portion of the application:

FSAR Tier 1, Table 2.2.1-5, Item 3.7 states that an analysis will be performed that assesses the LBB capability of the piping, interconnected component nozzles, and equipment given in Table 2.2.1-1. The ITAAC given are appropriate when plant-specific (e.g., as-built) information is used in the analyses.

3.6.3.4.7 COL Information Items

As stated in Section 3.6.3.5 of this report, FSAR Tier 2, Table 1.8-2, COL Information Item 3.6-3 exists for this analysis:

> A COL applicant that references the U.S. EPR design certification will confirm that the design LBB analysis remains bounding for each piping system and provide a summary of the results of the actual as-built plant specific LBB analysis, including material properties of piping and welds, stress analyses, leakage-detection capability, and degradation mechanisms.

This item is appropriate in that the COL applicant will need to verify that their plant-specific details (i.e., geometry, loads, etc.) can be shown to fall within the bounds of the ALL Diagrams to verify LBB applicability.

Combined License Information Items

Table 3.6.3-1 provides a list of LBB related COL information item numbers and descriptions from FSAR Tier 2, Table 1.8-2.

Table 3.6.3-1 U.S. EPR Combined License Information Items

Item No.	Description	FSAR Tier 2 Section
3.6-3	A COL applicant that references the U.S. EPR design certification will confirm that the design LBB analysis remains bounding for each piping system and provide a summary of the results of the actual as-built plant specific LBB analysis, including material properties of piping and welds, stress analyses, leakage detection capability, and degradation mechanisms.	3.6.3

The staff finds the above listing to be complete. Also, the list adequately describes actions necessary for the COL applicant or holder. No additional COL information items need to be included in FSAR Tier 2, Table 1.8-2 for LBB consideration.

3.6.3.5 *Conclusions*

As stated previously, 10 CFR Part 50, Appendix A, GDC 4 states that the dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is "extremely low" − defined as 1 x 10^{-6} per reactor year in the final rule on the modification to GDC 4 requirements (52 Federal Register (FR) 41288), October 27, 1987 − under conditions consistent with the design basis for the piping. Alternatively, a deterministic evaluation with verified design and fabrication, in addition to adequate inservice inspection, can meet the extremely low probability criterion. The deterministic evaluation is based on the requirement that structures and components are correctly engineered to meet the applicable regulations and NRC-endorsed industry codes.

This section presents the results of the staff's review of FSAR Tier 2, Section 3.6.3. The overall conclusion from this review is that, except for the open items presented in this document, the analyses and results are sufficient to demonstrate LBB for the systems selected and satisfy the requirements of GDC 4.

The staff evaluation concludes on a design-specific and piping system-specific basis that the acceptance criteria are satisfied, and, therefore, that dynamic effects of pipe rupture may be eliminated from design consideration. When dynamic effects of pipe rupture are eliminated, protective devices such as pipe whip restraints and jet impingement barriers are no longer needed. The staff determination is based on the following:

1. That water hammer, corrosion, creep, fatigue, erosion, environmental conditions, and indirect sources are extremely low causes of pipe rupture. That extremely low probability is based on deterministic evaluations that demonstrate that structures and components are correctly engineered to meet the applicable regulations and NRC-endorsed industry codes.

2. That a deterministic fracture mechanics evaluation has been completed and approved by the staff.

3. That leak detection systems are sufficiently reliable, redundant, diverse and sensitive, and that margin exists to detect the through-wall flaw used in the deterministic fracture mechanics evaluation.

3.9 Mechanical Systems and Components

3.9.1 Special Topics for Mechanical Components

3.9.1.1 *Introduction*

This section addresses the design transients and methods of analysis for Seismic Category I components, and supports, including both those designed as ASME B&PV Code, Section III, Division 1, Class 1, 2, 3, or core support and those not covered by the Code. Also included are the assumptions and procedures used for the inclusion of transients in the design and fatigue evaluations of ASME Code Class 1 and core support components, the computer programs used in the design and analyses of Seismic Category 1 components, and their supports, and experimental and inelastic analytical techniques.

In accordance with the guidelines in SRP Section 3.9.1, the staff reviewed the information in FSAR Tier 2, Revision 1, Section 3.9.1, "Special Topics for Mechanical Components," including:

- The design transients used in the design and fatigue evaluations for ASME Code Class 1 components, component supports, core support structures, and reactor internals
- The methods of analysis and computer programs used in the design and analysis for Seismic Category I components, component supports, core support (CS) structures, and reactor internals designated as ASME Code Class 1, 2, 3, and CS under ASME Code, Section III and those not covered by the ASME Code
- Experimental stress analysis techniques that may be used in lieu of theoretical stress analysis
- Elastic-plastic stress analysis methods which the applicant may elect to use in the design of the above-noted components

3.9.1.2 *Summary of Application*

The applicant has provided an FSAR Tier 2 system description in Section 3.9.1 summarized in part, as follows.

3.9.1.2.1 Design Transients

In FSAR Tier 2, Table 3.9.1-1, "Summary of Design Transients," the applicant provided the fluid system design transients for five operating conditions and the number of cycles for each transient considered in the design and fatigue analyses of reactor coolant system ASME Code Class 1 components, other ASME Code Class 1 components, RCS supports, and reactor internals. The operating conditions are as follows:

- ASME Service Level A—normal conditions
- ASME Service Level B—upset conditions, incidents of moderate frequency
- ASME Service Level C—emergency conditions, infrequent incidents
- ASME Service Level D—faulted conditions, low-probability postulated events
- Test Conditions

FSAR Tier 2, Section 3.9.1.1, "Design Transients," provides a list of the design transients and the number of cycles for the transients in FSAR Tier 2, Table 3.9.1-1. The transients are defined for equipment design purposes and are not intended to represent actual operating experience. The design transients define thermal-hydraulic conditions (i.e., pressure, temperature, and flow) for the reactor coolant pressure boundary. Bounding thermal-hydraulic design transients are defined for components of the RCPB and the secondary side pressure boundary (SSPB) with respect to their cyclic behavior and fatigue life.

The transients given in FSAR Tier 2, Table 3.9.1-1 are assumed for the design life of the plant. In accordance with the ASME Code, Section III, emergency and faulted conditions are not included in fatigue evaluations, with the exception that any significant emergency cycles in excess of 25 must be considered in the fatigue analyses. Significant emergency cycles are those that result in stresses higher than the endurance limits on the ASME design fatigue curves.

FSAR Tier 2, Table 3.9.3-1, "Load Combinations and Acceptance Criteria for ASME Code Class 1 Components," shows the load combinations and acceptance criteria for safety-related ASME Code Class 1, 2, and 3 components, component supports, and CS structures. U.S. EPR Topical Report ANP-10264NP-A, Table 3-1, "U.S. EPR Piping Analysis and Pipe Support Design Topical Report," Revision 0, September 2006, shows specific load combinations and acceptance criteria for ASME Code, Section III Class 1 piping systems and components. This Topical Report has been reviewed and approved by the staff on August 11, 2008, in a final safety evaluation report (FSER).

3.9.1.2.2 Computer Programs

The applicant used computer programs to analyze mechanical components. 10 CFR Part 50, Appendix B requires design control measures to verify the adequacy of the design of safety-related components. In SRP Section 3.9.1, the staff provides guidelines sufficient to meet 10 CFR Part 50, Appendix B. FSAR Tier 2, Section 3.9.1.2, "Computer Programs Used in Analysis," provides computer codes used in the structural dynamic and static analyses of mechanical loads, stresses, deformations, and in the thermal hydraulic transient loads analyses for Seismic Category I components and supports. This section also refers to FSAR Tier 2, Appendix 3C, "Reactor Coolant System Structural Analysis Methods," for a list of the computer programs used in the design of major safety-related components. The list includes programs to perform thermal hydraulic analyses to generate forcing functions, and the structural analyses subject to dynamic and static loads, resulting in stresses and deformations of Seismic Category I components and supports. In addition, the computer programs for the design and analyses of piping and supports are also discussed in Topical Report ANP-10264NP-A, Section 5.1.

3.9.1.2.3 Experimental Stress Analysis

FSAR Tier 2, Section 3.9.1.3, "Experimental Stress Analysis," indicates that experimental stress analysis is not used to evaluate stresses for Seismic Category I components and supports.

3.9.1.2.4 Considerations for the Evaluation of Faulted Conditions - Inelastic Analyses

In FSAR Tier 2, Section 3.9.1.4, "Considerations for the Evaluation of the Faulted Condition," the applicant referred to FSAR Tier 2, Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures," for the analytical methods used for Seismic Category I systems and components subjected to faulted conditions.

In FSAR Tier 2, Section 3.9.3, the applicant stated that all Seismic Category I equipment are evaluated for the faulted (ASME Code, Section III Service Level D) loading conditions identified in FSAR Tier 2, Table 3.9.3-1, "Load Combination and Acceptance Criteria for ASME Code Class 1 Component."

3.9.1.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 3.9.1, and are summarized below. Review interfaces with other SRP sections can also be found in NUREG-0800, Section 3.9.1.

1. GDC 2, as it relates to the design of mechanical components of systems being designed to withstand the effects of earthquakes without loss of capability to perform their safety function.

2. GDC 14, “Reactor Coolant Pressure Boundary,” as it relates to the design of the reactor coolant pressure boundary so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture.

3. GDC 15, “Reactor Coolant System Design,” as it relates to the design of mechanical components of the RCS with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences.

4. GDC 1 and 10 CFR 50.55a, as they relate to the design, fabrication, erection, construction, testing, and inspection of components important to safety in accordance with the requirements of applicable codes and standards commensurate with the importance of the safety function to be performed.

5. 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants,” as it relates to requirements that the components and component supports, and core support structures will be designed and built in accordance with the certified design.

6. 10 CFR Part 50, Appendix B, as it relates to design quality control.

7. 10 CFR Part 50, Appendix S, as it relates to the suitability of the plant design bases for mechanical components established in consideration of site seismic characteristics.

8. 10 CFR Part 100, Appendix A, as it relates to the suitability of the plant design bases for mechanical components established in consideration of site seismic characteristics.

Regulatory guidance provided to meet the above requirements includes:

1. RG 1.207, “Guidelines for Evaluating Fatigue Analyses Incorporating the Life Reduction of Metal Components Due to the Effects of the Light-Water Reactor Environment for New Reactors.”

2. RG 1.20, “Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing.”

3. NUREG-1061, Volume 4.

4. ASME B&PV Code, Section III, Appendix II: If experimental stress analysis methods are used in lieu of analytical methods for any Seismic Category I mechanical components, sufficient information to allow the staff to determine their acceptability when compared to the requirements of the ASME Code, Section III, Appendix II.

5. ASME B&PV Code, Section III, Appendix F: If inelastic analysis methods, including ASME Code, Section III, Service Level D limits, are used for any Seismic Category I mechanical components, compliance of the analytical methodology used to calculate stresses and deformations to the methods specified in the ASME Code, Section III, Appendix F.

3.9.1.4 *Technical Evaluation*

3.9.1.4.1 Design Transients

In accordance with the guidance in SRP Section 3.9.1(III), the staff reviewed FSAR Tier 2 design transients given in FSAR Tier 2, Table 3.9.1-1.

Based on its review, the staff requested discussion and justification of various transients given in FSAR Tier 2, Table 3.9.1-1.

FSAR Tier 2, Section 3.9.1.1 provides a list of the design transients and the number of cycles for the transients in FSAR Tier 2, Table 3.9.1-1. The applicant stated that the number of cycles is a conservative estimate of the magnitude and frequency of the temperature and pressure transients that may occur during plant operation. The number of design transients is based on a plant life of 60 years. The transients are defined for equipment design purposes and are not intended to represent actual operating experience. However, the applicant did not provide description of the transients. In accordance with SRP Section 3.9.1 III.1, the method for determining the number of occurrences of each event should conform to the same information on similar and previously licensed applications. Any deviations from previous accepted practice are noted, and the applicant should justify them. Previous accepted applications define the magnitude of these transients including temperature and pressure variations and duration.
In RAI 179, Question 03.09.01-1, the staff requested that the applicant (1) provide information for each of the given transients including the event background, involved systems, operating conditions (pressure, temperature, and flow), including heat-up and cool-down rate limits; (2) provide the basis for the Plant Operating Events and corresponding “number of events” given in FSAR Tier 2, Table 3.9.1-1; (3) confirm that the transients in FSAR Tier 2, Table 3.9.1-1 are for 60 years; (4) explain in detail for events relating to the unscheduled power variation, the unscheduled fluctuations at hot shutdown and the external induced transient; (5) discuss the basis for selecting one cycle for external induced transient, control rod ejection, main feedwater line break, and main steam line break; and (6) provide the number of occurrences for each of the emergency transients.

In a May 13, 2009, response to RAI 179, Question 03.09.01-1.1, the applicant stated that FSAR Tier 2, Section 3.9.1.1 will be revised to provide a description of each design transient. The level of detail provided for the description of each design transient conforms to similar information provided in other design certification applications in accordance with SRP Section 3.9.1, Section III.1. The plant operating events given in FSAR Tier 2, Table 3.9.1-1 correspond to the U.S. EPR design transients. The applicant further indicated that FSAR Tier 2, Table 3.9.1-1 will be revised to add the transient ID and the corresponding number of occurrences for each design transient. As noted in FSAR Tier 2, Section 3.9.1, the number of occurrences given in FSAR Tier 2, Table 3.9.1-1 is based on a 60-year design life for the U.S. EPR. In addition, the applicant provided a summary of events relating to the unscheduled power variation, the unscheduled fluctuations at hot shutdown, and the external induced transient along with the descriptions of transients as shown in a marked-up copy of the FSAR in the May 13, 2009, response to RAI 179, Question 03.09.01-1(A). The response noted that the basis for selecting one cycle for externally induced transient, control rod ejection, main feedwater line break, and main steamline break is that while these faulted conditions are not expected to occur, the plant is designed to mitigate a single occurrence. In a May 13, 2009, response to RAI 179, Question 03.09.01-1(F), the applicant stated that FSAR Tier 2, Table 3.9.1-1 will be revised to indicate the specific number of occurrences for each of the emergency transients. The staff reviewed proposed revisions to FSAR Tier 2, Section 3.9.1.1 and FSAR Tier 2, Table 3.9.1-1 and finds the additional information provided by the applicant in the May 13, 2009, response to RAI 179, Question 03.09.01-1.1 acceptable. The staff will confirm the upcoming FSAR revision to incorporate the proposed additional information. **RAI 179, Question 03.09.01-1.1 is being tracked as a confirmatory item.**

10 CFR Part 50, Appendix S specifies that applicants include seismic events in the design basis. In RAI 179, Question 03.09.01-1.2, the staff requested that the applicant provide the basis to justify not including earthquakes dynamic events at the rated operating power conditions in FSAR Tier 2, Table 3.9.1-1. In a February 26, 2009, response to RAI 179, Question 03.09.01-1.2, the applicant stated that FSAR Tier 2, Table 3.9.1-1 provides a summary of thermal design transients. The seismic design basis is addressed in FSAR Tier 2, Section 3.7.1, “Seismic Design Parameters,” Section 3.7.2, “Seismic System Analysis,” and Section 3.7.3, “Seismic Subsystem Analysis.” The staff’s FSER for Topical Report ANP-10264NP-A, “Piping Analysis and Pipe Support Design Topical Report,” states: “AREVA meets 10 CFR Part 50, Appendix S, requirements by designing the safety-related piping systems, with a reasonable assurance to withstand the dynamic effects of earthquakes with an appropriate combination of other loads of normal operation and postulated events with an adequate margin for ensuring their safety functions.” Additionally, per FSAR Tier 2, Table 3.9.3-1, the seismic inertial loads are included in the fatigue analysis of ASME Code Class 1 Components. The earthquake dynamic loads are included in the fatigue analysis of structures, systems, and components. The staff notes that FSAR Tier 2, Table 3.9.3-1 provides the loading combinations and corresponding stress design criteria per ASME Service Level for ASME Code Class 1 components. Topical Report ANP-10264NP-A, Section 3.4 states that the fatigue analysis be performed for all ASME Code Class 1 piping to meet 10 CFR Part 50, Appendix S and 10 CFR Part 100, Appendix A requirements. However, neither FSAR Tier 2, Table 3.9.3-1, nor Topical Report ANP-10264NP-A provides specific requirements for fatigue evaluation regarding the number of cycles, estimated magnitude, and frequency of the reversing dynamic seismic events that may occur during the 60 years plant operation. As a result, the staff closed RAI 179, Question 03.09.01-1.2, and in follow-up RAI 404, Question 03.09.01-5, the staff requested that the applicant clarify fatigue evaluation as indicated above. In a June 30, 2010, response to RAI 404, Question 03.09.01-5, the applicant indicated that FSAR Tier 2, Table 3.9.1-1 will be revised to include the information as shown in the markup to the response for fatigue evaluation. The staff will confirm the revised FSAR Tier 2, Table 3.9.1-1 in the upcoming revision. **RAI 404, Question 03.09.01-5 is being tracked as a confirmatory item.**

The turbine stop valve (TSV) closure transient effects on the main steam (MS) line piping and its supports is significant due to the pressure and flow changes with the instant closure of the TSV. The MS analysis including the TSV transient should be used or the design of MS piping and supports since the closure time of the TSV is very short, this results in a higher steam hammer load. In RAI 179, Question 03.09.01-1.3, the staff requested that the applicant provide the basis for not including the turbine stop valve closure induced loads in FSAR Tier 2, Section 3.9.1. In a

February 26, 2009, response to RAI 179, Question 03.09.01-1.3, the applicant stated that they had not been able to identify a precedent in which the NRC has requested a design certification applicant to provide such information. The staff notes that the TSV closure event has been considered to be the Service Level B fluid transient loads (i.e., steam hammer loads) resulting from the TSV closure event. Staff closed RAI 179, Question 03.09.01-1.3 and in follow-up RAI 404, Question 03.09.01-6, the staff requested that the applicant address why this transient due to TSV closure is not applicable to U.S. EPR design. In a June 30, 2010, response to RAI 404, Question 03.09.01-6, the applicant stated that turbine trip is accomplished by closure of the turbine valves (stop valves and control valves). Turbine Stop Valve closure is included in the upset transients in FSAR Tier 2, Section 3.9.1.1.2. Specifically, Upset Transient 2 is the plant response to a turbine trip (loss of load event). Since Upset Transient 2 currently considers turbine bypass to be unavailable, this transient is more conservative than looking at closure of the TSV by itself (with bypass remaining available). Since TSV closure is included in the upset transients in Section 3.9.1.1.2, the staff finds the applicant's response acceptable, and therefore considers this issue resolved.

NRC Bulletin (BL) 79-13, "Cracking in Feedwater System Piping," addressed the fatigue loading due to thermal stratification and high cycle thermal striping during low flow emergency feedwater injection. NRC BL 88-08, "Thermal Stresses in Piping Connected to Reactor Cooling Systems," and its supplements indicate that during low feed water flow, stratification flow conditions can result in significant differences in thermal fatigue cycles that have resulted in failures of the feedwater piping pressure boundary in pressurized water reactors. NRC BL 88-11, "Pressurizer Surge Line Thermal Stratification," called for consideration of the effects of thermal stratification on the pressurizer surge line dynamic loads. In RAI 179, Question 03.09.01-1.4, the staff requested that the applicant provide the basis for not considering the thermal stratification in FSAR Tier 2, Section 3.9.1.1, as it relates to the design transient in the piping design on fatigue.

In a February 26, 2009, response to RAI 179, Question 03.09.01-1.4, the applicant stated that FSAR Tier 2, Section 3.12.5.10, "Thermal Stratification," addresses thermal stratification for the pressurizer (PZR) surge line, PZR lower head, normal spray line, auxiliary spray line, main feedwater (MFW) line, and emergency feedwater (EFW) line. The contribution of normal and upset condition stratification cycles is considered in the fatigue analysis of these piping systems. In addition, the applicant referred the staff to its August 21, 2008, response to RAI 48, Question 03.06.03-3 and Question 03.06.03-4, for more information on thermal stratification of the PZR SL. The staff noted in RAI 179, Question 03.09.01-1.4, that the stratification has been extensively considered in the design of piping in FSAR Tier 2, Section 3.12.5.10 for fatigue analysis. However, the applicant has not defined or described the thermal stratification transient including information regarding the number of cycles for the transients and the magnitude and frequency of the transients that may occur during plant operation. The staff determined that the applicant's response was not acceptable. As a result, staff closed RAI 179, Question 03.09.01-1.4 in follow-up RAI 404, Question 03.09.01-7, requested that the applicant clarify thermal stratification transient as indicated above.

In a June 30, 2010, response, to RAI 404, Question 03.09.01-7, the applicant stated that stratification transients are not separate events. They occur because of particular combinations of flow and temperature and are an inherent part of several design transients. The number of cycles for each transient is provided, therefore, the number and frequency of stratification events can be derived based on the cycles of the transients in which they occur. The applicant also indicated that FSAR Tier 2, Table 3.9.1-1 will be revised to reflect the information as described in the markup attached to this response for the fatigue analysis of piping system. **Follow-up RAI 404, Question 03.09.01-7 is being tracked as a confirmatory item.**

SRP Section 3.9.2 states, "the instabilities in these flow fields can couple with acoustic and/or structural resonances, causing high dynamic loads throughout the steam system and reactor pressure vessel (RPV)." Thus, the vibrations from acoustic resonances are readily identifiable throughout an affected piping system. Regarding the vibration effects on the components and piping due to acoustic resonance stated in RG 1.20, Revision 3, in RAI 179, Question 03.09.01-1.5, the staff requested that the applicant provide the basis for not including this acoustic cyclic loading in FSAR Tier 2, Section 3.9.1.1, as it relates to cyclic loadings applying to Class 1 piping and components.

On the basis of this evaluation and the evaluation of the responses to RAI 179, Questions 03.09.01-1 through 03.09.01-4 and the follow-on RAIs, the staff concludes that the use of operating plant experience, adjusted for a 60-year plant life, plus additional cycles to account for seismic events, provides an acceptable basis for estimating the total number of cycles for each transient.

Therefore, the staff concludes that the information relative to the U.S. EPR design transients in FSAR Tier 2, Section 3.9.1.1, complies with the applicable guidelines in SRP Section 3.9, and is therefore acceptable.

3.9.1.4.2 Computer Programs

In accordance with SRP Section 3.9.1(III), in RAI 179, Question 03.09.01-2.1, the staff requested that the applicant (1) list all computer programs used for generating hydraulic forcing functions and performing structural analyses including preprocessors and postprocessors which are given in FSAR Tier 2, Section 3.9.1; FSAR Tier 2, Appendix 3C; and in the Topical Report ANP-10264NP-A, Section 5.1; and (2) confirm that the analyses of pipe supports were performed using the computer code GT STRUDL, which is not stated in the FSAR. In a May 13, 2009, response to RAI 179, Question 03.09.01-2.1, the applicant indicated that a list of computer programs used for generating hydraulic forcing functions and performing structural analyses along with their respective preprocessors and postprocessors is provided in FSAR Tier 2, Section 3.9.1.2; FSAR Tier 2, Appendix 3C; and AREVA Topical Report ANP-10264NP-A, Section 5.1. In addition to these computer programs, ROLAST and S-TRAC are also used for the calculation of dynamic hydraulic loads on piping systems. A description of these computer programs will be added to FSAR Tier 2, Section 3.9.1. As noted in ANP-10264NP-A, Section 3.3.1 and FSAR Tier 2, Section 3.9.1.2, the requested information regarding these computer programs is available for NRC inspection. Based on the February 26, 2009, response to RAI 179, Question 03.09.01-2.1, the applicant provided that information on the computer code GT STRUDL in Question (b) is included in marked up pages to the FSAR. The staff performed an audit on July 20 - 23, 2010, at the AREVA offices in Lynchburg, VA. As a result, the staff finds that computer codes used to design ASME Code Class 1 piping and components are well documented based on the information provided by the applicant during the audit. The staff will confirm the upcoming FSAR revision to incorporate the proposed additional information. **RAI 179, Question 03.09.01-2.1 is being tracked as confirmatory item.**

10 CFR Part 50, Appendix B requires provisions to assure that appropriate standards are specified and included in design documents including design methods and computer programs for the design and analysis of Seismic Category I, ASME Code Class 1, 2, 3, and core support structures and non-Code structures. In RAI 179, Question 03.09.01-2.2, the staff requested that the applicant confirm that the computer programs used for U.S. EPR design and given in FSAR Tier 2, Section 3.9.1.1, FSAR Tier 2, Appendix 3C, and Topical Report ANP-10264NP-A, Section 5.1, Computer Codes, programs.

In a February 26, 2009, response to RAI 179, Question 03.09.01-2.2, the applicant indicated that the AREVA Quality Assurance Plan for Design Certification of the U.S. EPR Topical Report ANP-10266A, Section 2.1.1, “Scope,” states, “The QAP assures that activities affecting quality are accomplished under suitably controlled conditions. It also provides for the development, control, and use of computer programs.” Additionally, Topical Report ANP-10266A, Section 3.6.2, “Design Control,” states, “Computer programs used for design analyses are certified or verified and validated as appropriate.” Accordingly, the computer codes given in the FSAR sections in this question are subject to the requirements of Topical Report ANP 10266A. The staff reviewed the information in Topical Report ANP-10264A and determined that there are computer programs that are not reviewed and approved by NRC. As a result, the staff conducted an audit of selected computer programs. The applicant indicated that, as stated in FSAR Tier 2, Section 3.9.1, the following information on computer codes is available for NRC inspection: Author; source; version date; program description; extent and limitation of the program application; and code solutions to test data. RAI 179, Question 03.09.01-2.2 remains open until the staff’s satisfactory completion of the technical audit of quality assurance of the computer codes. **RAI 179, Question 03.09.01-2.2 is being tracked as an open item.**

For applicability of the computer code BWSPAN, the applicant stated that the use of BWSPAN for Class 1 reactor coolant loop (RCL) analysis has previously been approved by the NRC, see letter David E. LaBarge (NRC) to W.R. McCollum, Jr. (Duke Energy Corporation), Oconee Nuclear Station, Units 1, 2, and 3, Re: Reactor Coolant Loop (RCL) Analysis Methodology for Steam Generator Replacement, September 6, 2001. This reference letter contains a safety evaluation in which the staff evaluated the steam generator replacement analysis for Oconee, but did not review the BWSPAN computer code. In RAI 179, Question 03.09.01-2.3, the staff requested that the applicant provide a summary of the verification and validation (V&V) for this program including benchmark problems. The V&V information is required by 10 CFR Part 50, Appendix B, and should be available as requested above. Instead of providing the V&V information, the applicant stated in a February 26, 2009, response to RAI 179, Question 03.09.01-2.3, that NRC FSER for Topical Report ANP-10264NP-A, Section 3.3.1 approved the use of the BWSPAN computer code for the U.S. EPR, and specifically accepted the referenced Oconee RCL analysis methodology for steam generator replacement as part of the basis for their approval. However, the staff determined that methodology for steam generator replacement using BWSPAN program does not relate to the review of the V&V information of BWSPAN. As a result, in follow-up RAI 404, Question 03.09.01-9, the staff requested that the applicant clarify the methodology for steam generator replacement using BWSPAN program. In a June 30, 2010, response to follow-up RAI 404, Question 03.09.01-9, the applicant indicated that the requested information is available for NRC inspection. This issue remains open until the staff’s satisfactory completion of technical audit of the BWSPAN computer code. **RAI 179, Question 03.09.01-2.3 is being tracked as an open item.**

Regarding the qualification of the SUPERPIPE computer program, the applicant indicated that previous versions were approved by the NRC as shown in previous license applications including the [Catawba Nuclear Station (CNS) Updated Final Safety Analysis Report (UFSAR) Revision 12, Table 3-68] and the Combustion Engineering (CE) System 80+ Design Certification (NUREG-1462, Section 3.12.3)). In RAI 179, Question 03.09.01-2.4, the staff requested that the applicant provide the dates and version number of the current version that was used for the design of U.S. EPR components and piping. In a February 26, 2009, response to RAI 179, Question 03.09.01-2.4, the applicant stated that Topical Report ANP-10264NP-A, Section 3.3.1 approved the use of the SUPERPIPE computer code for the U.S. EPR design certification and the documentation of the SUPERPIPE program is available for NRC inspection. The staff notes that this issue will be resolved by the audit to be conducted to resolve RAI 179, Question 03.09.01-2.2. RAI 179, Question 03.09.01-2.4 remains open until the staff's satisfactory completion of the technical audit of the SUPERPIPE computer codes. **RAI 179, Question 03.09.01-2.2 is being tracked as an open item.**

In FSAR Tier 2, Appendix 3C, there are computer codes requiring verification against the results from the analysis of a sample problem to the classical solution of the sample problem, each time they are executed. In RAI 179, Question 03.09.01-2.5, the staff requested that the applicant (a) explain why it is required to run a sample problem each time the computer code is used, (b) confirm whether the results of the sample problem runs are documented in the calculation, and (c) discuss how the quality assurance requirements will be satisfied for not using an executable file. In a February 26, 2009, response to RAI 179, Question 03.09.01-2.5, the applicant explained that (a) the purpose of running these test cases is to perform an additional check for computer-based codes and to demonstrate that computers with different hardware configurations do not adversely affect the performance of the program, and (b) the internally developed software programs have verification documents which consist of test cases the users are required to perform each time the program is used. For some programs, these test cases are classical textbook solutions. Regarding the QA Item (c), the applicant indicated that the computer programs described in FSAR Tier 2, Appendix 3C are verified per the AREVA QAP in accordance with the requirements of 10 CFR Part 50 Appendix B. The staff considers the applicant's response to have provided sufficient explanations regarding the computer programs stated above. Therefore, the staff considers RAI 179, Question 03.09.01-2.5 resolved.

ASME Code, Section III requires that the cumulative damage from fatigue be evaluated for all ASME Code Class 1 piping, components, and supports. In RAI 179, Question 03.09.01-2.6, the staff requested that the applicant identify the computer programs that were used to perform the fatigue analysis and confirm these analyses for ASME Code, Section III Class 1 components and piping for the fatigue evaluation including environmental effects in accordance with RG 1.207. In a February 26, 2009, response to RAI 179, Question 03.09.01-2.6, the applicant identified computer codes ANSYS, BWSPAN, and SUPERPIPE which are used to perform the fatigue analysis for ASME Code Class 1 piping and components. The applicant also stated that, as noted in FSAR Tier 2, Section 3.12.5.19, "Effects of Environment on Fatigue Design," the effects of reactor coolant environment, using the methodology described in RG 1.207, are considered when performing fatigue analyses for Class 1 piping and components. The applicant did not address the staff's question of whether the computer codes ANSYS, BWSPAN, and SUPERPIPE incorporate the environmental effects on fatigue.

As a result, in follow-up RAI 404, Question 03.09.01-10, the staff requested that the applicant address whether the computer codes ANSYS, BWSPAN, and SUPERPIPE incorporates the environmental effects on fatigue. In a June 30, 2010, response to follow-up RAI 404, Question 03.09.01-10, the applicant indicated that computer codes ANSYS, BWSPAN and SUPERPIPE perform ASME Code in-air fatigue analysis and do not incorporate the environmental effects on fatigue per RG 1.207. The calculations for consideration of the environmental effects on fatigue are performed by FatTool, a post-processing program to ANSYS, BWSPAN and SUPERPIPE. FatTool is developed for the fatigue analysis of ASME Code Class 1 Piping. FatTool consists of two modules, namely the In-Air and Environmentally Assisted Fatigue (EAF) modules. The EAF module is a postprocessor of the In-Air module and incorporates the effect of Light-Water Reactor environment on the fatigue resistance of piping per the requirements of RG 1.207. The analysis methodology complies with the piping stress analysis requirements in ASME B&PV Code, Section III, Division I, NB-3600 and the EAF criteria and fatigue curves provided in NUREG/CR-6909 and RG 1.207. The applicant also indicated it will revise FSAR Tier 2, Section 3.9.1.2 to include the description of computer code FatTool as provided in its response. **RAI 404, Question 03.09.01-10, is being tracked as a confirmatory item**.

The staff notes that structural and piping damages during seismic events were often caused by the foundation and anchor movements. To prevent such damages, the staff suggested that appropriate methodologies discussed in NUREG-1061 should be used for calculating the stresses and fatigue on piping and components subjected to multiple individual support motions. In RAI 179, Question 03.09.01-2.7, the staff requested that the applicant verify that all computer programs used for U.S. EPR design of piping that use the Independent Support Motion (ISM) Response Spectrum analysis method comply with the staff position for combining mode, group (absolute sum) and direction responses, as stated in NUREG-1061, Volume 4. In a February 26, 2009, response to RAI 179, Question 03.09.01-2.7, the applicant stated that conformance with NUREG-1061, Volume 4 was evaluated by the NRC in Topical Report ANP-10264NP-A, Section 3.2.3. The applicant did not address how the methodology in NUREG-1061, Volume 4 with absolute summation is appropriately incorporated in the U.S. EPR analyses when using computer codes ANSYS, BWSPAN, and SUPERPIPE in the piping analyses. As a result, in follow-up RAI 404, Question 03.09.01-11, the staff requested that the applicant provide the information discussed above.

In a July 30, 2010, response to, RAI 404, Question 03.09.01-11, the applicant stated that the provisions of NUREG-1061, Volume 4, for using the ISM method of analysis will be followed for U.S. EPR piping design. This includes combining the support group responses, for each mode and each direction, using the absolute sum method of combination. Computer codes BWSPAN and SUPERPIPE incorporate the above methodology for performing modal combinations when using independent support motions. The staff finds the applicant’s response regarding use of ISM methodology acceptable and, therefore, considers follow-up RAI 404, Question 03.09.01-11 resolved.

On July 20 to 23, 2010, the staff conducted a technical audit at the AREVA offices in Lynchburg, VA. The purpose of this regulatory review was to verify and validate the computer programs which are used for the design and analysis of the ASME Code Class 1, 2, and 3 components and piping. The review was performed in accordance with requirements of 10 CFR Part 50, Appendix B and ASME NQA-1 Code, the methodology and criteria described in the ASME Code Section III, and the FSAR in support of the U.S. EPR design certification. As a result, in RAI 458, Question 03.09.01-13, the staff requested that the applicant provide additional information as stated below.

FSAR Tier 2, Section 3.12.5.19 states, "the effects of reactor coolant environment, using the methodology described in RG 1.207, are considered when performing fatigue analyses for Class 1 piping and components." RG 1.207 recommends using the Fen method presented in NUREG/CR-6909. The staff reviewed the design basis documentation (Document # 32-9119032-002, "Fatigue Modules for U.S. EPR Piping System," July 16, 2010). The staff identified the discrepancies between the design basis document and NUREG/CR-6909 as follows:

NUREG/CR-6909		AREVA Document #: 32-9119032-002
Eq. 23	T* = T – 150 (150 < T ≤350°C)	T* = T – 302 (302 < T ≤662°F)
Eq. 25	ε* = ln(0.001) (ε* < 0.001%/s)	ε* = ln(1.0E-5) (ε* < 0.001%/s)

In RAI 458, Question 03.09.01-13, the staff requested that the applicant clarify and correct the differences.

The staff notes that the applicant's Document #: 32-9119032-002, "Fatigue Modules for U.S. EPR Piping System," July 16, 2010 indicated that the threshold strain amplitude of 0.1 percent is considered for all type material. However, the threshold strain amplitude of 0.07 percent for carbon steel and low alloy steel is defined in NUREG/CR-6909, Section 4.2.13. The staff requested that the applicant clarify the differences of the threshold strain amplitude used by AREVA and that are defined in NUREG/CR-6909.

The staff also noted that the applicant's weighted average Fen (environmental factor for fatigue usage) method is not consistent with the existing method identified in NUREG/CR-6909. The applicant stated that the weighted average Fen method is conservative. The staff's concern is that final Fen may be reduced if partial Fen=1.0 for some stress components due to threshold consideration. The staff requested that the applicant demonstrate the conservatism of the proposed weighted average Fen method. **RAI 458, Question 03.09.01-13 is being tracked as an open item.**

The staff notes that the design basis documentation (Document, "Program Verification of PC Version of P91232," 02/16/2001) P91232 stratification option is used to calculate equivalent linear profile using top and bottom temperature and non-linear portion of the top-to-bottom difference (ΔT4) to calculate local thermal stress.

The staff notes that the applicant's methodology developed to calculate the temperature profile on pipe cross-section (at mid-thickness) should be verified and benchmarked by a computer program which is recognized in the public domain and has had sufficient history of use to justify its applicability and validity (e.g. ANSYS). In RAI 458, Question 03.09.01-14, the staff requested that the applicant provide the benchmark comparison.

Revision 1 of the FatTool (Document No.: 32-9119032-001) defined that local thermal stratification stress equals to EαΔT4/ (1- ν). Revision 2 the FatTool (Document No. 32-9119032-002) defined that local thermal stratification stress equals to EαΔT4. In RAI 458, Question 03.09.01-14, the staff also requested that the applicant provide the benchmark justification for the local thermal stratification stress. **RAI 458, Question 03.09.01-14 is being tracked as an open item.**

The staff notes that FSAR Tier 2, Section 3.9.1.2 identified that computer program RESPECT is to be used for U.S. EPR design. The staff reviewed the supporting documents for RESPECT computer program that generates response spectrum from an acceleration time history for piping analysis. In RAI 458, Question 03.09.01-15, the staff requested that the applicant provide validation for this computer program. **RAI 458, Question 03.09.01-15 is being tracked as an open item.**

Except for the open items in Section 3.9.1.4.2 of this report, and based on the review of the information and the responses to the RAIs mentioned in this section. The staff concludes that the computer code qualification methods described in this section comply with the guidance of SRP Section 3.9.1, and are therefore acceptable.

3.9.1.4.3 Experimental Stress Analysis

FSAR Tier 2, Section 3.9.1.3 indicates that experimental stress analysis is not used to evaluate stresses for Seismic Category I components and supports. In RAI 179, Question 03.09.01-3, the staff requested that the applicant discuss the stress analysis methods used to verify the design adequacy for the design of U.S. EPR components such as snubbers, pipe whip restraints, and the prototype fine motion control rod drive. In a February 26, 2009, response to RAI 179, Question 03.09.01-3, the applicant indicated that the pipe whip restraints are designed using elastic and elastic-plastic methodologies in accordance with the guidance in SRP Section 3.6.2, and that experimental stress analysis is not used to evaluate stresses for the restraints.

Regarding the stress evaluation for the snubbers, the applicant referred to its October 31, 2008, response to RAI 107, Questions 03.09.03-13 and 03.09.03-14, and the February 24, 2009, response to RAI 178, Questions 03.09.03-19 and 03.09.03-20, where it addresses the snubbers as the linear supports which may be designed by experimental analysis or load rating methods in accordance with ASME Code, Section III, Subsections NF-3370 and NF-3380. The applicant noted that it does not design and manufacture snubber components; they are purchased from a qualified vendor to meet ASME Code requirements. Snubber vendors provide a certified load data sheet that states the design of its snubber meets the requirements of ASME Code, Section III, Subsections NCA and NF. ASME Code, Section III, Subsection NF, Paragraph NF-1214, “Standard Supports,” provides guidance on the design of snubbers. The design specifications require the snubber vendor to meet the design stress criteria of the applicable ASME Code standards.

The staff notes that it is the applicant’s responsibility to ensure that the vendor design of its snubbers meets the requirements of ASME Code, Section III, Subsections NCA and NF.
The applicant needs to confirm that if snubbers are designed by experimental stress analysis, that they meet the provisions of Appendix II to ASME Code, Section III, Division 1, in accordance with SRP Section 3.9.1.II acceptance criterion. Therefore, in follow-up RAI 404, Question 03.09.01-12, the staff requested that the applicant provide the information discussed above. In a July 30, 2010, response to follow-up RAI 404, Question 03.09.01-12, the applicant stated that experimental stress analysis is not used for Seismic Category I systems or components for the U.S. EPR. This is inconsistent with the information previously provided by the applicant and requires a clarification from the applicant. RAI 404, Question 03.09.01-12 is tracked as open item.

3.9.1.4.4 Considerations for the Evaluation of Faulted Conditions - Inelastic Analyses

In accordance with the guidance in SRP Section 3.9.1(III), the staff reviewed the evaluation of components under the faulted loading conditions identified in FSAR Tier 2, Tables 3.9.1-1 and 3.9.3-1.

In FSAR Tier 2, Section 3.9.1.4, the applicant referred to FSAR Tier 2, Section 3.9.3 for the analytical methods used for Seismic Category I systems and components subjected to faulted conditions. FSAR Tier 2, Section 3.9.3 states that calculation methods used to evaluate RCS components and their supports for faulted loading are provided in FSAR Tier 2, Appendix 3C. Calculation methods used to evaluate piping and supports are described in Topical Report ANP-10264NP-A, Sections 4 and 6. FSAR Tier 2, Table 3.9.3-1, Note 15 states that the rules given in ASME Code, Section III, Appendix F are used for analysis of faulted service condition loading. In RAI 179, Question 03.09.01-4.1, the staff requested that the applicant describe the computer programs that were used to evaluate the stresses for determining that the ASME Code, Section III, Appendix F, limits were met. In a May 13, 2009, response to RAI 179, Question 03.09.01-4.1, the applicant indicated that computer program ANSYS (see FSAR Tier 2, Section 3.9.1.2), which is a finite element program, is used to evaluate the stresses for the reactor pressure vessel, SG, PZR, and core support structures. Computer programs used to evaluate stresses for vendor supplied components (e.g., reactor coolant pumps) will be referenced in the ASME Code design reports; FSAR Tier 2, Section 3.9.1.2 will be revised accordingly. **RAI 179, Question 03.09.01-4.1 is being tracked as a confirmatory item.**

In FSAR Tier 2, Section 3.9.3, the applicant stated that calculation methods used to evaluate RCS components and their supports for faulted loading are provided in FSAR Tier 2, Appendix 3C. The staff determines that FSAR Tier 2, Appendix 3C describes the general static and dynamic structural analysis.

In RAI 179, Question 03.09.01-4.2, the staff requested that the applicant identify the components evaluated in FSAR Tier 2. Section 3.9.3, where the inelastic Service Level D limits were met under the faulted condition loads and load combinations described in FSAR Tier 2, Table 3.9.3-1. In a May 13, 2009, response to RAI 179, Question 03.09.01-4.2, the applicant stated that SG tube support plates are evaluated using the collapse load method in accordance with F-1341.3 for Level D service conditions specified in ASME Code, Section III. If any additional components are identified requiring inelastic analysis to meet Service Level D limits, these codes will be evaluated using F-1330, ASME Code, Section III, for elastic system analyses or F-1340 for plastic system analyses, and will be documented in the ASME Code design reports. The staff finds the applicant's response adequate and acceptable in accordance with SRP 3.9.1.II acceptable criteria. Therefore, the staff considers RAI 179, Question 03.09.01-4.2 resolved.

Based on its review and the responses to the RAI, the staff determines that the application of elastic and inelastic stress analyses are in compliance with ASME Code, Section III, Appendix F, and are therefore acceptable.

3.9.1.5 *Combined License Information Items*

Table 3.9.1-1 provides a list of special topics for mechanical components related COL information item numbers and descriptions from FSAR Tier 2, Table 1.8-2:

Table 3.9.1-1 U.S. EPR Combined License Information Items

Item No.	Description	FSAR Tier 2 Section
3.9-9	A COL applicant that references the U.S. EPR design certification will either use a piping analysis program based on the computer codes described in Section 3.9.1 and Appendix 3C or will implement an NRC-approved benchmark program using models specifically selected for the U.S. EPR.	3.9.1.2
3.9-10	Pipe stress and support analysis will be performed by a COL applicant that references the U.S. EPR design certification.	3.9.1.2

The staff reviewed COL Information Items 3.9-9 and 3.9-10 as they relate to FSAR Tier 2, Section 3.9.1. However, the staff will not be able to complete its evaluation of these COL information items until after the resolution of the open and confirmatory items.

3.9.1.6 *Conclusions*

Except for the open items discussed above, and based on the evaluations in Sections 3.9.1.1 through 3.9.1.4 of this report, the staff concludes that the design transients and resulting loads and load combinations with appropriate specified design and service limits for mechanical components are acceptable and meet the relevant requirements of GDC 1, GDC 2, GDC 14, GDC 15; 10 CFR Part 50, Appendix B; 10 CFR Part 50, Appendix S; and the guidelines in SRP Section 3.9.1, and therefore are acceptable.

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

3.9.3.1 *Introduction*

The structural integrity and functional capability of pressure-retaining components, their supports, and core support (CS) structures are ensured by designing them in accordance with ASME Code, Section III, or other industrial standards. This section addresses the loading combinations and their respective stress limits, the design and installation of pressure-relief devices, and the design and structural integrity of ASME Code Class 1, 2, and 3 components and component supports.

The criteria for the SSC design include the following considerations:

- Loading combinations, design transients, and stress limits
- Pump and valve operability assurance
- Design and installation criteria of Class 1, 2, and 3 pressure-relieving devices
- Component and piping supports

3.9.3.2 *Summary of Application*

FSAR Tier 1: Component design is addressed in the system description.

FSAR Tier 2: The applicant has provided an FSAR Tier 2 system description in FSAR Tier 2, Section 3.9.3, "ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures," summarized in part, as follows.

3.9.3.2.1 Loading Combinations, Design Transients, and Stress Limits

In FSAR Tier 2, Section 3.9.3.1, "Loading Combinations, System Operating Transients, and Stress Limits," the applicant describes the design and service loading combinations specified for ASME Code, Section III, components designated as ASME Code Class 1, 2, 3, and CS structures, including the appropriate system operating transients. The applicant defines the design loads and the loading combinations for the design of ASME Code Class 1, 2, and 3 components, supports, and core support structures. It also defines the stress limits applicable to the various loading combinations. The loading combinations and corresponding stress limits for ASME Code design are defined for the Design Condition, Service Levels A, B, C, and D (also known as normal, upset, emergency, and faulted conditions), and test conditions.

The applicant states that calculation methods used to evaluate reactor coolant system (RCS) components and their supports for faulted loading are provided in FSAR Tier 2, Appendix 3C. Calculation methods used to evaluate piping and supports are described in Topical Report ANP-10264NP-A, Sections 4 and 6. This Topical Report has been reviewed and approved by the staff in an FSER dated August 11, 2008.

3.9.3.2.2 Design and Installation of Pressure-Relief Devices

In FSAR Tier 2, Section 3.9.3.2, "Design and Installation of Pressure-Relief Devices," the applicant states that the Class 1 pressurizer safety relief valves (PSRVs) are designed to provide overpressure protection for the RCPB. The PSRVs connect to nozzles on the top head of the pressurizer and discharge through connected piping to the pressurizer relief tank.
The PSRVs, in conjunction with the main steam safety valves (MSSVs), prevent the RCPB from exceeding 110 percent of its design pressure with only safety classified systems in operation and the failure of the PSRV considered at the lowest set point.

The applicant states that the main steam relief isolation valves (MSRIVs) and the MSSVs are ASME Code, Section III, Class 2 pressure relief devices. The MSRIVs and the MSSVs provide overpressure protection for the secondary side of the steam generators. These valves are designed to the requirements of Subarticle NC-3500 of the ASME Code and American National

Standards Institute (ANSI) B16.34, "Valves-Flanged, Threaded, and Welding End," American National Standards Institute, 2004. Additional information on the MSRIVs and the MSSVs is provided in Section 10.3 of this report.

3.9.3.2.3 Pump and Valve Operability Assurance

In FSAR Tier 2, Section 3.9.3.3, "Pump and Valve Operability Assurance," the applicant states that pump operability is established initially by subjecting the pumps to factory tests prior to installation. These factory tests are followed by post-installation testing in the plant. Factory tests include a hydrostatic test for pressure retaining parts, pump seal leakage tests to the hydrostatic test pressure, and performance tests to establish pump head requirements. Post-installation testing includes cold hydrostatic tests and hot functional tests as part of the piping system testing.

The applicant also states that active valve operability is established initially by subjecting the valves to factory tests prior to installation. These tests are followed by post-installation testing in the plant. Factory tests include a shell hydrostatic test, a valve closure test, and a performance test to verify correct opening and closing of the valve. In addition to the factory tests, other post-installation tests are performed on these valves, including cold hydrostatic tests, hot functional tests, periodic inservice inspections, and periodic inservice operational tests.
In addition to the valve qualifications noted above, a representative sample of each valve type is tested for operability during a simulated plant condition event.

3.9.3.2.4 Component Supports

In FSAR Tier 2, Section 3.9.3.4, "Component Supports," the applicant states that core support structures and ASME Code, Section III, Class 1, 2, and 3 component and piping supports meet the stress criteria of the ASME Code using the loadings and combinations outlined in FSAR Tier 2, Section 3.9.3.1. In addition, the applicant states that Class 1 linear-type and plate-and-shell-type support structures are designed in accordance with the criteria in RG 1.124 and RG 1.130, "Service Limits and Loading Combinations for Class 1 Plate-and Shell-Type Component Supports."

ITAAC: There are ITAAC to address the component design and as-built configuration.

3.9.3.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 3.9.3 and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Section 3.9.3.

1. 10 CFR 50.55a and 10 CFR Part 50, Appendix A, GDC 1 as they relate to structures and components being designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

2. GDC 2 and 10 CFR Part 50, Appendix S, as they relate to structures and components important to safety being designed to withstand the effects of earthquakes without loss of capability to perform their safety functions.

3. GDC 4, as it relates to structures and components important to safety being designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents.

4. GDC 14, as it relates to the reactor coolant pressure boundary being designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

5. GDC 15, as it relates to the RCS and associated auxiliary, control and protection systems being designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

6. 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," requirements that the components and component supports, and core support structures will be designed and built in accordance with the certified design.

Acceptance criteria adequate to meet the above requirements include:

1. RG 1.124, "Service Limits and Loading Combinations for Class 1 Linear-Type Supports"

2. RG 1.130, "Service Limits and Loading Combinations for Class 1 Plate-and Shell-Type Component Supports"

3.9.3.4 *Technical Evaluation*

3.9.3.4.1 Loading Combinations, Design Transients, and Stress Limits

In accordance with SRP Section 3.9.3, the staff reviewed FSAR Tier 2 loading combinations, the design transients, and the stress limits that are used for the design of U.S. EPR safety-related ASME Code, Section III, Class 1, 2, and 3 components and component supports and CS structures. These are given in FSAR Tier 2, Table 3.9.3-1 for Class 1 components, and FSAR Tier 2, Table 3.9.3-2, "Load Combinations and Acceptance Criteria for ASME Code Class 2 and 3 Components," for ASME Code Class 2 and 3 components. FSAR Tier 2, Section 3.9.3.1.1, "Loads for Components, Component Supports, and Core Support Structure," describes the design and service level loadings used for the design of ASME Code Class 1, 2, and 3 components, piping, supports, and core support structures, including the appropriate system operating transients. FSAR Tier 2, Sections 3.9.3.1.2, "Load Combinations and Stress Limits for Class 1 Components," through 3.9.3.1.8, "Load Combinations and Stress Limits for Class 1, 2, and 3 Pipe Supports," define the loading combinations for the ASME Code Class 1, 2, and 3 components, piping, supports, and core support structures. These sections also define the stress limits applicable to the various load combinations. The loading combinations and corresponding stress limits for ASME Code design are defined for the Design Condition, Service Levels A, B, C, and D (also known as normal, upset, emergency, and faulted conditions), and test conditions.

FSAR Tier 2, Section 3.9.3 refers to the Topical Report ANP-10264NP-A for information related to the design and analysis of safety-related piping. This Topical Report presents ASME Code Class 1, 2, and 3 piping and pipe supports the U.S. EPR code requirements, acceptance criteria, analysis methods, and modeling techniques, which are also generally applicable to ASME Code Class 1, 2, and 3 components and component supports.

In FSAR Tier 2, Section 3.9.3, the applicant states that the U.S. EPR design is based on the 2004 Edition of the ASME Code. In RAI 107, Question 03.09.03-1, the staff requested that the applicant confirm that, for the design of components, component supports, and core support structures, the requirements of 10 CFR 50.55a(b) will be met without exception. In a December 1, 2008, response to RAI 107, Question 03.09.03-1, the applicant referred the staff to its September 30, 2008, response to RAI 51, Question 05.02.01.01-4, which states that the code of record for the design of the U.S. EPR is the 2004 Edition of the ASME Code. However, in their response to RAI 107, Question 03.09.03-1, the applicant did not confirm that the design of the components, component supports, and core support structures will meet the requirements of 10 CFR 50.55a(b) without exception. Therefore, in RAI 107, Question 03.09.03-17, the staff requested that the applicant clarify this issue, since the Topical Report uses the 2001 ASME Code, Section III, Division 1, 2003 Addenda as the base code with limitations identified in the 10 CFR 50.55a(b)(1).

In a February 24, 2009, response to RAI 107, Question 03.09.03-17, the applicant stated the U.S. EPR design complies with the 2004 ASME Code, Section III, Division 1, with no addenda, subject to the limitations and modifications identified in 10 CFR 50.55a(b)(1) and the piping analysis criteria and methods, modeling techniques, and pipe support criteria in the Topical Report. The staff finds this acceptable, because in a November 20, 2007 response to RAI EPR-3 on Topical Report ANP-10264NP-A, the applicant stated that the U.S. EPR piping design complies with the limitations in 10 CFR 50.55a(b)(1)(ii) Weld Leg, (iii) Seismic, (v) Independence of Inspection, and (vi) Inspection NH; and other limitations (i) Section III-Materials and (iv) Quality Assurance do not apply to the U.S. EPR design. Therefore, the staff considers RAI 107, Question 03.09.03-17 resolved.

In FSAR Tier 2, Section 3.9.3, there is no discussion on how each of the U.S. EPR pressure boundary safety-related mechanical components and component supports is designed. In RAI 107, Question 03.09.03-2, the staff requested that the applicant include in FSAR Tier 2, Section 3.9.3, a list of pertinent U.S. EPR pressure boundary safety-related components and component supports (with the respective design classifications), and provide a brief description of the design analysis and/or qualification methodologies for these components and component supports, including the codes and standards used. In a December 1, 2008, response to RAI 107, Question 03.09.03-2, the applicant stated that classification of pertinent U.S. EPR pressure boundary safety-related components, including the codes and standards used, is provided in FSAR Tier 2, Table 3.2.2-1. The functional design and qualification of active pumps and valves, and inservice testing programs to assess operability are addressed in FSAR Tier 2, Section 3.9.6, "Functional Design, Qualification, and Inservice Testing Programs, for Pumps, Valves, and Dynamic Restraints." The seismic and dynamic qualification of mechanical equipment is described in FSAR Tier 2, Section 3.10, "Seismic and Dynamic Qualification of Mechanical and Electrical Equipment."

The staff determined that for major mechanical components in the nuclear steam supply system, their supports, and components mounted on or within these components, such as reactor pressure vessel, reactor fuel assemblies, reactor internals, control rods, reactor coolant pumps, steam generators, and pressurizer, FSAR Tier 2, Appendix 3C provides adequate information regarding how the components and their supports are modeled for the finite element analysis and how pertinent dynamic loadings are considered in the qualification. Based on this additional information, the staff considers RAI 107, Question 03.09.03-2 resolved.

In FSAR Tier 2, Section 3.9.3.1, the applicant defined the loading combinations and corresponding stress limits for ASME Code design for the Design Condition, Service Levels A, B, C, and D, and test conditions.

However, a comprehensive description of plant conditions is not provided. In RAI 107, Question 03.09.03-3, the staff requested that the applicant provide a description for each of the plant conditions and their relations to frequency of occurrence, and describe major plant events accounted for in each of the plant conditions. In addition to the information provided in FSAR Tier 2, Tables 3.9.3-1 through 3.9.3-4, "Load Combinations and Acceptance Criteria for ASME Code Class 1, 2, and 3 Component Supports," the staff also requested that the applicant develop a table equivalent to SRP Section 3.9.3, Appendix A, Table I, which would more comprehensively illustrate U.S. EPR plant events, system operating conditions, service loading combinations, and service stress limits.

In a December 1, 2008, response to RAI 107, Question 03.09.03-3, the applicant stated that the information provided in FSAR Tier 2, Table 3.9.3-1 through Table 3.9.3-4 conforms to the guidance provided in RG 1.206 and SRP Section 3.9.3. Additionally, the information in these tables conforms to the information provided in Topical Report ANP-10264NP-A which has been approved by the NRC and was reviewed in accordance with the guidance of SRP Section 3.9.3. Based on the above, the staff determined that the information provided in the above mentioned tables, though not as comprehensive as the information provided in the guidance table in SRP Section 3.9.3, Appendix A, is not adequate in defining the design and service level loadings and their combinations, as well as the associated service level limits. The staff considers RAI 107, Question 03.09.03-3 unresolved. Therefore, in follow up RAI 503, Question 03.09.03-26, the staff requested that the applicant revise the service load combination for design pipe breaks. **RAI 503, Question 03.09.03-26 is being tracked as an open item.**

In FSAR Tier 2, Section 3.9.3, the applicant stated that a COL applicant that references the U.S. EPR design certification will prepare the design specifications and design reports for ASME Code Class 1, 2, and 3 components, piping, supports and core supports structures that comply with and are certified to the requirements of ASME Code, Section III. In order for the staff to reach a reasonable assurance finding based on the requirements of 10 CFR 52.47, however, certain additional information was required during the NRC review of the design certification application. Therefore, in RAI 107 Question 03.09.03-4, the staff requested that the applicant commit to provide the design specifications of risk-significant mechanical components, as a minimum, for NRC audit. This is to ensure that the components are ready for procurement, and that the FSAR design methodologies and criteria are adequately reflected in the associated component design specifications. As for the design reports, the staff requested that the applicant discuss in the FSAR its plan and schedule of making the as-designed design reports of U.S. EPR major mechanical components available for staff audit (e.g., through an ITAAC), to ensure that the applicant has established a procedure to verify the completion of the U.S. EPR component design.

In a December 1, 2008, response to RAI 107, Question 03.09.03-4, the applicant stated that it understands that the information requested pertains to the design specifications required for ASME Code Class 1, 2, and 3 components. In addition, the applicant also understands that NRC is interested in the availability of these design specifications in preparation for a staff audit.

Therefore, the applicant stated that a representative sample of the design specifications will be available for staff inspection beginning April 1, 2009.

The staff determined that the applicant's response to RAI 107, Question 03.09.03-4 was inadequate, because 10 CFR 52.47 requires that before design certification, information normally contained in procurement specifications be completed and available for audit by the staff if the information is necessary to make a safety determination. Therefore, to ensure a reasonable assurance finding, the design specifications of all risk-significant mechanical components, not just a representative sample, should be subjected to the staff's onsite review prior to the approval of U.S. EPR design certification application. In follow-up RAI 178, Question 03.09.03-18, the staff requested that the applicant provide design specifications of all safety-related mechanical components, not just a representative sample be available for staff onsite review, prior to the approval of the U.S. EPR design certification application.

In a February 24, 2009, response to RAI 178, Question 03.09.03-18, the applicant stated:

> As noted in the response to RAI 107, Question 03.09.03-4, and based on discussions with the NRC on January 26, 2009, regarding this question, AREVA NP understands that the information requested in this question pertains to the design specifications required for safety-related ASME Code Class 1, 2, and 3 components. As discussed with the NRC on January 26, 2009, the design specifications for safety-related ASME Code Class 1 reactor coolant system heavy components (e.g., reactor pressure vessel, steam generator, pressurizer), piping, and supports will be available for NRC inspection by April 1, 2009. For safety-related ASME Code Class 2 and 3 components, design specifications are prepared for the following types of components: pumps, valves, tanks, and heat exchangers. A typical design specification for the safety-related ASME Code Class 2 and 3 pumps and a typical design specification for safety-related ASME Code Class 2 and 3 valves will also be available for NRC inspection by April 1, 2009. A typical design specification for the ASME Code Class 2 and 3 heat exchangers and ASME Code Class 2 and 3 tanks will be available for NRC inspection by June 1, 2009. AREVA NP believes that this will provide NRC sufficient information to perform their review of the U.S. EPR design certification application, since the design specifications for each type of safety-related component have consistent design requirements.

The staff considers the applicant's response inadequate, since it does not comply with the requirements of 10 CFR 52.47. Therefore, in follow-up RAI 404, Question 03.09.03-24, the staff requested that the applicant provide the ASME Design Specifications of risk-significant mechanical components, at a minimum, for NRC audit prior to certification. This is to ensure that the components are ready for procurement, and that the FSAR design methodologies and criteria are adequately reflected in the associated component ASME Design Specifications. For the ASME design reports, the staff requested that the applicant discuss in the FSAR its plan and schedule for making the design reports of U.S. EPR mechanical components available for the NRC audit, (e.g., through an ITAAC, to ensure that AREVA has established a procedure to verify the completion of the U.S. EPR component design). In a December 1, 2008, response to RAI 404, Question 03.09.03-24, the applicant stated that a representative sample of the design specifications will be available for NRC inspection beginning April 1, 2009. In order for the staff to reach a reasonable assurance finding based on the requirements of 10 CFR 52.47, the staff requested that the applicant make available for audit the ASME design specifications for all risk-significant mechanical components, not just a representative sample, prior to the certification of the U.S. EPR design. **RAI 404, Question 03.09.03-24 is being tracked as an open item.**

In FSAR Tier 2, Section 3.9.3, the applicant states that a COL applicant that references the U.S. EPR design certification will provide a summary of the maximum total stress, deformation (where applicable), and cumulative usage factor values for each of the component operating conditions for ASME Code Class 1 components. In RAI 107, Question 03.09.03-6, the staff requested that the applicant elaborate on this COL commitment and explain the differences between this COL information item and the commitment discussed in a separate COL information item by which design specifications and design reports are required to be made available for staff audit.

In a December 1, 2008, response to RAI 107, Question 03.09.03-6, the applicant stated that this COL information item in FSAR Tier 2, Section 3.9.3, addresses the information requested in RG 1.206, Section C.I.3.9.3.1, Part C.III. Specifically, it requires that a summary of the maximum total stress, deformation, and cumulative usage factor values for each of the component operating conditions should be included for all ASME Code Class 1 components and component supports. It also requires that those values that differ from the allowable limits by less than 10 percent should be identified, and that the contribution of each of the loading categories (e.g., seismic, dead weight, pressure, and thermal) to the total stress for each maximum stress value identified in this range should be provided. The applicant also states that the COL information item related to design specification and design report applies to ASME Code Class 1, 2, and 3 components, piping, supports, and core support structures.

The staff determines the clarification provided by the applicant in their December 1, 2008, response to RAI 107, Question 03.09.03-6 adequately delineates the context of the above mentioned two COL information items, and is acceptable for ASME Code Class 1 items. Therefore, the staff considers RAI 107, Question 03.09.03-6 resolved. However, RG 1.206, Section C.I.3.9.3.1, Part C.III also requests that the applicant include a summary of the maximum total stress and deformation values for each of the components operating conditions for all ASME Code Class 2 and 3 components required to shut down the reactor or mitigate consequences of a postulated piping failure without offsite power. As a result, in follow-up RAI 211, Question 03.09.03-21, the staff requested that the applicant confirm that the existing COL information item for ASME Code Class 1 components will be revised to include ASME Code Class 2 and 3 components which are required to shutdown the reactor or mitigate consequences of a postulated piping failure without offsite power.

In a May 26, 2009, response to RAI 211, Question 03.09.03-21, the applicant stated that FSAR Tier 2, Section 3.9.3.1 COL information item will be revised to include ASME Code Class 2 and 3 components required for safe shutdown of the reactor and mitigation of consequences of a postulated piping failure without offsite power, and FSAR Tier 2, Section 3.9.3.1 and FSAR Tier 2, Table 1.8-2 will be revised as indicated in the May 26, 2009, response to RAI 211, Question 03.09.03-21. **RAI 211, Question 03.09.03-21 is being tracked as a confirmatory item.**

In FSAR Tier 2, Section 3.9.3.1, the applicant states that internal parts of components, such as valve discs, seats, and pump shafts, comply with the applicable ASME Code or Code Case criteria. In those instances where no ASME Code criteria exist, these components are designed so that no safety-related functions are impaired. In RAI 107, Question 03.09.03-7, the staff requested that the applicant identify what acceptable codes and standards are used for the design of these kinds of components if no ASME Code criteria exist, and what are the associated design analysis procedures and design criteria. In a December 1, 2008, response to RAI 107, Question 03.09.03-7, the applicant stated that when there are no specific ASME Code criteria for component internal parts (e.g., valve discs, seats, and pump shafts), the applicable industry and supplier standards are used (e.g., ASME QME-1). The applicant also stated that additional information on active pumps and valves is provided in FSAR Tier 2, Section 3.9.3.3. The staff considers the use of applicable industry standards for internal parts of the components as acceptable in cases where no ASME Code criterion exists, since GDC 1 requires that SSCs important to safety be designed to quality standards commensurate with importance of the safety functions to be performed. Therefore, the staff considers RAI 107, Question 03.09.03-7 resolved.

In FSAR Tier 2, Section 3.9.3.1.1, loads and stress criteria are provided for components and component supports. The section did discuss whether components and component supports remained operational and performed a safety function after a specified plant condition event. In RAI 107, Question 03.09.03-10, the staff requested that the applicant provide confirmation that safety-related components and component supports required to remain operational and to perform a safety function after a specified plant condition event are designed to lower ASME Code, Section III service level stress criteria.

In a December 1, 2008, response to RAI 107 03.09.03-10, the applicant stated that, in accordance with SRP Section 3.9.3, functional and operational capability requirements apply only to active components (and their supports) such as pumps, valves, snubbers and ASME Code Class 1, 2, and 3 piping components. Although design of equipment using lower ASME service limits is one acceptable method of demonstrating functional capability of the equipment, it is also acceptable to design to the service limits of ASME corresponding to the specified service condition provided operability and functionality of the component is demonstrated through additional analysis or testing or a combination of these methods. The staff agrees with the applicant's response about the different methods to demonstrate the functionality of the components; however, SRP, Section 1, Appendix A also states that the treatment of the functionality, including collapse and deflection limits, is not adequately treated by the ASME Code for all situations. Such factors should, therefore, be evaluated, and appropriate information should be developed for inclusion in the Design Specification or other referenced documents. In follow-up RAI 404, the staff requested that the applicant provide the specifications for ASME Code Class 1, 2, and 3 components and supports for audit. **RAI 404, Question 03.09.03-25 is being tracked as an open item.**

In RAI 404, Question 03.09.03-25, the applicant stated that the stress evaluation applied by ASME Service Level D requirements is intended to confirm that the pressure retaining integrity is maintained, but is not intended to confirm the operability of components. Pump and valve operability is verified by tests and analysis in accordance with SRP Section 3.10, "Seismic and Dynamic Qualification of Mechanical and Electrical Equipment," Acceptance Criteria II.A. Seismic and environmental qualification is described in FSAR Tier 2, Section 3.10 and FSAR Tier 2, Section 3.11, respectively. FSAR Tier 2, Section 3.9.6 provides details for the Inservice Testing Program for pumps and valves.

Safety-related pump and valve vendors will design against collapse and deflection for faulted conditions. The criteria will be developed on a component-specific basis at the time of procurement and specified in the design specifications.

The staff finds the response dated February 20, 2009, acceptable. The staff will perform an audit of design specifications and to confirm design specifications that contain the design criteria. **RAI 107, Question 03.09.03-10 is being tracked as a confirmatory item.**

FSAR Tier 2, Section 3.9.3.1.1 provides the loadings considered in the design of the components, piping, and support structures. The applicant states that design pressure is described in Topical Report ANP-10264NP-A, Section 3.3 and applies to ASME Code Class 1, 2, and 3 components and piping. The criteria for incorporating the effects of both internal and external pressures for components are described in ASME Code, Section III, Articles NB-3000, NC-3000, and ND-3000. Other pertinent design loadings considered for the U.S. EPR component design are described in the following sections.

Deadweight analyses consider the weight of the component, piping, or structure being analyzed and the additional weight of contained fluid, external insulation, and other appurtenances. For piping and components, the applicant states that the deadweight present during hydrostatic test loadings is also considered where such loadings exceed the normal operational deadweights. In addition, static and dynamic heads of liquid are also included in the deadweight analyses of components. Deadweight loads are further described in Topical Report ANP-10264NP-A, Sections 3.3.1.2 and 6.3.1.

The effects of restrained thermal expansion and contraction on piping and supports are described in Topical Report ANP-10264NP-A, Section 3.3.1.3 and Section 6.3.2. Detailed evaluation of the piping thermal expansion and piping supports is provided in Section 3.12 of this report.

Analyses of seismic inertial loads and anchor movements on piping systems and the RCS are described in Topical Report ANP-10264NP-A, Sections 3, 4, and 6; and FSAR Tier 2, Appendix 3C, respectively. In addition to the inertia and anchor movement stress effects due to a seismic event, the fatigue effects of such cyclic events are considered in the design of ASME Code, Class 1 components and piping. The number of safe shutdown earthquake stress cycles included in the fatigue analysis is identified in FSAR Tier 2, Section 3.7.3, and in Topical Report ANP-10264NP-A, Section 3.4.1.

Analyses of system operating transients, including fluid transient loadings, on piping systems and the RCS are discussed in Topical Report ANP-10264NP-A, Sections 3.3.1.5 and 6.3.4; and FSAR Tier 2, Appendix 3C, respectively. Thermal and pressure transients are described in Topical Report ANP-10264NP-A, Section 3.3.1.8; whereas water and steam hammer loads are described in Topical Report ANP-10264NP-A, Section 3.3.1.5. The analysis of these transients results in force-time histories for application in the piping analyses. Detailed evaluation of the piping analysis is provided in Section 3.12 of this report.

Wind and tornado loads are discussed in Topical Report ANP-10264NP-A, Sections 3.3.1.6, 6.3.5, and 6.3.6. As noted in Topical Report ANP-10264NP-A, should a COL applicant that references the U.S. EPR design certification find it necessary to route ASME Code Class 1, 2, and 3 piping not included in the U.S. EPR design certification so that it is exposed to wind and tornadoes, the design must withstand the plant design-basis loads for this event.

Loads due to pipe breaks are described in Topical Report ANP-10264NP-A, Section 3.3.1.7. Additionally, the leak-before-break methodology is used to eliminate the dynamic effects of pipe rupture for the main coolant loop, pressurizer surge line, and portions of the main steam line piping. Pipe break load design condition and service level evaluations are described in Topical Report ANP-10264NP-A, Sections 6.3.7, 6.3.8, and 6.3.9.

Friction loads are described in Topical Report ANP-10264NP-A, Section 6. In addition, minimum pipe support design loads are described in Topical Report ANP-10264NP-A, Section 6.3.

Minimum design loads are described in Topical Report ANP-10264NP-A, Section 6.3. Normal condition allowable stresses are applicable to the stresses resulting from the described applied loads. Use of this criterion does not eliminate the requirement to analyze supports for applicable service conditions. The minimum pipe support design loads are addressed in SRP Section 3.12. Detailed evaluation of the minimum pipe support design load is provided in Section 3.12 of this report.

In addition, thermal stratification, cycling, and stripping (including applicable NRC Bulletins 79-13, 88-08, and 88-11) are described in Topical Report ANP-10264NP-A, Section 3.7. The pressurizer surge line is analyzed with the main coolant loop piping and supports as described in FSAR Tier 2, Appendix 3C. As noted in the Topical Report ANP-10264NP-A, a COL applicant that references the U.S. EPR design certification will confirm that thermal deflections do not create adverse conditions during hot functional testing. The applicant also states that a COL applicant that references the U.S. EPR design certification will examine the feedwater line welds after hot functional testing prior to fuel loading and at the first refueling outage, in accordance with NRC Bulletin 79-13. A COL applicant that references the U.S. EPR design certification will report the results of inspections to the staff, in accordance with NRC Bulletin 79-13.

In FSAR Tier 2, Section 3.9.3.1.1, the applicant states that the effects of the environment on fatigue for Class 1 piping and components are addressed in FSAR Tier 2, Section 3.12, “ASME Code Class 1, 2, and 3 Piping Systems, Piping Components, and their Associated Supports,” and in Topical Report ANP-10264NP-A, Section 3.4. However, the staff determined that only piping is addressed in the mentioned references. In RAI 107, Question 03.09.03-8, the staff requested that the applicant explain where in the FSAR environment fatigue for ASME Code, Section III components is discussed. In a December 1, 2008, response to RAI 107, Question 03.09.03-8, the applicant stated that FSAR Tier 2, Section 3.12.5.19 will be revised as follows so as to include consideration of fatigue effects on ASME Code, Section III Class1 components: “The effects of reactor coolant environment, using the methodology described in RG 1.207, are considered when performing fatigue analyses for Class1 piping and components. If there are locations in the Class 1 systems where the cumulative usage factor (CUF) cannot be shown to be less than 1.0, based on the methodology described in RG 1.207, alternative methods for addressing environmental fatigue will be applied.” This change is acceptable to the staff, and therefore the staff considers RAI 107, Question 03.09.03-8 resolved.

FSAR Tier 2, Table 3.9.3-1 provides the loading combinations and corresponding stress design criteria for ASME Code Class 1 components; whereas, FSAR Tier 2, Table 3.9.3-2 provides the same information for ASME Code Class 2 and 3 components. Core support structures as described in FSAR Tier 2, Table 3.9.3-3, “Load Combinations and Acceptance Criteria for ASME Core Support Structures,” are reviewed in Section 3.9.5 of this report. In addition, load combinations and stress limits for Class 1, 2, and 3 piping are discussed in Section 3.12 of this report.

In FSAR Tier 2, Tables 3.9.3-1 and 3.9.3-3, load combinations and acceptance criteria for ASME Code Class 1 components and core support structures for primary plus secondary stress intensity category under upset condition are described. In RAI 107, Question 03.09.03-9, the staff requested that the applicant explain why earthquake inertial load is not given as a potential loading in these tables. Also, for faulted condition, the staff asked the applicant to explain why a secondary stress category is not included and why the anchor motion effect of SSE is not considered in the design of ASME Code Class 1 components.

In a December 1, 2008, response to RAI 107, Question 03.09.03-9, the applicant stated that FSAR Tier 2, Tables 3.9.3.-1 and 3.9.3-3 list earthquake inertial load as potential loading in fatigue analysis in ASME Code Class 1 components under upset conditions. As part of the fatigue analysis, the earthquake inertial load is considered while evaluating the maximum primary plus secondary stress intensity range in the upset condition. The applicant's response is acceptable to the staff because FSAR Tier 2, Tables 3.9.3-1 and 3.9.3-3 require that earthquake inertial load used in Level B (upset condition) stress intensity calculations is taken as 1/3 of the peak SSE inertial load or as the peak SSE inertial load. If the earthquake inertial load is taken as 1/3 of the peak SSE inertial load, then the number of cycles to be considered for earthquake loading shall be 300 (the equivalent number of 20 full SSE cycles as derived in accordance with Institute of Electrical and Electronic Engineers (IEEE) Standard (Std) 344-2004, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," Appendix D).

NRC has previously endorsed the use of IEEE Std 344-1987 in RG 1.100, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants," and has not endorsed IEEE Std 344-2004. However, a review of the both revisions of the IEEE Std 344 standard indicates that requirements of Appendix D are identical. In addition, the requirement of 20 full SSE cycles or 300 cycles at 1/3 of the peak SSE conforms to SRP Section 3.7.3.

In a December 1, 2008, response to RAI 107, Question 03.09.03-9, the applicant stated that for faulted conditions, all applicable primary stress intensity limits are considered in accordance with ASME Code Section III. The ASME Code does not require evaluation of secondary stresses for faulted conditions. The mechanical loads on the components at the attached nozzles and supports are developed considering the seismic anchor motion effects of the SSE. These loads are conservatively evaluated against the primary stress intensity criteria of the ASME Code; therefore, seismic anchor motions are considered in the design of the U.S. EPR Class 1 components. The staff has reviewed the applicant's response, and finds it to be acceptable, because FSAR Tier 2, Tables 3.9.3-1 and 3.9.3-3 require that rules of ASME Code, Section III, Appendix F be used for analysis of faulted condition loading. ASME Code Section III, Subsection NF-1310(c) prescribes limits on primary stresses for faulted (Level D) conditions, and that self-relieving (secondary stresses) need not be considered. In addition, since the applicant will determine nozzle loads on the equipment considering seismic anchor motion effects as a part of the SSE load, there is no need to have a separate loading category for seismic anchor motion effect. Therefore, the staff considers RAI 107, Question 03.09.03-9 resolved.

Topical Report ANP-10264NP-A, Table 3-1 and Table 3-2 provide the loading combinations and corresponding stress design criteria per ASME Service Level for ASME Code Class 1, 2, and 3 piping. The safety evaluation of load combinations and stress limits for Class 1, 2, and 3 piping is addressed in Section 3.12 of this report.

In FSAR Tier 2, Section 3.9.3.1.7, "Load Combinations and Stress Limits for ASME Code Class 1, 2, and 3 Component Supports," the applicant states that FSAR Tier 2, Table 3.9.3-4 provides the loading combinations and corresponding stress design criteria per ASME Service Level for ASME Code Class 1, 2, and 3 component supports. ASME Code Section III, Table NF-3131(a)-1 is a cross-reference to various sections of ASME Code, Section III, Subsection NF for allowable stress for specific types of component supports. The applicant also states that the allowable stress criteria are supplemented by RG 1.124 and RG 1.130 for Class 1 linear-type and plate-and-shell-type support structures, respectively. The staff reviewed the NF sections given in Table NF-3131(a)-1 and was not able to identify design stress criteria specifically applicable to snubbers. Therefore, in RAI 107, Question 03.09.03-13, the staff requested that the applicant provide the design stress criteria that are specifically applicable to snubbers and to discuss from where the criteria are referenced.

In a December 1, 2008, response to RAI 107, Question 03.09.03-13, the applicant stated that the snubbers are purchased from a qualified vendor to meet the ASME requirements. The applicant also stated that snubber vendors provide a certified load data sheet that states the design of its snubber meets the requirements of ASME Code, Section III, Subsections NCA and NF. ASME Code Section III, Subsection NF, subparagraph NF-1214, "Standard Supports," provides guidance on the design of snubbers. The design specifications require the snubber vendor to meet the design stress criteria of the applicable ASME Code standards. Furthermore, as described in FSAR Tier 2, Section 3.9.3.4.5, "Use of Snubbers," additional information on snubber supports for piping systems is described in Topical Report ANP-10264NP-A, Section 6.6, which has been approved previously by the staff. The staff determined that the applicant's response did not provide detailed information about the snubber design. In follow-up RAI 178, Question 03.09.03-19, the staff requested that FSAR Tier 2, Section 3.9.3.1.7 be revised to include pertinent design criteria, regardless of which party will be responsible for the final snubber design.

In a February 24, 2009, response to RAI 178, Question 03.09.03-19, the applicant stated that Subparagraph NF-1214 Item (c) lists snubbers as an example of standard supports. The last three rows in Table NF-3131(a)-1 provide cross references to design criteria for standard supports. ASME Code Section III, NF-3400 provides design rules for standard supports, and Table NF-3412.4 specifically addresses snubbers. ASME Code Section III, NF-3400 also requires that ASME Code Section III, NF 3300 be met as the snubber functions as a linear type support. Also, ASME Code Section III, NF-3370 and NF-3380 address the design of linear supports by experimental analysis or load rating methods. The staff has performed a detailed review of FSAR Tier 2, Table 3.9.3-4, Table NF-3131(a)-1, Subarticles NF-3300 and NF-3400, including Subarticles NF-3370 and 3380, and Subparagraph NF-1214 and determined that the design stress criteria for snubbers is adequately described in the FSAR and ASME Code Section NF. Therefore, the staff considers RAI 178, Question 03.09.03-19 resolved.

Topical Report ANP-10264NP-A, Table 6-1 provides the loading combinations and corresponding stress design criteria per ASME Service Level for ASME Code Class 1, 2, and 3 pipe supports. Analysis required establishing piping functionality is addressed in Topical Report ANP-10264NP-A, Section 3.5. The safety evaluation of load combinations and stress limits for

ASME Code Class 1, 2, and 3 piping and piping functionality is addressed in Section 3.12 of this report.

3.9.3.4.2 Design and Installation of Pressure-relief Devices

In FSAR Tier 2, Section 3.9.3.2, the applicant states that the design and installation criteria for pressure-relief devices are described in Topical Report ANP-10264NP-A, Section 3.8. Stress and load combination requirements are provided in Topical Report ANP-10264NP, Tables 3-1 and 3-2. Topical Report ANP-10264NP-A, Section 3.3.1.5.1 discusses relief valve thrust loads. Information on the structural response of the piping and support systems, including dynamic analyses (i.e., response spectrum or time history analyses) or the equivalent static load method is provided in Topical Report ANP-10264NP-A, Section 4.2.

In FSAR Tier 2, Section 3.9.3.2.1, “Class 1 Pressurizer Safety Relief Valves,” the applicant states that the PSRVs and their pilot operators are qualified to operate in saturated steam, water, and steam and water mixtures in hot or cold conditions. They are also designed to operate in hot conditions without electric or I&C inputs and are designed so that the I&C and power supply to the PSRV pilot operator will operate in the event of a single failure during cold shutdown conditions. Details on the design of the PSRVs are provided in FSAR Tier 2, Sections 5.2.2 and 5.4.13, “Safety and Relief Valves.”

The two types of discharge systems for pressure relief devices are described in FSAR Tier 2, Section 3.9.3.2.3, “Pressure Relief Device Discharge System Design and Analysis.” These are open discharge systems and closed discharge systems. An open discharge system discharges fluid directly to the atmosphere or to a vent pipe that is open to the atmosphere. A closed discharge system is hard piped to a distant location or closed tank. ASME Code, Section III, Appendix O describes the layout considerations and limits for both types of systems, as well as design equations and considerations for analysis of these systems. The applicant states that the U.S. EPR design complies with these requirements.

The design and installation of pressure–relief devices is described in FSAR Tier 2, Section 3.9.3.2. However, it does not address the testing requirements as a result of the Three Mile Island (TMI) accident. Therefore, in RAI 107, Question 03.09.03-12, the staff requested that the applicant provide in FSAR Tier 2, Section 3.9.3.2, a detailed description of the tests that are conducted to address the testing requirements in TMI Action Item II.D.1 of NUREG-0737, “Clarification of TMI Action Plan Requirements,” or provide a reference in the FSAR where this is discussed. In a December 1, 2008, response to RAI 107, Question 03.09.03-12, the applicant stated that TMI Action Plan Item II.D.1 concerns relief and safety valve test requirements. Conformance with the TMI requirements, in accordance with 10 CFR 50.34(f), is addressed in FSAR Tier 2, Table 1.9-3, “U.S. EPR Conformance with TMI Requirements (10 CFR 50.34(f)) and Generic Issues (NUREG-0933).” As noted in FSAR Tier 2, Table 1.9-3, Item (2)(x), the test program for reactor coolant system pressure relief valves is described in FSAR Tier 2, Section 3.9.6, Section 5.2.2, and Section 14.2, “Initial Plant Test Program.” The staff finds the response acceptable and therefore, considers RAI 107, Question 03.09.03-12 resolved.

3.9.3.4.3 Pump and Valve Operability Assurance

In FSAR Tier 2, Section 3.9.3.3, the applicant states that ASME Code Class 1 pump and valve design loadings and stress limits are addressed in FSAR Tier 2, Section 3.9.3.1.2, as Class 1 components. Similarly, ASME Code Class 2 and 3 pump and valve design loadings and stress limits are described in FSAR Tier 2, Section 3.9.3.1.3, “Load Combinations and Stress Limits for Class 2 and 3 Components,” as ASME Code Class 2 and 3 components. A list of active safety-related pumps and valves is provided in FSAR Tier 2, Section 3.9.6.

Seismic testing of safety-related active pumps is in accordance with IEEE Std 344-2004 or by an analysis that demonstrates that seismic deflections do not cause the rotor to bind or cause other unacceptable damage to critical pump parts. Details of the seismic qualification of the pumps are provided in Section 3.10 of this report. A detailed discussion of the functional design and qualification provisions and inservice testing programs for safety-related pumps is provided in Section 3.9.6 of this report.

The applicant states that in addition to the factory tests prior to installation and post-installation testing in the plant, a representative sample of each valve type is tested for operability during a simulated plant condition event. The valve is mounted so that it conservatively bounds possible plant mounting orientations. The valve includes operators, limit switches, and pilot valves that are normally attached to the valve in the plant. The details of the seismic qualification are provided in Section 3.10 of this report.

FSAR Tier 2, Section 3.9.3.3 describes the design of active valves and pumps and refers to FSAR Tier 2, Section 3.10 for seismic qualification of the pumps and valves. However, FSAR Tier 2, Section 3.9.3.3 does not clearly identify what are the allowable stresses for the valve bodies and pump casings. Therefore, in RAI 107, Question 03.09.03-11, the staff requested that the applicant confirm that the stresses in active valve bodies and pump casings comply with the requirements in SRP Section 3.10 for faulted conditions. In a December 1, 2008, response to RAI 107, Question 03.09.03-11, the applicant stated that in compliance with the guidance in SRP Section 3.9.3, stresses in active valve bodies comply with the guidance in SRP Section 3.10, where it states that FSAR Tier 2, Section 3.10 describes the methods and criteria for seismic qualification testing of Seismic Category I mechanical equipment. Additionally, FSAR Tier 2, Section 3.9.3.3.1, “Pump Operability,” and Section 3.9.3.3.2, “Active Valve Operability,” contain references to FSAR Tier 2, Section 3.10. The staff finds the applicant response to be acceptable, and therefore, the staff considers RAI 107, Question 03.09.03-11 resolved.

As an alternative, to verify that the valve operability, an equivalent static load representing the faulted load is applied to the top of the bonnet, and the pressure is increased until the valve actuates. A successful actuation within the design setpoint requirements verifies its operational overpressurization capabilities during a plant condition event.

3.9.3.4.4 Component Supports

In FSAR Tier 2, Section 3.9.3.4, the applicant provides design analysis methodologies, as well as acceptance criteria for ASME Code Class 1, 2, and 3 component supports. The applicant incorporates by reference Topical Report ANP-10264NP-A for most of the design considerations of component supports, including review areas such as jurisdictional boundaries, pipe support baseplate and anchor bolt design, use of energy absorbers and limit stops, pipe support stiffness, seismic self-weight excitation, design of supplemental steel, pipe support gaps and clearances, instrumentation line support criteria, pipe deflection limits, load combinations and stress limits for buried piping, and model isolation methods. These are considered acceptable except for revision of Topical Report ANP-10264NP-A, to incorporate the required changes resulting from the previous NRC review.

In FSAR Tier 2, Section 3.9.3.4.5, the applicant discusses the snubber supports for piping systems and provides a description of functional design and IST programs for snubbers. However, sufficient information is not provided for snubber production and qualification test programs. Therefore, in RAI 107, Question 03.09.03-14, the staff requested that the applicant address the following:

- Provide a description of the AREVA snubber production test program and qualification test program, for both mechanical and hydraulic snubbers.
- Provide justification if the production tests do not consider all snubbers in the population.
- Explain the basis of selecting samples for qualification tests, if sampling method is used.
- Discuss the procedures taken to demonstrate the required snubber load ratings.
- Discuss the acceptance criteria used to ensure that the snubber design comply with the specific requirements of ASME Code Section III, Subsection NF.
- Discuss the specific functional parameters (activation level, release rate, drag, dead band, etc.) considered for snubber production and qualification testing.
- Provide the acceptable codes and standards (including editions) used for the snubber production and qualification testing.
- Verify that the production operability testing for large-bore hydraulic snubbers (greater than 50 kips load rating) include the following:
    - A full Service Level D load test to verify sufficient load capacity
    - Testing at the full load capacity to verify proper bleed with the control valve closed
    - Testing to verify that the control valve closes within the specified velocity range
    - Testing to demonstrate that breakaway and drag forces are within the acceptable design limits

In a December 1, 2008, response to RAI 107, Question 03.09.03-14, the applicant stated that the snubber vendor is responsible for the snubber production and qualification test programs in accordance with the applicable ASME Code standards and the AREVA design specifications. The applicant also stated that information on the inservice testing of snubbers is provided in FSAR Tier 2, Section 3.9.6. Noting that the RAI is addressing the design aspects of the snubbers, instead of snubber IST programs, the staff determined the response to be inadequate in resolving the RAI. Therefore, in follow-up RAI 178, Question 03.09.03-20, the staff requested that the applicant provide the additional information requested in RAI 107, Question 03.09.03-14.

In a February 24, 2009, response to RAI 178, Question 03.09.03-20, the applicant addressed the eight questions given in RAI 107, Question 03.09.03-14 as follows:

- In accordance with ASME Code, Section III, a design specification for snubbers is generated. This specification addresses the qualification and production testing for mechanical and hydraulic snubbers, as applicable, in accordance with the guidance of ASME QME-1, Section QDR.

- Production tests address all snubbers in the population.

- Sampling techniques, if used, are in accordance with the guidance of ASME QME-1, Section QDR.

- Snubber load ratings are developed by the manufacturer using testing or analysis, as described in ASME Section III, Subsection NF. Certified design report summaries are provided by the manufacturer to document these load ratings.

- The acceptance criteria for snubber design are discussed in the response to RAI 178, Question 03.09.03-19.

- Functional parameters to be considered for testing are in the design specification, and are based on those identified in ASME QME-1, Section QDR.

- The snubber qualification and production testing is based on ASME QME-1, Section QDR.

- For large bore snubbers of greater than 50 kip capacity, the snubber design verification testing is in accordance with the recommendations of NUREG/CR-5416, “Technical evaluation of Generic Issue 113: Dynamic qualification and testing of large bore hydraulic snubbers.” This information will be added to FSAR Tier 2, Sections 3.9.3.4.5 and 3.9.3.5.

The staff reviewed the applicant’s responses to the eight items given above and finds them to be acceptable based on the justification noted below:

- ASME Section III, Subsection NF, paragraph, NF-3124 refers to ASME QME-1 (2007) for qualification and inservice testing information pertaining to snubbers. In addition, ASME QME-1, Subsection QDR-1100 states that Subsection QDR augments, but does not replace, the requirements of ASME Section III, Subsection NF.

- Snubber procurement specifications, production tests, sampling, load rating and qualification will be performed in accordance with the ASME Section III, Subsection NF, and ASME QME-1, Subsection QDR.

- SRP Section 3.9.3, Section II.3.B.iii.(5) states that specifications for large bore hydraulic snubbers with rated load capacities of 50 Kips or more should contain environmental, structural, and performance design verification tests, including the required dynamic qualification, testing and extrapolation methods supporting qualification in accordance with NUREG/CR-5416.

- The staff finds the responses to RAI 178, Question 03.09.03-20 acceptable; therefore, the staff considers RAI 178, Question 03.09.03-20 resolved.

As stated earlier, FSAR Tier 2, Appendix 3C, describes the structural analysis of the RCS and the RPV internals. The staff reviewed the structural analysis focusing on the adequacy of the modeling and analysis methodology for the ASME Code Class 1 RCS mechanical components. FSAR Tier 2, Appendix 3C, Section 3C.2 states that two mathematical models are developed for use in the structural analysis of the RCS and RPV internals; these are RCS four loop model and RPV isolated model. The RCS four loop model consists of representations of RPV, SGs and SG internals, RCPs and the RCP internals, PZR, and RCS component supports. It also consists of RCS piping, such as main coolant loop and surge line, as well as reactor building internal structures (RBIS). In this RCS four loop model, Loops 1, 2, and 4 contain simplified representations of the RCP and SG (and their internals) and Loop 3 contains detailed representations of the RCP and SG (and their internals). RPV isolated model, on the other hand, contains the detailed representation of the RPV and its internals, as well as the simplified representations of the SGs and RCPs in all four loops of the model.

In FSAR Tier 2, Appendix 3C, Section 3C.2, the applicant describes the modeling of the RCS four loop model. Beam elements are used to represent the pressure boundary of major RCS components (i.e., RPV, SGs, RCP, and PZR) and their internals, linear support elements (i.e., component support columns and SG upper lateral support struts), RPV vertical supports, PZR supports, and piping. Separate elements are used to represent the shear stiffness and the bending stiffness of the RBIS concrete. These beam elements that represent the pressure boundary are assigned distributed mass reflecting the weight of the pressure boundary, entrained fluid, and thermal insulation, and are assigned cross-sectional properties representative of the pressure boundary. Spring elements are used to represent the snubbers on the SGs, the snubbers on the RCPs, the SG lower lateral support bumpers, and the RPV horizontal supports. Rigid members are used to connect RBIS to the building end of the RCS component supports.

Component internals and fuel mass are lumped at the appropriate center of gravity locations. These internals include SG lower internals of tubesheet, tubes, tube support plates, anti-vibration bars, bundle wrappers, and seismic stops; SG upper internals of feedwater headers, platforms, separators, dryers, and supports; and RCP internals of shaft, impeller, bearings, seal packages, RCP motor and RCP motor stand. In addition, the mass of the RPV closure head appurtenances (i.e., control rod drive mechanisms (CRDMs) and closure head equipment (CHE)) is lumped at the top of the upper head. Lumped masses of RBIS are located eccentrically to represent the mass distribution of the physical structure accurately.

RPV is connected to the RBIS through representations of the RPV support ring, which provides support to the RPV primary nozzles. The support ring is represented by vertical beam elements under each primary nozzle and horizontal springs perpendicular to each primary nozzle centerline. The SG models in the four loops are connected to the RBIS through representations of the SG vertical columns, lower lateral support bumpers, upper lateral support snubbers, and upper lateral support struts. The applicant states that RCP models in the four loops are connected to the RBIS through representations of the RCP vertical columns and the lateral support snubbers. Beam elements are used to represent the vertical columns, and spring elements are used to represent the snubbers. In addition, the PZR model is connected to the

RBIS through representations of the PZR support lugs and PZR bumpers. Beam elements are used to represent the support lugs and bumpers.

The applicant states that including the RBIS in this model allows explicit consideration of how the RBIS affects the RCS response to static and dynamic loading, and allows application of the SSE excitations at a single point (i.e., the basemat) in the model.

Because of the gaps at the SG lower lateral support bumpers during operating conditions, the RCS four loop structural model is geometrically non-linear. The effect of this non-linearity to the RCS response is accounted for by a comparison of the dynamic responses with or without consideration of such non-linearity.

This model provides the RCS response (i.e., displacements, loads and accelerations (for dynamic loading only)), to static and dynamic loading at any RCS location, excluding RPV internals locations. The response of the RPV internals to static and dynamic loading is obtained from the RPV isolated structural model, where detailed representations of the RPV internals and RPV closure head appurtenances are included.

In FSAR Tier 2, Appendix 3C, Section 3C.2, concerning mathematical modeling of major components, the applicant states that local flexibilities of the RPV, SG, and RCP shells at the primary nozzle connections, and of the PZR shell at the support lug and lateral bumper connections are accounted for in the model. In RAI 107, Question 03.09.03-15, the staff requested that the applicant discuss how these local flexibilities are formulated for the beam elements representing the components, and how they are included in the mathematical model. In a December 1, 2008, response to RAI 107, Question 03.09.03-15, the applicant stated that the local flexibilities of the component shells at the attachment points are calculated using the design information contained in Welding Journal 34 (12), "Stresses from Radial Loads and External Moments in Cylindrical Pressure Vessels," Research Supplement, 608-s to 617-s (1955); and Welding Journal, 33(12), "Stresses from Radial Loads in Cylindrical Pressure Vessels," Research Supplement, 615-s to 623-s (1954), both by P.P. Bijlaard. The applicant stated that the design information is used to calculate localized deflections and rotations in the shell due to punching force and circumferential and longitudinal bending moments, based on the geometry of the shell and the attachment. The deflections and rotations are used to form a stiffness matrix which is incorporated in the four-loop structural model in BWSPAN (see FSAR Tier 2, Section 3.9.1.2 for a description of the computer code) at the appropriate location. The staff determined that the applicant's response was comprehensive in explaining how local shell flexibilities are incorporated in the finite element stick model in the four-loop dynamic analysis. Therefore, the staff considers RAI 107, Question 03.09.03-15 resolved.

Reactor Pressure Vessel Isolated Model

The RPV isolated model consists of detailed representations of the RPV pressure boundary, CRDMs, CRDM nozzles, CHE, lower internals, upper internals, and fuel assemblies. Beam elements are used to represent the pressure boundary; beam elements and spring elements are used to represent the internals and fuel assemblies to simulate the physical arrangement. A single beam element is used to represent the CRDMs and CRDM nozzles; beam elements are used to represent the CHE linear support elements (i.e., columns, cross braces, beams). The mass of the RPV entrained fluid and thermal insulation is also accounted for in the model.

The SGs, RCPs, PZR, their supports, the interconnecting piping (i.e., MCL and SL) and the RBIS are also included in the RPV isolated structural model. The models for these components, piping, and RBIS are the same as those described in the RCS four loop model, except the simplified representations of the SGs and RCPs are used in all four loops of the RPV isolated model. The models for the RCS component supports are also similar to those described in the RCS four loop model. Again, excluding the SL, piping attached to the primary loop is not included in the model because it meets the decoupling criteria as described in the Topical Report ANP-10264NP-A.

An important aspect of the dynamic analysis of the RPV and internals is the consideration of the hydrodynamic effects of the entrained fluid inside the RPV. Hydrodynamic coupling of the RPV shell to the core barrel (CB) and of the CB to the heavy reflector (HR) is simulated in the RPV isolated model through the following methodologies and considerations:

- Separate finite element models of the RPV, CB and HR structures are created, and the in-air beam mode frequencies are determined for each model.

- Separate finite element models of the annular fluid between the RPV shell and the CB, and between the CB and HR, with rigid boundary conditions at the structural surfaces, are created and the natural frequencies of these fluid cylinders are determined.

- The first bending modes of the two structural models and the modes of the fluid models are coupled, and the frequencies of the coupled fluid-structure models are determined.

- A "virtual mass" matrix, representing the mass necessary to reduce the CB and HR bending frequencies obtained from the in-air values down to the values obtained from the coupled fluid-structure models, is determined.

- The virtual mass matrix is included in the model of the RPV internals to capture the hydrodynamic mass coupling effects between the RPV shell, CB, and HR.

Including the RBIS in this model allows explicit consideration of how the RBIS affects the RPV response to static and dynamic loading, and allows application of the SSE excitations at a single point (i.e., the basemat) in the model. This model provides the RPV response (i.e., displacements, loads and accelerations (dynamic loading only)) to applied static and dynamic loading at any RPV location, including appurtenances and internals.

The staff reviewed the above detailed description of the mathematical models provided by the applicant for RCS major components and RPV internals. The staff finds the modeling approach used by the applicant is in accordance with the acceptable industry practice for the static and dynamic structural response calculations using the finite element method of analysis, and are therefore acceptable.

The RCS four loop model and the RPV isolated model are subjected to the applied static loadings of deadweight, internal pressure, steady state flow, and thermal effects, as well as to the dynamic loadings of high-energy line break (HELB) and SSE, in static and dynamic analyses, based on the recommended loading combinations of SRP Section 3.9.3. Stress and fatigue analyses of the RCS components and the stress analysis of the supports are subsequently performed using the loads obtained from the static and dynamic analyses.

The staff finds that the general approach used by the applicant for the component design analysis adequate.

FSAR Tier 2, Appendix 3C, Section 3C.4.2.2 describes the derivation of the seismic loadings used for the RCS four loop model and RPV isolated model. The applicant states that the seismic design basis of the U.S. EPR, as presented in FSAR Tier 2, Section 3.7.1.3, “Supporting Media for Seismic Category 1 Structures,” includes 12 SSE cases to represent 12 combinations of soil profile and control motions, ranging from soft soil through medium soil to hard rock. The applicant states that the response of the Nuclear Island Common Basemat Structure at the basemat elevation obtained from soil-structure interaction analysis considering these 12 cases serves as input to the seismic analyses of the RCS four loop model and the RPV isolated model. The staff evaluation of the development of the seismic design basis, including the development of the bounding seismic input loadings for the U.S. EPR RCS structural analysis, is provided in Section 3.7 of this report.

The applicant states that the seismic loads needed for the stress and fatigue analysis of the RCS four loop model pressure boundary components are generated through application of pertinent seismic cases, with the effect of the gaps at the SG lower lateral support bumpers considered. Basemat acceleration time histories are used to develop amplified response spectra at points of interest in the RCS. These include branch lines nozzle locations on the RCS primary piping and the MFW line and MS line nozzles on the SGs.

For the RPV isolated model, the applicant states that all 12 SSE cases described in FSAR Tier 2, Section 3.7.1.3, are considered in the seismic analysis of the RPV isolated model. The corresponding bounding seismic loadings are generated using an approach similar to that for the RCS four loop model. The seismic loads to be used for the primary stress analysis of the RPV, its internals, and RPV CHE are generated through the application of these bounding seismic loadings to the RPV isolated model, with consideration of gaps identified at various locations indicated in FSAR Tier 2, Appendix 3C, Section 3C.4.2.2.2, “Reactor Pressure Vessel Isolated Structural Model Seismic Analysis.” Since the RPV isolated model is geometrically non-linear due to these gaps, the direct step-by-step integration time history solution technique with Rayleigh damping is utilized. The seismic loads determined are then combined with other design-basis loads in the stress analyses of the RCS piping, components, internals, and components supports, based on the loading combinations described in Section 3.9.3.3.1 of this report.

In FSAR Tier 2, Appendix 3C, Section 3C.6, “Description of Computer Programs,” the applicant states that the computer codes used for the analysis of the RCS four loop model and RPV isolated model are certified (or verified), controlled, and maintained per administrative procedure. Files are maintained that provide the software author, source code, dated version, program description, extent and limitation of the program application, and the solutions to the test problems described. This is acceptable to the staff, as discussed in its discussion on the acceptance of a computer benchmark program in Section 3.9.1 of this report.

COL Information Item 3.9-2

The staff concerns with COL Information Item 3.9-2 dealing with provision of design specifications after issuance of the design certification are addressed in **RAI 107, Question 03.09.03-1 which is being tracked as an open item**. The staff questioned whether the design reports should be addressed using a COL information item. Procurement will not take place until after the COL is granted, so the staff questioned whether the required action should be included in ITAAC. Therefore, in RAI 156, Question 14.03.03-26, the staff requested that the applicant add an appropriate ITAAC in FSAR Tier 1 to address the issue. **RAI 156, Question 14.03.03-26 is being tracked as an open item.**

COL Information Item 3.9-11

COL Information Item 3.9-11 is discussed in detail above. The question of whether the item should include ASME Code Class 2 and 3 is addressed in RAI 211, Question 03.09.03-21.

In RAI 211, Question 03.09.03-21, the staff requested that the applicant confirm that the existing COL item for ASME Code Class 1 components be revised to include ASME Code Class 2 and 3 components which are required to shutdown the reactor or mitigate consequences of a postulated piping failure without offsite power. In a May 26, 2009, response to RAI 211, Question 03.09.03-21, the applicant stated that FSAR Tier 2, Section 3.9.3.1 COL information item will be revised to include ASME Code Class 2 and 3 components required for safe shutdown of the reactor and mitigation of consequences of a postulated piping failure without offsite power, and FSAR Tier 2, Section 3.9.3.1, and FSAR Tier 2, Table 1.8-2 will be revised as indicated in the May 26, 2009, response to RAI 211, Question 03.09.03-21. **RAI 211, Question 03.09.03-21 is being tracked as a confirmatory item.**

COL Information Items 3.9-3 and 3.9-4

In FSAR Tier 2, Section 3.9.3.1.1, the applicant indicated that a COL applicant referencing the U.S. EPR design certification will examine the feedwater line welds after hot functional testing prior to fuel load in accordance with NRC Bulletin 79-13. Specifically, in FSAR Tier 2, Table 1.8-2, Item No. 3.9-3, the applicant stated that a COL applicant referencing the U.S. EPR design certification will report the results of inspections to the NRC, in accordance with NRC Bulletin 79-13. 10 CFR 52.47(b)(1) states that a design certification application must contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act of 1954, and NRC regulations. The staff understands that the applicant is proposing to have COL applicants (or Holders in this case) address the final resolution of the issue. However, the staff's concern is that COL applicants must address all COL Items whether final action is taken before or after the license is issued. If the information is not provided, COL applicants need to meet RG 1.206 and inform the staff when and how the information will be provided. Given that it is acknowledged that the action will occur during construction, to allow the staff to perform necessary inspection of the report results ensuring the feedwater line welds has been examined, the staff finds that an ITAAC in the FSAR is necessary. Therefore in RAI 388, Question 03.09.03-22, the staff requested that the applicant add an appropriate ITAAC in FSAR Tier 1 to address the issue.

In a July 1, 2010, response to RAI 388, Question 03.09.03-22, the applicant stated that construction will be finished prior to completion of hot functional testing. Therefore, weld inspection after hot functional testing is not a construction issue. FSAR Tier 2, Section 14.2 will be revised to include this inspection as part of the initial test program. FSAR Tier 2, Section 14.2, Test #033 will be revised to also include the feedwater nozzle inspection in accordance with NRC Bulletin 79-13. This inspection will remain a COL Item because the inspection during the first refueling outage will occur after the license is issued.

The staff finds the applicant response acceptable for COL information items3.9-3. **RAI 388, Question 03.09.03-22 is being tracked as a confirmatory item.**

In FSAR Tier 2, Section 3.9.3.1.1, the applicant indicated that a COL applicant referencing the U.S. EPR design certification will confirm that the thermal deflections do not create adverse conditions during hot functional testing. Specifically, in FSAR Tier 2, Table 1.8-2, Item No. 3.9-4, the applicant stated that a COL applicant referencing the U.S. EPR design certification will confirm this issue. According to 10 CFR 52.47(b)(1), a design certification application must contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act of 1954, and NRC regulations.

The staff understands that the applicant is proposing to have COL applicants (or Holders in this case) address the final resolution of the issue. However, the staff concern is that COL applicants must address all COL Items whether final action will be taken before or after the license is issued. If the information is not provided, COL applicants need to meet RG 1.206 and inform the staff when and how the information will be provided. Given that the action will occur during the construction period, to allow the staff to perform necessary review or inspection confirming that the thermal deflections do not create adverse conditions during hot functional testing, the staff finds that an ITAAC in the FSAR is necessary. Therefore in RAI 388, Question 03.09.03-23, the staff requested that the applicant add an appropriate ITAAC in FSAR Tier 1 to address the issue.

In a July 1, 2010, response to RAI 388, Question 03.09.03-23, the applicant stated that in FSAR Tier 1, Table 2.2.1-5, Item 3.9 is an existing ITAAC that verifies gaps during hot functional testing. FSAR Tier 2, Section 14.2 will be revised to include this inspection as part of the initial test program. Specifically, FSAR Tier 2, Section 14.2.12.13.1 will be revised to also include the feedwater line measurements.

The staff finds the applicant response acceptable for COL information item 3.9-4. The applicant will revise FSAR Tier 2, Section 14.2.12.13.1 to include the feedwater line measurements. **RAI 388, Question 03.09.03-23 is being tracked as a confirmatory item.**

3.9.3.5 *Combined License Information Items*

Table 3.9.3-1 provides a list of ASME Code Class 1, 2, and 3 components, component supports, and core support structures related COL information item numbers and descriptions from FSAR Tier 2, Table 1.8-2:

Table 3.9.3-1 U.S. EPR Combined License Information Items

Item No.	Description	FSAR Tier 2 Section

Item No.	Description	FSAR Tier 2 Section
3.9-2	A COL applicant that references the U.S. EPR design certification will prepare the design specifications and design reports for ASME Class 1, 2, and 3 components, piping, supports and core support structures that comply with and are certified to the requirements of Section III of the ASME Code.	3.9.3
3.9-3	A COL applicant that references the U.S. EPR design certification will examine the feedwater line welds after hot functional testing prior to fuel loading and at the first refueling outage, in accordance with NRC Bulletin 79-13. A COL applicant that references the U.S. EPR design certification will report the results of inspections to the NRC, in accordance with NRC Bulletin 79-13.	3.9.3.1.1
3.9-4	As noted in ANP-10264(NP), a COL applicant that references the U.S. EPR design certification will confirm that thermal deflections do not create adverse conditions during hot functional testing.	3.9.3.1.1
3.9-5	As noted in ANP-10264(NP), should a COL applicant that references the U.S. EPR design certification find it necessary to route Class 1, 2, and 3 piping not included in the U.S. EPR design certification so that it is exposed to wind and tornadoes, the design must withstand the plant design-basis loads for this event.	3.9.3.1.1
3.9-11	A COL applicant that references the U.S. EPR design certification will provide a summary of the maximum total stress, deformation (where applicable), and cumulative usage factor values for each of the component operating conditions for ASME Code Class 1 components. For those values that differ from the allowable limits by less than 10 percent, the COL applicant will provide the contribution	3.9.3.1

Item No.	Description	FSAR Tier 2 Section
	of each of the loading categories (e.g., seismic, pipe rupture, dead weight, pressure, and thermal) to the total stress for each maximum stress value identified in this range.	

The staff reviewed the above COL information items and has concerns with the context and wording as addressed above in the Technical Evaluation. These are the open and confirmatory items discussed above.

3.9.3.6 *Conclusions*

Based on its review of the information provided in FSAR Tier 2, Section 3.9.3 and Appendix 3C, the staff concludes that the methodologies presented by the applicant for the design analysis of ASME Code Section III components and their supports are generally acceptable. However, because of pending resolution of the RAIs that remain open, the staff will defer its final conclusion about the U.S. EPR being designed to meet the requirements of 10 CFR Part 50, specifically 10 CFR 50.55a and GDC 1, GDC 2, GDC 4, GDC 14, and GDC 15, until all RAIs are resolved by the applicant.

3.9.4 Control Rod Drive System

3.9.4.1 *Introduction*

The control rod drive system (CRDS) consists of the control rods and related components which provide the means for mechanical movement. In the U.S. EPR design certification, the CRDS consists of the control rod drive mechanisms, and rod cluster control assemblies (RCCAs) with neutron absorber material over the length of the control rods. The CRDMs are equipped with a digital and analog position indication system, so the RCCA is measured over the height of the core by two diverse methods. They use natural air circulation and convention cooling, and are mounted on top of the reactor vessel head.

The staff's review under SRP Section 3.9.4 includes CRDMs up to their interface with the rod cluster control assemblies to ensure reliably controlling reactivity changes under anticipated operational occurrences and postulated accident conditions while maintaining structural integrity during normal and postulated accident conditions.

3.9.4.2 *Summary of Application*

FSAR Tier 1: The applicant has provided a description of the mechanical design features of the reactor coolant system in FSAR Tier 1, Section 2.2.1.

FSAR Tier 2: The applicant has provided a description of the control rod drive system in FSAR Tier 2, Section 3.9.4, "Control Rod Drive System." FSAR Tier 2, Section 14.2 includes a description of test abstracts for control rod drive mechanism testing.

ITAAC: The acceptance criteria described in FSAR Tier 2, Section 3.9.4 provide the basis for the ITAAC used in the following sections:

FSAR Tier 1, Section 2.2.1, Table 2.2.1. The inspections, tests, analyses, and acceptance criteria for individual systems are described in FSAR Tier 1, Chapter 2. This chapter identifies the activities to be performed to verify that the as-built system meets the required design.

FSAR Tier 2, Section 14.2 includes a description of test abstracts for the control rod drive pressure housing.

3.9.4.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 3.9.4 and are summarized below. Review interfaces with other SRP sections can be found in NUREG-0800, Section 3.9.4.

1. GDC 1 and 10 CFR 50.55a, "Codes and Standards," as it relates to the requirements regarding the quality standards to be applied to the CRDMs. Specifically, 10 CFR 50.55a identifies the ASME Code requirements contained in Sections III and Section XI, Code editions, and addenda that must be applied to pressure-retaining portions of the CRDMs that are of the highest importance to safety. The application of GDC 1 and 10 CFR 50.55a requirements to the design, fabrication, installation, and testing ensures the CRDMs meet quality standards that are adequate to provide assurance that these safety functions will be performed.

2. GDC 2, as it relates to the requirements regarding the ability of the CRDMs to withstand the effects of a design-basis earthquake without loss of capability to perform their safety function.

3. GDC 14, as it relates to the requirements for the portion of the CRDMs that form part of the reactor coolant pressure boundary. Application of the GDC 14 criteria to the CRDM components functioning as a part of the RCPB enhances safety by ensuring that the reactor coolant system's pressure boundary will have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture.

4. GDC 26, "Reactivity Control System Redundancy and Capability," as it relates to the requirements regarding the reactivity control systems' redundancy and capability. Application of GDC 26 criteria requires that one of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded.

5. GDC 27, "Combined Reactivity Control Systems Capability," as it relates to the requirements regarding the combined reactivity control systems capability. Requiring compliance with GDC 27 for the CRDMs ensures that the reactivity control systems shall be designed to have a combined capability, in conjunction with the addition of negative reactivity by the emergency core cooling system, of reliably controlling reactivity changes

to assure that under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained.

6. GDC 29, "Protection Against Anticipated Operational Occurrences," as it relates to the requirements regarding the capability of the CRDMs to have a high probability of accomplishing their safety function during anticipated operational occurrences.

Acceptance criteria adequate to meet the above requirements include:

1. RG 1.26, Revision 4, March 2007, identifies acceptable standards to be applied for pressure-retaining portions of the CRDS that are less important to safety but which may contain radioactive material.

2. RG 1.29, March 2007, RG 1.29 describes an acceptable method of identifying and classifying those plant features that should be designed to withstand the effects of the SSE.

3.9.4.4 *Technical Evaluation*

The staff reviewed FSAR Tier 2, Section 3.9.4 in accordance with SRP Section 3.9.4 and RG 1.206. The functional performance of the CRDS was reviewed to confirm that the system is capable of providing a safe shutdown, responding within the acceptable limits during an anticipated operational occurrence (AOO), and preventing or mitigating the consequences of postulated accidents to ensure compliance with the requirements of GDC 1, GDC 2, GDC 14, GDC 26, GDC 27, GDC 29, and 10 CFR 50.55a.

FSAR Tier 2, Section 3.9.4 describes the function of the CRDM and specifies the necessary requirements pertaining to its materials, design, inspection, and testing prior to and during service. The loading combinations and corresponding stress limits for ASME Code design are defined for the design condition, Service Levels A, B, C, and D (also known as normal, upset, emergency, and faulted conditions), and test conditions. FSAR Tier 2, Section 3.9.3.1 includes design and service level loadings with appropriate system operating transients; and FSAR Tier 2, Sections 3.9.3.1.1 through 3.9.3.1.8 define loading combinations for ASME Code Class 1, 2, and 3 components, piping, supports, and core support structures design.

In FSAR Tier 2, Section 3.9.4.3, "Design Loads, Stress Limits, and Allowable Deformations," the design conditions for CRDMs to withstand loading combinations, loading values, and the primary stresses to meet the ASME Code, Section III, Division I, Subsection NB requirements are: (1) Design pressure of 17,478 MPa (2535 psig), (2) Operating pressure of 15.513 MPa (2250 psig), (3) Design temperature of 351.1 °C (664 °F), and (4) Operating temperature of 250 °C (482 °F). The CRDMs are designed and qualified to operate in reactor pressure vessel environment.

In RAI 95, Question 03.09.04-1a, the staff requested that the applicant provide reference(s) that document CRDM qualification to operate in an RPV environment for 60 years. In a November 7, 2008, response to RAI 95, Question 03.09.04-1a, the applicant indicated that the Primary Stress Analysis will provide justification for 60 years design life of CRDM pressure boundary components. This report is based on the ASME B&PV Code requirements for Section III, Class 1 Components, and also on the applicant's 30 year proven U.S. EPR design experience of its operating plants. In FSAR Tier 1, Table 2.2.1-1, the CRDM Pressure Housing is given in the Equipment Description column and categorized as ASME Code Section III. In FSAR Tier 1, Table 2.2.1-5, ITAAC 3.1 states that the equipment given in FSAR Tier 1, Table 2.2.1-1 as ASME Code Section III, will be designed, welded, and hydrostatically tested in accordance with ASME Code Section III. Additionally, in the Acceptance Criteria column of FSAR Tier 1, Table 2.2.1-5, it states that ASME Code Section III Design Reports (NCA-3550) exist and concludes that equipment in FSAR Tier 1, Table 2.2.1-1 as ASME Code Section III, meet ASME Code Section III design requirements. Since FSAR Tier 1 provides the ITAACs which commit to verify that the CRDMs are designed to ASME standards, the staff finds the applicant's response acceptable, and considers RAI 95, Question 03.09.04-1a resolved.

FSAR Tier 2, Section 3.9.4 states that the prototype testing for the CRDS is comprised of performance, stability, and endurance tests. In RAI 95, Question 03.09.04-1b, the staff requested that the applicant provide additional details including status and results of the prototype testing program and the range of environmental conditions that supports it. In a November 7, 2008, response to RAI 95, Question 03.09.04-1b, the applicant provided details of prototype testing which includes performance, stability, and endurance tests. The performance tests verify the adequate performance of the equipment in a range of temperature, pressure, and flow rate conditions. Stability tests ensure proper functioning is achieved over an amount of time in normal conditions. Both of these tests have been completed for the U.S. EPR CRDS design. The endurance tests quantify the amount of time and/or number of steps during which no appreciable mechanical damage possibly altering the correct mechanical behavior is to be expected. The endurance testing is currently being conducted in Germany at the KOPRA test facility for the CRDS design. Therefore, the staff requested that the applicant submit acceptance endurance test reports or provide staff access for its review and/or acceptance.

On April 9, 2009, the staff performed an audit to review the prototype testing program for mechanical adequacy of the CRDM. During the audit, the applicant provided the prototype testing results for the CRDS design which consists of performance, stability, and endurance testing. The performance test results verified the adequacy of the performance of the equipment in a range of temperature, pressure, and flow conditions. The stability test results ensured proper functioning is achieved over time. The endurance test results quantified the number of steps during which no appreciable damage possibly altering the mechanical behavior is expected. The staff determined the test results acceptable; however, in follow-up RAI 245, Question 03.09.04-2b, the staff requested that the applicant provide the basis for enveloping the number of cycles or steps for 60-year design life.

In an October 29, 2009, response to follow up RAI 245, Question 03.09.04-2b, the applicant stated that evaluating the number of steps over the 60-year design life of the plant is based on three parameters: (1) Transients to be analyzed; (2) CRDM steps for a given transient; and (3) number of occurrences of the transient over the 60-year life of the plant. Based on these parameters, the number of steps performed during the endurance test was nine million. For base load units, the CRDM steps expected during 60-year plant life is approximately 750,000 steps. Since the test was conservatively defined and bound the condition expected during the 60-year plant life, the staff finds the basis for enveloping acceptable, and considers RAI 245, Question 03.09.04-2b resolved.

In RAI 95, Question 03.09.04-1c, the staff requested that the applicant provide the technical basis for the statement in FSAR Tier 2, Section 4.6.4, "Information for Combined Performance of Reactivity Systems," that mechanical failure or overheating of the CRDM causes failure of only one RCCA from inserting into the core by gravity, while other CRDMs remain functional. In a November 7, 2008, response to RAI 95, Question 03.09.04-1c, the applicant stated that the mechanical operation of each CRDM is independent of the mechanical operation of the other CRDM, and if a mechanical failure occurs (e.g., latch assembly failure or broken latch), other CRDMs would remain functional. The overheating of the operating coil on an individual CRDM assembly would not prevent other CRDMs from operating, and in case of overheating, if the electrical power was lost to the CRDM, the CRDM would fail in a safe condition. Once power is removed from the operating coils, the latches retract from the drive rod, and the RCCA inserts into the core by gravity. Failure to supply power to the operating coils of the CRDM does not result in a condition that would prevent the rods from inserting into the core. Therefore, the staff considers RAI 95, Question 03.09.04-1c resolved.

FSAR Tier 2, Section 3.9.4 also states that the CRDM pressure housing is constructed in compliance with the requirements of 10 CFR 50.55a, including design, materials, and quality assurance requirements as specified in ASME Code, Section III. FSAR Tier 2, Sections 3.9.4.1.1.3, “Latch Unit,” 3.9.4.1.1.4, “Coil Housing Assembly,” and 3.9.4.3 state that in addition to performance, stability, and endurance testing, each CRDM has a series of production tests performed to verify pressure housing and functional integrity of the CRDM. In RAI 95, Question 03.09.04-1d, the staff requested that the applicant define, “Hydrostatic Test Methods,” for the CRDM housing, and clarify if the hydrostatic test for the connection of rod travel housing to the latch assembly housing is done as part of the system hydrostatic test. In a November 7, 2008, response to RAI 95, Question 03.09.04-1d, the applicant stated that the pressure housing of the CRDM consists of two main sections, the latch unit section and the position indicator section. The CRDM may not be part of the system hydrostatic test at the site. However, the pressure housing of the CRDM will be hydrostatically tested prior to shipping, and installation at the site and the hydrostatic testing will be done in accordance with ASME Section III requirements for Class 1 components. Since ITAAC No. 3.1 in FSAR Tier 1, Table 2.2.1-5 commits to hydrostatic testing in accordance with ASME Code, Section III, the staff finds this ITAAC acceptable, and therefore, considers RAI 95, Question 03.09.04-1d resolved.

FSAR Tier 2, Section 4.6.4 states that in addition to the prototype testing program, functional tests are performed on the CRDMs to verify insertion and withdrawal times in stepping mode, and the drop times meet design requirements. In RAI 95, Question 03.09.04-1g, the staff requested that the applicant clarify if all CRDMs go through the function verification test, and at what stage. In a November 7, 2008, response to RAI 95, Question 03.09.04-1g, the applicant stated that the functional tests are performed on all CRDMs at the manufacturing facility prior to shipping, and as noted in FSAR Tier 2, Section 3.9.4.4, “CRDS Operability Assurance Program,” drop time tests are performed after installation at the reactor site, and therefore, are not recorded at the manufacturing facility. Drop tests are performed at the manufacturing facility to ensure the latch assembly operates and releases as required, but these tests are not timed. However, the function verification tests are conducted at the manufacturing facility prior to shipping of all CRDMs. The applicant revised FSAR Tier 2, Section 3.9.4.4 to clarify that the drop time testing is not conducted as part of functional tests at the manufacturing facility, but are performed to verify the mechanical functioning of the CRDM. Therefore, the staff considers RAI 95, Question 03.09.04-1g resolved.

3.9.4.5 *Combined License Information Items*

There are no COL information items for this section identified in FSAR Tier 2, Table 1.8-2, "U.S. EPR Combined License Information Items."

3.9.4.6 *Conclusions*

Based on the staff's review of the design information provided in FSAR Tier 2, Section 3.9.4, additional supporting sections, documents provided by the applicant, and for the reasons set forth above, the staff concludes that the design of the CRDS is structurally adequate and provides a reliable means of movement of the control rod assemblies within the reactor core under conditions of normal plant transients or under postulated accident conditions.

The review has determined the adequacy of applicant's proposed design criteria, design basis, safety classification of CRDS, and the requirements for providing a safe shutdown during normal operation, anticipated operational occurrences, and accident conditions. The staff also, concludes that the design of the U.S. EPR CRDS is acceptable and meets the requirements of 10 CFR 50.55a (Codes and Standards), and GDC 1, GDC 2, GDC 14, GDC 26, GDC 27, and GDC 29.

3.11 Environmental Qualification of Mechanical and Electrical Equipment

3.11.1 Introduction

Mechanical, electrical, and I&C, including digital I&C equipment designated as important to safety is addressed in the environmental qualification (EQ) program to verify it is capable of performing its design functions under all normal environmental conditions, anticipated operational occurrences, and accident and post-accident environmental conditions.

FSAR Tier 2, Section 3.11.1.2, "Definition of Environmental Conditions," defines service condition environments (harsh and mild) and identifies the equipment that is within the scope of 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants." Included in FSAR Tier 2, Section 3.11 is a description of the approach used by the applicant to environmentally qualify electrical, mechanical, and I&C (including analog and digital) equipment. Harsh environment is an environment resulting from a design basis event (i.e., loss-of-coolant accident, high-energy line break, and main-steam line break). Mild environment is an environment that would at no time is significantly more severe than the environment that would occur during plant operation, including anticipated operational occurrences.

The objectives of the staff's review are to confirm that the applicant's environmental qualification program for electrical and I&C equipment meets the requirements in 10 CFR 50.49, and that the set of equipment to be environmentally qualified includes safety-related equipment, non-safety-related equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of specified safety functions, and instrumentation to monitor parameters specified in RG 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants."

The objective of the staff's review is also to confirm that the applicant's environmental qualification program for safety-related and important to safety mechanical equipment complies with the requirements in 10 CFR Part 50, Appendix A, GDC 4, "Environmental and Dynamic Effects Design Bases."

3.11.2 Summary of Application

FSAR Tier 1: FSAR Tier 1 requirements for environmental qualification of mechanical, electrical, and I&C equipment are contained in 30 sections of FSAR Tier 1, Chapter 2, "System Based Design Descriptions and ITAAC Table of Contents." To avoid unnecessary repetition, these are enumerated below in the discussion of ITAAC. The FSAR Tier 1 requirements are those pertaining to the qualification for the environmental variables specified in 10 CFR 50.49(e) and those pertaining to qualification for electromagnetic compatibility (EMC). Some of the FSAR Tier 1 sections that contain requirements for environmental qualification of electrical, mechanical, and I&C equipment for the process variables specified in 10 CFR 50.49(e) also contain requirements for qualification to determine EMC.

FSAR Tier 2: The applicant has provided an FSAR Tier 2 description in FSAR Tier 2, Section 3.11, "Environmental Qualification of Mechanical and Electrical Equipment," summarized here, in part, as follows:

The applicant's approach to environmental qualification of electrical and I&C equipment must meet the applicable requirements of 10 CFR Part 50 Appendices A and B, and 10 CFR 50.49. In addition, with regard to 10 CFR Part 50, Appendices A and B, the applicant states that their qualification program meets the requirements of 10 CFR Part 50, Appendix A, GDC 1, GDC 2, GDC 4; and GDC 23; and 10 CFR Part 50, Appendix B, Criteria, III, "Design Control"; Criteria XI, "Test Control"; and Criteria XVII, "Quality Assurance Records." The applicant defines the scope of equipment for which qualification is required to include equipment essential for emergency reactor shutdown, core cooling, containment isolation, containment and reactor heat removal, and any equipment necessary to prevent a significant radioactive release to the environment. Also provided in the application is the process used by the applicant to qualify equipment identified in the list.

The applicant has also provided FSAR Tier 2, Appendix 3D, "Methodology for Qualifying Safety-Related Electrical and Mechanical Equipment," to describe the U.S. EPR environmental qualification program for qualifying electrical, mechanical, and I&C equipment in FSAR Tier 2, Section 3.11.

Environmental Qualification of Electrical and I&C Equipment

The applicant has identified areas of the plant that could be subjected to a harsh environment following an accident. Further, the information in FSAR Tier 2 includes a tabulation of plant equipment by equipment tag number, the area in which the equipment is located, and whether the area in which the equipment is located could be subjected to a harsh environment. FSAR Tier 2, Table 3.11-1, "List of Environmentally Qualified Electrical/I&C Equipment," of FSAR Tier 2, Section 3.11 includes a detailed listing by equipment tag number of electrical and I&C equipment located in an environmental harsh or radiation harsh environment that requires qualification. In addition, FSAR Tier 2, Table 3.11-2, "List of U.S. EPR Important to Safety

Systems Screened for the EQ Program," provides a listing of systems that are screened for inclusion in the EQ Program.

Although qualification of equipment located in mild environments is not discussed in FSAR Tier 2, Section 3.11 for electrical components, those used in digital I&C systems and located in a mild environment were included in the EQ program for EMC, where it involves testing to assure that electromagnetic interference (EMI) and radio frequency interference (RFI) would not adversely affect those I&C equipment.

Environmental Qualification of Mechanical Equipment

The applicant has described a methodology for the environment design and qualification of mechanical equipment in both harsh and mild environments. Mechanical equipment experiences the same environmental conditions as those defined in 10 CFR 50.49 for electrical equipment, and such conditions are used in qualifying mechanical equipment. Metallic components that form a pressure boundary are qualified by the nature of their pressure retention capability as demonstrated by the application of an ASME B&PV Code stamp. For mechanical equipment, the primary focus is on nonmetallic materials that are sensitive to environmental effects (e.g., seals, gaskets, lubricants, fluids for hydraulic systems, and diaphragms) needed for safety-related functions and to verify that the design of such materials, parts, and equipment is adequate. Nonmetallic materials are designed to meet the applicable environmental and service conditions. Nonmetallic materials located in harsh environments are qualified in accordance with QME-1-2007, Appendix QR-B, "Guide for Qualification of Nonmetallic Parts." Maintenance and surveillance programs to be developed by the COL applicant provide assurance that equipment qualification will be maintained during the operational life of the plant.

ITAAC: As discussed in the summary of FSAR Tier 1, a table is included below of all the ITAAC that pertain to environmental qualification of electrical, mechanical, and I&C equipment. It should be noted that FSAR Tier 1, Section 2.4 contain ITAAC that require the applicant to demonstrate that the subject equipment is qualified for EMC, as well as for the process variables specified in 10 CFR 50.49(e), (e.g., temperature, pressure, humidity, radiation, and chemical effects).

Table 3.11-1 U.S. EPR ITAAC Related to Environmental Qualification of Electrical and Mechanical Equipment

Tier 1 Section Number	Table Number	Commitment Wording
2.2.1	2.2.1-2/3	6.1, 6.2
2.2.2	2.2.2-2	6.1
2.2.3	2.2.3-2	6.1
2.2.4	2.2.4-2	6.1
2.2.5	2.2.5-2	6.1
2.2.6	2.2.6-2	6.1
2.2.7	2.2.7-2	6.1

Tier 1 Section Number	Table Number	Commitment Wording
2.3.1	2.3.1-2	6.1
2.3.3	2.3.3-2	6.1
2.4.1	2.4.1-1	4.10
2.4.2	2.4.2-1	4.4
2.4.4	2.4.4-1	4.1
2.4.5	2.4.5-1	4.3
2.4.11	2.4.11-1	4.2
2.4.13	2.4.13-1	4.1
2.4.14	2.4.14-1	6.1
2.4.17	2.4.17-1	6.1
2.4.19	2.4.19-1	5.1
2.4.22	2.4.22-1	6.1
2.6.3	2.6.3-2	6.1
2.6.4	2.6.4-2	6.1
2.6.6	2.6.6-2	6.1
2.6.8	2.6.8-3	6.1
2.7.1	2.7.1-2	6.1
2.7.2	2.7.2-2	6.1
2.7.3	2.7.2-2	6.1
2.7.5	2.7.5-2	6.1
2.8.1	2.8.1-2	6.1
2.8.2	2.8.2-2	6.1
2.8.6	2.8.6-2	6.1
2.8.7	2.8.7-2	6.1
2.9.3	2.9.3-2	6.1
2.9.5	2.9.5-1	5.1
3.5	3.5-2	6.1, 6.2

3.11.3 Regulatory Basis

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 3.11 and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Section 3.11.

1. 10 CFR 50.49, as it relates to the applicant establishing a program for qualifying electrical equipment important to safety located in a harsh environment.

2. 10 CFR Part 50, Appendix A, GDC 1, as it relates to components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed.

3. 10 CFR Part 50, Appendix A, GDC 2 as it relates to components important to safety be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety function.

4. 10 CFR Part 50, Appendix A, GDC 4, as it relates to components important to safety be designed to accommodate the effects of, and be compatible with, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss of coolant accidents (LOCAs).

5. 10 CFR Part 50, Appendix A, GDC 23, as it relates to protection systems be designed to fail in a safe state, or in a state demonstrated to be acceptable on some other defined basis, if conditions such as postulated adverse environments (e.g., extreme heat or cold, pressure, steam, water, or radiation) are experienced.

6. 10 CFR Part 50, Appendix B, Section III, as it relates to measures be established to ensure that applicable regulatory requirements and the associated design bases are correctly translated into specifications, drawings, procedures and instructions.

7. 10 CFR Part 50, Appendix B, Section XI, as it relates to a test control plan be established to ensure that all tests needed to demonstrate a component's capability to perform satisfactorily in service be identified and performed in accordance with written procedures that incorporate the requirements and acceptance limits contained in applicable design documents.

8. 10 CFR Part 50, Appendix B, Section XVII, as it relates to sufficient records be maintained to furnish evidence of activities affecting quality.

Acceptance criteria adequate to meet the above requirements includes:

1. RG 1.89, “Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants,” provides the principal guidance for implementing the requirements and criteria of 10 CFR 50.49 for environmental qualification of electrical equipment that is important to safety and located in a harsh environment.

2. RG 1.40, “Qualification Tests of Continuous-Duty Motors Installed inside the Containment of Water-Cooled Nuclear Power Plants,” endorses IEEE Std 334, “IEEE Standard for Qualifying Continuous Duty Class 1 Motors for Nuclear Power Generating Stations.”

3. RG 1.63, “Electrical Penetration Assemblies in Containment Structures for Nuclear Power Plants,” endorses IEEE Std 317, “IEEE Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations.”

4. RG 1.73, “Qualification Tests of Electric Valve Operators Installed inside the Containment of Nuclear Power Plants,” endorses IEEE Std 382, “IEEE Trial Use Guide for Type Test of Class 1E Electric Valve Operators for Nuclear Power Generating Stations.”

5. RG 1.97, “Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants,” provides guidance acceptable to the staff for the environmental qualification of the post-accident I&C monitoring equipment.

6. RG 1.211, “Qualification of Safety-Related Cables and Field Splices for Nuclear Power Plants” replaces RG 1.131, “Qualification Tests of Electric Cables and Field Splices for Light-Water-Cooled Nuclear Power Plants,” endorses IEEE Std 383-2003, “Standard for Type Test of Class 1E Electric Cables and Field Splices for Nuclear Power Generating Stations.”

7. RG 1.152, “Criteria for Digital Computers in Safety Systems of Nuclear Power Plants.”

8. RG 1.156, “Environmental Qualification of Connection Assemblies for Nuclear Power Plants,” endorses IEEE Std 572, “IEEE Standard for Qualification of Class 1E Connection Assemblies for Nuclear Power Generating Stations.”

9. RG 1.158, “Qualification of Safety-Related Lead Storage Batteries for Nuclear Power Plants,” endorses IEEE Std 535-1986, “IEEE Standard for Qualification of Class 1E Lead Storage Batteries for Nuclear Power Generating Stations.”

10. RG 1.180, “Guidelines for Evaluating Electromagnetic and Radio- Frequency Interference in Safety-Related Instrumentation and Control Systems,” provides guidance acceptable to the staff for determining EMC for I&C equipment during service.

11. RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” provides guidance acceptable to the staff for determining the radiation dose and dose rate for equipment during postulated accident conditions.

12. RG 1.209, “Guidelines for Environmental Qualification of Safety-related Computer Based Instrumentation and Control Systems in Nuclear Power Plants,” provides guidance acceptable to the staff for determining the environmental qualification procedures for safety-related computer-based I&C systems for service within nuclear power plants.

13. NUREG-0588, “Interim Staff Position on Environmental Qualification of Safety Related Electrical Equipment,” Category I guidance may be used if relevant guidance is not provided in RG 1.89.

14. Staff Requirements Memorandum (SRM)-SECY-05-0197, “Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria,” provides the use of a license condition for operational program implementation milestones that are fully described or referenced in the final safety analysis report.

3.11.4 Technical Evaluation

3.11.4.1 *Environmental Qualification of Electrical and I&C Equipment*

The staff reviewed FSAR Tier 2, Section 3.11 that describes the applicant's approach for satisfying 10 CFR 50.49 requirements pertaining to the EQ of equipment located in a harsh environment and identifies equipment that is within the scope of 10 CFR 50.49. The review evaluates whether the applicant's information presented in the FSAR Tier 2, Section 3.11 is sufficient to support the conclusion that all items of equipment that are important to safety are capable of performing their design safety functions under: (1) Normal environmental conditions (e.g., startup, operation, refueling, shutdown); (2) anticipated operational occurrences (e.g., plant trip and testing); (3) design-basis accidents (e.g., LOCA and high-energy line break) and post-accident environmental conditions.

The specific equipment scoped in for the EQ review is mechanical, electrical, and I&C, including digital I&C equipment associated with systems that are (1) essential to emergency reactor shutdown, containment isolation, reactor core cooling, and containment and reactor heat removal, or (2) otherwise are essential in preventing significant release of radioactive material to the environment. The scoped equipment includes:

- Equipment that initiates the above functions automatically
- Equipment that is used by the operators to initiate the above functions manually
- Equipment whose failure can prevent the satisfactory accomplishment of one or more of the above safety functions
- Safety-related and non-safety-related electrical equipment
- Certain post-accident monitoring (PAM) equipment

The approach described in SRP Section 3.11 requires compliance with the following relevant requirements:

- 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants"
- 10 CFR Part 50, Appendix A, GDC 1, "Quality Standards and Records"
- 10 CFR Part 50, Appendix A, GDC 2, "Design Basis for Protection Against Natural"
- 10 CFR Part 50, Appendix A, GDC 4, "Environmental and Dynamic Effects Design Bases"
- 10 CFR Part 50, Appendix A, GDC 23, "Protection System Failure Modes"
- 10 CFR Part 50, Appendix B, "Quality Assurance Criteria," Criteria III, XI, and XVII

3.11.4.1.1 Compliance with 10 CFR 50.49

Compliance with 10 CFR 50.49 requires that the applicant establish an EQ program for qualifying electrical equipment important to safety located in a harsh environment. The program ensures that equipment will be able to perform acceptably during all anticipated operating conditions, even after being degraded due to exposure to service conditions during its qualified life. The methodology used to develop the EQ program is described in FSAR Tier 2, Section 3.11, Appendix 3D, "Methodology for Qualifying Safety-Related Electrical and Mechanical Equipment."

Once the applicant identifies all equipment in the scope of the EQ review was identified, screening was performed to establish the EQ list for electrical and I&C equipment, based on the guidelines provided according to provisions 10 CFR 50.49(b)(1), (b)(2), and (b)(3) where:

- 10 CFR 50.49(b)(1) − safety-related electrical equipment that is relied on to remain functional during and after design-basis events to ensure that certain functions are accomplished

- 10 CFR 50.49(b)(2) − non-safety-related electrical equipment whose failure under the postulated environmental conditions could prevent satisfactory performance of the safety functions of the safety-related equipment

- 10 CFR 50.49(b)(3) − certain post-accident monitoring equipment and RG 1.97

The applicant explained that a three step approach was used: (1) The first step in this process screens the SSCs to determine the equipment safety function, based on the plant safety analysis and the regulatory definition of safety-functions to identify the safety-related equipment. (2) The next step determines and screens equipment that is not safety related, but whose failure could prevent the performance of a safety function. This equipment is identified as "NS-AQ" (i.e., non-safety, but having augmented quality). (3) The third step screens and determines PAM equipment that is required to monitor in accordance with RG 1.97.

The equipment screened for the qualification is located in three plant areas: Reactor, Safeguard, and Fuel Buildings in the Nuclear Island (NI); Switchgear and Turbine Buildings in Turbine Island (TI); and the balance-of-plant (BOP) areas. The Reactor Building (RB) within NI is considered an environmental and radiation harsh area. Equipment in the RB, which is within the scope of 10 CFR 50.49, requires consideration for the environmental stressors such as temperature, radiation, pressure, humidity, moisture, steam, water immersion, and chemicals. The applicant stated that equipment in the TI and BOP Buildings are considered to be in a mild temperature, pressure, and radiation environmental zone.

The environmental conditions considered for the EQ program include normal, anticipated operational occurrences, and accident and post-accident environments due to design-basis events (DBEs). The applicable environmental parameters include pressure, radiation, temperature, chemical spray, humidity, submergence, aging, margins, and synergistic effects in specific plant building and room locations. Service conditions are the actual environmental, physical, mechanical, electrical, and process conditions experienced by equipment during service. Plant operation includes both normal and abnormal operations. Abnormal operation occurs during plant transients, system transients, natural phenomena, or in conjunction with certain equipment or system failures. The service condition falls into two general categories:

(1) harsh and (2) mild environments (see FSAR Tier 2, Section 3.11, Figure 3.11-1). FSAR Tier 2, Section 3.11.1.2 defines harsh environments as plant areas where the environmental conditions significantly exceed the normal design (service) conditions as a direct result of a DBE. An environmentally harsh environment is a location (inside or outside containment) that is subject to a break in the reactor coolant system, steam, or other HELB piping that significantly alters the environmental parameters of temperature, pressure, humidity, chemistry, and/or flooding. A radiation harsh environment is a location inside or outside containment where the radiation levels exceed the following thresholds:

- Greater (>) than 10^4 Rads gamma for mechanical equipment including non-metallics or consumables (e.g., O-rings, seals, packing, gaskets, lube oil, diaphragms)

- Greater (>) than 10^3 Rads gamma for electronic devices and components

Mild environments are defined plant areas where the environment at no time would be significantly more severe than the environment that would occur during normal plant operation, including anticipated operational occurrences. FSAR Tier 2, Appendix 3D, Table 3D-1, “Typical Mild Environmental Parameter Limits,” provided typical mild environmental parameter limits on temperature, pressure, humidity, radiation, chemical spray, and submergence.

In RAI 96, Question 03.11-2, Part (i), the staff requested that the applicant clarify why FSAR Tier 2, Section 3.11.1.2 defined areas with radiation levels of 10^4 rads (10^3 rads for electrical) as both harsh and mild radiation environments for the purposes of equipment qualification. In an October 17, 2008, response to RAI 96, Question 03.11-2, Part (i), the applicant explained that this statement was incorrect. The staff confirmed that Revision 1 of the FSAR, dated May 29, 2009, contains the changes committed to in the RAI response (i.e., to indicate that areas with radiation levels greater than 10^4 rads (10^3 rads for electrical) will be defined as harsh radiation environments for the purposes of equipment qualification). Accordingly, the staff finds that the applicant has adequately addressed this issue and, therefore, considers RAI 96, Question 03.11-2, Part (i) resolved.

In RAI 393, Question 03.11-36, the staff requested that the applicant correct an apparent inconsistency between FSAR Tier 2, Section 3.11.1.2 and FSAR Tier 2, Appendix 3D, Section 3.11, Table 3D-1, concerning the equipment identified for a harsh radiation level as “greater than 1.0E03 Rads gamma for electrical or digital equipment,” while FSAR Tier 2, Appendix 3D, Section 3.11, indicated as “greater than 1.0E03 Rads gamma electronic devices and components.” In a June 8, 2010, response to RAI 393, Question 03.11-36, the applicant corrected FSAR Tier 2, Section 3.11.1.2 to coincide with FSAR Tier 2, Appendix 3D, Section 3.11, Table 3D-1 as “greater than 1.0E03 Rads gamma for electronic devices and components.” Therefore, the staff considers RAI 393, Question 03.11-36 resolved.

In RAI 249, Question 03.11-9, the staff requested that the applicant explain why FSAR Tier 2, Section 3.11 does not contain any discussion on synergistic effects when those effects are believed to have a significant effect on electrical, mechanical and I&C equipment qualification programs, as required per 10 CFR 50.49(e)(7). In a July 10, 2009, response to RAI 249, Question 03.11-9, was revised in Revision 2 to include a statement on synergistic effects when its effects are determined to have significant effect on equipment performance. The staff has reviewed the changes in FSAR Tier 2, Revision 2, Section 3.11.1.2, and finds that revisions were made by the applicant as committed in the July 10, 2009, response to RAI 249, Question 03.11-9. Therefore, the staff considers RAI 249, Question 03.11-9 resolved.

For the equipment that is required to be environmentally qualified for its safety functions, such as reactor trip, engineered safeguards actuation, post-accident monitoring, or containment isolation, the U.S. EPR environmental design assigned a period of required post-accident operability duration as: Immediate operability (2 hours); short-term (24 hours); medium-term (4 months); or long-term (1 year) for each safety function.

FSAR Tier 2, Section 3.11.1.3, “Equipment Operability Times,” elaborated that: (1) The immediate operability includes components that must be operational for a maximum of 2 hours after onset of the event; (2) the short-term operability includes components that must remain operational for a maximum of 24 hours after onset of the event; (3) the medium-term operability includes replacement, repair, or recalibration of equipment accessible outside containment or inaccessible instrumentation inside containment required for post-accident monitoring; and (4) the long-term operability includes equipment needed to operate for the entire duration of the accident, as well as into the start of the recovery phase. With the above identified a period of operability, the staff can ensure applicable equipment’s functionality for the appropriate length of time (i.e., during and following a design basis accident). The staff finds that the applicant’s identification of a period of operability of components is consistent with the requirements of 10 CFR 50.49(d)(1), which requires appropriate performance specifications under DBA conditions.

FSAR Tier 2, Section 3.11 also stated that the qualification time is established based on a conservative estimate that conforms to the analyses of when and for how long the component is required to function, plus margin, per IEEE Std 323-1974, “IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations.” In RAI 249, Question 03.11-5, the staff requested that the applicant change the ITAAC commitment wordings for ITAAC Items 6.1 and 6.2 in the FSAR Tier 1, ITAAC Tables in Sections 2.2, 2.3, 2.6, 2.7, 2.8, and 3.5 pertinent to EQ testing to state that equipment given in a harsh environment is required to be functional “before and during design basis accidents.” The existing wording states from that specified in 10 CFR 50.49(d)(1) which requires the equipment to remain functional “during and following design basis events.” In a July 10, 2009, response to RAI 249, Question 03.11-5, the applicant stated that the July 24, 2009, response to RAI 182, Question 14.03-10, Part H has corrected all affected EQ ITAAC verbiages from “before and during design basis accidents” to “during and following design basis events” to be consistent with 10 CFR 50.49(d)(1). The staff has reviewed FSAR Tier 1, Revision 2 and finds that all revisions were made by the applicant as committed to in the July 10, 2009, response to RAI 249, Question 03.11-5. Therefore, the staff considers RAI 249, Question 03.11-5 resolved.

In RAI 249, Question 03.11-6, the staff requested that the applicant discuss how margins were applied for the FSAR Tier 2, Appendix 3D, Table 3D-3 for the EQ test profiles. In a July 10, 2009, response to RAI 249, Question 03.11-6, the applicant referred to FSAR Tier 2, Appendix 3D, Section 3D.4.8 which states, “Table 3D-3, EQ Program Margin Requirements, presents the margins for various environmental parameters. The margins shown in the table are those recommended in IEEE Standard 323-1974.” The applicant also stated that the service conditions for the U.S. EPR, by definition, do not include margin. However, these margins (e.g., production variations and inaccuracies in test instruments) are required to be added by the vendors to the service conditions when conducting the qualification activities for equipment being qualified. The staff reviewed Equipment Qualification Data Package (EQDP), Appendix 3D, Attachment A, where the equipment required to address how margins were applied based on IEEE Std 323-1974, and also to elaborate how much margins were applied to account for local environmental conditions cited in 10 CFR 50.49(e)(8). The staff finds the applicant has adequately addressed this issue and, therefore, considers RAI 249, Question 03.11-6 resolved.

For post accident operability durations, FSAR Tier 2, Appendix 3D, Figures 3D-1 through 3D-6 provided typical combined loss of coolant/steam line break (LOCA/SLB) EQ curves for inside and outside containment temperature and pressure envelopes. For accident EQ radiation dose, FSAR Tier 2, Appendix 3D, Table 3D-9 provided accident cumulative radiation dose for each post-accident operability duration and various plant locations. In RAI 249, Question 03.11-14, the staff requested that the applicant provide updated temperature and pressure curves shown on FSAR Tier 2, Appendix 3D, Figures 3D-1 through 3D-2, according to the method that was described by the staff (RAI 209, Question 06.02.01-14) that uses a multi-node containment model method. Should the results show significant peak temperature and pressure changes, those aforementioned curves should be revised in FSAR Tier 2, Appendix 3D, Figures 3D-1 through 3D-6. Since the staff has not yet seen the final updated curves for the above figures in FSAR Tier 2, Appendix 3D, **RAI 249, Question 03.11-14 is being tracked as an open item.**

By using the aforementioned screening process based on 10 CFR 50.49 criteria, the final step resulted in the equipment being "screened in" and placed in the EQ list, FSAR Tier 2, Table 3.11-1 for electrical and I&C equipment located in an environmental harsh or radiation harsh environment that requires qualification. FSAR Tier 2, Table 3.11-1 also includes an extensive list of PAM equipment. The information consists of: Equipment tag number of electrical and I&C equipment, location, environment condition (environmental harsh or radiation harsh); designated functions (reactor trip, engineering safeguards, PAM, seismic categories); safety class (S-safety related, NS-AQ-Non-safety augmented quality, Class 1E, EMC, C/NM-Consumables/Non-metallics); and EQ program designation (EQ electrical, EQ radiation-electrical, EQ radiation harsh-consumables, etc). In addition, FSAR Tier 2, Table 3.11-2 provides a list of important to safety systems that are screened for inclusion in the EQ Program.

In RAI 96, Question 03.11-3, the staff requested that the applicant explain why certain electrical equipment which required seismic EQ qualification is given in FSAR Tier 2, Table 3.11-1, but not in FSAR Tier 2, Table 3.10-1. In an October 17, 2008, response to RAI 96, Question 03.11-3, the applicant explained that FSAR Tier 2, Table 3.10-1 includes only that electrical equipment that had been identified as requiring only seismic EQ qualification. Electrical equipment that requires additional EQ qualification (such as for electromagnetic compatibility, radiation harsh environment, or full EQ electrical), is given in FSAR Tier 2, Table 3.11-1. Since the applicant clarified the difference between Tables 3.10-1 and 3.11-1, the staff finds that the applicant adequately addressed the issue. Therefore, the staff considers RAI 96, Question 03.11-3 resolved.

For qualification test results, a sample format of the summaries and results of qualification tests for electrical equipment and components in the harsh environment areas is documented as FSAR Tier 2, Appendix 3D, Attachment A, "Equipment Qualification Data Packages (EQDP)." For seismic qualification tests for electrical and mechanical equipment and components, a sample format is documented as FSAR Tier 2, Appendix 3D, Attachment F, "Seismic Qualification Data Packages (SQDP)." Qualification of electrical and mechanical equipment located in a mild environment is based on the manufacturer's certificates of conformance to the appropriate engineering specifications.

In RAI 317, Question 03.11-15, the staff requested that the applicant provide the basis for the proposed deletion of the humidity design parameter for the U.S. EPR. With no humidity parameter limit, the humidity level could reach over 100 percent for the equipment in the main control room (MCR), where condensation could form water and damage I&C equipment. In an October 29, 2009, response to RAI 317, Question 03.11-15, the applicant stated that the MCR and other rooms within the Control Room Envelope (CRE) are cooled by the control room air conditioning system (CRACS). Subsequently, FSAR Tier 2, Section 9.4.1.1 will be revised to indicate that the relative humidity within the MCR and the CRE I&C/Computer Rooms is maintained at or above 20 percent and less than or equal to 70 percent. The staff has reviewed the changes in Revision 2 of the FSAR Tier 2 and revised humidity parameter limits of 30-60 percent by the applicant as committed to in the October 29, 2009, response to RAI 317, Question 03.11-15. Since the new humidity level is much lower than 100 percent, the staff considers RAI 317, Question 03.11-15 resolved.

The borated water spray can affect equipment operation. Thus, the applicant has reviewed the effects of chemical spray that affected equipment during normal and abnormal operating conditions. FSAR Tier 2, Section 3.11 stated that chemical spray is not used to mitigate design basis events (DBEs) for the U.S. EPR design. In fact, the chemical spray is only used for a severe accident heat removal system (SAHRS) to prevent the pressure and temperature within the containment from exceeding design limits. However, given the deliberate steps required to initiate and actuate the SAHRS, inadvertent actuation of this system is not considered as a credible event. The use of chemicals has been only considered for pH control and their effects. Containment sump pH is adjusted to the range of 7.0 and higher if the containment is flooded during a severe accident condition.

In RAI 249, Question 03.11-12, the staff requested that the applicant provide additional details regarding how the U.S. EPR EQ program considers the chemical effects of the borated spray. In a July 10, 2009, response to RAI 249, Question 03.11-12, the applicant referred to FSAR Tier 2, Section 3.11.5.1 that states, “chemical spray is not considered for the U.S. EPR, because chemical spray is not used to mitigate a DBE.” Therefore, in FSAR Tier 2, Appendix 3D, Table 3D-1, chemical spray is shown as “Not applicable,” and in FSAR Tier 2, Appendix 3D, Table 3D-4, “Normal Operating Environments,” the chemistry parameter is identified as “None.” For consistency, FSAR Tier 2, Appendix 3D, Table 3D-7, “Abnormal Room Conditions,” will be revised to indicate that the chemical environment also does not apply to the conditions identified in this table. The applicant further stated that the SAHRS is a system dedicated to support mitigation of beyond DBEs and is classified as a non-safety system. Thus, the SAHRS (chemical spray system) is not required to be environmentally qualified per 10 CFR 50.49. On the basis of the above review, the staff finds that the applicant incorporated appropriate changes in FSR Tier 2, Appendix 3D, Table 3D-7. The staff identified the changes in Revision 2 of the FSAR Tier 2 and finds that revisions were made by the applicant as committed in the July 10, 2009, response to RAI 249, Question 03.11-12. Therefore, the staff considers RAI 249, Question 03.11-12 resolved.

Radiation environments are reviewed for normal and accident conditions. FSAR Tier 2, Section 3.11 states that normal operation radiation doses are calculated for initial plant start-up conditions. Radiation doses are continuously monitored during plant operation. If the actual measured radiation doses are higher than the original calculated doses, the U.S. EPR database will be revised and qualified life adjustments identified through the EQ program. In addition, if area doses exceed the qualified dose of an item of interest, a component specific dose calculation may be performed to determine doses at the specific equipment location, as well as the need for qualified life adjustments.

The normal operation dose rates for equipment qualification are derived from direct gamma emitted by components that contain radioactive fluids. Because the structural walls of these components shield beta particles, beta radiation is omitted. Bremsstrahlung radiation is also neglected, because it is a small contributor compared to the normal operations source term gamma contribution. FSAR Tier 2, Revision 1, Appendix 3D, Section 3D.5.1.1, "Normal Radiation Dose," refers FSAR Tier 2, Revision 1, Appendix 3D, Table 3D-8, "Bounding Normal EQ Dose," where it lists the bounding normal operational dose rates, as well as cumulative dose values for various building areas assuming 60 years of continuous operation and steady-state operating conditions.

The applicant stated that bounding accident dose rates for equipment qualification are calculated based on the guidance of RG 1.183 and include a submersion dose and a direct dose contribution. The submersion dose is primarily from engineered safety features (ESF) leakage; the dose contribution is derived from both the gamma and beta radiation. The beta radiation may be attenuated by low-density equipment enclosures. Alpha radiation is also neglected from both the normal and accident equipment qualification dose rates, because the alpha particle is easily attenuated by air, and it is primarily a personnel committed dose concern.

In RAI 96, Question 03.11-2, Part (ii), the staff requested that the applicant clarify the 55 year integration period used by the applicant for the Annulus and Containment Buildings (calculated by dividing the cumulative dose values given in the table by the corresponding dose rates given) which appeared to contradict the 60 year integration period discussed in FSAR Tier 2, Appendix 3D, Section 3D.5.1.1. In an October 17, 2008, response to RAI 96, Question 03.11-2, Part (ii), the applicant clarified that the difference in integration times resulted from multiplying the dose rate by a capacity factor of 0.92 in order to account for maintenance outages. On the basis of its review, the staff finds that the applicant provided adequate basis for the 60 years of continuous operation. Therefore, the staff considers RAI 96, Question 03.11-2, Part (ii) resolved.

In RAI 96, Question 03.11-2, Part (iii), the staff requested that the applicant correct some apparent typographical errors, including a missing closed parentheses in FSAR Tier 2, Appendix 3D, Table 3D-4 and an incorrect reference in Table 3D-4 to FSAR Tier 2, Table 3D-7 to describe the radiation normal operating environment. In an October 17, 2008, response to RAI 96, Question 03.11-2, Part (iii), the applicant explained that the missing parentheses and the reference to FSAR Tier 2, Table 3D-7 were errors. FSAR Tier 2, Table 3D-4 will be revised to include the closed parenthesis and the correct reference to FSAR Tier 2, Table 3D-8. The staff confirmed Revision 1 of the FSAR, dated May 29, 2009, Appendix 3D, Table 3D-4 (Sheet 4 of 5), contains the changes committed to in the RAI response. Accordingly, the staff finds that the applicant has adequately addressed this issue and, therefore, the staff considers RAI 96, Question 03.11-2, Part (iii) resolved.

In RAI 96, Question 03.11-2, Part (iv), the staff requested that the applicant clarify what actions will be required if dose rates exceed thresholds for a mild radiation environment for the various pieces of equipment. Currently, the thresholds used are for 10^4 Rad for mechanical and 10^3 Rad for electronic devices and components. In an October 17, 2008, response to RAI 96, Question 03.11-2, Part (iv), the applicant explained that once the integrated dose for a certain area exceeds the above thresholds, the equipment located in that area could no longer be considered to be in a “mild” radiation environment and, therefore, it would have to be replaced, unless it was shown that the actual dose to the component was lower due to appropriate shielding. On the basis of its review and the clarification provided in the October 17, 2008, response to RAI 96, Question 3.11-2, Part (iv), the staff finds that the applicant has adequately addressed this issue and, therefore, the staff considers RAI 96, Question 03.11-2, Part (iv) resolved.

In RAI 249, Question 03.11-11, the staff requested that the applicant clarify whether mechanical, electrical, and I&C equipment are required to be qualified for 60 years. In a July 10, 2009, response to RAI 249, Question 03.11-11, the applicant stated that the mechanical and electrical equipment qualified life is 60 years, based on the design life of the plant. The qualified life is verified using methods and procedures of qualification and documentation as stated in IEEE Std 323. Each vendor is required to determine the qualified life for each component of equipment within their scope of supply. Any components that do not meet the qualified life will be evaluated for extension of the qualified life or replaced to meet the 60 year design life of the plant. On the basis of its review and the clarification provided in the July 10, 2009, response to RAI 249, Question 3.11-11, the staff finds that the applicant has adequately addressed this issue and, therefore, the staff considers RAI 249, Question 03.11-11 resolved.

For qualification methods, the applicant stated that equipment may be qualified by testing, analysis, operating experience, or combination of methods prescribed by IEEE Std 323-1974. FSAR Tier 2, Appendix 3D, Sections 3D.6 and 3D.7 provide details on each methodology. In the above analysis (3D.6.2) method, it further discusses similarity (FSAR Tier 2, Appendix 3D, Section 3D.6.2.1) and substitution (FSAR Tier 2, Appendix 3D, Section 3D.6.2.2). According to 10 CFR 50.49(f), the EQ of equipment located in a harsh environment shall be demonstrated by appropriate type testing, testing supported by analyses, or analyses supported by experience data and/or partial test data.

In RAI 317, Question 03.11-16, the staff requested that the applicant revise FSAR Tier 2, Section 3D.6.2, “Analysis,” to include test information or test data for the analysis section to demonstrate equipment qualification. This demonstration is to ensure that analysis alone would not be used to demonstrate equipment qualification. In an October 29, 2009, response to RAI 317, Question 03.11-16, the applicant stated that FSAR Tier 2, Appendix 3D, Section 3D.6.2 will be revised to clarify the limits and requirements for analysis. The revision will include analysis with type test data for material properties, equipment rating, and environmental tolerances may be used to demonstrate qualification. The staff has reviewed the October 29, 2009, response to RAI 317, Question 03.11-16, and finds that the bases of the analysis will include the type test information, test data, or operating data as appropriate. The staff reviewed the changes in Revision 2 of the FSAR Tier 2 and finds that revisions were made by the applicant as committed to in the October 29, 2009, response to RAI 317, Question 03.11-16. Therefore, the staff considers RAI 317, Question 03.11-16 resolved, as revised FSAR Tier 2, Section 3D.6.2, “Analysis,” to include test information or test data for the analysis section to demonstrate equipment qualification.

In RAI 326, Question 03.11-19, the staff requested that the applicant identify and delete all “analyses” from all ITAAC tables listed in FSAR Tier 1, as analyses alone could have interpreted as an acceptable way for demonstrating for the EQ qualification. This is contrary to 10 CFR 50.49(f)(4), as it must include test information or test data for the analysis section to demonstrate equipment qualification. In a January 15, 2010, response to RAI 326, Question 03.11-19, the applicant stated that the term, “Type tests, analyses, or a combination of type tests and analyses,” will be revised to state, “Type tests or type tests and analyses,” which will be consistent with the guidance of IEEE Std 323. In RAI 393, Question 03.11-34, the staff identified additional ITAAC items that these changes were made. Therefore, the staff considers RAI 326, Question 03.11-19 and RAI 393, Question 03.11-34 resolved.

In RAI 317, Question 03.11-17, the staff requested that the applicant revise FSAR Tier 2, Appendix 3D, Section 3D.6.2.1, “Similarity,” to include consideration of key material properties and aging characteristics (e.g., application/failure mode-specific activation energy) that can affect accelerated aging equivalent degradation and end-of-life harsh environment durability and performance. In an October 29, 2009, response to RAI 317, Question 03.11-17, the applicant stated that FSAR Tier 2, Appendix 3D, Section 3D.6.2.1 will be revised to clarify and expand the qualification criteria further. The supporting analysis, used to qualify one piece of equipment based upon testing performed on another piece of equipment, will consider information such as material properties and aging characteristics, including specific/failure mode based activation energy. The staff has reviewed the response that similarity analyses will consider items such as critical materials properties, aging characteristics, and applicable-specific harsh environment performance. The staff finds the applicant’s response acceptable. The staff reviewed the changes in Revision 2 of the FSAR Tier 2 and finds that that revisions were made by the applicant as committed to in the October 29, 2009, response to RAI 317, Question 03.11-17. Therefore, the staff considers RAI 317, Question 3.11-17 resolved.

In RAI 317, Question 03.11-18, the staff requested that the applicant revise FSAR Tier 2, Appendix 3D, Section 3D.6.2.2, “Substitution,” to reflect analysis of substitute parts or materials that assumes the material properties required in a harsh environment and manufacturing processes that support the analyses as prescribed by 10 CFR 50.49(f). In an October 29, 2009, response to RAI 317, Question 03.11-18, the applicant stated that FSAR Tier 2, Section 3.11.2.2 and FSAR Tier 2, Appendix 3D will be revised to delete the term, ”form, fit and function,” as those elements alone are not sufficient for substitution. The applicant will revise to take materials or manufacturing process into account. Since the applicant replaced the above term, “form, fit, and function,” with a reference to the material evaluations described in FSAR Tier 2, Section 3.11.2.2.5, the staff finds that the material evaluation will be consistent with the analysis prescribed by 10 CFR 50.49(f). The staff has reviewed the changes in Revision 2 of the FSAR Tier 2 and finds that revisions were made by the applicant as committed to in the October 29, 2009, response to RAI 317, Question 03.11-18. Therefore, the staff considers RAI 317, Question 03.11-18 resolved.

As for satisfying 10 CFR 50.49(i) on the EQ documentation, the applicant stated that the U.S. EPR equipment qualification program documentation consists of: (1) Equipment qualification data package; (2) equipment qualification test reports; and (3) equipment maintenance requirements.

Based on the staff’s review of the applicant’s EQ program provided in FSAR Tier 2, Section 3.11, Appendix 3D, the staff finds that the program includes qualification criteria (Mild vs. harsh environments, qualified life, operability time), design specification (normal and abnormal operating conditions for temperature or radiation) qualification methods (type test, and combination of testing and analysis), and documentation (EQDP and maintenance records) needed to support electrical and I&C equipment qualification prescribed by 10 CFR 50.49. The staff finds that the applicant complies with the requirements of 10 CFR 50.49.

3.11.4.1.2 Conformance to RG 1.89 and RG 1.97

RG 1.89 is used as a principal guidance for implementing the requirements and criteria of 10 CFR 50.49 for environmental qualification of electrical equipment that is important to safety and located in a harsh environment. RG 1.89 endorses IEEE Std 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," which provides guidance for demonstrating the qualification of Class 1E equipment by including test procedures and analysis methods. When these qualification requirements are met, the electrical and I&C equipment that is important to safety will perform its design function under normal, abnormal, DBE, post DBE, and containment test conditions. FSAR Tier 2, Section 3.11 states that NUREG-0588, Category I guidance has been used to enhance the guidance provided in RG 1.89. The FSAR further states that electrical equipment identified in FSAR Tier 2, Table 3.11-1 will be environmentally qualified by type testing or type testing and analysis using the guidance provided in IEEE Std 323-2003, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations."

In RAI 96, Question 03.11-1, the staff requested that the applicant change all references to IEEE Std 323-2003 for environmental qualification of electrical equipment that is located in a harsh environment be changed to IEEE Std 323-1974 in FSAR Tier 2, Section 3.11 and FSAR Tier 2, Appendix 3D, or provide appropriate justification for deviating from IEEE Std 323-1974. In a May 6, 2009, response to RAI 96, Question 03.11-1, the applicant agreed to endorse IEEE Std 323-1974 and revised all references to IEEE Std 323-1974 for FSAR Tier 2, Section 3.11 and FSAR Tier 2, Appendix 3D, with the exception of RG 1.209 where the IEEE Std 323-2003 is referenced for safety-related computer-based I&C systems located in a mild environment. Subsequently, the applicant provided the updated pages. The staff confirmed that Revision 1 of the FSAR dated May 29, 2009, contains the changes committed to in the RAI response. Accordingly, the staff finds that the applicant has properly adopted IEEE Std 323-1974 in FSAR Tier 2, Section 3.11 and FSAR Tier 2, Appendix 3D. Therefore, the staff considers RAI 96, Question 03.11-1 resolved.

In RAI 249, Question 03.11-8, the staff requested that the applicant identify the elements of NUREG-0588, Category 1 that will be used to enhance the guidance provided in the EQ Program. In a July 10, 2009, response to RAI 249, Question 03.11-8, the applicant stated that the guidance in NUREG-0588 and RG 1.89 "may be used" to meet the requirements of 10 CFR 50.49. The use of any guidance documents will be determined and documented later in the procurement phase after vendors respond to the applicant's specification requirements for the specific component, and it will be documented in the EQ program implementation in the EQ data package (EDDP). Based on the clarification provided in the applicant's response, the staff agrees that the enhancement will be discussed by the vendor and will be documented in EQDP if NUREG-0588, Category I is used in the EQ program. Therefore, the staff considers RAI 249, Question 03.11-8 resolved.

FSAR Tier 2, Section 3.11 states that PAM equipment will be environmentally qualified in accordance with RG 1.97, Revision 4 that endorses IEEE Std 497-2002, "IEEE Standard Criteria for Accident Monitoring Instrumentation for Nuclear Power Generating Stations.". The method used to identify and qualify this equipment is described in FSAR Tier 2, Section 7.5, "Information Systems Important to Safety." The PAM equipment is identified as type A, B, C, D or E, according to RG 1.97, Revision 4. While type E variables are not required to be environmentally qualified, FSAR Tier 2, Section 3.11 states that type A, B, C, and D will be environmentally qualified as required by 10 CFR 50.49 and the guidelines provided in the Branch Technical Position (BTP) 7-10, "Guidance on Application of Regulatory Guide 1.97."

Qualification of electrical equipment and components in a mild location is based on certificates of conformance to the purchaser/procurement specification. The applicant's EQ program addressed the acceptability of important to safety electrical equipment located in a mild environment (not subject to 10 CFR 50.49) as follows:

- A periodic maintenance, inspection or replacement program based on sound engineering practice and recommendation of the equipment manufacturer, which is updated as required by the results of an equipment surveillance program

- A periodic testing program used to verify operability of safety-related equipment within its performance specification requirements. System level testing of the type typically required by the plant technical specifications may be used

- An equipment surveillance program that includes periodic inspections, analysis of equipment and component failures, and a review of the results of the preventive maintenance and periodic testing program

The staff finds that the U.S. EPR EQ program uses correct guidance documents: (1) RG 1.89 which endorses IEEE Std 323-1974 for the electrical equipment and (2) RG 1.97, Revision 4 that endorses IEEE Std 497-2002 for the post accident monitoring I&C equipment that are important to safety and located in a harsh environment. Thus, the staff concludes that U.S. EPR EQ program conforms to the guidance of RG 1.89 and RG 1.97 in satisfying the requirements of 10 CFR 50.49.

3.11.4.1.3 Compliance with 10 CFR Part 50, Appendix A

The applicant is required to comply with the following acceptance criteria of 10 CFR Part 50, Appendix A:

3.11.4.1.3.1 *GDC 1, "Quality Standards and Records"*

GDC 1 addresses requirements for quality standards that must be met, and records that must be kept concerning the quality standards for design, fabrication, erection, and testing of components important to safety. Components in the GDC 1 scope must have auditable records to document that environmental design and qualification requirements have been met.

All electrical and I&C equipment important to safety will be designed, fabricated, and qualified by methods for quality standard prescribed by IEEE Std 323-1974. This standard is used as a principal guidance for implementing the requirements and record keeping criteria of 10 CFR 50.49 for environmental qualification of electrical equipment that is important to safety and located in a harsh environment. All qualification records per FSAR Tier 2, Appendix 3D, Section 3D.8, "Documentation," will be documented and maintained in an auditable form for the entire installed life for quality standards. Records will be kept concerning the quality standards for design, fabrication, erection, and testing of components. Therefore, by satisfying the acceptance requirements of 10 CFR 50.49(j), the staff finds that equipment quality standard and records complies with the requirements of GDC 1.

3.11.4.1.3.2 *GDC 2, "Design Bases for Protection against Natural Phenomena"*

GDC 2 addresses the design bases for components important to safety must withstand the effects of the most severe natural phenomena without loss of capability to perform their safety function.

Components in the GDC 2 scope are designed with consideration of the environmental conditions or stressors resulting from natural phenomena as part of the environmental conditions outlined in 10 CFR 50.49 evaluated. The applicant stated that equipment quality standards GDC-2 requires testing (type) appropriate combinations of the effects of normal and accident conditions and for the effects of the natural phenomena. Satisfying the qualification testing requirements of 10 CFR 50.49 assures that equipment will be designed to withstand the effects associated with natural phenomena without loss of capability to perform their safety functions during and after DBEs. The staff finds that this complies with the requirements of GDC 2.

3.11.4.1.3.3 *GDC 4, "Environmental and Dynamic Effects Design Bases"*

GDC 4 requires that components important to safety be designed to protect against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures, and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs.

10 CFR 50.49(f) describes the methodology used to qualify equipment that can perform its safety functions, under the specified conditions such as applicable normal, abnormal, and DBE service conditions during its qualified life. Since all EQ equipments are tested and qualified for the requirements of 10 CFR 50.49 (i.e., to simulated the effects, or analyzed with test data for equipment failures) to withstand the aforementioned normal operations, maintenance, and postulated accidents, including LOCAs, the applicant stated that they are protected against dynamic effects that may result from equipment failures. The staff finds that this complies with the requirements of GDC 4.

3.11.4.1.3.4 *GDC 23, "Protection System Failure Modes"*

GDC 23 requires that protection systems be designed to fail in a safe state, or in a state demonstrated to be acceptable on some other defined basis, if conditions such as postulated adverse environments (e.g., extreme heat or cold, steam, water, or radiation) are experienced.

FSAR Tier 2, Appendix 3D, Section 3D.6, "Qualification Methods," describes the qualification methods used that depend on factors such as materials used in construction of the equipment, applicable normal, abnormal, and DBE service conditions, and dynamic characteristics such as disconnection of system, loss of energy, or postulated adverse environments of the expected failure modes of equipment. The applicant stated that components in this scope are subject to environmental design, and qualification requirements must consider the failure mode of the equipment. Since the qualification methods used to test its protection systems include the above dynamic characteristics of the expected failure modes of equipment, the staff finds that this complies with the requirements of GDC 23.

3.11.4.1.4 Compliance with 10 CFR Part 50, Appendix B

Compliance with 10 CFR Part 50, Appendix B, Criterion III, “Design Control, requires that measures be established to ensure that applicable regulatory requirements and the associated design bases are correctly translated into specifications, drawings, procedures, and instructions. This criterion is applicable since it includes requirements for test programs that are used to verify the adequacy of a specific design feature. Such test programs include suitable qualification testing of a prototype unit under the most adverse design conditions.

The applicant stated that compliance with 10 CFR 50.49(f) requires that the environmental qualification process under EQ program includes appropriate qualification testing of a prototype unit under the most adverse design conditions to verify the adequacy of a specific design feature. The staff finds that EQ related testing under the most adverse design conditions complies with 10 CFR Part 50, Appendix B, Criterion III.

Compliance with 10 CFR Part 50, Appendix B, Criterion XI, “Test Control,” requires development of a test control plan to ensure that all tests needed to demonstrate a component’s capability to perform satisfactorily in service be identified and performed in accordance with written procedures that incorporate the requirements and acceptance limits contained in applicable design documents. RG 1.89 that endorses IEEE Std 323-1974 outlines a planned sequence of test conditions (test plan) that meet or exceed the expected or specified service conditions. Since RG 1.89 provides the guidance for satisfying qualification testing 10 CFR 50.49, the staff finds that meeting the requirements of 10 CFR 50.49 that outline a planned sequence of test plan assures compliance with 10 CFR Part 50, Appendix B, Criterion XI.

Compliance with 10 CFR Part 50, Appendix B, Criterion XVII. “Quality Assurance Records,” requires that sufficient records be maintained to furnish evidence of activities affecting quality. The EQ records must include inspections, tests, audits, monitoring of work performance, and materials analysis. Records pertaining to quality assurance must be identifiable and retrievable.

Complying with 10 CFR 50.49 (j) requires that records must be maintained to furnish evidence of activities affecting quality. Environmental design and qualification must have identifiable and retrievable records that document the fact that they meet these requirements.

Meeting the requirements of 10 CFR Part 50, Appendix B, Criterion XVII provides assurance that identifiable and retrievable records are maintained to furnish evidence of activities affecting quality, which includes environmental design and qualification.

The U.S. EPR equipment qualification program documentation consists of: (1) Equipment qualification data package (EQDP) and (2) equipment qualification test reports. The staff finds that the aforementioned U.S. EPR equipment qualification documentation for 10 CFR 50.49 (j) complies with the requirements of 10 CFR Part 50, Appendix B, Criterion XII

Based on the above, the staff finds that the U.S, EPR EQ program complies with the requirements of 10 CFR Part 50, Appendix B, Criteria III, “Design Control”; XI, “Test Control”; and XVII “Quality Assurance Records.”

3.11.4.1.5 Conformance of RGs that are specific to electrical and I&C equipment

The electromagnetic compatibility per RG 1.180 is included as a service condition that must be considered to address proper operation under adverse conditions for digital I&C equipment and considered as one of the screening criteria for the EQ list in FSAR Tier 2, Table 3.11-1. Addressing EMC involves testing to show that critical equipment will not be adversely affected by electromagnetic interference or radio frequency interference in the plant environment. In addition, only selected TI and BOP electrical components (e.g., switchgear, motor control centers (MCCs), transformers) that might be susceptible to electromagnetic compatibility (EMC) per RG 1.180 or non-safety-related equipment whose failure could prevent a safety function per 10 CFR 50.49(b)(2), are considered in their application.

The U.S. EPR referenced RGs used to address specific equipment for their qualification on motors (RG 1.40), penetration assemblies (RG 1.63), valve operators (RG 1.73), cables (RG 1.211 replaces RG 1.131), digital computers (RG 1.152), connection assemblies (RG 1.156), lead storage batteries (RG 1.158), EMI/RFI (RG 1.180), alternate radiological source (RG 1.183), and computer based I&C system (RG 1.209). FSAR Tier 2, Table 3.11-4, “Summary of Comparison of IEEE Endorsed Standards versus Latest IEEE Standards,” provides a summary comparison of the current IEEE standards to be used for equipment qualification and the associated RGs and revision that endorse them. FSAR Tier 2, Table 3.11-5, “Summary of IEEE Non-Endorsed Standards,” provides a summary of the related qualification standards that are not associated with an RG. Some of above RGs include new (non-endorsed) standards. The staff finds the applicant’s usage of the above IEEE standards with the provided justifications is sufficient, because it provides additional information for electrical and I&C equipment that are important to safety.

3.11.4.2 *Environmental Qualification of Mechanical Equipment*

The staff reviewed FSAR Tier 1, Section 2.0, FSAR Tier 2, Sections 3.2 and 3.11, and FSAR Tier 2, Appendix 3D for compliance with NRC regulations for the environmental design and qualification of safety-related and important-to-safety mechanical equipment to be used in U.S. EPR nuclear power plants. Environmental design means that components shall be designed to accommodate the effects of environmental conditions and is required for all safety-related and important-to-safety equipment in mild and harsh environments. Environmental qualification means verification of design, limited to demonstrating that equipment is capable of performing its safety function under significant and environmental stresses (i.e., harsh environments) resulting from design-basis events in order to avoid common cause failure. GDC 1 and GDC 4 in 10 CFR Part 50, Appendix A and 10 CFR Part 50, Appendix B, Criterion III and Criterion XVII provide the following general requirements related to environmental design and qualification of mechanical equipment. (1) The components shall be designed to have the capability of performing their design safety functions under all anticipated operational occurrences and in normal, accident, and post-accident environments, including LOCA, and for the length of time for which the function is required; (2) the environmental qualification of components located in harsh environments shall be demonstrated by appropriate testing and analysis; and (3) a quality assurance program meeting 10 CFR Part 50, Appendix B, shall be established and implemented to provide assurance that all requirements have been satisfactorily accomplished.

The staff followed the acceptance criteria in SRP Section 3.11 in reviewing the environmental qualification of safety-related mechanical equipment in the U.S. EPR. In particular, mechanical components must be designed to be compatible with postulated environmental conditions, including those associated with a LOCA. A process must be established to determine the suitability of materials, parts, and equipment needed for safety-related functions, and to verify that the design of such materials, parts, and equipment is adequate. Equipment records must be maintained with the results of test, and material analyses used a part of the environmental design and qualification process for each component. FSAR Tier 2, Chapter 3 provides for the qualification of metallic equipment and its reference to ASME Boiler & Pressure Vessel Code, Section III, as incorporated by reference in 10 CFR 50.55a, and ASME Standard QME-1-2007, as accepted in RG 1.100, Revision. FSAR Tier 2. Section 3.11 addresses supplemental details regarding the environmental qualification of nonmetallic parts. For the environmental qualification of nonmetallic parts of mechanical equipment, the staff concentrated its review on materials that are sensitive to environmental effects (e.g., seals, gaskets, lubricants, fluids for hydraulic systems, and diaphragms). The staff's review included the following objectives: (1) Identify safety-related mechanical equipment located in harsh environment areas, including required operating time; (2) identify non-metallic subcomponents of such equipment; (3) identify the environmental conditions for which the equipment must be qualified; (4) identify non-metallic material capabilities; and (5) evaluate environmental effects. For mechanical equipment located in a mild environment, design/purchase specifications that can be used to demonstrate acceptable environmental design. Maintenance and surveillance programs developed by the COL applicant provide assurance that equipment qualification is maintained during the operational life of the plant.

FSAR Tier 2, Section 3.2, "Classification of Structures, Systems, and Components," states that U.S. EPR SSCs are categorized as safety-related (as defined in 10 CFR 50.2) or non-safety-related. Safety-related SSCs are those relied upon to remain functional during and following design-basis events to ensure: (1) The integrity of the reactor coolant pressure boundary; (2) the capability to shut down the reactor and maintain it in a safe condition; or (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the applicable guideline exposures set forth in 10 CFR 50.34(a)(1). Safety-related SSCs must conform to the QA requirements of 10 CFR Part 50, Appendix B. Non-safety-related SSCs have QA provisions applied commensurate with the importance of the SSC's function.

FSAR Tier 2, Appendix 3D.6.2.3, "Qualification of Safety-Related Mechanical Equipment," states that FSAR Tier 2, Section 3.11.2.2 describes the qualification of mechanical equipment. Engineering design specifications are generated and used to procure equipment, components, and parts. Under the procurement program, compliance with GDC 4 through the evaluation of non-metallic parts in mechanical components is based on material evaluations as described in FSAR Tier 2, Section 3.11.2.2.5. FSAR Tier 2, Table 3D-10, "Mechanical Equipment Components Requiring Environmental Qualification," provides a summary of the types of non-metallic or consumable parts in mechanical components that will be screened for EQ. The list of specific non-metallic components by tag number screened in the EQ program is provided in FSAR Tier 2, Section 3.10, Table 3.10-1, "List of Seismically and Dynamically Qualified Mechanical and Electrical Equipment."

FSAR Tier 2, Section 3.11.2.2 describes the environmental design and qualification process for mechanical equipment located in a harsh environment. FSAR Tier 2, Section 3.11.2.2 states that mechanical equipment located in harsh environmental zones is designed to perform under the appropriate environmental conditions. For mechanical equipment, the staff's primary focus is on materials that are sensitive to environmental effects (e.g., seals, gaskets, lubricants, fluids for hydraulic systems, and diaphragms) needed for safety-related functions and to verify that the design of such materials, parts, and equipment is adequate. This process involves:

- Identifying safety-related mechanical equipment located in harsh environment areas
- Identifying nonmetallic subcomponents of this equipment
- Identifying the environmental conditions and process parameters for which this equipment must be qualified
- Identifying nonmetallic material capabilities
- Evaluating environmental effects on the nonmetallic components of the equipment

FSAR Tier 2, Section 3.11.2.2 states that mechanical components are designed to be compatible with postulated environmental conditions, including those associated with LOCAs. The environmental qualification of equipment located in harsh environments shall be demonstrated by appropriate testing and analyses using applicable service conditions as required by GDC 4 and discussed in FSAR Tier 2, Appendix 3D.

FSAR Tier 2, Section 3.11.2.2 states that the potential impact of adverse environmental conditions is considered in the functional design and qualification of pumps, valves and dynamic restraints (see FSAR Tier 2, Section 3.9.6). For example, electric motors might produce less torque under high temperature conditions than under ambient conditions, which could impact their capability to operate their individual pumps or valves.

FSAR Tier 2, Section 3.11.2.2 states that since most mechanical equipment interfaces with process fluid, the effect of the fluid on the environmental conditions is considered for the design and qualification of mechanical equipment.

FSAR Tier 2, Sections 3.11.2.2.1 through 3.11.2.2.5 describe the environmental design and qualification process for nonmetallic materials located in harsh environments.

FSAR Tier 2, Section 3.11.2.2.1, "Identifying Safety-Related Mechanical Equipment Located in Harsh Environment Areas," states that safety-related mechanical equipment located in harsh environmental areas are identified in FSAR Tier 2, Table 3.10-1.

FSAR Tier 2, Section 3.11.2.2.2, "Identifying Nonmetallic Subcomponents of this Equipment," states that non-metallic subcomponents of safety-related mechanical equipment located in harsh environments are identified in FSAR Tier 2, Table 3.10-1, and states that engineering design specifications are used in the procurement of equipment, components, and parts that are to be qualified. The procurement specifications identify the environmental conditions for which the components must be qualified.

FSAR Tier 2, Section 3.11.2.2.3, "Identifying the Environmental Conditions and Process Parameters for which this Equipment Must Be Qualified," states that mechanical equipment experiences the same environmental conditions as those defined in 10 CFR 50.49 for electrical equipment, and such conditions are used in the qualifying of mechanical equipment. FSAR Tier 2, Section 3.11.1.2 describes the environmental conditions for which the equipment is qualified. The environmental parameters (e.g., radiation, temperature, chemical spray, humidity from steam, pressure, flooding) applicable to the various environmental conditions in specific plant building and room locations are specified in FSAR Tier 2, Appendix 3D, Section 3D.5 and in tables and figures provided in FSAR Tier 2, Appendix 3D.

FSAR Tier 2, Section 3.11.2.2.4, "Identifying Nonmetallic Material Capabilities," describes the process for identifying non-metallic material capabilities. Mechanical equipment is designed to comply with GDC 4 by verifying the ability of the components to perform their required safety functions when exposed to internal and external, normal and abnormal operating conditions, and when exposed to external postulated accident environments. The engineering design process and program evaluates both metallic and non-metallic components to meet environmental conditions (e.g., radiation, temperature, pressure) for safety related and important to safety mechanical equipment. Operating temperatures and pressures are compared to the design parameters of each component to confirm and demonstrate that design limits are not exceeded. The effects of radiation are considered in the evaluations.

FSAR Tier 2, Section 3.11.2.2.5, "Evaluating Environmental Effects on the Non-metallic Components of the Equipment," describes the process for evaluating environmental effects on the non-metallic components. Mechanical equipment is designed to have the capability of performing its design safety functions for the length of time for which its function is required. Non-metallic components, such as greases, gaskets, lubricant, are shown to be capable of performing their intended functions under all postulated service conditions. Non-metallic components in harsh environments are qualified in accordance with QME-1-2007, Appendix QR-B as endorsed by RG 1.100, Revision 3. RG 1.100, Revision 3 states that if a licensee commits to the use of non-mandatory appendices in ASME QME-1-2007 for its qualification of active mechanical equipment, then the criteria and procedures delineated in those non-mandatory appendices become part of the requirements for its qualification program.

FSAR Tier 2, Section 3.11.2.2, "Environmental Qualification of Mechanical Equipment," describes the environmental design process for mechanical equipment located in a mild environment. FSAR Tier 2, Section 3.11.2.2 states that for mechanical equipment located in a mild environment, acceptable environmental design is demonstrated by the design and purchase specifications for the equipment. The specifications contain a description of the functional requirements for a specific environmental zone during normal environmental conditions and anticipated operational occurrences.

FSAR Tier 2, Section 3.11.2.2, "Environmental Qualification of Mechanical Equipment," states that maintenance and surveillance programs, as described in FSAR Tier 2, Section 3.11.2.2.6, provide reasonable assurance that the safety function of the equipment, related to environmental considerations established during design, is maintained on a continuing basis.

FSAR Tier 2, Section 3.11.2.2.6, "Maintaining Mechanical Equipment Qualification," states that compliance with GDC 4 is maintained through the engineering design, procurement, maintenance, and surveillance programs. These plant programs include inspections, testing, analyses, repairs, and replacement. For mechanical equipment, qualification is maintained through implementation of the preventive maintenance program, surveillance program, and periodic testing of mechanical equipment. Operating and maintenance programs related to the environmental qualification of electrical and mechanical equipment are the responsibility of the

COL applicant as described in FSAR Tier 2, Section 13.5.2. As stated in response to RAI 435, Question 03.11-38 dated November 29, 2010, these programs are the responsibility of the COL applicant, and in order to fully describe the EQ operational programs, the programs must include an EQ Master Equipment List (EQMEL) and describe the operational aspects of the EQ maintenance and surveillance programs such as: (1) Evaluation of EQ results for design life to establish activities to support continued EQ; (2) determination of surveillance and preventive maintenance activities based on EQ results; (3) consideration of EQ maintenance recommendations from equipment vendors; (4) evaluation of operating experience in developing surveillance and preventive maintenance activities for specific equipment; (5) development of plant procedures that specify individual equipment identification, appropriate references, installation requirements, surveillance and maintenance requirements, post-maintenance testing requirements, condition monitoring requirements, replacement part identification, and applicable design changes and modifications; (6) development of plant procedures for reviewing equipment performance and EQ operational activities, and for trending the results to incorporate lessons learned through appropriate modifications to the EQ operational program; (7) development of plant procedures for the control and maintenance of EQ records; and (8) development of an EQMEL that includes the length of time for which the safety-related component function is required.

NUREG-0800, Section 3.11 states that equipment shall be designed to the capability of performing its design safety functions under all anticipated operational occurrences and in normal, accident, and post-accident environment, and for the length of time for which its function is required. FSAR Tier 2, Table 3.10-1, “List of Seismically and Dynamically Qualified Mechanical and Electrical Equipment,” and FSAR Tier 2, Table 3.11-1, “List of Environmentally Qualified Electrical/I&C Equipment,” do not identify the function time for the equipment. Therefore, in RAI 326, Question 03.11-33, the staff requested that the applicant revise FSAR Tier 2, Section 3.11 to address the length of time for which the function of each mechanical component is required. In an April 19, 2010, response to RAI 326, Question 03.11-11, the applicant stated that the operability times for electrical and mechanical equipment listed in FSAR Tier 2, Tables 3.10-1 and 3.11-1 are documented in the Equipment Qualification Data Packages (EQDPs) and the Seismic Qualification Data Packages (SQDPs). Additionally, ITAAC exist for the SQDP and the EQDP. Function times are described above for the components identified in the U.S.EPR FSAR Tier 1 tables. Based on the above description that operability times for electrical and mechanical equipment are to be documented in the SQDP and the EQDP and that ITAAC specify that the function times are identified, the staff finds that the applicant has adequately described the process to document the function time for electrical and mechanical components. Therefore, the staff considers RAI 326, Question 03.11-33 resolved.

As part of a COL application review, the staff will evaluate the full description of the operational program for the environmental qualification of electrical and mechanical equipment provided by the COL applicant to supplement the general program description outlined in the FSAR. For example, the COL applicant will need to confirm the program scope as part of its development of a full description of the environmental qualification program on a plant-specific basis and address operational aspects for maintaining equipment qualification. The COL applicant will implement the environmental qualification program, and the staff will inspect the program during plant construction.

In summary, the staff reviewed FSAR Tier 1, Section 2.0; FSAR Tier 2, Sections 3.2 and 3.11; and FSAR Tier 2, Appendix 3D for the environmental qualification of safety-related mechanical equipment used in the U.S. EPR. The FSAR establishes an EQ methodology for applicable safety-related and important-to-safety mechanical equipment and their non-metallic subcomponents. Based on its review, the staff finds the FSAR acceptable with respect to the environmental qualification of safety-related and important-to-safety mechanical equipment for the U.S. EPR design certification.

3.11.5 Combined License Information Items

Table 3.11-2 provides a list of environmental qualification related COL information item numbers and descriptions from FSAR Tier 2, Table 1.8-2:

Table 3.11-2 U.S. EPR Combined License Information Items

Item No.	Description	FSAR Tier 2 Section
3.11-1	A COL applicant that references the U.S. EPR design certification will maintain the equipment qualification test results and qualification status file during the equipment selection, procurement phase and throughout the installed life in the plant.	3.11
3.11-2	A COL applicant that references the U.S. EPR design certification will identify additional site specific components that need to be added to the environmental qualification list in Table 3.11-1.	3.11.1.1
3.11-3	If the equipment qualification testing is incomplete at the time of the COL application, a COL applicant that references the U.S. EPR design certification will submit an implementation program, including milestones and completion dates, for NRC review and approval prior to installation of the applicable equipment.	3.11-3

3.11.6 Conclusions

As set forth above, the staff has reviewed all of the relevant information that is applicable to FSAR Tier 2, Section 3.11 for EQ of mechanical and electrical equipment and evaluated for compliance with the requirements of GDC 1, GDC 2, GDC 4, and GDC 23; 10 CFR Part 50, Appendix B, Quality Assurance Criteria III, XI, and XVII; and 10 CFR 50.49 and conformance with applicable regulatory guides and standards committed by the applicant. The staff also reviewed the COL information items in FSAR Tier 2, Table 1.8-2 for the COL applicant. Except for confirmatory and open items, the staff concludes that the applicant has provided sufficient information in the FSAR to support the bases for their conclusions and identified site-specific COL action items that required COL applicants to develop their EQ program. The staff concludes that the U.S. EPR EQ program supports the conclusion that mechanical, electrical, and I&C equipment, including digital I&C equipment, that are important-to-safety are capable of performing their design safety function under normal environmental conditions, anticipated operational occurrences, design-basis accidents, and post-accident environmental conditions.

The staff reviewed the U.S. EPR design certification application for compliance with NRC regulations for the environmental qualification of safety-related and important-to-safety mechanical equipment to be used in U.S. EPR nuclear power plants. Based on its review, the staff finds that the general description of methodology for the environmental qualification for mechanical equipment to be used in a U.S. EPR nuclear power plant satisfies NRC regulations for a design certification application. U.S. EPR EQ methodology adequately describes that: (1) The components shall be designed to have the capability of performing their design safety functions under all anticipated operational occurrences and in normal, accident, and post-accident environments, including a LOCA, and for the length of time for which the function is required; (2) the environmental qualification of components located in harsh environments shall be demonstrated by appropriate testing and analysis; and (3) a quality assurance program meeting 10 CFR Part 50, Appendix B, shall be established and implemented to provide the assurance that all requirements have been satisfactorily accomplished. Therefore, the staff concludes that FSAR Tier 2, Section 3.11 is acceptable, except for open item related to RAI 249, Question 03.11-14.

Additionally, the COL applicant is to provide a full description and a milestone for program implementation for the environmental qualification program that includes completion of plant-specific components. As part of the review of a COL application for a U.S. EPR nuclear power plant, the staff will evaluate the full description of the operational program for the environmental qualification of electrical and mechanical equipment provided by the COL applicant. The staff will also confirm the completion of the applicable ITAAC for U.S. EPR components during plant construction to ensure that design reports for piping systems and ASME components are satisfactory and to confirm that NRC regulations are met.

3.12 ASME Code Class 1, 2, and 3 Piping Systems, Piping Components, and their Associated Supports

3.12.1 Introduction

FSAR Tier 2, Section 3.12 addresses the design of the piping systems and piping supports used in Seismic Category 1, Seismic Category II, and non-safety-related systems. The staff evaluated the structural integrity and functional capability of safety-related piping systems associated with the design of the U.S. EPR standard plant. The review includes not only ASME B&PV Code Class 1, 2, and 3 piping and pipe supports, but also buried piping, instrumentation lines, interaction of Non Safety-Related Seismic Category I piping with Seismic Category I piping. The following sections of this report provide the staff's evaluation of the adequacy of the U.S. EPR piping analysis methods, design procedures, acceptance criteria, and verification of the design. The staff's evaluation included the following:

- Regulatory criteria
- Applicable codes and standards
- Methods to be used in the piping design
- Modeling of piping systems
- Pipe stress analysis criteria
- Pipe support design criteria

The application provides piping design criteria and process at this preliminary stage and the plan the COL applicant will follow to complete the final design.

3.12.2 Summary of Application

FSAR Tier 1/ITAAC: Piping design is addressed in the system description and there are ITAAC to address the design and as-installed piping analyses.

FSAR Tier 2: The applicant provided the methods used for the design and analyses of piping systems in Section 3.12. The applicant previously submitted Topical Report ANP-10264NP-A, which is referred to in the application. In addition, the applicant has used additional information provided in the FSAR Tier 2, Sections 3.7.2; 3.7.3; 3.9.1; 3.9.2, "Dynamic Testing and Analysis of Systems, Components, and Equipment"; 3.9.3; and 5.2, "Integrity of the Reactor Coolant Pressure Boundary," as references to support the information provided in FSAR Tier 2, Section 3.12. Completion of the piping design is addressed in Section 14.3.3, "Tier 1, Chapter 3, Non-System Based Design Descriptions and ITAAC."

3.12.3 Regulatory Basis

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 3.12 and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Section 3.12.I.

1. GDC 1 and 10 CFR 50.55a, as they relate to piping systems, pipe supports, and components being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed.

2. GDC 2 and 10 CFR Part 50, Appendix S, as it relates to design transients and resulting load combinations for piping and pipe supports necessary to withstand the effects of earthquakes combined with the effects of normal or accident conditions.

3. GDC 4, as it relates to piping systems and pipe supports important to safety, being designed to accommodate the effects of, and to be compatible with, the environmental conditions of normal as well as postulated events, such as loss-of-coolant accident and dynamic effects.

4. GDC 14, as it relates to the reactor coolant pressure boundary of the primary piping systems being designed, fabricated, constructed, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

5. GDC 15, as it relates to the reactor coolant systems and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences.

6. 10 CFR 52.47(b)(1), which requires that a design certification application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the design certification is built and will operate in accordance with the design certification, the provisions of the Atomic Energy Act of 1954, and NRC regulations.

3.12.4 Technical Evaluation

The NRC established requirements in 10 CFR Part 50 to ensure the pressure boundary leakage integrity of the piping components and structural integrity of the pipe supports in nuclear power plants. The staff evaluated the design, materials, fabrication, erection, inspection, testing, and inservice surveillance of piping and pipe supports using the industry codes and standards, RGs, and staff technical reports.

The staff has previously reviewed Topical Report ANP-10264NP-A and prepared a safety evaluation report, “Final Safety Evaluation by the Office of New Reactors, Topical Report ANP-10264NP-A, Revision 0, EPR Piping Analysis and Pipe Support Design Topical Report, AREVA NP, Inc (Docket No. 52-050).” In the FSER, the staff concluded that piping systems important to safety are designed to quality standards commensurate with importance safety. In a November 4, 2008, letter, the applicant submitted an NRC acceptable version of Topical Report ANP-10264NP-A for referencing in the U.S. EPR licensing application.
The safety determination basis for the piping design is fully documented in the FSER for Topical Report ANP-10264NP-A Codes and Standards.

GDC 1 requires that SSCs important to safety be designed, fabricated, erected, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. In 10 CFR 50.55a, the NRC requires that certain systems and components of boiling- and pressurized-water-cooled nuclear power reactors must meet certain requirements of the ASME Code. The regulation specifies the use of the latest edition and addenda endorsed by the NRC and any limitations discussed in the regulations. In RG 1.84, “Design, Fabrication, and Materials Code Case Acceptability, ASME Section III,” October 2007, the staff lists acceptable ASME Code cases for design and materials acceptability (Section III) and any conditions that apply to them.

3.12.4.1.1 ASME Boiler and Pressure Vessel Code

For ASME Code Class 1, 2, and 3 piping analysis and piping support design, the applicant used the 2004 Edition ASME Code, Section III, Division 1 as the base code with limitations identified in 10 CFR 50.55a(b)(1). FSAR Tier 2, Section 5.2.1.1, "Compliance with 10 CFR 50.55a," specifies the 2004 edition (no addenda) of the ASME Code for U.S. EPR design. The applicant addressed seismic design limitation specified in 10 CFR 50.55a(b)(1)(ii) by using Sub-articles NB/NC/ND-3650 of the 1993 addenda of the ASME Code. This position meets the regulatory position. The applicant states that Quality Group D piping will be analyzed based upon ASME B31.1, 2004 edition, no addenda. Section 5.2.1.1 of this report discusses the staff's evaluation of the ASME Code edition and the process for changing ASME Code editions and addenda. For the reason set forth in that section, the staff concludes that the ASME Code Class 1, 2, and 3 piping and support design will conform to the appropriate ASME code editions and addenda and NRC regulations. Using ASME B31.1 Code for Quality Group D piping design satisfies RG 1.26.

3.12.4.1.2 ASME Code Cases

The acceptable ASME Code cases that may be used for the design of the ASME Code Class 1, 2, and 3 piping systems in the U.S. EPR standard plants are those either conditionally or unconditionally approved in RG 1.84 in effect at the time of design certification. However, the COL applicant may submit with its COL application for staff review and approval future ASME Code cases that have been endorsed in RG 1.84 at the time of COL application, provided the ASME Code cases do not alter the staff's safety findings on the U.S. EPR certified design.

FSAR Tier 2, Section 3.12.2, "Codes and Standards," refers to Topical Report ANP-10264NP-A for acceptable Code Cases that may be used for the U.S. EPR design of ASME Code Class 1, 2, and 3 pressure-retaining components and their supports. The ASME Code Cases that are given in Topical Report ANP-10264NP-A, Section 2.2 for the RCPB components, which are applicable to the U.S. EPR piping and pipe support design are given below:

- ASME Code Case N-122-2, "Procedure for Evaluation of the Design of Rectangular Cross Section Attachments on Class 1 Piping, Section III, Division 1"

- ASME Code Case N-318-5, "Procedure for Evaluation of the Design of Rectangular Cross Section Attachments on Class 2 or 3 Piping, Section III, Division 1"

- ASME Code Case N-319-3, "Alternate Procedure for Evaluation of Stress in Butt Welding Elbows in Class 1 Piping, Section III, Division 1"

- ASME Code Case N-391-2, "Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Class 1 Piping, Section III, Division 1"

- ASME Code Case N-392-3, "Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Class 2 and 3 Piping, Section III, Division 1"

In RG 1.84, Revision 34, October 2007, the staff endorsed ASME Code Cases N-122-2, N-318-5, N-319-3, N-391-2, and N392-3. On this basis, the staff finds use of the ASME Code cases proposed by the applicant acceptable.

3.12.4.1.3 Design Specifications

ASME Code, Section III, Subarticle NCA-3250 requires that a design specification be prepared for Class 1, 2, and 3 components such as pumps, valves, and piping systems. The design specification is intended to become a principal document governing the design and construction of these components and should specify loadings and their combinations; design, service, and test limits; and other design data inputs. ASME Code, Subarticle NCA-3260 also requires a design report for ASME Code Class1, 2, and 3 piping and components. In Topical Report ANP-10264NP-A, the applicant committed to construct all safety-related piping systems to applicable requirements of the ASME Code, Section III.

FSAR Tier 2, Section 3.9.3 and FSAR Tier 2, Table 1.8-2 require that a COL applicant that references the U.S. EPR design certification to prepare the design specifications and design reports for ASME Code Class 1, 2, and 3 components, piping, supports and core support structures that comply with and are certified to the requirements of Section III of the ASME Code. A review of FSAR Tier 2, Section 1.8.1, "COL Information Items," and FSAR Tier 2, Table 1.8-2 indicates that the design specifications and design reports will be made available to the NRC before fuel load. The piping ITAAC are included to address the final piping design. The staff reviewed the FSAR Tier 2, ITAAC information with plan for implementing design ITAAC and documented the staff's evaluation in Section 14.3.3 of this report.

3.12.4.1.4 Codes and Standards Evaluation Summary

On the basis of the staff's evaluation of Topical Report ANP-10264NP-A and the staff's review of the FSAR, the staff concludes that the piping systems important to safety are designed to quality standards commensurate with their importance to safety. The staff's conclusion is based on the following:

- The applicant satisfies the requirements of GDC 1 and 10 CFR 50.55a by specifying appropriate codes and standards for the design and construction of the safety-related piping and pipe supports.

- The applicant committed in the Topical Report ANP-10264NP-A to design all ASME Code Class 1, 2, and 3 pressure retaining components and their supports in accordance with ASME Code Section III using the 2004 Edition. In addition, ASME Code cases given in Topical Report ANP-10264NP-A meet the guidelines of RG 1.84, which have been reviewed and endorsed by the staff.

The COL applicants will make available to the staff design specifications and design reports demonstrating and documenting that as-designed piping and pipe support configurations adhere to the requirements of the design specifications as required by the ASME Code. The piping ITAAC are included to address the final piping design. The staff reviewed FSAR Tier 1, ITAAC information with plan for implementing design ITAAC and documented the staff's evaluation in Section 14.3.3 of this report.

3.12.4.2 *Piping Analysis Methods*

3.12.4.2.1 Experimental Stress Analysis

In FSAR Tier 2, Section 3.12.3.1, "Experimental Stress Analysis Methods," the applicant stated that experimental stress analysis methods will not be used to qualify piping for the U.S. EPR design. The staff finds this acceptable per SRP Subsection 3.12 II.A.i.

3.12.4.2.2 Response Spectrum Method with Uniform Support Motion

In FSAR Tier 2, Section 3.12.3.2, "Modal Response Spectrum Method," the applicant states that the uniform support response spectrum method used in the analyses for piping systems is addressed in Topical Report ANP-10264NP-A, Section 4.2. Topical Report ANP-10264NP-A, Section 4.2.2 describes the dynamic analysis procedure using the response spectrum (RS) method with uniform support motion (USM) using enveloped floor response spectra or independent support motion (ISM) using multiple floor response spectra.

In Topical Report ANP-10264NP-A, Section 4.2.2, the applicant states that the effects of the ground motion during an SSE event are transmitted through structures to the piping system at support and equipment anchorage locations. The floor response spectra are developed which represent the maximum acceleration responses of idealized single-degree-of-freedom damped oscillators as a function of natural frequency to the vibratory input motion of the structure. These floor response spectra are applied to the piping system at locations of structural attachment, such as support or equipment locations in each of three orthogonal directions. The total seismic response of the system is determined by combining the modal and spatial results.

In Topical Report ANP-10264NP-A, Section 4.2.2.2.1, the applicant also states that for a piping system supported at points with different dynamic excitations, an enveloped response spectrum of all attachment points is used in the USM method of analysis. For a given direction, the modal responses are combined in accordance with the methods described in Topical Report ANP-10264NP-A, Section 4.2.2.3. Following the modal combinations, the responses due to each of the three orthogonal earthquake motion inputs are combined using the square-root-of-the-sum-of-the-square (SRSS) method as stated in Topical Report ANP-10264NP-A, Section 4.2.2.4.

For U.S. EPR piping analyses, all modes with frequencies below the zero period acceleration frequency (i.e., cutoff frequency) are included. Above this frequency, in the rigid range, the effects of all additional modes are also included by the application of the missing mass correction as discussed in Topical Report ANP-10264NP-A, Sections 4.2.2.3.2 and 4.2.3. The cutoff frequency for a given spectra is the frequency at which the response curves for all damping values converge to the same acceleration value ZPA and remain at this value for all frequencies above this cutoff frequency. In Topical Report ANP-10264NP-A, Section 4.2, the applicant states that for the U.S. EPR the cutoff frequency is 40 Hz or as defined by RG 1.92, Figures 2 and 3, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," Revision 2. This approach conforms to the industry practice and SRP Section 3.9.2.

The staff notes that, for piping systems that are anchored and restrained to floors and walls of structures that have differential movements during a seismic event, additional forces and moments due to the differential supporting structure movements are induced in the system.

The applicant has committed to performing additional static analyses, as described in Topical Report ANP-10264NP-A, Section 4.2.2.5 to determine responses to these structure movements. The support displacements are imposed in a conservative manner using the static analysis method for each orthogonal direction with all dynamic supports active. This is known as seismic anchor movement (SAM) analysis. For the USM method of analysis, the results of the SAM analysis are combined with the results of the dynamic analysis by absolute sum method in accordance with SRP Section 3.9.2.

The staff reviewed Topical Report ANP-10264NP-A description of the RS method with USM and concluded that it conforms to the applicable guidelines in SRP Section 3.9.2, Subsection II.2 and RG 1.92, Revision 2, and is therefore acceptable.

3.12.4.2.3 Response Spectrum Method with Independent Support Motion

In FSAR Tier 2, Section 3.12.3.3, "Response Spectra Method (or Independent Support Motion Method)," the applicant states that the independent support motion response spectrum method used in the analyses for piping systems is addressed in Topical Report ANP-10264NP-A, Section 4.2.

As an alternative to the enveloped response spectrum method, the RS method with ISM may be used. The theory and development of the governing equations of motion for this method are the same as the USM RS method. Additional requirements associated with the application of this method are described in Topical Report ANP-10264NP-A, Section 4.2.2.2.2. This section states that when the ISM method of analysis is used, the following conditions must be met. First, a support group is defined by supports which have the same time history input. This usually means all supports located on the same floor, or portions of a floor, of a structure. Second, the responses from motions of supports in two or more different groups are combined by the absolute sum procedure. The modal and directional responses are then combined similar to those discussed for the USM RS method and as discussed in Topical Report ANP-10264NP-A, Sections 4.2.2.3 and 4.2.2.4, respectively.

In addition to the inertial response, the effects of relative support displacements, similar to that discussed in the USM method above, are performed to obtain the SAM responses, as discussed in Topical Report ANP-10264NP-A, Section 4.2.2.5.

The current staff position for modal and group combinations in the ISM method of analysis is presented in NUREG-1061, Volume 4, "Evaluation of Other Dynamic Loads and Load Combinations". For inertial or dynamic components, group responses are combined by the absolute sum method. Both modal and directional responses are combined by the SRSS method; the modal combination is performed without considering the effects of closely spaced frequencies. For SAM components, the maximum absolute responses from each directional input for each group are combined by the absolute sum method, and the directional responses are combined by the SRSS method. For the total response, the dynamic and SAM responses are combined by the SRSS rule.

In Topical Report ANP-10264NP-A, Sections 4.2.2.2 and 4.2.2.3, the applicant specifies that the combinations of modal responses and spatial components for systems analyzed using ISM are performed in conformance to the recommendations in NUREG-1061, Volume 4. Since this position meets the current staff position on ISM method of analysis, the staff finds this acceptable.

3.12.4.2.4 Time History Method

In FSAR Tier 2, Section 3.12.3.4, "Time History Method," the applicant states that Topical Report ANP-10264NP-A, Section 4.2.3 addresses the TH methods used in the analyses of the piping systems. Additional information about the TH method of analysis is provided in FSAR Tier 2, Section 3.7.2.

Typically, a TH analysis may be performed using either the modal superposition method, direct integration method in the time domain, or the complex frequency response method in the frequency domain. The applicant described its use of the modal superposition method in Topical Report ANP-10264NP-A, Section 4.2.3 for Seismic Category 1 and Category 2 piping systems.

FSAR Tier 2, Appendix C, Section 3C.4.2.2.1, "Reactor Coolant System Four Loop Structural Model Seismic Analysis," states that modal superposition TH solution technique is used to calculate the response of the linear RCS four loop structural model which includes RCL and pressurizer surge line piping. The direct step-by-step integration TH solution technique is used to calculate the response of the non-linear RCS four loop structural model.

Topical Report ANP-10264NP-A, Section 4.2.3 states that missing mass will be accounted for in the TH modal superposition analyses in accordance with RG 1.92, Revision 2, Appendix A. The mode shapes and frequencies are determined as they are in the RS analysis. The cutoff frequency for the determination of modal properties is 40 Hz or as defined by RG 1.92, Revision 2, Figures 2 and 3, as this is expected to encompass all of the important response frequencies of the system. Since the applicant's position is consistent with the staff's recommendation of RG 1.92, the staff finds this acceptable.

In Topical Report ANP-10264NP-A, the applicant included a requirement to perform time step studies for three of the ASME Code Class 1 attached piping models to determine the smallest integration time step required for convergence. The smallest integration time step required for convergence in these sample analyses will be used for all other ASME Code Class 1 piping that the applicant will analyze. The staff finds this approach is acceptable, because it will ensure convergence of the solution.

To account for uncertainties in the TH analyses for seismic loading, the applicant has included a peak shifting approach in Section 4.2.3 of Topical Report ANP-10264NP-A. This approach is similar to one described in Topical Report ANP-10264NP-A, Section 4.2.2.1.2, for the response spectra analyses. The staff has reviewed this approach as part of Topical Report ANP-10264NP-A FSER and finds this approach acceptable.

In Topical Report ANP-10264NP-A, Section 4.2.3, the applicant states that input time histories are analyzed for each of the three orthogonal directions of input motion and the three directional time history inputs are statically independent, and they are applied simultaneously in one analysis. The total response at each step is calculated as the algebraic sum of the three directional results. Alternatively, the three time histories may be applied individually and the responses combined by the SRSS method. On the basis that the applicant's analysis method is consistent with RG 1.92, Revision 1, Section B.2, the staff finds this combination method acceptable.

3.12.4.2.5 Inelastic Analysis

In FSAR Tier 2, Section 3.12.3.5, "Inelastic Analysis Method," the applicant states that inelastic analysis will not be used to qualify piping for the U.S. EPR design. The ASME Code design criteria are based on elastic theory. The staff finds this acceptable per SRP Subsection 3.12.II.A.v.

3.12.4.2.6 Small Bore Piping Method

In FSAR Tier 2, Section 3.12.3.6, "Small Bore Piping Method," the applicant defines small bore piping as ASME Code Class 1 piping that is 25.4 mm (1 in.) nominal pipe size (NPS) and smaller or ASME Code Class 2, Class 3, and Quality Group D piping that is 50.8 mm (2 in.) NPS or smaller. The applicant identifies two analysis options for the seismic analysis of these systems. The options include the response spectra methods and the equivalent static method.

The response spectrum method is an acceptable seismic analysis methodology for the analysis of both small bore and large bore piping. Section 3.12.4.2 of this report presents the staff's evaluation regarding the use of the response spectrum in the U.S. EPR.

In Topical Report ANP-10264NP-A, Section 4.2.4, the applicant describes the equivalent static load method as an alternative method of analyzing the effects of the SSE on piping system. That allows a simpler technique, but is known to yield more conservative results. The equivalent static load analysis method is used when a simplified analysis is considered with the mass of the piping and components as lumped masses at their center of gravity locations. The seismic response forces due to these masses are then statically determined by multiplying the contributing mass by an appropriate seismic acceleration coefficient at each location.

In Topical Report ANP-10264NP-A, Section 4.2.4, the applicant states that when the equivalent static load method is used, justification will be provided that the use of simplified model is realistic and the results are conservative. The staff finds this provision meets the recommendation as described in SRP Section 3.9.2.II.2.A(ii)(1), and is therefore acceptable.

In Topical Report ANP-10264NP-A, Section 4.2.4, the applicant states that to obtain an equivalent static load for multiple degree of freedom systems, the peak acceleration of the appropriate floor response spectra will be multiplied by 1.5; for single degree of freedom systems with a known fundamental frequency or rigid systems, a factor of 1.0 may be used with the highest spectral acceleration at that frequency or any higher frequency as may be the case for multiple peak input spectra. The staff finds these load factors meet the equivalent load factor as recommended in SRP Section 3.9.2.II.2.A(ii)(3), and is therefore acceptable.

In Topical Report ANP-10264NP-A, Section 4.2.4, the applicant states that this analysis is performed for all three directions of the seismic input motion. The results of these three analyses are then combined using the SRSS method, as in the response spectrum analyses. The relative motions of support locations (seismic anchor motions) are considered. The staff finds the applicant's SAM consideration meets the recommendation as described in SRP Section 3.9.2.II.2.A(ii)(2), and is therefore acceptable.

3.12.4.2.7 Non-Seismic/Seismic Interaction (II/I)

All Non-Seismic Category I piping (or other systems and components) should be isolated from Seismic Category I piping. This isolation may be achieved by designing a seismic constraint or barrier or by locating the two sufficiently apart to preclude any interaction. If it is impractical to isolate the Seismic Category I piping system; the adjacent Non-Seismic Category I system should be evaluated to the same criteria as the Seismic Category I system.

FSAR Tier 2, Section 3.12.3.7, “Non-Seismic/Seismic Interaction (II/I),” states that design and analysis considerations for the interaction of non-seismic and seismic piping will be in accordance with Topical Report ANP-10264NP-A, Section 4.4. Topical Report ANP-10264NP-A, Section 4.4 requires that for Non-Seismic Category I piping systems attached to Seismic Category I piping systems, the dynamic effects of the Non-Seismic Category I system are to be considered in the analysis of the Seismic Category I piping. In addition, the Non-Seismic Category I piping from the attachment point to the first anchor is evaluated to ensure that, under all loading conditions, it will not cause a failure of the Seismic Category I piping system. This is consistent with RG 1.29, RG Regulatory Position C.3.

In Topical Report ANP-10264NP-A, Section 4.4, the applicant also states that the primary method of protection for seismic piping is isolation (by physical separation or physical barrier as discussed in Topical Report ANP-10264NP-A, Section 4.4.1) from all non-seismically analyzed piping. Otherwise, the adjacent non-seismic piping is classified as Seismic Category II and analyzed and supported such that an SSE event will not cause an unacceptable interaction with the Seismic Category I piping. Alternatively, an interaction evaluation (as discussed in Topical Report ANP-10264NP-A, Section 4.4.2) may be performed to demonstrate that the interaction will not prevent the Seismic Category I piping system from performing its safety-related function.

The staff has previously approved the procedure noted above in the Topical Report ANP-10264NP-A, FSER. Therefore, the non-seismic/seismic interaction criteria referenced in FSAR Tier 2, Section 3.12.3.7, and discussed in Topical Report ANP-10264NP-A, Sections 4.4.1 and 4.4.2 is acceptable to the staff.

3.12.4.2.8 Seismic Category 1 Buried Piping

FSAR Tier 2, Section 3.12.3.8, “Seismic Category I Buried Piping,” states that Topical Report ANP-10264NP-A, Section 3.10 addresses the seismic criteria for buried piping systems.
In Topical Report ANP-10264NP-A, Section 3.10, the applicant states that ASME Code Class 2 and three Seismic Category I buried piping systems in the U.S. EPR will be analyzed for pressure, weight, thermal expansion and seismic loads using dynamic or equivalent static load methods. The acceptance criteria are the same as those used for non-buried piping systems described in Topical Report ANP-10264NP-A, Table 3-2 with additional consideration of the following differences:

- Deformations imposed by either seismic waves traveling through the surrounding soil or by differential deformations between the soil and anchor points and lateral earth pressures acting on buried piping will be considered.

- The effects of static resistance of the surrounding soil on piping deformations or displacements, anchor movements and pipe geometry will be considered using the theory of structures on elastic foundations.

- The effects of local soil settling will be considered when applicable.

- It is also assumed that soil liquefaction and fault displacement will be avoided.

- Seismic loads experienced by buried piping are primarily generated by soil strains and, therefore, are self-limiting and considered secondary in nature.

Design conditions, load combinations, and stress criteria to be used in the qualification of buried piping are addressed in Topical Report ANP-10264NP-A, Table 3-4. Topical Report ANP-10264NP-A, Section 3.10 states:

- The methods developed for the U.S. EPR buried piping meet the requirements of SRP Section 3.7.3, Revision 3; ASCE Standard 4-98 and ASCE Report,"Seismic Response of Buried Pipes and Structural Components."

- Live loads such as those imposed by trucks, rail, construction equipment or other construction conditions shall be considered in the buried pipe analysis and design. American Lifeline Alliance Report, "Guideline for Design of Buried Steel Pipe," may be used to determine impact factor and surface loads effects on buried pipe.

- For utilities buried below groundwater table, vertical force due to buoyancy should be considered.

- Buried piping systems must be designed to meet the external pressure load criteria of ASME Code NC/ND-3133.

- ASME Code Class 2 and 3 buried piping design conditions, load combination and stress criteria are provided in Topical Report ANP-10264NP-A, Table 3-4.

- The staff finds the approach to addressing Seismic Category I buried pipe acceptable, because inertial effects due to earthquake, static resistance of the surrounding soil on piping deformations, differential movement of seismic anchors, and applicable effects of local soil settlements and arching are adequately considered in the proposed analyses of the buried piping.

3.12.4.2.9 Piping Analysis Methods Evaluation Summary

On the basis of these piping analysis method evaluations, the staff concludes that the analysis methods to be used for all Seismic Category 1 piping systems, as well as non-Seismic Category I piping systems that are important to safety, are acceptable. The analysis methods utilize piping design practices that are commonly used in the industry and provide an adequate margin of safety to withstand the loadings as a result of normal operating, transient, and accident conditions. The staff concludes that the applicant satisfies the requirements of GDC 2 by specifying appropriate analysis methods for designing piping and pipe supports against seismic loads.

3.12.4.3 *Piping Modeling Techniques*

The staff has evaluated the piping modeling techniques used in the design of the U.S. EPR, as presented in the FSAR, as described below.

3.12.4.3.1 Computer Codes

FSAR Tier 2, Section 3.9.1.2 provides information on the computer programs used in the U.S. EPR analysis. The staff reviewed FSAR Tier 2, Sections 3.9.1.2, 3.12, and FSAR Tier 2, Appendix 3C.6 for the computer programs used in the dynamic and static analysis of mechanical loads, stresses, and deformations, and in the hydraulic load analyses of piping, components and supports. The staff noted that GT STRUDL, CASS, and EBDynamics were identified as part of FSAR Tier 2, Sections 3.12 and Appendix 3C but were not identified in FSAR Tier 2, Section 3.9.1.2. Therefore, in RAI 161, Question 03.12-11, the staff requested that the applicant revise FSAR Tier 2, Section 3.9.1.2 for consistency. In a February 27, 2009, response to RAI 161, Question 03.12-11, the applicant stated that FSAR Tier 2, Section 3.9.1.2 will be revised to address this RAI. The staff confirmed that Revision 1 of the FSAR, dated May 29, 2009, contains the changes committed to in the RAI response. Accordingly, the staff finds that the applicant has adequately addressed this issue and, therefore, the staff considers RAI 161, Question 03.12-11 resolved.

The applicant also stated that AREVA computer codes are certified (or verified), controlled, and maintained per administrative procedure. Files are maintained that provide the software author, source code, dated version, program description, extent and limit of the program application, and the solutions to the test problems described above. Based on the applicant's description, the staff finds the computer codes used for U.S. EPR meet the recommendation as described in SRP Section 3.9.1.II.2.C. Section 3.9.1 of this report contains a more detailed discussion of this issue.

As stated in COL Information Item 3.12-2, the COL applicant will either use a piping analysis program based on the computer codes described in FSAR Tier 2, Section 3.9.1 and FSAR Tier 2, Appendix 3C or will implement an NRC-approved benchmark program using models specifically selected for the U.S. EPR. The staff finds this approach acceptable, because other computer programs have not been verified.

3.12.4.3.2 Dynamic Piping Model

In Topical Report ANP 10264NP-A, Section 5.2, the applicant describes the procedures used for analytical modeling of piping systems. For dynamic analysis, the piping system is idealized as a three dimensional model using finite element analysis programs. The analysis model consists of a sequence of nodes connected by pipe elements (both straight and bend elements) with stiffness properties representing the piping and other inline components. Nodes are typically modeled at points required to define the piping system geometry as well as lumped mass locations, support locations, locations of structural or load discontinuities, and at other locations of interest along the piping. System supports are idealized as springs with appropriate stiffness values for the restrained direction.

In the dynamic mathematical model, the applicant also states that the distributed mass of the system, including pipe, contents (fluid or gas), and insulation weight, is represented as either a consistent (distributed) mass or lumped masses placed at each node. For the latter case, in order to adequately determine the dynamic response of the system, elements may be subdivided and additional mass points added. The minimum number of degrees of freedom in the model is to be equal to twice the number of modes with frequencies below the ZPA frequency. Maximum mass point spacing may be no greater than one half of the span length of a simply supported beam with stiffness properties and distributed mass equal to that of the piping cross-section and a fundamental frequency equal to the cutoff frequency. The applicant further states that concentrated weights of in-line components, such as valves, flanges, and instrumentation, are also modeled as lumped masses. Torsional effects of eccentric masses are included in the analysis. For rigid components (those with natural frequencies greater than the ZPA cutoff frequency), the lumped mass is modeled at the center of gravity of the component with a rigid link to the pipe centerline. Flexible components (those with natural frequencies less than the ZPA cutoff frequency) are included in the model using beam elements and lumped mass locations to represent the dynamic response of the component.

Additionally, a portion of the weight of component type supports (such as snubbers, struts, spring hangers, etc.) is supported by the pipe and is considered in the piping analysis model. The mass contributed by the support is included in the analysis when it is greater than 10 percent of the total mass of the adjacent pipe span (including pipe, contents, insulation and concentrated masses). The adjacent span is defined as the piping including the applicable support and bounded by the adjacent restraint on each side of this support in each direction. The applicant also states that because the mass of a given support will not contribute to the piping response in the direction of the support, only the unsupported directions need to be considered, unless the support is flexible in the supported direction.

A review of the impact of contributing mass of supports on the piping analysis will need to be performed by the COL applicant(s) following the final support design to confirm that the mass of the support is no more than 10 percent of the mass of the adjacent pipe span. This is identified as COL-Action Item 5 in Topical Report ANP-10264NP-A, Table 1-1 and is also given in FSAR Tier 2, Table 1.8-2 as COL Information Item 3.12-1.

The staff reviewed Topical Report ANP-10264NP-A, Section 5.2 and found these requirements acceptable because they comply with SRP Section 3.7.3 and good engineering practice.

In Topical Report ANP-10264NP-A, Section 5.4, the applicant discusses the model boundaries based on defining terminal points. Piping system analysis models are typically terminated by one of three techniques: (1) Structural boundaries; (2) termination based on decoupling criteria; or (3) termination by model isolation methods. Structural boundaries and the use of decoupling criteria are the preferred methods. However, after applying these first two methods, further division of the piping system may be desired to create more manageable models for analysis. This may be accomplished using the model isolation methods. The structural boundary and the model isolation methods are discussed here. The decoupling criteria are discussed in Section 3.12.4.3.4 of this report.

The applicant states that structural model boundaries, such as equipment nozzles or penetrations, provide isolation of the effects of the piping on one side of the boundary to the piping on the opposite side. For large piping systems, the applicant states that an in-line anchor is used as intermediate structural boundary during design to allow for further division of the analysis model. The applicant also stated that the addition of an in-line anchor generally creates stiffer piping systems and may cause significant increases in stress and support loads on lines with high thermal movements. Additionally, the use of in-line anchors on high energy lines adds additional postulated terminal end pipe rupture locations. Therefore, additional in-line anchors are only added if they are determined to be practical.

In Topical Report ANP-10264NP-A, Section 5.4.3, the applicant describes two model isolation methods, namely overlap region method and influence zone method, to divide large seismic piping systems that cannot be separated by structural methods or decoupling criteria. Both these methods are similar in technique in that a section of the piping system is used as the boundary of the models. This section of the system is defined such that the effects of the piping beyond one end of the region do not significantly affect the piping beyond the opposite end of the region. In Topical Report ANP-10264NP-A, Section 5.4.3.1, the applicant suggests for the overlap region method that, as a minimum, an overlap region must contain at least four seismic restraints in each of three perpendicular directions and at least one change in direction. The overlap region should be selected in a rigid area of the piping system and is modeled in two or more piping analyses. A dynamic analysis of the overlap region shall be made with pinned boundaries extended beyond the overlap region either to the next actual support or to a span length equal to the largest span length within the region. The fundamental frequency determined from this analysis shall be greater than the frequency corresponding to the ZPA.

In Topical Report ANP-10264NP-A, Section 5.4.3.2, the applicant states that the main difference between the influence zone and the overlap region is that in using the influence zone, all piping and supports are qualified by a single model. This is achieved by first determining the qualification boundary between models. Each model is then extended to a termination point such that the response of the piping at the termination of the model will not influence the response of the piping within the qualification boundary. The influence zone is then defined by the section of piping between the qualification boundary and the model termination point. However, when using this methodology versus the overlap region method, a significantly larger section of piping may be required to be included in two or more models.

The overlap methodology provided in Topical Report ANP 10264NP-A, Section 5.4.3.1 conforms to the recommendations of NUREG/CR-1980. The zone of influence (ZOI) method is provided as an option when the requirement for a rigid section of piping cannot be met in order to use the overlap methodology. In this method, all piping must be modeled to a point where boundary conditions and loadings no longer impact the piping being qualified. This will typically be more piping than is required by the overlap method and the validity of the boundary is required to be demonstrated during the analysis. Since these methods use four seismic restraints in each of three perpendicular directions, and at least one change in direction conforms to the recommendations in NUREG/CR-1980, the staff finds this acceptable.

3.12.4.3.3 Piping Benchmark Program

In Topical Report ANP-10264NP-A, Section 5.3, the applicant states that pipe stress and support analysis will be performed by the COL applicant. If the COL applicant chooses to use a piping analysis program other than those given in Topical Report ANP-10264NP-A, Section 5.1, the applicant will implement the benchmark program using models specifically selected for the U.S. EPR. This is identified as COL-Action Item 6 in Topical Report ANP-10264NP-A, Table 1-1. This item is also given in FSAR Tier 2, Table 1.8-2 as COL Information Item 3.12-2.

In accordance with guidance in SRP Section 3.12, the COL applicants who will complete the piping analysis and finalize the piping designs will verify their computer programs in accordance with the NRC benchmark program specific to the standardized plant design. Under a piping benchmark program, the COL applicant applies its computer program to construct a series of selected piping system mathematical models that are representative of the standard plant piping designs. The results of the analyses must be compared with the results of independent benchmark problem analyses developed by the staff. The COL applicant must document and submit any deviations from these values, as well as justification for such deviations, to the staff for review and approval before initiating final piping analyses. The benchmark program provides assurance that the computer program used to complete the piping design and analyses produces results that conform to results considered acceptable to the staff.

Additionally, the applicant performed its piping computer code BWSPAN benchmark verification using NRC piping benchmark BM3 model. The benchmark model BM3 was documented and established in NUREG/CR-6645, “Reevaluation of Regulatory Guidance on Modal Response Combination Methods for Seismic Response Spectrum Analysis.” The staff reviewed the benchmark result and compared with result of NUREG/CR-6645. On the basis of result comparison of frequencies, support reactions, and pipe end moments, the staff concludes that the piping analysis benchmark is acceptable.

3.12.4.3.4 Decoupling Criteria

In Topical Report ANP 10264NP-A,Section 5.4.2, the applicant defines smaller branch lines as those lines that can be decoupled from the analytical model used for the analysis of the main run piping to which the branch lines are attached. Branch lines can be decoupled when the ratio of run to branch pipe diameter is 3 to 1, or greater, or moment of inertia is 25 to 1, or greater; and with sufficient flexibility to prevent restraint of movement of the main run pipe. The decoupling criteria may also be applied for in-line pipe size changes (such as at a reducer or reducing insert). The applicant’s decoupling criteria meet the technical position on industry recommendation in the Welding Research Council (WRC) Bulletin 300, which the staff has found to be acceptable as documented in NUREG-1793. On this basis, the staff finds this acceptable.

In Topical Report ANP-10264NP-A, Section 5.5, the applicant provides the criteria for analyzing non-seismic piping affecting the Seismic Category I piping. The model boundary at a non-seismic/seismic piping interface may consist of structural isolation, decoupling, or model isolation methods similar to those discussed in Topical Report ANP-10264NP-A, Section 5.4. However, additional considerations are required to ensure that the dynamic effects of the non-seismic piping on the Seismic Category I piping are considered.

The applicant states that the Seismic Category I design requirements extend to the first seismic restraint beyond the seismic system boundary. The non-seismic piping and supports beyond this location that impact the dynamic analysis of the Seismic Category I piping are reclassified as Seismic Category II and included in the model. The extent of piping classified as Seismic Category II may be bounded by the same three methods discussed in Topical Report ANP-10264NP-A, Section 5.4. The staff evaluation of these sections is discussed in this section as well as in Section 3.3.2 of this report. The applicant states that, when structural boundaries are used to terminate the Seismic Category II region, all piping and supports between the Seismic Category I design boundary and the structural anchor, or the final restraint of a restrained elbow or tee, are classified as Seismic Category II. When the decoupling criteria are used, all piping and restraints beyond the Seismic Category I boundary up to the decoupled location are classified as Seismic Category II.

In all three cases cited in Topical Report ANP-10264NP-A, Section 5.5, the Seismic Category II portion of the system is analyzed with the Seismic Category I piping for the SSE load case as well as loads resulting from the potential failure of the non-seismic piping and pipe supports. This is accomplished by the application of a plastic moment in each of three orthogonal directions at the termination of the model. Each moment is applied and evaluated in a separate analysis and the results of the three analyses are enveloped.

Since all methods described in Topical Report ANP-10264NP-A provide assurance that the Seismic Category I piping is adequately designed to include the effects from the non-seismic piping during an earthquake, the staff determined all three methods to be acceptable.

3.12.4.3.5 Piping Modeling Techniques Evaluation Summary

On the basis of the discussions in the above subsections and evaluation of Topical Report ANP-10264NP-A, Section 5.0, the staff concludes that U.S. EPR design control measures are acceptable to ensure quality of computer programs and piping modeling methods. The staff's conclusion is based on the following:

- The applicant satisfies the requirements of GDC 2 and acceptance criteria described in SRP 3.12.II.B by providing criteria for the seismic design and analysis of all Seismic Category I piping and pipe supports using prescribed modeling techniques and design methods that are in conformance with generally recognized engineering practice.

- The applicant meets 10 CFR Part 50, Appendix B by demonstrating the applicability and validity of the computer programs for performing piping seismic analysis.

- Computer programs to be used by the COL applicant to complete its analyses of the U.S. EPR piping systems will be verified and validated.

3.12.4.4 *Pipe Stress Analysis Criteria*

GDC 1 requires that the piping and pipe supports should be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. 10 CFR Part 50, Appendix B requires that design quality should be controlled for ensuring structural and functional integrity of Seismic Category I components. GDC 2 requires that the piping and pipe supports should withstand the effects of earthquake loads. GDC 4 requires that the piping and pipe supports should withstand the dynamic effects of equipment failures including missiles and blowdown loads associated with the loss-of-coolant accident. The basis for design of ASME Code Class 1, 2, and 3 piping components sufficiently defines the design and service load combinations, including the system operating transients, and associated design and service stress limits considered for all normal, abnormal and accident conditions.

GDC 14 requires that the RCPB components should be designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross failure. GDC 15 requires that the reactor coolant system should be designed with sufficient margin to assure that the design conditions are not exceeded.

3.12.4.4.1 Seismic Input Envelop versus Site-Specific Spectra

In Topical Report ANP-10264NP-A, Section 4.2.1, the applicant states that the response spectra curves for the U.S. EPR are being developed to cover an appropriate range of possible soil conditions with the ground motion anchored to peak ground acceleration (PGA) of 0.3g. The PGA in the vertical design ground motion is equal to the horizontal design ground motion

PGA. The input design ground motion response spectra for the U.S. EPR standard plant is documented in FSAR Tier 2, Section 3.7.2.5, “Development of Floor Response Spectra,” and is evaluated in Section 3.7.2.5 of this report.

The staff recognizes that the site-enveloping response spectra for the U.S. EPR plant would contain conservatisms that may be excessive for certain site-specific conditions. If amplified building response spectra are generated using site-dependent properties, then the approach and method used must be submitted to the staff for review and approval as part of the COL application. In Topical Report ANP 10264NP-A, Section 4.2.2.1, the applicant describes the method of analysis to be used in developing the floor response spectra for the structures using the guidelines provided in RG 1.122, Revision 1. The applicant also states that either the response spectra peak broadening method or peak shifting method is used to address the uncertainties in the structural frequencies. The staff’s evaluation of the development of seismic input and floor response spectra is documented in Section 3.7.2 of this report.

3.12.4.4.2 Design Transients

In FSAR Tier 2, Section 3.9.1.1, the applicant discussed the design transient for ASME Code Class 1 components and supports. FSAR Tier 2, Table 3.9.1-1 lists the design transient for five groups of plant operating conditions and the number of cycles for each event within the group that are normally used for fatigue evaluation of components including ASME Code Class 1 piping systems.

The staff evaluated FSAR Tier 2, Section 3.9.1.1 and documented its evaluation in Section 3.9.1 of this report.

3.12.4.4.3 Loading and Load Combinations

The applicant described loads and load combinations in Topical Report ANP-10264NP-A, Section 3.3 for piping stress analysis and Topical Report ANP-10264NP-A, Section 6.3 for pipe support analysis. Loadings applicable to the U.S. EPR piping design include:

- Pressure
- Deadweight
- Thermal expansion (includes thermal anchor movements)
- Seismic (includes seismic anchor movements)
- Fluid transients (includes relief valve thrust, valve closure and water/steam hammer)
- Wind/tornado (identified as Topical Report ANP-10264NP-A, Table 1-1, COL Action Item 3)
- Design basis pipe breaks (includes pipe whip, jet impingement, dynamic effects)
- Thermal and pressure transients
- Hydro tests

The applicant states that the zero thermal load temperature is 21.1 °C (70 °F), and that piping systems with an operating temperature equal to or less than 65.6 °C (150 °F) do not require a thermal analysis. The staff reviewed the criteria described above and noted that the value of minimum temperature for thermal expansion analysis is consistent with industry practice and this position has been approved and used in many nuclear power plants. Based on these precedents, the staff finds this acceptable.

The applicant also states that the ground motion of the operating basis earthquake for the U.S. EPR is equal to one-third of the ground motion of 0.3g for the SSE. In case of a seismic event greater than the OBE ground motion, in accordance with 10 CFR Part 50, Appendix S, plant shut down is required and Seismic Category I piping and supports are required to be inspected to ensure no loss of function or physical damage has occurred. Both inertial and SAM effects are considered as Service Level D loads, since the U.S. EPR is not designed to an OBE loading. This conforms to SECY-93-087, and is therefore acceptable to the staff.

In Topical Report ANP 10264NP-A, Section 3.3.1.7, the applicant states that design basis pipe break loads must be evaluated for the appropriate service condition. However, pipe breaks in the RCL, main steam and pressurizer surge lines which meet the leak-before-break (LBB) size criteria are eliminated from consideration based on LBB analysis. The staff evaluated LBB criteria and documented its evaluation in Section 3.6.3 of this report.

In Topical Report ANP-10264NP-A, Section 3.3.2, the applicant further stated that using the methodology and equations from the ASME Code, pipe stresses are calculated for various load combinations. The ASME Code includes design limits for design conditions, Service Levels A, B, C, and D and testing. Design conditions, load combinations, and stress criteria for ASME Code Class 1 piping are given in Topical Report ANP-10264NP-A, Table 3-1 and Table 3-2 for ASME Code Class 2 and 3 piping.

The staff concludes that appropriate combinations of normal, operating transients and accident loadings are specified to provide a conservative design envelope for the design of piping systems. The load combinations conform to the guidelines provided in SRP Section 3.9.3 and the staff position associated with the SRM on SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” for elimination of an OBE. Therefore, the staff finds the load combination for the U.S. EPR piping design acceptable.

The staff reviewed FSAR Tier 2, Appendix 3C.4.1.3, “Steady State Flow,” and noted that the applicant states that, under 100 percent power steady flow conditions, the RCS components and piping are subjected to flow loads at locations where flow direction or flow area change. In RAI 161, Question 03.12-7, the staff requested that the applicant describe the method for applying this load in its analysis model and how to apply the results (stress, support load) of this loading.

In a February 27, 2009, response to RAI 161, Question 03.12-7, the applicant stated that the 100 percent power steady state flow loads are obtained from the RCS four loop hydraulic analysis using CRAFT2. The steady state axial hydraulic forces are transferred to the structural program by orienting the force time-histories using the post-processing program BWHIST. Once oriented and applied to the structural model of the RCS (in BWSPAN), the loads are evaluated on the piping, components, and supports using principles of statics. The steady state flow load is included in the piping, component, and supports stress analysis as an applied mechanical load (see FSAR Tier 2, Table 3.9.3-1 and Topical Report ANP-10264NP-A, Table 3-1t). The applicant indicated that steady state flow loads are considered as part of the applied mechanical loads, and applied mechanical loads are considered in load combinations and acceptance criteria. The staff finds this acceptable, because steady state flow loads are appropriately considered in design.

The staff reviewed the loading conditions and noted that the applicant did not address inter-building settlement difference in piping design. In RAI 161, Question 03.12-10, the staff requested that the applicant clarify if building settlement cases are considered for piping design. In a February 27, 2009, response to RAI 161, Question 03.12-10, the applicant stated that building settlement cases are considered in the piping design as non-repeated anchor movement load cases. FSAR Tier 2, Section 3.12.5.3, “Loadings and Load Combinations,” refers to Topical Report ANP-10264NP-A, Section 3.3 for the loads and load combinations used for piping design. Topical Report ANP-10264NP-A, Tables 3-2 and 3-4 include non-repeated anchor movement loads as a normal/upset loading condition. The staff concurs that building settlement loading is a non-repeated anchor movement load case. The staff finds this acceptable, because the applicant has appropriately addressed non-repeated anchor movement loads in Topical Report ANP-10264NP-A.

In RAI 211, Question 03.12-16, the staff requested that the applicant clarify the support load combination for this building settlement case. In a May 26, 2009, response to RAI 211, Question 03.12-16, the applicant stated that building settlement loads and other non-repeated anchor movement loads are combined with normal loads for pipe support design and stated that FSAR Tier 2, Section 3.12.5.3 will be revised accordingly. The staff reviewed the applicant’s response to RAI 211, Question 03.12-16, and verified that FSAR Revision 2 has been updated accordingly. Therefore, the staff considers RAI 211, Question 03.12-16 resolved.

3.12.4.4.4 Damping Value

The staff reviewed Topical Report ANP-10264NP-A, Section 4.2.5 and FSAR Tier 2, Section 3.7.1. Topical Report ANP-10264NP-A, Section 4.2.5 states that per RG 1.61, “Damping Values for Seismic Design of Nuclear Power Plants,” Revision 1, four percent damping value will be used for U.S. EPR piping design. The staff determines that the applicant’s action meets the recommendation of SRP Section 3.12. However, FSAR Tier 2, Section 3.7.1.2, “Percentage of Critical Damping Values,” states that the analysis of piping that uses the uniform support motion response spectrum method is performed with five percent damping. FSAR Tier 2, Table 3.7.1-1, “Damping Values for Safe Shutdown Earthquake,” also states that five percent damping is used for piping analysis. Therefore, in RAI 161, Question 03.12-11, the staff requested that the applicant make an appropriate revision to resolve the difference between FSAR Tier 2, Table 3.7.1-1 and Topical Report ANP-10264NP-A.

In a February 27, 2009, response to RAI 161, Question 03.12-11, the applicant stated that, as noted in the FSER for Topical Report ANP-10264NP-A, Section 3.4.4, the applicant agreed to use damping values provided in RG 1.61, Revision 1 for uniform support motion response spectrum analysis, independent support motion response spectrum analysis, and time history analysis. FSAR Tier 2, Section 3.7.1.2 and Table 3.7.1-1 will be revised accordingly. The staff reviewed FSAR Tier 2, Revision 2, Section 3.7.1.2 and Table 3.7.1-1. FSAR Tier 2, Revision 2, Section 3.7.1.2 states that piping analyzed for the U.S. EPR uses damping in accordance with

RG 1.61, Revision 1. The staff finds this acceptable, because the proposed damping value conforms to RG 1.61.

Topical Report ANP-10264NP-A, Section 4.2.5, the applicant stated that when composite modal damping is applied in a dynamic analysis (either time history or response spectra), each model subgroup (piping, supports, equipment, etc) is assigned an appropriate damping value per RG 1.61, Revision 1. The equivalent modal damping matrix or composite modal damping matrix is calculated for each mode. The staff reviewed the applicant's position for the composite damping. On the basis that the methods used meet the requirement of SRP Section 3.7.2, the staff finds this acceptable.

3.12.4.4.5 Combination of Modal Responses

The staff reviewed Topical Report ANP-10264NP-A, Section 4.2.2.3 which addressed the modal combination methods used in response spectrum analyses for piping. The applicant stated that the inertial response of a piping system in a seismic response spectrum analysis is considered in two parts. First, the modal analysis calculates the peak response of the piping system for all low frequency (or non-rigid) modes with seismic excitation frequencies up to the frequency (known as the cutoff frequency) at which spectral accelerations return to the ZPA. Modal combinations associated with this part are evaluated in this section. Second, at modal frequencies above the cutoff frequency, pipe members are considered rigid. The acceleration associated with these rigid modes is usually small. However, in certain situations the response to high frequency modes can significantly affect support loads, particularly axial restraints on long piping runs. To account for these effects, the applicant presented a method of calculating the missing mass correction in Topical Report ANP-10264NP-A, Section 4.2.2.3.3. The staff evaluation of this methodology is documented in Section 3.12.4.4.6 of this report.

In Topical Report ANP-10264NP-A, Section 4.2.2.3, the applicant states that RG 1.92, Revision 2, is used for combining of modal response. On the basis that the methods conform to NRC guidelines, the staff finds this acceptable.

3.12.4.4.6 High-frequency Modes

In Topical Report ANP-10264NP-A, Section 4.2.2.3, the applicant states that response from high frequency modes must be included in the response of the piping system. Guidance for including the missing mass effects is provided in RG 1.92, Revision 2 for USM. The U.S. EPR will use the method present in RG 1.92 or the left-out-force method for calculating and applying the response of the high frequency modes based on applying a missing mass correction.

The staff reviewed the above two methods presented by the applicant. The applicant using the method in RG 1.92, Appendix A is consistent with the staff guidance, and is therefore acceptable. The left-out-force method is also acceptable to the staff as documented in NUREG-1793.

The applicant also states that as an alternative, when using the Lindley-Yow method, the Static ZPA method for calculating a total mass rigid response presented in RG 1.92, Section C.1.4.2 may be used. The staff finds this acceptable on the basis that this alternative is consistent with the staff guidance in RG 1.92.

3.12.4.4.7 Fatigue Evaluation for ASME Code Class 1 Piping

In Topical Report ANP-10264NP-A, Section 3.4.1, the applicant states that ASME Code Class 1 piping will be evaluated for the effects of fatigue as a result of pressure and thermal transients and other cyclic events including earthquakes. The staff reviewed the applicant's determination for the cycle numbers for the OBE stress range. The applicant followed SRM on SECY-93-087 to determine the number of OBE stress cycles. On this basis, the staff finds this acceptable.

The applicant states that the environmental effects of the reactor coolant on fatigue will be accounted for in the ASME Code Class 1 piping fatigue analyses using methodology described in RG 1.207. On the basis that the methodology conforms to the NRC guideline, the staff finds this acceptable.

3.12.4.4.8 Fatigue Evaluation for ASME Code Class 2 and 3 Piping

In Topical Report ANP 10264NP-A, Section 3.4.2, the applicant states that ASME Code Class 2 and 3 piping is evaluated for fatigue due to thermal cycles by following the requirements of the ASME Code. The staff concludes that this is acceptable, because the proposed fatigue evaluation meets 10 CFR 50.55a ASME Code mandatory requirements. The applicant also states that environmental impact on fatigue of ASME Code Class 2 and 3 piping will follow guidelines established by the NRC at the time of analysis. Therefore, the staff finds this acceptable.

3.12.4.4.9 Thermal Oscillation in Piping Connected to the Reactor Coolant System

The applicant addressed thermal oscillation issues identified in NRC Bulletin 88-08 in FSAR Tier 2, Section 3.12.5.9, "Thermal Oscillations in Piping Connected to the Reactor Coolant System." The applicant described a two-step approach based on thermal management guidelines provided in EPRI Reports TR-1011955, "Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines" and TR-103581, "Thermal Stratification, Cycling, and Striping (TASCS)."

The first step included identification, screening, and evaluation of thermal cycling for normally stagnant, non-isolable lines attached to the RCS. The second step was an evaluation of susceptible lines using operational test data taken at AREVA-designed foreign plants on similar piping configurations.

The applicant follows the EPRI generic methodology and indicated that thermal stratification will occur in the following lines:

- Residual Heat Removal/Safety Injection System/Extra Borating System (RHR/SIS/EBS) injection piping (for all four divisions)

- RHR/SIS suction piping (for two of four divisions)

The applicant stated that the inherent uncertainty in predicting detailed thermal hydraulic phenomena for piping systems is indicated by the differences in swirling penetration distances predicted by the generic EPRI methodology and the applicant's operational test data.
The applicant stated that specific measurements taken at AREVA designed foreign plants on piping configurations that are representative of U.S. EPR piping systems indicated that smaller

range and shorter vortex penetration occurred than predicted by the EPRI methodology. Therefore, the applicant concluded that test data shows thermal stratification does not occur in any horizontal segment of the U.S. EPR RCS attached piping.

The applicant also stated that the differences in swirling penetration distances between the generic EPRI methodology and the applicant's measurements indicate the inherent uncertainty in predicting detailed thermal phenomena for piping systems and the need to instrument/monitor conditions during initial plant operation. The RCS attached piping will be instrumented and monitored during the first cycle of the first U.S. EPR initial plant operation to verify that operating conditions have been considered in the design unless data from a similar plant's operation demonstrates that thermal oscillation is not a concern for piping connected to the RCS.

The staff reviewed the applicant's claims and determined that RCS attached piping has to be monitored to confirm its stratification susceptibility. The staff noted that this monitoring activity is not given in FSAR Tier 2, Table 1.8-2 as a COL information item. Therefore, in RAI 161, Questions 03.12-1 and RAI 211, Question 03.12-12, the staff requested that the applicant clarify who is responsible for this monitoring activity and requested that the applicant provide a description of the monitoring program/methodology for confirming the integrity of the RCS attached piping.

In a February 27, 2009, response to RAI 161, Question 03.12-1 the applicant stated that since the test information in the referenced FSAR section is incorporated by reference (IBR) in the COL, it becomes a COL responsibility. Tests to confirm system integrity are addressed in FSAR Tier 2, Section 14.2 (see Test Nos. 30, 32, 33, 35, 37, 168,186, 195, and 197) and the Technical Specifications (e.g., Surveillance Requirement (SR) 3.4.12, SR 3.4.14). The staff reviewed and evaluated test/monitoring program and documented the review in Section 14.2 of this report. The applicant also provided a May 26, 2009, response to RAI 211, Question 03.12-12 to clarify that this activity will be performed by the COL applicant and FSAR Tier 2, Table 1.8-2 will be revised. The staff reviewed and verified that Table 1.8-2 was revised in FSAR Tier 2, Revision 2 to clarify that the COL applicant is responsible for this activity. Therefore, the staff considers RAI 161, Question 03.12-1 and RAI 211, Question 03.12-12 resolved.

The staff also noted that the applicant's thermal stratification discussion described the RCS non-isolable piping flow turbulent penetration without mentioning valve leakage cases. In RAI 161, Question 03.12-9, the staff requested that the applicant provide an approach to address NRC Bulletin 88-08 issues and ensure that valve leakage cases are evaluated and addressed.

In a February 27, 2009, response to RAI 161, Question 03.12-9, the applicant stated that it has analyzed and evaluated thermal stratification issues for RCS non-isolable piping by considering valve leakage as discussed in NRC Bulletin 88-08 and the thermal management guidelines provided in EPRI TR-1011955 and TR-103581.

The applicant also responded as follows:

The EPRI criteria used in the evaluation of the U.S. EPR piping systems attached to the RCS for susceptibility to thermal stratification due to valve leakage are summarized below:

- For piping that extends vertically upward from the RCS followed by a horizontal section, a cold water source from leaking valve must exist in order to have the potential for thermal oscillations.
    - o It is assumed that any single valve could leak. Sections with two or more valves in series are assumed to not create enough leakage to cause thermal oscillations.
    - o There is a high pressure differential capable of forcing leakage.
    - o There is a temperature difference between the fluid in the non-isolable piping section and the fluid from the leakage source.
- Sections of piping that are less than or equal to 25.4 mm (1 in.) nominal pipe size are not susceptible to thermal stratification.
- If a sufficient continuous flow rate exists within the RCS attached piping, thermal oscillations will not occur.
- For any un-isolable piping attached to the RCS with the first vertical-to-horizontal elbow L/Di greater than 20, thermal stratification does not occur in the branch line considered, since the swirl penetration does not reach the horizontal segment of the first isolation valve or check valve. For this term, L is defined as the length from inside face of the RCS to a location on the branch pipe, and Di is the branch line inside diameter.
- Piping oriented downward from the RCS followed by a horizontal section is not susceptible to thermal stratification due to valve leakage, based on operating plant experience presented in EPRI TR-1011955 Appendix B.

The RCS-attached piping out to first normally-closed valve including the safety-injection system/residual heat removal lines, the normal spray lines, the pressurizer surge line, and the chemical and volume control system (CVCS) let down and charging lines have been evaluated. Of these systems, the CVCS let down and charging lines both have a non-isolable section attached to top of the RCS loops followed by a horizontal portion with a check valve. The length of the first upward vertical-to-horizontal elbow piping connected to the RCS is greater than 22 Di for the CVCS let down and charging lines, where Di is the inside diameter of the CVCS piping. The EPRI evaluation criterion based on the geometry (fourth bullet above) indicates that thermal stratification from valve leakage will not occur in the CVCS let down and charging lines.

As documented in NRC Bulletin 88-08 (including supplements) and EPRI Report TR-1011955, safety-injection systems at operating plants (e.g., Farley, Tihange, Dampierre) have been susceptible to valve leakage-induced cyclic thermal stratification. The U.S. EPR design incorporates lessons learned from this operating experience in that the injection line (SIS/RHR) continually rises in elevation from the check valve; therefore, it is not susceptible to valve leakage-induced cyclic thermal stratification."

The staff reviewed the applicant's February 27, 2009, response to RAI 161, Question 03.12-9 and in follow-up RAI 211, Question 03.12-13, the staff requested that the applicant further clarify its response as below:

- The applicant’s response (Bullet 4) stated that for any un-isolable piping attached to the RCS with the first vertical-to-horizontal elbow L/Di greater than 20, thermal stratification does not occur in the branch line considered. However, EPRI TR-1011955 (which references EPRI report MRP-132, “Materials Reliability Program Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines”) stated that the swirl penetration distance is a function of the line diameter with greater penetration in larger diameter lines. Thermal stratification is dependent on the swirl penetration distance which is not a constant of L/Di=20. The staff requested that the applicant clarify the difference.

- The applicant‘s response (Bullet 5) stated that piping oriented downward from the RCS followed by a horizontal section is not susceptible to thermal stratification due to valve leakage, based on operating plant experience presented in the EPRI guidelines Appendix B. The piping oriented downward from the RCS followed by a horizontal section has been identified as DH piping branch configuration in EPRI TR-1011955. The staff noted that EPRI guideline TR-1011955 stated that valve inleakage is not required for thermal cycling to occur in DH line configurations. The staff requested that the applicant clarify the response (Bullet 5) statement related to DH not susceptible to thermal stratification due to valve leakage.

- The applicant also stated that the U.S. EPR design incorporates lessons learned from operating experience in that the injection line (SIS/RHR) continually rises in elevation from the check valve; therefore, it is not susceptible to valve leakage-induced cyclic thermal stratification. However, FSAR Tier 2, Section 3.12.5.9 stated that RHR/SIS/EBS injection piping is subject to thermal stratification. The staff requested that the applicant clarify the inconsistency.

In a May 26, 2009, response to follow-up RAI 211, Question 03.12-13, the applicant provided the following information to address questions raised by the staff:

- The applicant has used screening criteria from EPRI TR-1011955 based on swirl penetration as a function of line diameter. Accordingly, the fourth bullet in the response to RAI 161, Question 3.12-9 should have stated:

  For any un-isolable attached piping to the RCS with the first vertical-to-horizontal elbow that meets L/Di of the screening criteria of EPRI TR-1011955 for DH lines, thermal stratification does not occur in the branch line considered, since the swirl penetration does not reach the horizontal segment of the first isolation valve or check valve. For this term, L is defined as the length from inside face of the RCS to a location on the branch pipe and Di is the branch line inside diameter.

The staff reviewed the May 26, 2009, response to follow-up RAI 211, Question 03.12-13, and concluded that the screening criteria meet the recommendation of EPRI TR-1011955. On this basis, the staff finds the response acceptable.

In May 26, 2009, response to RAI 211, Question 03.12-13, the applicant also provided the following clarification:

- The applicant acknowledges that valve leakage is not required for thermal cycling in DH line configuration. However, as stated in EPRI TR-1011955, Section 2.2.3, “Note that while cyclic valve out-leakage has been previously attributed to failure in one DH configuration plant leakage event, it is generally believed that the cyclic penetration and retreat of the thermal interface is a fundamental mechanism for thermal cycling in drain lines, residual heat removal suction lines, and similar lines. Valve leakage effects are not considered in the methodology for DH line configurations." The applicant also provided its proposed FSAR revision to clarify this issue. The revision states that, "Operation plant experiences presented in reference 3 support this finding and indicate that DH piping does not require valve leakage for thermal cycling to occur, but instead thermal stratification in DH line was governed by the cyclic penetration and retreat of thermal front due to turbulent penetration."

The staff reviewed the applicant's clarification and FSAR revision. The staff finds this acceptable, because the applicant's revision clarified formation for thermal stratification in DH line.

The applicant further clarified the inconsistency between the last response and the FSAR as follows:

- While FSAR Tier 2, Section 3.12.5.9 addresses thermal stratification potential for RHR/SIS/EBS injection piping, in RAI 161, Question 03.12-9, the staff specifically requested that the applicant evaluate thermal stratification with respect to valve leakage cases. Accordingly, in a February 27, 2009, response to RAI 161, Question 03.12-9, the applicant justified SIS/RHR/EBS injection piping line not being subject to valve inleakage induced thermal stratification. However, it is subject to turbulent penetration induced thermal stratification as stated in FSAR Tier 2, Section 3.12.5.9. FSAR Tier 2, Section 3.12.5.9 will be revised to incorporate clarifications provided by this response.

The staff reviewed the applicant's proposed FSAR Tier 2 revision which stated that the U.S. EPR design incorporates lessons learned from operating experience in that the injection line (SIS/RHR) continually rises in elevation from the check valve; therefore, it is not susceptible to valve leakage-induced cyclic thermal stratification. This statement is unclear to the staff. Therefore, in RAI 306, Question 03.12-20, the staff requested that the applicant clarify why the piping is not susceptible to valve leakage-induced cyclic thermal stratification with continual rises in elevation from the check valve and rise to what kind of level/elevation will not be susceptible to cyclic thermal stratification. The staff also noted that the cyclic thermal stratification occurring within such RCS attached piping is affected by the line orientation and geometry. The staff also requested that the applicant provide detailed line geometry information (e.g., L/Di, DH/ up-horizontal and horizontal (UH/H) configuration) for each of the FSAR mentioned lines in order to determine that the thermal stratification does not occur in any horizontal segment of the RCS attached piping. The applicant stated that AREVA-designed foreign plants on piping testing information show that thermal stratification does not occur in any horizontal segment of the aforementioned (RHR/SIS/EBS injection, RHR/SIS suction) RCS attached piping. The staff noted that if the applicant uses its specific test information to justify that thermal stratification does not occur on any RCS attached piping for U.S. EPR design, the applicant has to provide test information for review. Additionally, in RAI 306, Question 03.12-20, the staff also requested that the applicant provide detailed test information for review and approval.

In a June 2, 2010, response to RAI 306, Question 03.12-20, the applicant stated, "RHR/SIS/EBS injection and RHR/SIS suction RCS attached piping are both located in a downward-horizontal (DH) configuration from the RCS. The requested detailed piping information for the RCS attached RHR/SIS/EBS injection lines of trains 1 to 4 is shown in Figure 03.12-20-1 and Figure 03.12-20-2, and the RCS attached SIS/RHR suction lines of trains 1 to 4 is shown in Figure 03.12-20-3 and Figure 03.12-20-4. The cumulative pipe lengths, with respect of the pipe inner diameter (Di), are indicated within parentheses."

EPRI Topical Report MRP-146, "Management of Thermal Fatigue in Normally Stagnant Non-Isolable Reactor Coolant System Branch Lines," Section 2.2.3 stated that valve inleakage is not required for thermal cycling to occur in DH line configurations.

The staff reviewed the geometry configuration as shown in Figures 03.12-20-1 thru 4. The staff confirmed that all four figures have shown that these RCS attached piping are located in a downward-horizontal (DH) configuration. The staff concurred that these piping systems are subjected to thermal cycling.

In the June 2, 2010, response to RAI 306, Question 03.12-20, the applicant also stated

> AREVA did not use specific test information to justify that thermal stratification does not occur in any RCS attached piping for the U.S. EPR design. Instead, AREVA followed the EPRI thermal stratification guidelines (Reference 4 of U.S. EPR FSAR Tier 2, Section 3.12.7). When the EPRI thermal stratification guidelines predict the occurrence of thermal stratification, the branch lines of concern are monitored. U.S. EPR FSAR Tier 2, Section 3.12.5.9 will be revised to clarify that the assessment of thermal oscillations in piping connected to the RCS was based on the EPRI thermal stratification guidelines and EPRI thermal fatigue guidelines (Reference 3 and Reference 4 of U.S.EPR FSAR Tier 2, Section 3.12.7). The COL information item in U.S. EPR FSAR Tier 2, Section 3.12.5.9 and U.S. EPR FSAR Tier 2, Table 1.8-2 will be revised to clarify that the RCS attached piping branch lines of concern are monitored.

The staff reviewed FSAR Tier 2, Revision 2, Section 3.12.5.9, to confirm that the applicant has removed the testing information taken from AREVA-designed foreign plants as the basis that thermal cycling/stratification does not occur in any horizontal segment of the RCS attached piping.

The applicant also revised FSAR Tier 2, Section 3.12.5.9, by adding that a COL applicant that references the U.S. EPR design certification will monitor the RHR/SIS/EBS injection piping from the RCS to the first isolation valve (all four trains) and RHR/SIS suction piping from the RCS to the first isolation valve (trains 1 and 4) during the first cycle of the first U.S. EPR initial plant operation to verify the operating conditions have been considered in the design unless data from a similar plant's operation demonstrates that thermal oscillation is not a concern for piping connected to the RCS. The additional monitoring activity provides safety verification. Therefore, the staff finds this acceptable.

The staff also confirmed that the applicant has revised FSAR Tier 2, Section 3.12 to clarify that the assessment of thermal cycling for RCS un-isolable piping was based on the EPRI thermal stratification guidelines and EPRI thermal fatigue guidelines. The staff finds this acceptable.

The staff noted that the thermal cycling model is based on valve inleakage establishing a cold stratified layer in horizontal pipe run which interacts with branch line swirling resulting in cyclic thermal loads applied to a region of the horizontal pipe segment for branch pipe with up-horizontal and horizontal geometry configuration. The staff does not expect valve-inleakage during initial plant operation. In RAI 306, Question 03.12-21, the staff requested that the applicant clarify how to simulate valve inleakage to verify that operating conditions have been considered in the design.

In a March 11, 2010, response to RAI 306, Question 03.12-21, the applicant agrees that valve inleakage during initial plant operation is not expected and simulation of valve inleakge in the RCS attached piping is not needed for the following reasons:

- The RCS attached branch lines that are susceptible to thermal stratification are downward horizontal and do not depend on valve inleakage to create cyclic thermal loads per EPRI guidelines.

- The UH/H configurations are part of the CVCS let down and suction lines and normal spray lines (Loop 2 and 3). For these specific lines, EPRI screening methodology has shown that even through valve inleakage occur, these lines do not experience thermal stratification and fatigue-related phenomena.

EPRI Topical Report, MRP-146, Table 2-1, does provided guidance for UH/H geometry configuration to eliminate thermal stratification phenomena. On the basis that the AREVA UH/H configurations are committed to follow EPRI screening methodology. The staff finds this acceptable and, therefore, considers RAI 306, Question 03.12-21 resolved.

3.12.4.4.10 Thermal Stratification

The term "thermal stratification" applies to any condition where fluid is thermally layered due to buoyancy differences between the layers. Thermal stratification occurs in horizontal piping when flow and boundary conditions result in two layers of fluid at different temperatures without appreciable mixing. In cases where the top of pipe temperature is higher than the bottom of pipe temperature, pipe stresses occur due to pipe deflection and changes in support loads. The staff's evaluation for the thermal stratification is presented in the following sections.

3.12.4.4.10.1 *Pressurizer Surge Line Stratification (NRC Bulletin 88-11)*

The applicant stated that the U.S. EPR pressurizer surge line design addresses structural integrity issues raised by NRC Bulletin 88-11, with several features and operational procedures that minimize surge line stratification as given below:

- The pressurizer surge line connects to the top of the hot leg with a vertical take-off. There are no horizontal sections of pressurizer surge line piping. The surge line is sloped at approximately five degrees between the vertical take-off at the hot leg and the vertical leg at the pressurizer which promotes mixing of the colder and hotter fluid layered in the line.

- The vertical take-off from the hot leg is of sufficient length such that continuous bypass spray flow will prevent cooler water from the hot leg from entering the surge line beyond the take-off.

- During normal at-power operation, a continuous bypass spray flow of sufficient magnitude is maintained to suppress turbulent penetration from the hot leg flow.

- The pressurizer versus RCS temperature differential is controlled during heatup to limit the pressurizer-to-hot leg temperature difference. Also, the pressurizer on/off heaters are energized during initial RCS heatup to maintain a constant outsurge of fluid from the pressurizer reducing the number of insurges and the thermal cycles between pressurizer and hot leg temperature.

The applicant also states that the pressurizer surge line temperatures will be monitored during the first cycle of the first U.S. EPR initial plant operation to verify that the design transients for the surge line are representative of actual plant operations unless data from a similar plant's operation determines that monitoring is not warranted. The monitoring program, if required, includes temperature measurements at several locations along the pressurizer surge line and plant parameters including pressurizer temperature, pressurizer level, hot leg temperature, and reactor coolant pump status.

The staff reviewed the applicant's design features and operational procedures that minimize surge line stratification. The staff determines this acceptable to minimize the stratification. However, the severity of the stratification has to be confirmed by the monitoring program. The staff noted that this monitoring activity is not given in FSAR Tier 2, Table 1.8-2 as part of the COL information items. Therefore, in RAI 161, Question 03.12-02 and RAI 211, Question 03.12-12, the staff requested that the applicant clarify who is responsible for this activity and requested that the applicant provide a description of the monitoring program/methodology for confirming the pressurizer surge line integrity. In a February 27, 2009, response to RAI 161, Question 03.12-02, and a May 26, 2009, response to RAI 211, Question 03.12-12, the applicant stated that tests that confirm system integrity are addressed in FSAR Tier 2, Section 14.2 (Test Nos. 30, 32, 33, 35, 37,168,186,195, and 197) and the Technical Specifications (e.g., Surveillance Requirement (SR) 3.4.12, SR 3.4.14). The staff reviewed and evaluated the test/monitoring program and documented it in Section 14.2 of this report. The applicant also provided its response to clarify that this activity will be performed by the COL applicant and FSAR Tier 2, Table 1.8-2 will be revised. The staff verified that Revision 2 of the FSAR Tier 2 was revised accordingly. The staff finds this acceptable and, therefore, considers RAI 161, Question 03.12-02, and RAI 211, Question 03.12-12 resolved.

The staff noted that plant heatup and cooldown methods will control the system ΔT (pressurizer temperature – coolant loop temperature). In order to use the first U.S. EPR initial plant operation to verify that the design transients for the SL are representative, the applicant has to assure that all U.S. EPR plants have to use the same heatup and cooldown methods. In RAI 215, Question 03.12-17, the staff requested that the applicant explain how it would ensure that all U.S. EPR plants will use the same heatup and cooldown methods. In a June 18, 2009, response to RAI 215, Question 03.12-17, the applicant stated that specific heatup and cooldown methods will be prescribed in plant operating procedures, which are the COL applicant's responsibility. Since the response did not address the staff's concern, in follow-up RAI 306, Question 03.12-19, the staff requested that the applicant explain why only first plant surge line transients are monitored without standard heatup/cooldown procedures. In a June 2, 2010, response to follow-up RAI 306, Question 03.12-19, the applicant stated that the COL information item in FSAR Tier 2, Section 3.12.5.10.1 and FSAR Tier 2, Table 1.8-2 will be revised to clarify that the pressurizer surge line temperatures will be monitored during the first cycle of initial plant operation for each plant. The staff verified that Revision 2 of the FSAR Tier 2 was revised accordingly. The staff finds this acceptable and, therefore, considers RAI 306, Question 03.12-19 resolved.

3.12.4.4.10.2 *Pressurizer Stratification*

The applicant states that insurges due to momentary fluctuations in RCS inventory occur during normal operation. These fluctuations result in a stratified thermal front of cooler fluid (near hot leg temperature) being moved up into the lower section of the pressurizer. These insurges result in a step change in the pressurizer bottom fluid temperature. Consideration of these temperature changes is included in the design basis of the pressurizer. On the basis that this has been considered in the design basis of the U.S. EPR pressurizer, the staff determined that this is acceptable. The pressurizer component qualification is addressed in Section 3.9.3 of this report.

3.12.4.4.10.3 *Spray Line Stratification*

The applicant states that the normal spray lines contain stratified liquid and steam during the initial part of the heatup as the horizontal sections in each of the two lines are filled from the cold leg at the same time that the pressurizer is being filled. The normal spray line temperatures will be monitored during the first cycle of the first U.S. EPR initial plant operation to verify that the design transients for the normal spray are representative of actual plant operations unless data from a similar plant's operation determines that monitoring is not warranted.

The staff noted that this monitoring activity is not given in FSAR Tier 2, Table 1.8-2 as part of the COL information items. Therefore, in RAI 161, Questions 03.12-4 and 03.12-12, the staff requested that the applicant clarify who is responsible for this activity and requested that the applicant provide a description of the monitoring program/methodology for confirming the integrity of the normal spray. In a February 27, 2009, response to RAI 161, Question 03.12-4 and a May 26, 2009, response to RAI 211, Question 03.12-12, the applicant stated that tests to confirm system integrity are addressed in FSAR Tier 2, Section 14.2 (Test Nos. 30, 32, 33, 35, 37, 168,186, 195, and 197) and the Technical Specifications (e.g., SR 3.4.12, SR 3.4.14). The staff reviewed and evaluated the test/monitoring program and documented it in Section 14.2 of this report. The applicant also clarified that this activity will be performed by the COL applicant. The staff confirmed that FSAR Tier 2, Table 1.8-2 was revised appropriately. The staff finds this acceptable and, therefore, considers RAI 161, Question 02.12-4, and RAI 211, Question 03.12-12 resolved.

3.12.4.4.10.4 *Feedwater Line Stratification (NRC Bulletin 79-13)*

The applicant stated that U.S. EPR main feed water lines are designed to minimize the potential for thermal stratification to occur. This design feature addresses NRC Bulletin 79-13 concerns with thermal fatigue loading in feed water piping. In FSAR Tier 2, Section 3.12.5.10.4, "Feedwater Line Stratification (NRC Bulletin 79-13)," that the U.S. EPR main feed water lines are designed to minimize thermal stratification. The main feed water nozzle on each steam generator and the attached feed water line are sloped downward to minimize the potential for thermal stratification. The applicant further stated that continuous feed water flow during operation prevents thermal stratification in the piping. However, during low flow actuation and flow shutdown, thermal stratification in the main feed water line near the steam generator occurs. The applicant stated that main feed water line temperatures will be monitored during the first cycle of the first U.S. EPR initial plant operation to verify that the design transients for the main feed water lines are representative of actual plant operations unless data from a similar plant's operation determines that monitoring is not warranted. In RAI 161, Question 03.12-5 and RAI 211, Question 03.12-12, the staff noted that this monitoring activity is not given in FSAR Tier 2, Table 1.8-2 as a part of the COL information items. The staff requested that the applicant clarify who is responsible for this activity and requested that the applicant provide a description of the monitoring program/methodology for confirming the main feedwater line integrity.

In a February 27, 2009, response to RAI 161, Question 03.12-5 and a May 26, 2009, response to RAI 211, Question 03.12-12, the applicant stated that tests to confirm system integrity are addressed in FSAR Tier 2, Section 14.2 (Test Nos. 30, 32, 33, 35, 37, 168,186, 195, and 197) and the Technical Specifications (e.g., SR 3.4.12, SR 3.4.14). The staff reviewed and evaluated the test/monitoring program and documented it in Section 14.2 of this report. In the February 27, 2009, response to RAI 161, Question 03.12-5 and the May 26, 2009, response to RAI 211, Question 03.12-12, the applicant also provided clarification that this activity will be performed by the COL applicant and that FSAR Tier 2, Table 1.8-2 was revised in Revision 2 of the FSAR. The staff finds this acceptable and confirmed that FSAR Tier 2, Table 1.8-2 has been revised. Therefore, the staff considers RAI 161, Question 03.12-5 and RAI 211, Question 03.12-12 resolved.

In FSAR Tier 2, Section 3.9.3.1.1, the applicant states that a COL applicant that references the U.S. EPR design certification will examine the feedwater line welds after hot functional testing prior to fuel loading and at the first refueling outage, and report the results of inspection to the NRC, in accordance with NRC Bulletin 79-13. The staff noted that the proposed inspection per NRC Bulletin 79-13 may not detect weld damage as a result of thermal cycling at this early stage of the plant operation. However, since this inspection will be performed in accordance with the NRC guidance in NRC Bulletin 79-13, the staff finds this acceptable.

The emergency feed water system is not actuated during normal or upset operations. The EFW system piping layout minimizes thermal stratification during emergency and faulted plant operation. The staff noted that the above statement does not justify why thermal stratification will be minimized by EFW system piping layout. Therefore, in RAI 161, Question 03.12-8, the staff requested that the applicant provide detailed justification to substantiate that EFW system thermal stratification is minimized. The staff also requested that the applicant explain what the layout is and how the layout can minimize thermal stratification.

In a February 27, 2009, response to RAI 161, Question 03.12-8, the applicant provided the following information:

The emergency feedwater system is composed of four trains that supply water to their respective steam generator, or to any other steam generator, via a common cross-connect discharge header. For each EFW system train, the water runs from a water storage pool (cold source) and is pumped toward the steam generator (hot source). During emergency and faulted plant operations, the thermal stratification in the emergency feedwater piping layout is minimized for the following reasons:

- The piping layout of the EFW system is physically independent of the main feedwater system. The EFW system and MFW system have a separate nozzle connected to each steam generator, such that the EFW system piping is not affected when MFW system is being injected into the steam generator. Based on operating experience from previous plant designs, such as physical EFW system to MFW system, separation reduces the frequency of thermal cycling and the susceptibility of thermal stratification in the EFW system nozzle.

- Each EFW system train is a continuously descending piping run 10.16 cm (4 in.) piping from the steam generator to the pump, with a 90 degree elbow oriented downward at each steam generator downcomer nozzle. For each train, the length of the first vertical-to-horizontal elbow piping connected to each steam generator is greater than 38 Di, where Di is the inside diameter of the EFW system piping. Due to the relatively long length and the relatively low steam velocities in the vicinity of the EFW nozzles, turbulent penetration does not occur in this first horizontal section upstream of the steam generator.

The staff reviewed the applicant's February 27, 2009, response RAI 161, Question 03.12-8 which explained that the layout provides a long riser, and physical EFW system to MFW system separation reduces the frequency of thermal cycling and the susceptibility of thermal stratification in the EFW system nozzle. The staff finds this acceptable and, therefore, considers RAI 161, Question 03.12-8 resolved.

3.12.4.4.11 Safety Relief Valve Design, Installation, and Testing

In Topical Report ANP-10264NP-A,Section 3.8.1, the applicant states that the design and installation of safety and relief valves for overpressure protection are performed to the criteria specified in ASME Code, Appendix O, "Rules for the Design of Safety Valve Installations," 2001 Edition, 2003 Addenda. In addition, the design and installation will comply with the additional criteria in SRP Section 3.9.3, Paragraph II.2. In Topical Report ANP-10264NP-A, Section 3.8.2, the applicant describes analysis requirements for pressure relieving devices when the discharge is directly to the atmosphere (open discharge) and to headers or tanks (closed discharge). The applicant's position meets staff's recommendation and the staff finds this acceptable. The testing is discussed in Section 14.2 of this report.

3.12.4.4.12 Functional Capability

All ASME Code Class 1, 2, and 3 piping systems that are essential for safe shutdown must retain their functional capability for all Service Level D loading conditions as required by GDC 2. Designs meeting the recommendations in NUREG-1367, "Functional Capability of Piping Systems," are accepted by the staff as satisfying the functional capability requirements.

In Topical Report ANP-10264NP-A, Section 3.5, the applicant states that all ASME Code Class 1, 2, and 3 piping systems that are essential for safe shutdown under the postulated events given in the Topical Report ANP-10264NP-A, Table 3-3 are designed to meet the guidance and code requirements in NUREG-1367. In no case, shall the piping stress exceed the limits designated for Service Level D in ASME Code, Section III. The Service Level D limits are 3.0 S_m (not to exceed 2.0 S_y) for ASME Code Class 1 piping and 3.0 S_h (not to exceed 2.0 S_y) for ASME Code Class 2 and 3 piping. In addition, the criteria also include: (1) The ratio of pipe NPS and the wall thickness (D_o/t) not to exceed 50; (2) dynamic responses for reversing dynamic loads (e.g., earthquake, building hydrodynamic loads) based on an elastic response spectrum with 15 percent peak broadening with not more than 5 percent damping;

(3) the external pressure not to exceed the internal pressure; and finally, (4) steady state stresses from dead weight loads not to exceed 0.25 S_y. Since the applicant committed to satisfy all guidance and code requirements of NUREG-1367, the staff finds this acceptable.

3.12.4.4.13 Combination of Inertial and Seismic Anchor Motion Effects

In Topical Report ANP 10264NP-A, Section 4.2.2.5, the applicant states that the results of the SAM analysis shall be combined with the seismic inertia analysis results from absolute summation method when an enveloped uniform support motion is used for the dynamic analysis, per SRP 3.7.3. When independent support motion is used in the inertial analysis, the response is due to the relative displacements, and those due to inertia are combined by the SRSS method, per NUREG-1060. Since the applicant's position is consistent with the staff guidance, the staff finds this acceptable.

3.12.4.4.14 Operating Basis Earthquake as Design Load

In SECY-93-087, the staff recommended eliminating the OBE from the design for ALWRs. The NRC approved the staff recommendations in its July 21, 1993, SRM. The SRM includes specific supplemental criteria for fatigue, seismic anchor motion, and piping stress limits that should be applied when the OBE is eliminated. The staff position on the use of a single-earthquake design for SSCs is discussed in Section 3.12.4.4.3 for load combinations and Section 3.12.4.4.7 for fatigue evaluation of this report. The effects of SAM due to the SSE should be considered in combination with the effects of other normal operational loadings that might occur concurrently. For fatigue evaluation, two SSE events with 10 maximum stress cycles per event (or an equivalent number of fractional cycles) should be considered.

In Topical Report ANP 10264NP-A,Section 3.4.1, the applicant states that the fatigue evaluation of ASME components will take into consideration two SSE events with 10 peak stress cycles per event. Alternately, an equivalent number of fractional vibratory cycles (i.e., 300 cycles) may be used (but with an amplitude not less than one-third of the maximum SSE amplitude) when derived in accordance with IEEE Std 344-1987, Appendix D, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations." The staff finds this acceptable, since the commitment conforms to the NRC guidance document previously discussed above and the NRC-approved staff recommendations on the issue of OBE elimination.

3.12.4.4.15 Welded Attachments

For the analysis of local stresses at welded attachments to piping (e.g., lugs, trunnions, or stanchions), in Topical Report ANP-10264NP-A, Section 3.6, the applicant states that the support and restraint designs that require welded attachments to the pipe for transfer of the pipe loads to the supporting structure will adhere to industry practices and ASME Code Cases identified in Topical Report ANP-10264NP-A, Section 2.2. Since this will ensure the quality of these welded attachments, the staff finds this acceptable.

3.12.4.4.16 Modal Damping for Composite Structures

For subsystems that are composed of different material types (e.g., welded steel pipe and pipe supports), either a mass or stiffness weighted method can be used to determine the composite modal damping value. Composite modal damping for coupled building and piping systems can be used for piping systems that are coupled to the primary coolant loop system and the interior concrete building.

The composite modal damping ratio can be used when the modal superposition method of analysis (either TH or RS) is used, as required by SRP Section 3.7.2, II.13. In Topical Report ANP 10264NP-A, the applicant stated that when composite modal damping is applied in a dynamic analysis, each modal subgroup (piping, supports, equipments, etc.) is assigned an appropriate damping value per RG 1.61, Revision 1. The staff finds that the applicant's method meets the requirements of SRP Section 3 7.2, and is therefore acceptable.

3.12.4.4.17 Minimum Temperature for Thermal Analyses

In Topical Report ANP-10264NP-A, Section 3.3.1.3, the applicant states that the zero thermal load temperature is 21.1 °C (70 °F) and for piping systems with an operating temperature equal to or less than 65.56 °C (150 °F), a thermal analysis is not required. These criteria are typically used by the industry and the staff finds this acceptable as discussed in NUREG-1793.

3.12.4.4.18 Intersystem Loss-of-Coolant Accident

In SECY-90-016, "Evolutionary Light Water Reactor (LWR) Certification Issues and their Relationships to Current Regulatory Requirements," January 12, 1990, the staff discussed the resolution of the Intersystem LOCA (ISLOCA) issue for advanced light water reactor plants by requiring that low pressure piping systems that interface with the RCPB be designed to withstand full RCS pressure to the extent practicable. In its June 26, 1990, SRM, the NRC approved these staff recommendations, provided that all elements of the low-pressure systems are considered.

In Topical Report ANP-10264NP-A, Section 3.9, the applicant states that low pressure piping systems that interface with the RCL and are thus subjected to the full RCL pressure will be designed for the full operating pressure of the RCL. The appropriate minimum wall thickness of the piping will then be calculated for each system by using Equation 1 of NB-3640 of the ASME Code for Class 1 piping or Equation 3 of NC/ND-3640 for ASME Code Class 2 and 3 piping. The piping will be analyzed to the requirements in NB/NC/ND-3650. Since this satisfies the ASME Code and ensures the low pressure piping to withstand a full RCL pressure, the staff finds this acceptable.

3.12.4.4.19 Effect of Environment on fatigue Design

The staff reviewed environmental effect on fatigue design as described in FSAR Tier 2, Section 3.12.5.19. The applicant states that the effects of reactor coolant environment, using the methodology described in RG 1.207 are considered when performing fatigue analyses for ASME Code Class 1 piping. If there are locations in the ASME Code Class 1 systems where the cumulative usage factor cannot be shown to be less than 1.0, based on the methodology described in RG 1.207, alternative methods for addressing environmental fatigue will be applied. The applicant's four alternatives are given below:

- Redefinition of the normal and upset transients affecting the location in question to reduce the severity of the transients or to reduce the number of cycles associated with the transients

- Redefinition of the in-air design fatigue curves and/or environmental correction factor (F_{en}) curves using data obtained from testing of samples representative of U.S. EPR materials, configurations, and environment

- Fatigue monitoring of the affected locations

- Augmented inspection (beyond 10 year inservice inspection requirements) of the affected locations

The staff does not agree that the cumulative fatigue usage factor is allowed to exceed 1.0 during the design stage. Therefore, in RAI 306, Question 03.12-18, the staff requested that the applicant address allowing the cumulative fatigue usage factor to exceed 1.0. Further, the staff noted that redefinition of the normal and upset transient affecting the location in question to reduce the severity of the transients or to reduce the number of cycles associated with the transient requires a license amendment. The redefinition of the in-air design fatigue curves and/or Fen penalty factors also requires a license amendment. The staff requested that the applicant clarify that the applicant submit a license amendment for NRC review and approval for use of these two alternative methods. The staff also noted that fatigue monitoring and augmented inspections are expected for operating plants. The staff does not agree that the design requirement for fatigue and cumulative fatigue usage factors for piping and components can be changed. The staff requested that the applicant provide other alternatives or to follow the staffapproved methods.

In a December 4, 2009, response to RAI 306, Question 03.12-18, the applicant proposed to revise transients and proposed alternative approaches. The staff reviewed the response and determined the response is not sufficient without inclusion of the actual approach. Therefore, in follow-up RAI 388, Question 03.12-24, the staff requested that the applicant address this issue. In a July 1, 2010, response to RAI 388, Question 03.12-24, the applicant removed all four alternatives including the discussion of redefinition of transients, redefinition of Fen, fatigue monitoring and augmented inspections from FSAR Tier 2, Section 3.12.5.19. On the basis that the applicant's position is now consistent with RG 1.207, the staff finds this acceptable. The staff also confirmed that FSAR Tier 2, Section 3.12.5.19 has been revised appropriately. Therefore, the staff considers RAI 306, Question 03.12-18, and RAI 388, Question 03.12-24 resolved.

3.12.4.4.20 Pipe Stress Analysis Criteria Evaluation Summary

Except for open items mentioned above and on the basis of the discussion in the above subsections and evaluation of Section 3.12.4.4 of this report, the staff concludes that the pipe stress analysis methods for the U.S. EPR piping design are acceptable. The staff's conclusion is based on the following:

- The applicant meets GDC 1 and 10 CFR Part 50, Appendix B with regard to piping systems being designed, fabricated, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed, and with appropriate quality control.

- The applicant meets GDC 2 and 10 CFR Part 50, Appendix S with regard to design transients and resulting load combinations for piping and pipe supports to withstand the effects of earthquakes combined with the effects of normal or accident conditions.

- The applicant meets GDC 4 with regard to piping systems important to safety being designed to accommodate the effects of and to be compatible with the environmental conditions of normal and accident conditions.

- The applicant meets GDC 14 with regard to the reactor coolant pressure boundary of the primary piping systems being designed, fabricated, constructed, and tested to have an extremely low probability of abnormal leakage, of rapid propagating failure, and of gross rupture.

- The applicant meets GDC 15 with regard to the reactor coolant piping systems being designed with specific design and service limits to assure sufficient margin that the design conditions are not exceeded.

3.12.4.5 *Piping Support Design Criteria*

GDC 1 requires that the piping and pipe supports should be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. 10 CFR Part 50, Appendix B requires that design quality should be controlled for ensuring structural and functional integrity of Seismic Category I components. GDC 2 requires that the piping and pipe supports should withstand the effects of earthquake loads. The supporting elements should be capable of carrying the sum of all concurrently acting loads and designed to provide the required support to the piping system and allow pipe movement with thermal changes without causing overstress. All parts of the supporting equipment or structure should be fabricated and assembled so that they would not be disengaged by movement of the supported piping.

In Topical Report ANP 10264NP-A, Section 6.0, the applicant states that the pipe support elements will be designed to meet the requirements of the appropriate design codes and conform to the code requirements of the overall piping system.

3.12.4.5.1 Applicable Codes

Pipe supports include hangers, snubbers, struts, spring hangers, frames, energy absorbers and limit stops and can be plate and shell type supports, linear type supports or commercially available standard piping supports. In Topical Report ANP-10264NP-A, Section 6.1, the applicant states that for Service Levels A, B, and C, the Seismic Category I pipe supports will be designed, manufactured, installed, and tested in accordance with ASME Code, Subsection NF and ASME Code, Section III, Appendix F, Service Level D will be utilized. In addition, the welding requirements for A500, Grade B tube steel from AWS D1.1 will be utilized.

The applicant also states that plate and shell type supports such as skirts or saddles are fabricated from plate elements and loaded to create a biaxial stress field. Linear type supports (i.e., beams, columns, frames, and rings) are essentially subjected to a single component of direct stress, but may also be subjected to shear stresses. Standard supports are made from typical support catalog items such as springs, rigid struts, and snubbers and are typically load rated items, but they may be also qualified by plate and shell or linear analysis methods.

Further, the applicant states that Seismic Category II pipe supports are designed to ANSI/American Institute of Steel Construction (AISC) N690, "Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities." Non-seismic category pipe supports are designed using guidance from the AISC Manual of Steel Construction. In addition to the pipe support design codes mentioned above, expansion anchors and other steel embedments in concrete shall be designed for concrete strength in accordance with American Concrete Institute's ACI-349, "Code Requirements for Nuclear Safety Related Concrete Structures."

The applicant further states that typically the stress limits for pipe supports are in accordance with ASME III, Subsection NF and Appendix F. The design of all supports for the non-nuclear piping satisfies the requirements of ASME/ANSI B31.1, "Power Piping Code," Paragraph 120 for loads on pipe supporting elements and Paragraph 121 for design of pipe supporting elements.

The applicant also stated that for all Seismic Category II pipe supports, other than standard component supports, design, manufacturing, installation, and testing, will meet the requirements of ANSI/AISC N690. Standard component supports will be designed, manufactured, installed and tested to ASME Code, Subsection NF. Any structural members used as part of a pipe support also containing standard components will be designed, manufactured, installed, and tested to ANSI/AISC N690. The reference to ANSI/AISC N690 in Topical Report ANP 10264NP-A, includes Supplement 2 (2004), this is in accordance with SRP Sections 3.8.3 and 3.8.4 and RG 1.84.

The applicant further stated that for non-seismic pipe supports used for piping analyzed to ASME/ANSI B31.1, the requirements of ASME/ANSI B31.1 for supports (Paragraphs 120 and 121) will be met, where applicable. In addition, the structural elements will meet the requirements of the AISC Manual. For standard components used in such supports, vendors' catalog requirements will be utilized, which also meet B31.1 requirements.

Use of the ASME Code Section III, Subsection NF and Appendix F, along with the other associated design documents for U.S. EPR design Seismic Category II and non-seismic pipe supports, meets SRP recommendation, RG 1.84, and quality industry standards and is acceptable to the staff.

3.12.4.5.2 Jurisdictional Boundaries

In Topical Report ANP-10264NP-A, Section 6.2, the applicant states that all piping supports are designed in accordance with the rules of Subsection NF of the ASME Code up to the building structure interface as defined by the jurisdictional boundaries in ASME Code, Subarticle NF-1130. For attachments to building steel, the boundary is taken at the interface with the building steel, with the weld being designed to the rules of ASME Code, Subsection NF. For attachments to concrete building structures, the boundary is generally at the weld of the support member to a base plate or embedded plate, with the weld designed to the specifications of ASME Code, Subsection NF.

The jurisdictional boundary between the pipe and its support structure will follow the guidance of Paragraph NB-1132, NC-1132, or ND-1132, as appropriate for the ASME Code Class of piping involved. For piping analyzed to B31.1, the jurisdictional boundary guidance of Paragraph ND-1132 will be utilized. The staff's review of the jurisdictional boundaries in the ASME Code concludes that they are sufficiently defined to ensure a clear division among the piping, pipe support, and the building structure. Therefore, the staff finds this acceptable.

3.12.4.5.3 Load and Load Combinations

Topical Report ANP-10264NP-A, Section 6.3 defines the support loads, and their combination methods for the design of piping supports correspond to those used for design of the supported pipe. The loadings for the pipe support design include:

- Deadweight (includes pipe and fittings, contents and support itself)
- Thermal (for all four service levels: normal, upset, emergency, and faulted)
- Friction (due to thermal expansion movement)
- System operating transients (safety/relief valve thrust, fast valve closure, water/steam hammer)
- Design basis pipe break (includes jet impingement or pipe whip)
- Main steam/feedwater pipe break
- LOCA
- Seismic (safe shutdown earthquake and seismic anchor movement)

In Topical Report ANP 10264NP-A, Section 6.3.11, the applicant provided a minimum design load criteria that will be used for all supports so that uniformity is obtained in the load carrying capability of the supports. All supports will be designed for the largest of the following three loads: 125 percent of the Level A condition load; the weight of a standard ASME B31.1 span of water filled, Schedule 80 pipe; a minimum value of 150 lb. Topical Report ANP 10264NP-A, Table 6-1 provides the specific load combinations that will be used in the design of pipe supports. The acceptance criteria associated with the Service Levels will be as described in ASME Code, Subsection NF, ANSI/AISC N690 or the AISC Manual of Steel Construction, as appropriate.

The applicant states that since signed thermal loadings may cancel other signed loadings, the cold condition must also always be considered for support loads. In Topical Report ANP 10264NP-A, Section 6.3.2, the applicant states consideration for local, radial thermal expansion of the pipe cross section must be made. This effect is addressed by having small gaps around the pipe for such thermal growth, while still maintaining relatively tight constraints for seismic loadings. The staff evaluated the gap criteria in FSAR Tier 2, Section 3.12.4.5.11 and finds the applicant's small gap criteria acceptable.

To clarify the load combinations for different types of supports, the applicant also clarified that Topical Report ANP 10264NP-A, Table 6-1 includes three faulted load combinations which contain SSE loads. In addition, Note 3 of Table 6-1 states that SSE includes inertia and SAM loads combined by the absolute sum method. These would all apply to ASME Code Class 1, 2, and 3 pipe supports. In addition, struts and anchors/guides will be analyzed to all load combinations shown in the table. Snubbers will be designed to all but the normal level load combinations shown in the table.

With regards to wind/tornado loads, the applicant stated that in the Topical Report ANP 10264NP-A, Section 3.3.1.6 for design certification, no ASME Code Class 1, 2 and 3 piping is exposed to wind and tornado loads, and further stated that if a COL applicant creates such an exposed piping condition, it will be addressed at that time. The staff finds this acceptable.

The applicant also stated that forces due to friction of the piping on the support shall be considered under combined deadweight and thermal loading normal to the applicable support member. Topical Report ANP 10264NP-A, Table 6-1 includes the effects of system operating transients (SOT) with pipe break, LOCA, and SSE loads, both in the Level C and the Level D cases.

The applicant further stated that loads due to dynamic events are combined considering the time phasing of the events (i.e., whether the loads are coincident in time). When the time phasing relationship can be established, dynamic loads may be combined by the SRSS method, provided it is demonstrated that the non-exceedance criteria given in NUREG-0484, "Methodology for Combining Dynamic Responses", is met. When the time phasing relationship cannot be established or when the non-exceedance criteria in NUREG-0484 are not met, dynamic loads are combined by absolute summation. The applicant also stated that SSE and high energy line break (i.e., LOCA and secondary side pipe rupture) loads are always combined using the SRSS method. Note, that any steady state effects from the system operating transients will be added to the combinations.

Since the load combinations presented in Topical Report ANP 10264NP-A, Table 6-1 conform to the industry practice using ASME Code, Subsection NF, ANSI/AISC N690, or AISC Manual for Steel Construction for Service Level A, B, C and D loads, and conform to NUREG-0484 for dynamic load combinations, the staff finds this acceptable.

3.12.4.5.4 Pipe Support Baseplate and Anchor Bolt Design

In Topical Report ANP 10264NP-A, Section 6.4, the applicant states that the use of base plates with expansion anchors will be minimized in the U.S. EPR design. The concrete will be evaluated using ACI-349, Appendix B subject to the conditions and limitations of RG 1.199, "Anchoring Components and Structural Supports in Concrete." This guidance accounts for the proper consideration of anchor bolt spacing and distance to a free edge of concrete. In addition, all aspects of the anchor bolt design, including base plate flexibility and factors of safety will be utilized in the development of anchor bolt loads, as addressed in IE Bulletin 79-02, Revision 2, "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts."

SRP Section 3.12.II.D.iv states that the design of the pipe support baseplates and anchor bolts should comply with guidance provided in NRC BL 79-02, Revision 2.

On the basis that the applicant's position meets staff's recommendation, the staff finds this acceptable.

3.12.4.5.5 Use of Energy Absorbers and Limit Stops

In Topical Report ANP 10264NP-A, Section 6.5, the applicant states that energy absorbers and limit stops for pipe supports utilizing normal design loadings will not be used for the U.S. EPR piping design. On the basis that energy absorber and limit stops are not part of the U.S. EPR design, the staff finds that this portion of the SRP review is not applicable to U.S. EPR.

3.12.4.5.6 Use of Snubber

The operating loads on snubbers are the loads caused by dynamic events during various operating conditions. Snubbers restrain piping against response to dynamic excitation and to the associated differential movement of the piping system support anchor points. The loads calculated in the piping dynamic analysis cannot exceed the snubber load capacity for design, normal, upset, emergency, and faulted conditions. Snubbers are generally used in situations where dynamic support is required, because thermal growth of the piping prohibits the use of rigid supports. The snubber locations and support directions are first decided by estimation so that the stresses in the piping system have acceptable values.

In Topical Report ANP 10264NP-A, Section 6.6, the applicant states that typical snubber components are manufactured standard hardware, and may be either hydraulic or mechanical in operation. The applicant also stated that “Other design/analysis considerations for snubbers are related to the ability of the snubbers to properly activate for their design loadings. For snubbers which might experience high thermal growth rates, the analysis should ensure that such growth rates do not exceed the snubber lock-up velocity. Also, for parallel snubbers utilized in the same support, care must be taken to ensure that total fitting clearances are not mismatched between the tandem snubbers such that one will activate before the other.

The applicant also states that design specifications provided to the snubber suppliers will include the codes and standards, functional requirements, operating environment (both normal and post-accident), materials (construction and maintenance), functional testing and certification, and requirements for construction to meet ASME Code, Subsection NF. The proper installation and operation of snubbers will be verified by the COL applicant, utilizing visual inspections, hot and cold position measurements, and observance of thermal movements during plant startup. This is identified as a COL information item in FSAR Tier 2, Table 1.8-2. The staff evaluated the snubber operability and documented its acceptance in Section 3.9.6 of this report.

3.12.4.5.7 Pipe Support Stiffness

Topical Report ANP 10264NP-A, Section 6.7 provides information about modeling the stiffness of pipe supports with either the actual stiffness or an arbitrary rigid stiffness. Also, the applicant discusses two deflection checks that will be performed for each support modeled as rigid in the piping analysis. The first check will compare the deflection in the restrained direction(s) to a maximum of 1.5875 mm (1/16 in.) for SSE loadings or the minimum support design loadings of Topical Report ANP 10264NP-A, Section 6.3.11. The second check will compare the deflection in the restrained direction(s) to a maximum of (1/8 in.) for the worst case deflection for any load case combination. In the development of the support deflections, dynamically flexible building elements beyond the support jurisdictional boundaries will also be considered by the applicant.

In Topical Report ANP 10264NP-A, Section 6.8.2, the applicant stated that the initial piping analyses will assume all supports rigid (except for the few cases where the actual support structures are included in the piping model) and, will therefore, utilize the default rigid support stiffness values contained in the analysis code. In addition, the initial pipe support designs will be developed to create a rigid support based on the deflection check criteria given in Topical

Report ANP 10264NP-A, Section 6.7. If for some reason, a rigid support cannot be achieved, actual support stiffness will be determined for all supports within that model and will be used in a reanalysis of the piping along with the mass of the support. Therefore, the dynamic characteristics of supports that are not rigid will be included in the piping analysis. Use of the actual support stiffness in the piping analysis model is acceptable to the staff.

The staff noted that WRC Bulletin 353, “Position Paper on Nuclear Plant Pipe Supports”, discusses the use of deflection checks to determine stiffness of supports. It discusses the use of a 1.5875 mm (1/16 in.) deflection for Level B checks, with no more than a maximum of (1/4 in.) for typical piping systems in the range of 3 to 9 Hz. frequency. The deflection check criteria used in Topical Report ANP 10264NP-A, have been used in other plants and falls within the bounds of the criteria of this document. Since the deflection criteria and the process described provide reasonable assurance for the, piping structural integrity and also meets the recommendation of WRC Bulletin 353 that provides technical justification, the staff finds this acceptable.

3.12.4.5.8 Seismic Self-Weight Excitation

In Topical Report AN-10264NP-A, Section 6.8, the applicant states that the response of the support structure itself to SSE loadings is to be included in the pipe support analysis. In general, the inertial response of the support mass will be evaluated using a response spectra analysis similar to that performed for the piping. Damping values for welded and bolted structures are given in RG 1.61. This support self weight SSE response, the piping inertial load SSE response, and the SSE loads from SAM are to be combined by the absolute sum method. The staff concludes that this method takes into consideration the service loading combinations resulting from postulated events and the designation of appropriate service limits for pipe support seismic loads and is consistent with SRP Section 3.9.3, and is therefore acceptable.

3.12.4.5.9 Design of Supplementary Steel

Topical Report AN-10264NP-A, Section 6.9 provides design criteria for the design of pipe supports using supplementary steel. Supplementary steel for pipe supports are designed in accordance with ASME Code, Section III, Subsection NF or ANSI/AISC N690-1994 including Supplement 2 (2004), or AISC Manual (for non-seismic supports).

The use of ASME Code, Section III, Subsection NF meets the staff recommendation of SRP Section 3.12.II.D.xi. The use of ANSI/AISC N690-1994 including Supplement 2(2004) meets the staff recommendation in RG 1.84. Therefore, the staff finds this acceptable. Non-seismic supports to be design with AISC Manual of Steel Construction is an industry practice acceptable to the staff, since it was developed by a professional society and voluntary consensus standards organization and has proven to provide adequate design guidelines for the design of structural steel.

3.12.4.5.10 Consideration of Friction Forces

In Topical Report AN-10264NP-A, Section 6.10, the applicant describes the criteria for considering the effect of friction forces due to thermal movements. The friction forces are calculated using the deadweight and thermal loads normal to the applicable support member. Specifically, to calculate the friction forces, a force will only need to be calculated if the thermal movement in the applicable unrestrained direction(s) is greater than 1.5875 mm (1/16 in.).

If this threshold is met, the force will be calculated using C x N, where C is the appropriate coefficient of friction and N is the total force normal to the movement. The coefficient of friction will be taken as 0.3 for steel-to-steel conditions and 0.1 for low friction slide/bearing plates. If support stiffness information is readily available, this calculated force can be reduced by using the force of K x D (if less than C x N), where K is the support stiffness in the movement direction and D is the movement. The staff notes that the coefficients of friction are reasonable values commonly used in the nuclear industry that have been validated through years of design experience, and therefore are found acceptable for use by the staff.

3.12.4.5.11 Pipe Support Gaps and Clearances

In Topical Report ANP 10264NP-A, Section 6.11, the applicant states that for rigid guide pipe supports in the piping analysis, the typical industry design practice is to provide small gaps between the pipe and its surrounding structural members. These small gaps allow radial thermal expansion of the pipe, as well as allow rotation of the pipe at the support. The normal design practice for the U.S. EPR will be to use a nominal cold condition gap of 1.5875 mm (1/16 in.) on each side of the pipe in the restrained direction. This will lead to a maximum total cold condition gap around the pipe for a particular direction of 3.175 mm (1/8 in.). For gaps around the pipe in an unrestrained direction, the gap magnitudes should be specified large enough to accommodate the maximum movement of the pipe.

The staff noted that 1.5875 mm (1/16 in.) cold condition gap on each side of pipe in the restrained direction may not provide sufficient radial expansion of the pipe in the restrained direction for ASME Code Class 1 and 2 large bore piping. Therefore, in RAI 331, Question 03.12-22, the staff requested that the applicant demonstrate that the 1.5875 mm (1/16 in.) gap has accounted radial expansion. The staff also noted that a 1.5875 mm (1/16 in.) cold gap indicates the pipe is not supported vertically during cold condition. The March 2, 2010, response to RAI 331, Question 03.12-22 was not acceptable to the staff because it did not provide the technical basis to support SRP Section 3.12. Therefore, RAI 331, Question 03.12-22 was closed and the staff issued follow-up RAI 377, Question 03.12-23. In RAI 377, Question 03.12-23 the staff requested that the applicant demonstrate that the pipe support design with 1.5875 mm (1/16 in.) cold gap is adequate during a cold condition. In an April 9, 2010, response to RAI 377, Question 03.12-23, the applicant stated that U.S.EPR piping was evaluated to determine that maximum radial thermal expansion is (0.12 in.) for main steam piping at its design temperature. The applicant also stated that “Under cold conditions, U.S. EPR piping design provides zero gap in the downward direction for horizontal piping where gapped vertical supports are required for dynamic restraint. This is standard practice and is done to provide required deadweight support.” On the basis that the total cold condition gap of (1/8 in.) is greater than the maximum radial expansion, to provide sufficient clearance as recommended in SRP Section 3.12.II.D.xi and standard practice for vertical support with zero gap, the staff finds this acceptable and, therefore, considers RAI 377, Question 03.12-23 resolved.

3.12.4.5.12 Instrument Line Support Criteria

In Topical Report ANP 10264NP-A, Section 6.12, the applicant states that the design and analysis loadings, load combinations and acceptance criteria to be used for instrumentation line supports will be similar to those used for pipe supports. The applicable design loads will include deadweight, thermal expansion, and seismic loadings (where appropriate). The applicable loading combinations will similarly follow those used for normal and faulted levels in Topical

Report ANP 10264NP-A, Table 6-1, utilizing the design loadings mentioned above. The acceptance criteria are ASME Code, Subsection NF for Seismic Category I instrumentation lines, ANSI/AISC N690 for Seismic Category II instrumentation lines and the AISC Manual of Steel Construction for non-seismic instrumentation lines.

The staff notes that the use of pipe support design criteria for instrumentation line supports is a conservative design approach and utilizes standards developed by a professional society and voluntary consensus standards organization, which are acceptable to the staff. Therefore, these criteria are acceptable for use in the design of the U.S. EPR instrumentation line supports.

3.12.4.5.13 Pipe Deflection Limits

In Topical Report ANP 10264NP-A, Section 6.13, the applicant states that for pipe supports utilizing standard manufactured hardware components, the manufacturer's recommendations for limitations in its hardware will be followed. Examples of these limitations are travel limits for spring hangers, stroke limits for snubbers, swing angles for rods, struts and snubbers, alignment angles between clamps or end brackets with their associated struts and snubbers, and the variability check for variable spring supports. In addition to the manufacturer's recommended limits, allowances will be made in the initial designs for tolerances on such limits. This is especially important for snubber and spring design where the function of the support can be changed by an exceeded limit. The staff finds these additional tolerances acceptable, because of the added assurance that the component movement will remain within intended design limits of the component supports, thus ensuring the functionality of supports.

3.12.4.5.14 Piping Support Design Criteria Evaluation Summary

The staff concludes that piping systems important to safety are designed to quality standards commensurate with their importance to safety. The staff also concludes the following:

- The applicant meets the requirements of GDC 1 and 10 CFR 50.55a by specifying methods and procedures for the design and construction of safety-related piping systems in conformance with general engineering practice.

- The applicant meets the requirements of GDC 2 and GDC 4 by designing and constructing the safety-related piping systems to withstand the effects of normal operation as well as postulated events such as LOCAs and dynamic effects resulting from the SSE.

- The applicant meets 10 CFR Part 50 requirements by identifying applicable codes and standards, design and analysis methods, design transients and load combinations, and design limits and service conditions to ensure adequate design of all safety-related piping and pipe supports in the U.S. EPR for their safety functions.

- The applicant meets 10 CFR Part 52 requirements by providing reasonable assurance that the piping systems will be designed and built in accordance with the certified design. The implementation of these pre-approved methods and satisfaction of the acceptance criteria will be verified through the performance of the ITAAC by the COL holder to ensure that the as-constructed piping systems are in conformance with the certified design for their safety functions.

- The applicant meets 10 CFR Part 50, Appendix S, requirements by designing the safety-related piping systems, with a reasonable assurance to withstand the dynamic effects of earthquakes with an appropriate combination of other loads of normal operation and postulated events with an adequate margin for ensuring their safety functions.

- The applicant meets the requirements of GDC 14 by following the ASME Code requirements with regard to the RCPB of the primary piping systems being designed, fabricated, constructed, and tested to have an extremely low probability of abnormal leakage, of rapid propagating failure, and of gross rupture.

- The applicant meets the requirements of GDC 15 by following the ASME Code requirements with regard to the reactor coolant piping systems being designed with specific design and service limits to assure sufficient margin that the design conditions are not exceeded.

3.12.5 Combined License Information Items

Table 3.12-1 provides a list of ASME Code Class 1, 2, and 3 piping systems, piping components, and piping supports related COL information item numbers and descriptions from FSAR Tier 2, Table 1.8-2.

Table 3.12-1 U.S. EPR Combined License Information Items

Item No.	Description	FSAR Tier 2 Section
3.9-2	A COL applicant that references the U.S. EPR design certification will prepare the design specifications and design reports for ASME Class 1, 2, and 3 components, piping, supports and core support structures that comply with and are certified to the requirements of Section III of the ASME Code.	3.9-2

Item No.	Description	FSAR Tier 2 Section
3.12-1	A COL applicant that references the U.S. EPR design certification will perform a review of the impact of contributing mass of supports on the piping analysis following the final support design to confirm that the mass of the support is no more than 10 percent of the mass of the adjacent pipe span.	3.12.4.2
3.12-2	As indicated in Section 5.3 of topical report ANP-10264(NP), pipe and support stress analysis will be performed by the COL applicant that references the U.S. EPR design certification. If the COL applicant that references the U.S. EPR design certification chooses to use a piping analysis program other than those given in Section 5.1 of the topical report, the COL applicant will implement a benchmark program using models specifically selected for the U.S. EPR.	3.12.4.3
3.12-3	A COL applicant that references the U.S. EPR design certification will monitor the RHR/SIS/ EBS injection piping from the RCS to the first isolation valve (all four trains), and RHR/SIS suction piping from the RCS to the first isolation valve (trains 1 and 4) during the first cycle of the first U.S. EPR initial plant operation to verify that operating conditions have been considered in the design unless data from a similar plant's operation demonstrates that thermal oscillation is not a concern for piping connected to the RCS.	3.12.5.9
3.12-4	A COL applicant that references the U.S. EPR design certification will monitor pressurizer surge line temperatures during the first fuel cycle of initial plant operation to verify that the design transients for the surge line are representative of actual plant operations.	3.12.5.10.1

Item No.	Description	FSAR Tier 2 Section
3.12-5	A COL applicant that references the U.S. EPR design certification will monitor the normal spray line temperatures during the first cycle of the first U.S. EPR initial plant operation to verify that the design transients for the normal spray are representative of actual plan operations unless data from a similar plant's operation determines that monitoring is not warranted.	3.12.5.10.3
3.12-6	A COL applicant that references the U.S. EPR design certification will monitor the temperature of the main feedwater lines during the first cycle of the first U.S. EPR initial plant operation to verify that the design transients for the main feedwater lines are representative of actual plant operations unless data from a similar plant's operation determines that monitoring is not warranted.	3.12.5.10.4

3.12.6 Conclusions

The specified design and service combinations of loadings as applied to ASME Code Class 1, 2, and 3 piping systems are acceptable and meet the requirements of 10 CFR Part 50.55a, 10 CFR Part 52.47(b)(1), and GDC 1, GDC 2, GDC 4, GDC 14, and GDC 15, except for open items. This conclusion is based on the following.

The applicant has proposed quality assurance programs to correlate the test measurements with the analysis results. The programs constitute an acceptable basis for demonstrating the compatibility of the results from tests and analyses, through consistency between mathematical models used for different loadings, and the validity of the interpretation of the test and analysis results. Therefore, the applicant has met the relevant requirements of GDC 1 with respect to piping systems being designed and tested to quality standards commensurate with the importance of the safety functions to be performed.

The applicant has met the criteria with respect to the design and analyses of systems and components important to safety. These systems are designed to withstand the effects of earthquakes and the appropriate combinations of the effects of normal and postulated accident conditions with the effects of the SSE and, therefore, meets the requirements of GDC 2 and GDC 4.

The applicant has met the relevant requirements of GDC 2, GDC 4, and 10 CFR Part 50, Appendix S, by including seismic events in design transients which serve as design basis to withstand the effects of natural phenomena.

The applicant has met the applicable criteria including ASME Section III NB-3600, with respect to the design of the RCPB by ensuring that there is a low probability of rapidly propagating failure, gross rupture and that design conditions are not exceeded during normal operation, including anticipated operational occurrences. The applicant has provided an acceptable vibration, thermal expansion, and dynamic effects test program which will be conducted during startup and initial operation on specified high-and moderate-energy piping, and all associated systems, restraints and supports and therefore has met the relevant requirements of GDC 14 and GDC 15.

The applicant proposed piping ITAAC addresses as-designed piping and as-built piping. The applicant meets the requirements of 10 CFR 52.47(b)(1). The applicant meets the requirements of 10 CFR 50.55a, 10 CFR 52.47(b)(1), and GDC 1, GDC 2, GDC 4, GDC 14, and GDC 15 with respect to piping systems important to safety and these systems are designed to quality standards commensurate with the importance of the safety function to be performed.