ArevaEPRDCPEm Resource

From:	Pederson Ronda M (AREVA NP INC) [Ronda.Pederson@areva.com]
Sent:	Tuesday, March 31, 2009 8:16 PM
To:	Getachew Tesfaye
Cc:	BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC); VAN NOY Mark (EXT); HEDRICK Gary E (AFS)
Subject:	Response to U.S. EPR Design Certification Application RAI No. 155, Supplement 1
Attachments:	RAI 155 Supplement 1 Response US EPR DC.pdf

Getachew,

AREVA NP Inc. (AREVA NP) provided responses to 5 of the 78 questions of RAI No. 155 on February 13, 2009. The attached file, “RAI 155 Supplement 1 Response U.S. EPR DC” provides technically correct and complete responses to 20 of the remaining 73 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 155 Supplement 1 Questions 03.08.01-04, 03.08.01-05, 03.08.01-21, 03.08.02-09, 03.08.03-02, 03.08.03-09, 03.08.05-03, and 03.08.05-04.

The following table indicates the respective page(s) in the response document, “RAI 155 Supplement 1 Response U.S. EPR DC,” that contain AREVA NP’s response to the subject questions.

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The schedule for technically correct and complete responses to the remaining 53 questions is unchanged and provided below:

Question RAI 155 #	Response Date
RAI 155 — 03.08.01-03	October 30, 2009
RAI 155 — 03.08.01-06	October 30, 2009

RAI 155 — 03.08.01-07	April 30, 2009
RAI 155 — 03.08.01-08	May 29, 2009
RAI 155 — 03.08.01-09	May 29, 2009
RAI 155 — 03.08.01-10	May 29, 2009
RAI 155 — 03.08.01-11	June 30, 2009
RAI 155 — 03.08.01-12	May 29, 2009
RAI 155 — 03.08.01-16	May 29, 2009
RAI 155 — 03.08.01-17	April 30, 2009
RAI 155 — 03.08.01-20	October 30, 2009
RAI 155 — 03.08.01-22	May 29, 2009
RAI 155 — 03.08.01-24	October 30, 2009
RAI 155 — 03.08.01-27	May 29, 2009
RAI 155 — 03.08.02-01	June 30, 2009
RAI 155 — 03.08.02-02	July 31, 2009
RAI 155 — 03.08.02-03	April 30, 2009
RAI 155 — 03.08.02-04	June 30, 2009
RAI 155 — 03.08.02-05	May 29, 2009
RAI 155 — 03.08.02-06	May 29, 2009
RAI 155 — 03.08.02-07	July 31, 2009
RAI 155 — 03.08.02-08	July 31, 2009
RAI 155 — 03.08.02-10	May 29, 2009
RAI 155 — 03.08.03-03	May 29, 2009
RAI 155 — 03.08.03-04	July 31, 2009
RAI 155 — 03.08.03-05	April 30, 2009
RAI 155 — 03.08.03-06	May 29, 2009
RAI 155 — 03.08.03-10	May 29, 2009
RAI 155 — 03.08.03-11	May 29, 2009
RAI 155 — 03.08.03-12	May 29, 2009
RAI 155 — 03.08.03-14	April 30, 2009
RAI 155 — 03.08.03-15	April 30, 2009
RAI 155 — 03.08.03-16	July 31, 2009
RAI 155 — 03.08.03-17	July 31, 2009
RAI 155 — 03.08.04-02	April 30, 2009
RAI 155 — 03.08.04-03	May 29, 2009
RAI 155 — 03.08.04-04	May 29, 2009
RAI 155 — 03.08.04-05	May 29, 2009
RAI 155 — 03.08.04-06	October 30, 2009
RAI 155 — 03.08.05-01	July 31, 2009
RAI 155 — 03.08.05-02	May 29, 2009
RAI 155 — 03.08.05-05	April 30, 2009
RAI 155 — 03.08.05-06	May 29, 2009
RAI 155 — 03.08.05-07	June 30, 2009
RAI 155 — 03.08.05-08	July 31, 2009
RAI 155 — 03.08.05-10	July 31, 2009

RAI 155 — 03.08.05-11	April 30, 2009
RAI 155 — 03.08.05-12	April 30, 2009
RAI 155 — 03.08.05-13	June 30, 2009
RAI 155 — 03.08.05-14	June 30, 2009
RAI 155 — 03.08.05-15	June 30, 2009
RAI 155 — 03.08.05-16	June 30, 2009
RAI 155 — 03.08.05-18	June 30, 2009

Sincerely,
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From: Pederson Ronda M (AREVA NP INC)
Sent: Friday, February 13, 2009 7:18 PM
To: 'Getachew Tesfaye'
Cc: BENNETT Kathy A (OFR) (AREVA NP INC); DELANO Karen V (AREVA NP INC); VAN NOY Mark (EXT); HARRIS Carolyn A (AREVA NP INC)
Subject: Response to U.S. EPR Design Certification Application RAI No. 155, FSAR Ch. 3

Getachew,
Attached please find AREVA NP Inc.'s (AREVA NP) response to the subject request for additional information (RAI). The attached file, "RAI 155 Response US EPR DC.pdf" provides technically correct and complete responses to 5 of the 78 questions.

Appended to this file are affected pages of the U.S. EPR Final Safety Analysis Report in redline-strikeout format which support the responses to RAI 155 Questions 03.08.01-15, 03.08.01-18, 03.08.01-19, and 03.08.01-26.

The following table indicates the respective pages in the response document, "RAI 155 Response US EPR DC.pdf," that contain AREVA NP's response to the subject questions.

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A complete answer is not provided for 73 of the 78 questions. The schedule for a technically correct and complete response to these questions is provided below.

Question #	Response Date
RAI 155 — 03.08.01-01	March 31, 2009
RAI 155 — 03.08.01-02	March 31, 2009
RAI 155 — 03.08.01-03	October 30, 2009
RAI 155 — 03.08.01-04	March 31, 2009
RAI 155 — 03.08.01-05	March 31, 2009
RAI 155 — 03.08.01-06	October 30, 2009
RAI 155 — 03.08.01-07	April 30, 2009
RAI 155 — 03.08.01-08	May 29, 2009
RAI 155 — 03.08.01-09	May 29, 2009
RAI 155 — 03.08.01-10	May 29, 2009
RAI 155 — 03.08.01-11	June 30, 2009
RAI 155 — 03.08.01-12	May 29, 2009
RAI 155 — 03.08.01-13	March 31, 2009
RAI 155 — 03.08.01-16	May 29, 2009

RAI 155 — 03.08.01-17	April 30, 2009
RAI 155 — 03.08.01-20	October 30, 2009
RAI 155 — 03.08.01-21	March 31, 2009
RAI 155 — 03.08.01-22	May 29, 2009
RAI 155 — 03.08.01-23	March 31, 2009
RAI 155 — 03.08.01-24	October 30, 2009
RAI 155 — 03.08.01-25	March 31, 2009
RAI 155 — 03.08.01-27	May 29, 2009
RAI 155 — 03.08.02-01	June 30, 2009
RAI 155 — 03.08.02-02	July 31, 2009
RAI 155 — 03.08.02-03	April 30, 2009
RAI 155 — 03.08.02-04	June 30, 2009
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RAI 155 — 03.08.02-06	May 29, 2009
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RAI 155 — 03.08.02-08	July 31, 2009
RAI 155 — 03.08.02-09	March 31, 2009
RAI 155 — 03.08.02-10	May 29, 2009
RAI 155 — 03.08.03-01	March 31, 2009
RAI 155 — 03.08.03-02	March 31, 2009
RAI 155 — 03.08.03-03	May 29, 2009
RAI 155 — 03.08.03-04	July 31, 2009
RAI 155 — 03.08.03-05	April 30, 2009
RAI 155 — 03.08.03-06	May 29, 2009
RAI 155 — 03.08.03-07	March 31, 2009
RAI 155 — 03.08.03-08	March 31, 2009
RAI 155 — 03.08.03-09	March 31, 2009
RAI 155 — 03.08.03-10	May 29, 2009
RAI 155 — 03.08.03-11	May 29, 2009
RAI 155 — 03.08.03-12	May 29, 2009
RAI 155 — 03.08.03-13	March 31, 2009
RAI 155 — 03.08.03-14	April 30, 2009
RAI 155 — 03.08.03-15	April 30, 2009
RAI 155 — 03.08.03-16	July 31, 2009
RAI 155 — 03.08.03-17	July 31, 2009
RAI 155 — 03.08.04-01	March 31, 2009
RAI 155 — 03.08.04-02	April 30, 2009
RAI 155 — 03.08.04-03	May 29, 2009
RAI 155 — 03.08.04-04	May 29, 2009
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RAI 155 — 03.08.04-06	October 30, 2009
RAI 155 — 03.08.05-01	July 31, 2009
RAI 155 — 03.08.05-02	May 29, 2009
RAI 155 — 03.08.05-03	March 31, 2009

RAI 155 — 03.08.05-04	March 31, 2009
RAI 155 — 03.08.05-05	April 30, 2009
RAI 155 — 03.08.05-06	May 29, 2009
RAI 155 — 03.08.05-07	June 30, 2009
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RAI 155 — 03.08.05-12	April 30, 2009
RAI 155 — 03.08.05-13	June 30, 2009
RAI 155 — 03.08.05-14	June 30, 2009
RAI 155 — 03.08.05-15	June 30, 2009
RAI 155 — 03.08.05-16	June 30, 2009
RAI 155 — 03.08.05-17	March 31, 2009
RAI 155 — 03.08.05-18	June 30, 2009

Sincerely,

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From: Getachew Tesfaye [mailto:Getachew.Tesfaye@nrc.gov]
Sent: Wednesday, January 14, 2009 9:33 AM
To: ZZ-DL-A-USEPR-DL
Cc: Jim Xu; Samir Chakrabarti; Sujit Samaddar; Michael Miernicki; Joseph Colaccino; ArevaEPRDCPEm Resource
Subject: U.S. EPR Design Certification Application RAI No. 155 (1671, 1831,1672, 1834, 1833, 1836), FSAR Ch. 3

Attached please find the subject requests for additional information (RAI). A draft of the RAI was provided to you on December 12, 2008, and discussed with your staff on January 13, 2009. No changes were made to the Draft RAI Questions 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: 368

Mail Envelope Properties (5CEC4184E98FFE49A383961FAD402D31CA05A6)

Subject: Response to U.S. EPR Design Certification Application RAI No. 155, Supplement 1
Sent Date: 3/31/2009 8:15:49 PM
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MESSAGE	12934	3/31/2009 8:15:57 PM
RAI 155 Supplement 1 Response US EPR DC.pdf		1654106

Options
Priority: Standard
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Response to

Request for Additional Information No. 155, Supplement 1

01/14/2009

U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 03.08.01 - Concrete Containment
SRP Section: 03.08.02 - Steel Containment
SRP Section: 03.08.03 - Concrete and Steel Internal Structures of Steel or Concrete Containments
SRP Section: 03.08.04 - Other Seismic Category I Structures
SRP Section: 03.08.05 - Foundations
Application Section: FSAR Section 3.8

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

Question 03.08.01-1:

FSAR Section 3.8.1.1 states that the reactor containment building (RCB) accommodates the calculated pressure and temperature conditions resulting from a loss of coolant accident (LOCA) without exceeding the design leakage rate and with sufficient margin. The FSAR indicates that the design pressure is 62 psig and the design temperature is 309.2 ºF. For calculation of the ultimate pressure capacity of the containment, Table 3.8-6 identifies that the maximum design basis temperature is 395 ºF. For performance of the in-service inspection (ISI) of the containment, Table 3.8-7 provides the ISI schedule. Depending on the number of years from construction, either Pd (design pressure) or Pa (accident pressure) is specified. FSAR Section 6.2.1.1.2 states that the design pressure and temperature of the containment are 62 psig and 338°F, respectively. Based on this information, AREVA is requested to address the following:

1. If the containment design pressure (Pd) is 62 psig, explain what is the containment accident pressure (Pa) used in the ISI schedule. If they are different values explain the basis for selecting the accident pressure.

2. Explain why the containment design temperature of 309.2 ºF, presented in Section 3.8.1.1, is not consistent with the maximum design basis temperature of 395 ºF, presented in Table 3.8-6, nor consistent with the design temperature of 338°F, presented in Section 6.2.1.1.2.

Response to Question 03.08.01-1:

1. The containment design pressure (P_d) was conservatively assumed to be 62 psig in the structural analyses. .The containment accident pressure (P_a) is the calculated peak internal pressure associated with a postulated design basis accident and is the pressure identified for ISI.

2. The 395°F and 338°F temperatures are atmospheric temperatures associated with DBAs in U.S. EPR FSAR Tier 2, Section 6.2 and supporting analyses as part of containment design. The 309°F temperature represents the saturation temperature at the inner surface of the containment liner with consideration of condensation at the face. This temperature is based in the containment DBA analyses with consideration of boundary conditions at the containment wall. The 309°F temperature design temperature is then used to establish a thermal gradient through the containment wall to determine the design loads and moments used in design of the wall section, reinforcement and associated design components.

FSAR Impact:

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

Question 03.08.01-2:

FSAR Section 3.8.1.1.3 states that the liner plate is not used as a strength element to carry design basis loads. However, in the same section it states that no load transfer attachments are used at the bottom of the liner plate to transfer loads from the concrete reactor building (RB) internal structure into the lower portion of the nuclear island (NI) common basemat foundation. Instead the RB internal lateral reaction loads are transferred through the liner plate by lateral bearing on the haunch wall. If the entire lateral load from the RB internal structure is resisted by the haunch wall then describe how the lateral load and overturning moment from the internal structure were considered in the analysis and design of the haunch wall and NI basemat. This should include a description of how this behavior was represented in the finite element model (FEM), and how it was demonstrated that no uplift occurred between the containment internal structure and the containment liner as well as uplift between the containment liner and the NI basemat due to the overturning loads.

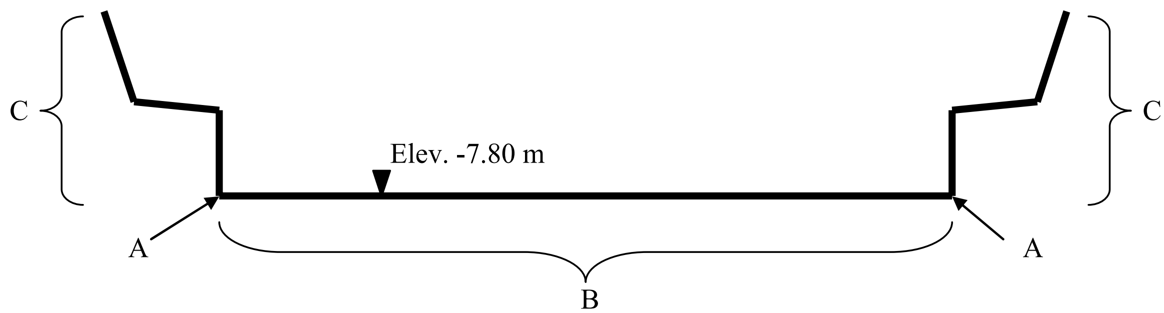
Response to Question 03.08.01-2:

A global static finite element model (FEM) was created for the entire Nuclear Island structure to facilitate application of many of the design loadings. A separate FEM was created to investigate stability of the Reactor Building Internal Structures (RBIS). This additional FEM utilized only the RBIS portion of the global static model, with boundary conditions representative of the interface between the RBIS and the Nuclear Island common basemat. The RBIS model boundary conditions were applied as follows:

- The nodes around the periphery of the circular base (see Point A in Figure 03.08.01-2-1) were fixed against translation in all three directions. These nodes are free to rotate in all three directions.
- The remaining nodes located on the bottom of the circular base (see Point B in Figure 03.08.01-2-1) were fixed against translation in both lateral directions. These nodes are free to rotate in all three directions, and to translate in the vertical direction.
- The remainder of the nodes along the interface surface (see point C in Figure 03.08.01-2-1) are free to translate and rotate in all directions. This is conservative since these surfaces, if restrained in the model, would provide additional resistance against sliding and overturning.

Figure 03.08.01-2-1

Interface Between RBIS and Common Basemat Structure



The load combinations recommended by NUREG-0800, Sections 3.8.3 and 3.8.5, were investigated. The total number of combinations to be considered for RBIS stability analysis was reduced based on the following observations:

- The smallest factors of safety (FS) for RBIS stability analysis (i.e., for sliding or overturning) will result from load combinations containing significant lateral and uplift loads.
- Hurricane wind (W) and tornado wind (W_t) can result in significant lateral loads; however they are not applicable for RBIS stability analysis since the RBIS is completely enclosed by the Shield Building and the Reactor Building.
- Flotation is not an issue since the RBIS is not a submerged or partially submerged structure.
- Only those load combinations which contain seismic loads (E') or accident pipe reactions (R_a) will result in significant lateral loads on the RBIS.

After reduction of the load combinations as discussed above, the following load combinations remained for RBIS stability analysis:

- B-05 $D + L + H + F + F_b + J + E' + F_a + P_a + T_a + R_a + R_r$
- H-05 $D + H + F + F_b + E'$

where B-05 and H-05 are identification numbers assigned to these load combinations

The independent loads which make up these load combinations are defined as follows:

- D = dead load
- L = live load
- H = lateral earth pressure loads
- F = hydrostatic loads
- F_b = buoyancy loads
- J = post-tensioning loads
- E' = seismic loads
- F_a = flooding loads
- P_a = accident pressure loads
- T_a = accident temperature loads
- R_a = accident pipe reaction loads
- R_r = pipe rupture reaction loads

Several of these independent loadings were not considered for RBIS stability analysis, based on the following:

- H the RBIS is not exposed to soil.
- F_b the RBIS is not a submerged or partially submerged structure.
- J the RBIS is not post-tensioned.
- F_a internal flood loads will not generate any significant net lateral or uplift loadings on the RBIS.
- P_a accident pressure loads will not generate any significant net lateral or uplift loadings on the RBIS.
- T_a accident temperature loads will not generate any net significant lateral or uplift loadings on the RBIS.
- R_r pipe rupture loads are local loads and are not considered in global, stability type analyses.

After removal these independent loadings, the two controlling load combinations reduced to:

- B-05 $D + L + F + E' + R_a$
- H-05 $D + F + E'$

These load combinations were then expanded to account for the directionality of the various independent loadings (i.e., 100-40-40 combinations of E' and reversible values of R_a). This resulted in a total of 32 possible permutations for B-05 and 24 possible permutations for H-05. It was further determined that the permutations which include significant "down" loadings resulting from E' or R_a need not be considered, since these downward loadings increase the stability of the RBIS. This resulted in a total of 16 remaining permutations of B-05 and 12 remaining permutations of H-05.

Seismic loadings were applied to the FEM as static equivalent loadings (i.e., zero period acceleration or ZPA). The U.S. EPR is designed to envelop a total of twelve different soil conditions, which result in twelve different sets of ZPAs. The ZPA values applicable to the RBIS were reviewed, and a total of four sets were determined to be potentially controlling for RBIS stability analysis. This resulted in a total of 4*16 = 64 permutations of B-05 and 4*12 = 48 permutations of H-05.

Following solution of the FEM for each load combination, the following additional tasks were performed to check the stability of the RBIS against sliding:

- The reactions at the base of the RBIS were determined, where;

  R_{hx} = horizontal reaction in x-direction

  R_{hy} = horizontal reaction in y-direction

  R_v = vertical reaction

- The sliding force was determined as: $F_{slide} = (R_{hx}^2 + R_{hy}^2)^{1/2}$.
- The resisting force was determined as: $F_{resist} = \mu * R_z$, where μ (friction coefficient) is 0.45.

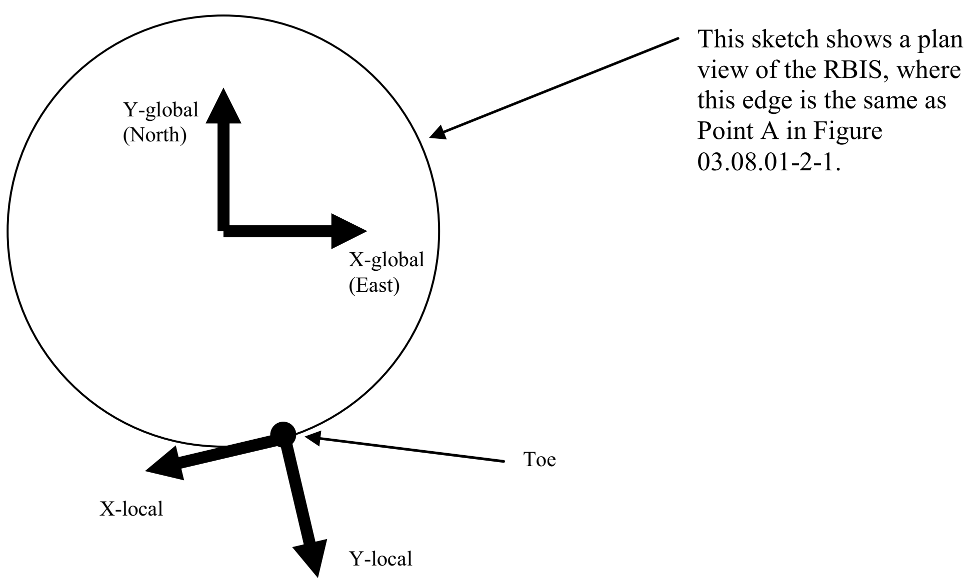
- The factor of safety (FS) against sliding was determined as: $FS_{slide} = F_{resist} / F_{slide}$.
- The minimum FS against sliding was identified: $FS_{min} = 0.16$.

Following solution of the FEM for each load combination, the following additional tasks were performed to check the stability of the RBIS against overturning:

- The vertical support node (see Point A in Figure 03.08.01-2-1) which carries the largest compressive reaction load is identified. This is the point that the RBIS is trying to overturn about (i.e., the "toe") for the current loading condition.
- A local coordinate system (CS) is established. The origin of the local CS is located at the toe. The X-axis of the local CS is located tangent to the periphery of the RBIS foundation. The direction of the X-axis is such that a moment about this axis is positive if the moment is toward the center of the RBIS, and negative otherwise. See Figure 03.08.01-2-2.
- The summation point (for use in the following ANSYS *FSUM* command) is relocated to the origin of the new local CS.
- The results CS is changed from the global CS to the new local CS.
- The reactions and moments about the toe, in the new local CS, are determined using the ANSYS *FSUM* command.
- The primary item of interest from the ANSYS *FSUM* command is the moment about the local x-axis, which is referred to below as the net moment (ΣM).

Figure 03.08.01-2-2

RBIS Overturning Moment Calculation



To allow determination of the FS against overturning, the associated resisting moment and overturning moment must be determined. The net moment (ΣM) about the toe is obtained as discussed above. The net moment is defined as:

- $\Sigma M = M_{resist} - M_{ot}$

where: M_{resist} = the total resisting moment on the RBIS about the toe

M_{ot} = the total overturning moment on the RBIS about the toe

To determine the total resisting moment, the resisting moment due to each of the appropriate independent loads is first determined. From a review of the applicable independent loads, it was determined that only the dead (D), live (L) and hydrostatic (F) loads will contribute significantly to the total resisting moment. All other independent loadings will result primarily in overturning moment, and are therefore ignored for determination of resisting moment. Once the individual resisting moments for D, L, and F are determined, the total resisting moment for each load combination is determined.

For independent loadings D, L, and F, the resisting moment is determined as:

- $M_{resist} = R_z$*arm

where: R_z = the vertical reaction force (defined earlier)

arm = distance between the toe & the center of gravity (CG) of the applied loading

From a review of the geometry of the RBIS and the location of the loadings applied by D, L, and F, it was determined to be reasonable to use the distance from the geometric center of the structure to the toe as the "arm" value (i.e., D, L, and F are estimated as symmetrically applied loads). Since the toe is always located at a given radius from the geometric center of the RBIS, the "arm" is a constant value.

For all permutations of loading combination B-05, the total M_{resist} value is determined as:

- $M_{resist} = 1.0^*M_{resist}$(from D) + 1.0^*M_{resist}(from F) + 0.25^*M_{resist}(from L)

For all permutations of loading combination H-05, the total M_{resist} value is determined as:

- $M_{resist} = 1.0^*M_{resist}$(from D) + 1.0^*M_{resist}(from F)

The total overturning moment (M_{ot}) about the toe is calculated as:

- $M_{ot} = -(M_{resist} - \Sigma M)$

The factor of safety (FS) against overturning was determined as: $FS_{ot} = - M_{resist} / M_{ot}$.

The minimum FS against overturning was identified: $FS_{min} = 1.22$.

The required FS against sliding, as recommended by NUREG-0800, Section 3.8.5, is 1.10. Since the minimum FS is much less than the required FS, the RBIS can slide if not properly restrained at the base against lateral movement.

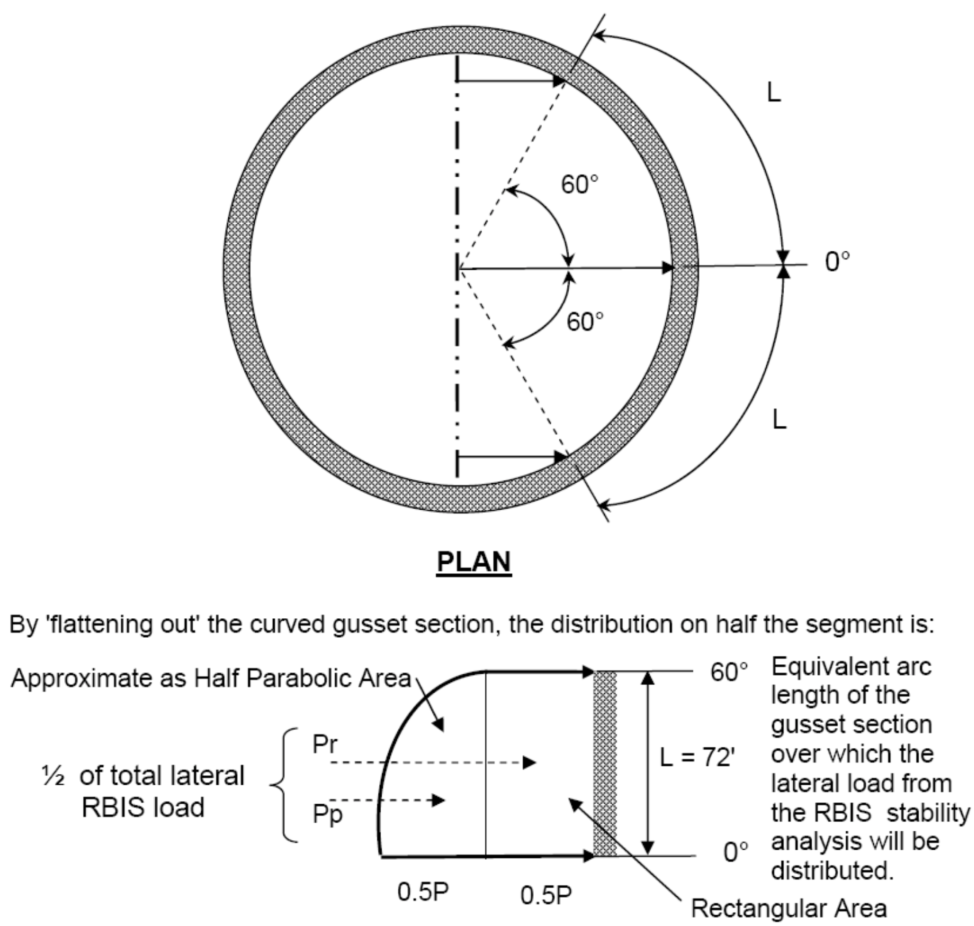
The required FS against overturning, as recommended by NUREG-0800, Section 3.8.5, is 1.10. Since the minimum FS is greater than the required FS, the RBIS cannot overturn.

To restrain the RBIS against sliding, the haunch (i.e., the transition area between the containment wall and the NI basemat) was designed to resist the entire sliding load (F_{slide}). As discussed above, the design sliding load is obtained from a separate FEM developed specifically to investigate RBIS stability. Therefore, the sliding load was added to the appropriate design loadings obtained from the global static FEM.

For the haunch, F_{slide} was applied to a 120º arc of the haunch. Therefore, the entire sliding load is assumed to be carried by one-third of the total length of the haunch. The intensity of F_{slide} applied to the haunch varied along the length from a maximum (cos 0º = 1.0) at the center of the arc to a minimum (cos 60º = 0.5) at the edges of the arc. See Figure 03.08.01-2-3 for clarification.

Figure 03.08.01-2-3

RBIS Lateral Load Distribution



Due to the rigidity of the RBIS baseslab and haunch area, the intensity of F_{slide} was applied uniformly in the vertical direction. Therefore, for a given vertical slice of haunch, the pressure due to F_{slide} did not vary from top to bottom, and the bending moment in the haunch was determined using a moment arm equal to half the free height of the haunch. The resulting shear and bending moments were added to the haunch design loadings obtained from the global static FEM.

The RBIS basemat was designed for loadings obtained from the global static FEM.

A vertical uplift between the liner plate and the internal structure or basemat is not credible because overturning is resisted by the configuration of the internal structure/haunch and the gravity loads due to the mass of the internal structure is greater than the vertical acceleration.

FSAR Impact:

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

Question 03.08.01-4:

FSAR Section 3.8.1.2 describes the codes, standards, and specifications followed for the design, fabrication, construction, testing and inservice inspection of the RCB. AREVA is requested to explain the following items:

1. Since the RCB is founded on the same NI basemat as several other seismic category I structures, explain where is the ASME containment jurisdictional boundary defined for the EPR plant which must satisfy the code requirements of the ASME Section III, Division 2. The response should consider the fact that the containment basemat is integrally connected to the rest of the NI foundation, and thus additional peripheral volume of concrete and anchorage of the containment shell reinforcement beyond the containment wall should be included in the jurisdictional boundary. In addition, AREVA is requested to confirm that all loads (e.g., wind, lateral earth pressure, etc.) arising from the evaluation of the common basemat outside the rules of ASME Code Section III, Division 2, are considered in combination with those specified for the ASME Code Section III, Division 2 basemat.
2. ASCE Standard 4-98, Seismic Analysis of Safety-Related Nuclear Structures and Commentary is identified under the heading of applicable codes in Sections 3.8.1.2.1 and 3.8.2.2.1 of the FSAR. AREVA should recognize that this Standard is not a code and should explain where this standard is utilized in the design of the containment. AREVA should preferably not reference this Standard because the NRC staff has not generically endorsed it for seismic analysis of nuclear power plants, or alternatively AREVA should explain the specific provisions from this Standard that were utilized and provide the technical basis for their use. This also needs to be addressed for FSAR sections 3.8.2 – 3.8.5.
3. ASCE/SEI Standard 43-05, Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities is also identified under the heading of applicable codes in Sections 3.8.1.2.1 and 3.8.2.2.1 of the FSAR. AREVA should recognize that this Standard is not a code and should explain where this standard is utilized in the design of the containment. AREVA should preferably not reference this Standard because the NRC staff has not generically endorsed it for seismic analysis of nuclear power plants, or alternatively AREVA should explain the specific provisions from this Standard that were utilized and provide the technical basis for their use. This also needs to be addressed for FSAR sections 3.8.2 – 3.8.5.

Response to Question 03.08.01-4:

1. ASME Section III Division 2 (subarticle CC-1140) does not provide specific guidance on where to locate the jurisdictional boundary between ASME Section III Division 2 and other design codes (i.e., ACI 349). As per ASME Section III Section NCA (subparagraph NCA-3254.2), the jurisdictional boundary shall be specified in the Design Specification. This indicates that the specific location of the jurisdictional boundary is to be determined by the designer. The jurisdictional boundary selected for the U.S. EPR, as applicable to the common basemat, is a cylinder aligned with the outside face of the reactor containment building wall. The complete jurisdictional boundary is shown in Figure 03.08.01-4-1. Together, the turquoise and blue areas form the portions of the common basemat structure under the jurisdiction of ASME Section III Division 2. All loads arising from the evaluation of the common basemat outside the rules of ASME Code Section III, Division 2, are considered in combination with those specified for the ASME Code Section III, Division 2 basemat. For example, when analyzing load combinations for design of the containment

wall, Wind Loads (W) and Lateral Earth Pressure Loads (H) are applied to the appropriate exterior walls, even though these walls do not form a part of the containment structure. This assures that any effects from these loadings, which can be transferred through the common basemat, are accounted for in the design of the containment wall. This philosophy is also applied to the analysis and design of the Reactor Building Internal Structures (RBIS) and the Other Category I structures, which are also supported by the common basemat. For example, Post Tension Loads (J), which are <u>directly</u> applicable only to the Containment Building, are also included with all load combinations used for analysis and design of the RBIS, Safeguard Buildings, and Fuel Building. This ensures that any effects induced on non-containment structures due to the application of Post Tension Loads (J) to the containment structure are accounted for.

2. It is acknowledged that ASCE Standard 4-98 is not a code and the reference will be removed from the list of Codes in U.S. EPR FSAR, Tier 2, Sections 3.8.1.2.1 and 3.8.2.2.1.

   – Provisions of ASCE Standard 4-98 were utilized in analysis with regard to:

     a. The 100-40-40 percent rule for combining the three components of an earthquake.

     b. Seismic induced soil pressures.

     c. Hydrodynamic loads.

   – The respective technical basis for utilization is:

     a. In RAI 155, Question 03.08.03-10, NRC requested that AREVA “provide the technical basis which demonstrates the adequacy of the 100-40-40 method taken from ASCE 4-98.” Therefore, the request for the technical basis is redundant to RAI 155, Question 03.08.03-10 and it will be provided in response to that question.

     b. The use of ASCE 4-98 for calculating dynamic soil pressures is explicitly stated as acceptable in SRP 3.8.1, under SRP Acceptance Criteria paragraph 4.E and in SRP 3.8.4 under SRP Acceptance Criteria 4.H.

     c. In determining hydrodynamic loads, the requirements of both ASCE 4-98 and USAEC TID-7024 are met.

3. It is acknowledged that ASCE Standard 43-05 is not a code and the reference will be removed from the list of Codes in U.S. EPR FSAR, Tier 2, Sections 3.8.1.2.1 and 3.8.2.2.1.

   – ASCE/SEI 43-05 is referenced only in regards to the reserve energy approach used to estimate sliding distances. As per the response to RAI 3.8.5-4.8, this approach has not been used.

4. Although not identified in the question, it was self-identified that the Codes section also listed ASCE/SE Standard 7-05 and SEI/ASCE Standard 37-02. These Standards were also removed from the list of Codes as indicated on the enclosed markup.

FSAR Impact:

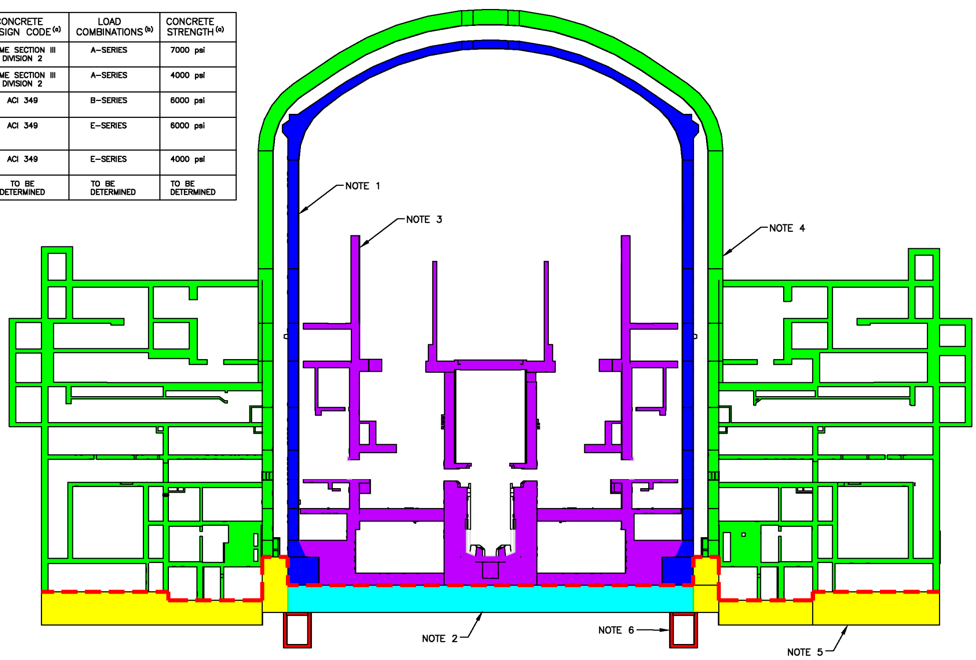
U.S. EPR FSAR Tier 2, Section 3.8.1.2.1 and Section 3.8.2.2.1 will be revised as described in the response and indicated on the enclosed markup.

Figure 03.08.01-4-1

Jurisdictional Boundaries

CONCRETE DESIGN BOUNDARIES FOR THE
U.S. EPR COMMON BASEMAT STRUCTURES

NOTE	STRUCTURE CLASSIFICATION[(a)]	CONCRETE DESIGN CODE[(a)]	LOAD COMBINATIONS[(b)]	CONCRETE STRENGTH[(a)]
1	CONTAINMENT (WALLS & DOME)	ASME SECTION III DIVISION 2	A–SERIES	7000 psi
2	CONTAINMENT (BASESLAB)	ASME SECTION III DIVISION 2	A–SERIES	4000 psi
3	R.B. INTERNAL STRUCTURES	ACI 349	B–SERIES	6000 psi
4	OTHER CATEGORY I STRUCTURE (ALL EXCEPT BASESLAB)	ACI 349	E–SERIES	6000 psi
5	OTHER CATEGORY I STRUCTURE (BASESLAB)	ACI 349	E–SERIES	4000 psi
6	TO BE DETERMINED	TO BE DETERMINED	TO BE DETERMINED	TO BE DETERMINED



Question 03.08.01-5:

FSAR Section 3.8.1.3.1 - Design Loads, defines the various loads to be utilized for the analysis and design of the containment. AREVA is requested to address the following items related to design loads:

1. For dead loads (D), explain whether the term "permanent equipment" used in the definition includes the weight of components such as cable tray systems, conduit systems, HVAC systems, etc. in addition to individual equipment/components. Provide the magnitude of the "permanent equipment load" and "other loads" used in addition to the dead weight of the structural element. Explain why the dead weight of the piping and its contents are included under "Pipe Reactions (Ro)" rather than under dead loads (D). Typically, Ro is reserved for piping reaction loads arising from loads other than dead load and earthquake. Treating the pipe dead load as Ro results in its elimination in some load combinations. Explain why hydrostatic loads (F) due to water stored in pools and tanks are defined separately from dead loads. This has resulted in its elimination from the load combinations as noted in RAI 3.8.1-7.
2. For live loads (L), explain what magnitude was utilized for analysis and design, and the basis for this load magnitude.
3. For SSE (E'), the FSAR indicates that SSE loads are considered due to applied inertial loads, including dead loads, live loads, and hydrodynamic loads (i.e., water in storage pools and tanks). Explain whether the entire dead load, including the weight of all components discussed under item 1 above, were included as mass in the seismic model(s) to develop the member forces used for design. Explain what portion of the live load (discussed under item 2 above) was included as mass (in addition to the dead load mass) in the seismic model(s) to develop the member forces for design. Explain where does the FSAR provide a description of all the storage pools and tanks used in all seismic category I structures.

Response to Question 03.08.01-5:

U.S. EPR FSAR Tier 2, Section 3.8.1.3.1, Paragraph Reaction Loads (R_o), third line will be revised to read as follows:

> "The dead weight of the piping and its contents are not included."

1. The term "permanent equipment" as used in the definition of Dead Loads (D) does include the weight of components such as cable tray systems, conduit systems, HVAC systems, etc., in addition to individual equipment/components.

   The dead weight of piping, including its contents as well as all associated supports and restraints is included in Dead Loads (D), not in Pipe Reactions (R_o). U.S. EPR FSAR Tier 2, Section 3.8.1.3.1, will be revised to indicate that these loads are not included as Pipe Reactions (R_o).

   The static finite element model of the Nuclear Island includes discrete dead loads for all equipment (including cable tray, conduit, HVAC, etc.) identified as having a concentrated dead weight of 100 kN (22,481 lbs) or more. To account for smaller concentrated dead loads as well as various distributed dead loads, uniform dead loads were applied to the floor and wall systems as follows:

Reactor Building -	300 psf on concrete floors, 50 psf on each face of concrete walls
Fuel Building -	300 psf on concrete floors, 25 psf on each face of concrete walls
Safeguard Buildings -	200 psf on concrete floors, 25 psf on each face of concrete walls
All Steel Platforms -	25 psf

Hydrostatic Loads (F) are Dead Loads (D), and they are treated as such in the static finite element model of the Nuclear Island. Hydrostatic Load (F) is included in all load combinations, and is multiplied by the same load factor as the Dead Load (D). However, hydrostatic loads are a unique type of dead load, in that they can fluctuate due to filling or emptying of a pool or tank. Hydrostatic Loads (F) were defined separately from Dead Loads (D) purely as a bookkeeping measure, due to their somewhat unique status as potentially "variable" dead loads, and to allow them to be quickly identified during the analysis and design process.

2. The following typical Live Load (L) values are utilized for analysis and design of the Nuclear Island:

Reactor Building -	500 psf on concrete floors, 175 psf on steel platforms
Fuel Building -	500 psf on concrete floors, 175 psf on steel platforms
Safeguard Buildings -	300 psf on concrete floors, 175 psf on steel platforms
All Roof slabs -	100 psf

Larger values are utilized in selected areas such as the Fuel Building loading hall (due to transporters carrying new and spent fuel canisters) and in the Reactor Building near the equipment hatch (due to staging of equipment during a refueling outage).

These live loads were selected following a review of (a) the live loads used for the analysis and design of the European version of the EPR and (b) typical live loads found in various U.S. nuclear plants.

3. The entire Dead Load (D), including the weight of all components discussed under item 1 above and the Hydrostatic Load (F), were included as mass in the seismic model(s) to develop the member forces used for design.

Twenty-five percent of the Live Load (L) discussed under item 2 above was included as mass (in addition to the dead load mass) in the seismic model(s) to develop the member forces for design. One exception was the Live Load (L) applied to the roof slabs, which is controlled by precipitation loads. Seventy-five percent of the roof Live Load (L) was included as mass (in addition to the dead load mass) in the seismic model(s) to develop the member forces for design.

Information related to U.S. EPR storage tanks and pools can be found in the following sections of U.S. EPR FSAR Tier 2:

Seismic Category I

- 1.2.3.3.2 In-Containment Refueling Water Storage Tank.
- 3.8.3.1.9 In-Containment Refueling Water Storage Tank.
- 5.4.11 Pressurizer Relief Tank.
- Table 5.4-8 Pressurizer Relief Tank Design Parameters.
- 6.3.2.2.2 In-Containment Refueling Water Storage Tank.
- 9.1.3.2.2 Fuel Building and Reactor Building Pools.
- 9.2.2.2.2 Component Cooling Water System Surge Tanks.
- 9.2.2.2.2 Dedicated Component Cooling Water System Surge Tanks.
- Table 9.2.2-1 Component Cooling Water System Design Parameters.
- 9.2.7.2.2 Buffer Tanks.
- 9.2.8.2.2 Diaphragm Expansion Tanks.
- 9.3.4.2.2 Volume Control Tank.
- Table 9.3.4-1 Major CVCS Component Design Data.
- 9.5.4.2.2 Main Fuel Oil Storage Tanks.
- 9.5.4.2.2 Fuel Oil Day Tanks.
- 9.5.5.2.2 Cooling System Expansion Tank.
- 9.5.7.2.2 Engine Lube Oil Sump Tank.
- 9.5.7.2.2 Auxiliary Lube Oil Tank.
- 10.4.6.2.2 Spent Resin Tank.
- 10.4.7.2.2 Deaerator-Feedwater Storage Tanks.
- 10.4.8.2.2 Blowdown Flash Tank.
- 10.4.9.2.2.2 Emergency Feedwater Storage Pools.
- Table 10.4.9-1 Emergency Feedwater System Component Data.

Radwaste Seismic

- 11.2.2.4.1 Liquid Waste Storage Tanks.
- 11.2.2.4.1 Chemical Tanks.
- 11.2.2.4.1 Concentrate Tanks.
- 11.2.2.4.1 Monitoring Tanks.
- 11.2.2.4.1 Activity Measurement Tank.
- 11.2.2.4.2.1 Distillate Tank.

Radwaste Seismic (continued)

- 11.2.2.4.2.2 Sludge Tank.
- Table 11.2-2 Various tanks.
- 11.3.2.3.2 Sealing Liquid Tanks.
- 11.3.2.3.16 Condensate Collecting Tank.
- Table 11.3-2 Various tanks.
- 11.4.2.3.2 Resin Proportioning Tank..
- 11.4.2.3.2 Concentrate Buffer Tank.
- 11.4.2.3.2 Condensate Collection Tank.
- 11.4.2.3.2 Scrubber Tank.

FSAR Impact:

For item 1 the U.S. EPR FSAR Tier 2, Section 3.8.1.3.1 will be revised as described in the response and indicated on the enclosed markup. For items 2 and 3, the U.S. EPR FSAR will not be changed as a result of this question.

Question 03.08.01-13:

FSAR Section 3.8.1.4.1 - Computer Programs, refers only to the ANSYS computer code for analysis of the RCB and other structures. FSAR Section 3.8.4 discusses the use of another computer code GT STRUDL. AREVA is requested to address the following items related to the use of computer programs for all aspects of structural analysis and design:

1. Identify all versions of the computer programs that are utilized for all aspects of analysis and design of structures. This should include identification of the programs that are used for postprocessing of results of one computer code for use in another and combining output results.
2. For each of these computer programs, identify the program name and version number, describe what analyses they are used for, and how they were validated.
3. Confirm for each of these programs that the validation methods used are consistent with those described in SRP3.8.1 II.4.F.

Response to Question 03.08.01-13:

1. ANSYS Version 10.0 Service Pack 1 (64bit) is utilized for the design and analysis of all Nuclear Island common basemat structures. The mesh of the Nuclear Island common basemat structures static model is generated using the Preprocessor of ANSYS Version 10.0 Service pack 1 (64bit) and all results of the static analysis are obtained through the Postprocessor of ANSYS Version 10.0 Service Pack 1 (64bit).

   GT STRUDL Version 27 is utilized to generate the finite element model of the Emergency Power Generating Building and Essential Service Water Buildings for translation to a soil-structure interaction analysis which is performed using SASSI2000 Version 3.1.

   GT STRUDL Version 28 is utilized to generate the finite element model (stick model) of the Nuclear Island common basemat structures and is converted to a SASSI model for the purpose of soil-structure interaction analysis which is performed using SASSI Version 4.1B. RESPEC Version 1.1A is used to generate spectra from the SASSI soil-structure interaction analysis.

   GT STRUDL Versions 29 and 29.1 are utilized to generate the finite element model for the static analysis and design of the Emergency Power Generating Building and Essential Service Water Buildings. All results for design of the Emergency Power Generating Building and Essential Service Water Buildings are obtained through GT STRUDL Version 29.1.

   SASSI Versions 4.1B, 4.2, and SASSI2000 Version 3.1 are used to analyze soil structure interaction of the Nuclear Island Basemat, Emergency Power Generating Building and Essential Service Water Buildings. RESPEC Version 1.1A is used to generate spectra from the SASSI soil-structure interaction analysis.

   SHAKE2000 Version 1.1 is used to compute the required cross correlation functions used for the analysis of the Emergency Power Generating Building and Essential Service Water Buildings. The Fourier spectrum for the final time histories are obtained using SHAKE2000.

BSIMQKE Release B1-4PC is used to convert the seed time histories for the Emergency Power Generating Building and Essential Service Water Buildings to make them spectrum-compatible to their respective target response spectrum.

DATAN Release C1-4PC is used to compute the cross-correlation functions of the response spectrum matched time histories in x, y and z directions for the analysis of the Emergency Power Generating Building and Essential Service Water Buildings.

2. ANSYS 10.0 Service Pack 1 (64bit) is validated through a software installation test record exclusive to each computer system used for analysis. ANSYS is used to analyze the Nuclear Island including the reactor building and the reactor interior structures. ANSYS is validated through a series of test problems upon installation provided with the quality assurance agreement.

   GT STRUDL Versions 27, 28, 29 and 29.1 are used to compose finite element models. Versions 27, 29 and 29.1 are used for the Emergency Power Generating Building and Essential Service Water Buildings. Version 28 is used to model the Nuclear Island Basemat. GT STRUDL is validated through a generic model provided with the quality assurance program.

   SASSI Versions 4.1B, 4.2, and SASSI2000 Version 3.1 are used to analyze soil structure interaction of the Nuclear Island Basemat, Emergency Power Generating Building and Essential Service Water Buildings. SASSI Versions 4.1B, 4.2, and SASSI2000 Version 3.1 are validated through meeting an allowable percentage to a chain of test problems.

   SHAKE2000 Version 1.1 is used to compute the required cross correlation functions used for the analysis of the Emergency Power Generating Building and Essential Service Water Buildings. The Fourier spectrum for the final time histories are obtained using SHAKE2000. SHAKE2000 is validated by running test problems with known results and confirming those results are the same prior to performing the calculation.

   BSIMQKE Release B1-4PC is used to convert the seed time histories for the Emergency Power Generating Building and Essential Service Water Buildings to make them spectrum-compatible to their respective target response spectrum. BSIMQKE is validated by running test problems with known results and confirming those results are the same prior to performing the calculation.

   DATAN Release C1-4PC is used to compute the cross-correlation functions of the response spectrum matched time histories in x, y and z directions for the analysis of the Emergency Power Generating Building and Essential Service Water Buildings. DATAN is validated by running test problems with known results and confirming those results are the same prior to performing the calculation.

   RESPEC Version 1.1A is used to generate the response spectra corresponding to acceleration time histories from GT STRUDL, ANSYS, and SASSI. Test problems are used to verify through comparison of the RESPEC, Version 1.1A and its results.

3. The computer programs were validated with methods consistent with those described in SRP 3.8.1 II.4.F. Further descriptions are provided as follows:

   ANSYS Version 10.0 Service Pack 1 (64bit) is validated by confirming the computer program's solutions to a series of test problems substantially identical to those obtained from classical solutions. The test problems are demonstrated to be similar to or within the range of applicability of the classical problems analyzed to justify acceptance of the program. There is a software installation test record sheet which confirms this and it is supplied with each document where the software is used.

   GT STRUDL Versions 27, 28, 29 and 29.1 are validated by confirming the computer program's solutions to a series of test problems substantially identical to those obtained from classical solutions. Input files are supplied and used in the program to correlate supplied output files. These results must meet a required allowance. There is a software installation test record sheet which confirms this and it is supplied with each document that the software is used.

   SASSI Versions 4.1B, 4.2, and SASSI Version 3.1 are validated by confirming the computer program's solutions to a series of test problems substantially identical to those obtained from classical solutions. The test problems are demonstrated to be similar to or within the range of applicability of the classical problems analyzed to justify acceptance of the program.

   RESPEC Version 1.1A is validated by confirming the computer program's solutions to a series of test problems substantially identical to those obtained from classical solutions. The test problems are demonstrated to be similar to or within the range of applicability of the classical problems analyzed to justify acceptance of the program.

   SHAKE2000 Version 1.1 is validated by confirming the computer program's solutions with known results prior to performing the calculation.

   BSIMQKE Release B1-4PC is validated by confirming the computer program's solutions with known results prior to performing the calculation.

   DATAN Release C1-4PC is validated by confirming the computer program's solutions with known results prior to performing the calculation.

FSAR Impact:

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

Question 03.08.01-21:

FSAR Table 3E.1-1 lists the loads considered in the FEM of the RCB, and Table 3E.1-2 lists the loads not considered in the FEM but evaluated separately and added to the other loads for design. AREVA is requested to explain why the construction loads and combustion gas load C, which are defined in FSAR Section 3.8.3.1 are not also considered. In addition, explain why P_a in Table 3E.1-1 is only considered for the containment wall, since the jurisdictional boundary of the containment should include the basemat foundation and liner as well.

Response to Question 03.08.01-21:

Construction loads as defined in U.S. EPR FSAR Tier 2, Section 3.8.1.3.1 have not been applied to the Nuclear Island (NI) finite element model (FEM). Construction loadings and their effect on particular NI designs depend on actual fabrication scenarios and sequences. These loadings will be incorporated into the structural design, in combination with other loadings, as needed to produce an overall design. U.S. EPR FSAR Tier 2, Table 3E.1-2 will be revised to add the construction loading category.

The NRC's request is redundant to RAI 155, Question 03.08.01-6. Therefore, AREVA will provide the combustion gas loads, design methodology and results as part of the response to RAI 155 Question 03.08.01-6.

The containment accident pressure loads (P_a) are applied to the basemat. U.S. EPR FSAR Tier 2, Table 3E.1-1 notes that this load is for the containment wall to distinguish it from sub-compartment pressurization (Pa) loads not applied to the FEM, listed in U.S. EPR FSAR Tier 2, Table 3E.1-2. U.S. EPR FSAR Tier 2, Table 3E.1-1 will be modified to clarify.

U.S. EPR FSAR Tier 2, Table 3E.1-2 will add an additional line to the table to include the following independent load not considered in the FEM:

"CL Construction Loads"

U.S. EPR FSAR Tier 2, Table 3E.1-1 will add an additional line to the table to include the following independent load considered in the FEM:

"C Combustible Gas"

FSAR Impact:

U.S. EPR FSAR Tier 2, Table 3E.1-1 and Table 3E.1-2 will be revised as described in the response and indicated on the enclosed markup.

Question 03.08.01-23:

FSAR Section 3E.1.1 describes the element forces and moments obtained from the ANSYS FEM of the containment in accordance with Figure 3E.1-1. These element forces are in terms of shell element forces (e.g., membrane forces, shear forces, and bending forces) across the entire concrete section not the individual brick elements that make up the through wall section of the wall. Tables for the governing design data for the critical sections also provide such loads across the entire concrete section. Explain how these shell type section forces are developed when the FEM utilizes solid brick elements through the thickness of the walls.

Response to Question 03.08.01-23:

From the Nuclear Island (NI) Analysis, solid element output is always in terms of stresses, forces and moments at the existing nodes. Due to mesh density and geometry of the NI model, it was necessary for design to sum forces and moments on a surface that is not aligned with the existing node set. A numerical tool, ANSYS macro "SolidMF.mac", was employed to calculate the resultants on the desired surface. For a design surface, this macro interpolates stresses from solid element results and integrates these stresses to calculate equivalent shell results of forces and moments per unit length. The calculation of equivalent shell results from solid elements facilitates the use of common design approaches and tools for structural components of the Nuclear Island whether modeled with shell or solid elements.

FSAR Impact:

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

Question 03.08.01-25:

FSAR Section 3.E.1.3 states that a separate analysis was performed to determine the magnitude of in-plane shear produced by accidental torsion in the various walls of the NI common basemat structures. Describe the separate analysis including computer codes that is used to determine the in-plane torsional shear in the RCB and how these loads are combined with other loads in the structure.

Response to Question 03.08.01-25:

The primary analyses referenced in U.S. EPR FSAR Tier 2, Appendix 3E.1.3, first paragraph is the global analysis for the NI structure, but the primary analysis did not include the effects of accidental torsion. The accidental torsion load effects were specifically addressed using the alternative analysis method described under acceptance criteria of the Standard Review Plan (Section 3.7.2, page 3.7.2-16, Item 11, NUREG 0800) utilizing the ANSYS 10.0 SP1 computer code. This alternate analysis supplements the loads obtained from the global analyses by adding the in-plane shear from the accidental torsion analysis to the in-plane shear force from all other load combinations that include seismic loads.

FSAR Impact:

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

Question 03.08.02-9:

In FSAR Section 3.8.2.3.2, under Level B Service Limits, it states that if a component screens out of analysis for cyclic operation, Level B service limit load combinations may be eliminated. Define the technical basis for “screening out of analysis for cyclic operation.” If the screening criteria are based on Subsection NE of the ASME III Division 1 Code, identify the specific Code paragraph. If not based on the Code, describe what precedents exist for the criteria applied.

Response to Question 03.08.02-9:

ASME boiler and pressure vessel (BPV) Code, Section III, Division 1, Subsection NE, Subparagraph NE-3221.5 provides the technical basis for screening of analysis for cyclic operation. It states that if the specified Service Loadings for the component meet all of the requirements of NE-3221.5(d), analysis for cyclic service is not required.

This will be clarified by adding the ASME technical basis to the U.S. EPR FSAR Tier 2, Section 3.8.2.3.2, Level B Service Limits statement.

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 3.8.2.3.2, Level B Service Limits, will be revised as described in the response and indicated on the enclosed markup.

Question 03.08.03-1:

FSAR Section 3.8.3.1.1 provides some description of the reactor vessel (RV) support structure and reactor cavity. Since this description and associated figures are not sufficient to understand the structural elements, connections, and load path from the components to the containment internal structures, provide the following additional information:

1. Provide additional details which show how the RV ring is embedded into the concrete and the anchorage details.
2. Provide details of the components described in the second paragraph of FSAR Section 3.8.3.1.1 which include the large penetrations in the circular RV support concrete wall, permanently installed cavity seal ring, and neutron shield assembly resting on the embedded ring at the top of the wall.
3. Provide details of the embedment plates, baseplates, grout (if applicable) and anchorages for the RV; vertical and horizontal supports of the steam generators, reactor coolant pumps, and pressurizers; and the polar crane steel plate support brackets.

Response to Question 03.08.03-1:

1. The Reactor Pressure Vessel (RPV) is supported by the nozzles of the four hot legs and four cold legs of the Reactor Coolant System (RCS). Each nozzle of the RCS bears on a support block incorporated into the upper plate of the support ring. The load is transferred through vertical stiffener plates and the cylindrical ring plate to the lower plate that bears on the top, inside edge of the reactor cavity wall. The base of the RPV ring is at approximate elevation 13 ft, 5 inches.

   Lateral restraint of the RPV ring is provided by eight vertical keys that are welded to the bottom plate; the vertical keys fit into vertical block-outs formed in the inner face of the concrete reactor cavity wall.

   Once the RPV ring is set in place and leveled, grout is placed in the vertical block-outs in the void between the concrete wall and vertical keys and under the lower plate.

   Under each RCS leg, anchor bolts connect the lower plate to a plate embedded in the reactor cavity wall.

   U.S. EPR FSAR Tier 2, Figure 5.4-9 shows the Reactor Pressure Vessel support ring.

2. The large penetrations in the circular RPV concrete wall are block-outs that allow the legs of the RCS to pass through. There is a block-out for each leg of the RCS. See Figure 03.08.03-1-1.

   The cavity seal ring and neutron shield is seismically supported and permanently attached to the reactor vessel support wall. Specific details of the neutron shield design will be developed later in the design process.

3. The details of the RPV ring were discussed in response to Part 1 of this response above.

   Each steam generator (SG) is supported vertically at its base by four steel columns with pinned joints. The columns attach to a floor at approximate elevation 4 ft, 11 inches. The base of each column is supported on a circular base plate that bears on a bed of grout. The base plate is connected to the floor by embedded rods that extend through the concrete floor and bolt to an anchor frame. See U.S. EPR FSAR Tier 2, Figure 5.4-10 for details.

Each steam generator is supported laterally by four struts attached to the upper shell at approximate elevation 64 feet. Each upper strut is attached to the wall of the steam generator cavity with anchor rods connecting to an embedded plate. Grout is applied under the base plate. See Figure 03.08.03-1-2.

Each steam generator is also supported laterally at its base by two lateral restraints. The connection of the lateral restraint to the concrete wall consists of embedded rods bolted through a base plate. Grout is applied under the base plate. See Figure 03.08.03-1-3.

Each reactor coolant pump (RCP) is supported vertically at its base by three steel columns with pinned joints. The columns attach to a floor at approximate elevation 4 ft, 11 inches. The reactor coolant pumps are connected to the concrete floor in a manner similar to the steam generators. See U.S. EPR FSAR Tier 2, Figure 5.4-11.

Each reactor coolant pump is supported laterally by two supports attached to the concrete walls of the RCP cubicle at approximate elevation 28 feet. The RCPs are attached to the concrete wall in a manner similar to the upper lateral restraints of the SGs. See U.S. EPR FSAR Tier 2, Figure 5.4-11.

The pressurizer is supported vertically by three welded supports attached to the side of its lower cylindrical shell. Each welded support is attached to a floor at approximate elevation 49 feet using threaded rods. The threaded rods extend through the concrete floor and bolt to the underside. Base plates are used under the welded supports and on the underside of the floor. The base plates are welded to a steel sleeve and this assembly acts as an anchor frame for the threaded rods. A bed of grout is placed under the base plate supporting the welded support. See Figure 03.08.03-1-4.

The pressurizer is supported laterally by eight restraints at approximate elevation 68 feet. The connection of the lateral restraint to the concrete floor consists of embedded rods and a base plate. Each rod screws into a threaded sleeve welded to the back of a plate with corresponding holes. Grout is applied under the base plate. See Figure 03.08.03-1-4.

The polar crane support brackets are fabricated of plate material. The bracket extends from the interior face of the containment building wall. A section cut through the bracket shows vertical plates that provide vertical support for the polar crane rail support beams. Plates are welded to the top and bottom of the vertical plates to form a box cross section. This assembly is welded to a base plate which forms a part of the RCB liner plate. Bars with 90° hooks are welded around the perimeter of the back of the assembly to resist the tensile force developed in the bracket. The vertical support provided by the bracket is transferred to the concrete RCB wall by steel plates positioned at an approximate angle of 30° to horizontal. There are approximately 45 total brackets supporting the circular girder of the polar crane. See Figure 03.08.03-1-5.

The following figures were referenced in response to this RAI question:

Figure 03.08.03-1-1	Reactor Vessel Support Ring
Figure 03.08.03-1-2	Steam Generator Upper Lateral Support
Figure 03.08.03-1-3	Steam Generator Lower Lateral Support
Figure 03.08.03-1-4	Pressurizer Support
Figure 03.08.03-1-5	Polar Crane Girder Support Bracket

FSAR Impact:

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

Figure 03.08.03-1-1

Reactor Vessel Support Ring

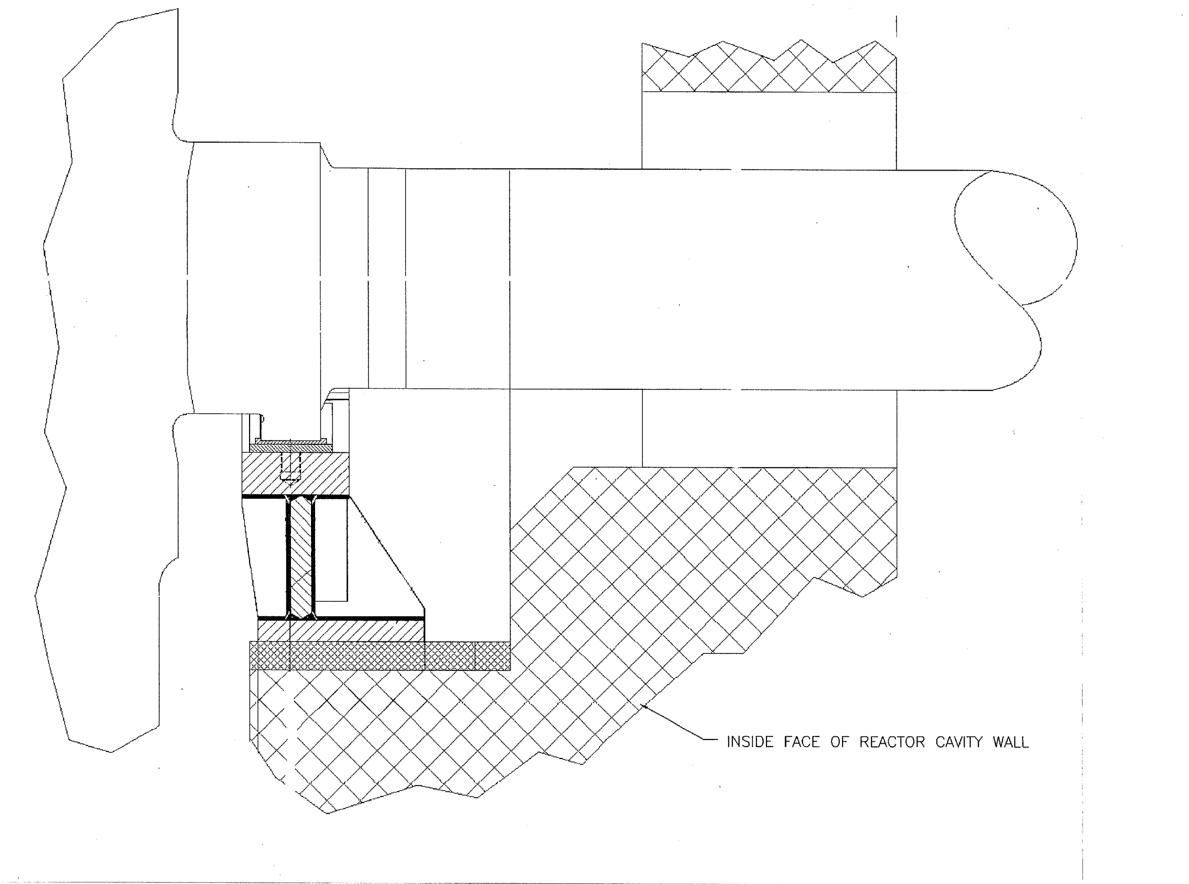


Figure 03.08.03-1-2

Steam Generator Upper Lateral Support

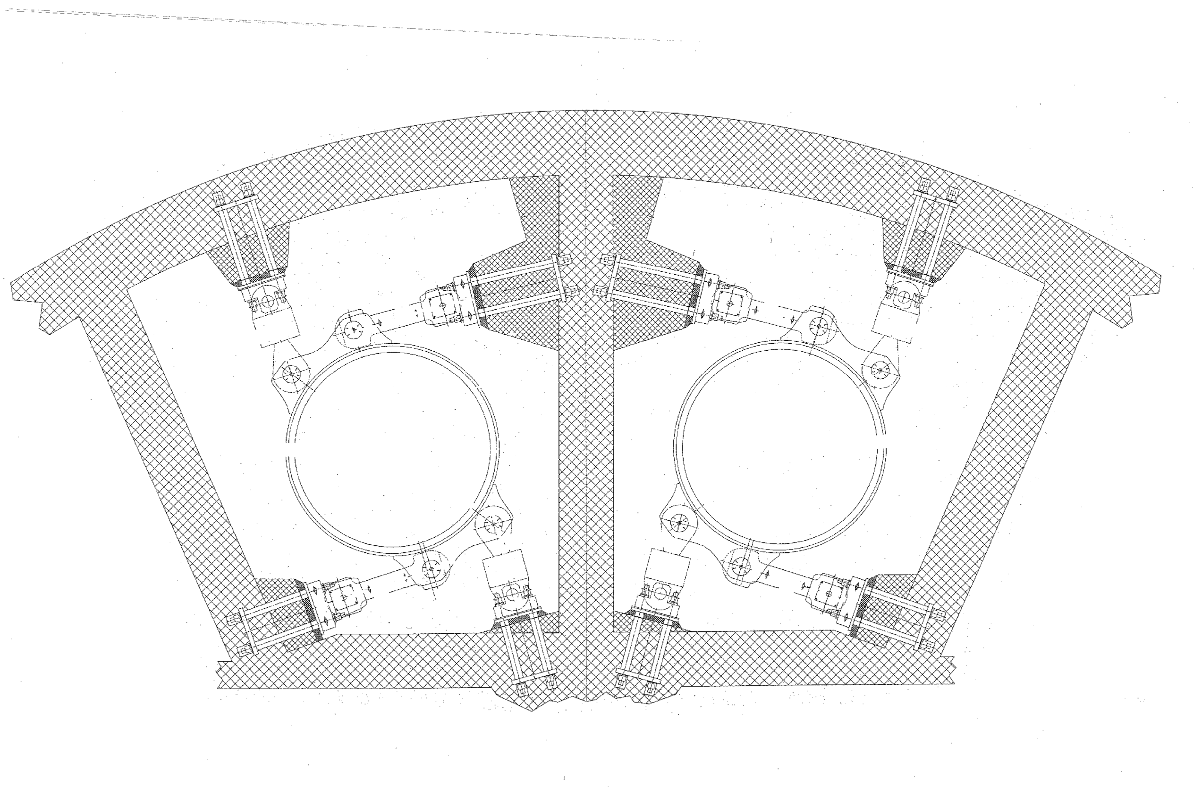


Figure 03.08.03-1-3

Steam Generator Lower Lateral Support

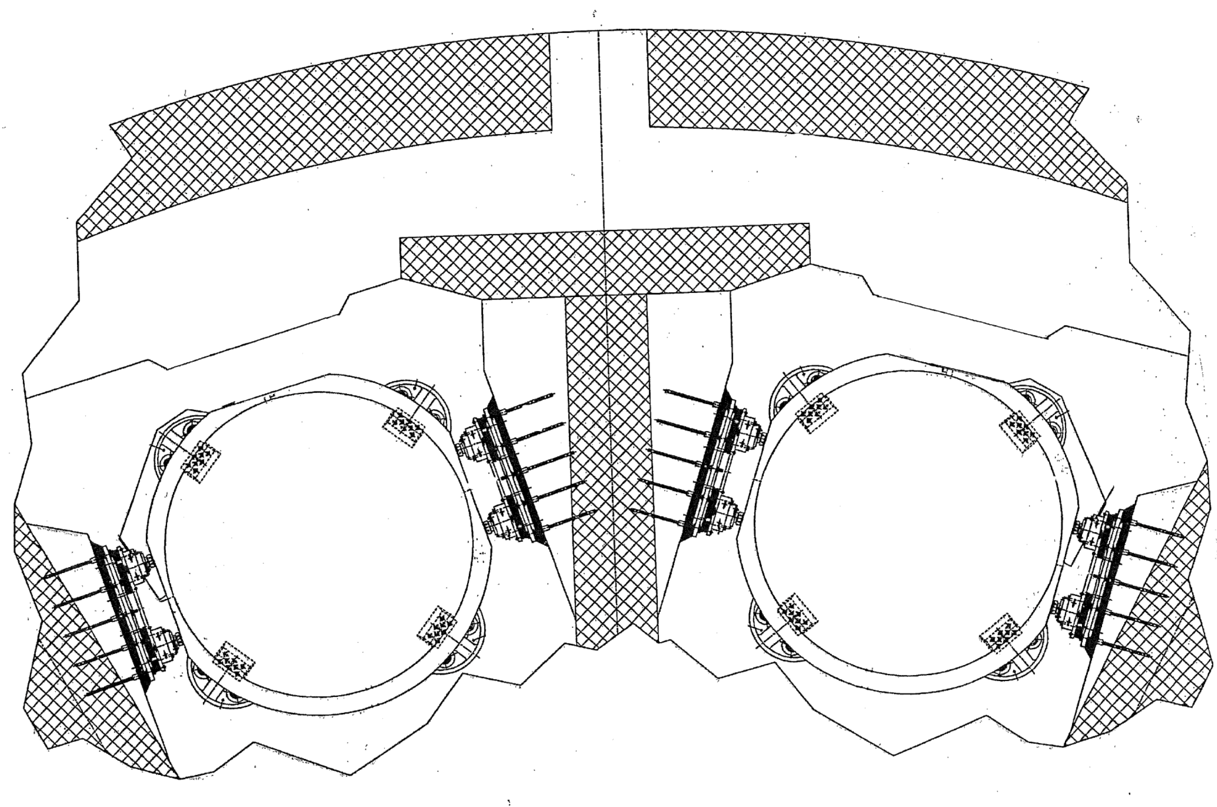


Figure 03.08.03-1-4

Pressurizer Support

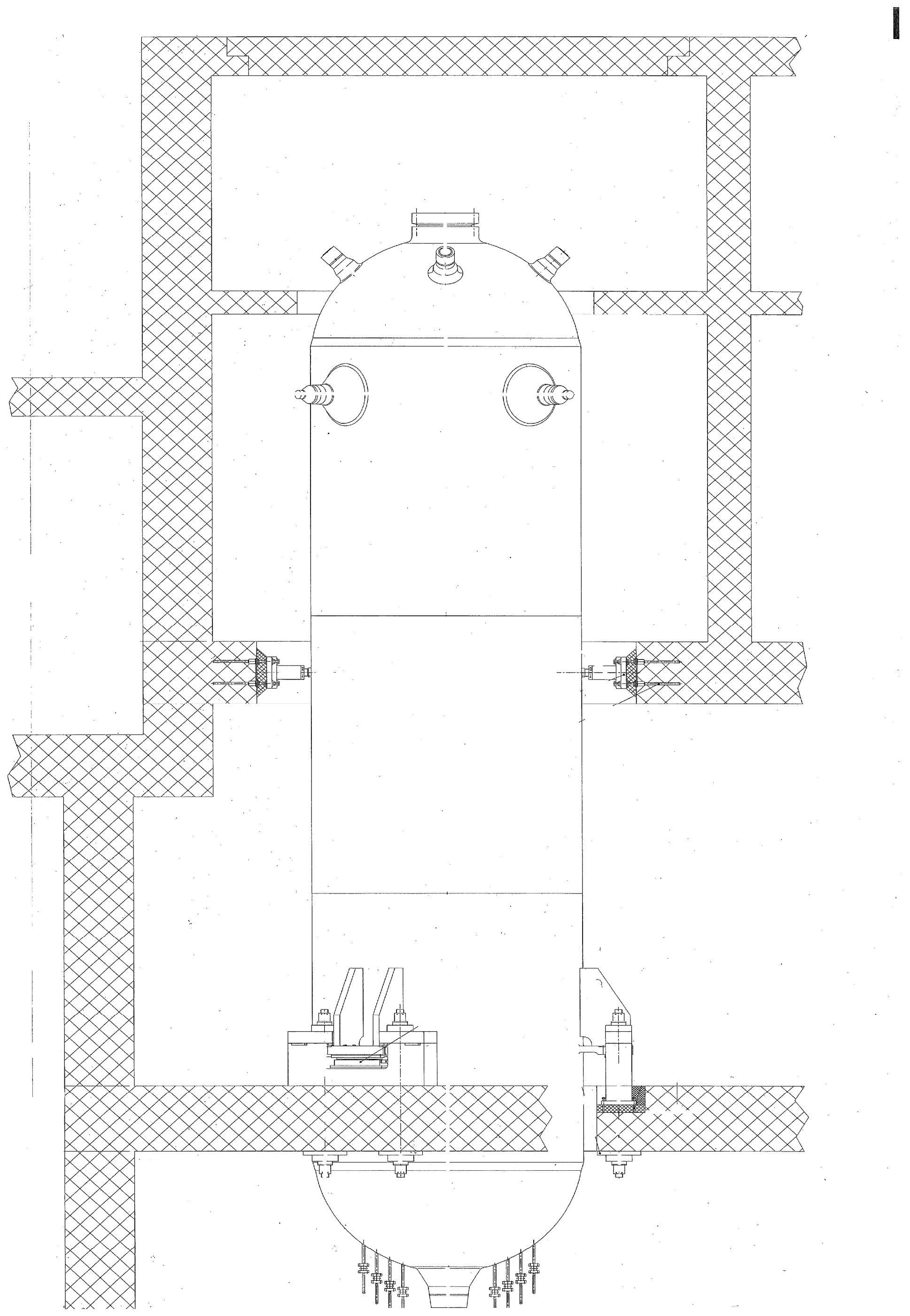
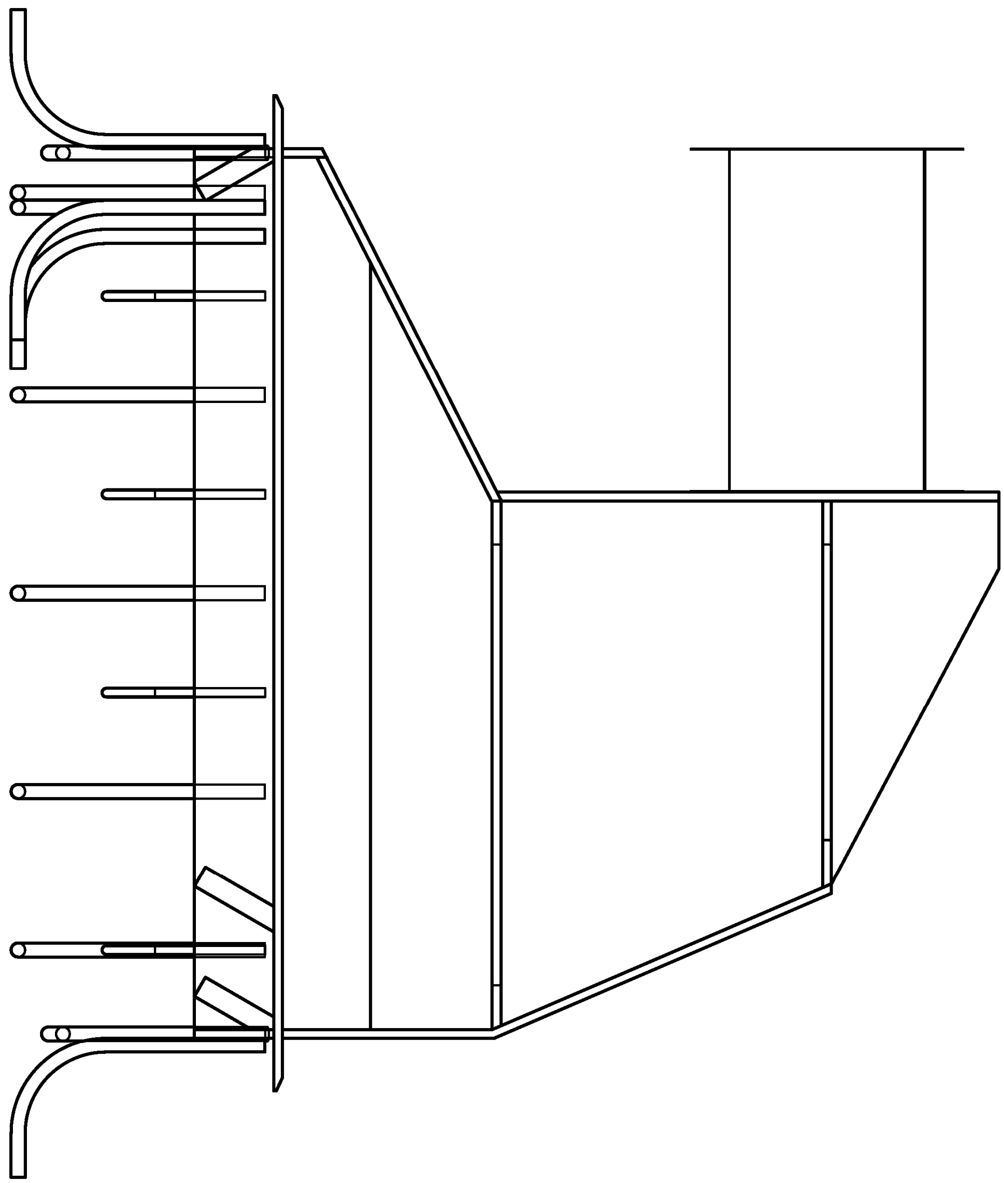


Figure 03.08.03-1-5

Polar Crane Girder Support Bracket



Question 03.08.03-2:

FSAR Section 3.8.3.1.10 - Distribution System Supports, indicates that structural steel supports are provided for distribution systems as part of the RB internal structures. These include pipe supports, equipment supports, cable tray and conduit supports, HVAC duct supports, and other component supports. Distribution system supports are primarily constructed of steel shapes and tubing, which are anchored to the concrete RB internal structures using embedded steel plates, cast-in place anchor bolts, and drilled-in concrete anchors. For concrete anchors of all types that are discussed in FSAR Sections 3.8.1 through 3.8.5, for all components attached to concrete structural elements (not just distribution systems), AREVA is requested to explain whether the criteria listed below is utilized and to insert the criteria the FSAR, or explain why not:

1. The design and installation of all anchor bolts are performed in accordance with Appendix B to ACI 349-01 - “Anchoring to Concrete,” subject to the conditions and limitations specified in RG 1.199 (November 2003).
2. The design and installation of all anchor bolts are also performed in accordance with the information presented in NRC IE Bulletin 79-02, Revision 2, which includes criteria for anchor bolt safety factors, baseplate flexibility, and other criteria.

Response to Question 03.08.03-2:

1. The following sentence will be added to U.S. EPR FSAR Tier 2, Sections 3.8.1.5, 3.8.3.5 and 3.8.5.5:

   “Limits for allowable loads on concrete embedments and anchors are in accordance with Appendix B of ACI 349-2006 and guidance given in RG 1.199.”

   This sentence already appears in U.S. EPR FSAR Tier 2, Section 3.8.4.5.

   The design and installation of all concrete anchors are in accordance with Appendix B to ACI 349-06 subject to the conditions and limitations in RG 1.199 (November 2003) and apply to the following:

   - Concrete containment (U.S. EPR FSAR, Tier 2, Section 3.8.1).
   - Reactor Building internal structures (U.S. EPR FSAR, Tier 2, Section 3.8.3).
   - Other Seismic Category I structures (U.S. EPR FSAR, Tier 2, Section 3.8.4).
   - Foundations for Seismic Category I structures (U.S. EPR FSAR, Tier 2, Section 3.8.5).

   The use of ACI 349-01 and RG 1.199 is not applicable to the steel containment (U.S. EPR FSAR Tier 2, Section 3.8.2).

2. NRC IE Bulletin 79-02, Revision 2, discusses the use of concrete expansion anchors and their deficiencies. The issues addressed in the bulletin pertain to base plate flexibility and the capacity, installation, and testing of expansion anchors.

   Since the issuance of IE Bulletin 79-02, Revision 2, extensive work and research has been performed in the area of concrete anchors. The knowledge gained by testing and research has been incorporated into ACI 349-2001, Appendix B, to adequately address the issues of capacity, installation, and testing raised in the bulletin. Therefore, concrete anchors are designed in accordance with ACI 349, Appendix B. Additionally, the guidelines of RG 1.199, including those pertaining to installation, inspection, and sampling of post-installed anchors are followed.

The issue of base plate flexibility presented in IE Bulletin 79-02, Revision 2, is addressed by following the guidelines for base plate design according to the American Institute of Steel Construction (AISC). Additional anchor forces resulting from base plate flexibility (i.e., prying action) are included in the design of the anchorage when the base plate does not meet rigid plate requirements.

Therefore, the design and installation of anchor bolts is performed in accordance with the information presented in NRC IE Bulletin 79-02, Revision 2, by:

- Adhering to the design requirements of Appendix B to ACI 349.
- Adhering to the regulatory positions of RG 1.199.
- Following the AISC guidelines for base plate design.

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 3.8.1.5, Section 3.8.3.5 and Section 3.8.5.5 will be revised as described in the response and indicated on the enclosed markup.

Question 03.08.03-7:

FSAR Section 3.8.3.1.2 describes removable panels in the interior walls of each steam generator (SG) cubicle and states that these reinforced concrete wall panels are keyed into the side walls of the SG cubicles and to the slab at the bottom of the panels to prevent dislodgement during seismic events. As the panels must maintain their structural integrity and remain in place under a combination of loads, provide the method of analysis used for qualification of such non-integral concrete structural systems. Also describe how the reaction loads from these panels are imposed on the side walls and slab of the SG cubicle.

Response to Question 03.08.03-7:

Per U.S. EPR FSAR Tier 2, Figure 3B-11, there are two removable panels at Elevation 93 feet, 6 inches enclosing the inside face of the Steam Generator enclosures. The stiffness and mass of these panels was accounted for in the dynamic model with an appropriate pinned boundary condition. Therefore, forces and moments due to seismic loadings are calculated as well as the reaction loads from these panels on the supporting surfaces. Additionally, design forces, moments and reactions; dead load, LOCA pressures, etc., needed to complete a given loading combination are computed using hand calculations. Combined results are used for the panel design and combined reaction forces are used for connection design. The calculated additional forces and moments are used in combination with the ANSYS finite element results for the design of the supporting walls and slab.

FSAR Impact:

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

Question 03.08.03-8:

FSAR Section 3.8.3.4.1 states that for RB internal structures, localized abnormal loads are not included in the overall analysis. These loads include sub compartment pressure loads, pipe break thermal loads, accident pipe reactions, pipe break loads, and local flood loads. Instead local analyses are used to address these localized loads. Some additional information on the local analysis and design is provided in FSAR Section 3.8.3.4. In order to understand how these analyses and design are performed AREVA is requested to address the items listed below. This information is also requested for the localized analyses for other Category I structures described in FSAR Sections 3.8.4 and 3.8.5 (if applicable):

1. Provide the method and basis for performing the localized analysis for each type of abnormal load. This should include the potential effects of concrete cracking due to accident thermal loads and redistribution of member forces due to cracking of concrete if significant.
2. Describe how the results of the localized analyses are combined with the results of the overall structural analyses for other loads.

Response to Question 03.08.03-8:

1. Abnormal loads are those loads generated by a postulated high-energy pipe break accident. This event is classified as a “Design Basis Accident”. Included in this category are: Internal Flooding loads (Fa), Buoyant Force (Fb), Pressure loads (Pa), Thermal loads (Ta), Accident Pipe Reaction loads (Ra), and Pipe Break loads (Rr). The Pipe Break load is subcategorized as: Pipe Break Reaction loads (Rrr), Pipe Break Jet Impingement loads (Rrj), and Pipe Break Missile Impact loads (Rrm). These loadings include an appropriate dynamic load factor to account for the dynamic nature of the load, unless a time-history analysis is performed to justify otherwise.

   In the global analysis of the Nuclear Island System (NIS), the nuclear steam supply system (NSSS) is the only high-energy line considered in developing Ra and Rr. Furthermore, Rrr is the only component of Rr considered in the global analysis of the NIS. Rrj and Rrm are considered for structural members designed to accommodate the effects of, and be compatible with, the environmental conditions associated with these loads. Their effects on the overall response of the NIS are not considered significant and are not included in the formation of load combinations for the global system.

   With respect to thermal loads (Ta), thermal stress was evaluated considering the stiffness of the member and the rigidity of the section and the degree of restraint of the structure. The evaluation may be based on cracked section properties, provided the following conditions are met (ACI 349, Appendix A):

   (a) The tensile stress for any section exceeds the tensile stress at which the section is considered cracked.
   (b) Redistribution of internal forces and strains due to cracking are included.
   (c) All concurrent loads, as specified in the load combinations, are considered.

   When thermal stress is combined with the stress due to other loads to determine a design stress, the magnitude of the design stress must not be less than the magnitude of the stress due to other loadings alone unless the following are considered:

(a) The effect of cracking in the tensile zone of flexural members on reduction of the flexural rigidity and on the redistribution of stress.
(b) The reduction of long term stresses due to creep.
(c) Stress combinations that reduce the magnitude of the stress due to other loads utilizing actual temperatures and temperature distributions which act concurrently with the other loads.

2. Design loads for the Reactor Building Internal Structures (RBIS) are combined in accordance with the review criteria specified in the SRP, Section 3.8.3, which references the load criteria of ACI 349 for concrete and AISC N690 for steel, as applicable, with exceptions as noted in the SRP. The specific analyses differentiate between loads used for the design of concrete (Ultimate Strength Design) and steel (Working Stress Design or Plastic Design) in addition to design condition. The design conditions and number of load combinations in each is summarized as follows:

   (a) Normal + Abnormal (contains 8 load combinations).
   (b) Normal + Extreme Environmental + Abnormal (contains 32 load combinations).

FSAR Impact:

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

Question 03.08.03-9:

FSAR Section 3.8.3.4.2 indicates that openings in walls and slabs of RB internal structures are shown on construction drawings and that openings are acceptable without analysis if they meet the criteria identified in ACI 349, Section 13.4.2. This referenced section of ACI 349 is applicable to openings in slabs, not walls. Therefore, provide the technical justification for the use of these criteria for walls or revise the approach described in the FSAR to be consistent with the provisions in ACI 349 for design of openings in concrete walls, which among other provisions must also meet the requirements of Chapter 21 – Special Provisions for Seismic Design.

Response to Question 03.08.03-9:

It is acknowledged that unanalyzed openings are limited to slabs for which the criteria of ACI Section 13.4.2 are satisfied. U.S. EPR FSAR Tier 2, Section 3.8.3.4.2 will be revised to include the following:

> "Openings in slabs are acceptable without analysis if they meet the criteria identified in ACI, Section 13.4.2."

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 3.8.3.4.2 will be revised as described in the response and indicated on the enclosed markup.

Question 03.08.03-13:

FSAR Sections 3.8.3.6.5, 3.8.4.6.3, and 3.8.5.6.3 provide a brief description of modular construction methods and composite type structural members used in the EPR. Provide a more detailed description, including figures, of each specific type of module or composite member used in the EPR. Also provide a description of the analysis and design approach used for each type of module and composite member. FSAR Sections 3.8.3.6.5 and 3.8.4.6.3 also state that decking, plates, and beams, as well as other types of formwork, may be left in place and become a permanent part of the structure. Provide details and a description of the analysis and design approach used for each of these items.

Response to Question 03.08.03-13:

U.S. EPR FSAR Tier 2, Sections 3.8.3.6.5, 3.8.4.6.3, and 3.8.5.6.3 state that all structures are constructed using proven methods common to heavy industrial construction. No special, new, or unique construction techniques are used.

No designated modules or three dimensional structural units are used in the U.S. EPR. The construction process will prefabricate sections and the pre-assembled modules/sections will be transported and placed. These members are designed to withstand the loads that occur during transportation and rigging activities. Examples of such structural members are listed in U.S. EPR FSAR Tier 2, Sections 3.8.3.6.5, 3.8.4.6.3, and 3.8.5.6.3. Examples of prefabrication include the in-containment refueling water storage tank (IRWST) liner, refuel canal liner, spent fuel pool liner, reinforcing, and concrete formwork.

Construction aids such as steel decking, plates, and supporting steel beams that may be used to form concrete of general access during construction may also be left in place. In these instances, the design structural thicknesses do not credit the structural aid. Construction aids that become a permanent part of the structure conform to code requirements and are designed and permanently attached to prevent their failure from interacting with Seismic Category 1 structures, systems and components (SSC).

FSAR Impact:

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

Question 03.08.04-1:

FSAR Section 3.8.4 does not discuss the design of Radwaste Structures. It is also noted that FSAR Section 3.8.4.2.5 does not reference RG 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in LWR Plants." FSAR Tables 3.2.2-1 and 3.7.2-29 state that the Nuclear Auxiliary Building (NAB) and the Radioactive Waste Processing Building (RWPB) are Radwaste Structures and are designed in accordance with guidance for RW-IIa structures in RG 1.143. Since these structures are part of the design certification and are designed in accordance with RG 1.143, provide in FSAR Section 3.8.4 the design details for these structures comparable to that provided for other Category I structures. The staff notes that FSAR Section 1.2.3.1.2 states that the NAB and RWPB are described in FSAR Section 3.8.4.

Response to Question 03.08.04-1:

Safety-related structures, systems, and components (SSC), including their foundations and supports, are classified as Seismic Category I per RG 1.29. Additionally, certain SSC specifically identified in RG 1.29 are also classified as Seismic Category I. Radioactive Waste Management Systems (RWMS) are explicitly excluded from RG 1.29 as Seismic Category I SSC.

SSC that by definition are not safety related but to which a "significant licensing requirement or commitment" applies are classified as NS-AQ as described in U.S. EPR FSAR Tier 2, Section 3.2.1. Per RG 1.143, both Radioactive Waste (Processing) Building (RWB) and Nuclear Auxiliary Building (NAB) are RW IIa structures designated as NS-AQ (non-safetyrelated, augmented quality). The RWB and the NAB are classified as neither 'Seismic Category I' nor 'safety related'. Therefore, they (and reference to RG 1.143) are not included in U.S. EPR FSAR Tier 2, Section 3.8.4.

For seismic design, RWMS SSC that are classified as RW-IIa per RG 1.143 are subject to the applicable seismic requirements tabulated in the RG. These SSC are designed for loads up to 1/2 SSE and are seismically categorized as Radwaste Seismic (RS). RWMS SSC classified as other than RW-IIa per RG 1.143, are categorized as Conventional Seismic (CS) and also designed to the seismic requirements of the RG 1.143.

For the NAB, requirements for interaction with Seismic Category I SSC per U.S. EPR FSAR Tier 2, Section 3.7.2.8 apply.

The U.S. EPR FSAR, Tier 2, Section 1.2.3.1.2, cross-references to Section 3.8.4 for the NAB and RWB were previously revised to read as follows:

> "The interaction of the Nuclear Auxiliary Building with Seismic Category I structures is described in Section 3.7.2."
>
> "The interaction of the Radioactive Waster Processing Building with Seismic Category I structures is described in Section 3.7.2."

The seismic category and safety classification for the Nuclear Auxiliary Building (NAB) and the Radioactive Waste (Processing) Building (RWB) are appropriately noted in U.S. EPR FSAR Tier 2, Tables 3.2.2-1 and 3.7.2-29.

FSAR Impact:

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

Question 03.08.05-3:

FSAR Section 3.8.5.4.1 states that the design of steel structures used for Seismic Category I foundations is performed in accordance with ANSI/AISC N690. Clarify where this specification will be used for foundation design since the FSAR does not describe any steel Seismic Category I foundations. If any steel foundations are used in the EPR design, provide descriptions of these foundations and information comparable to that provided for the concrete foundations.

Response to Question 03.08.05-3:

Per U.S. EPR FSAR Tier 2, Sections 3.8.5.1.1, 3.8.5.1.2 and 3.8.5.1.3, all foundations for Seismic Category I structures are constituted of either reinforced concrete slabs or reinforced concrete shear walls. Therefore, there are no foundations for Seismic Category I structures composed of structural steel. The references to structural steel foundations will be removed from the FSAR.

The following paragraph in U.S. EPR FSAR Tier 2, Section 3.8.5.4.1, will be deleted to remove reference to steel foundations.

> “Design of steel structures used for Seismic Category I foundations is performed in accordance with ANSI/AISC N690-1994 (R2004), including Supplement 2 (GDC 1). Steel member design uses the allowable stress design methods of ANSI/AISC N690.”

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 3.8.5.4.1 will be revised as described in the response and indicated on the enclosed markup.

Question 03.08.05-4:

FSAR Section 3.8.5.4.1 includes a discussion of general procedures applicable to Seismic Category I foundations. With regard to the discussion in this section, AREVA is requested to provide the following information:

1. FSAR Table 3.8-11 provides minimum required factors of safety against overturning, sliding and flotation for foundations for various load combinations that are consistent with SRP 3.8.5. FSAR Table 3.8-12 provides the corresponding minimum factors of safety for the NI Common Basemat Structure foundation. For the load combinations including W, Wt, and Fb, explain the method used to calculate the reported minimum factors of safety.
2. FSAR Table 3.8-12 refers to FSAR Section 3.8.5.4.2 for the minimum factors of safety for overturning and sliding for the load combination including E.' No values are provided in this section. However, FSAR Section 3.8.5.5 states that for the load combination containing seismic loads, the calculated minimum factors of safety are less than the values provided in NUREG 0800. These calculated factors of safety for overturning and sliding for this load combination should be provided in the FSAR along with a description of the methods used to determine these factors of safety. The need for additional information on this issue is discussed under RAI 3.8.5-8.
3. In the discussion of lateral earth pressure loads, it is stated that lateral earth effects are considered in structure sliding and overturning analyses. If the sliding resistance is the sum of the shear friction along the basemat and passive pressures induced by embedment effects, describe the contribution of each in determining the overall factor of safety against sliding. This should consider the fact that in order to develop the full passive resistance sufficient sliding deformation is required. Once sliding occurs then the full static coefficient of friction cannot be utilized.
4. How has the potential effect of saturated soils from groundwater, flood, or water infiltration from the surface been considered in all seismic soil structure interaction (SSI) analyses, overall NI structural analysis, and the second model used for bearing, sliding, and overturning calculations. This explanation should include the development of soil springs for the overall NI structure (beneath the foundation and the side walls), the brick element layer beneath the basemat in the second model, the coefficient of friction for sliding, calculation of lateral earth pressures, and other calculations.
5. If lateral earth pressure loads are needed to resist the structure sliding and overturning, presumably at the same time, provide the seismic pressure distribution used in the design of the foundation walls and compare them to the maximum calculated soil pressure load distribution from the sliding and overturning seismic analysis.
6. It is stated that justification is provided for live loads that are included in loading combinations when evaluating structures for the effects of sliding and overturning. Provide specific examples and bases for the types of live loads that are considered and the expected effect when determining the factor of safety for sliding and overturning.
7. It is stated that the effects of differential foundation settlements are applied concurrently with the dead load using the same load factors. Describe how the effects of differential foundation settlements are applied concurrently with dead load and in which load combinations these are considered.
8. It is stated that sliding distance estimates may be computed using the reserve energy approach described in ASCE/SEI 43-05 as a conservative alternate to time-history

computed sliding displacements. Explain whether this alternate approach has been used. If it has been used or it is still desired to remain as an option, then as noted in RAI 3.8.1-4, ASCE/SEI 43-05 has not been generically endorsed by the NRC. Therefore, technical justification for the use of this method should be submitted for review and approval.

Response to Question 03.08.05-4:

1. The sliding, overturning, and flotation factors of safety are calculated using results from the static model according to the following formulas (for sliding and overturning in the northerly direction):

$$\text{FS Sliding} = \frac{\left(F_{z\_dead} + F_{z\_wind}\right)\mu}{F_{y\_wind\_shear} + F_{y\_soil\_pressure\_shear}},$$

$$\text{FS Overturning} = \frac{\left(F_{z\_dead} + F_{z\_soil\_pressure} + F_{z\_wind}\right)L_a}{M_{x\_dead} + M_{x\_soil\_pressure} + M_{x\_wind}},$$

$$\text{FS Floatation} = \frac{F_{z\_dead}}{F_{z\_buoyant}},$$

   where μ is the minimum coefficient of static friction, L_a is the distance from the center of gravity to the outside edge of the NI in the direction of overturning, F_{z_dead}, for example, is the total fixed base reaction in the z-direction due to dead loading, and M_{x_dead}, for example, is the overturning moment generated by dead loads.

2. As requested by NRC, the minimum factors of safety for overturning and sliding for the load combination including E' will be provided in response to RAI 155 Question 3.8.5-8.

3. In the sliding and overturning analyses, lateral passive pressures account for less than 1 percent of the total sliding resistance. As stated in U.S. EPR FSAR Tier 2, Section 3.8.5.4.2, full passive resistance does not occur until a horizontal displacement of 1 percent of the embedded depth (i.e., 0.41 ft). The coefficient of friction used in the dynamic model, μ= 0.7, is considered to be the dynamic coefficient of friction as well as the minimum static coefficient of friction.

4. For all structural analyses of the NI building (static model, sliding and overturning, and lateral earth pressures), the submerged unit weight of the soil was used. Hydrostatic and hydrodynamic loads resulting from groundwater are also considered.

   The shear modulus used in development of soil spring is based on saturated soil conditions. Coefficient of friction for sliding is assumed to be 0.7. Confirmation of this value is a COL item as incorporated by the response to RAI Batch 130 Question 03.07.02-22.

5. Seismic soil pressure used in the design of foundation walls varies with depth with a maximum of 3.11 ksf (see Figure 03.08.05-4-1 for pressure distribution). Maximum bearing pressure from seismic sliding analysis is 34.56 ksf (see Figure 03.08.05-4-2 for bearing pressure distribution under sliding event).

6. In accordance with the acceptance criteria of SRP 3.7.2.II.3.D, 25 percent of the live load is included in the sliding and overturning analysis. Twenty-five percent of the live load corresponds to less than 5 percent of the dead load and is, therefore, not expected to have a significant effect on the factor of safety.

7. Analyses of the U.S. EPR Nuclear Island common foundation basemat were performed with and without differential settlement. The analyses considered bearing pressure and internal stresses of the basemat for soil case 1u, which is the softest (i.e., least stiff) soil case. The analyses found that there is a negligible difference in both the bearing pressures and the internal stresses of the basemat when the NI is subjected to an initial settlement of 1.0 inch per 50 ft (twice the maximum allowable differential settlement). Consequently, no effect of differential foundation settlement is applied with the dead load.

8. The reserve energy approach as described in ASCE/SEI 43-05 has not been used in the analyses of the U.S. EPR. Therefore, the U.S. EPR FSAR Tier 2, Section 3.8.5.4.1, reference paragraph will be deleted. The deleted text is as follows:

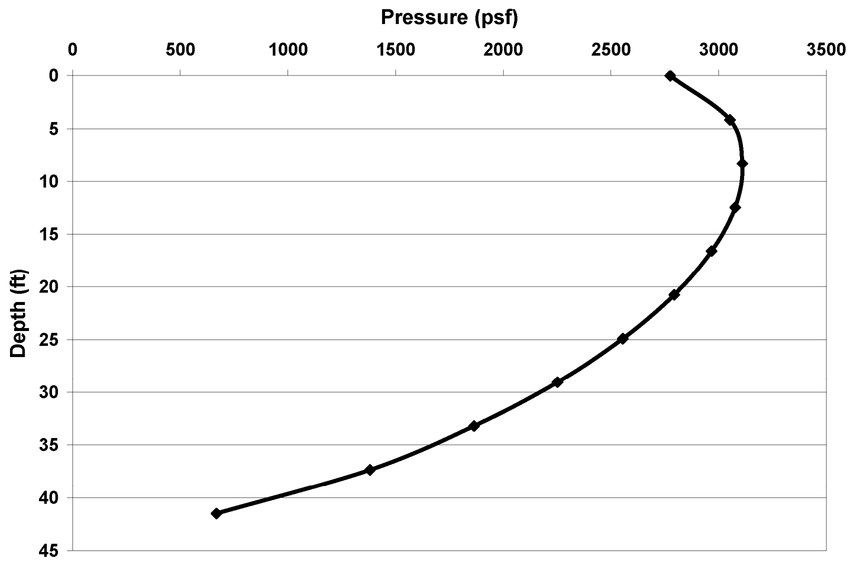
    "Sliding distance estimates may be computed using the reserve energy approach described in ASCE/SEI 43-05 as a conservative alternate to time-history computed sliding displacements."

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 3.8.5.4.1 will be revised as described in the response and indicated on the enclosed markup.

Figure 03.08.05-4-1

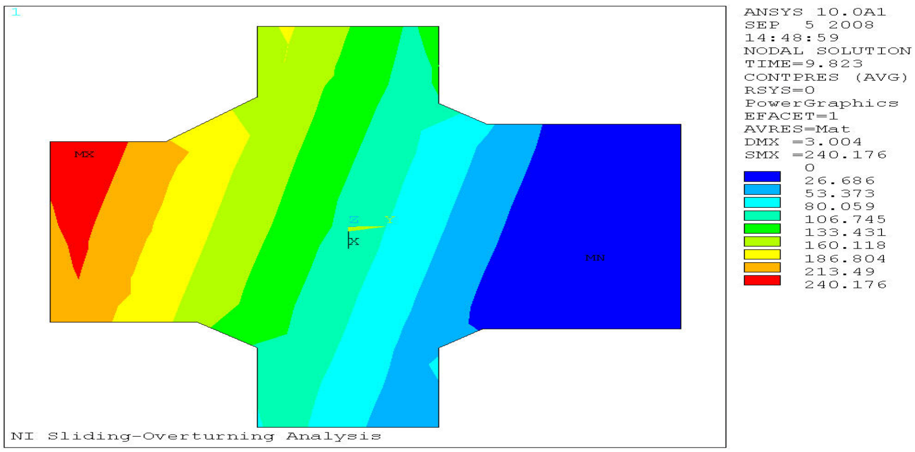
Seismic Soil Pressure Distribution used in the Design of Foundation Walls



Seismic Soil Pressure Distribution used in the Design of the Foundation Walls

Figure 03.08.05-4-2

Bearing Pressure Distribution Under Sliding Event



Question 03.08.05-9:

FSAR Section 3.8.5.4.3 for the EPGB and FSAR Section 3.8.5.4.4 for the ESWB state that elastic boundary conditions are included in the finite element model for each structure in order to simulate the stiffness of the supporting soil. As these structures are designed for an envelope of soil conditions, describe how the stiffness of the soil springs are determined for each of the soil cases and how an envelope of design loads is produced for each structure.

Response to Question 03.08.05-9:

The stiffness of the soil springs was determined using the same methodology as described in U.S. EPR FSAR Tier 2, Section 3.8.5.4.2 for the Nuclear Island (NI) basemat (i.e., using Gazetas' equations, with the static modulus taken as half of the dynamic modulus). Stiffness of soil springs was not determined for each soil case. Out of the ten soil profiles considered for the standard plant design (refer to U.S. EPR FSAR Tier 2, Table 3.7.1-6, for listing), based on the seismic response characteristics obtained from SASSI analyses, the following three representative subgroups were identified for the purpose of enveloping basemat design:

- Soft soil group (1u and 1n2u).
- Sedium soil group (2u and 2n3u).
- Hard soil group (2sn4u, 3r3u, 3u, 4u, 5a, and 5u).

An enveloped seismic response (applied as equivalent static load) was considered for each soil group. To maximize the basemat design forces and moments, the softest soil case (in terms of the magnitude of the effective static shear modulus) was considered as the controlling case for each of these groups. Controlling soil spring values were thus considered for the following three soil cases: 1u, 2u, 2sn4u. Soil case 5a was further evaluated for the Emergency Power Generating Building (EPGB) as a separate soil case from the third group as it corresponds to hard rock site, which has a very high shear modulus. The effective static modulus was determined for layered soil profile(s) using elastic half-space theory.

An elliptical distribution of vertical subgrade modulus, similar to that described in U.S. EPR FSAR Tier 2, Section 3.8.5.4.2, was determined for soil case 1u only. The following distribution for the EPGB was used (with K_o equal to 30.2 kcf):

$$K(x,z) = K_o\left(4.88 - 4.2\sqrt{1-\left(\frac{x^2+z^2}{2L^2}\right)}\right) \text{ (kcf)}$$

EPGB analysis results using elliptical subgrade modulus distribution were compared with those using a uniform distribution. The comparison showed that the uniform subgrade modulus distribution assumption resulted in larger basemat design forces and moments, while it underestimated soil bearing pressures. The bearing pressure requirements are controlled by the larger and more massive NI structure; as such, the focus for EPGB and Essential Service Water Building (ESWB) was to maximize their respective basemat design forces/moments. Based on this consideration, a uniform subgrade modulus distribution was deemed satisfactory for other soil cases.

Use of Gazetas' equation for the EPGB resulted in the following K_o values for 2u, 2sn4u, and 5a: 166.4 kcf, 289.5 kcf, and 13,944 kcf, respectively. Similarly, the use of Gazetas' equation for the ESWB resulted in the following K_o values for 1u, 2u, 2sn4u: 25.1 kcf, 137.7 kcf, and 339.2 kcf, respectively. Individual nodal soil springs were calculated as the modulus of subgrade reaction multiplied by the tributary area associated with the basemat node.

No variation in the K_o values was considered for any of the soil cases. This is considered acceptable for the following reasons:

- Conservatively enveloped (equivalent static) seismic loads were combined with the softest soil cases.
- Because of their thicknesses, the concrete basemats for both structures are essentially rigid relative to the underlying soil (except for Case 5a, which does not control). This makes the basemat insensitive to the soil spring variation.
- It typically takes an order of magnitude variation in the soil spring before appreciably affecting the basemat forces and moments. An order of magnitude variation is not credible.

In summary, an enveloped seismic response (applied as equivalent static load) was considered for each soil group. Furthermore, the softest soil cases were considered from each group in order to maximize the basemat design forces and moments. This approach thus resulted in enveloping the combined effects of seismic and static loads.

FSAR Impact:

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

Question 03.08.05-17:

FSAR Section 3E.2.1 for the EPGB foundations and FSAR Section 3E3.1 for the ESWB foundations describe the basemat typical reinforcement configurations in FSAR Figures 3E.2-3 and 3E.3-3, respectively. These figures indicate the horizontal reinforcement pattern for each foundation design, but do not indicate whether this reinforcement is in the top or bottom of the slab. Provide additional figures showing key cross sections of the slabs that indicate the size, location and spacing of the top and bottom reinforcement, as well as any vertical reinforcement. Also, please reconcile 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 03.08.05-17:

The reinforcement shown in U.S. EPR FSAR Tier 2, Figures 3E.2-3 and 3E.3-3, is required on both top and bottom faces. The information provided in the U.S. EPR FSAR is limited due to scale . The specific requested detail is provided in Figures 03.08.05-17-1 through 03.08.05-17-4.

No vertical (shear) reinforcement is required for the Emergency Power Generating Building (EPGB) foundation. The vertical (shear) reinforcement is not required for the revised Essential Service Water Building (ESWB) foundation configuration. The remaining information regarding the ESWB requested by this RAI was also part of the subject matter of RAI 130, Question 03.07.02-27. Therefore, this portion of the response 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.

FSAR Impact:

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

Figure 03.08.05-17-1

Emergency Power Generating Building Reinforcement

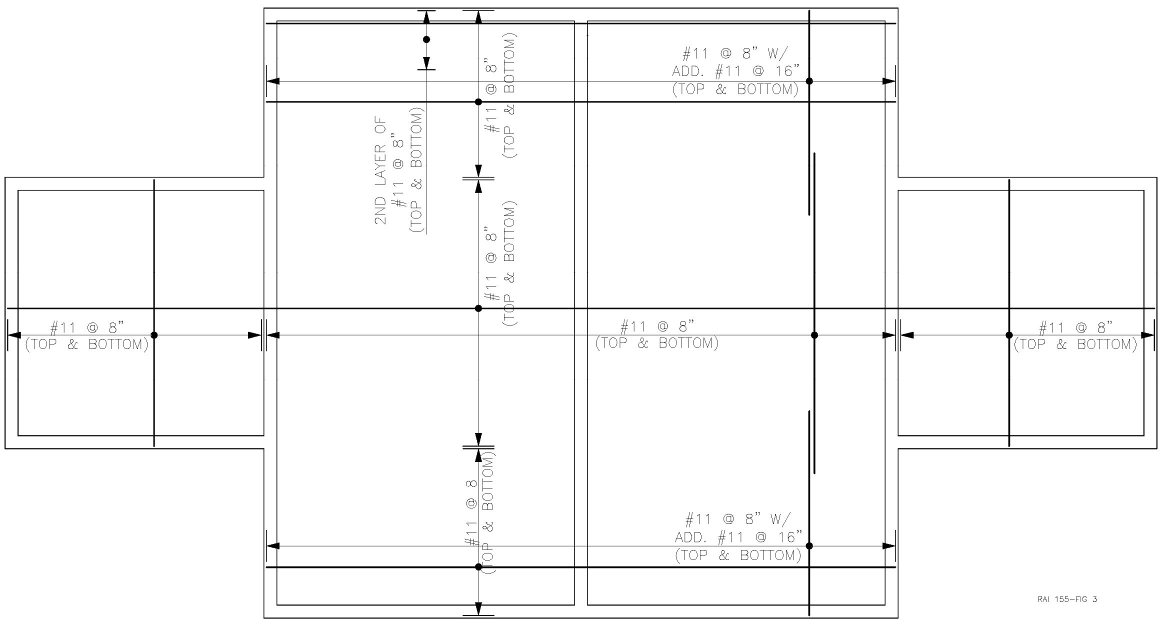


Figure 03.08.05-17-2

Essential Service Water Building Reinforcement

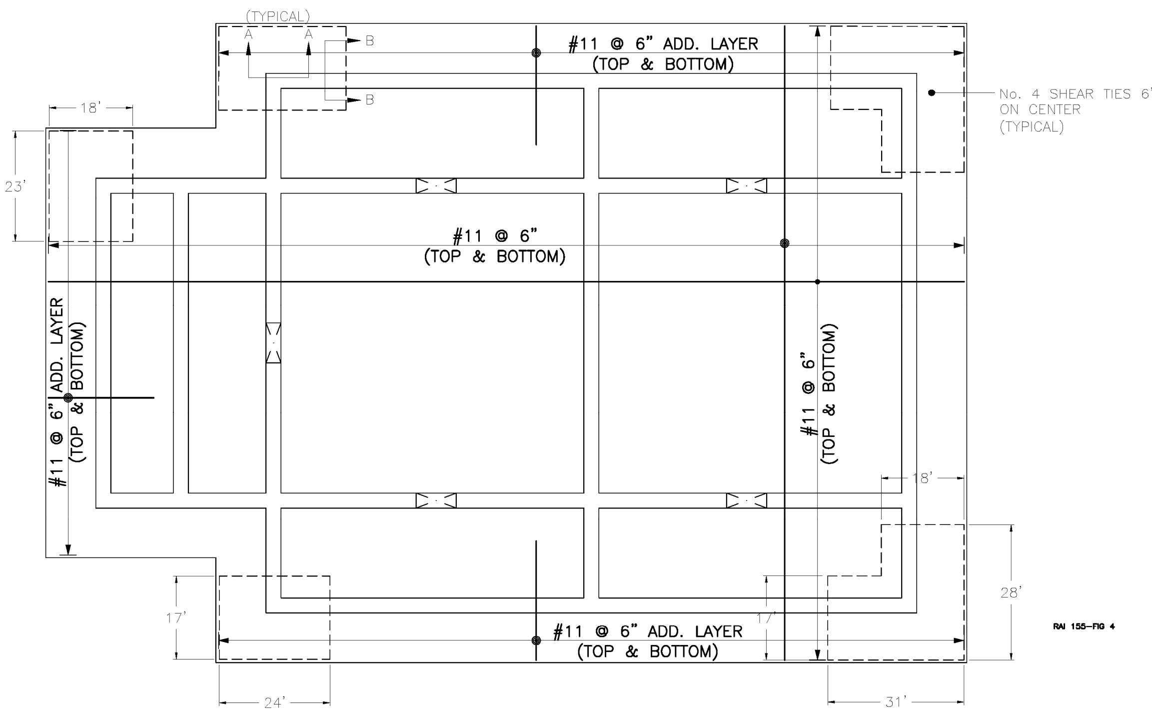
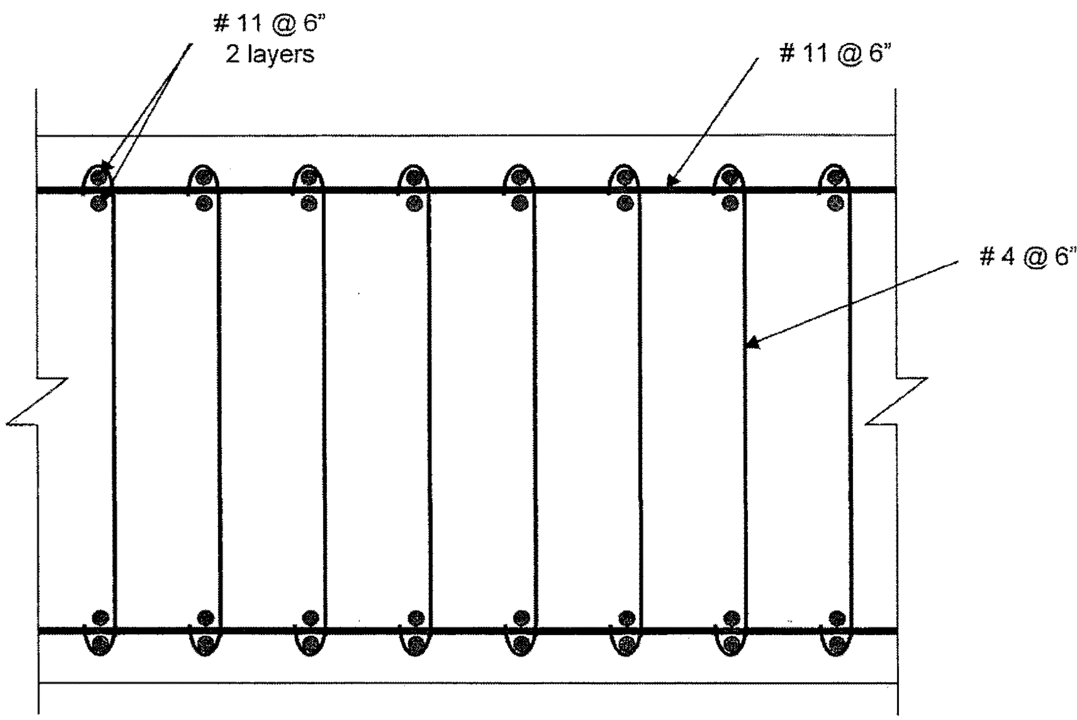


Figure 03.08.05-17-3

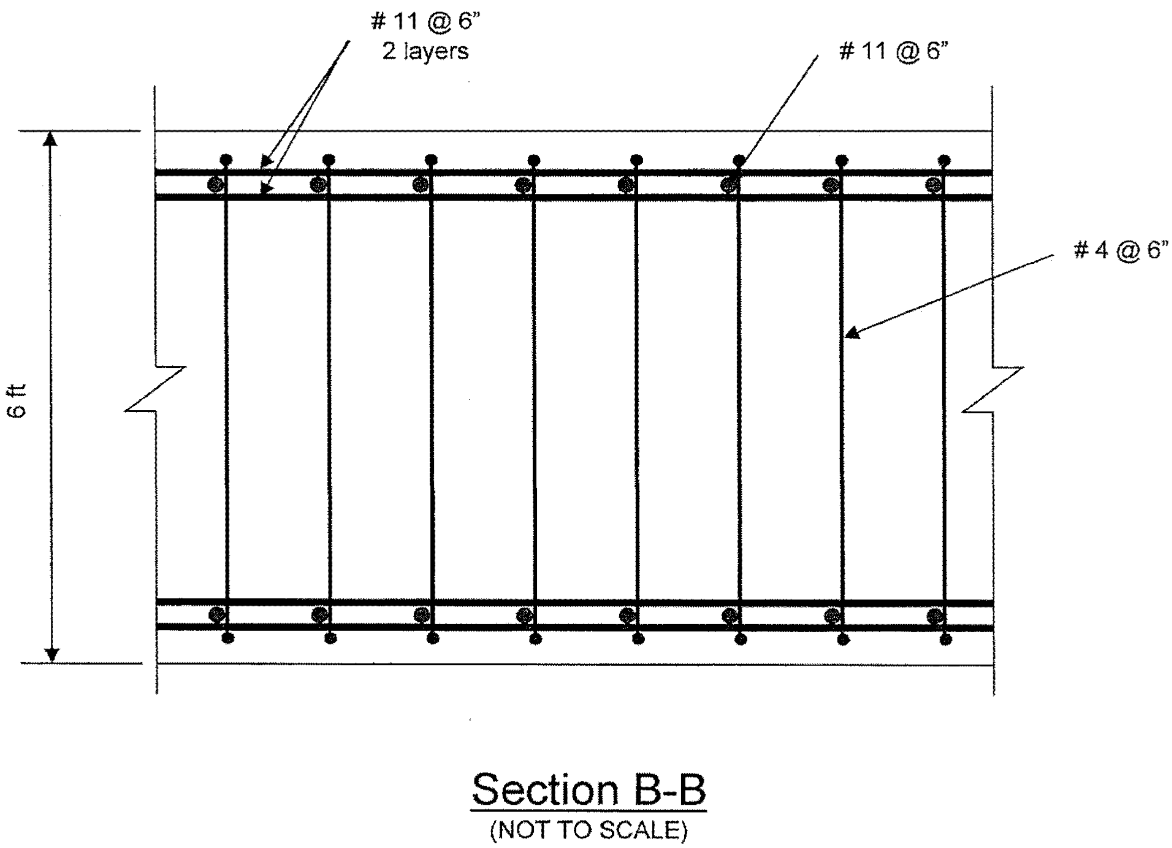
Essential Service Water Building Shear Reinforcement, Section A-A



Section A-A
(NOT TO SCALE)

Figure 03.08.05-17-4

Essential Service Water Building Shear Reinforcement, Section B-B



U.S. EPR Final Safety Analysis Report Markups

03.08.01-4

- ANSI/AISC N690-1994 (R2004), Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities, including Supplement 2 (Reference 14).
- ~~ASCE Standard 4-98, Seismic Analysis of Safety-Related Nuclear Structures and Commentary (Reference 15).~~
- ~~ASCE/SEI Standard 7-05, Minimum Design Loads for Buildings and Other Structures (Reference 16).~~
- ~~ASCE/SEI Standard 43-05, Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities (Reference 17).~~
- ANSI/AWS D1.1/D1.1M-2006, Structural Welding Code – Steel (Reference 18).
- ANSI/AWS D1.4-2005, Structural Welding Code - Reinforcing Steel (Reference 19).
- ANSI/AWS D1.6 - 1999, Structural Welding Code – Stainless Steel (Reference 20).
- ASME BPV Code - 2004 Edition.
  - Section II - Material Specifications.
  - Section III, Division 2 - Code for Concrete Reactor Vessels and Containments.
  - Section V - Nondestructive Examination.
  - Section VIII - Pressure Vessels.
  - Section IX - Welding and Brazing Qualifications.
  - Section XI – Rules for Inservice Inspection of Nuclear Power Plant Components.
- Acceptable ASME BPV Code cases per RG 1.84, Revision 33, August 2005.

03.08.01-4

- ASME NOG-1-04, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder) (Reference 21).
- ~~SEI/ASCE Standard 37-02, Design Loads on Structures During Construction (Reference 2).~~

3.8.1.2.2 Standards and Specifications

Industry standards (e.g., those published by the ASTM) are used to specify material properties, testing procedures, fabrication, and construction methods. Section 3.8.1.6 lists the applicable standards used.

– Hydrostatic Loads (F) – Hydrostatic loads due to water stored in pools and tanks are considered in the design of RB internal structures that exert reaction loads on the RCB and NI Common Basemat Structure foundation basemat. Hydrodynamic loads resulting from seismic excitation of fluids are included as a component of the safe shutdown earthquake (SSE) load. There are no hydrostatic loads from groundwater or external floods on the RCB because it is surrounded by other Seismic Category I structures that subsequently provide a shield. Buoyancy loads are addressed in Section 3.8.5 for foundation design.

– Thermal Loads (T_o) – Thermal loads consist of thermally induced forces and moments resulting from normal plant operation and environmental conditions. Thermal loads and their effect are based on the critical transient or steady-state condition. Thermal expansion loads due to axial restraint, as well as loads resulting from thermal gradients, are considered.

The ambient air temperatures listed below are for normal operation. Normal operation temperatures are given as a maximum value during summer and a minimum value during winter.

RB internal ambient temperatures:

- During normal operation:
Equipment Area: 131°F (maximum), 59°F (minimum).
Service Area: 86°F (maximum), 59°F (minimum).

- During normal shutdown: 86°F (maximum), 59°F (minimum).

RB annulus internal ambient temperatures:

- During normal operation: 113°F (maximum), 45°F (minimum).

03.08.01-5

– Pipe Reactions (R_o) – Pipe reactions are those loads applied by piping system supports during normal operating or shutdown conditions based on the critical transient or steady state conditions. The dead weight of the piping and its contents are not included. Appropriate dynamic load factors are used when applying transient loads, such as water hammers.

– Post-Tension Loads (J) – Post-tension loads are those loads developed from applying strain on the containment tendons.

– Relief Valve Loads (G) – Relief valve loads are those loads resulting from the actuation of a relief valve or other high-energy device.

– Pressure Variant Loads (P_v) – Pressure variant loads are those external pressure loads resulting from pressure variation either from inside or outside of containment.

– Construction Loads – Construction loads are those loads to which the structure may be subjected during construction of the plant. Construction loads will be applied to evaluate partially completed structures, temporary structures, and torus. However, membrane failure at the transition region is the limiting condition. The ultimate pressure capacity reported is the median pressure capacity.

Pressure capacities were evaluated for the reinforced area around the equipment hatch opening. The evaluation considered a horizontal plane and a vertical plane section passing through the centerline of the opening. The vertical plane section, which corresponds to hoop stress direction, was the weaker of the two planes. The ultimate pressure capacity reported is the median pressure capacity for the vertical plane section.

The equipment hatch cover and cylinder, shown inFigure 3.8-25—Equipment Hatch General Assembly has a cover ultimate pressure capacity based on ASME Section II, Part D material specification minimum required strengths and an elastic, perfectly plastic stress-strain relationship at 400°F. The internal pressure from containment is applied to the convex surface of the cover and non-embedded portion of the cylinder. The ultimate pressure capacity reported corresponds to ASME Service Level C stress limits for the hatch cover and cylinder.

3.8.1.4.12 Design Report

Design information and criteria for Seismic Category I structures are provided in Sections 2.0, 2.4, 2.5, 3.3, 3.5, 3.8.1, 3.8.2, 3.8.3, 3.8.4, and 3.8.5. Design results are presented in Appendix 3E for Seismic Category I structure critical sections.

3.8.1.5 Structural Acceptance Criteria

The limits for RCB allowable stresses, strains, deformations and other design criteria are in accordance with the requirements of Subsection CC-3400 of the ASME BPV Code, Section III, Division 2 and RG 1.136 (GDC 1, GDC 2, GDC 4, GDC 16, and GDC 50). This applies to the overall containment vessel and subassemblies and appurtenances that serve a pressure retaining function, except as noted in Section 3.8.2. Specifically, allowable concrete stresses for factored loadings are in accordance with Subsection CC-3420 and those for service loads are in accordance with Subsection CC-3430.

The limits for stresses and strains in the liner plate and its anchorage components are in accordance with ASME BPV Code, Section III, Division 2, Tables CC-3720-1 and CC-3730-1.

03.08.03-2

Limits for allowable loads on concrete embedments and anchors are in accordance with Appendix B of ACI-349-2006 and guidance given in RG 1.199.

Section 3.8.1.6 describes minimum requirements for concrete, reinforcing, post-tensioning tendons, and the liner plate system for the RCB.

RSB and FB concrete. Figure 3.8-31—Fuel Transfer Tube Penetration (Conceptual View) illustrates the fuel transfer tube penetration.

3.8.2.2 Applicable Codes, Standards, and Specifications

The following codes, standards, specifications, design criteria, regulations, and regulatory guides are used in the design, fabrication, construction, testing, and inservice inspection of steel portions of the RCB that are intended to resist pressure, but are not backed by structural concrete (GDC 1, GDC 2, GDC 4, GDC 16 and GDC 50).

Section 3.8.1.2 describes codes, standards, and specifications applicable to the containment steel liner.

3.8.2.2.1 Codes

- ANSI/AISC N690-1994 (R2004), Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities, including Supplement 2.
- ANSI/AWS D1.1/D1.1M-2006, Structural Welding Code – Steel.
- ANSI/AWS D1.6-1999, Structural Welding Code – Stainless Steel.

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- ~~ASCE Standard 4-98, Seismic Analysis of Safety-Related Nuclear Structures and Commentary.~~
- ASME BPV Code – 2004 Edition:
  - Section II – Material Specifications.
  - Section III, Division 1 – Nuclear Power Plant Components.
  - Section V – Nondestructive Examination.
  - Section VIII – Pressure Vessels.
  - Section IX – Welding and Brazing Qualifications.
- Acceptable ASME BPV Code cases per RG 1.84.

03.08.01-4

- ~~ASCE/SEI Standard 7-05, Minimum Design Loads for Buildings and Other Structures.~~
- ~~ASCE/SEI Standard 37-02, Design Loads on Structures During Construction.~~
- ~~SEI/ASCE Standard 43-05, Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities.~~

RG 1.57 contains load combinations which include P_{g3}, defined as pressure resulting from postaccident inerting. The U.S. EPR does not utilize a postaccident inerting hydrogen control system. Therefore, load combinations containing P_{g3} are not applicable.

Level B Service Limits

These service limit load combinations include the loads subject to Level A service limits, plus the additional loads resulting from natural phenomena during which the plant must remain operational (GDC 2, GDC 4, and GDC 50). For the load effects of the OBE, only the contribution to cyclic loading needs to be considered because the OBE is defined as one-third of the SSE. If a component screens out of an analysis for cyclic operation, based on ASME Section III, Division I, Subsection NE, Subparagraph NE-3221.5, Level B service limit load combinations may be eliminated.

03.08.02-9

$$P^* = D + L + T_o + R_o + P_v + E$$

$$P^* = D + L + T_a + R_a + P_a + E$$

Level C Service Limits

These service limit load combinations include the loads subject to Level A service limits, plus the additional loads resulting from natural phenomena for which safe shutdown of the plant is required (GDC 2, GDC 4, GDC 50).

$$P^* = D + L + T_o + R_o + P_v + E'$$

$$P^* = D + L + T_a + R_a + P_a + E'$$

$$P^* = D + P_{g1} + P_{g2.}$$

In the last load combination, $P_{g1} + P_{g2}$ should not be less than 45 psig and evaluation of instability is not required as specified by the code.

Level D Service Limits

These service limit load combinations include other applicable service limits and dynamic loads for which containment function is required (GDC 2, GDC 4, and GDC 50).

$$P^* = D + L + T_a + R_a + P_a + R_{rr} + R_{rj} + R_{rm} + E'$$

$$P^* = D + L + F_a + E.$$

- ACI 308R-01, Guide to Curing Concrete (Reference 50).
- ACI 308.1-98, Standard Specification for Curing Concrete (Reference 39).
- ACI 311.4R-05, Guide for Concrete Inspection (Reference 40).
- ACI 347-04, Guide to Formwork for Concrete.
- ACI 349-01/349-R01, Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary on Code Requirements for Nuclear Safety Related Concrete Structures (exception described in 3.8.4.4 and 3.8.4.5) (GDC 1).
- ACI 349.1R-07, Reinforced Concrete Design for Thermal Effects on Nuclear Power Plant Structures (Reference 41).
- AISC 303-05, Code of Standard Practice for Steel Buildings and Bridges (Reference 42).
- ANSI/AISC N690-1994, Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities, including Supplement 2, 2004 (GDC 1).
- ANSI/AISC 341-05, Seismic Provisions for Structural Steel Buildings, American Institute of Steel Construction, including Supplement 1 (Reference 43).

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- AISC 348-04/2004 RCSC, Specification for Structural Joints Using ASTM A325 and A490 Bolts (Reference 44).
- ~~ASCE Standard 4-98, Seismic Analysis of Safety-Related Nuclear Structures and Commentary.~~
- ~~ASCE/SEI Standard 7-05, Minimum Design Loads for Buildings and other Structures.~~
- ~~ASCE/SEI Standard 43-05, Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities.~~
- ANSI/AWS D1.1/D1.1M 2006, Structural Welding Code - Steel.
- ANSI/AWS D1.4-2005, Structural Welding Code - Reinforcing Steel.
- ANSI/AWS D1.6-1999, including January 6, 2005 update, Structural Welding Code – Stainless Steel.
- ANSI/AWS D1.8-2005, Structural Welding Code – Seismic Supplement (Reference 45).
- ASME Boiler and Pressure Vessel Code - 2004 Edition, Section III, Division 2 - Code for Concrete Reactor Vessels and Containments (GDC 1).

- ASME Boiler and Pressure Vessel Code - 2004 Edition, Section III, Division 1 – Nuclear Power Plant Components (GDC 1).

03.08.01-4

- ASME NOG-1-04, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder).
- ~~SEI/ASCE Standard 37-02, Design Loads on Structures During Construction.~~

3.8.3.2.2 Specifications

Industry standards (e.g., those published by the ASTM) are used to specify material properties, testing procedures, fabrication methods, and construction methods. Section 3.8.3.6 addresses the applicable standards used.

Structural specifications cover areas related to the design and construction of the RB internal structures. These specifications emphasize important points of the industry standards for these structures and reduce options that otherwise would be permitted by the industry standards. These specifications cover the following areas:

- Concrete material properties.
- Mixing, placing, and curing of concrete.
- Reinforcing steel and splices.
- Structural steel.
- Stainless steel liner plate and embedments.
- Miscellaneous and embedded steel.
- Anchor bolts.
- Expansion anchors.
- Polar crane.
- Miscellaneous cranes and hoists.

3.8.3.2.3 Design Criteria

- ACI 349-01/349-R01, Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary on Code Requirements for Nuclear Safety-Related Concrete Structures (GDC 1).
- ANSI/AISC N690-1994, Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities, including Supplement 2 (2004) (GDC 1).

Local analysis and design consider the same member and element forces and moments as described for overall design. In addition, local effects (e.g., punching shear and transfer of anchorage loads to the structure) are considered. Local analyses also are used for design of secondary structures (e.g., platforms, equipment supports, crane supports).

The recommendations of ACI 349-2001 and its appendices, including the exceptions in RG 1.142, are followed for concrete element and member local design (GDC 1).

Design of concrete embedments and anchors conforms to Appendix B of ACI 349-2001 and guidelines of RG 1.199. Ductility is provided by designing anchorage systems so that a steel failure mode controls the design.

ANSI/AISC N690-1994 (R2004), including Supplement 2, and ANSI/AISC 341-05, are followed for local steel member design (GDC 1).

The design of bolted connections is in accordance with ANSI/AISC N690, Section Q1.16 and AISC 348-04/2004 RCSC. Bolted in connections are fully tensioned, regardless of design methodology, unless justified otherwise.

The design of welded connections is in accordance with ANSI/AWS D1.1/D1.1M 2006 and ANSI/AWS D1.6-99, including January 6, 2005 update.

The design of bolted connections in combination with welded connections is in accordance with Section Q.15.10 of ANSI/AISC N690.

03.08.03-9

Openings in walls and slabs of RB internal structures are shown on construction drawings. Openings in slabs are acceptable without analysis if they meet the criteria identified in ACI 349, Section 13.4.2. Round pipe sleeves are used in lieu of rectangular penetrations, where possible. Corners of rectangular openings in walls or slabs are provided with diagonal reinforcing to reduce cracking due to stress concentrations at these locations in accordance with ACI 349, Section 14.3.7.

Appendix 3E provides a description of analysis and design results for critical areas of the RB internal structures.

Section 5.4.14 describes the design of interfacing steel assemblies which support the NSSS components and attach to, or interact with, embedments in the concrete. Steel supports for the RCS components and piping, including the base plates at the face of concrete structures, are designed in accordance with ASME Section III Division 1, Subsection NF. Embedded portions of RCS component and pipe supports, which are beyond the jurisdictional boundary of the ASME Code, are designed in accordance with ACI 349-2001, including Appendix B, and also in accordance with ANSI/AISC N690-1994 (R2004), including Supplement 2.

Structure Type	Percent of Critical Damping
• Welded Steel	4
• Bolted Steel, Slip-Critical Connections	4
• Bolted Steel, Bearing Connections	7
• Reinforced Concrete	7

Hydrodynamic Load Analyses

03.08.01-4

Hydrodynamic loads are applied to the IRWST and refueling canal walls and floors to account for the impulsive and impactive effects of water moving and sloshing in the tank as a result of seismic excitation. These loads are considered as part of the seismic SSE loads, and components of these loads in the three orthogonal directions are combined in the same manner as other seismic loads. Methodology consistent with ASCE Standard 4-98 and USAEC TID-702.4 are~~is~~ used to determine hydrodynamic loadings. The effect of tank structure flexibility on spectral acceleration is included when determining the hydrodynamic pressure on the tank walls for the impulsive mode.

Design for hydrodynamic loads is within the elastic range of concrete and steel members and elements.

Polar Crane Seismic Analyses

Design of the RCB for seismic loads from the polar crane is performed with the crane in positions that result in maximum stresses on the supporting containment wall. See Section 3.8.1 for additional information on the design of the RCB.

For seismic load combinations, the polar crane design is based on the trolley being located in different positions along the bridge girders. Seismic evaluations are performed with and without the critical load raised to different positions for the trolley locations to determine which hook position produces the primary response of the crane. For analysis purposes, the critical load is defined as that of the reactor head. The design of the crane includes seismic restraints (up-kick lugs), which prevent the bridge and trolley from dislodging from their respective rails.

Refer to Section 9.1.5 for additional information on the polar crane.

Pipe Rupture Loads

Local analyses of the RB internal structures consider the following abnormal loads:

- Sub-compartment pressure loads (P_a).

static equivalent of these loads. Elasto-plastic behavior may be assumed with appropriate ductility ratios, provided that excessive deflections do not result in the loss of function of any safety-related SSC. Appendix C of ACI 349-2001 is used to determine pipe break reactions, jet impingement, and missile impact impulsive and impactive loads. The design of the RB internal structures for these loads conforms to the procedures described in Section 3.5 for internally generated missiles. Section 3.5 also describes ductility limits that are met for impactive and impulsive loadings.

Local flood loads (F_a) are applied to walls and floors of the RB internal structures in the overall ANSYS computer model. Concrete and steel members are designed to accommodate these flood loads within the elastic range of their section strength.

3.8.3.4.5 Design Report

Design information and criteria for Seismic Category I structures are provided in Sections 2.0, 2.4, 2.5, 3.3, 3.5, 3.8.1, 3.8.2, 3.8.3, 3.8.4, and 3.8.5. Design results are presented in Appendix 3E for Seismic Category I structure critical sections.

3.8.3.5 Structural Acceptance Criteria

Limits for allowable stresses, strains, deformations, and other design criteria for reinforced concrete RB internal structures are in accordance with ACI 349-2001, and its appendices, including the exceptions specified in RG 1.142, with the exception that the shear strength reduction factor of 0.85 is used as allowed in ACI 349-2006. The exceptions specified in RG 1.142 (GDC 1, GDC 2, GDC 4 and GDC 50) are considered.

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Limits for allowable loads on concrete embedments and anchors are in accordance with Appendix B of ACI 349-200~~1~~6 and guidance given in RG 1.199.

Limits for the allowable stresses, strains, deformations and other design criteria for structural steel RB internal structures are in accordance with ANSI/AISC N690-1994, including Supplement 2 (GDC 1, GDC 2, GDC 4 and GDC 50).

Limits for allowable stresses, strains, and deformations on steel RCS component and pipe supports, including the base plates for these supports at the face of concrete structures, are in accordance with ASME Section III Division 1, Subsection NF.

The design of RB internal structures is generally controlled by load combinations containing SSE seismic loads. Stresses and strains are within the ACI 349-2001 and ANSI/AISC N690-1994 limits.

Appendix 3E provides design results for critical areas of the RB internal structures.

- ACI 311.4R-05 - Guide for Concrete Inspection (Reference 40).
- ACI 347-04 - Guide to Formwork for Concrete.
- ACI 349-01/349-R01 - Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary on Code Requirements for Nuclear Safety Related Concrete Structures (exception described in 3.8.4.4 and 3.8.4.5) (GDC 1).
- ACI 349.1R-07 - Reinforced Concrete Design for Thermal Effects on Nuclear Power Plant Structures.
- ACI 350-06 - Code Requirements for Environmental Engineering Concrete Structure (Reference 58).
- ACI 350.3-06 - Seismic Design of Liquid-Containing Concrete Structures (Reference 59).
- AISC 303-05 - Code of Standard Practice for Steel Buildings and Bridges.
- ANSI/AISC N690-1994 - Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities, including Supplement 2 (2004) (GDC 1).
- ANSI/AISC 341-05 - Seismic Provisions for Structural Steel Buildings, American Institute of Steel Construction, including Supplement 1.
- ANSI/ANS-6.4-2006 - Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants (Reference 4).
- AISC 348-04/2004 RCSC - Specification for Structural Joints Using ASTM A325 and A490 Bolts.
- ANSI/AWS D1.1/D1.1M 2006 - Structural Welding Code – Steel.
- ANSI/AWS D1.4-2005 - Structural Welding Code - Reinforcing Steel.
- ANSI/AWS D1.6-99, including January 6, 2005 update - Structural Welding Code – Stainless Steel.

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- ANSI/AWS D1.8 2005 - Structural Welding Code – Seismic Supplement.
- ~~ASCE Standard 4-98 - Seismic Analysis of Safety-Related Nuclear Structures and Commentary.~~
- ASME BPV Code - 2004 Edition, Section III, Division 2 – Code for Concrete Reactor Vessels and Containments.
- ASME NOG-1-2004 - Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girders).

- ASME B31.3 - 1996 - Process Piping, American Society of Mechanical Engineers (Reference 60).
- ASME B31.4 - 1992 - Liquid Transportation System for Hydrocarbon, Liquid Petroleum Gas, Anhydrous Ammonia, and Alcohols (Reference 61).

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- ASME B31.8 - 1995 - Gas Transportation and Distribution Piping Systems.
- ~~ASCE/SEI Standard 7-05 – Minimum Design Loads for Buildings and Other Structures (Reference 62).~~
- ~~ASCE/SEI Standard 43-05 – Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities.~~
- ~~SEI/ASCE Standard 37-02 – Design Loads on Structures During Construction.~~

3.8.4.2.2 Specifications

Industry standards (e.g., those published by the ASTM) are used to specify material properties, testing procedures, fabrication methods, and construction methods.

Structural specifications cover areas related to the design and construction of other Seismic Category I structures. These specifications emphasize important points of the industry standards for these structures and reduce options that would otherwise be permitted by the industry standards. These specifications cover the following areas:

- Concrete material properties.
- Mixing, placing, and curing of concrete.
- Reinforcing steel and splices.
- Structural steel.
- Steel liner plate and embedments.
- Miscellaneous and embedded steel.
- Anchor bolts.
- Expansion anchors.
- Cranes and hoists.

3.8.4.2.3 Design Criteria

- ACI 349-01/349-R01 - Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary on Code Requirements for Nuclear Safety Related Concrete Structures (GDC 1).

Design and analysis procedures for Seismic Category I foundations are the same as those described in Sections 3.8.1.4 and 3.8.4.4 for the respective structures that apply loads on the foundations.

Seismic Category I concrete foundations are designed in accordance with ACI 349-01 and its appendices (GDC 1). Exceptions to code requirements specified in RG 1.142 are incorporated into the design and are accommodated in the loading combinations described in Section 3.8.5.3. In addition, the portion of the NI Common Basemat Structure foundation basemat that supports the RCB is designed in accordance with the ASME BPV Code–2004 Edition, Section III, Division 2 for support and anchorage of the concrete RCB as described in Section 3.8.1.

The design of concrete foundations for Seismic Category I structures is performed using the strength-design methods described in ACI 349-01, with the exception that a shear reduction factor of 0.85 is used as allowed in ACI 349-06 (Reference 39). The ductility provisions of ACI 349-01 are satisfied to provide a steel reinforcing failure mode and to prevent concrete failure for design basis loadings.

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~~Design of steel structures used for Seismic Category I foundations is performed in accordance with ANSI/AISC N690-1994 (R2004), including Supplement 2 (GDC 1). Steel member design uses the allowable stress design methods of ANSI/AISC N690.~~

Foundation design is performed for the spectrum of soil cases described in Section 3.7.1. Section 2.5 and Section 3.7 describe seismic parameters and design methods used for analyzing and designing Seismic Category I structures.

Soil-structure interaction and structure-soil-structure interaction effects are considered in the seismic analyses of Seismic Category I structures as described in Section 3.7.2. Figure 3B-1 illustrates separation distances between Seismic Category I structures upon which these interaction evaluations are based.

The NI Common Basemat Structure is designed for an average static soil bearing pressure of 14,500 pounds per square foot and a maximum static bearing pressure of 22,000 pounds per square foot. Accordingly, Seismic Category I foundations are sized and reinforced to accommodate these bearing pressure values.

The following criteria apply for load combinations for concrete and steel Seismic Category I foundations:

- The one-third increase in allowable stresses for concrete and steel members due to seismic (E') or wind (W and W_t) loadings is not permitted.
- Where any load reduces the effects of other loads, the corresponding coefficient for that load is 0.9 if it can be demonstrated that the load is always present or occurs simultaneously with other loads.

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When the effects of vertical seismic acceleration are included in the stability check against sliding, the unfactored dead weight of the structure is used to calculate the resistance to sliding due to friction. ~~Sliding distance estimates may be computed using the reserve energy approach described in ASCE /SEI 43-05 as a conservative alternate to time-history computed sliding displacements.~~

Buoyancy effects of saturated soil due to a groundwater level of elevation -3.3 feet below finished grade or to a flood water level of elevation -1.0 feet below finished grade are considered when performing sliding and overturning analyses. For uplift evaluations (i.e., flotation and seismic overturning), dead load includes the weight of water permanently stored in pools and tanks. Justification is provided for live loads that are included in loading combinations when evaluating structures for the effects of sliding and overturning.

The effects of differential foundation settlements are applied concurrently with the dead load using the same load factors. Also, the effects of varying settlements between adjacent foundations are considered for the design of mechanical and electrical systems (e.g., piping, cables) that are routed between structures founded on separate basemats. See Section 3.8.4.4.5 for analysis and design procedures for Seismic Category I buried items that interface with structures on separate foundations.

3.8.5.4.2 Nuclear Island Common Basemat Structure Foundation Basemat

The NI Common Basemat Structure foundation basemat is analyzed and designed using the ANSYS V10.0 SP1 finite element overall computer model (a static model) for NI Common Basemat Structure Seismic Category I structures, which is described in Section 3.8.1.4.1. The NI Common Basemat Structure model includes the RCB, RB internal structures, RSB, FB, and SBs, as well as the NI Common Basemat Structure foundation basemat. This model is also used to determine the static bearing pressure on the supporting soils. A second model (a dynamic model) is used to determine dynamic soil bearing pressures as well as sliding and overturning factors of safety.

ANSYS SOLID45 solid elements are used to model the concrete basemat foundation in the NI Common Basemat Structure static analysis. SOLID45 is a three-dimensional, eight-node element that is suitable for moderately thick structures. Depending on the thickness of the basemat, between three to five layers of SOLID45 elements are used in the model, with an average of four elements in the typical 10 feet thick basemat areas. Figure 3.8-103—Nuclear Island Common Basemat Structure Foundation Basemat ANSYS Model illustrates the model used for design of the basemat.

Springs are used to represent soil that provides support for the concrete foundation basemat in the ANSYS model. These springs represent the compressibility of the soil and were developed to reflect the pressure distribution under the NI Common Basemat Structure. Springs values vary for each soil case based on the soil properties walls, slabs, and beams) and the foundation basemat. Analysis of the ESWB includes all applicable design loads and design load combinations described in Section 3.8.4.3. Figure 3.8-105—Essential Service Water Building Foundation Basemat Model illustrates the foundation basemat portion of the overall ESWB finite element model.

The GT STRUDL finite element model representing the ESWB foundation basemat consists of SBHQ6 rectangular elements, each with six degrees of freedom. This element type is capable of capturing both in-plane and out-of-plane behavior. Elastic boundary conditions are included in the finite element model in order to simulate the stiffness of the supporting soil. Basemat flexibility and SSI are addressed by inclusion of the basemat section properties and aforementioned soil spring boundary conditions in the finite element model. Illustrations of the complete finite element model representing the ESWB are provided in Section 3.7.2.

Detailed analysis and design procedures are described in the critical sections presented in Appendix 3E for the ESWBs.

3.8.5.4.5 Design Report

Design information and criteria for Seismic Category I structures are provided in Sections 2.0, 2.4, 2.5, 3.3, 3.5, 3.8.1, 3.8.2, 3.8.3, 3.8.4, and 3.8.5. Design results are presented in Appendix 3E for Seismic Category I structure critical sections.

3.8.5.5 Structural Acceptance Criteria

Limits for allowable stresses, strains, deformations, and other design criteria for Seismic Category I concrete foundations are in accordance with ACI 349-01 and its appendices, with the exception that the shear reduction factor of 0.85 is used as allowed in ACI 349-06 (GDC 1, GDC 2 and GDC 4). Limits for concrete design include the exceptions specified in RG 1.142. In addition, the portion of the NI Common Basemat Structure foundation basemat that supports the RCB is in accordance with the ASME BPV Code and RG 1.1.36 for containment loadings as described in Section 3.8.1.

Limits for the allowable stresses, strains, deformations, and other design criteria for structural steel elements of Seismic Category I foundations are in accordance with ANSI/AISC N690-1994 (R2004), including Supplement 2 (GDC 1, GDC 2 and GDC 4).

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The design of Seismic Category I foundations is generally controlled by load combinations containing SSE seismic loads. Stresses and strains are within the ACI 349-01 limits, with the exceptions previously listed. Limits for allowable loads on concrete embedments and anchors are in accordance with Appendix B of ACI 349-2006 and guidance given in RG 1.199. Portions of the NI Common Basemat Structure foundation basemat that support the RCB are within the limits in accordance with ASME BPV Code, Section III, Division 2.

Table 3E.1-1—Independent Loads Considered in the FEM

D	Dead Loads
L	Live Loads
J	Post-tensioning Loads
H	Lateral Earth Pressure Loads
F	Hydrostatic Loads
F_b	Buoyancy Loads
E’	Seismic Loads
R_o	Piping Loads (normal operating conditions)
R_a	Piping Loads (accident conditions)
W	Wind Loads (severe environmental)
W_t	Wind Loads (extreme environmental)
P_t	Pressure Loads (test conditions)
P_a (only for containment wall)	Pressure Loads (accident conditions)
T_a (only for containment wall)	Temperature Loads (accidental conditions)
C	Combustible Gas

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Table 3E.1-2—Independent Loads Not Considered in the FEM

G	Relief Valve Loads
R_r	Pipe Rupture Loads
F_a	Compartment Flood Loads
T_o	Temperature Loads (normal operating)
T_t	Temperature Loads (test conditions)
P_v	Containment Wall Pressure Variant Loads
P_a	Sub-compartment pressurization
CL	Construction Loads

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