Response to

Request for Additional Information No. 82
(1082, 1096, 1107, 1113, 1125, 1151, 1098, 1097), Revision 0

10/03/2008

U. S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 06.02.01 - Containment Functional Design
SRP Section: 06.02.01.02 - Subcompartment Analysis
SRP Section: 06.02.01.03 - Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs)
SRP Section: 06.02.01.04 - Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures
SRP Section: 06.02.01.05 - Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies
SRP Section: 06.02.02 - Containment Heat Removal Systems
SRP Section: 06.02.04 - Containment Isolation System
SRP Section: 06.02.06 - Containment Leakage Testing
Application Section: FSAR Ch. 6
SPCV Branch

Question 06.02.01-12:

RAI 6.2.1.1-1

a. Steam Line Break Calculations (FSAR Section 6.2.1.1.3)

1. Only one power level (50%) was investigated in the large span between 20% initial power level and 80% initial power level. This analysis at 50% initial power level produced the limiting temperature and pressure for the design of the U.S. EPR containment. Perhaps the peak containment pressure and temperature lie at an intermediate power level. Provide additional analysis for double ended main steam line breaks at intermediate power levels so that the power level producing the most severe containment results may be identified.

2. Section 6.2.1.4.1.3 states that emergency feedwater flow to the affected steam generator is assumed to be terminated 30 minutes (1800 seconds) after the break by the plant operators. Figures 6.2.1-34 and 6.2.1-35 provide containment pressure and temperature analyses for only 500 seconds. Since there are no active safety systems to provide containment atmospheric cooling at EPR, extend the containment analysis until steam flow from the postulated main steam line break is terminated.

3. For the spectrum of main steam line breaks analyzed, the calculated containment vapor temperature for some cases exceeded the specified containment design temperature of 338°F. Explain why exceeding the design temperature is acceptable. Provide appropriate COL interface requirements (COL Information Item) for instrumentation within the containment so that adequately qualified equipment may be installed.

b. Negative Containment Pressure Analysis (FSAR Section 6.2.1.1.1)

1. Section 6.2.1.1.1 lists 5 potential events which cause negative pressure across the containment wall. So that the staff may perform a review, provide a complete description of the calculation which was performed in each case including the assumptions and justification that the assumptions and methodology are conservative for containment analysis. For example for the post accident cooldown scenario, the leakage of air from the containment before isolation should be evaluated and the details of the evaluation should be described in the response.

2. A sudden containment temperature reduction is said to produce the largest negative pressure of 2.92 psi which is said to be within the external design of the building. Provide the maximum negative differential pressure that would be within the structural design of the reactor building and provide reference to the FSAR section where the structural design is described.

c. Containment Atmospheric Mixing and Heat Transfer Modeling (FSAR Section 6.2.1.1.3)

1. Describe and justify the heat transfer correlations that are used with the GOTHIC containment building model to describe heat transfer to the containment heat structures following a LOCA. For both LOCA and MSLB calculations describe and

justify the differences in assumptions for heat transfer coefficients between vertical and horizontal surfaces within containment.

2. Provide an analysis of IRWST pool stratification following a large break LOCA and include the following information.

   i. Justify that the assumptions made for pool surface temperature in calculations of atmospheric heat transfer to the pool are conservative.

   ii. FSAR Figures 3.8.2 and 6.3-5 appear to show a vertical partition bisecting the IRWST. The IRWST drawings in ANP-10293 do not appear to show such a partition. Describe the function of the partition and its effect on IRWST mixing.

   iii. FSAR Figures 3.8-11, 3.8.12 and 3.8.13 seem to show that the section of ceiling over the IRWST which is under the pressurizer is about 3 feet lower than the rest of the IRWST ceiling. Discuss the effect of the lowered ceiling area on heat transfer to the IRWST surface in particular for raised post-accident IRWST water levels.

3. The Containment Building is separated into a central portion containing the reactor system and a peripheral lower temperature portion containing equipment. Separation is accomplished by compartment walls, foils, doors, and dampers. The foils are located above the steam generator compartments and are designed to open at a fraction of a psi. The doors and dampers located at lower elevations and must also open to avoid stratification so that steam flowing to the containment dome can circulate down the containment walls to reach the heat structures an the containment lower elevations. The doors and dampers are designed to open at various pressures from a few psi to greater than 13 psi. The staff is concerned that the foils above the reactor system will open and cause pressure to be equalized throughout the containment building. With the pressure equalized the doors and dampers needed to promote circulation and prevent stratification may not open. Provide justification that sufficient compartment dampers and doors will open and to discuss impact on containment circulation if only a portion of the dampers and doors are open following a LOCA or a main steam line break accident.

   1. Describe the testing program by which the opening characteristics of the foils, doors and dampers assumed in the analyses will be verified.

   2. In the absence of containment atmospheric sprays and fan coolers, the containment internal heat structures (heat sinks) play a vital role in removing steam from the containment atmosphere following a high energy line break within containment. The expected heat sink inventory is given in FSAR Table 6.2.1.5. Describe the pre-operational inspections which will be performed to ensure that the heat sinks given in Table 6.2.1.5 are present in the as built plant.

   3. Section 6.2.2 of the FSAR contends that long-term hydrogen mixing experiments at the Battelle Model Containment (BMC) facility show that adequate containment mixing will occur under post-LOCA conditions at EPR. At BMC flashing of superheated liquid in the containment sump was reported to be the agent for containment mixing. FSAR Section 6.2.1.1.3 describes how following a LOCA subcooled water spills out of the postulated break on to the heavy floor

and into the IRWST promoting steam condensation. The staff does not understand how the same water source can provide both heating and cooling. Describe this process in greater detail and provide justification that the processes which occurred at the test facility will occur at EPR. Provide a scaling analysis of the BMC and EPR containments to demonstrate that it is appropriate to apply BMC test results to the EPR.

d. Containment Compartments and Flow Paths (FSAR Section 6.2.1.1.2)

1. Additional Flow Paths

From examining FSAR Figures 3.8-1 through 3.8-13, the NRC staff is concerned that significant flow paths might have been omitted form Table 6.2.1-07-3. For example: The vertical grating openings from UJA rooms 15-003 to 18-003 (elevation +30.77 ft) and from 23-003 to 29-003 (elev +64.8 ft) are included in Table 6.2.1-07. There should also be openings from room 11-003 to 15-003 (elev +17 ft) and from 18-003 to 23-003 (elev +45 ft) because the steam generator rooms form a vertical stack. We believe that the flow paths described in the attached Tables 1 and 2 may exist. Provide data for elevation, opening type and area for these flow paths or provide justification that the flow paths do not exist or are insignificant. For initially closed doors, flaps and dampers provide the differential pressure required to open.

2. Room volumes:

The Reactor Building rooms of US-EPR are identified in FSAR Figures 3.8-1 through 3.8-13. Table 6.2.1-07-02 of RAI 6.2.1 lists the elevation and free volumes of these rooms. The staff could not find UJA rooms 15-026, 15-027, 18-026, 18-027, 23-026, 23-027, 29-025 and 29-026 from the Chapter 3.8 figures on the table. Does Areva believe that these rooms will not affect the results from multi-noded containment analyses of design basis accidents? If not, provide information for these rooms similar to that of Table 6.2.1-07-02 including the associated containment heat structures. Otherwise the staff will leave them out of the multi-noded containment model which we are building.

3. Opening direction of doors:

The pressure differentials required to open the doors between the UJA rooms in FSAR figures 3.8-1 through 3.8-13 are identified in Table 6.2.1-07-03 of RAI 6.2.1. Should the staff assume that the doors are able to open only to positive direction (one-way opening), or should we model the doors as opening to both directions? If the doors are capable of opening in the reverse opening direction provide the reverse opening pressures.

Table 1. Continuously open connections.

From room	**To room**
07017	04002
04003	07016
04004	07019
04005	07022

04006 07023
04012 07012
07012 11012
07013 11020
07017 11022
07018 07014
07020 07023
07020 07028
07021 07027
07022 07021
07023 11024
07024 07020
07024 07022
07024 07027
07026 07023
07026 11024
07028 07027
07028 07027
07028 11023
07029 07026
07029 07028
07029 11024
11002 11003
11002 15002
11003 15003
11004 11005
11004 15004
11005 15005
11006 11007
11006 15006
11007 15007
11008 11009
11008 15008
11009 15009
11010 15010
11012 15012
11013 15013
11014 11013
11014 15014
11015 11016
11015 15015
11016 15016
11019 15019
11019 15018
11021 15025
11021 11002
11021 11023
11021 07021
11022 11023

11023 07028
11024 11009
11024 11023
11031 11025
11031 11026
11032 11027
11032 11028
15001 15017
15010 18010
15011 18011
15012 18012
15013 15014
15013 18013
15014 18014
15015 15016
15015 18015
15016 18016
15018 15019
15018 18018
15019 18019
15020 15021
15023 29013
15026 18026
15027 18027
18002 18003
18002 23002
18003 23003
18004 18005
18004 23004
18005 23005
18006 23006
18007 18006
18007 23007
18008 23008
18009 18008
18009 23009
18010 23010
18011 23011
18012 23012
18013 18014
18013 23013
18014 23014
18015 18016
18015 23015
18016 23016
18018 18019
23002 23003
23004 23005
23006 23007

23009 23008
23010 29023
23011 29011
23012 29012
23013 23014
23014 29023
23015 23016
23017 11003
23018 11008
23019 29019
23042 23014
29003 29004
29003 34003
29004 29005
29004 34004
29005 34005
29006 34006
29007 29006
29007 34007
29008 29007
29008 34008
29011 34011
29012 34012
29013 40001
29016 40001
29013 29016
29014 29018
29014 34014
29015 34015
29019 34019
34003 34004
34003 23017
34004 34005
34007 34006
34008 34007
18025 18013
15014 15026
15015 15027
34020 29020
34021 29021
34018 23018

Table 2. Doors.

From room To room
07019 07018
07026 07023
07027 07022
07028 07024

11019 11018
11021 11013
11031 11014
11032 11015
15014 15026
15015 15027
15020 15013
15024 15001
15025 15013
15026 15014
15027 15015
18002 18025
18014 18026
18015 18027
18025 18013
23002 23020
23009 23031
23014 23026
23015 23027
23017 23026
23042 23014
29022 29015
34014 34018
34020 34014

34021 34015

Response to Question 06.02.01-12:

a. Steam Line Break Calculations (FSAR Section 6.2.1.1.3)

1. A response to this question will be provided by May 22, 2009.
2. A response to this question will be provided by May 22, 2009.
3. A response to this question will be provided by May 22, 2009.

b. Negative Containment Pressure Analysis (FSAR Section 6.2.1.1.1)

1. A response to this question will be provided by December 19, 2008.
2. The U.S. EPR FSAR, Tier 2, Section 3.8.1.1 states that the Reactor Containment Building (RCB) is designed for a negative internal pressure of -3.0 psig. The structural design of the RCB is described in U.S. EPR FSAR, Tier 2, Section 3.8.1 and Section 3E.1 of Appendix 3E.

c. Containment Atmospheric Mixing and Heat Transfer Modeling (FSAR Section 6.2.1.1.3)

1. A response to this question will be provided by January 28, 2009.

   AREVA NP Inc. will provide responses to a number of these RAI questions in a comprehensive Technical Report that will provide the basis for the containment pressure and temperature response modeling of the U.S. EPR following postulated loss of coolant accidents (LOCA). The Technical Report will assess the principal containment heat removal phenomena to demonstrate conformance with design and regulatory criteria. The RAI questions that will be answered in the Technical Report are:

6.2.1.1-1.c.1
6.2.1.1-1.c.2
6.2.1.1-1.c.3 (not including 6.2.1.1-1.c.3-1 and 6.2.1.1-1.c.3-2)
6.2.1.1-1.c.3-3
6.2.1.3-1.c
6.2.1.3-1.d
6.2.1.3-1.f
6.2.1.3-1.i
6.2.1.3-1.j
6.2.1.3-1.k
6.2.1.3-1.l
6.2.1.3-1.m
6.2.1.3-1.n
6.2.1.3-1.o

2. A response to this question will be provided in a Technical Report by January 28, 2009. See the response to Question 6.2.1.1-1.c.1 for additional details.

3. A response to this question will be provided in a Technical Report by January 28, 2009. See the response to Question 6.2.1.1-1.c.1 for additional details.

3-1. A response to this question will be provided by May 22, 2009.

3-2. A response to this question will be provided by May 22, 2009.

3-3. A response to this question will be provided in a Technical Report by January 28, 2009. See the response to Question 6.2.1.1-1.c.1 for additional details.

d. Containment Compartments and Flow Paths (FSAR Section 6.2.1.1.2)

1. A response to this question will be provided by May 22, 2009.
2. A response to this question will be provided by May 22, 2009.
3. A response to this question will be provided by May 22, 2009.

FSAR Impact:

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

Question 06.02.01.02-1:

a. Conservativeness of Differential Pressure Calculations (Relates to SRP Section 6.2.1.2)

Provide the following information concerning the subcompartment differential pressure calculations:

1. Provide additional justification that the use of the homogeneous equilibrium model (HEM) is conservative for the prediction of break flow for subcompartment analysis. The response to RAI 6.2.1-08 states that the results of an EPRI study concludes that for L/D>1.5, HEM shows good agreement with test data. Provide a comparison of the L/D for the assumed breaks in the EPR subcompartment analyses with those of the test series to show that the postulated break locations for EPR fall with the range of data from which the EPRI observations were made and show that use of HEM is conservative.

2. The operating temperature used for the SIS/RHR line is 77°F, a rather low value. See FSAR tables 6.2.1.10 and 6.2.1.15. Provide justification that this low assumed operating temperature is conservative for the calculation of break mass and energy to be used in the pressurization analyses of the associated subcompartments.

3. Not all subcompartments with high energy lines were considered in the for pressure evaluation. Only those that were calculated to undergo the highest concentrated loading conditions in FSAR Chapter 3 were evaluated. Justify the omission of other subcompartments with high energy lines. What would be the consequences if the design pressure were exceeded in these subcompartments?

4. The initial conditions (e.g. containment pressure, temperature, relative humidity) at the receiving node and surrounding nodes were said to be imposed to maximize the resultant differential pressure across the affected node. All initial conditions for each subcompartment were not presented. Table 6.2.1-4 in FSAR Tier 2 lists the overall containment initial conditions. Section 6.2.1.1 (page 6.2-3), states that the initial pressure for subcompartment transient differential pressure analysis is 14.7 psia which is consistent with the pressure at time zero in Figures 6.2.1-5 through 6.2.1-9. The selection of initial conditions should maximize the calculated differential pressure. Justify that this was done and provide the initial conditions for all subcompartments analyzed.

5. The evaluation of subcompartment pressure is dependant on the input coefficients of inertia and the flow loss coefficient. Provide and justify the method by which the flow and inertia coefficients were chosen. Values of 1.5 were used for the flow loss coefficients. Page 225 of the 1994 Handbook of Hydraulic Resistance by Idelchik indicates that the coefficient varies with the length to diameter ratio of the hole and is between 2.85 and 1.55. Provide sensitivity studies showing the effect on subcompartment pressure to uncertainty in flow loss coefficients and provide justification that the approach taken is conservative.

6. A Nodalization sensitivity study was performed by dividing each critical room circumferentially in four nodes. According to the sensitivity analysis, the circumferential nodalization affects the local peak pressure by several psi. NUREG-0609 Chapter 3.2.2 recommends that the subcompartments be analyzed by subdividing them into a number of control volumes or nodes. Provide additional noding sensitivity studies including the effects of axial and radial noding. Discuss how the compartment pressure variations within will be included in the Chapter 3 loading evaluations and justify that this representation is conservative. Provide the sub-node volumes and flow path inputs used in the noding sensitivity studies.

7. Provide and justify to be conservative the heat transfer assumptions that are used in the GOTHIC models for subcompartment analysis.

8. The last paragraph of FSAR (Rev. 0) Section 6.2.1.2.2 (page 6.2-10), describes that the vent paths considered in the subcompartment analysis include open doors as well as grates and through wall openings. It is also stated that the effects of vent areas that become available after the occurrence of a pipe break are specifically noted and conservatively treated. Provide more details about how this is done. Tables 6.2.1-11 through 6.2.1-14 show all doors to remain closed in the analyses of critical subcompartments. Are any foils or dampers considered in the subcompartment analyses? If so their theatment should be described. Descibe the treatment of initially closed vent paths for the remander of the subcompartments that were analyzed as shown in Table 6.2.1-10.

9. FSAR Tables 6.2.1-11 through 6.2.1-14 indicate that the large lumped volumes are connected by doors to the break volume and that these doors remain closed. With only the smaller volumes considered in the analysis, one would expect pressures in the remaining volumes to trend upward as the blowdown continues. Figures 6.2.1-5 through 6.2.1-9 indicate that once the initial inertial spike is passed that the pressure in the break volume approaches an constant value. Describe the processes within the GOTHIC code which mitigate the pressure increase and justify that the analysis is conservative for determining subcompartment differential pressures.

10. Maximum calculated accident pressures are strongly affected by the area of the connecting vent paths. Describe the preoptional measures, inspections, ITAAC etc., that will be taken to ensure that the as-built subcompartments are consistent with the assumptions made in FSAR Section 6.2.1.2.

b. Subcompartment Pressure Loads (Relates to SRP Section 6.2.1.2)

1. FSAR Tier 2, Table 6.2.1-10 shows “accident pressures” for critical subcompartments but not for all listed rooms. The values shown differ from calculated pressures in FSAR Tier 2, Figures 6.2.1-5 through 6.2.1-9. What is the relation between “accident pressure” and calculated pressure? Provide the pressures calculated for all subcompartments and the pressures that are utilized in the compartment loading analyses of Chapter 3.

2. For each subcompartment for which the pressure response to a high energy pipe break was calculated, provide a comparison of the calculated subcompartment pressure with the maximum pressure allowed by the subcompartment design and justify that sufficient margin is available.

3. The subcompartment pressure analyses shown in FSAR Figures 6.2.1.5 through 6.2.1.7 show considerable variation in the peak pressure around the compartment circumference. Discuss how the pressure variations in both time and location are considered in the Chapter 3 loading evaluations and justify that this treatment is conservative.

4. FSAR Page 3E-11, states that “the upper portion of the of the SG/RCP wing wall and SG separation wall are subject to a sub-compartment pressurization load of 20 psi.” However, the calculated accident pressures in FSAR Tier 2, Table 6.2.1-10 in room UJA29-004 is 31.07 psia, which results in 16.4 psi pressure load assuming atmospheric pressure on the other side of the wall. Page 6.2-12 (FSAR Tier 2), states that a factor of 1.4 is used in peak pressure predictions which results in 23 psi pressure load in this particular subcompartment. The calculated pressure is somewhat higher than that used in the design of the structures. The design pressures of the subcompartment and how they are applied to the compartment load analyses should be clarified.

Response to Question 06.02.01.02-1:

a.1 Use of Homogeneous Equilibrium Model (HEM)

The mass and energy discharge rates used in the U.S. EPR subcompartment analysis are based on unrestricted flow with zero length/diameter (L/D). The actual L/D ratios vary depending on the pipe routing and the break location. For a break at a terminal end, the short-end L/D can be less than 1.5, while the long-end L/D can be much greater than 10. Thus, it is reasonable to assume that the combined effective L/D of a break is greater than 1.5.

The test series in the EPRI report referenced in the question (Reference 1) address L/D ratios ranging from 0.33 to 3.7. Comparison of the measured critical mass flux with HEM (illustrated in Figures 5.1 through 5.9 of the EPRI report) shows that the HEM model is adequate for calculating the mass and energy discharge rates for pipes with L/D ratios greater than 1.5.

As discussed in AREVA NP’s response to RAI No. 1, Question 6.2.1-08, the impact of using HEM on the pressurization of the critical rooms is negligible, and the existing subcompartment analysis results and conclusions for these rooms are reasonable and acceptable.

a.2 Operating Temperature Used for the SIS/RHR Line (77°F)

A response to this question will be provided by May 22, 2009.

a.3 Omission of Subcompartments with High-Energy Lines

The subcompartments selected for design and analysis in the U.S. EPR FSAR, Tier 2, Chapter 3 are those that experience the highest concentrated loading conditions. The highest loading conditions are generated from high-energy line pressures and occur in those subcompartments that support the nuclear steam supply system (NSSS) components. This envelopes the worst case loading scenario when used in the loading combinations for the structural members, and therefore justifies the omission of other subcompartments that experience a pressure load but do not support NSSS components.

Structural members, including subcompartments, are designed and sized to account for applicable loads and load combinations, including pressure, as specified in the codes and standards. Therefore, no consequences are expected from line breaks in these compartments.

a.4 Initial Conditions

Critical rooms were analyzed at zero percent relative humidity and at the following maximum allowable temperature and minimum allowable pressure:

Temperature: 131°F (maximum operating temperature in the non-accessible space)

Pressure: 14.656 psia (1.2” water gauge vacuum, normal operating pressure in the non-accessible space)

This is consistent with Standard Review Plan (SRP) 6.2.1.2 where it is stated that an acceptable model would assume air at the maximum allowable temperature, minimum absolute pressure, and zero percent relative humidity in order to maximize the resultant differential pressure.

a.5 Input Coefficients of Junction Inertia and the Flow Loss Coefficients

The junction inertias are defined as the ratio of junction effective length (L) and junction flow area (A):

$$Inertia = \frac{L}{A}$$

The junction effective length is approximated from the geometry of the connecting volumes as follows.

For vertically stacked volumes: $L = \frac{1}{2}(H_1 + H_2)$

For horizontally connected volumes: $L = \frac{1}{2}(W_1 + W_2)$

Where,

L = effective flow length of the junction
H = volume height
W = volume width

This approach is equivalent to assuming that the flow stream extends from cell center to cell center without undergoing any contraction or expansion. In reality, there is significant contraction upstream of the junction and expansion downstream of the junction. Treating the flow stream as an incompressible tube of constant cross-section increases the junction inertia, and thereby, the differential pressure.

The flow loss coefficient of 1.5 used in the subcompartment analysis is based on a sharp-edged restriction with an entrance coefficient of 0.5 and a discharge coefficient of 1.0. In a highly restrictive flow path the loss coefficient can be as high as 2.85. However, according to the *Handbook of Hydraulic Resistance* cited in the question, this requires an infinite contraction ratio. Major flow paths in the critical rooms with highest differential pressure (steam generator cavities and the reactor coolant pump rooms) have small contraction ratios, and therefore loss coefficients of about 1.5. More importantly, the peak differential pressure in the critical rooms is primarily due to the inertial pressure drop and is relatively insensitive to the loss coefficient. This is illustrated in U.S. EPR FSAR, Tier 2, Figures 6.2.1-5 through 6.2.1-9 which show that the peak pressure is experienced long before the steady-state pressure is reached.

a.6 Nodalization Sensitivity Study

The loading evaluation described in U.S. EPR FSAR, Tier 2, Chapter 3 is based on a uniform differential pressure load applied as an equivalent static load. The differential pressure is derived by adjusting the calculated peak pressure upward by an amount determined from the nodalization sensitivity study, as described in the response to item b.1 of this question (see below). Pressure gradients are not credited and the same peak pressure is assumed to prevail across the entire structure.

The structural evaluations apply a dynamic load factor to the differential pressure calculated from the peak pressure, and apply that value to the affected structural members in the associated room. For rooms with a distinctive pressure peak the dynamic load factor is 1.4, otherwise it is 1.0.

AREVA NP recognizes the limitation of the nodalization sensitivity studies performed in satisfying the requirements of NUREG-0609, “Asymmetric Blowdown Loads on PWR Primary Systems,” January 1981. NUREG-0609 addresses the issue of asymmetric blowdown loads on the reactor vessel, including its internals and supports. Capturing the pressure gradients within a subcompartment is essential to proper dynamic and structural evaluation of the primary system components. For the structural evaluation of the walls and floors, it is important to capture the maximum inter-compartmental differential pressure, but not necessarily the pressure gradients within the compartment.

As described above, the U.S. EPR FSAR, Tier 2, Chapter 3 loading evaluation is based on maximum inter-compartmental differential pressure applied uniformly without regard to spatial pressure decay. Because of this, and because adjustments were made to account for nodalization and dynamic effects, the structural evaluation results are conservative and appropriate for the design of the containment structures.

The sub-node volumes used in the sensitivity study are shown by dotted lines in U.S. EPR FSAR, Tier 2, Figure 6.2.1-1 through Figure 6.2.1-4. The sub-nodes have the same height as the parent volume and occupy one-quarter of the parent volume, as summarized in Table 06.02.01.02-1-1.

Table 06.02.01.02-1-1—Sub-Node Properties

a.7 Heat Transfer Assumptions Used in the GOTHIC Model

The GOTHIC models used in subcompartment analysis assume direct contact heat transfer from the fluid to the heat sinks. The heat transfer coefficient is based on Uchida condensing heat transfer with a minimum value of 2 Btu/hr-ft^2-°F and a maximum value of 278. Convective and radiative heat transfer modes are not credited.

Containment heat sink data used in the GOTHIC models are detailed in the response to RAI 40, Table 06.02.01-11-2.

a.8 Vent Paths Involving Doors and Foils

The vent paths for the subcompartments were assigned by considering the free openings, gratings, doors, and rupture disks listed in the response to RAI 40, Table 06.02.01-11-2. This table shows the vent types, areas, loss coefficients, and the burst pressures for all room-to-room connections in the Containment Building. In developing the subcompartment pressurization models, vent paths of similar types between adjoining rooms were combined to define an equivalent vent path.

In the GOTHIC model, the vent paths in the critical rooms and the adjacent subcompartments requiring a burst pressure to open are assumed to have a one second delay after the burst pressure is reached. Because the peak pressure is reached within the first half second following a break, these vent paths are effectively treated as being permanently closed.

a.9 Pressure Trend

The GOTHIC model used in subcompartment analyses of the critical rooms represents the entire containment building with approximately 30 nodes interconnected by vent paths. The subcompartments containing the critical rooms are further subdivided so that the critical rooms are represented as individual nodes. Following a break, the discharging fluid passes from the break compartment to the adjacent compartments, and is distributed throughout the Containment Building. Because the subcompartment pressurization analyses are short duration transients (one-half second), pressurization of the adjacent compartments during this short period is insignificant, except for large breaks such as the feedwater line break in steam generator compartments. As for the observed pressure trend, there is no built-in process in GOTHIC to arrest the pressure rise, other than the phenomena inherently modeled, such as condensation. The U.S. EPR FSAR figures might give the illusion of the pressure approaching a constant value because the subsiding pressure oscillations are going through a crest or a trough, the blowdown mass and energy discharge rates are decreasing, or both. If the transients are run beyond the initial 0.5 seconds with constant blowdown discharge, a gradual rise of the pressure in the break compartment as well as the adjacent compartments would be observed. This has been found unnecessary for subcompartment pressurization because the peak pressure occurs within a fraction of a second after the break.

The differential pressure assigned to the critical rooms is conservative for the following reasons:

- The pressure rise in the adjacent compartments is ignored.
- The peak pressure, which occurs shortly after the break, is used as the basis for the differential pressure.

a.10 As-Built Subcompartment Preoperational Inspections

A response to this question will be provided by May 22, 2009.

b.1 Accident Pressure vs. Calculated Pressure

The accident pressure given in U.S. EPR FSAR, Tier 2, Table 6.2.1-10 is derived by adjusting the calculated peak pressure upward by an amount determined from the nodalization sensitivity studies. As described in U.S. EPR FSAR, Tier 2, Section 6.2.1.2.3, each critical room is analyzed twice: first using a lumped nodalization that modeled the room as a single node, and then using a refined nodalization that divided the room into four circumferential nodes. These analyses show that the peak pressure varies by less than two psi due to the finer nodalization. To account for this variation, the calculated peak pressures are adjusted upward as follows:

$p_{accident} = p_4$ if $p_4 - p_1 < 1$ psi

$p_{accident} = p_4 + 2(p_{4e} - p_1)$ if $p_4 - p_1 \geq 1$ psi

The subscripts 1 and 4 designate the single node and four node models, respectively.

This upward-adjusted pressure is referred to as the “accident pressure” in U.S. EPR FSAR, Tier 2, Table 6.2.1-10. The data used to derive the “accident pressure” for the critical rooms given in Table 6.2.1-10 are provided in Table 06.02.01.02-1-2.

Table 06.02.01.02-1-2—Calculated Room Pressures

Room Name	Room Description	Calculated Pressure		Difference	Accident Pressure
		p_1 (psia)	p_4 (psia)	p_4-p_1 (psia)	$p_{accident}$ (psia)
30UJA 23-004	SG Cavity	22.66	24.56	1.9	28.36
30UJA 29-004	SG Cavity	26.66	28.13	1.47	31.07
30UJA 29-019	PZR Cavity	17.49	17.82	0.33	17.82
30UJA 34-019	PZR Cavity	18.02	18.59	0.57	18.59
30UJA 15-006	RCP Room	23.26	25.18	1.92	29.02

The pressure values used in the compartment loading analyses of the U.S. EPR FSAR, Tier 2, Chapter 3 are based on the accident pressures shown in U.S. EPR FSAR, Tier 2, Table 6.2.1-10 with the application of the 1.4 factor to the peak (if a peak exists), as recommended by the Standard Review Plan, Section 6.2.1.2. The final values used are rounded up, and in case of the steam generator cavities and reactor coolant pump rooms from elevation 64 ft to 94 ft, they are further increased by 4 psi as detailed in Table 06.02.01.02-1-3.

Table 06.02.01.02-1-3—Calculated Pressure Adjustments

From Elevation (ft)	To Elevation (ft)	SG Cavities	RCP Rooms	PZR Cavity
5	45	0.0	0.0	0.0
45	64	20.0 = (28.36-14.7) x 1.4	20.0 = (28.36-14.7) x 1.4	4 = (18.59-14.7) x 1.4
64	79	27.0 = (31.07-14.7) x 1.4	27.0 = (31.07-14.7) x 1.4	4 = (18.59-14.7) x 1.4
79	94	0.0	0.0	4 = (18.59-14.7) x 1.4

b.2 Subcompartment Pressure Margin

A response to this question will be provided by December 19, 2008.

b.3 Application of Pressure Profiles

Structural evaluations presented in the U.S. EPR FSAR, Tier 2, Chapter 3 utilize a factor of 1.4 on the converged enveloped peak differential pressure values for all four quadrants of each subcompartment. This allows pressure variation through time to be considered, and the value applied to the affected structural members in that room (all locations).

b.4 Design Pressure of Subcompartments

The subcompartment pressure design values used in U.S. EPR FSAR, Tier 2, Chapter 3 envelope the pressure values listed in U.S. EPR FSAR, Tier 2, Table 6.2.1-10.

The pressure value of 31.07 psia identified in the question for room number UJA29-004 is for the area of the cubicle above the critical section area presented in U.S. EPR FSAR, Tier 2, Chapter 3 Appendix 3E. The subcompartments selected for design and analysis in the U.S. EPR FSAR, Tier 2, Chapter 3 are those that undergo the highest concentrated loading conditions. The highest loading conditions are generated from high-energy line pressures and occur in those subcompartments that support NSSS components. This envelopes the worst case loading scenario when used in the loading combinations for the structural members, and justifies the omission of other subcompartments that undergo a pressure load but do not support NSSS components for the standard plant critical areas, such as UJA29-004.

The answer to item b.1 above lists the design pressures and how they are applied to the compartment load analysis in U.S. EPR FSAR, Tier 2, Chapter 3.

References:

1. EPRI NP-2192, "Critical Flow Data Review and Analysis," Electric Power Research Institute, January 1982.

FSAR Impact:

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

Question 06.02.01.03-1:

RAI - 6.2.1.3-1 (Relates to FSAR Section 6.2.1.3, Mass and Energy Release Analyses for Postulated LOCAs)

a. The initial reactor power level for the LOCA mass and energy release calculations is 4612 MWt, which is the rated thermal power level plus a calorimetric uncertainty. The uncertainty is given in FSAR Tier 2, Section 6.2.1.1.3 as 0.5 percent. Describe how the 0.5 percent calorimetric uncertainty was established and justify that it is conservative for containment analysis. What uncertainties were considered in the uncertainty analysis? What values were used for the uncertainties? How were the uncertainties combined? Describe the interface requirements (COL Information Item) which will be transmitted to COL applicants to ensure that the assumed power uncertainty is maintained for the as-built plant.

b. Describe how reactor shutdown was calculated in the RELAP5 code. Was control rod entry assumed? If so provide or reference the evaluation that the control rods would insert against the forces generated by a large break LOCA. Was a stuck control rod assembly assumed in the calculations? What was the worth of the assembly?

c. Provide an evaluation of the effect of chugging in the reactor core on the mass and energy release rate. Provide the change in the steam release rate to the containment in case of a DEG hot leg break and a DEG pump suction break if chugging is eliminated from the calculations (core flow is assumed to be smooth).

d. Page vii of BAW -10252 states that the models and methods described therein follow the guidance of NUREG-0800 (SRP Section 6) where appropriate. Provide a comparison of the assumptions used in the LOCA mass and energy release calculations with the acceptance criteria listed in SRP Section 6.2.1.3. If the acceptance criteria were not followed include a description of assumptions used to replace the SRP criteria and provide justification that they are conservative for containment analysis.

e. Table 6.2.1-1 provides a summary of the assumptions for the various loss of coolantaccidents evaluated for the containment. Table 6.2.1-20 identifies the mass and energy results from a cold leg pump discharge break as long-term Case B. The staff could not find long-term Case B on table 6.2.1-1. Provide the assumptions used in this analysis.

f. Table 6.2.1-23 gives the end of core reflood as 3957 seconds for a double ended hot leg break and 4000 seconds for double ended breaks in a cold leg pump suction or discharge. These reflood times are longer than the staff is familiar with for operating plants. Generally short reflood times are conservative for containment analysis since energy is transferred to the containment at a faster rate. Provide the criteria that are used to determine the end of reflood for the US-EPR. Discuss the relationship of the reflood calculation to core quench as discussed in SRP 6.2.1.3 and justify that the results are conservative.

g. For the limiting hot and cold leg breaks provide the temperature history of the reactor system and secondary system components to indicate that the sensible heat from the reactor system and steam generators is being accounted for and is conservatively removed by the calculation. Provide initial values, those at the end of blowdown, those at the end of reflood, those at the time of peak pressure, those at the time of the switch between the RELAP5 and GOTHIC analysis and those at the end of 24 hours. Provide the assumptions made for heat transfer between the primary medal surfaces and the fluid within the reactor system that are used in the RELAP5 analysis. The staff requests similar heat transfer information for the GOTHIC reactor system model under item “m” of this RAI.

h. Section 6.2.1.3.3.2-d of the FSAR for Midland indicates that complete steam condensation as a result of the mixing of steam and water flowing together in a pipe should not be assumed below a threshold velocity as determined by test data. Describe and justify the threshold velocity model that is used in the RELAP5 and GOTHIC mass and energy models to determine steam and water mixing within the reactor system of US-EPR following a LOCA.

i. The EPR is equipped with 4 trains of safety injection. These are cross-connected so that trains 1 and 2 interconnect and trains 3 and 4 interconnect. If the break is in the loop fed by train 1, train 2 undergoes a single failure and train 3 is out for maintenance trains 1 and 4 would be available to deliver ECCS water to the core. The water injected into loop one might be lost from the break but by the cross-connects all loops would be fed. If the failure were in train 4, train 3 was out for maintence and train 2 were operable then only loops 1 and 2 would be fed with ECCS. For double ended breaks of the hot leg and at the reactor coolant pump suction and discharge evaluate various single failure possibilities for the safety injection trains to identify the worst case.

j. One factor which might affect the steam release from the reactor system is the filling of low points in the cold leg piping at the pump suction (loop seals) in the intact cold legs following a large cold leg break. With the intact loop cold legs plugged with water all steam from the core might exit from the break without mixing and being condensed with the ECCS water. The loop seals might be filled during the course of the accident by back flow of ECCS water at the pump discharge or by entrainment of liquid from the core through the steam generators. Provide an evaluation of the potential for and the effect of loop seal filling on the steam release to the containment following a postulated break at the reactor system pump suction.

k. Provide the noding diagrams for the RELAP5 simulation of the reactor system used to predict mass and energy release from large hot and cold leg breaks. Justify the noding selected is adequate.

l. At a time between 5000 seconds and 11000 seconds depending on the break location the mass and energy release calculation is switched from RELAP5 to a one node GOTHIC model. Describe the switching criteria used to determine the time of solution transfer between the RELAP5 and GOTHIC models. Describe the precautions taken to ensure that energy is conserved between the two models at the time of the switch.

m. Provide a complete description of the GOTHIC models used to predict the long term mass and energy releases for a large hot leg, cold leg pump discharge and cold leg pump suction models. Describe the location of reactor system heat structures for each break location as to whether they are wetted or not. Justify the heat transfer options used with the wetted and unwetted structures. Describe and justify the reactor core two-phase level swell model that is used.

n. For all break locations the steam flow from the break is eventually predicted to reach or approach zero in the GOTHIC simulation. The staff does not understand how the steam release could ever be zero for a break at the reactor coolant pump suction. Provide justification that the GOTHIC model is accurately evaluating the steam and water flow and condensation phenomena within the reactor system.

o. As cooler water is injected into the cold legs during the long term post reflood phase, the flow of vapor from the break may reverse so that the containment atmosphere is drawn into the reactor system. Consider the case of a double ended pump suction break. Demonstrate that reverse flow at the reactor vessel side of the break will not cause non-condensables to be drawn in from the containment so that the steam condensation effectiveness at the SIS injection locations is reduced. Under these conditions a greater than expected fraction of steam might flow directly to the containment through the steam generator side of the break than predicted if the effect of non-condensables were not modeled.

p. Provide justification for decreasing the core decay heat multiplier from 1.2 to 1.1 in the mass and energy release calculations for the long term post reflood phase.

Response to Question 06.02.01.03-1:

a. A response to this question will be provided by December 19, 2008.

b. A response to this question will be provided by December 19, 2008.

c. A response to this question will be provided in a Technical Report by January 28, 2009. See the response to Question 06.02.01.01-1.c.1 for additional details.

d. A response to this question will be provided in a Technical Report by January 28, 2009. See the response to Question 06.02.01.01-1.c.1 for additional details.

e. A response to this question will be provided by December 19, 2008.

f. A response to this question will be provided in a Technical Report by January 28, 2009. See the response to Question 06.02.01.01-1.c.1 for additional details.

g. A response to this question will be provided by May 22, 2009.

h. A response to this question will be provided by May 22, 2009.

i. A response to this question will be provided in a Technical Report by January 28, 2009. See the response to Question 06.02.01.01-1.c.1 for additional details.

j. A response to this question will be provided in a Technical Report by January 28, 2009. See the response to Question 06.02.01.01-1.c.1 for additional details.

k. A response to this question will be provided in a Technical Report by January 28, 2009. See the response to Question 06.02.01.01-1.c.1 for additional details.

l. A response to this question will be provided in a Technical Report by January 28, 2009. See the response to Question 06.02.01.01-1.c.1 for additional details.

m. A response to this question will be provided in a Technical Report by January 28, 2009. See the response to Question 06.02.01.01-1.c.1 for additional details.

n. A response to this question will be provided in a Technical Report by January 28, 2009. See the response to Question 06.02.01.01-1.c.1 for additional details.

o. A response to this question will be provided in a Technical Report by January 28, 2009. See the response to Question 06.02.01.01-1.c.1 for additional details.

p. A response to this question will be provided by December 19, 2008.

Question 06.02.01.04-1:

RAI - 6.2.1.4, Conservativeness of the Secondary System Break Mass and Energy Release Calculations (Relates to FSAR Section 6.2.1.4 and SRP Section 6.2.1.4)

a. Describe how energy stored in secondary system metal (steam generator vessel, tubing, tubesheets, steam line, feedwater line) was treated in the mass and energy release calculations. Describe the heat transfer models that were used and justify that they are conservative. Was nucleate boiling heat transfer used below the two phase level in the affected steam generator? If a different heat transfer model was used, justify that use of the model is conservative.

b. Identify the break discharge model (HEM, Moody, others) and discharge coefficient that was used for the main steam line break analysis and justify that the assumptions are conservative for containment analysis.

c. What decay heat model was used? **[**

]

Provide justification that the model is conservative for containment analysis.

d. From BAW-10169 which is referenced, it is understand that a stuck control rod assembly was assumed in the calculations. Verify that that was the case. The effect of a stuck control rod may be a return to power within the reactor core which may increase the energy available to be released to the containment. Show the effect of the stuck control rod on reactor power by providing a plot of reactor power for the limiting case and justify that the reactor power calculated by RELAP5 is conservative for MSLB mass and energy release calculations.

e. So that the NRC staff may perform additional confirmatory containment analyses provide mass and energy release data for the double ended steam line break cases with 20% and 80% power.

f. FSAR Section 6.2.1.4.1.2 states that during periods when liquid entrainment out of the break is predicted by RELAP5, that the energy of the fluid is set to saturated steam. Provide more detail of how this is done. Were code modifications made? If so, the modifications should be described and justified.

g. For the postulated double ended break of a steam line at 50% power provide the energy and mass content of the primary system metal and fluid and the secondary system metal and fluid at the beginning of the accident and at the end of the blowdown.

h. FSAR Section 6.2.1.4.3.3 states that the volume of water in the unisolated section of main feedwater piping is considered small and is not significant for containment analysis and therefore not considered. Provide additional justification such as comparing the unisolated feedwater mass to the total mass of the affected steam generator.

i. FSAR Table 6.2.1-22 which provides the mass and energy flow from the limiting main steam line break shows an initial flow of 7956 lbm/sec of steam which increases to 13691 lbm/sec at 5 seconds. Since steam pressure will be greatest at the beginning of the event, describe the mechanisms by which the calculated steam flow is lower at the beginning of the event and justify that this treatment is conservative.

j. FSAR Section 6.2.1.4.2 states that the RELAP5 code was used to determine the mass and energy released during the blowdown. If other methodology was used to determine the mass and energy release after the blowdown period, describe this methodology and justify that it is conservative for containment analysis.

k. Postulated feedwater Line Break Accidents were not addressed in FSAR Section 6.2.1.4. Provide evaluations of the containment consequences of postulated feedwater line break accidents.

l. FSAR Section 6.2.1.4.2 states that the RELAP5 code was used to determine the mass and energy released during the blowdown. Provide a diagram of the RELAP5 noding diagram and justify that it is appropriate for EPR MSLB analyses.

Response to Question 06.02.01.04-1:

a. A response to this question will be provided by May 22, 2009.

b. [

] is described in Topical Report BAW-10193PA, “RELAP5/MOD2-B&W For Safety Analysis of B&W-Designed Pressurized Water Reactors” by reference in Topical Report BAW-10252PA, “Analysis of Containment Response to Postulated Pipe Ruptures Using GOTHIC.”

c. The decay heat model used in the analysis is [

] This decay heat assumption is consistent with Topical Report BAW-10252PA, “Analysis of Containment Response to Postulated Pipe Ruptures Using GOTHIC.”

The primary factor in the containment peak pressure and temperature during an MSLB event is the mass and energy in the affected steam generator at the time of the break. Compared to the initial mass and energy in the steam generator inventory, the heat generation rate from decay heat is not a significant contributor to the containment peak pressure and temperature, which occur relatively early in the transient. Furthermore, as steam generator inventory is released through the break, the heat transfer across the steam generator tubes degrades when the tube bundle becomes uncovered. Degraded heat transfer across the steam generator tubes reduces the effect of decay heat on containment peak pressure and temperature.

Although decay heat is not a significant contributor to the containment peak pressure and temperature, greater decay heat is conservative for the MSLB analysis. The use of [

] and is therefore conservative.

d. A response to this question will be provided by December 19, 2008.

e. A response to this question will be provided by June 23, 2009.

f. Entrained liquid out the break is treated as saturated steam by using the RELAP5/MOD2-B&W control variables. These control variables track the mass and thermodynamic properties of both the liquid and steam release separately. The control variables perform a routine manipulation of the entrained liquid mass and energy, and the energy of the entrained liquid is set to saturated steam only within the control variables. These control variables do not affect the thermodynamic properties of any junction or control volume used by RELAP5/MOD2-B&W, and do not affect the evolution of the transient. The control variables are solely for RELAP5/MOD2-B&W minor edit output for GOTHIC input.

g. A response to this question will be provided by June 23, 2009.

h. A response to this question will be provided by June 23, 2009.

i. A response to this question will be provided by June 23, 2009.

j. A response to this question will be provided by June 23, 2009.

k. A feedwater line break produces a rapid blowdown of the affected steam generator through the feedwater nozzle. Depending on the size and location of the break, this transient can vary from a rapid heatup to a rapid cooldown. Both the MSLB and feedwater line break (FWLB) analyses are performed to generate mass and energy release boundary conditions to determine the containment response to postulated breaks in the secondary system piping. The breaks' sizes and locations vary; however, the general plant system response is similar for the secondary system pipe ruptures. The feedwater system continues to feed the faulted steam generator until closure of the feedwater isolation valves.

   Given the similarities in the plant system response between the two scenarios, the same initial and boundary conditions are used for the MSLB and the FWLB analyses. The transient progression of an MSLB and a FWLB is quite similar; however, the MSLB yields the most limiting mass and energy release rate for a secondary pipe rupture due to:

   1. Break size

      Due to the physical size and arrangement of the piping, the spectrum of break sizes that must be examined for the MSLB bounds the spectrum of break sizes that must be examined for the FWLB.

   2. Integrated energy

      The integrated energy of the break fluid is always greater for a MSLB than for the FWLB. Once the steam generator (SG) tubes uncover during a MSLB, any additional feedwater injected into the SG prior to complete isolation of the feedwater system will accept energy from the primary system. This additional energy is directly deposited in containment as the SG inventory flashes and exits the break. In the case of a FWLB any main feedwater that is injected into the containment through the break does not have an opportunity to acquire additional primary system energy. Thus, the integrated energy for a MSLB is always greater than that of a FWLB.

3. Break effluent conditions

   Due to the break location for a FWLB, the break effluent is initially a liquid discharge, then progresses to a two-phase release, and then to a single-phase steam discharge. However, due to the break location in a MSLB, the break effluent is initially a single-phase steam release, then progresses to a two-phase discharge, and then back to a single-phase steam release.

For these reasons, the MSLB is more limiting than the FWLB for the U.S. EPR containment response.

l. A response to this question will be provided by December 19, 2008.

FSAR Impact:

For items b, c, f, and k, the U.S. EPR FSAR will not be changed as a result of this question.

Question 06.02.01.05-1:

a. Instead of the conservative heat transfer coefficients recommended by BTP 6-2 the minimum containment pressure analysis for the US-EPR realistic LOCA used heat transfer coefficients that were 1.7 times the Uchida correlation which were benchmarked against 1.0 times the Tagami correlation and then 1.2 times the Uchida correlation. These were described as best estimate. Provide: 1. Justification that the heat transfer correlations selected for the EPR minimum containment pressure analysis are indeed best estimate and 2. Provide the basis for the uncertainty in these coefficients so that these uncertainties may be applied in the realistic LOCA calculations.

b. FSAR Section 6.2.1.5.2 states that inside the containment, the IRWST water temperature is expected to be at the containment temperature of approximately 60°F, but could range as high as 120°F, which is the Technical Specification maximum value. The realistic large break LOCA methodology uses the value of 120°F for the minimum containment pressure calculation. Provide justification for the apparently non-conservative value selected for IRWST temperature.

c. According to FSAR Tier 2, Section 6.2.1.5.3, the passive heat sinks and thermo-physical properties were derived in accordance with Branch Technical Position 6-2, “Minimum Containment Pressure Model for PWR ECCS Performance Evaluation.” The BTP states that the data on passive heat sinks was compiled from previous reviews and used as a basis for the simplified model which it contains. This simplified model is stated to be acceptable for minimum containment pressure analyses for construction permit applications until a complete identification of available heat sinks can be made. Table 6.2.1.5 of FSAR Tier 2 (Containment heat sink inventory), shows the heat sink components, including their material, thickness and areas for EPR in detail. The staff further understands that the heat sink inventory for EPR is greater than that of operating plants. To demonstrate that the BTP containment heat sink inventory is conservative for EPR minimum containment pressure analysis, provide the results of a sensitivity study of the minimum containment pressure for which the heat sinks of Table 6.2.1.5 are used in place of those in the BTP.

d. Provide an evaluation of the effect of containment atmosphere leakage though containment openings before isolation including equipment and personnel hatches. What would be the effect of such leakage on the minimum calculated containment pressure calculation?

e. The staff understands that a realistic model as discussed in 10 CFR 50.46(a)(1)(i) is used to evaluate ECCS performance for EPR and that this model is used to provide the mass and energy release for the minimum containment pressure analysis. Section 6.2.1.5.1 of the FSAR states that the mathematical model that calculate the mass and energy releases to the containment for the minimum containment pressure analysis conforms the requirements for deterministic ECCS evaluation models in 10 CFR Part 50, Appendix K. Clarify if the model to calculate the mass and energy release is realistic as discussed in 10 CFR 50.46(a)(1)(i) or if the model is an follows Appendix K to 10CFR50 as prescribed in 10 CFR 50.46(a)(1)(ii).

f. Regulatory Guide 1.206 C.I.6.2.1.5(1) requests that for the minimum containment pressure analysis that applicants provide for the most severe break, the mass and energy release data used for the minimum containment pressure analysis. The mass and energy of safety injection fluid that is assumed to spill from the break directly to the containment floor should be included. The purpose this request is so that the staff may make independent containment pressure assessments. This information was provided in response to RAI 6.2.1-09a. The staff cannot use the mass and energy release data in a containment analysis computer code since the nitrogen accumulator gas release is lumped with the steam and water. Provide separate tables one containing the steam and water and the other containing the nitrogen release. Provide justification that input to the ICECON model in S-RELAP5 is properly accounting for the separate entry of steam and water as well as nitrogen.

g. The given initial pressure for the minimum containment pressure analysis is the normal atmospheric pressure. The initial temperatures inside and outside of the containment are lower than normally expected temperatures; thus, the initial condition appears to be conservative. However, it is not clear how the cold outside temperature was taken into account in the analyses. For instance, what initial temperature distribution through the containment wall was used? In addition, it is not clear whether the initial values have been varied and what range of variation was assumed for realistic LOCA. Provide this information.

h. Minimum containment pressure is calculated by the ICECON module embedded in S-RELAP5. Provide a noding diagram of the ICECON containment model and justify that the noding is conservative for calculating minimum containment pressure. In a presentation to the NRC staff January 29, 2008, Areva presented a sensitivity study showing that a multi-node GOTHIC model of the EPR produced containment pressure several psi lower than the single node model. Perform a similar noding sensitivity study using ICECON to show that noding detail is being conservatively accounted for.

Response to Question 06.02.01.05-1:

a. A response to this question will be provided by June 12, 2009.

b. A response to this question will be provided by May 22, 2009.

c. A response to this question will be provided by May 22, 2009.

d. A response to this question will be provided by May 22, 2009.

e. A response to this question will be provided by May 22, 2009.

f. A response to this question will be provided by June 12, 2009.

g. A response to this question will be provided by May 22, 2009.

h. A response to this question will be provided by June 12, 2009.

Question 06.02.02-1:

Containment Heat Removal Systems (FSAR Section 6.2.2)

a. **Justification for fouling factors used in the LHSI heat exchangers**

   Meeting SRP Section 6.2.2 requires that heat exchanger surface fouling should be taken into account in containment heat removal capability. The application should discuss the results of the fouling effect analysis. FSAR Tier 2, Section 6.3.2.2.2 states that conservative fouling factors in the LHSI heat exchangers have been used in the analyses. The application states in FSAR Tier 2, Section 6.2.1.1.3 that the "GOTHIC heat exchanger model was benchmarked against heat exchanger performance data to provide conservative representation." The FSAR does not provide sufficient information for confirmation of the conservatism of the applied fouling resistance factors or provide sufficient information for verification that the heat exchanger model in GOTHIC gives conservative estimates with respect to measured heat exchanger performance data. Describe how was it established that the applied fouling factors are conservative? Describe how is it known that the applied GOTHIC heat exchanger model gives conservative estimates in comparison to measured heat exchanger data?

b. **Post accident operation of the Severe Accident Heat Removal System (SAHRS) for containment mixing and cooling.**

   Following the accident at Three Mile Island Unit 2, the staff required that licensees develop symptom oriented emergency procedures for the mitigation of accidents involving multiple failures, operator error or unforeseen phenomena (NUGEG-0737 Action Item I.C.1). The staff understands that actuation of the SAHRS in the containment spray mode on high containment pressure will be included in the emergency procedures. Provide these procedures including the high containment pressure actuation criterion. Provide an evaluation of the time required to bring the SAHRS containment spray function into operation. Provide any requirements concerning SAHRS operability that will be in place during EPR operation.

c. **Post Accident IRWST Cooling:**

   FSAR Figures 6.3-1 and 6.3-2 indicate paths where the IRWST can be cooled by the SIS through the LPSI heat exchangers without injecting into the reactor system. Describe conditions under which this capability might be utilized including instrumentation to be monitored by the operating staff in affecting this mode of operation and the emergency procedure which would be followed.

Response to Question 06.02.02-1:

a. Justification for fouling factors used in the LHSI heat exchangers

The fouling factors applied to the GOTHIC low head safety injection (LHSI) heat exchanger model used in the containment heat removal analysis are:

- Tube side fouling factor = 0.000170 (ft^2-hr-°F)/BTU
- Shell side fouling factor = 0.000284 (ft^2-hr-°F)/BTU

These fouling factors are based on standard plant design and approximate those typically used for reactor coolant and closed cooling water systems. For the U.S. EPR, the applied fouling factors can be monitored with available systems information (i.e., LHSI flow, LHSI heat exchanger inlet and outlet temperatures, and component cooling water (CCWS) heat exchanger inlet and outlet temperatures).

The GOTHIC LHSI heat exchanger stand-alone model (i.e., the model based on nominal process conditions that does not include fouling factors) was benchmarked to an overall heat transfer coefficient, UA, value of 2.4738E+06 BTU/(hr-°F). The nominal conditions used were for trains 1 and 4, as listed in U.S. EPR FSAR, Tier 2, Table 6.2.1-3. This benchmarked stand-alone model was then made more conservative in the integrated containment design basis loss of coolant accident (LOCA) analysis by applying the given fouling factors. Also incorporated were the bounding shell side (CCWS) flow (608.5 lbm/s, as compared to the benchmarked data of 828.9 lbm/s) and the inlet temperature (113.0°F, as compared to the benchmarked data of 104.0°F), in order to provide bounding heat exchanger performance in the accident analysis.

(Note: The U.S. EPR FSAR, Tier 2, Table 6.3-5, the design overall heat transfer coefficient, UA, value of the LHSI heat exchanger was calculated at 3.5361E+06 BTU/(hr-°F)—higher than the benchmarked value—as a result of a more conservative effective heat transfer area applied to the GOTHIC model.)

With these applied conservatisms, the GOTHIC LHSI heat exchanger model was determined to provide sufficient margin to the containment design pressure and temperature in the containment design basis LOCA analysis.

b. Postaccident operation of the SAHRS for containment mixing and cooling

As stated in the U.S. EPR FSAR, Tier 2, Section 6.2.2, "The U.S. EPR does not credit active cooling by fan coolers or sprays inside containment during a postulated LBLOCA." The severe accident heat removal system (SAHRS) is a beyond-design-basis, non-safety-related system used only to mitigate severe accidents. As a dedicated severe accident response system, the SAHRS performance is not measured against the regulatory acceptance criteria in 10 CFR 50.46 or GDC 38, 39, and 40. Procedures addressing the unique role and operation of the SAHRS are to be documented in the U.S. EPR Severe Accident Management Guidelines and not in the Emergency Operating Procedures (EOP) addressing design-basis accidents.

c. Postaccident IRWST cooling

In U.S. EPR FSAR, Tier 2, Figures 6.3-1 and 6.3-2, paths to the in-containment refueling water storage tank (IRWST) —via the LHSI miniflow lines—allow the IRWST to be cooled by the safety injection system (SIS) through the LHSI heat exchangers without injecting into the reactor coolant system, during both normal and design basis accident conditions. However, no credit is taken for cooling of the IRWST by the LHSI heat exchangers via the LHSI miniflow lines in any design basis accidents analyzed.

The IRWST temperature is continually monitored by sensors located in each sump during design basis events.

FSAR Impact:

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

Question 06.02.04-8:

10 CFR 50.34(f)(xiv)(D) states that containment isolation systems shall be provided that utilize a containment set point pressure for initiating containment isolation as low as is compatible with normal operation.

DCA Tier 2, Sections 6.2.4.2.4 and 7.3.1.2.9, state that containment isolation signals occur at Max 1p (stage 1) from containment pressure and activity, and Max 2p (stage 2) from containment pressure.

What are the specific values of Max 1p and Max 2p?

Response to Question 06.02.04-8:

The values of Max 1p and Max 2p are:

Max 1p pressure: 18.7 psia

Max 1p activity: 100x background

Max 2p pressure: 36.3 psia

The setpoints are provided in U.S. EPR FSAR, Tier 2, Chapter 16 Tables 3.3.1-2, B.9.a, B.9.c, and B.9.d.

FSAR Impact:

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

Question 06.02.06-3:

Provide correct reference to the ANSI standard endorsed by RG 1.163, or justify why the referenced standard is an acceptable alternative.

FSAR sections 6.2.6, and 6.2.8, reference 10, refer to ANSI/ANS 56.8, 1987 and 2002. The forward to ANSI/ANS 56.8-2002 states that it was issued as an update to the 1994 version of the ANS standard and that its intended purpose is to consolidate the guidance from RG 1.163, NEI 94-01, 1995, and ANS 56.8-1994 into one document that could be referenced in the Tech Specs. The NRC has not yet reviewed nor accepted the 2002 version.

Modify the FSAR section 6.2.8, reference 10 to refer to ANS 56.8-1994, or submit ANSI/ANS 56.8-2002 for formal NRC review and approval and provide an explanation of how it comports with RG 1.163, NEI 94-01, 1995, and ANS 56.8-1994.

Response to Question 06.02.06-3:

Reference 10 of U.S. EPR FSAR, Tier 2, Section 6.2.8 will be modified to refer to ANSI/ANS 56.8-1994.

FSAR Impact:

U.S. EPR FSAR, Tier 2, Section 6.2.8 will be revised as described in the response and indicated on the enclosed markup.

U.S. EPR Final Safety Analysis Report Markups

containment pressure boundary remain intact during the harshest expected conditions, thereby precluding release of radioactivity to the environment.

6.2.8 References

1. BAW-10252(NP)-A, Revision 0, "Analysis of Containment Response to Postulated Pipe Ruptures Using GOTHIC," Framatome ANP, September 2005.

2. BAW-10168P-A, Revision 3, "BWNT Loss-of-Coolant Accident Evaluation Model for Recirculating Steam Generator Plants – Volume I – Large Break," B&W Nuclear Technologies, December 1996.

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

4. NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," Revision 1, U.S. Nuclear Regulatory Commission, July 1981.

5. BAW-10169P-A, "B&W Safety Analysis Methodology for Recirculating Steam Generator Plants," B&W Fuel Company, October 1989.

6. Karwat, H., "State of the Art Report on Containment Thermal Hydraulics and Hydrogen Distribution," NEA/CSNI, June 1999.

7. ANSI/ANS–56.2, "Containment Isolation Provisions for Fluid Systems After a LOCA," American National Standards Institute/American Nuclear Society, 1989.

8. ANP-10268P, Revision 0, "U.S. EPR Severe Accident Evaluation," AREVA NP Inc., October 2006.

06.02.06-3

9. ANSI N45.4, "Leakage Rate Testing of Containment Structures for Nuclear Reactors," American National Standards Institute, 1972.

10. ANSI/ANS 56.8, "Containment System Leakage Testing Requirements," American National Standards Institute/American Nuclear Society, ~~1987 & 2002~~1994.

11. SRM-SECY-04-0032, "Review of Operational Programs in a Combined License Application and Generic Emergency Planning Inspections, Tests, Analyses, and Acceptance Criteria," U.S. Nuclear Regulatory Commission, 2004.

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

13. NEI 94-01, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," Nuclear Energy Institute, 1995.