

ArevaEPRDCPEm Resource

From: BRYAN Martin (EXT) [Martin.Bryan.ext@areva.com]
Sent: Thursday, March 11, 2010 3:02 PM
To: Tesfaye, Getachew
Cc: DELANO Karen V (AREVA NP INC); ROMINE Judy (AREVA NP INC); BENNETT Kathy A (OFR) (AREVA NP INC); HAMMOND Philip R (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 306, FSAR Ch. 3, Supplement 1
Attachments: RAI 306 Supplement 1 Response US EPR DC.pdf

Getachew,

AREVA NP Inc. (AREVA NP) provided a schedule for a technically correct and complete response to RAI No. 306 on December 4, 2009. The attached file, "RAI 306 Supplement 1 Response US EPR DC" provides technically correct and complete responses to 8 of the remaining 10 questions, as committed.

Appended to this file are affected pages of the U.S. EPR Final Safety Analysis Report in redline-strikeout format which support the response to RAI 306 Questions 03.08.01-42 and 3.12-18.

The following table indicates the respective pages in the response document, "RAI 306 Supplement 1 Response US EPR DC," that contain AREVA NP's response to the subject questions. Please note that AREVA NP requests an opportunity for interaction with the staff regarding environmentally-assisted fatigue as it relates to the response to question 03.12-18.

Question #	Start Page	End Page
RAI 306 — 03.03.01-4	2	2
RAI 306 — 03.08.01-39	3	8
RAI 306 — 03.08.01-40	9	11
RAI 306 — 03.08.01-41	12	13
RAI 306 — 03.08.01-42	14	14
RAI 306 — 03.08.01-43	15	16
RAI 306 — 03.12-18	17	18
RAI 306 — 03.12-19	19	19
RAI 306 — 03.12-20	20	20
RAI 306 — 03.12-21	21	21

The schedule for technically correct and complete responses to the remaining 2 questions has been changed due to administrative reasons and is provided below:

Question #	Response Date
RAI 306 — 03.12-19	May 12, 2010
RAI 306 — 03.12-20	May 12, 2010

Sincerely,

Martin (Marty) C. Bryan

Licensing Advisory Engineer
AREVA NP Inc.
Tel: (434) 832-3016
Martin.Bryan.ext@areva.com

From: Pederson Ronda M (AREVA NP INC)
Sent: Friday, December 04, 2009 4:08 PM
To: 'Tesfaye, Getachew'
Cc: BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC); HAMMOND Philip R (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 306, FSAR Ch. 3

Getachew,

Attached please find AREVA NP Inc.'s response to the subject request for additional information (RAI). The attached file, "RAI 306 Response US EPR DC.pdf," provides the schedule for technically correct and complete responses to these questions.

The following table indicates the respective pages in the response document, "RAI 306 Response US EPR DC.pdf," that contain AREVA NP's response to the subject questions.

Question #	Start Page	End Page
RAI 306 — 03.03.01-4	2	2
RAI 306 — 03.08.01-39	3	3
RAI 306 — 03.08.01-40	4	4
RAI 306 — 03.08.01-41	5	5
RAI 306 — 03.08.01-42	6	6
RAI 306 — 03.08.01-43	7	7
RAI 306 — 03.12-18	8	8
RAI 306 — 03.12-19	9	9
RAI 306 — 03.12-20	10	10
RAI 306 — 03.12-21	11	11

The schedule for technically correct and complete responses to these questions is provided below.

Question #	Response Date
RAI 306 — 03.03.01-4	March 12, 2010
RAI 306 — 03.08.01-39	March 12, 2010
RAI 306 — 03.08.01-40	March 12, 2010
RAI 306 — 03.08.01-41	March 12, 2010
RAI 306 — 03.08.01-42	March 12, 2010
RAI 306 — 03.08.01-43	March 12, 2010
RAI 306 — 03.12-18	March 12, 2010
RAI 306 — 03.12-19	March 12, 2010
RAI 306 — 03.12-20	March 12, 2010
RAI 306 — 03.12-21	March 12, 2010

Sincerely,

Ronda Pederson

ronda.pederson@areva.com

Licensing Manager, U.S. EPR Design Certification
AREVA NP Inc.

An AREVA and Siemens company

3315 Old Forest Road
Lynchburg, VA 24506-0935
Phone: 434-832-3694
Cell: 434-841-8788

From: Tesfaye, Getachew [mailto:Getachew.Tesfaye@nrc.gov]

Sent: Wednesday, November 04, 2009 12:14 PM

To: ZZ-DL-A-USEPR-DL

Cc: Patel, Jay; Xu, Jim; Hawkins, Kimberly; Hsu, Kaihwa; Dixon-Herrity, Jennifer; Miernicki, Michael; Colaccino, Joseph; ArevaEPRDCPEm Resource

Subject: U.S. EPR Design Certification Application RAI No. 306(3642,3787,3755), FSAR Ch. 3

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on October 9, 2009, and discussed with your staff on November 4, 2009. No changes were made to the draft RAI as a result of that discussion. The schedule we have established for review of your application assumes technically correct and complete responses within 30 days of receipt of RAIs. For any RAIs that cannot be answered within 30 days, it is expected that a date for receipt of this information will be provided to the staff within the 30 day period so that the staff can assess how this information will impact the published schedule.

Thanks,
Getachew Tesfaye
Sr. Project Manager
NRO/DNRL/NARP
(301) 415-3361

Hearing Identifier: AREVA_EPR_DC_RAIs
Email Number: 1226

Mail Envelope Properties (BC417D9255991046A37DD56CF597DB7105875A91)

Subject: Response to U.S. EPR Design Certification Application RAI No. 306, FSAR Ch. 3, Supplement 1
Sent Date: 3/11/2010 3:01:39 PM
Received Date: 3/11/2010 3:01:51 PM
From: BRYAN Martin (EXT)

Created By: Martin.Bryan.ext@areva.com

Recipients:

"DELANO Karen V (AREVA NP INC)" <Karen.Delano@areva.com>

Tracking Status: None

"ROMINE Judy (AREVA NP INC)" <Judy.Romine@areva.com>

Tracking Status: None

"BENNETT Kathy A (OFR) (AREVA NP INC)" <Kathy.Bennett@areva.com>

Tracking Status: None

"HAMMOND Philip R (AREVA NP INC)" <Philip.Hammond@areva.com>

Tracking Status: None

"Tesfaye, Getachew" <Getachew.Tesfaye@nrc.gov>

Tracking Status: None

Post Office: AUSLYNCMX02.adom.ad.corp

Files	Size	Date & Time
MESSAGE	4922	3/11/2010 3:01:51 PM
RAI 306 Supplement 1 Response US EPR DC.pdf		194715

Options

Priority: Standard

Return Notification: No

Reply Requested: No

Sensitivity: Normal

Expiration Date:

Recipients Received:

Response to

Request for Additional Information No. 306, Supplement 1

11/04/2009

U.S. EPR Standard Design Certification

AREVA NP Inc.

Docket No. 52-020

SRP Section: 03.03.01 - Wind Loading

SRP Section: 03.08.01 - Concrete Containment

**SRP Section: 03.12 - ASME Code Class 1, 2, and 3 Piping Systems and Piping
Components and Their Associated Supports**

Application Section: FSAR Ch 3

QUESTIONS for EPR Projects Branch (NARP)

QUESTIONS for Structural Engineering Branch 2 (ESBWR/ABWR Projects) (SEB2)

QUESTIONS for Engineering Mechanics Branch 1 (AP1000/EPR Projects) (EMB1)

Question 03.03.01-4:

The staff has determined that COL Information Item 3.3-1 in FSAR Tier 2, Table 1.8-2 does not distinguish between site parameters and site characteristics as defined in 10 CFR 52.1(a). Please revise COL information Item 3.3-1 in FSAR Tier 2, Table 1.8-2, accordingly, or as suggested below:

A COL applicant that references the U.S. EPR DC will determine site-specific wind and tornado ~~design parameters characteristics~~ and compare these to the standard plant criteria. If the site-specific wind and tornado ~~parameters characteristics~~ are not bounded ~~by the site parameters, postulated for the certified design~~, then the COL applicant will evaluate the design for site-specific wind and tornado events and demonstrate that these loadings will not adversely affect the ability of safety-related structures to perform their safety functions during or after such events.

The staff also requests the applicant to review and revise any additional COL Information Items in FSAR Tier 2, Table 1.8-2, and throughout the all necessary sections of the FSAR, that clearly do not use the terms "site parameters" and "site characteristics" in accordance with 10 CFR 52.1 (a), as necessary.

Response to Question 03.03.01-4:

The Response to RAI 274, Supplement 1, Question 02-1 will address this question.

FSAR Impact:

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

Question 03.08.01-39:

Follow-up to RAI Question No. 3.8.1-16

The RAI response provided information regarding a parametric study performed to address the issue of the variation of material properties and the use of best estimate values for material properties in the design of the Reactor Containment Building (RCB). The following information is needed to resolve this RAI:

1. Provide the range of values used in the parametric study and demonstrate that these range of values are appropriate by comparing them to the properties (or range of properties) used in the design of the RCB. This comparison of properties between the study and design values should consider the variation of properties corresponding to the range of temperatures for the containment under the different loading conditions.
2. Confirm whether the values in FSAR Tables 3.8-1 through 3.8-4 are best-estimate values used for the analysis and design of the RCB, because it appears that some of these values (e.g., modulus of elasticity for concrete) may be based on code specified values instead. As requested in the RAI, provide the technical basis for using the properties listed in the FSAR Tables 3.8-1 through 3.8-4 (i.e., identify the source for the values). Where a reference to an industry code, standard, guide, or textbook is not available, provide the technical basis for using the listed values. Also, explain why the best-estimate values are used for design purposes and not a conservative value which would account for potential uncertainties inherent in the parameters, as is done in design codes.
3. Provide the same information in Item 2 above for FSAR Sections 3.8.2 through 3.8.5.
4. Explain how the detrimental effects of radiation were considered for the concrete and steel structures in and within the primary and secondary shield walls.
5. The response to RAI Number 3.8.1-14 states that the axisymmetric model of the RCB was also used to study the effect of the variations in the temperature of the annulus relative to the 79F value used to date, and that the results of this study would be given in the response to this RAI. Since the RAI response only marginally mentions this issue, provide a complete discussion on the results of this study.

Response to Question 03.08.01-39:

1. Table 03.08.01-39-1 provides the parameters and range of values used in the material variation parametric study of the Reactor Containment Building (RCB).

Thermal and structural analyses are performed under accidental temperature and pressure transients using the design values for the RCB as part of the parametric study. These analyses establish reference design forces and moments that can be compared with the results from the adjusted parameter models. Parameter adjustments were made by comparing material property values at the average normal room temperature with material property values at the maximum design temperature under accident conditions.

Experimental data from Reference 1 indicates that the values of specific heat of concrete increase slightly with a rise in temperature. The value of 15 percent was selected based on Reference 1, Figure 7.

The modulus of elasticity for concrete experiences a reduction in value at elevated temperatures. Reference 1, Figure 2 shows the upper and lower bound E_c curves for test results. At 300°F, the upper bound curve indicates a modulus of 0.9 E_c while the lower bound curve indicates a modulus of 0.45 E_c . For this parametric study, the average of these bounding curves was considered.

Considering the effect of a rise in temperature from ambient to 300°F on the specific heat of steel, NUREG/CR-6900, Figure 4 indicates that a 10 percent increase is an appropriate variation.

2. Table 03.08.01-39-2 shows the values from U.S. EPR FSAR Tier 2, Tables 3.8-1 through 3.8-4 and the information source for that data. The values generally fall into one of three categories: code or standard specified values, design specifications, or engineering estimates that can be justified by experience or a technical basis. The effect of variation of select engineering estimated values has been quantified in the parametric study discussed in Item 1 of this question. The study concluded that this variation had an insignificant effect on the resulting forces and moments of the RCB wall.
3. Thermal properties for concrete and steel in U.S. EPR FSAR Tier 2, Sections 3.8.2 through 3.8.5 are consistent with the properties listed in U.S. EPR FSAR Tier 2, Tables 3.8-1 through 3.8-4. Material properties for concrete and steel in U.S. EPR FSAR Tier 2, Sections 3.8.2 through 3.8.5, including unit weight and Poisson's ratio, are consistent with the values listed in the subject tables. The specified nominal compressive strength (f'_c) varies as described in U.S. EPR FSAR Tier 2, Sections 3.8.3.6, 3.8.4.6, and 3.8.5.6. The modulus of elasticity for concrete is calculated based on the compressive strength using the formula from ACI 349-01, Section 8.5.1.
4. Primary and secondary shield wall thicknesses are determined by selecting the maximum thickness based on radiation shielding requirements described in ANSI/ANS-6.4-2006 or structural requirements contained in ACI 349-01 and ACI 349.1R-07. Concrete aggregates conforming to ASTM C637 will be used in radiation shielding applications where applicable. No material variation is expected for primary or secondary shield walls because industry operating experience has not indicated a loss of strength for reinforced concrete exposed to radiation. The governing civil/structural design codes and standards for structural steel and reinforced concrete design in nuclear applications do not contain design considerations that indicate a variation in material properties or allowables for structural materials exposed to radiation.
5. The axisymmetric model of the RCB was used to study the variation of annulus temperatures ranging from a minimum of 45°F to a maximum of 113°F. The results at four critical time points from this study are used in the design of the RCB wall. The four critical time points were selected by choosing time points where maximum forces and moments occurred for different sections of the RCB under accidental temperature and pressure distribution. In the parametric study, the results for the thermal analysis at these four critical time points indicate that the minimum annulus temperature of 45°F results in larger design forces and moments while the maximum annulus temperature of 113°F results in a reduction of the design forces and moments in the RCB wall. However, the combination of the variation in thermal properties, mechanical properties, and annulus temperature has an insignificant effect on the resulting forces and moments of the RCB wall.

References for Question 03.08.01-39:

1. M.K. Kassir, K.K. Bandyopadhyay, and M. Reich, "Thermal Degradation of Concrete in the Temperature Range From Ambient to 315°C (600°F)," June 1993.

FSAR Impact:

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

Table 03.08.01-39-1—Range of Values Used in the RCB Material Variation Parametric Study

	Parameter	Minimum Study Value	Maximum Study Value	Design Value
Concrete	Specific Heat	1000 J/kg°C	1150 J/kg°C	1000 J/kg°C
	Modulus of Elasticity	3.22x10 ⁶ psi	4.77x10 ⁶ psi	4.77x10 ⁶ psi
Steel	Specific Heat	434 J/kg°C	477.4 J/kg°C	434 J/kg°C

**Table 03.08.01-39-2—RCB Design and Analysis Values
(2 Sheets)**

	Property	Value	Source/Justification
Concrete	Thermal Conductivity (kW/m°C)	0.0023	Thermal Property ¹
	Specific Heat (J/kg°C)	1000	Thermal Property ¹
	Modulus of Elasticity (ksi)	4769	ACI 349-01 Sec. 8.5.1
	Poisson's Ratio	0.17	Poisson's Ratio for concrete usually falls in the range of 0.15 to 0.20. The selected value was chosen to be near the middle of this typical range. Section 3-5, page 74 of "Reinforced Concrete Mechanics and Design," Fourth Edition, by James G. MacGregor and James K. Wight
	Nominal Strength f_c (ksi)	7	The nominal compressive strength of concrete is a design specification which will be verified by testing.
	Unit Weight (lb/ft ³)	150	PCI Design Handbook Precast and Prestressed Concrete, 5 th Edition, Section 2.2.5
	Film Coefficient (BTU/hr*ft ² *°F)	1.41	Thermal Property ¹
	Thermal Diffusivity (ft ² /hr)	0.037	Thermal Property ¹
Steel	Thermal Conductivity (kW/m°C)	0.041	Thermal Property ¹
	Specific Heat (J/kg°C)	434	Thermal Property ¹
Post-Tensioning Cable	Modulus of Elasticity (ksi)	28000	"Estimating Prestress Losses", by Paul Zia, H. Kent Preston, Norman L. Scott, and Edwin B. Workman, published in Concrete International magazine
	Poisson's Ratio	0.30	AISC Manual of Steel Construction: Allowable Stress Design, 9 th Edition, Page 6-37
	Nominal Strength F_{pu} (ksi)	270	ASTM A416
	Unit Weight (lb/ft ³)	490	AISC Code of Standard Practice for Steel Buildings and Bridges, Adopted Effective September 1, 1986, Section 9.2.1

**Table 03.08.01-39-2—RCB Design and Analysis Values
(2 Sheets)**

		Property	Value	Source/Justification
Tendon Friction Losses	Hoop	K (per foot)	0.00050	The values for the wobble coefficient 'K' and the curvature coefficient ' μ_p ' are determined based on experimental testing. A range of values is given in Table R18.6.2 of ACI 318-05. The values selected for analysis, as shown in Table 3.8-3, are chosen based on design experience with a similar post-tensioned system in European applications of the EPR. The values of 'K' for hoop and dome tendons match the lower end of the range given in Table R18.6.2. The value for vertical tendons is taken as half of the dome and hoop tendon value as the vertical tendons are stressed from one end only. The value of the curvature coefficient for all types of tendons is close to the lower end of the range given in Table R18.6.2 for grouted 7-wire strand tendons. Based on these considerations and the experience gained from European applications of the EPR, the selected coefficients are considered to be conservative.
		μ (per radian)	0.18	
	Vertical	K (per foot)	0.00025	
		μ (per radian)	0.16	
	Dome	K (per foot)	0.00050	
		μ (per radian)	0.16	
Reinforcing Bar	Modulus of Elasticity (ksi)		29000	ACI 349-01 – Sec. 8.5.2
	Poisson's Ratio		0.30	AISC Manual of Steel Construction: Allowable Stress Design, 9 th Edition, Page 6-37
	Nominal Strength F_y (ksi)		60	The nominal strength of reinforcing steel is a design specification. Conventional reinforcement steel will conform to ASTM A615.
	Unit Weight (lb/ft ³)		490	AISC Code of Standard Practice for Steel Buildings and Bridges, Adopted Effective September 1, 1986, Section 9.2.1

Notes:

1. Thermal Properties for concrete are dependent on concrete mix design. As the mix design will be determined based upon field testing, the thermal properties cannot be definitively determined during Design Certification. The values selected for analysis for concrete and steel are best estimate values based on experience with European applications of the EPR design.

Question 03.08.01-40:

Follow-up to RAI Question No. 3.8.3-8

The first part of the response to RAI Number 3.8.3-8 lists the abnormal loads generated by a postulated high-energy pipe break accident, as described in the FSAR. In addition, the response indicates that the nuclear steam supply system is the only high-energy line considered in developing abnormal loads R_a and R_r , and that pipe break loads R_{rj} and R_{rm} were not considered in the global analysis of the RBIS. Finally, the response also mentions the methodology for evaluating thermal stresses per ACI 349 Appendix A.

The staff considers that the RAI response does not address the question raised in the RAI. To resolve the first part of this RAI, provide the method and basis for performing the localized analysis for each type of abnormal load, including the potential effects of concrete cracking due to accident thermal loads and redistribution of member forces due to cracking of concrete if significant. In other words, provide further elaboration on: (1) why certain abnormal loads are considered "localized" loads and "not included in the formation of load combinations for the global system"; simply stating that the localized loads "are not considered significant" is not sufficient; (2) how the member forces and stresses due to these localized loads were determined, e.g. describe the more refined finite element sub-models mentioned in FSAR Section 3.8.3.4.2 (first paragraph) and elaborate on the analysis approach; and (3) how the potential effects of concrete cracking due to accident thermal loads were considered in the finite element models or sub-models.

The staff further notes that the response states that pipe break reaction load " R_{rr} is the only component of R_r considered in the global analysis of the NIS." This statement appears to contradict FSAR Section 3.8.3.4.1, which identifies pipe break reaction, jet impingement and missile loads (R_{rr} , R_{rj} , and R_{rm}) as localized abnormal loads that are not included in the overall analysis. Explain this inconsistency.

In response to the second part of RAI Number 3.8.3-8, the RAI response summarizes the design conditions and number of load combinations corresponding to normal plus abnormal loads, as well as to normal plus extreme environmental plus abnormal loads.

The staff considers that the response does not address the question raised in the RAI. To resolve the second part of this RAI, describe how the results (e.g., member forces and stresses from the differing global and localized finite element models and sub-models) of the localized analyses are combined with the results of the global structural analyses for other loads, since location of these element forces from the two models do not necessarily match.

Response to Question 03.08.01-40:

1. Critical section design results in U.S. EPR FSAR Tier 2, Appendix 3E utilize the results of the global Nuclear Island (NI) static finite element model (FEM) and results of various localized analyses for loads not included in the global NI static FEM (including the potential effects of concrete cracking due to accident thermal loads and redistribution of member forces due to cracking of concrete where significant). The newly developed and implemented critical section selection methodology (as discussed in the Response to RAI 155, Supplement 7, Question 03.08.01-20) will result in a revision of U.S. FSAR Tier 2, Appendix 3E. As stated in the Response to RAI 155, Supplement 7, Question 03.08.01-20,

the revised U.S. EPR FSAR Tier 2, Appendix 3E with results of critical section design and with descriptions of applicable loadings, analysis, and modeling techniques (both global and localized), and design methods will be provided in the Response to RAI 155, Supplement 9, Question 03.08.04-6.

The Response to RAI 155, Supplement 9, Question 03.08.01-24 maintains clarity and detail for each critical section by introducing a method to standardize U.S. EPR FSAR Tier 2, Appendix 3E input. The Response to Question 03.08.01-24 indicates that future additions or changes to U.S. EPR FSAR Tier 2, Appendix 3E will include the detail and discussion necessary to further clarify the critical section design results, associated analyses (both global & localized), and design methodologies.

Following is a response to the three specific sub-parts of this question:

A. There are two categories of loads which were not included in the global model:

1) Loads which were not available at the time the global model was ran.

Some loads are not included in the formation of load combinations because these loads were not available at the initiation of the global model. These loads were addressed locally on a case-by-case basis and incorporate the loads into the final global NI static FEM results (by hand calculations or with sub-models). Examples include relief valve loads and pipe rupture loads.

2) Loads which would result in excessive load combination permutations.

Some loads are not included in the formation of load combinations for the global model because the inclusion of these loads would result in an unmanageable number of combination permutations. In these cases, it is appropriate to address the loads locally on a case-by-case basis and incorporate the loads into the final global model results (by hand calculations or with sub-models). Examples include compartment flood loads, containment wall pressure variant loads, and sub-compartment pressure loads.

B. The Response to RAI 155, Supplement 7, Question 03.08.01-20 includes a list of selected critical sections. Detailed design results for these critical sections will be provided with the Response to RAI 155, Supplement 9, Question 03.08.04-6. The Response to RAI 155, Supplement 9, Question 03.08.04-6 will address the analysis methodology for the determination of member forces and moments due to localized loads or loads which were not included in the global NI static FEM (i.e., hand calculations and sub-modeling techniques).

C. The Response to RAI 155, Supplement 7, Question 03.08.01-20 includes a list of selected critical sections. Detailed design results for these critical sections will be provided with the Response to RAI 155, Supplement 9, Question 03.08.04-6. The Response to RAI 155, Supplement 9, Question 03.08.04-6 will address the potential effects of concrete cracking due to the accident thermal loads

2. The statement in the Response to RAI 155, Supplement 1, Question 03.08.03-8 that the pipe break reaction load, R_{tr} , is included in the global analysis of the NIS is inaccurate. U.S. EPR FSAR Tier 2, Section 3.8.3.4.1 correctly defines R_{tr} as a localized abnormal load that is not included in the overall global NI static analysis.

3. As discussed in Part 1 of this response, the Response to RAI 155, Supplement 7, Question 03.08.01-20 includes a list of critical sections. Detailed design results for these critical sections will be provided with the Response to RAI 155, Supplement 9, Question 03.08.04-6, which will address the inclusion of localized analysis results with the results of the overall structural analysis.

FSAR Impact:

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

Question 03.08.01-41:

Follow-up to RAI Question No. 3.8.3-11

The response to item (1) of this RAI explains that the jurisdictional boundary between the polar crane assembly and the RCB is the location at which the crane runway system (girders) attaches to the crane runway support brackets. The crane support brackets are included in the design of the RCB. The RAI response provides a detail showing the jurisdictional boundary. The description of the jurisdictional boundary between the polar crane assembly and the RCB, which is discussed in the RAI response, needs to be included in the appropriate locations of the FSAR Section 3.8.

The response to item (3) of this RAI indicates that the crane girder and intervening structural steel members will be designed in accordance with the requirements for design and materials specified in AISC N690. Since the RAI response indicates that the intervening structural steel members (i.e., crane support brackets) are within the jurisdictional boundary of the RCB, explain why AISC N690 is utilized rather than the applicable ASME Code for containment. It should be noted that ASME Code, Section III, Division 2, Subsection CC, indicates that the design of steel members not backed by concrete shall meet the requirements of NE-3000 and Subsection NCA. Also, Article NE-1000 of ASME Code, Section III, Division 1, explains the jurisdictional boundary for the containment and any attachments to the containment, and provides a figure with typical examples. The jurisdictional boundaries identified in the RAI response do not appear to be consistent with the jurisdictional boundaries given in the ASME Code. Therefore, explain the basis for the ASME Code jurisdictional boundaries described in the RAI response with respect to the crane support brackets.

Response to Question 03.08.01-41:

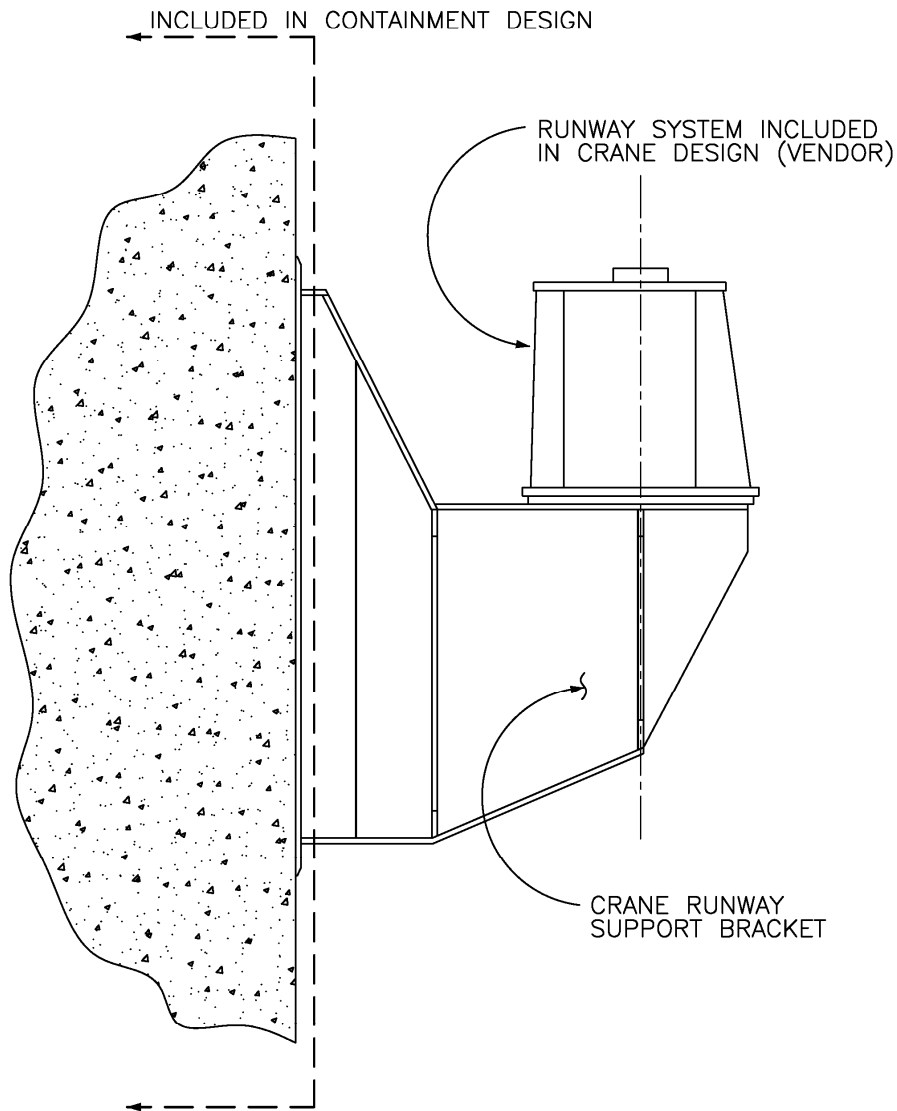
The design of the polar crane support brackets is not included in the jurisdictional ASME containment design boundary. Figure 03.08.03-11-1 of the Response to RAI 155, Question 03.08.03-11, has been revised for clarity and is included with this response as Figure 03.08.01-41-1.

The use of AISC N690 is appropriate in the design of the polar crane support brackets. The U.S. EPR FSAR description of the ASME containment design boundary excludes the design of the crane support brackets, and no changes are required.

FSAR Impact:

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

Figure 03.08.01-41-1—Boundary between Containment Structure and Crane Assembly



SECTION

(BOUNDARY BETWEEN CONTAINMENT STRUCTURE
AND CRANE ASSEMBLY)

FIGURE 03.08.03-11-1

(Crane Runway Support Bracket)

Question 03.08.01-42:

Follow-up to RAI Question No. 3.8.3-12

The response to item (1) of this RAI states that steel materials ASTM A333, A537 and A633 are listed as acceptable in ASME NOG-1-2004, ASME NUM-1-2004, and by extension, in NUREG-0554, when used in the construction of the polar crane and its support system. Nevertheless, the staff notes that: (1) the response to RAI 3.8.3-11 indicates that the crane runway support brackets are within the jurisdictional boundary of the RCB and (2) the response to RAI 3.8.3-11 also indicates that the crane and crane runway system are vendor supplied items that lie outside the jurisdictional boundary of the RCB. In light of this information, the applicant should confirm whether the polar crane assembly (minus the support brackets) is considered separately as a vendor supplied and qualified piece of equipment (i.e., not Seismic Category I structure), in which case the information related to the crane assembly should be removed from Table 3.8-8. In addition, if the resolution of RAI 3.8-11 determines that the crane brackets are within the jurisdictional boundary of the containment, as currently described in the RAI, then confirm that the listed materials are in conformance with the applicable ASME Code for containment.

The response to item (2) of this RAI states that material specifications, procurement and supplemental requirements for structural steel materials will be developed later in the design process. However, the structural design calculations described in the FSAR are based on allowable stresses that depend on specific material specifications, grades, and associated/supplemental requirements. Consequently, these specific material designations should be reflected in the FSAR (Table 3.8-8) at this time. To resolve item (2) of this RAI, the applicant is again requested to provide the materials specifications, along with procurement and supplemental requirements, for the actual steel structural materials to be used.

Response to Question 03.08.01-42:

The Response to Question 03.08.01-41 clarifies the jurisdictional boundary of the crane runway support brackets. The polar crane and the crane runway system (minus the support brackets) are vendor supplied items and are qualified separately. U.S. EPR FSAR Tier 2, Table 3.8-8 will be revised to reflect the use of the materials ASTM A333, A537 and A633 specific to the polar crane and vendor supplied crane runway system.

The design of the structural steel members are based on the conservative use of the minimum allowable material stress values provided in U.S. EPR FSAR Tier 2, Table 3.8-8. Where ranges of the material tensile and yield stresses are provided, the design specifies a particular minimum value to be used in the fabrication of the component. Actual materials used in fabrication are reported via certified material test reports and certificates of conformance to confirm the use of the material exceeding the minimum specified stress allowable value.

FSAR Impact:

U.S. EPR FSAR Tier 2, Table 3.8-8 will be revised as shown on the enclosed markup.

Question 03.08.01-43:

Follow-up to RAI Question No. 3.8.5-17

The response to RAI Number 3.8.5-17 provided additional information about the reinforcement in the EPGB and ESWB foundations. To complete the evaluation of this RAI response, provide the following information:

1. The U.S. EPR FSAR should be changed to include Figures 03.08.05-17-1 through 03.08.05-17-4 provided with the RAI response since these figures provide more complete information on the reinforcement design for the EPGB and ESWB foundations than the current figures in the FSAR.
2. Explain the following statement in the response: "The vertical (shear) reinforcement is not required for the revised Essential Service Water Building (ESWB) foundation configuration." Figures 3.8.5-17-3 and 4 are not consistent with this statement since both figures show shear reinforcement in the ESWB foundation.
3. Explain what additional information regarding the ESWB requested by this RAI will be addressed in the response to RAI Batch 130 Question 03.07.02-27, which will be provided as committed by the AREVA NP response to RAI 130. Clarify if this response has been submitted to the NRC.
4. As requested in the original RAI, provide information that reconciles the difference in the reinforcement for the NI foundation specified in FSAR Table 3E.1-37 and shown in FSAR Figure 3E.1-75.

Response to Question 03.08.01-43:

1. Figures 03.08.05-17-2 through 03.08.05-17-4 of the Response to RAI 155, Question 03.08.05-17 do not reflect the current design configuration of the EPGB and the ESWB. The current configuration of the EPGB and the ESWB is described in the Response to RAI 130, Supplement 2, Question 03.07.02-27, which was submitted to the NRC on April 17, 2009. The Response to RAI 155, Supplement 7, Question 03.08.01-20 states that the U.S. EPR FSAR Tier 2, Appendix 3E content, tables, and figures will be revised with the Response to RAI 155, Supplement 9, Question 03.08.04-6 to include updated design information and reinforcement configurations resulting from the newly implemented critical section selection methodology.
2. Figures 03.08.05-17-2 through 03.08.05-17-4 of the Response to RAI 155, Question 03.08.05-17 do not reflect the current design configuration of the ESWB. The current configuration is described in the Response to RAI 130, Supplement 2, Question 03.07.02-27. Inconsistencies such as this item will be clarified with the Response to RAI 155, Supplement 9, Question 03.08.04-6.
3. Additional information regarding the ESWB basemat slab extension and compatibility with soil-structure interaction (SSI) analysis is addressed in the Response to RAI 130, Supplement 2, Question 03.07.02-27 which was sent to the NRC on April 17, 2009.
4. The reinforcement for the Nuclear Island (NI) foundation specified in U.S. EPR FSAR Tier 2, Table 3E.1-37 does not reflect the current analysis results and design. Because these design results are subject to change (see Parts 1 and 2 of this response), Table 3E.1-37

and Figure 3E.1-75 will be revised in the Response to RAI 155, Supplement 9, Question 03.08.04-6.

FSAR Impact:

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

Question 03.12-18:

In FSAR Section 3.12.5.19, AREVA stated that alternative methods for addressing environmental fatigue will be applied and presented examples of alternative methods as follows:

- Redefinition of the normal and upset transients and number of cycles
- 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
- Augmented inspection

The staff noted that redefinition of the normal and upset transient affecting the location in question to reduce the severity of the transients or to reduce the number of cycles associated with the transient requires license amendment. The redefinition of the in-air design fatigue curves and/or F_{en} penalty factors also requires license amendment. The staff asks the applicant to clarify that the applicant will submit license amendment for NRC review and approval for taking these two alternative methods.

The staff also noted that fatigue monitoring and augmented inspections are for operating plants. The staff does not agree that design requirement for fatigue and cumulative fatigue usage factors for piping and components can be changed. The staff requests the applicant to provide other alternatives or to follow the staff approved methods.

Response to Question 03.12-18:

In RAI 179, Question 03.09.01-1, the NRC requested that a description of design transients, including “heat-up and cool-down rate limits,” be added to the U.S. EPR FSAR. The Response to RAI 179, Question 03.09.01-1 revised U.S. EPR FSAR Tier 2, Section 3.9.1.1.1 to provide the requested information. The transient descriptions in U.S. EPR FSAR Tier 2, Section 3.9.1.1.1 will be revised to clarify that the transient temperature rates are upper bound limits. It is AREVA’s understanding that modification of the transient temperature rates should not require regulatory approval if the modified values do not exceed the limits specified in U.S. EPR FSAR Tier 2, Section 3.9.1.1.1.

RG 1.207, Section D (Implementation) states that a licensee can propose acceptable alternate methods for complying with specified portions of NRC regulations. If alternate methods using data obtained from testing of samples representative of U.S. EPR materials, configurations, and environment are used for redefinition of the in-air fatigue curves or calculating F_{en} penalty factors, these methods would be proposed to the NRC consistent with the guidance in RG 1.207 Section D. Other alternate analyses methods, such as those endorsed by ASME (in the form of code cases), may also be proposed to the NRC consistent with the guidance in RG 1.207 Section D.

The regulatory basis is unclear for the position stated in the question that fatigue monitoring and augmented inspections are for operating plants exclusively. NUREG/CR-6909 acknowledges that the ASME Code fatigue design procedures are “quite conservative” and that the “ASME Code permits new and improved approaches to fatigue evaluations (e.g., finite-element analysis, fatigue monitoring, and improved K_e factors) that can significantly reduce the

conservatism in the current fatigue evaluation procedures.” Furthermore, the design transients are conservative compared to the actual plant operating transients, and fatigue monitoring is a generally accepted method to evaluate the actual fatigue usage of components.

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 3.9.1.1.1 will be revised as described in the response and indicated on the enclosed markup.

Question 03.12-19:

Follow-up to RAI Question No. 03.12-17

In response to Question 03.12-17, AREVA indicated that heatup/cooldown procedures are plant-specific. In order to use the first US EPR initial plant operation to verify the design transients for the surge line are representative, AREVA has to assure that all U.S. EPR plants will use the same heatup/cooldown methods. The staff asks AREVA to address this item and explain why only first plant surge line transients are monitored without standard heatup/cooldown procedures.

Response to Question 03.12-19:

A response to this question will be provided by May 12, 2010.

Question 03.12-20:

In FSAR Section 3.12.5.9, AREVA stated that the EPRI generic methodology indicated that thermal stratification will occur in RHR/SIS/EBS injection, RHR/SIS suction piping. AREVA also stated that specific measurements taken at AREVA NP designed foreign plants on piping configurations that are representative of U.S.EPR piping system indicate small range and shorter vortex penetration than the EPRI methodology. Thus, testing information shows that thermal stratification does not occur in any horizontal segment of the aforementioned (RHR/SIS/EBS injection, RHR/SIS suction) RCS attached piping.

The staff noted that the cyclic thermal stratification occurring within such RCS attached piping is affected by the line orientation and geometry. The staff requests AREVA to provide detailed line geometry information (e.g. L/Di, DH/H/LH configuration) for each of the above mentioned lines in order to determine that the thermal stratification does not occur in any horizontal segment of the RCS attached piping.

If AREVA uses its specific test information to justify that thermal stratification does not occur in any RCS attached piping for EPR design. The staff requests AREVA to provide detailed test information for review and approval.

AREVA stated that the U.S. EPR design incorporates lessons learned from operating experience in that the injection line (SIS/RHRS) continually rises in elevation from the check valve; therefore, it is not susceptible to valve leakage-induced cyclic thermal stratification. The staff requests AREVA to explain why the piping is not susceptible to valve leakage-induced cyclic thermal stratification with continual rises in elevation from the check valve and rise to what kind of level/elevation will not be susceptible to cyclic thermal stratification.

Response to Question 03.12-20:

A response to this question will be provided by May 12, 2010.

Question 03.12-21:

In FSAR 3.12.5.9, AREVA stated that a COL applicant that references the U.S. EPR design certification will monitor the RCS attached piping during the first U.S. EPR initial plant operation to verify that operating conditions have been considered in the design. However, the up-horizontal and horizontal (UH/H) configuration thermal cycling model is based on valve in-leakage establishing a cold stratified layer in horizontal pipe run which interacts with branch line swirl resulting in cyclic thermal loads applied to a region of the horizontal pipe segment.

The staff does not expect valve in-leakage during initial plant operation. The staff requests AREVA to explain how to simulate valve in-leakage during the first initial plant operation to verify that operating conditions have been considered in the design.

Response to Question 03.12-21:

AREVA NP agrees that valve in-leakage during initial plant operation is not expected. Simulation of valve in-leakage in the reactor coolant system (RCS) attached piping is not needed for the following reasons:

- The RCS attached branch lines that are susceptible to thermal stratification are downward horizontal and do not depend on valve in-leakage to create cyclic thermal loads per EPRI guidelines.
- The up-horizontal and horizontal (UH/H) configurations are part of the chemical and volume control system (CVCS) let down and suction lines and normal spray lines (Loop 2 and 3). For these specific lines, EPRI screening methodology has shown that even though valve in-leakage may occur, these lines do not experience thermal stratification and fatigue-related phenomena.

FSAR Impact:

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

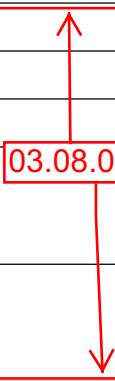
U.S. EPR Final Safety Analysis Report Markups

Table 3.8-8—Materials for Structural Steel Shapes and Plates
Sheet 1 of 2

ASTM Designation	Minimum F_y	Minimum F_u
A36	36 ksi	58 to 80 ksi
A53 (Type E or S) (Gr. B)	35 ksi	60 ksi
A106		
Grade A	30 ksi	48 ksi
Grade B	35 ksi	60 ksi
Grade C	40 ksi	70 ksi
A167	27 to 39 ksi	73 to 94 ksi
A240		
Austenitic	25 to 70 ksi	70 to 125 ksi
Duplex	58 to 80 ksi	87 to 116 ksi
Ferritic or Martensitic	25 to 90 ksi	55 to 115 ksi
A242	42 to 50 ksi	63 to 70 ksi
A276		
Austenitic	25 to 125 ksi	70 to 145 ksi
Austenitic-ferritic	65 to 105 ksi	90 to 125 ksi
Ferritic	30 to 60 ksi	60 to 75 ksi
Martensitic	30 to 100 ksi	60 to 125 ksi
A312	25 to 62 ksi	70 to 115 ksi
A333		
Grades 3, 7	35 ksi	65 ksi
Grades 4, 6	35 ksi	60 ksi
A441	40 to 50 ksi	60 to 70 ksi
A479		
Austenitic	25 to 125 ksi	70 to 145 ksi
Austenitic-ferritic	65 to 85 ksi	90 to 118 ksi
Ferritic	25 to 55 ksi	60 to 70 ksi
A276 (Martensitic)	40 to 100 ksi	70 to 130 ksi
A500 (round)		
Grade A	33 ksi	45 ksi
Grade B	42 ksi	58 ksi
Grade C	46 ksi	62 ksi
Grade D	36 ksi	58 ksi
A500 (square & rectangular)		
Grade A	39 ksi	45 ksi
Grade B	46 ksi	58 ksi
Grade C	50 ksi	62 ksi
Grade D	36 ksi	58 ksi
A501	36 ksi	58 ksi
A514	90 to 100 ksi	100 to 130 ksi

Table 3.8-8—Materials for Structural Steel Shapes and Plates
Sheet 2 of 2

ASTM Designation	Minimum F_y	Minimum F_u
A515	32 to 38 ksi	60 to 90 ksi
A516	30 to 38 ksi	55 to 90 ksi
A537	40 to 60 ksi	65 to 100 ksi
A570	30 to 55 ksi	49 to 70 ksi
A572	42 to 65 ksi	60 to 80 ksi
A588	42 to 50 ksi	63 to 70 ksi
A607	45 to 70 ksi	60 to 85 ksi
Class I	45 to 70 ksi	55 to 80 ksi
Class II	45 to 70 ksi	55 to 80 ksi
A618	46 to 50 ksi	67 to 70 ksi
Grade Ia, Ib & II	50 ksi	65 ksi
Grade III	50 ksi	65 ksi
A633 (Grades A, C, & D)	42 to 60 ksi	63 to 100 ksi
A709	36 to 50 ksi	58 to 80 ksi
A913	50 to 70 ksi	65 to 90 ksi
A992	50 to 65 ksi	65 ksi

03.08.01-42


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. The term “stretch-out operation” refers to the maximum fuel cycle length based on an assumed T_{AVG} reduction of 10°F, followed by a coastdown to 70 percent full power. This corresponds to approximately 40 effective full-power days (EFPD) at the end of the fuel cycle.

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.

3.9.1.1.1 Normal Conditions

The following RCS transients are considered normal conditions:

3.9.1.1.1.1 3.9.1.1.1.1 Plant Heatup and Cooldown

Heatup transients are analyzed for initial conditions following the reactor pressure vessel (RPV) head being removed (typically due to a refueling outage) or for a cold shutdown (CSD) condition where the RPV head has not been removed. Similarly, cooldown transients are also analyzed with these conditions as final conditions.

The following Normal transients represent U.S. EPR heatup and cooldown operations:

Transient 1 - Plant Startup from CSD to Full Load with the RPV Open at CSD

Transient 1A - Complete Plant Startup from CSD to Full Load, with RPV Open at CSD – RCS Heated up then Reactor Coolant Pump (RCP) Start

This transient is based on normal operations during a complete plant startup from cold shutdown following a refueling outage to full load. During the startup that follows a short outage, the RPV is heated from ambient to 122°F by residual heat before starting the RCPs. The RCS is heated to hot shutdown (HSD) at an upper rate limit of 72°F per hour followed by an increase to 100 percent full power (FP).

↑
03.12-18

The temperature gradient is maximized during the heatup phase since the maximum gradient should be lower than 72°F/hr based on the expected residual heat from the core and RCP heat.

Transient 1B - Complete Plant Startup from CSD to Full Load, with RPV Open at CSD – RCS not heated, cold RCP start

This transient is based on normal operations during a complete plant startup from cold shutdown following a refueling outage to full load. During a startup that follows a longer outage (or during commissioning) with minimum residual heat, the RCPs are started at an RCS temperature of 60°F. The RCS is heated to HSD at an upper rate limit of 72°F per hour followed by an increase to 100 percent FP.

03.12-18 →

↑
03.12-18

The temperature gradient is maximized during the heatup phase since the maximum gradient should be lower than 72°F/hr based on the expected residual heat from the core and reactor coolant pump heat.

Transient 1C - Plant Startup from CSD to Full Load with the RPV Closed at CSD

This transient is based on normal operations during a complete plant startup from cold shutdown to full load. This transient is identical to normal transient 1A except that the initial RCS temperature is 122°F.

Transient 2 - Complete Plant Shutdown from Full Load to CSD

Transient 2A - Plant Shutdown from Full Load to CSD with RPV Open

This transient is based on normal operations during a complete plant shutdown from full load to HSD and then to cold shutdown with the RPV open for refueling. The rate of power decrease is 5 percent FP per minute. The RCS cooldown continues to a final RCS temperature of 60°F.

Transient 2B - Plant Shutdown from Full Load to CSD with RPV Closed

This transient represents the normal operations during a complete plant shutdown from full load to HSD and then to CSD without removal of the reactor head. This transient is identical to Normal transient 2A except that the reactor head is closed at CSD and the final RCS temperature is 122°F.

Transient 3 - Heatup to 250°F with Subsequent Shutdown

This transient consists of a heatup of the RCS from CSD to 250°F followed by a return to CSD. The heatup and cooldown are identical to those described in normal transients 1A and 2A except that the maximum RCS temperature is 250°F.