AREVA

December 17, 2008
NRC:08:098

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Response Supplement 1 to U.S. EPR Design Certification Application RAI No. 82

Ref. 1: E-mail, Getachew Tesfaye (NRC) to Ronda Pederson, et al (AREVA NP Inc.), "U.S. EPR Design Certification Application RAI No. 82," October 3, 2008.

Ref. 2: E-mail, Ronda Pederson, et al (AREVA NP Inc.), to Getachew Tesfaye (NRC), "U.S. EPR Design Certification Application RAI No. 82," November 3, 2008.

In Reference 1, the NRC provided a request for additional information (RAI) regarding the U.S. EPR design certification application. Reference 2 provided technically correct and complete responses to 3 of the 8 questions. Technically correct responses to portions of 4 of the 5 remaining questions are enclosed with this letter as committed.

The enclosed response consists of the following:

Question #	Start Page	End Page
RAI 82 — 06.02.01-12b.1	8	10
RAI 82 — 06.02.01.02-1b.2	14	14
RAI 82 — 06.02.01.03-1a	17	18
RAI 82 — 06.02.01.03-1b	18	18
RAI 82 — 06.02.01.03-1e	18	18
RAI 82 — 06.02.01.03-1p	19	20
RAI 82 — 06.02.01.04-1a	22	22
RAI 82 — 06.02.01.04-1d	22	24
RAI 82 — 06.02.01.04-1l	23	28

The schedule for technically correct and complete responses to the remaining RAI No. 82 questions, as committed in Reference 2, is provided below and remains unchanged.

D077
NRO

Question #	Response Date
RAI 82 — 06.02.01-12	May 22, 2009
RAI 82 — 06.02.01.02-1	May 22, 2009
RAI 82 — 06.02.01.03-1	May 22, 2009
RAI 82 — 06.02.01.04-1	June 23, 2009
RAI 82 — 06.02.01.05-1	June 12, 2009

AREVA NP considers some of the material contained in the enclosure to be proprietary. As required by 10CFR2.390(b), an affidavit is enclosed to support the withholding of the information from public disclosure. Proprietary and non-proprietary versions are enclosed with this response.

Sincerely,

Sandra M. Sloan

Sandra M. Sloan, Manager
New Plants Regulatory Affairs
AREVA NP Inc.

Enclosures

cc: J. Rycyna
G. Tesfaye
Docket No. 52-020

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Ronda M. Pederson. I am Licensing Manager, U.S. EPR Design Certification, Regulatory Affairs for New Plants Deployment, for AREVA NP Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA NP to determine whether certain AREVA NP information is proprietary. I am familiar with the policies established by AREVA NP to ensure the proper application of these criteria.

3. I am familiar with the AREVA NP information contained in "Response to Request for Additional Information No. 82, Supplement 1" and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA NP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information".

6. The following criteria are customarily applied by AREVA NP to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA NP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA NP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA NP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA NP, would be helpful to competitors to AREVA NP, and would likely cause substantial harm to the competitive position of AREVA NP.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b) and 6(c) above.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in this Document has been made available, on a limited basis, to others outside AREVA NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

Donda Peden

SUBSCRIBED before me this 17th

day of December, 2008.

Smcfad

Sherry L. McFaden
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 10/31/2010
Registration # 7079129

SHERRY L. MCFADEN
Notary Public
Commonwealth of Virginia
7079129
My Commission Expires Oct 31, 2010

Response to

Request for Additional Information No. 82 Supplement 1

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. To support the U.S. EPR design, AREVA NP reviewed the possible scenarios leading to a negative containment pressure. The scenarios selected and described in the U.S. EPR FSAR were conservatively examined to determine the limiting scenario. The basic elements of each analysis are summarized below.

   Sudden Containment Temperature Reduction

   This event postulates that the ambient temperature inside the Containment Building suddenly drops due, for example, to severe weather outside the building or the activation of HVAC within the building. The temperature drop removes heat from the Containment Building which condenses the water out of the containment atmosphere and reduces the pressure in the building. The specific cause of the pressure reduction is not significant, since the final temperature is conservatively assumed to be the minimum allowed temperature of 59°F.

   Assumptions

   a. Heat losses to the containment from operating equipment is conservatively neglected
   b. In-leakage from the annulus is conservatively neglected
   c. External ambient pressure is initially 14.696 psia
   d. External ambient pressure remains constant
   e. Initial Containment Building relative humidity is 70%
   f. Initial Containment Building temperature is 122°F, the maximum for normal power operation
   g. Initial Containment Building pressure is 14.496 psia, the normal ambient condition
   h. The final containment temperature of 59°F is the minimum analyzed value.

   The calculation used the definition of relative humidity, partial pressures, and the ideal gas law to determine the resulting pressure drop. The calculation conservatively assumes that the end-state removes all the moisture from the air. This simplifies the ideal gas equation since there is now only one constituent. The effect of reduced containment volume due to conversion of water vapor into liquid that falls to the containment floor was conservatively ignored. As described in U.S. EPR FSAR, Tier 2, Section 6.2.1.1.1, this is the limiting event.

Removal of IRWST Inventory

This event postulates that the water in the IRWST ($7.22x10^4$ ft^3) is removed; this results in an increased containment volume that causes a sudden pressure drop. The final volume analyzed conservatively assumes the entire IRWST inventory is removed.

Assumptions

a. Containment Building is sealed

b. Internal ambient pressure is initially 14.496 psia

c. External ambient pressure remains constant at 14.696 psia.

A calculation using the ideal gas law equation for a constant specific heat, adiabatic process was performed.

HVAC Pulldown of Containment Pressure

This event postulates that the Containment Building ventilation system enters a purge mode which evacuates the building to the maximum extent possible by HVAC. Since fans and ductwork for U.S. EPR are specified later in the design process, representative fan curves were used to determine the maximum evacuation possible.

Assumptions

a. Containment Building is sealed

b. Internal ambient pressure is initially 14.496 psia

c. External ambient pressure remains constant at 14.696 psia

d. HVAC exhausts to ambient without containment in-leakage.

The calculation determined that the dead head suction pressure drop inside containment is bounded by one inch of water gauge pressure.

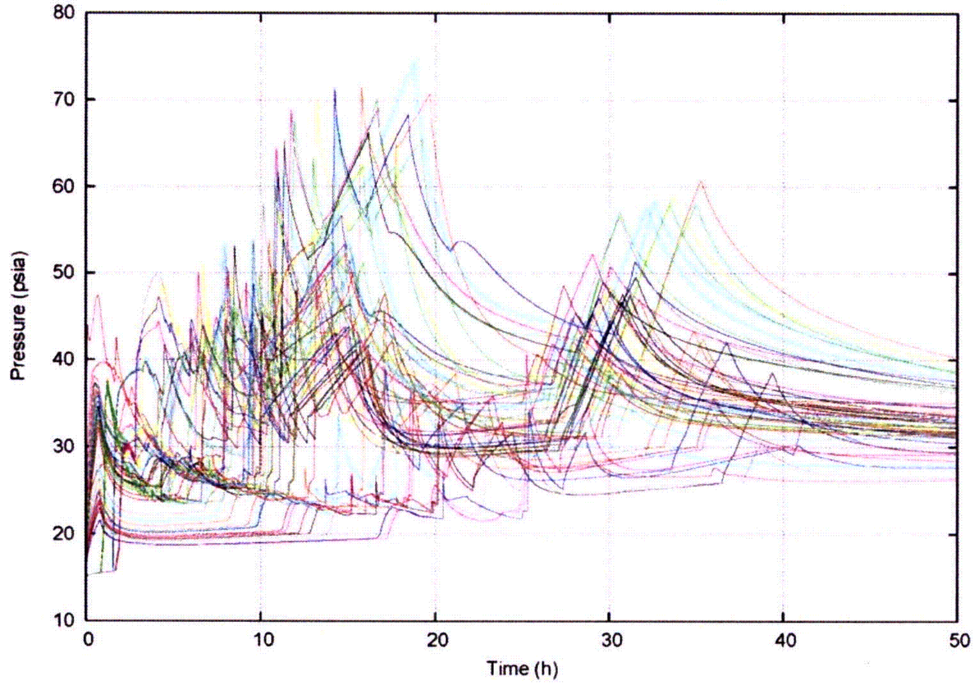
Postaccident Cooldown

Accidents that affect containment pressure (including loss of coolant accidents (LOCA), main steam line breaks, and small-break LOCAs) are described in a variety of calculations in the U.S. EPR FSAR. None of these calculations indicate negative containment pressures during the analyzed event. Therefore, the hypothetical pressure response from these events is bounded by the sudden containment temperature reduction event described above.

Post-Severe Accident Cooldown

AREVA NP performed an uncertainty analysis that included 59 separate cases to support containment performance following a severe accident. These cases are described in U.S. EPR FSAR, Tier 2, Section 19.2.4.2.3. As shown in Figure 06.02.01-12-1, none of the cases showed negative containment pressure. Therefore, the pressure response from these events is bounded by the sudden containment temperature reduction event described above.

Figure 06.02.01-12-1: Containment Pressure vs. Time for 59 Cases



2. This question was answered in AREVA NP's original response to RAI 82.

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

a.1 Use of Homogeneous Equilibrium Model (HEM)

This question was answered in AREVA NP's original response to RAI 82.

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

This question was answered in AREVA NP's original response to RAI 82.

a.4 Initial Conditions

This question was answered in AREVA NP's original response to RAI 82.

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

This question was answered in AREVA NP's original response to RAI 82.

a.6 Nodalization Sensitivity Study

This question was answered in AREVA NP's original response to RAI 82.

a.7 Heat Transfer Assumptions Used in the GOTHIC Model

This question was answered in AREVA NP's original response to RAI 82.

a.8 Vent Paths Involving Doors and Foils

This question was answered in AREVA NP's original response to RAI 82.

a.9 Pressure Trend

This question was answered in AREVA NP's original response to RAI 82.

a.10 As-Built Subcompartment Preoperational Inspections

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

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

b.1 Accident Pressure vs. Calculated Pressure

This question was answered in AREVA NP's original response to RAI 82.

b.2 Subcompartment Pressure Margin

A comparison of the calculated subcompartment differential pressure with the maximum differential pressure allowed by the subcompartment design is shown in Table 06.02.01.02-1-b.2.

Table 06.02.01.02-1-b.2: Subcompartment Pressures

From Elevation (ft)	To Elevation (ft)	SG Cavities		RCP Rooms		PZR Cavity	
		Design Δp (psi)	Allowable Δp (psi)	Design Δp (psi)	Allowable Δp (psi)	Design Δp (psi)	Allowable Δp (psi)
45	64	20	21.5	20	21.5	4	13
64	79	27	31	27	31	4	13
79	94					4	13

The allowable differential pressures were determined in accordance with ACI 349 and RG 1.142 for seismic Category I concrete structures. Because this process involves different categories of loads and load combinations, a design margin for differential pressure alone cannot be adequately defined. However, as shown above, the design differential pressure is bounded by the allowable differential.

b.3 Application of Pressure Profiles

This question was answered in AREVA NP's original response to RAI 82.

b.4 Design Pressure of Subcompartments

This question was answered in AREVA NP's original response to RAI 82.

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. Core Power Uncertainty

The 0.5 percent calorimetric uncertainty considers the individual uncertainties of the parameters that affect the primary-to-secondary heat balance. The individual uncertainties were statistically combined in accordance with ASME OTC 19.1-1998 as described by the following equation.

$$\frac{\Delta Q}{Q} = \sqrt{\sum \left(\frac{\partial Q}{\partial x_i} \Delta x_i \right)^2}$$

where Q stands for the reactor power, and x stands for independent parameters in the heat balance equation.

These independent parameters, along with their uncertainties, are listed in Table 06.02.01.03-1a. The end result of this calculation is an uncertainty of 0.40 percent based on a nominal core power of 4590 MW_{th}. This value has been increased by 25% to 0.50 percent, and is used in the U.S. EPR FSAR.

Table 06.02.01.03-1a: Calorimetric Parameters and Uncertainties

Parameter	Unit	Nominal Value (x)	Relative Uncertainty	Absolute Uncertainty (Δx)
Feedwater Flow Rate	lbm/hr	2.088E+07	0.28%	5.85E+04
Feedwater Pressure	psia	1132.8		25
Feedwater Temperature	°F	446.36		0.6
Steam Pressure	psia	1108.9		25.4
Steam Moisture Content	%	0.25		0.25
Blowdown Flow Rate	lb/hr	2.088E+05	5.00%	1.04E+04
Blowdown Pressure	psia	1124		30.00
Letdown Flow Rate	lb/hr	79366	4.00%	3.17E+03
Letdown Pressure	psia	2218	3.00%	66.54
Letdown Temperature	°F	563.5	3.00%	16.91
Charging Flow Rate	lb/hr	79366	4.00%	3.17E+03
Charging Pressure	psia	2340	3.00%	70.20
Charging Temperature	°F	518	3.00%	15.54
RCP Heat	Btu/hr	2.64E+07	20.00%	5.28E+06
RCS Heat Loss/Gain	Btu/hr	2.66E+07	20.00%	5.32E+06

The core power uncertainty used in the U.S. EPR FSAR is based on current industry practices with realistic sensor and measurement uncertainties, with the 25 percent margin added to the final calculated value for conservatism. Core power will be measured using a secondary side heat balance with feedwater flow rate. Feedwater flow rate will be measured using an ultrasonic flow meter.

The uncertainty requirements are transmitted to the COL applicant in U.S. EPR FSAR, Tier 2, Section 1.1.4.

b. Use of Control Rods for Reactor Shutdown in RELAP5.

For loss of coolant accident (LOCA) mass and energy release analysis, insertion of negative reactivity by the control rods (reactor trip) was not credited during blowdown. The negative reactivity inserted by the massive voiding caused by a LOCA was sufficient to trip the reactor. The negative reactivity inserted by the control rods and boron from the accumulators was credited during the reflood stage to maintain subcriticality in the core. This was accomplished by activating a reactor trip table 15 seconds after the break. The actual insertion time of the control rods, based on test data, is 3.5 seconds.

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. LOCA Long Term Case B.

Long Term Case B is Case 31 in U.S. EPR FSAR, Tier 2, Table 6.2.1-1, as indicated by the title of Table 6.2.1-20. The assumptions used in this case are listed in Table 6.2.1-1.

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. Core Decay Heat Multiplier.

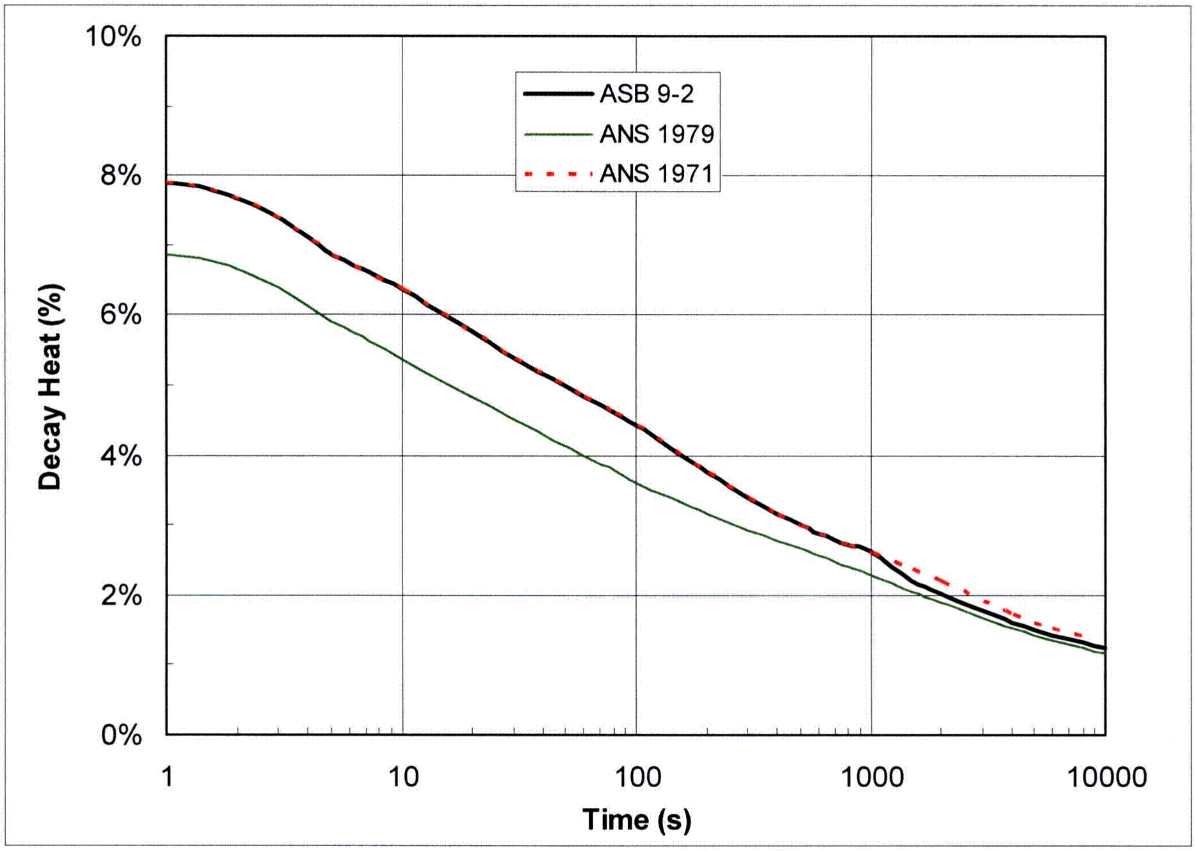
The core decay heat multiplier was defined in conformance with NUREG-0800 Branch Technical Position ASB 9-2. The ASB states that "In calculating the fission produced decay heat, a 20% uncertainty factor (K) should be added for any cooling time less than 10^3 seconds, and a factor of 10% should be added for cooling times greater than 10^3 but less than 10^7 seconds." The multipliers used in the LOCA mass and energy release model correspond to these uncertainty factors applied to the ANS 1971 decay heat curve.

The Standard Review Plan Section 9.2.5 references ANS 1979 as being acceptable for calculating the decay heat. A comparison of ANS 1979 decay heat rate with ASB 9-2 is provided in Figure 06.02.01.03-1p. The comparison shows that ASB 9-2 with 1.2/1.1 multiplier bounds ANS 1979 with a 2σ multiplier, where σ is the standard deviation set to 0.03 in accordance with AREVA NP's NRC approved realistic large break LOCA methodology. Additional information for the basis and the conservatism of the core decay heat rate used in U.S. EPR safety analyses is provided in the response to RAI 34 Question 15-7.

FSAR Impact:

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

Figure 06.02.01.03-1p: Comparison of ANS 1971 and ASB 9-2 Decay Heat Rates



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. Energy Stored in the Secondary System Metal

AREVA NP calculated the release of the energy stored in the secondary system metal explicitly with the RELAP5/MOD2-B&W computer code by using heat conductors to represent the metal structures in the model.

The RELAP5/MOD2-B&W computer code incorporates a full spectrum of heat transfer modes, including single phase convection, nucleate boiling, critical heat flux, transition film boiling, film boiling, and condensation. The appropriate heat transfer correlation is determined by RELAP5/MOD2-B&W for each heat conductor in the model based on the calculated thermodynamic conditions at each time step. [

] This modeling provides a detailed heat transfer profile in the steam generator secondary side tube region during the main steam line break analysis.

As a transient evolves, the heat transfer mode for a given heat conductor may change in response to changing thermodynamic conditions. For the main steam line break event at time zero, some nodes of the steam generator tube section are in single phase liquid convection heat transfer mode. In this mode, the heat transfer coefficient is the maximum of the Dittus-Boelter, Rohsenow-Choi, or the natural convection correlation.

Other nodes of the steam generator tube section are in saturated nucleate boiling heat transfer mode. In this mode, the heat transfer coefficient is calculated using the Chen correlation. These heat transfer correlations provide a representative heat transfer calculation for the main steam line break analysis.

b. This question was answered in AREVA NP's original response to RAI 82.

c. This question was answered in AREVA NP's original response to RAI 82.

d. Stuck Control Rod

In the shutdown reactivity calculations for the analysis, the most reactive control rod assembly was assumed to be stuck out of the core. A plot of reactor power for the limiting case is provided in Figure 06.02.01.04.d-1.

RELAP5/MOD2-B&W has a point kinetics model with six delayed neutron groups to perform reactor physics calculations. For the main steam line break (MSLB) mass and energy release analysis, it has provisions for fuel temperature feedback, moderator temperature feedback, and tripped rod reactivity. The most negative moderator and Doppler temperature feedback coefficients at end of cycle are selected to conservatively predict the core power response. The NRC has approved RELAP5/MOD2-B&W with its point kinetics model for loss of coolant accident (LOCA) analyses in topical report BAW 10192P-A ("BWNT Loss of Coolant Accident Evaluation Model for Once-Through Steam Generator Plants") and non-

LOCA transient analyses in topical report BAW 10193P-A ("RELAP5/MOD2-B&W For Safety Analysis of B&W Designed Pressurized Water Reactors").

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

f. This question was answered in AREVA NP's original response to RAI 82.

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. This question was answered in AREVA NP's original response to RAI 82.

l. RELAP5 Noding Diagram used in the MSLB Analysis.

The RELAP5 noding diagram used in the MSLB analysis is provided in Figure 06.02.01.04.l -1 through Figure 06.02.01.04.l -4.

FSAR Impact:

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

Figure 06.02.01.04.d-1: MSLB Reactor Power vs. Time (50% Rated Power)

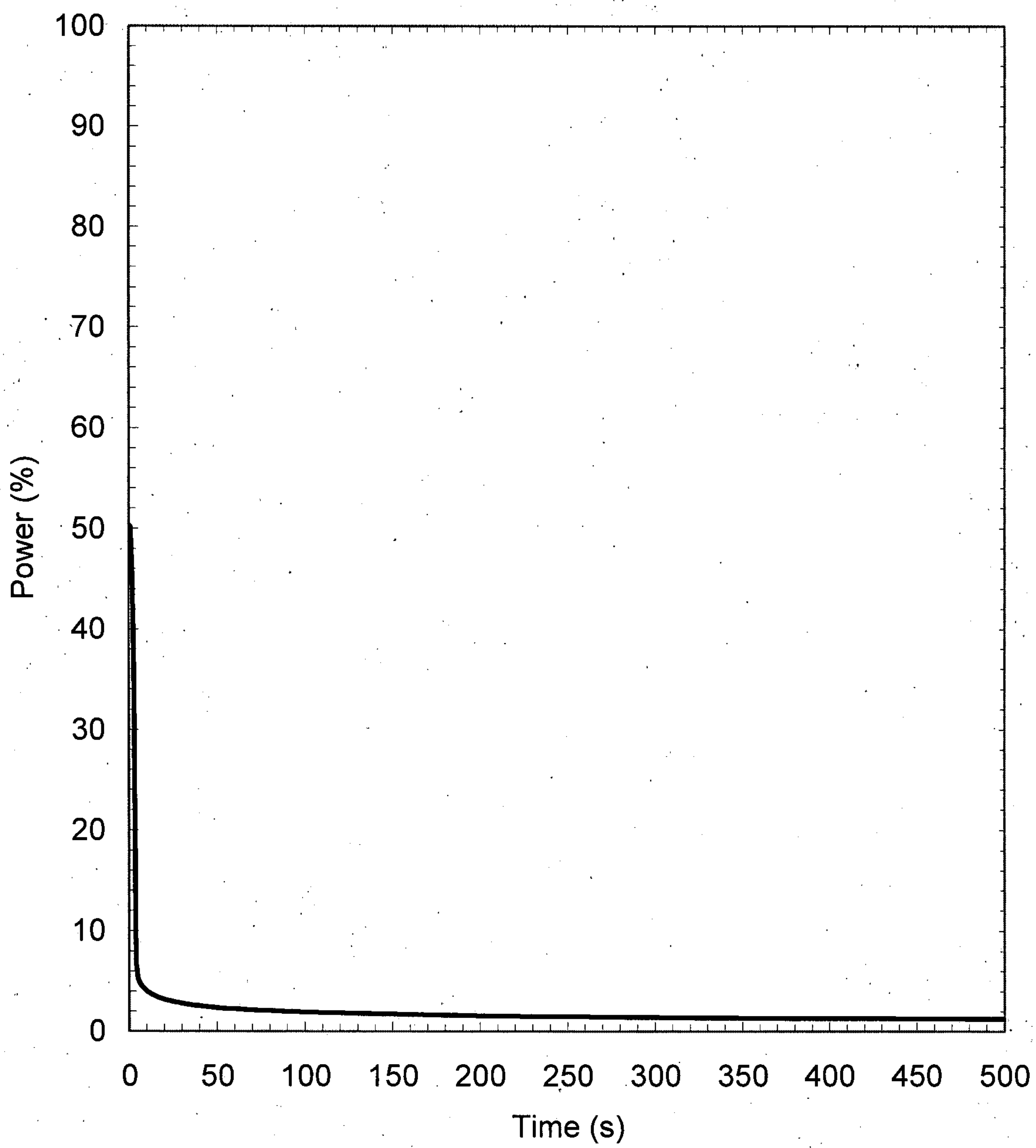


Figure 06.02.01.04.I-1: RELAP5 MSLB Noding Diagram

Figure 06.02.01.04.I-2: Reactor Vessel Nodalization

Figure 06.02.01.04.I-3: Single Loop Steam Generator Nodalization

Figure 06.02.01.04.I-4: Triple Loop Steam Generator Nodalization