Response to

Request for Additional Information No. 107 (1285, 1256, 1268), Revision 0

10/31/2008

U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 03.06.02 - Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping
SRP Section: 03.09.03 - ASME Code Class 1, 2, and 3 Components
Application Section: FSAR Ch. 3

QUESTIONS for Engineering Mechanics Branch 2 (ESBWR/ABWR Projects) (EMB2)

Question 03.06.02-1:

Branch Technical Position (BTP) 3-4, Part B, Item A(iv) states that in complex systems such as those containing arrangements of headers and parallel piping running between headers, the designer should identify and include all such piping within the designated run in order to postulate the number of breaks required by the criteria in Item A(iii). The staff did not find in U.S. EPR FSAR Section 3.6.2 the use of this criterion for postulating pipe break locations. Clarify if this criterion is applicable to U.S. EPR for postulating pipe break locations in high-energy piping.

Response to Question 03.06.02-1:

The criterion of BTP 3-4, Part B, Section A(iv) depends on the postulated number of breaks required by the criteria in BTP 3-4, Section A(iii). For the U.S. EPR, there are no break postulation criteria in BTP 3-4, Section A(iii) that would cause the number and location of breaks to change depending on how the piping runs are defined (see U.S. EPR FSAR Tier 2, Section 3.6.2.1.1.2). Therefore, no discussion of the criterion of BTP 3-4, Part B, Item A(iv) in U.S. EPR FSAR Tier 2, Section 3.6.2 is provided.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 03.06.02-2:

BTP 3-4, Part B, Items A(i) and B(i) do not allow exclusion from pipe break postulations based on systems with complete train separation and redundancy for both high- and moderate-energy systems inside and outside the containment, respectively. However, BTP 3-3, Item B.1c, allows such exclusion criteria for systems outside the containment, but not for systems inside the containment. In U.S. EPR FSAR Subsection 3.6.2.1.1.4, AREVA provides the separation criteria for postulating break and crack locations for high-energy systems separated from essential systems and components. It states that the criteria in FSAR Subsections 3.6.2.1.1.2 and 3.6.2.1.1.3 are only postulated if the consequences of the rupture can be shown to have an environmental effect on the essential equipment, such as an increased temperature in a room containing essential equipment that results from a high-energy line break in a nearby, separate room. It further states that rupture and target interactions need not be evaluated in cases where the essential system targets are in systems with complete train separation and redundancy. Similar criteria are also included in FSAR Subsection 3.6.2.1.2.4 for postulating pipe breaks in moderate energy systems. The staff determined that AREVA should clarify the criteria presented in FSAR Subsections 3.6.2.1.1.4 and 3.6.2.1.2.4 for high- and moderate-energy piping, respectively. Clarify the separation criteria, including those systems with complete train separation and redundancy, for systems inside and outside the containment for both high- and moderate-energy piping.

Response to Question 03.06.02-2:

AREVA NP will revise U.S. EPR FSAR Tier 2, Section 3.6.2.1.1.4 and Section 3.6.2.1.2.4 to clarify the separation criteria process as follows:

- Each rupture (break or crack) is postulated to follow the criteria of BTP 3-4, Part B, Sections A(iii), A(v), and B(iii) for high-energy and moderate-energy piping systems.

- Each postulated rupture is evaluated for its effects on essential systems and components, as required by GDC 4.

- Separation and redundancy for the respective system is used to eliminate the need for detailed evaluation and protection of a rupture’s essential targets. For example, if the essential system target is in one train of a four train redundant system, with complete separation of each train; the target can then be considered lost due to the rupture and no further evaluation of the dynamic and environmental effects is required.

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 3.6.2.1.1.4 and Section 3.6.2.1.2.4 will be revised as described in the response and indicated on the enclosed markup.

Question 03.06.02-3:

BTP 3-4, Part B, Item B(v) [see note 5 on page BTP 3-4-8] defines that piping subject to short operational period of time as 2 percent or less of the time the system operates under high-energy system conditions qualify as moderate-energy piping for the major operational period. In U.S. EPR FSAR Subsection 3.6.2.1.2.5, AREVA defines that leakage cracks, instead of breaks, are postulated in systems that are high-energy for only a short operational period of time (i.e., system operates under high energy system conditions is 2 percent or less of the time that the system operates under moderate-energy system conditions, or is less than or equal to 1 percent of the plant operating time), and moderate-energy for most of the time. However, the staff noted that the system definition relating to less than or equal to 1 percent of the plant operating time will also qualify high-energy systems to be considered as moderate-energy system. Provide technical justification for the use of this criterion relating to <u>1 percent or less of the plant operating time</u> and compare this with the criterion relating to <u>2 percent or less of the time system operates under high-energy conditions</u>.

Response to Question 03.06.02-3:

As noted in NUREG-1793, Section 3.6.2.1, the NRC determined that that no matter which definition of short operational period is used (1 or 2 percent); the resulting time from either definition is short enough that the likelihood of a break occurring during either period is small. Therefore, the NRC concluded that the definitions of high- and moderate-energy systems are consistent with that of the SRP and BTP.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 03.06.02-4:

In U.S. EPR FSAR Section 3.6.2.2, AREVA states that a COL applicant that references the U.S EPR design certification will provide information regarding the implementation of the design criteria relating to protective assemblies or guard pipes, including their final design and arrangement of the access openings used to examine the process pipe welds within such protective assemblies to meet the requirements of the in-service inspection program for the plant. The staff did not find this COL information item in FSAR Table 1.8-2. Justify why this COL information item is not included as one of the COL information items in FSAR Table 1.8-2.

Response to Question 03.06.02-4:

U.S. EPR FSAR Tier 2, Section 3.6.2.2 should not have included the combined license (COL) information item associated with RAI No. 107, Question 03.06.02-4. U.S. EPR FSAR Tier 2, Sections 3.6.2.1.1.1.3 and 3.6.2.2 address the design of guard pipe assemblies consistent with the guidance in BTP 3-4. U.S. EPR FSAR Tier 2, Section 3.6.2.2 will be revised to delete this COL information item.

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 3.6.2.2 will be revised as described in the response and indicated on the enclosed markup.

Question 03.06.02-5:

In U.S. EPR FSAR Section 3.6.2.3, AREVA provides details regarding assumptions in the piping dynamic analysis. The staff noted that SRP Section 3.6.2, item III.2.A provides dynamic analysis criteria and discusses material capacity limitations for a crushable material type of whip restraint, while SRP Section 3.6.2, item III.2.B discusses various methods of analyses. In addition, ANSI/ANS-58.2-1988, Section 6.3 presents several different types of dynamic analysis methods. Provide answers to the following:

(a) Acceptable dynamic models suggested in the SRP include lumped parameter analysis models, energy balance analysis models, and static analysis models. Also, alternate analytical approaches are discussed in ANS standard Sections 6.3.1 through 6.3.5. U.S. EPR FSAR Section 3.6.2.3 presents only two specific approaches: pseudo-dynamic analysis and dynamic time-history analysis methods. Clarify if any other analytical (nonlinear) methods and modeling techniques (discussed in the SRP and ANS standard) will be used for U.S. EPR plants.

(b) Discuss acceptable procedures and computer programs (for structural, thermo-hydraulic analyses) to be used to calculate the pipe whip dynamic responses of the piping. Also, include those computer codes (if different) to be used to evaluate the dynamic effects of essential piping and pipe components attached to the broken pipe as described in FSAR Section 3.6.2.4.

(c) Discuss the validation and verification (V/V) of the computer programs (per SRP Section 3.9.1) which the NRC staff has not yet approved.

Response to Question 03.06.02-5:

(a) In addition to the dynamic and pseudo-dynamic analyses methods provided in U.S. EPR FSAR Tier 2, Section 3.6.2.3, non-linear dynamic analyses and alternate analytical approaches—as discussed in ANS 58.2 Sections 6.3.1 through 6.3.5—may be used on a case-by-case basis. For example, a conservative static analysis approach, as described in Section 6.3.5 of ANS 58.2, may be used for pipe-whip analysis of smaller pipes that do not impart large loads to the concrete interface. For scenarios with large whip forces, a detailed dynamic analysis with material non-linearities imposed on the pipe and restraint sections may be performed to obtain more accurate loads on the interface concrete.

(b) U.S. EPR FSAR Tier 2, Section 3.9.1.2 provides a list of the computer codes that are used in the dynamic and static analyses of mechanical loads, stresses, and deformations, and in the hydraulic transient load analyses of Seismic Category I components and supports. These include:

- ANSYS, which is used to perform the dynamic non-linear structural analyses of pipe whip restraints for restraint systems.
- BWSPAN and SUPERPIPE, which are structural and piping analyses programs used to perform analyses using either the methods described in U.S. EPR FSAR Tier 2, Section 3.6.2.3 or the alternate approaches described in ANS 58.2 Sections 6.3.1 through 6.3.5. BWSPAN and SUPERPIPE are described in ANP-10264NP-A, "U.S. EPR Piping Analysis and Pipe Support Design Topical Report," which has been approved by the NRC (Reference 1).

U.S. EPR FSAR Tier 2, Section 3.9.1.2 and Section 3.9.1.5 will be revised to add computer code GT STRUDL, which has been approved by the NRC for U.S. EPR piping support analyses (Reference 1). The following tables and sections of U.S. EPR FSAR Tier 2, will also be revised to reference the approved version of ANP-10264NP-A (Reference 2): Table 1.6-1, Table 1.8-2, Section 3.6.1.4, Section 3.6.3.8, Section 3.7.3.2, Section 3.7.3.15, Section 3.8.6, Section 3.9.2.2.2, Section 3.9.2.7, Section 3.9.3.1.1, Section 3.9.3.5, Section 3.10.5, Section 3.12.1, Section 3.12.3.6, Section 3.12.4.3, Section 3.12.7, Section 3A.1, Section 3A.4, Section 3C.2.1, Section 3C.6, and Section 3C.7.

(c) Validation and verification of computer programs is provided in U.S. EPR FSAR Tier 2, Section 3.9.1 consistent with the guidance in SRP Section 3.9.1.

References for Question 03.06.02-5:

1. Letter, Getachew Tesfaye (NRC) to Ronnie L. Gardner (AREVA NP Inc.), “Final Safety Evaluation Report Regarding ANP-10264NP, ‘U.S. EPR Piping Analysis and Pipe Support Design Topical Report’ (TAC No. MD3128),” August 11, 2008.
2. Letter, Ronnie L. Gardner (AREVA NP Inc.) to Getachew Tesfaye (NRC), “Publication of ANP-10264NP-A ‘U.S. EPR Piping Analysis and Pipe Support Design Topical Report,’” NRC:08:086, November 7, 2008.

FSAR Impact:

U.S. EPR FSAR Tier 2, Table 1.6-1, Table 1.8-2, Section 3.6.1-4, Section 3.6.3.8, Section 3.7.3.2, Section 3.7.3.15, Section 3.8.6, Section 3.9.1.2, Section 3.9.1.5, Section 3.9.2.2.2, Section 3.9.2.7, Section 3.9.3.1.1, Section 3.9.3.5, Section 3.10.5, Section 3.12.1, Section 3.12.3.6, Section 3.12.4.3, Section 3.12.7, Section 3A.1, Section 3A.4, Section 3C.2.1, Section 3C.6, and Section 3C.7 will be revised as described in the response and indicated on the enclosed markup.

Question 03.06.02-6:

In FSAR Section 3.6.2.3.1, the applicant referred to Section 6.2 of ANSI/ANS 58.2-1988 for the development of the dynamic jet forces. The applicant also referred to simplified methodology provided in Appendix B of that standard. In addition, the applicant stated that the thermal hydraulic problem is solved using standard computer program or the analysis is simplified using the methodology from Appendix B of ANS 58.2 Standard. However, the precise physics, numerical methods, and level of validation of the referenced computational modeling are not given. The applicant is requested to supply some combination of references and written description yielding a complete picture of the physics, numeric, and validation status of the intended modeling. The applicant should also explain the criteria used to determine when the simplified modeling is to be applied and when the more detailed modeling is needed. In addition, the applicant is requested to consider the staff's concerns that are related to ANS 58-2 Standard as identified in RAIs 3.6.2-8 through 13. The applicant is requested to either substantiate the validity of the methods of ANS 58.2 for this analysis or to provide and substantiate a validated substitute.

Response to Question 03.06.02-6:

The simplified methodology provided in Appendix B of ANS 58.2 and U.S. EPR FSAR Tier 2, Section 3.6.2.3.2.1, for developing jet discharge forces, is the preferred method for jet impingement calculations. A description of the thrust and jet Impingement evaluation for the reactor coolant system (RCS) components using this methodology is provided in U.S. EPR FSAR Tier 2, Appendix C, Section 3C.3.1 and Section 3C.3.2. For cases where the stress qualification of the component is not possible by using the simplified methodology in ANS 58.2, either a more detailed jet thrust evaluation will be performed using a computational modeling approach or jet shields will be used to protect the component from the jet force. For example, a description of the thermal-hydraulic model for the reactor coolant loop piping and components, using the computer program CRAFT2, is provided in U.S. EPR FSAR Tier 2, Appendix C, Section 3C.1. A description of CRAFT2 and its verification is provided in U.S. EPR FSAR Tier 2, Appendix C, Section 3C.6 and U.S. EPR FSAR Tier 2, Section 3.9.1.2.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 03.06.02-7:

In U.S. EPR FSAR Section 3.6.2.4.1, AREVA provides criteria for evaluation for jet impingement. Such jet impingements depend on the placement of the target relative to the break, the jet energy discharged, the jet shape, and also the target characteristics (shape, structural, and dynamic characteristics). Clarify how the methods discussed are applied to safety-related SSCs (for other than piping systems) for establishing their structural integrity, and ensuring their operability.

Response to Question 03.06.02-7:

A response to this question will be provided by February 27, 2009.

Question 03.06.02-8:

(Introduction to RAI 3.6.2-8 through 13

The staff has considered the recent scrutiny of the ANS 58.2 expanding jet models by the Advisory Committee on Reactor Safeguards (ACRS) [Wallis - ADAMS ML050830344, Ransom - ADAMS ML050830341], which has revealed several inaccuracies that may lead to nonconservative assessments of the strength, zone of influence, and space and time-varying nature of the loading effects of supersonic expanding jets on neighboring structures. Wallis and Ransom also point out that initial blast waves are unaccounted for in the Standard.

The ACRS review of the ANS 58.2 jet models was motivated by Generic Safety Issue (GSI) 191, which addresses the blockage of strainers upstream of emergency sump pumps by particulate. The particulate is formed by fibrous ceramic insulation, which can be broken loose by blast waves and/or jets emanating from nearby pipe ruptures. The Wallis and Ransom critiques were cited in ACRS Safety Evaluation letters to the Chairman of the NRC (ACRSR-2097 - ML0429203342, and ACRSR-2110 ML043450346), and are cited in Section III.3 of Rev. 3 of SRP 3.6.2, dated March 2007. Also, examples of inconsistencies between existing standards for simulating the effects of LOCAs on neighboring structures are listed in the Knowledge Base for Emergency Core Cooling System Recirculation Reliability, February 1996, Issued by the NEA/CSNI, http://www.nea.fr/html/nsd/docs/1995/csni-r1995-11.pdf). Although the focus of the ACRS and the NEA/CSNI report was on debris generation and sump blockage, their comments directly impact the assessments of postulated pipe breaks on neighboring SSCs. The following RAIs (3.6.2-8 through 13) summarize the ACRS criticisms that relate specifically to possible non-conservatisms in the ANS 58.2. The applicant is advised that, as stated in section III.3 of Rev. 3 of SRP 3.6.2, the ANS 58.2 standard is no longer universally acceptable for modeling jet expansion in nuclear power plants.)

RAI 3.6.2-8

In the event of a high pressure pipe rupture, the first significant fluid load on surrounding structures would be induced by a blast wave. A spherically expanding blast wave is reasonably approximated to be a short duration transient and analyzed independently of any subsequent jet formation. Since the blast wave is not considered in the ANS 58.2 or the EPR FSAR for evaluating the dynamic effects associated with the postulated pipe rupture, omission of blast wave considerations is clearly non-conservative. Explain how the effects of blast wave loads on neighboring SSCs will be accounted for.

Response to Question 03.06.02-8:

The impact of a blast wave on neighboring structures, systems, and components (SSC) becomes smaller as the spherical wave expands. In an open cavity such as the reactor building, for example, the blast wave strength reduces rapidly. Therefore, the blast load on neighboring SSCs is considered negligible compared to a jet impingement load.

As described in U.S. EPR FSAR Tier 2, Appendix C, Section 3C1.4 and Section 3C3.4, asymmetric cavity pressurization (ACP) analyses for reactor coolant system (RCS) components are performed in accordance with NUREG-0609 to account for the blast waves and compartment pressurization from a high energy line break (HELB). The ACP loads are developed using the computer code COMPAR2 (see U.S. EPR FSAR Tier 2, Section 3.9.1.2), for the RCS loop HELB analysis, which considers the specific geometry of the room where the pipe rupture occurs. The input variables are the mass and energy release time history during the HELB event.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 03.06.02-9:

In the characterization of supersonic jets given by ANS 58.2, some physically incorrect assumptions underlie the approximating methodology. The model of the supersonic jet itself is given in Figures C-1 and C-2 of the Standard and contains references to supposedly universal jet characteristics that are not reasonable. A fundamental problem is the assumption that a jet issuing from a high pressure pipe break will always spread with a fixed 45 degree angle up to an asymptotic plane and subsequently spread at a constant 10 degree angle. Each of these characteristics is generally inapplicable and far from universal. Initial jet spreading rate is highly dependent on the ratio of the total conditions of the source flow to the ambient conditions. In reality, subsequent spreading rates depend, at a given axial position, on the ratio of the static pressure in the outermost jet flow region to the ambient static pressure. In the Standard, the asymptotic plane is described as the point at which the jet begins to interact with the surrounding environment. In his critique, Dr. Wallis takes this to mean that the jet is subsonic downstream of the asymptotic plane. In fact, as shown by Wallis and Ransom, supersonic or not, the jet is highly dependent on the conditions in the surrounding medium, and, at a given distance from the issuing break, will spread or contract at a rate depending on the local jet conditions relative to the surrounding fluid pressure.

Supersonic jet behavior can persist over distances from the break far longer than those estimated by the standard, extending the zone of influence of the jet, and the number of SSCs that could be impacted by a supersonic jet. For example, tests in the Seimens-KWU facility in Karlstein, Germany showed that significant damage from steam jets can occur as far away as 25 pipe diameters from a rupture (Knowledge Base for Emergency Core Cooling System Recirculation Reliability, NEA/CSNI/R (95)11, February 1996, Issued by the NEA/CSNI, http://www.nea.fr/html/nsd/docs/1995/csni-r1995-11.pdf).

The applicant is requested to:

(a) Explain what analysis and/or testing has been used to substantiate the use of the ANS 58.2 Appendix C for defining conservatively which SSCs are in jet paths and the subsequent loading areas on the SSCs.

(b) The applicant states in FSAR Section 3.6.1.1.1 that ‘Components that are in the path of steam or subcooled liquid that can flash at the break are assumed to fail if they are within a distance of ten (10) pipe diameters (broken pipe outside diameter) from the break’. Substantiate this assumption in light of the findings in NEA/CSNI/R (95)11 that steam jets can cause significant damage at distances up to 25 pipe diameters.

Response to Question 03.06.02-9:

A response to this question will be provided by February 27, 2009.

Question 03.06.02-10:

The ANS 58.2 Standard’s formulas for the spatial distribution of pressure through a jet cross-section are incorrect, as pointed out by Wallis and Ransom. In some cases, the Standard’s assumption that the pressure within a jet cross section is maximum at the jet centerline is correct (near the break, for instance), but far from the break, the pressure variation is quite different, often peaking near the outer edges of the jet. Applying the Standard’s formulas could lead to non-conservative pressures away from the jet centerline.

The applicant is requested to:

(a) Explain what analysis and/or testing has been used to substantiate use of ANS 58.2 Appendix D for defining conservatively the net jet impingement loading on SSCs in light of the information presented by Ransom and Wallis (ADAMS ML050830344, ADAMS ML050830341), which challenges the accuracy of the pressure distribution models presented in ANS 58.2.

(b) Expand the table of all postulated break types (FSAR Table 3.6.1-1 on pages 3.6-12 to 3.6-13) to include the properties of the fluid internal and external to the ruptured pipe. The table should specify what type of jet the applicant assumes will emanate from each pipe break – incompressible nonexpanding jet, or compressible supersonic expanding jet - along with how impingement forces will be calculated for each jet. Specific examples of jet impingement loading calculations made using the ANS 58.2 Standard for the postulated piping breaks in an EPR should be given, along with proof that the calculations lead to conservative impingement loads in spite of the cited inaccuracies and omissions in the ANS 58.2 models pointed out by Ransom and Wallis.

Response to Question 03.06.02-10:

A response to this question will be provided by February 27, 2009.

Question 03.06.02-11:

On FSAR page 3.6-37, the applicant stated that "…the jet impingement load on a target is the force exerted on the target by the jet. This dynamic problem is not only dependent on the jet forcing function, but also on the dynamic characteristics of the target in question. It can be solved dynamically with a model of the target, and utilizing the jet forcing function; however, it can also be solved using an equivalent static approach." In either approach, there does not appear to be any consideration of potential feedback between the jet and any nearby reflecting surface(s), which can increase substantially the dynamic jet forces impinging on the nearby target component and the dynamic thrust blowdown forces on the ruptured pipe through resonance.

The applicant should consider that supersonic expanding jets are known to be unsteady, particularly those impinging on nearby structures. The applicant should examine the following reference, "Knowledge Base for Emergency Core Cooling System Recirculation Reliability, February 1996, Issued by the NEA/CSNI," (http://www.nea.fr/html/nsd/docs/1995/csni-r1995-11.pdf), which states that tests in Germany's Heissdampfreactor (HDR) showed high dynamic (oscillating) loads in the immediate vicinity of breaks.

Free jets are notoriously unsteady and, in the case of supersonic jets, such strong unsteadiness will tend to propagate in the shear layer and induce unsteady (time-varying oscillatory) loads on obstacles in the flow path. Pressures and densities vary nonmonotonically with distance along the axis of a typical supersonic jet and this in turn feeds and interacts with shear layer unsteadiness. In addition, for a typical supersonic jet, interaction with obstructions will lead to backward-propagating transient shock and expansion waves that will cause further unsteadiness in downstream shear layers.

In some cases, synchronization of the transient waves with the shear layer vortices emanating from the jet break can lead to significant amplification of the jet pressures and forces (a form of resonance) that is not considered in the ANS 58.2 Standard or FSAR Tier 2. Should the dynamic response of the neighboring structure also synchronize with the jet loading time scales, further amplification of the loading can occur, including that at the source of the jet. These feedback phenomena are well-known to those in the aerospace industry who work with aircraft that use jets to lift off and land vertically [see, for example Ho, C.M., and Nosseir, N.S., "Dynamics of an Impinging Jet. Part 1. The Feedback Phenomenon," Journal of Fluid Mechanics, Vol. 105, pp. 119-142, 1981]. Some general observations by past investigators are that strong discrete frequency loads are observed when the impingement surface is within 10 diameters of the jet opening, and that when resonance within the jet occurs, significant amplification of impingement loads can result (Ho and Nosseir show a factor of 2-3 increase in pressure fluctuations at the frequency of the resonance). The applicant is requested to:

(a) Provide information that establishes that the applicant's interpretation of the jet impingement force as static is conservative.

(b) Explain whether any postulated pipe break locations are within 10 diameters of a neighboring SSC (or barrier/shield), and if so, how jet feedback/resonance and resulting dynamic load amplification are accounted for.

(c) Clarify whether dynamic jet loads are to be considered, and if so, using what methods. Also, should the dynamic loading include strong excitation at discrete frequencies corresponding to resonance frequencies of the SSC impinged upon, provide the basis for assuming a static analysis with a dynamic load factor of two is conservative.

Response to Question 03.06.02-11:

A response to this question will be provided by February 27, 2009.

Question 03.06.02-12:

Explain quantitatively how reflections of jets by neighboring structures will be considered.

Response to Question 03.06.02-12:

The loads on structures, systems, and components (SSC) due to reflection of jets from neighboring structures are negligible because the jets lose energy upon impact and rebound. The reflected jet spray loads for the SSCs are small compared to the safe shutdown earthquake and pipe break loads they are designed to withstand.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 03.06.02-13:

The applicant describes the use of barriers or shields between the postulated rupture and essential equipment (FSAR Sections 3.6.2.4.1 and 3.6.2.5.1.1). If the barriers or shields are close to postulated jets, these nearby surfaces can induce feedback and resonance within the jets, potentially destroying the barrier or shield. Explain how the barriers or shields will be designed so that they will not be damaged or destroyed by dynamic jet resonant loading.

Response to Question 03.06.02-13:

ANS 58.2 Appendices C and D are used to determine the jet impingement load. Structures such as barriers, shields, or enclosures are conservatively designed to withstand jet impingement and other large loads.

As stated in ANS 58.2 section 7.3, either a dynamic analysis or an equivalent static analysis may be performed for jet impingement loads to evaluate targets and barriers. In the absence of a dynamic analysis, an equivalent static analysis may be used where a dynamic load factor (DLF) of two is applied to determine the jet load. A time history analysis or equivalent static load application is used to determine the structural response of barriers, shields, and enclosures.

Barriers or shields that are identified as necessary by the pipe break hazards evaluation are designed for worst-case pipe break loads. The closest high-energy pipe location and resultant loads are used to size the barriers or shields.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 03.06.02-14:

In U.S. EPR FSAR Section 3.6.2.4.2, AREVA provides criteria for significant dynamic loads and displacements in essential system piping attached to the broken pipe. A factor in the definition of such a dynamic problem is the existence and location of check valves in the remaining unbroken piping. With closing check valves, the faster the closing time for the valve the higher the dynamic loading in the remaining piping. No details of this analysis method are included. Discuss the analysis methods and assumptions that will be used to evaluate the dynamic effects of essential system piping attached to the broken pipe.

Response to Question 03.06.02-14:

Check valve closure following postulated pipe breaks upstream of the check valve location is part of the analysis for Seismic Category 1 piping. The analysis includes conservative boundary conditions (e.g., a double ended guillotine break and bounding fluid conditions feeding the break). The methodology contains mechanistic models of lift or swing check valves and is capable of predicting the motion of the valve disk using the general equation of motion of a rigid body. Viscous effects as the check valve disk moves are included, as applicable. The methodology yields accurate predictions of check valve closing transients due to pipe breaks and subsequent water hammer surges and loadings in the analyzed piping system.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 03.06.02-15:

U.S. EPR FSAR Section 3.6.2.5.1.2 states that pipe whip restraints are designed for a one-time accident event; so they are designed to undergo deformation as long as the whipping pipe is fully restrained for the entire time of the blowdown event. The staff noted that the loads to be evaluated in combination with pipe break forces are not discussed. Discuss the loads and load combination methods to be used in the design of whip restraints.

Response to Question 03.06.02-15:

For pipe whip restraints that are designed for a one-time accident event, the deadweight of the pipe (when applicable) would be combined with the pipe break forces. The applicability of applying deadweight depends on the orientation of the pipe and the whip restraint.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 03.06.02-16:

SRP Section 3.6.2, Item III.4 states that analyses of pipe-break dynamic effects on mechanical components and supports should include the effects of both internal reactor pressure vessel asymmetric pressurization loads and expanded asymmetric compartment pressurization loads, as appropriate, as discussed for pressurized water reactor (PWR) primary systems in NUREG-0609. Explain how the effects of both internal RPV asymmetric pressurization loads and expanded asymmetric compartment pressurization loads, as appropriate, are included in the analysis of pipe-break dynamic effects on mechanical components and supports.

Response to Question 03.06.02-16:

Per NUREG-0609, the high energy line break (HELB) dynamic analyses of the reactor coolant loop and its components considers the effects of reactor pressure vessel (RPV) internal pressurization loads and compartment asymmetric cavity pressurization (ACP) loads.

The RPV internal pressure loads are calculated by creating a detailed hydraulic model of the RPV internals and reactor coolant system (RCS) loop, and performing a blowdown analysis due to a HELB. These blowdown loads are then imposed on the structural model of the RCS to obtain the effects of dynamic HELB. Descriptions of the hydraulic modeling, hydraulic loading analysis, and structural loading analysis are provided in U.S. EPR FSAR Tier 2, Appendix C, Section 3C, Section 3C.3, and Section 3C.4, respectively.

The ACP loads are evaluated by creating a compartment hydraulic model and performing a cavity pressurization analysis using the mass and energy release due to a HELB as input. The forces on the components due to ACP are evaluated and then imposed on the structural model to obtain the effects of dynamic HELB. The ACP hydraulic modeling is described in U.S. EPR FSAR Tier 2, Appendix C, Section 3C.1.4. Further information on the ACP wave loading analysis and dynamic structural loading analysis are provided in U.S. EPR FSAR Tier 2, Appendix C, Section 3C.3.4 and Section 3C.4.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 03.06.02-17:

10 CFR 52.47(b)(1), which requires that a design certificate application contain the proposed inspections, tests, analyses, and acceptance criteria (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 with the design certification, the provisions of the Atomic Energy Act, and the NRC's regulations. As described in SRP 14.3.3, Piping Systems and Components – ITAAC, one ITAAC item that should be included is to complete a pipe break evaluation report that documents that structures, systems, and components (SSCs) that are required to be functional during and following an SSE have adequate high-energy pipe break mitigation features. The design description should discuss the criteria used to postulate pipe breaks, the analytical methods used to performed pipe breaks, and the methods to confirm the adequacy of the results of the pipe break analyses. The design description should be verified in a pipe break analysis report that provides assurance that the postulated pipe break analyses have been completed. The pipe break analysis report shall conclude that, for each postulated pipe failure, the reactor can be shut down safely and maintained in a safe, cold shutdown condition without offsite power. Detailed information that supports this ITAAC should be contained in FSAR Tier 2 Chapter 3. Furthermore, an as-built reconciliation review of this pipe break analysis report should be included in the ITAAC.

Based on its review of the EPR FSAR, the staff found that the applicant has not provided sufficient information related to the pipe break analysis report as described above. Specifically, EPR Tier 1 ITAAC Table does not contain any ITAAC item related to pipe break analysis report. In addition, FSAR Tier 2 Chapter 3.6.2 does not contain a section that lists/summarizes the specific information that will be included in the pipe break analysis report. Moreover, the applicant did not provide any justification and closure milestone for the COL Holder Items identified in FSAR Tier 2, Section 3.6.2 as described in RG 1.206 C.III.4.3. The applicant is requested to:

(a) In Tier 1 ITAAC table, include information regarding the as-built pipe break evaluation report. If the applicant intends to address the completion of as-designed pipe break evaluation with the closure of an as-designed ITAAC item, then that ITAAC item should also be included. Also, the closure milestone of that as-designed pipe break evaluation report will need to be addressed.

(b) In Tier 2 Section 3.6.2, identify a list of information that will be included in that pipe break evaluation report.

(c) Provide justification including the closure milestone as described in RG. 1.206 C.III.4.3 for the COL Holder Items identified in Section 3.6.2.

Response to Question 03.06.02-17:

a) See the response to RAI No. 80, Question 03.06.01-2.

b) As noted in U.S. EPR FSAR Tier 2, Section 3.6.2.1, the combined license (COL) applicant is responsible for performing the pipe break hazards analysis. There is no guidance in RG 1.206 or SRP 3.6.2 to identify in the FSAR a list of information that will be included in a pipe break hazards analysis. In the response to RAI No. 80, Question 03.06.01-2, AREVA NP provided a new ITAAC (i.e., U.S. EPR Tier 1, Table 2.1.1-7, item 4.15) for the pipe break hazards analysis. The acceptance criteria for this ITAAC states that this analysis will confirm whether:

- Piping stresses in the containment penetration area are within allowable stress limits.
- Pipe whip restraints and jet shield designs can mitigate pipe break loads.
- Loads on safety-related structures, systems, and components (SSCs) are within design load limits.
- SSCs are protected or qualified to withstand the environmental effects of postulated failures.

c) The COL applicant is responsible for the information requested in RG 1.206 C.III.4.3, including implementation schedules and closure milestones.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 03.09.03-1:

In FSAR Section 3.9.3, AREVA states that the EPR design is based on the 2004 edition of ASME Code, Section III, Division 1, with no addenda. 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.

Response to Question 03.09.03-1:

See the response to RAI No. 51, Question 05.02.01.01-4.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 03.09.03-2:

In FSAR Section 3.9.3, there is no discussion on how each of the EPR pressure boundary safety-related mechanical components and component supports is designed. The staff requests that AREVA include in FSAR Section 3.9.3 a list of pertinent 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.

Response to Question 03.09.03-2:

Classification of pertinent U.S. EPR pressure boundary safety-related components, including the codes and standards used, is provided in U.S. EPR 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 U.S. EPR FSAR Tier 2, Section 3.9.6. The seismic and dynamic qualification of mechanical and electrical equipment is described in U.S. EPR FSAR Tier 2, Section 3.10.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 03.09.03-3:

In FSAR Section 3.9.3.1, AREVA states that 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. However, a comprehensive description of plant conditions is not provided. The staff requests that AREVA provide a description for each of the plant conditions and their relations to frequency of occurrence, and describe major plant events accounted for in the plant conditions. In addition to the information provided in Table 3.9.3-1 through 3.9.3-4 in EPR FSAR Tier 2, the staff also requests that AREVA develop a table equivalent to Table I of Appendix A to SRP 3.9.3 which would more comprehensively illustrate EPR plant events, system operating conditions, service loading combinations, and service stress limits, as described in the Appendix.

Response to Question 03.09.03-3:

AREVA NP does not understand the regulatory basis for this question, since Appendix A to SRP 3.9.3, Table I is a summary of the guidance provided in this appendix and not necessarily information identified as to be provided by the applicant in the FSAR. The information provided in U.S. EPR FSAR Tier 2, Table 3.9.1 through Table 3.9.3-4 is consistent with the guidance provided in RG 1.206 and SRP 3.9.3. Additionally, the information in these tables is consistent with the information provided in ANP-10264NP-A, “U.S. EPR Piping Analysis and Pipe Support Design Topical Report,” which has been approved by the NRC and was reviewed in accordance with the guidance of SRP 3.9.3 (see Section 3.4.3 of Reference 1)

Reference for Question 03.09.03-3:

1. Letter, Getachew Tesfaye (NRC) to Ronnie L. Gardner (AREVA NP Inc.), “Final Safety Evaluation Report Regarding ANP-10264NP, ‘U.S. EPR Piping Analysis and Pipe Support Design Topical Report,’ (TAC No. MD3128),” August 11, 2008.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 03.09.03-4:

In FSAR Section 3.9.3, AREVA states that 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 supports structures that comply with and are certified to the requirements of Section III of the ASME Code. In order for the staff to reach a reasonable assurance finding based on the requirements of 10 CFR 52.47, however, certain information is required during the NRC review of the design certification application. The staff requests that AREVA 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 to be procured, and that the DCD design methodologies and criteria are adequately reflected in the associated component design specifications. As for the design reports, the staff requests that AREVA discuss in the DCD its plan and schedule of making the as-designed design reports of EPR major mechanical components available for NRC audit, e.g., through an ITTAC, to ensure that a vehicle of verifying the completion of the EPR component design is established.

Response to Question 03.09.03-4:

Based on discussions with the NRC on October 29, 2008, regarding this question, AREVA NP understands that the information requested in this question pertains to the design specifications required for ASME Code Class 1, 2, and 3 components. As noted in the above question, and based on the discussions with NRC, AREVA NP also understands that NRC is interested in the availability of these design specifications in preparation for an NRC audit. A representative sample of the design specifications will be available for NRC inspection beginning April 1, 2009. The design specifications for Class 1 components are located in our Lynchburg, VA office and the design specifications for Class 2 and 3 components are located in our Charlotte, NC office.

AREVA NP does not understand the regulatory basis for including the plan and schedule for the design specifications in the FSAR. As noted in the question and in U.S EPR FSAR, Tier 2, Section 3.9.3, the combined license (COL) applicant is responsible for preparing 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. Additionally, inspection, test, analysis, and acceptance criteria (ITAAC) have been established to implement this COL information (e.g., U.S. EPR FSAR Tier 1, Table 2.2.1-5).

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 03.09.03-6:

In FSAR Section 3.9.3, AREVA 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. The staff requests AREVA to elaborate on this COL commitment and explain the differences between this COL item and the commitment discussed in a separate COL item by which design specifications and design reports are required to be made available for NRC audit.

Response to Question 03.09.03-6:

1) The combined license (COL) information item in U.S. EPR FSAR Tier 2, Section 3.9.3 addresses the information that the NRC requests of the COL applicant in Part C.III of RG 1.206, Section C.I.3.9.3.1:

> "Include the following information for ASME Code Class 1 components and component supports, if applicable:
>
> (3) summary of the maximum total stress, deformation, and cumulative usage factor values for each of the component operating conditions for all ASME Code Class I components."

As noted in the referenced section of RG 1.206, this COL information item is to "identify those values that differ from the allowable limits by less than 10 percent, and provide 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."

2) There is no specific COL item for design specifications and design reports to be made available for NRC audit. As noted in the response to RAI No. 107, Question 03.09.03-4, U.S EPR FSAR, Tier 2, Section 3.9.3 contains a COL information item that COL applicant is responsible for preparing 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.

Thus, the COL information item in item 1 above is applicable to ASME Code Class 1 components, whereas the COL information item in item 2 above applies to ASME Code Class 1, 2, and 3 components, piping, supports, and core support structures

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 03.09.03-7:

In FSAR Section 3.9.3.1, AREVA 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. The staff requests AREVA to 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.

Response to Question 03.09.03-7:

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). Additional information on active pumps and valves is provided in U.S. EPR FSAR Tier 2, Section 3.9.3.3. Also, refer to the response to RAI No. 49, Question 03.09.06-1.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 03.09.03-8:

In FSAR Section 3.9.3.1.1, AREVA states that the effects of the environment on fatigue for Class 1 piping and components are addressed in FSAR Section 3.12 and in Section 3.4 of Reference 2 (i.e., Piping Topical Report, ANP-10264NP). However, the staff found that only piping is addressed in the mentioned references. Explain where in the FSAR environmental fatigue for ASME Code Section III components is discussed.

Response to Question 03.09.03-8:

U.S. EPR FSAR Tier 2, Section 3.12.5.19 will be revised to address both Class 1 piping and components.

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 3.12.5.19 will be revised as described in the response and indicated on the enclosed markup.

Question 03.09.03-9:

In FSAR Table 3.9.3-1 and Table 3.9.3-3 (Load Combinations and Acceptance Criteria for ASME Class 1 Components, for primary plus secondary stress intensity category under Upset condition), explain why earthquake inertial load is not listed as a potential loading. Also, for faulted condition, explain why a secondary stress category is not included and why the anchor motion effect of SSE is not considered in the design of components. Reference SECY 93-087.

Response to Question 03.09.03-9:

A response to this question will be provided by February 27, 2009.

Question 03.09.03-10:

In FSAR Section 3.9.3.1, for loads for components and component supports, 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 Section III service level stress criteria.

Response to Question 03.09.03-10:

A response to this question will be provided by February 27, 2009

Question 03.09.03-11:

In FSAR Section 3.9.3.3, confirm that the stresses in active valve bodies and pump casings conform to the requirements in SRP Section 3.10 for faulted conditions.

Response to Question 03.09.03-11:

Consistent with the guidance in SRP 3.9.3, stresses in active valve bodies conform to the guidance in SRP 3.10. U.S. EPR FSAR Tier 2, Section 3.9.3 states: "Section 3.10 describes the methods and criteria for seismic qualification testing of Seismic Category I mechanical equipment." Additionally U.S. EPR FSAR Tier 2, Section 3.9.3.3.1 and Section 3.9.3.3.2 contain references to U.S. EPR FSAR Tier 2, Section 3.10.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 03.09.03-12:

In FSAR Section 3.9.3.2, provide a detailed description of the tests that are conducted to address the testing requirements in TMI Action Item II.D.1 of NUREG-0737, or provide a reference in FSAR where this is discussed.

Response to Question 03.09.03-12:

Three Mile Island (TMI) Action Item II.D.1 of NUREG-0737 concerns relief and safety valve test requirements. Conformance with the TMI requirements, in accordance with 10 CFR 50.34(f), is addressed in U.S. EPR FSAR Tier 2, Table 1.9-3. As noted in U.S. EPR FSAR Tier 2, Table 1.9-3, item (2)(x), the test program for reactor coolant system pressure relief valves is described in U.S. EPR FSAR Tier 2, Section 3.9.6, Section 5.2.2, and Section 14.2.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 03.09.03-13:

In FSAR Section 3.9.3.1.7, AREVA states that Table 3.9.3-4 provides the loading combinations and corresponding stress design criteria per ASME Service Level for ASME Class 1, 2, and 3 component supports. ASME Code Section III, Subsection NF, Table NF-3131(a)-1 is referenced for providing a cross-reference to various sections of NF for stress allowables for specific types of component supports. AREVA also states that the allowable stress criteria are supplemented by RGs 1.124 and 1.130 for Class 1 linear-type and plate-and-shell-type support structures, respectively. In looking through the NF sections listed in Table NF-3131(a)-1 the staff was not able to identify design stress criteria which are specifically applicable to snubbers. The staff requests AREVA to provide the design stress criteria that are specifically applicable to snubbers, and discuss where the criteria are referenced.

Response to Question 03.09.03-13:

AREVA NP 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 Section III, Subsections NCA and NF. ASME 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. As described in U.S. EPR FSAR Tier 2, Section 3.9.3.4.5, additional information on snubber supports for piping systems is described in Section 6.6 of ANP-10264NP-A, “U.S. EPR Piping Analysis and Pipe Support Design Topical Report,” which has been approved by the NRC (Reference 1).

Reference for Question 03.09.03-13:

1. Letter, Getachew Tesfaye (NRC) to Ronnie L. Gardner (AREVA NP Inc.), “Final Safety Evaluation Report Regarding ANP-10264NP, ‘U.S. EPR Piping Analysis and Pipe Support Design Topical Report,’ (TAC No. MD3128),” August 11, 2008.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 03.09.03-14:

In FSAR Section 3.9.3.4.5, AREVA 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. The staff requests AREVA to address the following: (1) provide a description of the AREVA snubber production test program and qualification test program, for both mechanical and hydraulic snubbers; (2) provide justification if the production tests do not consider all snubbers in the population; (3) explain the basis of selecting samples for qualification tests, if sampling method is used; (4) discuss the procedures taken to demonstrate the required snubber load ratings; (5) discuss the acceptance criteria used to ensure that the snubber design comply with the specific requirements of ASME Code Section III, Subsection NF; (6) discuss the specific functional parameters (activation level, release rate, drag, dead band, etc.) considered for snubber production and qualification testing; (7) provide the acceptable codes and standards (including editions) used for the snubber production and qualification testing; (8) verify that the production operability testing for large-bore hydraulic snubbers (greater than 50kips load rating) includes (i) a full Service Level D load test to verify sufficient load capacity, (ii) testing at the full load capacity to verify proper bleed with the control valve closed, (iii) testing to verify that the control valve closes within the specified velocity range, and (iv) testing to demonstrate that breakaway and drag forces are within the acceptable design limits.

Response to Question 03.09.03-14:

See the response to RAI No. 107, Question 03.09.03-13. The snubber vendor is responsible for the snubber production and qualification test programs in accordance with the applicable ASME Code standards and the AREVA NP design specifications. Information on the inservice testing of snubbers is provided in U.S. EPR FSAR Tier 2, Section 3.9.6.

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

Question 03.09.03-15:

In FSAR Tier 2, Appendix 3C, Section 3C.2, concerning mathematical modeling of major components, AREVA 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. The staff requests that AREVA discuss how these local flexibilities are formulated for the beam elements representing the components, and how they are included in the mathematical model.

Response to Question 03.09.03-15:

The local flexibilities of the component shells at the attachment points are calculated using the design information contained in References 1 and 2. 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 calculate local stiffnesses. The local stiffness values in each direction are assembled to form a stiffness matrix which is incorporated in the four-loop structural model in BWSPAN (see U.S. EPR FSAR Tier 2, Section 3.9.1.2) at the appropriate location.

References for Question 03.09.03-15:

1. Bijlaard P.P., “Stresses from Radial Loads and External Moments in Cylindrical Pressure Vessels,” Welding Journal 34 (12), Research Supplement, 608-s to 617-s (1955).
2. Bijlaard P.P., ‘Stresses from Radial Loads in Cylindrical Pressure Vessels,” Welding Journal, 33 (12), Research Supplement, 615-s to 623-s (1954).

FSAR Impact:

The U.S. EPR FSAR will not be changed as a result of this question.

U.S. EPR Final Safety Analysis Report Markups

03.06.02-5

Table 1.6-1—Reports Referenced
Sheet 1 of 4

Report No. (See Notes 1, 2, and 3)	Title	Date Submitted to NRC	FSAR Section Number(s)
ANF-89-060P-A ANF-89-060NP-A plement 1	Generic Mechanical Design Report High Thermal Performance Spacer and Intermediate Flow Mixer	3/28/91	4.2
ANP-10263P-A ANP-10263NP-A	Codes and Methods Applicability Report for the U.S. EPR	11/06/07	4, 5.1, 15, 16, and 19
ANP-10264NP-A	U.S. EPR Piping Analysis and Pipe Support Design Topical Report	~~9/29/06~~11/07/08	3.6, 3.7, 3.8, 3.9, 3.10, 3.12, App. 3A, and App. 3C
ANP-10266-A	AREVA NP Inc. Quality Assurance Plan (QAP) for Design Certification of the U.S. EPR Topical Report	06/18/07	7.1, 17.1, 17.2, 17.3, 17.5, 18.1, 18.7, and 18.11
ANP-10268P-A ANP-10268NP-A	U.S. EPR Severe Accident Evaluation Topical Report	~~10/31/06~~ 2/26/08	6.2.5, 15.4, 19.1, and 19.2
ANP-10269P-A ANP-10269NP-A	The ACH-2 CHF Correlation for the U.S. EPR Topical Report	~~11/30/06~~ 3/10/08	4.4, 5, 7, 15, and 19
ANP-10272	Software Program Manual TELEPERM XS™ Safety Systems Topical Report	12/21/06	7.1 and 7.6
ANP-10273P ANP-10273NP	AV42 Priority Actuation and Control Module Topical Report	11/28/06	7 and 16
ANP-10275P-A ANP-10275NP-A	U.S. EPR Instrument Setpoint Methodology Topical Report	~~3/26/07~~ 2/26/08	7 and 16
ANP-10278P ANP-10278NP	U.S. EPR Realistic Large Break Loss of Coolant Accident Topical Report	3/26/07	6.2 and 15
ANP-10279	U.S. EPR Human Factors Engineering Program Topical Report	1/29/07	3.4, 7.1, 13.1, and 18
ANP-10281P ANP-10281NP	U.S. EPR Digital Protection System Topical Report	3/27/07	3.1.3, 4.6, 7, and 8.1
ANP-10282P ANP-10282NP	POWERTRAX/E Online Core Monitoring Software for the U.S. EPR Technical Report	11/27/07	4.4
ANP-10283P ANP-10283NP	U.S. EPR Pressure-Temperature Limits Methodology for RCS Heat-Up and Cool-Down Technical Report	12/6/07 ~~(Note 4)~~	5.3 and 16
ANP-10284	U.S. EPR Instrumentation and Controls Diversity and Defense-in-Depth Methodology Topical Report	6/20/07	7

Table 1.8-2—U.S. EPR Combined License Information Items
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Item No.	Description	Section	Action Required by COL Applicant	Action Required by COL Holder
3.8-6	A COL applicant that references the U.S. EPR design certification will confirm that site-specific loads lie within the standard design envelope for RB internal structures, or perform additional analyses to verify structural adequacy.	3.8.3.3	Y	
3.8-7 03.06.02-5	A COL applicant that references the U.S. EPR design certification will confirm that site-specific conditions for Seismic Category I buried conduit, electrical duct banks, pipe, and pipe ducts satisfy the requirements specified in Section 3.8.4.4.5 and those specified in AREVA NP Topical Report ANP-10264NP-A.~~(NP), U.S. EPR Piping Analysis and Support Design, September 2006.~~	3.8.4.5	Y	
3.8-8	A COL applicant that references the U.S. EPR design certification will address site-specific Seismic Category I structures that are not described in this section.	3.8.4.1	Y	
3.8-9	A COL applicant that references the U.S. EPR design certification will describe site-specific foundations for Seismic Category I structures that are not described in this section.	3.8.5.1	Y	
3.8-10	A COL applicant that references the U.S. EPR design certification will evaluate site-specific methods for shear transfer between the foundation basemats and underlying soil for soil parameters that are not within the envelope specified in Section 2.5.4.2.	3.8.5.5	Y	
3.8-11	A COL applicant that references the U.S. EPR design certification will evaluate and identify the need for the use of waterproofing membranes and epoxy coated rebar based on site-specific groundwater conditions.	3.8.5.6.1	Y	
3.8-12	A COL applicant that references the U.S. EPR design certification will describe the program to examine inaccessible portions of below-grade concrete structures for degradation and monitoring of groundwater chemistry.	3.8.5.7	Y	

Table 1.8-2—U.S. EPR Combined License Information Items
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03.06.02-5

Item No.	Description	Section	Action Required by COL Applicant	Action Required by COL Holder
3.9-4	As noted in ANP-10264NP-A~~(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		Y
3.9-5	As noted in ANP-10264NP-A~~(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	Y	
3.9-6	A COL applicant that references the US EPR design certification will identify any additional site-specific valves in Table 3.9.6-2 to be included within the scope of the IST program.	3.9.6.3	Y	
3.9-7	A COL applicant that references the U.S. EPR design certification will submit the preservice testing (PST) program and IST program for pumps, valves, and snubbers as required by 10 CFR 50.55a.	3.9.6		Y
3.9-8	A COL applicant that references the US EPR design certification will identify any additional site-specific pumps in Table 3.9.6-1 to be included within the scope of the IST program.	3.9.6.2	Y	
3.9-9	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~~a U.S. EPR benchmark program using models specifically selected for the U.S. EPR.	3.9.1.2		Y
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		Y

Table 1.8-2—U.S. EPR Combined License Information Items
Sheet 20 of 42

03.06.02-5

Item No.	Description	Section	Action Required by COL Applicant	Action Required by COL Holder
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 ten percent of the mass of the adjacent pipe span.	3.12.4.2		Y
3.12-2	As indicated in Section 5.3 of topical report ANP-10264NP-A~~(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 listed 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		Y
3.13-1	A COL applicant referencing the U.S. EPR design certification will submit the inservice inspection ~~plan~~program for ASME Code Class 1, Class 2, and Class 3 threaded fasteners, to the NRC prior to performing the first inspection. The program will identify the applicable edition and addenda of ASME Section XI and ensure compliance with the requirements of 10CFR50.55a(b)(2)(xxvii).	3.13.2		Y
3E-1	A COL applicant that references the U.S. EPR design certification will address critical sections relevant to site-specific Seismic Category I structures.	3E	Y	
5.2-1	~~A COL applicant that references the U.S. EPR design certification will identify subsequent ASME Code editions or addenda that may be used and will determine the consistency of the U.S. EPR design with construction practices (including inspection and examination methods) reflected within the subsequent code editions and addenda identified in the COL application.~~Deleted	~~5.2.1.1~~	~~Y~~	

features perform properly. Furthermore, as noted in Section 3.6.1.2.1, where separation is not practical, deflectors, shielding, and barriers are used to protect essential components and equipment. Therefore, failure modes and effects have been evaluated to verify that the consequences of failures of high- and moderate-energy lines do not affect the ability to safely shut down the plant.

3.6.1.4 References

1. Branch Technical Position 3-3, Revision 3, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," U.S. Nuclear Regulatory Commission, March 2007.

2. Branch Technical Position 3-4, Revision 2, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," U.S. Nuclear Regulatory Commission, March 2007.

3. NUREG-0800, Revision 3, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, March 2007.

4. NUREG/CR-2913, "Two-Phase Jet Loads," U.S. Nuclear Regulatory Commission, January 1983.

5. ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Facility Components," The American Society of Mechanical Engineers, 2004.

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6. ANP-10264NP-A, Revision 0, "U.S. EPR Piping Analysis and Pipe Support Design," AREVA NP Inc., ~~September 2006~~November 2008.

If a structure separates a high-energy line from essential equipment, the structure is designed to withstand the consequences of a break in that line that yields the greatest effect at the structure, even if the criteria described in Section 3.6.2.1.1.2 may not require the break location to be postulated.

5. Environmental Qualification of Safety-Related Mechanical and Electrical Equipment.

   Safety-related equipment is environmentally qualified according to Section 3.11. The environmental qualification of electrical equipment, inside and outside containment, includes postulated breaks and cracks in the design bases. Section 3.11 also addresses the qualification of mechanical equipment.

3.6.2.1.1.3 Leakage Crack Locations in High-Energy Piping Systems

1. ASME Code, III, Division 1 – Class 1 Piping in Areas other than Containment Penetration Areas.

   Leakage cracks in Class 1 piping are postulated at axial locations where the stress range calculated by Equation 10 from Paragraph NB-3653 in Section III of the ASME Code, exceeds $1.2S_m$.

2. ASME Code, III, Division 1 – Class 2, 3, and non-ASME Code Class Piping in Areas other than Containment Penetration Areas.

   With the exception of the portions of piping identified in Item 2 in Section 3.6.2.1.1.1 above, leakage cracks in ASME Code Class 2, 3, and non-ASME Code piping are postulated at axial locations where the stress calculated by the sum of Equations 9 and 10 from Paragraph NC/ND-3653 in Section III of the ASME Code, exceeds 0.4 times the sum of the stress limits given in NC/ND-3653.

3. Unanalyzed Non-Safety Class Piping.

   Non-safety-class piping that does not have detailed stress information has a through-wall crack postulated at axial locations that yield the most severe environmental consequences.

3.6.2.1.1.4 High-Energy Fluid Systems Separated From Essential Systems and Components

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~~For high-energy lines that are separated from essential systems and components, break and crack locations based on the criteria in Section 3.6.2.1.1.2 and Section 3.6.2.1.1.3 are only postulated if the consequences of the rupture can be shown to have an environmental effect on the essential equipment, such as an increased temperature in a room containing essential equipment that results from a high-energy line break in a nearby, separate room. Similarly, rupture and target interactions need not be evaluated in cases where the essential system targets are in systems with complete train separation and redundancy.~~ For essential system targets identified for a high-energy line rupture that are part of a system with complete separation and redundancy, the evaluation of such targets need only identify this separation and redundancy, as the target may be considered lost due to the rupture without having an adverse impact on essential equipment. The U.S. EPR design has many essential systems with redundant safety trains located in each of four separate Safeguard Buildings. This four train separation and redundancy allows for one train to be lost due to the rupture, while a second train is lost due to single active failure and a third is down due to normal maintenance. With the fourth train still capable of operating the system, protection for the dynamic and environmental effects of these ruptures need not be considered.~~the effects of ruptures need not be specifically evaluated or protection provided.~~

3.6.2.1.2 Locations of Leakage Cracks in Moderate Energy Lines

3.6.2.1.2.1 Leakage Crack Locations in Fluid Systems in Containment Penetration Areas

Leakage cracks are not postulated in those portions of moderate-energy lines that extend from the containment wall up to and including the inboard and outboard containment isolation valves where they meet the requirements of Subarticle NE-1120 in Section III of the ASME Code, and where the Level A or Level B stress calculated by the sum of Equations 9 and 10 from Paragraph NC-3653 does not exceed 0.4 times the sum of the stress limits given in NC-3653.

3.6.2.1.2.2 Leakage Crack Locations in Fluid Systems in Areas other than Containment Penetration Areas

With the exception of the portions of piping identified in Section 3.6.2.1.2.1, leakage cracks are postulated at the following locations:

1. Through-wall cracks are postulated in piping located adjacent to safety-related SSCs except:

   A. When the piping is exempted by the criteria in Section 3.6.2.1.2.1 or Section 3.6.2.1.2.3.

   B. Where the ASME Code Class 1 piping stress range is calculated by Equation 10 from Paragraph NB-3653 in Section III of the ASME Code is less than 1.2 S_m.

   C. Where ASME Code Class 2, 3, and non-safety piping stresses are calculated by the sum of Equations 9 and 10 from Paragraph NC/ND-3653 in Section III of the ASME Code are less than 0.4 times the sum of the stress limits in NC/ND-3653.

2. Leakage cracks, unless exempted by Item 1 above, are postulated at circumferential and axial locations that yield the most severe environmental consequences.

3. Leakage cracks are postulated in piping designed to non-seismic standards at locations where the resultant leakage impacts the functional capability of affected essential equipment.

3.6.2.1.2.3 Moderate-Energy Fluid Systems in Close Proximity to High-Energy Fluid Systems

Leakage cracks are not postulated in moderate-energy lines where the crack is in close proximity to a high-energy line break, as long as the crack does not result in more limiting environmental conditions than the high-energy line break. When a leakage crack in a moderate-energy line causes more limiting environmental conditions than a proximate high-energy line break, the provisions from Section 3.6.2.1.2.2 are used.

3.6.2.1.2.4 Moderate-Energy Fluid Systems Separated from Essential Systems and Components

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~~For moderate-energy lines that are separated from essential systems and components, leakage crack locations based on the criteria inSection 3.6.2.1.2.2 are postulated if the consequences of the crack can be shown to have an effect on the essential equipment, such as an increased temperature in a room containing essential equipment that results from a moderate-energy line leakage crack in a nearby, separate room. Similarly, rupture and target interactions need not be evaluated in cases where the essential system targets are in systems with complete train separation and redundancy.~~For essential system targets identified for a moderate-energy line crack that are part of a system with complete separation and redundancy, the evaluation of such targets need only identify this separation and redundancy, as the target may be considered lost due to the rupture without having an adverse impact on essential equipment. The U.S. EPR design has many essential systems with redundant safety trains located in each of four separate Safeguard Buildings. This four train separation and redundancy allows for one train to be lost due to the rupture, while a second train is lost due to single active failure and a third is down due to normal maintenance. With the fourth train still capable of operating the system, protection for the effects of these ruptures (e.g., spray wetting) need not be considered.~~the effects of ruptures need not be specifically evaluated, or protection provided.~~

3.6.2.1.2.5 Fluid Systems that Qualify As Moderate-Energy Systems Based on Short Operational Periods as High-Energy

Leakage cracks, instead of breaks, are postulated in systems that are high-energy for only a short operational period of time and moderate-energy for most of the time. The operational period is defined as “short” if the time that the system operates under high-energy system conditions is two percent or less of the time that the system operates under moderate-energy system conditions, or is less than or equal to one percent of the plant operating time.

- Guard pipe assemblies are tested to a pressure not less than their design pressure.
- Design pressure and temperature are not less than the maximum operating pressure and temperature of the enclosed pipe during normal plant conditions.
- Containment design pressure and temperature, combined with safe shutdown earthquake loading, does not cause stress in the guard pipe assemblies to exceed Level C service limits from Subarticle NE-3220 in Section III of the ASME Code.
- Guard pipes do not prevent access for performing inservice inspections of piping welds, as required by Item 3 in Section 3.6.2.1.1.1. Additional information on inservice inspection and testing of the reactor coolant pressure boundary is provided in Section 5.2.4 and 6.6.8.

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- ~~A COL applicant that references the U.S. EPR design certification will provide information regarding the implementation of the design criteria relating to protective assemblies or guard pipes, including their final design and arrangement of the access openings used to examine the process pipe welds within such protective assemblies to meet the requirements of the inservice inspection program for the plant.~~

3.6.2.3 Analytical Methods to Define Forcing Functions and Response Models

Movement of pipe, due to pipe breaks and cracks, is analyzed to show that the motion does not result in overstress of any structure, system, or component important to safety. This section will address the criteria for dynamic or pseudo-dynamic analysis of piping systems, targets, and protection devices. Criteria for the dynamic analysis that will be followed are:

- For each postulated pipe break an analysis of the dynamic response of the broken pipe is performed.
- In the case of circumferential pipe breaks, the need for a pipe whip dynamic analysis is determined based on the driving energy of the fluid.
- Mass inertia and stiffness properties of the systems, elastic and inelastic deformation of piping systems, impact and rebound, and support boundary conditions are adequately accounted for when calculating the dynamic response of piping and restraints.
- Loading condition (pressure, temperature, and inertial effects) prior to rupture is used in the evaluation of postulated breaks. For piping pressurized during normal power operation, the initial conditions are the greater of system energy at hot standby or at 102 percent of rated power.
- Crushable material used to dissipate the energy of a moving pipe is limited to 80 percent of its rated energy dissipating capacity. A 10 percent increase of the design yield strength (S_Y) is used to account for strain rate effects.

10. ANP-10264NP-A, Revision 0, "U.S. EPR Piping Analysis and Pipe Support Design Topical Report," AREVA NP Inc., ~~September 2006~~November 2008.

11. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," The American Society of Mechanical Engineers, 2004.

12. NUREG/CR-6446, "Fracture Toughness Evaluation of TP304 Stainless Steel Pipes," U.S. Nuclear Regulatory Commission, February 1997.

13. NUREG/CR-4082, Volume 8, "Degraded Piping Program - Phase II," Semiannual Report," U.S. Nuclear Regulatory Commission, March 1989.

14. NUREG/CR-4599, "Short Cracks in Piping and Piping Welds," First Semiannual Report, U.S. Nuclear Regulatory Commission, March 1991.

15. ASTM Standard E1820-01, "Standard Test Method for Measurement of Fracture Toughness," American Society for Testing and Materials International, 2001.

16. NUREG/CR-6177, "Assessment of Thermal Embrittlement of Cast Stainless Steels," U.S. Nuclear Regulatory Commission, May 1994.

17. EPRI NP-3395, "Calculation of Leak Rates Through Cracks in Pipes and Tubes," Electric Power Research Institute, 1983.

18. NUREG/CR-3464, "The Application of Fracture Proof Design Methods Using Tearing Instability Theory to Nuclear Piping Postulating Circumferential Through Wall Cracks," Nuclear Regulatory Commission, September 1983.

19. NUREG/CR-1319, "Cold Leg Integrity Evaluation," U.S. Nuclear Regulatory Commission, February 1980.

20. NUREG/CR-5128, Revision 1, "Evaluation and Refinement of Leak-Rate Estimation Models," U.S. Nuclear Regulatory Commission, June 1994.

21. SQUIRT: Seepage Quantification of Upsets in Reactor Tubes, User's Manual, Windows Version 1.1, Battelle, March 24, 2003.

22. NUREG/CR-6861, "Barrier Integrity Research Program," U.S. Nuclear Regulatory Commission, December 2004.

23. EPRI NP-5596, "Elastic-Plastic Fracture Analysis of Through-Wall and Surface Flaws in Cylinders," Electric Power Research Institute, January 1988.

24. NUREG/CR-4878, "Analysis of Experiments on Stainless Steel Flux Welds," Nuclear Regulatory Commission, April 1987

25. NUREG/CR-6837, Volume 2, "The Battelle Integrity of Nuclear Piping (BINP) Program Final Report Summary and Implications of Results," Appendices, U.S. Nuclear Regulatory Commission, June 2005.

spectrum analyses. Relative motion of support locations, when determined to be significant, is considered. Every seismic support is considered active in this analysis.

3.7.3.2 Determination of Number of Earthquake Cycles

Criteria are established for the evaluation of distribution subsystems and for mechanical and electrical equipment for the effects of seismic-induced fatigue when fatigue is expected to have a significant effect on the design. Because the U.S. EPR design does not consider OBE load cases, the effects of seismic-induced fatigue are evaluated in accordance with SECY 93-087 (Reference 5) and SRP 3.7.3 of NUREG-0800 (Reference 6).

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Seismic-induced fatigue of piping systems is described in the AREVA NP Topical Report ANP-10264NP-A~~10264 NP~~ (Reference 1)~~ and the AREVA NP letter NRC:07:028 (Reference 2)~~. The consideration of low-level seismic effects (i.e., fatigue) is required by IEEE Std 344-2004[1] (Reference 7) to qualify electrical and mechanical equipment with the equivalent of five OBE events followed by one SSE event (with 10 maximum stress cycles per event). This consideration includes the seismic qualification process based on the approach provided in Reference 5 and outlined in SRP 3.10.III.3.C of Reference 6. To meet this requirement, earthquake cycles included in the fatigue analysis are composed of five one-third SSE events followed by one full SSE event. A number of fractional peak cycles equivalent to the maximum peak cycles for five one-half SSE events may be used in accordance with Appendix D of Reference 7 when followed by one full SSE event. This approach results in consideration of fractional peak cycles.

The effects of seismic-induced fatigue on distributed subsystems other than piping and electrical and mechanical equipment are evaluated, and when determined as appropriate the effects are evaluated using the same guidance from Reference 5 and SRP 3.7.3 of Reference 6 for piping systems. To meet this requirement, earthquake cycles included in the fatigue analysis are composed of two SSE events, with 10 maximum stress-cycles each, for a total of 20 full cycles. This is considered equivalent to the cyclic load basis of one SSE and five OBEs. Alternatively, the methods of Appendix D of Reference 7 may be used to determine a number of fractional vibratory cycles equivalent to 20 full SSE cycles. When this method is used, the amplitude of the vibration is taken as one-third of the amplitude of the SSE resulting in 300 fractional SSE cycles to be considered.

1. Section 3.11 provides the justification for the use of the latest version of the IEEE standards referenced in this section that have not been endorsed by existing Regulatory Guides. AREVA NP maintains the option to use current NRC-endorsed versions of the IEEE standards.

shall be limited to one percent and two percent of the pipe diameter, respectively. To eliminate compressive wrinkling of the pipe, the allowable axial strain is computed. These strain limits apply to both encased pipes and pipes surrounded by soil. For the case of pipes anchored to a building with potential for ground settlement, total allowable strain limit, ε_a, is limited to four percent of the pipe diameter in addition to satisfying the axial strain limit.

Section 3.8.4.1.8 describes requirements placed on the COL applicants to provide a description of Seismic Category I buried conduit and duct banks.

3.7.3.13 Methods for Seismic Analysis of Category I Concrete Dams

There are no Seismic Category I concrete dams in the U.S. EPR design. A COL applicant that references the U. S. EPR design certification will provide a description of methods used for seismic analysis of site-specific Category I concrete dams, if applicable.

3.7.3.14 Methods for Seismic Analysis of Aboveground Tanks

Dynamic pressure on fluid containers in the in-containment refueling water storage tank (IRWST), spent fuel pool, and other fluid reservoirs due to the SSE are considered in accordance with ASCE 4-98 (Reference 4). Section 3.7.1.3 presents damping values for seismic analysis of aboveground tanks. Damping values for concrete aboveground tanks are seven percent of critical for impulsive modes and 0.5 percent for sloshing mode. These damping values are taken from Table 3.7.1-1.

Seismic analyses of concrete above-ground tanks consider impulsive and convective forces of the water, as well as the flexibility of the tank walls and floor, and ceiling of the tank. For the spent fuel pool, cask loading pit, cask washdown pit, and fuel transfer canal, the impulsive loads are calculated by considering a portion of the water mass responding with the concrete walls (see Section 3.7.2.3). Impulsive forces are calculated by conventional methods for tanks determined to be rigid. For non-rigid tanks, the effect of tank flexibility on spectral acceleration is included when determining the hydrodynamic pressure on the tank wall for the impulsive mode.

The IRWST is analyzed using finite element methods by including it in the 3D FEM model of the internal structures described in Section 3.7.2 and detailed in Section 3.8.3.

3.7.3.15 References:

1. ANP-10264NP-A, "U.S. EPR Piping Analysis and Support Design Topical Report," AREVA NP Inc., ~~September 2006~~November 2008.

2. Deleted. ~~Ronnie L. Gardner (AREVA NP Inc.) to Document Control Desk (NRC), AREVA NP letter NRC:07:028 dated July 13, 2007, "Response to a Request for~~

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~~Additional Information Regarding AREVA NP Topical Report, ANP-10264NP, 'U.S. EPR Piping Analysis and Support Design,' (TAC No. MD3128)," NRC: 07:028, July 13, 2007.~~

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3. ASCE "Seismic Response of Buried Pipe and Structural Components," ASCE Committee on Seismic Analysis of Nuclear Structures and Material, American Society of Civil Engineers, 1983.

4. ASCE Standard 4-98, "Seismic Analysis of Safety-Related Nuclear Structures and Commentary," American Society of Civil Engineers, September 1986.

5. SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water (ALWR) Designs," U.S. Nuclear Regulatory Commission, July 1993.

6. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, March 2007.

7. IEEE 344-2004, "Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," Institute of Electrical and Electronics Engineers, 2004.

8. NUREG-1061, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee," U.S. Nuclear Regulatory Commission, (Vol. 1) August 1984, (Vol. 2) April 1985, (Vol. 3) November 1984, (Vol. 4) December 1984, (Vol. 5) April 1985.

9. W.S. Tseng, "Equipment Response Spectra Including Equipment–Structure Interaction Effects," 1989 Pressure Vessel and Piping Conference, ASME PVP, Volume 155.

10. ACI 349-01/349R-01, Appendix C, "Code Requirements for Nuclear Safety Related Concrete Structures and Commentary," American Concrete Institute, January 2001.

Next File

34. ASTM A519-06, "Standard Specification for Seamless Carbon and Alloy Steel Mechanical Tubing," American Society for Testing and Materials, 2006.

35. ASTM A576-06, "Standard Specification for Steel Bars, Carbon, Hot-Wrought, Special Quality," American Society for Testing and Materials, 2006.

36. ASTM A416-06, "Standard Specification for Steel Strand, Uncoated Seven-Wire for Prestressed Concrete," American Society for Testing and Materials, 2006.

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37. ANP-10264NP-A, Revision 0, "U.S. EPR Piping Analysis and Pipe Support Design," AREVA NP Inc., ~~September 2006~~November 2008.

38. ASTM C94-06, "Standard Specification for Ready-Mixed Concrete." American Society for Testing and Materials, 2006.

39. ACI 308.1-98, "Standard Specification for Curing Concrete," American Concrete Institute, Inc., 1998.

40. ACI 311.4R-05, "Guide for Concrete Inspection," American Concrete Institute, Inc., 2005.

41. ACI 349.1R-07, "Reinforced Concrete Design for Thermal Effects on Nuclear Power Plant Structures," American Concrete Institute, Inc., 2007.

42. AISC 303-05, "Code of Standard Practice for Steel Buildings and Bridges," American Institute of Steel Construction, Inc.

43. ANSI/AISC 341-05, "Seismic Provisions for Structural Steel Buildings," American National Standards Institute, 2005.

44. AISC 348-04, "Specification for Structural Joints Using ASTM A325 or A490 Bolts," American Institute of Steel Construction, 2004.

45. ANSI/AWS D1.8-2005, "Structural Welding Code – Seismic Supplement." American National Standards Institute, 2005.

46. ASTM F1554-07, "Standard Specification for Anchor Bolts, Steel, 36, 55, and 105 ksi Yield Strength," American Society for Testing and Materials, 2007.

47. ASTM C150-07, "Standard Specification for Portland Cement," American Society for Testing and Materials, 2007.

48. ASTM C595-07, "Standard Specification for Blended Hydraulic Cements," American Society for Testing and Materials, 2007.

49. ACI 306R-88, "Cold-Weather Conditioning," American Concrete Institute, 1988.

50. ACI 308R-01, "Guide to Curing Concrete," American Concrete Institute, 2001.

51. ASTM A82-07, "Standard Specification for Steel Wire, Plain, for concrete Reinforcement," American Society for Testing and Materials, 2007.

the fatigue analyses. Significant emergency cycles are those that result in stresses higher than the endurance limits on the ASME design fatigue curves.

The transient conditions selected for equipment fatigue evaluation are based on a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from various operating conditions in the plant. The transients selected are representative of operating conditions which are considered to occur during plant operation and are severe or frequent enough to be of possible significance to component cyclic behavior and fatigue life. The transients selected are a conservative representation of transients which, when used as a basis for component fatigue evaluation, provide confidence that the component is appropriate for its application over the design life of the plant.

Although the U.S. EPR will be operated as a base-loaded plant, the reference U.S. EPR design provides robust features for the effects of load follow. Similarly, the structural design and analysis of the RCS, RCS components, RCS component internals, and systems ancillary to the RCS account for the effects of load follow.

3.9.1.2 Computer Programs Used in Analyses

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The following computer codes are used in the dynamic and static analyses of mechanical loads, stresses, and deformations, and in the hydraulic transient load analyses of Seismic Category I components and supports. ~~A complete list of programs will be included in the ASME Code design reports.~~ ~~As noted in AREVA NP letter NRC:07:028 (Reference 3),~~ T~~t~~he 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 the test problems described in Appendix C and Reference 2 and 3. 03.06.02-5

- ANSYS and ANSYS CFX: ANSYS is a commercially available finite element analysis code for structural, stress, fatigue, and heat transfer analysis. It is used to perform stress and fatigue analyses of pressure vessels and their internals, as well as other complex geometries. Static and transient temperatures and pressures and applied mechanical loads can be modeled.

  ANSYS CFX is a commercially available finite element analysis code for computational fluid dynamic analysis. It is used in the analysis of the RPV internals to generate temperature profiles considering fluid heat transfer and internal heat generation (gamma heating).

  ANSYS and ANSYS CFX are owned and maintained by ANSYS, Inc. Validation of the ANSYS and ANSYS CFX computer codes is accomplished by executing verification cases and comparing the results to those provided by ANSYS, Inc. Each document that describes an ANSYS analysis includes information regarding the verification analysis and its results. Error notices from ANSYS, Inc. are processed and records pertaining to error notification, tracking, and disposition are available for NRC inspection.

- BWSPAN: Information on this computer code is provided in Section 5.1 of Reference 2 ~~and in Reference 3~~.

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- BWHIST, BWSPEC, COMPAR2, CRAFT2, P91232, and RESPECT: Information on these computer codes is provided in Appendix 3C.

- RELAP B&W: This is an advanced system analysis computer code designed to analyze a variety of thermal-hydraulic transients in light water reactor systems. As a system code, it provides simulation capabilities for the reactor primary coolant system, secondary system, feedwater trains, control systems, and core neutronics. Special component models include pumps, valves, heat structures, electric heaters, turbines, separators, and accumulators. Code applications include the full range of safety evaluation transients, loss of coolant accidents, and operating events. The code has been benchmarked to test facility data as documented in RELAP5/MOD2-B&W – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis (Reference 4).

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- S-RELAP5: Information on this computer code is provided in Section 15.0.2. S-RELAP5 evolved from the AREVA NP ANF-RELAP code. S-RELAP5 was benchmarked against a series of LOFT experiments and against ANF-RELAP simulations.

- SUPERPIPE: Information on this computer code is provided in Section 5.1 of Reference 2. ~~and in Reference 3.~~

- GTSTRUDL: Information on this computer code is provided in Section 5.1 of Reference 2.

As addressed in ~~Reference 3~~Reference 2, there are three representative calculations from the analyses for the U.S. EPR design certification to be used in the benchmark program. These calculations utilize the piping analysis codes identified in Section 5.1 of Reference 2. As noted in Reference 2, pipe stress and support analysis will be performed by a COL applicant that references the U.S. EPR design certification. 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~~a U.S. EPR benchmark program using models specifically selected for the U.S. EPR.

3.9.1.3 Experimental Stress Analysis

No experimental stress analysis methods are used for Category I systems or components.

3.9.1.4 Considerations for the Evaluation of the Faulted Condition

Section 3.9.3 describes the analytical methods used to evaluate stresses for Seismic Category I systems and components subjected to faulted condition loading.

3.9.1.5 References

1. ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Facility Components," The American Society of Mechanical Engineers, 2004.

2. ANP-10264NP-A, Revision 0, "U.S. EPR Piping Analysis and Pipe Support Design," AREVA NP Inc., ~~September 29, 2006~~November 2008.

3. Deleted. ~~Letter, Ronnie L. Gardner (AREVA NP, Inc.) to Document Control Desk (NRC), "Response to a Request for Additional Information Regarding AREVA NP Topical Report ANP-10264(NP), 'U.S. EPR Piping Analysis and Support Design Topical Report,' (TAC No. MD3128)," NRC:07:028, July 13, 2007.~~

4. BAW-10164P-A, Revision 6, "RELAP5/ MOD2-BAW – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analyses," AREVA NP Inc., June 2007.

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3.9.2.2.1 Seismic Qualification Testing

The methods and criteria for seismic qualification testing of Seismic Category I mechanical equipment ~~and a description of the seismic operability criteria~~ are described in Section 3.10.

3.9.2.2.2 Seismic Sub-System Analysis Methods

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Descriptions of the U.S. EPR seismic analysis methods for safety-related piping are provided in Reference 2. ~~and AREVA NP letter NRC:07:028 dated July 13, 2007 (Reference 4).~~ Methods for seismic analysis of systems, components, and equipment are also addressed in Section 3.7.3, Appendix 3A, and Appendix 3C.

3.9.2.2.3 Determination of Number of Earthquake Cycles

See Section 3.7.3.

3.9.2.2.4 Basis for Selection of Frequencies

See Section 3.7.3.

3.9.2.2.5 Three Components of Earthquake Motion

See Section 3.7.3, Appendix 3C, and Section 4.2 of Reference 2.

3.9.2.2.6 Combination of Modal Responses

See Section 4.2.2.3 of Reference 2.

3.9.2.2.7 Analytical Procedures for Piping

See Appendix 3C and Section 4.2 of Reference 2.

3.9.2.2.8 Multiple-Supported Equipment Components with Distinct Inputs

See Section 4.2.2.2 of Reference 2.

3.9.2.2.9 Use of Constant Vertical Static Factors

See Section 3.7.3.

3.9.2.2.10 Torsional Effects of Eccentric Masses

See Section 5.2 of Reference 2.

3.9.2.2.11 Buried Seismic Category I Piping Conduits, and Tunnels

See Section 3.7.3 and Section 3.10 of Reference 2.

Analysis of the RPV internals for blowdown loads resulting from a guillotine break of the safety injection line nozzles on the hot and cold legs is performed using direct step-by-step integration methods. Note that breaks are not considered in the main coolant loop piping (hot and cold legs), pressurizer surge line, and main steam line piping (from the steam generators to the first anchor point location) due to the application of leak-before-break methodology to these lines (see Section 3.6.3). The forcing functions obtained from hydraulic analysis of the safety injection line breaks are defined at points in the RPV internals where changes in cross-section or direction of flow occur, such that differential loads are generated during the blowdown transient. Additional details of the structural analysis of the RPV Isolated Model for LOCA loading are given in Appendix 3C.

Analysis of the RPV internals for safe shutdown earthquake (SSE) loading uses direct step-by-step time-history analysis techniques. The SSE analysis of the RPV Isolated Model is described in Appendix 3C.

The response of the RPV internals to SSE loading are combined with their response to the safety injection line breaks by the square-root-of-the-sum-of-the-squares method. Section 3.9.3 provides the faulted load combinations considered in the stress and fatigue analyses of the RPV internals.

3.9.2.6 Correlations of Reactor Internals Vibration Tests with the Analytical Results

The results of the dynamic analysis of the RPV internals are compared to the results of preoperational tests, and this comparison verifies that the analytical model provides appropriate results. If the predicted responses differ significantly from the measured values, the vibration responses are determined with the measured forcing function as input.

3.9.2.7 References

1. ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Facility Components," The American Society of Mechanical Engineers, 2004.

2. ANP-10264NP-A, Revision 0, "U.S. EPR Piping Analysis and Pipe Support Design Topical Report," AREVA NP Inc., ~~September 2006~~November 2008.

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3. ASME OM-S/G-2000, "Standards and Guides for Operation and Maintenance of Nuclear Power Plants," The American Society of Mechanical Engineers, 2000.

4. Deleted. ~~Ronnie L. Gardner (AREVA NP Inc.) to Document Control Desk (NRC), AREVA NP letter NRC:07:028 dated July 13, 2007, "Response to a Request for Additional Information Regarding AREVA NP Topical Report, ANP-10264(NP), 'U.S. EPR Piping Analysis and Support Design,' (TAC No. MD3128)," NRC:07:028, July 13, 2007.~~

deadweight analyses of components. Deadweight loads are further described in Sections 3.3.1.2 and 6.3.1 of Reference 2.

Thermal Expansion

The effects of restrained thermal expansion and contraction on piping and supports are described in Section 3.3.1.3 and Section 6.3.2 of Reference 2.

Seismic

Analyses of seismic inertial loads and anchor movements on piping systems and the RCS are described in Sections 3, 4, and 6 of Reference 2 and 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 Class 1 components and piping. The number of safe shutdown earthquake (SSE) stress cycles included in the fatigue analysis is identified in FSAR Section 3.7.3 and in Section 3.4.1 of Reference 2.

System Operating Transients

Analyses of system operating transients, including fluid transient loadings, on piping systems and the RCS are discussed in Sections 3.3.1.5 and 6.3.4 of Reference 2 and Appendix 3C, respectively. Thermal and pressure transients are described in Section 3.3.1.8 of Reference 2. Section 3.3.1.5 of Reference 2 also describes water and steam hammer loads. The analysis of these transients results in force time histories for application in the piping analyses.

Wind and Tornado

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Wind and tornado loads are discussed in Sections 3.3.1.6, 6.3.5, and 6.3.6 of Reference 2. As noted in ANP-10264NP-A~~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.

Pipe Break

Loads due to pipe breaks are described in Section 3.3.1.7 of Reference 2. 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 (see Section 3.6.3).

Pipe break load design condition and service level evaluations are described in Sections 6.3.7, 6.3.8, and 6.3.9 of Reference 2. Design basis pipe breaks are categorized as Level C. Main steam and main feedwater pipe breaks and LOCA are categorized as Level D.

Friction

Friction loads are described in Section 6.10 of Reference 2.

Minimum Pipe Support Design Loads

Minimum design loads are described in Section 6.3.11 of Reference 2. 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.

Thermal Stratification, Cycling, and Striping

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Thermal stratification, cycling, and striping (including applicable NRC Bulletins 79-13, 88-08, and 88-11) are described in Section 3.7 of Reference 2. The pressurizer surge line is analyzed with the main coolant loop piping and supports as described in Appendix 3C. As noted in ANP-10264NP-A~~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.

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.

Environmental Fatigue

The effects of the environment on fatigue for Class 1 piping and components are addressed in FSAR Section 3.12 and in Section 3.4 of Reference 2.

3.9.3.1.2 Load Combinations and Stress Limits for Class 1 Components

Table 3.9.3-1—Design Conditions, Load Combinations, and Stress Criteria for ASME Class 1 Components provides the loading combinations and corresponding stress design criteria per ASME Service Level for ASME Class 1 components.

3.9.3.1.3 Load Combinations and Stress Limits for Class 2 and 3 Components

Table 3.9.3-2—Load Combinations and Acceptance Criteria for ASME Class 2 and 3 Components provides the loading combinations and corresponding stress design criteria per ASME Service Level for ASME Class 2 and 3 components.

3.9.3.1.4 Load Combinations and Stress Limits for Class 1 Piping

Table 3-1 of Reference 2 provides the loading combinations and corresponding stress design criteria per ASME Service Level for ASME Class 1 piping.

3.9.3.4.7 Seismic Self-Weight Excitation

Seismic self-weight excitation, including the response of the support structure to SSE loadings, is described in Section 6.8 of Reference 2.

3.9.3.4.8 Design of Supplemental Steel

The design of supplemental steel is described in Section 6.9 of Reference 2.

3.9.3.4.9 Pipe Support Gaps and Clearances

The use of pipe support gaps in the piping analysis is described in Section 6.11 of Reference 2.

3.9.3.4.10 Instrumentation Line Support Criteria

Instrumentation line support criteria are described in Section 6.12 of Reference 2.

3.9.3.4.11 Pipe Deflection Limits

Pipe deflection limits are described in Section 6.13 of Reference 2.

3.9.3.4.12 Load Combinations and Stress Limits for Buried Piping

As noted in Section 3.10 of Reference 2, Code Class 2 and 3 Seismic Category I buried piping systems are analyzed for pressure, weight, thermal expansion, and seismic loads using dynamic or equivalent static load methods. Further information on this analysis is provided in Section 3.10 of Reference 2. Table 3-4 of Reference 2 provides the design conditions, load combinations, and stress criteria for the qualification of buried piping.

3.9.3.4.13 Model Isolation Methods

The overlap region and influence zone model isolation methods are used to divide large seismic piping systems that cannot be separated by structural methods or decoupling criteria. These methods are similar, in that a section of the piping system is used as the boundary of the models. These methods are further described in Section 5.4.3 and Figure 5-3 of Reference 2.

3.9.3.5 References

1. ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Facility Components," The American Society of Mechanical Engineers, 2004.

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2. ANP-10264NP-A, "U.S. EPR Piping Analysis and Pipe Support Design Topical Report," AREVA NP Inc., ~~September 2006~~November 2008.

3. NUREG-1482, Revision 1, “Guidelines for Inservice Testing at Nuclear Power Plants,” U.S. Nuclear Regulatory Commission, January 2005.

4. Generic Letter 96-05, “Periodic Verification of Design Basis Capability of Safety-Related Motor-Operated Valves,” U.S. Nuclear Regulatory Commission, September 18, 1996.

5. Generic Letter 89-10, “Safety-Related Motor-Operated Valve testing and Surveillance,” U.S. Nuclear Regulatory Commission, June 28, 1989.

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6. ANP-10264NP-A, Revision 0, “U.S. EPR Piping Analysis and Pipe Support Design Topical Report,” AREVA NP Inc., ~~September 2006~~November 2008.

7. ASME Boiler and Pressure Vessel Code, Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” The American Society of Mechanical Engineers, 2004 edition.

8. Generic Letter 90-09, “Alternative Requirements for Snubber Visual Inspection Intervals and Corrective Actions,” U.S. Nuclear Regulatory Commission, May 14, 1990.

the supporting structure is properly considered in the testing or analysis. The equipment mounting considered in the analysis or testing is identified in the SQDP.

If qualified by analysis, the critical support component stresses, and deflections if applicable, are determined and are compared to allowable levels per applicable codes and regulations (e.g., ASME Boiler and Pressure Vessel Code). If qualified by testing, the test response spectra must envelop the RRS at the mounting location of the support, over the frequency range of interest.

3.10.4 Test and Analysis Results and Experience Database

The results of seismic qualification testing and analysis, per the criteria in Section 3.10.1, Section 3.10.2, Section 3.10.3, are included in the corresponding SQDP (see Appendix 3D, Attachment F). A COL applicant that references the U.S. EPR design certification will create and maintain the SQDP file during the equipment selection and procurement phase. If the seismic and dynamic 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.

Complete and auditable plant-specific records and reports are available and are maintained at a central location for the life of the plant. The reports describe the qualification methods used for the equipment in sufficient detail to document compliance with the specified criteria. These records are updated and maintained current as equipment is replaced, modified, further tested, or requalified.

The equipment seismic qualification file contains a list of the systems' equipment and the equipment support structures. The equipment list identifies which equipment is NSSS supplied and which equipment is balance-of-plant supplied. The equipment qualification file includes qualification summary data sheets for each mechanical and electrical component of each system which summarizes the component's qualification. See Appendix 3D, Attachment F for a sample SQDP and Appendix 3D, Attachment A for a sample equipment qualification data package.

3.10.5 References

1. NUREG-1030, "Seismic Qualification of Equipment in Operating Nuclear Power Plants," U.S. Nuclear Regulatory Commission, February 1987.

2. European Utility Requirement for LWR Nuclear Power Plants, Volume 3, EPR Subset, December 1999.

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3. ANP-10264NPA, Revision 0, "U.S. EPR Piping Analysis and Pipe Support Design," AREVA NP Inc., ~~September 2006~~November 2008.

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

3.12.1 Introduction

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This section addresses the design of the piping systems and piping supports used in Seismic Category I, Seismic Category II, and non-safety-related systems. The information in this section is primarily supported by AREVA NP Topical Report ANP-10264NP-A~~10264(NP)~~ (Reference 1). This topical report focuses on Seismic Category I and Seismic Category II systems, but also addresses the interaction of non-seismic piping with Seismic Category I piping. Further supporting information is provided in Sections 3.7.2, 3.7.3, 3.9.1, 3.9.2, 3.9.3, 3.13, and 5.2.

3.12.2 Codes and Standards

Applicable codes and standards for piping and pipe supports are detailed in Section 2.0 and in Section 6.1 of Reference 1.

3.12.3 Piping Analysis Methods

3.12.3.1 Experimental Stress Analysis Methods

Experimental stress analysis methods are not used in lieu of analytical methods for Seismic Category I piping.

3.12.3.2 Modal Response Spectrum Method

The uniform support response spectrum method used in the analyses for piping systems is addressed in Section 4.2 of Reference 1.

3.12.3.3 Response Spectra Method (or Independent Support Motion Method)

The independent support motion response spectrum method is addressed in Section 4.2 of Reference 1.

3.12.3.4 Time History Method

Section 4.2.3 of Reference 1 addresses the time history methods used in the analyses of piping systems. Additional information is given in FSAR Section 3.7.2.

3.12.3.5 Inelastic Analysis Method

Inelastic analysis will not be used to qualify piping for the U.S. EPR design.

3.12.3.6 Small Bore Piping Method

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As noted in ~~AREVA NP letter NRC:07:028 dated July 13, 2007, "Response to a Request for Additional Information Regarding AREVA NP Topical Report, ANP-10264(NP)" (~~Reference ~~2~~1~~)~~, small bore piping is defined as ASME Class 1 piping that is 1 in NPS and smaller and Class 2, Class 3 and QG D piping that is 2 in NPS and smaller. This piping may be analyzed using response spectrum methods described in Section 4.2.2 of Reference 1 or the equivalent static method described in Section 4.2.3 of Reference 1.

3.12.3.7 Nonseismic/Seismic Interaction (II/I)

Section 4.4 of Reference 1 addresses design and analysis considerations for the interaction of non-seismic and seismic piping.

3.12.3.8 Seismic Category I Buried Piping

Section 3.10 of Reference 1 addresses the seismic criteria for buried piping systems.

3.12.4 Piping Modeling Techniques

3.12.4.1 Computer Codes

Section 5.1 of Reference 1 addresses the computer codes used in the analysis of safety-related piping systems (i.e., BWSPAN and SUPERPIPE). Further information on these computer codes is provided in Reference 2.

3.12.4.2 Dynamic Piping Model

Section 5.2 of Reference 1 addresses the dynamic piping modeling techniques. 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 ten percent of the mass of the adjacent pipe span.

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3.12.4.3 Piping Benchmark Program

As indicated in Section 5.3 of topical report ANP-10264NP-A~~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 listed 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.

3.12.4.4 Decoupling Criteria

Section 5.4.2 of Reference 1 addresses piping decoupling criteria.

3.12.5.18 Intersystem Loss-of-Coolant Accident

Section 3.9 of Reference 1 addresses intersystem LOCA. Additional information is provided in FSAR Section 19.2.

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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. 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. Examples of alternative methods are:

- 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 F_{en} environmental penalty factors 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 ten year inservice inspection requirements) of the affected locations.

3.12.6 Piping Support Design Criteria

3.12.6.1 Applicable Codes

Section 2.0 and Section 6.1 of Reference 1 address the applicable codes, code cases, and standards for the U.S. EPR piping supports.

3.12.6.2 Jurisdictional Boundaries

Section 6.2 of Reference 1 addresses the jurisdictional boundaries for pipe supports.

3.12.6.3 Loads and Load Combinations

Section 3.12.5.3 addresses loads and load combinations for pipe supports.

3.12.6.4 Pipe Support Baseplate and Anchor Bolt Design

Section 6.4 of Reference 1 addresses the design of pipe support baseplates and anchor bolts.

3.12.6.5 Use of Energy Absorbers and Limit Stops

Section 6.5 of Reference 1 addresses energy absorbers for pipe supports and gapped rigid supports (limit stops).

3.12.6.6 Use of Snubbers

Section 6.6 of Reference 1 addresses the use of snubbers in the piping design.

3.12.6.7 Pipe Support Stiffnesses

Section 6.7 of Reference 1 addresses the consideration of pipe support stiffnesses in the piping analyses and also provides support deflection criteria.

3.12.6.8 Seismic Self-Weight Excitation

Section 6.8 of Reference 1 addresses the consideration of seismic excitation of pipe supports in the analyses of the supports.

3.12.6.9 Design of Supplementary Steel

Section 6.9 of Reference 1 addresses the design of supplemental steel used in piping supports.

3.12.6.10 Consideration of Friction Forces

Section 6.10 of Reference 1 addresses consideration of pipe-to-pipe support friction forces in the analyses of pipe supports.

3.12.6.11 Pipe Support Gaps and Clearances

Section 6.11 of Reference 1 addresses pipe support gaps and clearances used in the design of pipe supports.

3.12.6.12 Instrumentation Line Support Criteria

Section 6.12 of Reference 1 addresses instrumentation line support design criteria.

3.12.6.13 Pipe Deflection Limits

Section 6.13 of Reference 1 addresses the allowable deflections for standard pipe support components (e.g., snubbers, struts, spring hangars) that are used in the design of piping.

3.12.7 References

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1. ANP-10264NP-A~~10264(NP) Revision 0~~, "U.S. EPR Piping Analysis and Pipe Support Design," AREVA NP Inc., ~~September 2006~~November 2008.

2. Deleted. ~~Ronnie L. Gardner (AREVA NP Inc.) to Document Control Desk (NRC), AREVA NP letter NRC:07:028 dated July 13, 2007, "Response to a Request for Additional Information Regarding AREVA NP Topical Report, ANP-10264(NP), 'U.S. EPR Piping Analysis and Support Design,' (TAC No. MD3128)," NRC:07:028, July 13, 2007.~~

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3. EPRI Technical Report 1011955, "Management of Thermal Fatigue in Normal Stagnant Unisolable Reactor Coolant System Branch Lines (MRP 146)," EPRI Proprietary Licensed Material, Electric Power Research Institute, June 2005.

4. EPRI Technical Report 103581, "Thermal Stratification, Cycling, and Striping (TASCS)," EPRI Proprietary Licensed Material, Electric Power Research Institute, March 1994.

5. NUREG-1367, "Functional Capability Of Piping Systems," U.S. Nuclear Regulatory Commission, November 1, 1992.

Next File

3A Criteria for Distribution System Analysis and Support

This appendix provides the design criteria for the U.S. EPR distribution system analysis and supports. As noted in Section 3.7.3, this appendix describes criteria for design of supports for:

- Piping.
- Heating, ventilation, and air conditioning (HVAC) ducts.
- Cable trays.

3A.1 Piping and Supports

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Information on piping, instrumentation, and supports is provided in AREVA NP Topical Report ANP-10264NP-A, "U.S. EPR Piping Analysis and Pipe Support Design," (Reference 1).

3A.2 Heating, Ventilation, and Air Conditioning Ducts and Supports

HVAC ductwork and its associated support structures are designed to withstand the loadings and load combinations presented in Section 3A.2.2 and Section 3A.2.3, based on the Codes and Standards provided in Section 3A.2.1. A typical HVAC duct system includes structural components (e.g., sheet metal ducts, duct stiffeners, duct supports) and inline components (e.g., heaters and dampers).

Safety-related, Seismic Category I HVAC ductwork, supports, and restraints meet the stress allowables provided in paragraph SA-4220 of ASME AG-1 (Reference 2). Seismic Category II HVAC ductwork, supports, and restraints are analyzed to make sure that a failure would not adversely impact safety-related equipment or components. Seismic Category II requirements are satisfied by conservatively analyzing the Seismic Category II HVAC ductwork, supports, and restraints to the same criteria as Seismic Category I.

Non-Seismic HVAC ductwork meets Sheet Metal and Air Conditioning Contractors National Association (SMACNA) standards (Reference 5). Non-Seismic HVAC ductwork support and restraint systems meet the analysis requirements of the American Institute of Steel Construction (AISC) Manual (Reference 3).

3A.2.1 Codes and Standards

HVAC ductwork, ductwork supports, and ductwork restraints conform to the following codes and standards:

- ASME AG-1-2003, Code on Nuclear Air and Gas Treatment, with 2004 Addenda (Reference 2).

3A.3.6 Seismic Analysis

The methods for seismic analysis are the same as described in Section 3A.2.4.4.

3A.4 References

03.06.02-5

1. ANP-10264NP-A~~10264(NP)~~, “U.S. EPR Piping Analysis and Support Design,” AREVA NP Inc., ~~September 2006~~November 2008.

2. ASME AG-1-2003, “Code on Nuclear Air and Gas Treatment,” The American Society of Mechanical Engineers, 2003 with 2004 Addenda.

3. AISC “Manual of Steel Construction,” Ninth Edition, American Institute of Steel Construction, April 2002.

4. AISI, “North American Specification for the Design of Cold-Formed Steel Structural Members,” American Iron and Steel Institute, 2001 Edition with 2003 Errata.

5. SMACNA, “HVAC Duct Construction Standards, Metal and Flexible,” Sheet Metal and Conditioning Contractors National Association, Third Edition, 2005.

6. AWS D1.1/D1.1M: 2004, “Structural Welding Code-Steel,” American Welding Society with errata through June 2005.

7. AWS D1.3-98, “Structural Welding Code – Sheet Steel,” American Welding Society, 1998.

8. ANSI/AISC-N690-1994, AISC “Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities,” American National Standards Institute/American Institute of Steel Construction, with Supplement 2, October 2004.

9. NUREG-0484, “Methodology for Combining Dynamic Responses,” U.S. Nuclear Regulatory Commission, May 1980.

  - RPV (Section 3C.2.1.1)
  - SGs and SG internals (Section 3C.2.1.2)
  - RCPs and the RCP internals (Section 3C.2.1.3)
  - Pressurizer (PZR) (Section 3C.2.1.4)
- RCS component supports (described in the component sections).
- RCS piping (Section 3C.2.1.5).
  - Main Coolant Loop (MCL)
  - Surge Line (SL)
- Reactor Building Internal Structures (RBIS) (Section 3C.2.1.6).

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Beam elements represent the RCS components, linear support elements (i.e., component support columns and SG upper lateral support struts), RPV vertical supports, PZR supports, piping, and RBIS sections. Springs represent the snubbers on the SGs, the snubbers on the RCPs, the SG lower lateral support bumpers, and the RPV horizontal supports. Excluding the surge line, piping attached to the MCL is not included in the model because it meets the decoupling criteria described in Section 5.4.2 of ANP-10264NP-A, U.S. EPR Piping Analysis and Pipe Support Design Topical Report (Reference 1). Loops 1, 2, and 4 of the model 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).

In addition to the mass of the model elements described above, the mass of the RPV closure head appurtenances (i.e., control rod drive mechanisms (CRDMs) and closure head equipment (CHE)) and the RPV internals are included in the RCS four loop structural model. The entrained fluid and thermal insulation mass for the components and piping is also accounted for in the model.

The RCS four loop structural model is developed using nominal dimensions.

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 safe shutdown earthquake (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. This non-linearity is accounted for in the dynamic analyses of the model as described in Section 3C.4.2.

Equation 3C-11) are determined and applied in the analysis in the same manner as described in Section 3C.4.2.1.

The integration time step used in these non-linear analyses is 0.0005 seconds. Such a small time step is required to ensure that the various gaps are properly accounted for in the solution. A time step study is performed where one SSE case is reanalyzed with the integration time step halved to 0.00025 seconds. The RCS response from this analysis is compared to that from the corresponding analysis with the original time step (0.0005 seconds). The maximum change in response is less than 6 percent, thereby validating the original integration time step (0.0005 seconds) as sufficient to allow convergence of the solution.

3C.4.3 Load Combinations

The load combinations used in the stress analyses of the RCS piping, components, component internals, and component supports are described in Section 3.9.3.

3C.5 Amplified Response Spectra Generation

Basemat acceleration time histories representing the SSE cases considered in the seismic analysis of the RCS four loop structural model are used to develop Amplified Response Spectra (ARS) at points of interest in the RCS. These include branch line nozzle locations on the RCS primary piping and the MFW line and MS line nozzles on the SGs. The ARS is generated for the various damping levels needed for seismic analysis of the attached piping (see Table 3.7.1-1). ARS is generated using the computer code RESPECT (see Section 3C.6). RESPECT generates ARS using input basemat time histories and the RCS structural properties as obtained from the BWSPAN output from the seismic analysis of the RCS four loop structural model.

3C.6 Description of Computer Programs

The following computer programs are used in the loading analyses of the RCS four loop structural model and the RPV isolated structural model:

- BWHIST: This code converts pressure time histories generated by CRAFT2 and COMPAR2 into force time histories by integrating the pressures over the component area on which the pressure acts. BWHIST also orients the resulting force time history for direct input into BWSPAN. Earlier versions of BWHIST were certified by comparing the output from the analysis of sample problems to the results obtained from hand calculations for the same sample problems. As additional options were added to the code, test cases were run to confirm that results did not change from the previous version. BWHIST is a certified computer code that is maintained in a controlled location (users can only access an executable file, not the source code).

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- BWSPAN: Information on this computer code is provided in Section 5.0 of Reference 1. ~~and AREVA NP letter NRC:07:028 dated July 13, 2007, "Response to a~~

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~~Request for Additional Information Regarding AREVA NP Topical Report, ANP-10264(NP), 'U.S. EPR Piping Analysis and Support Design,' (TAC No. MD3128)," (Reference 9).~~

- BWSPEC: This code tabulates displacements, pipe and structure loads, support loads, and spring loads for selected locations using output from a BWSPAN analysis. Tabulations can be made for static, response spectrum, and time history load cases. Earlier versions of BWSPEC were certified by comparing the results it generated using BWSPAN output from sample problems to the actual output from BWSPAN. As additional options were added to the code, test cases were run to confirm that results did not change from the previous version. BWSPEC is a personal computer based code that is verified, by comparing its output to the BWSPAN output from which it is reading, each time it is executed.

- CASS: This code performs structural analysis of general structures subjected to static and dynamic loading using finite element analysis techniques and is typically used to perform modal analysis of structures. CASS is a purchased code that runs on a personal computer that is verified, by comparing the results from analysis of a sample problem to the classical solution of the same problem, each time it is executed.

- COMPAR2: This code performs hydraulics analysis of fluid systems (generally containment cavities). The system is modeled as a series of control volumes and flow paths, so that the behavior of a pressure wave caused by a pipe break can be predicted. Pressure time histories can be obtained for any structure included in the model. COMPAR2 is the AREVA NP version of COMPARE-MOD1, which is described in NUREG-0609 as being applicable and conservative for use in Asymmetric Cavity Pressurization analyses. No difference exists between these two codes except that COMPAR2 provides an additional output file containing a tabulation of nodal pressures for subsequent input to BWHIST. COMPAR2 is certified, by comparing results obtained from analyses of test configurations to actual test data and to hand calculations. COMPAR2 is a certified computer code that is maintained in a controlled location (users can only access an executable file, not the source code).

- CRAFT2: This code performs hydraulics analysis of fluid systems including piping and components. The system is modeled as a series of control volumes and flow paths such that the behavior of a pressure wave caused by a pipe break can be predicted. Pressure time histories can be obtained at changes in area or changes in flow direction. The NRC has approved CRAFT2 for use in simulating the effect of pipe ruptures on the RCS (Reference 4). CRAFT2 is certified, by comparing results obtained from analyses of test configurations to actual test data and to hand calculations. As additional options were added to the code, test cases were run to confirm that results did not change from the previous version. CRAFT2 is a certified computer code that is maintained in a controlled location (users can only access an executable file, not the source code).

- EBDynamics: This code performs dynamic analysis of general structures and fluids subjected to dynamic loading and is typically used to find the time domain solution of coupled fluid-structure problems. EBDynamics is a personal computer based program that is verified, by comparing the results from analysis of a sample problem to the classical solution of the same problem, each time it is executed.

- P91232: This code calculates through-wall gradient temperatures and stresses given pipe or nozzle geometry and thermal characteristics (i.e., time dependant fluid temperature and film coefficients or flow rates). P91232 is a personal computer based code that is verified, by comparing results from analysis of sample problems to hand calculated results, each time it is executed.

- RESPECT: This code generates ARS given the frequency and mode characteristics of the system in question (from BWSPAN) and the acceleration time history applicable to the base of the structure. RESPECT is used to generate seismic ARS at the branch nozzle locations in a model of a piping system. Earlier versions of RESPECT were certified by comparing the output obtained from analysis of sample problems to results obtained from hand calculations for the same sample problems. As additional options were added to the code, test cases were run to confirm that results did not change from the previous version. RESPECT is a certified computer code that is maintained in a controlled location (users can only access an executable file, not the source code).

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 limitation of the program application, and the solutions to the test problems described above.

3C.7 References

03.06.02-5

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