Response to

Request for Additional Information No. 1, Revision 0

4/9/2008

U. S. EPR Standard Design Certification

AREVA NP Inc.

Docket No. 52-020

SRP Section: 6.2.1 – Containment Functional Design

Application Section: 6.2

SPCV Branch

Question 6.2.1-01

(FSAR Section 6.2.1) The NRC staff is reviewing the AREVA evaluation model for the containment functional design following a loss of coolant accident. The evaluation model includes methodology for determining the steam release and water spillage from the reactor system following a piping rupture as well as the methodology for determining the thermodynamic condition within the containment building. The EPR containment design is unlike that of operating PWRs in the US in that it does not include safety related containment sprays or fan coolers and instead relies on other containment heat removal mechanisms to a greater extent than credited in the containment analyses of operating PWRs. These heat removal mechanisms are the quenching of steam within the reactor system by incoming ECCS water and the removal of heat from the containment atmosphere by the internal containment structures.

Although the NRC staff has previously approved the AREVA evaluation model for use in safety analyses for the core reloads of operating plants, the staff did not review the methodology for application to the safety analyses of a plant without active safety related systems able to cool and mix the containment atmosphere following a postulated design basis LOCA. The NRC staff therefore requests that AREVA follow Regulatory Guide 1.203 in seeking approval of the EPR containment evaluation model. RG 1.203 describes a process that the NRC staff considers acceptable in developing and assessing evaluation models that will be used to analyze transient and accident behavior that is within the design basis of a nuclear power plant. The staff recognizes that portions of the EPR containment model have undergone a process similar to the one described in RG 1.203 for operating plants. For the portions of the model that have undergone a process similar to RG 1.203, the staff requests that AREVA provide a description or reference to the prior process and explain why the prior process bounds the application of the model for the EPR.

a. Regulatory Guide (RG)-1.203, “Transient and Accident Methods,” Regulatory Element 1 provides 20 steps for a process of evaluation model development and assessment. These elements discuss how computer codes will be assessed for adequacy for specific applications, describes their usage with other computer codes and their qualification for the specific applications for which they will be applied. Please address each of these 20 steps and show how the recommendations are met by the AREVA evaluation model for the US-EPR containment safety analysis.

b. Regulatory Element 2 of RG-1.203 deals with quality assurance. Appendix B to 10 CFR Part 50 describes NRC requirements regarding quality assurance for nuclear power plants. Please provide descriptions of how the evaluation model for US-EPR containment analysis meets these requirements.

c. Regulatory Element 3 of RG-1.203 deals with documentation. Please provide or provide available reference to the following documentation for the US-EPR containment evaluation model:

   c.1. Requirements
   The requirements for the evaluation model as developed in the phenomena identification and ranking table (PIRT) should be established and documented. Step 4 of Regulatory Position 1 to RG-1.203 deals with the development of the PIRT for the various applications for the evaluation model. The PIRT provides a means of determining those processes and phenomena for which code assessment should be

demonstrated. Please provide PIRTs for the evaluation of the containment functional design following a LOCA. Provide the qualifications of the PIRT panel members.

c.2. Methodology
Please provide methodology documentation as described in the regulatory guide. You should include noding diagrams as well as the selection of input options and justify the selection of each option chosen. The noding arrangement for the containment building and the reactor system should be provided and justified.

c.3. User manuals and user guidelines
Provide documentation describing user instructions for the application of the evaluation model software for US-EPR containment analysis.

c.4. Scaling Reports
Scaling analyses should be conducted to ensure that the data and the models based on those data, will be applicable to the full-scale analyses performed for the US-EPR. Provide scaling reports for the test facilities used in the validation for the US-EPR containment evaluation model as discussed in the regulatory guide.

c.5. Assessment Reports
As described in the regulatory guide, the evaluation model should be assessed against applicable experimental data to ensure that the significant phenomena and processes for containment analysis as identified in the PIRT are being adequately modeled.

c.6. Uncertainty Analysis Reports
Please provide documentation of any uncertainty analysis performed for use of US-EPR containment evaluation model.

Response to Question 6.2.1-01:

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

AREVA NP Inc. will provide responses to RAI questions 6.2.1-01, -02, -03, and -07d in a comprehensive Technical Report (TR) that provides the basis for the modeling of containment pressure and temperature response of the U.S. EPR following postulated loss of coolant accidents (LOCA). The TR will provide an assessment of the principal phenomenology concerning containment heat removal to demonstrate conformance with design and regulatory criteria. The scope of the TR will include the following:

- The containment pressure mitigation strategy of the U.S. EPR, including a discussion of any unique features.
- An overview of containment test facility data and an assessment of applicability to the U.S. EPR.
- Validation of code applicability for the U.S. EPR phenomenology, including code-to-code comparisons.
- A discussion of containment analytical methods, including nodalization and treatment of phenomenological uncertainties.
- Benchmarking and sample problem analyses.

The TR will provide the basis for demonstrating that the passive heat sinks within containment are sufficient to limit the pressure response while affording a well-mixed environment to allow containment pressure to decrease to less than half the peak pressure in less than 24 hours following a postulated accident. AREVA NP Inc. plans to have progressed sufficiently in the development of the TR by the end of June 2008 to support further discussions with the NRC staff.

Question 6.2.1-02:

(FSAR Section 6.2.1) The following is a list of issues involving the mass and energy release calculations that the staff believes should be investigated in the application of RG 1.203 to the AREVA post LOCA containment evaluation model for US-EPR. Please indicate by cross-reference where each of the following items is addressed in the response to RAI #1.

a. Blowdown
   Demonstrate that the RELAP5 reactor core heat transfer assumptions are conservative for containment analysis.

b. Refill
   Historically no refill time was assumed so that the water level in the reactor vessel at the end of blowdown was set at the bottom of the core when the refill started. If AREVA makes other assumptions they should be justified.

c. Reflood
   The liquid predicted to be carried out of the top of the core for US-EPR to the steam generator tubes should be compared with carryout rate fraction (CRF) measurements for similar reactor core conditions such as those in the FLECHT series. SRP 6.2.1.3 defines the CRF as the mass ratio of liquid exiting the core to the liquid entering the core. Heat transfer from the steam generator tubes should be justified. Steam quenching in the cold legs by the incoming ECCS water should be justified. Core flow reversals predicted by RELAP5 which act to quench steam generated by the core in the lower plenum volume need to be justified by comparisons with experimental data or other methodology which does not produce flow reversals should be used.

   For a postulated double ended cold leg break provide graphs of the core inlet flow rate, the core exit liquid flow rate, and the CRF. Plot resolution should be sufficiently detailed so that flow oscillations are displayed eg. ½ second of transient time. Provide comparison plots of the CRF predicted by the RELAP5 code with the predictions of applicable experiments.

d. Post Reflood
   Boiling in the core at low pressure will cause a two phase level to rise into the steam generators. The level swell assumptions used in the RELAP5 and GOTHIC computer models of the reactor system should be justified. Heat transfer from the steam generator tubes should be justified. Steam quenching in the cold legs by the incoming ECCS water should be justified. Note that SRP 6.2.1.3 recommends that “Steam from decay heat boiling in the core should be assumed to flow to the containment by the path which produces the minimum amount of mixing with the ECCS injection water.” If other assumptions are made for steam/water mixing, justification is required.

Response to Question 6.2.1-02:

A response to this question will be provided by January 28, 2009. See the response to Question 6.2.1-01 for additional details.

Question 6.2.1-03:

(FSAR Section 6.2.1) The following issues involve the containment pressure and temperature calculations. The staff believes that these should be investigated in the application of RG 1.203 to the AREVA post LOCA containment evaluation model for US-EPR. Please indicate by cross-reference where each of these items is addressed in the response to RAI #1.

a. Models for determining the thermodynamic condition within the reactor building should be justified including the noding detail selected for the various reactor building compartments and for the internal flow paths.

b. Heat transfer assumptions for the internal containment structures should be justified. Separate validation should be performed for the vertical and horizontal heat transfer surfaces.

Response to Question 6.2.1-03:

A response to this question will be provided by January 28, 2009. See the response to Question 6.2.1-01 for additional details.

Question 6.2.1-04:

(FSAR Section 6.2.1.3) Please provide the staff with the SRELAP5 input used to determine M&E release for the double ended cold leg break at a reactor coolant pump suction that is described in the FSAR.

Response to Question 6.2.1-04:

The M&E release for the containment response to loss of coolant accidents and main steam line break accidents was calculated using the RELAP5/MOD2-B&W computer program rather than S-RELAP5. The RELAP5MOD2-B&W computer program is based on RELAP5/MOD2 whereas S-RELAP is based on RELAP5/MOD2.5. The S-RELAP data previously transmitted to the NRC should provide sufficient information to develop an input deck for the validation of the M&E results.

The S-RELAP5 input deck and updates were submitted by AREVA NP Inc. to the NRC via the following correspondence:

- Letter, Ronnie L. Gardner (AREVA NP Inc.) to Document Control Desk (NRC), “Response to a Request from the NRC for an S-RELAP5 Input Deck for the U.S. EPR,” NRC:06:063, December 21, 2006.

- Letter, Ronnie L. Gardner (AREVA NP Inc.) to Document Control Desk (NRC), “Response to Questions Regarding the S-RELAP5 Input Deck for the U.S. EPR,” NRC:07:036, August 17, 2007.

- Letter, Ronnie L. Gardner (AREVA NP Inc.) to Document Control Desk (NRC), “Additional Information Regarding the S-RELAP Input Deck for the U. S. EPR,” NRC:08:005, January 24, 2008.

FSAR Impact:

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

Question 6.2.1-05:

(FSAR Section 6.2.1) The staff has the following questions regarding the GOTHIC model (short-term LOCA, case 7A) which was provided to the staff by AREVA. There are some apparent inconsistencies between the FSAR descriptions of the containment analysis and the EPR GOTHIC input. Please provide clarification for the following:

a. FSAR Table 6.2.1-5 does not specify the presence of stainless steel; it appears from the GOTHIC input that the three IRWST heat sinks are lined with stainless steel. Stainless steel is also specified for the containment shell in the GOTHIC input and the stainless steel is input as painted; is that correct? Provide a material property table for the passive heat sinks which includes material density, thermal conductivity and heat capacity for each material type.

b. FSAR Section 6.2.1.1.3 indicates that the heat sinks listed are all exposed to the containment atmosphere; however it appears that in the GOTHIC input the three IRWST heat sinks are exposed to the pool; which is correct?

c. Containment initial and boundary conditions are listed in FSAR Table 6.2.1-4 which do not appear to reflect the intended operating conditions, i.e., the accessible areas of the containment have an operating range different from the inaccessible areas. The GOTHIC input for initial conditions uses the upper bound pressure and lower bound temperature based on the FSAR table. The heat sinks have various initial temperatures. Please justify that the initial conditions selected in the GOTHIC input are those which provide the most conservative result.

d. In FSAR Section 6.2.1.1.3, it appears that nitrogen gas from all 4 accumulators is assumed to be released into the containment. The GOTHIC input seems to assume only 3 accumulators release nitrogen. Please address this apparent inconsistency. Justify that the inputting of nitrogen from only 3 accumulators is acceptable.

Response to Question 6.2.1-05a:

The GOTHIC input provides an accurate material representation of the passive heat sinks inside containment. The following materials are specified from the inside to the outside surface:

- Containment shell: paint–stainless steel liner–air gap–concrete.
- IRWST heat sinks: stainless steel liner–concrete.
- Accessible and inaccessible area walls (vertical and horizontal): paint–concrete.
- Steel (thick, medium and thin) heat sinks: paint–carbon steel.

FSAR Table 6.2.1-5 will be revised to be consistent with the GOTHIC input.

The material properties for the passive heat sinks are provided in Table 6.2.1-05-1 and include material density, thermal conductivity and heat capacity for each material type.

Response to Question 6.2.01-05b:

The passive heat sinks inside the primary containment consist of all painted and unpainted concrete, steel structures and liner for the containment shell, and in-containment refueling water storage tank (IRWST) surfaces.

The GOTHIC input provides an accurate representation of these heat sinks. The three IRWST heat sinks are exposed to the pool. The remaining heat sinks are exposed to the containment atmosphere. The relevant statement in FSAR Section 6.2.1.1.3 describing the heat sinks listed will be revised as shown in the enclosed markup to be consistent with the GOTHIC input.

Response to Question 6.2.1-05c:

During power operation the inaccessible areas inside containment experience higher temperatures than the accessible areas because they are directly exposed to the hot walls of the nuclear steam supply system. The inaccessible areas encompass the “equipment compartments.” The accessible areas are cooler and encompass the “service compartments” away from the hot equipment.

The minimum initial containment temperature value of 86°F, given in FSAR Table 6.2.1-4, represents the highest expected temperature of the accessible areas (service compartments). The maximum initial containment temperature value of 131°F, given in FSAR Table 6.2.1-4, represents the highest temperature in the inaccessible areas (equipment compartments). FSAR Table 6.2.1-4 will be clarified to make this distinction.

Heat sinks were assigned input initial temperatures based on their physical location inside containment. Higher initial temperatures were used within a given location to be conservative because higher temperatures result in lower heat sink absorption and, thereby, higher peak containment pressures.

The heat sinks located inside the inaccessible areas were conservatively assigned the GOTHIC input value (initial temperature) of 131°F, corresponding to the highest temperature encountered in that space during power operation. The heat sinks located inside the accessible area were assigned the GOTHIC input value of 86°F corresponding to the highest expected temperature encountered in that space during power operation.

The IRWST heat sinks were assigned an initial temperature that is an average of the accessible and inaccessible temperatures (i.e., 108.5°F). This GOTHIC input value used for the IRWST heat sinks is conservative because the IRWST is at the lowest point inside containment (i.e., below equipment locations).

The accessible area temperature (86°F) was used as the GOTHIC input for initial containment temperature. This input selection is conservative because it results in a larger mass of non-condensables and, thereby, a reduction in the Uchida condensing heat transfer coefficient assumed for the conductor surfaces exposed to steam due to the decrease in steam/air mass ratio; thus, leading to higher peak containment pressures.

Response to Question 6.2.1-05d:

The GOTHIC input assumes all four accumulators release nitrogen into the containment.

The accumulator non-condensable (i.e., nitrogen) cover gas mass and energy release rates are supplied as input boundary condition forcing functions in the GOTHIC models. In GOTHIC, a constant mass flow rate was assigned. The flow rate is calculated based on the conservative assumption that the available nitrogen within the accumulators is released in 20 seconds. The nitrogen release rate is calculated as follows:

$$massN_2Accum = 4\frac{(652.7\,psia - 14.7\,psia)(1942.5\,ft^3 - 1324.3\,ft^3)}{55.15\frac{ft*lbf}{lbm*R}(90.5^o\,F + 459.67R)}\left(144\frac{in^2}{ft^2}\right)$$

$$massN_2Accum = 7487.4\ lbm$$

$$massN_2FlowRate = \frac{7487.4\ lbm}{20\sec} = 374.4\frac{lbm}{\sec}$$

Where:

Number of Accumulators is	4
Accumulator initial pressure is	652.7 psia
Accumulator total volume is	1942.5 ft^3
Accumulator initial liquid volume is	1324.3 ft^3
Accumulator normal operating temperature is	90.5°F
Gas constant of nitrogen is	55.15 ft-lbf/lbm-R

The accumulator nitrogen injection calculated above was implemented in the GOTHIC model by Boundary Condition 12 with mass flow rate specified by Function 26, which is a constant 374.4 lbm/sec for the first 20 seconds and zero thereafter.

Table 6.2.1-05-1—Passive Heat Sink Material Properties

Material	Density kg/m^3	Thermal Conductivity W/m-°K	Heat Capacity J/kg-°K
Stainless steel	7850	16	500
Carbon steel	7850	45	500
Concrete	2400	1.75	960
Paint	1200	0.174	1674
Air gap	1.182	0.028	1004

FSAR Impact:

FSAR, Tier 2, Table 6.2.1-4, Table 6.2.1-5, and Section 6.2.1.1.3 will be revised as described in the response and indicated on the enclosed markup.

Question 6.2.1-06:

(FSAR Section 6.2.1) To enable the staff to better compare the results from audit calculations with AREVA's containment analysis results for US-EPR, please provide clarification for the following:

a. FSAR Table 6.2.1-3 lists different operating characteristics for CCW trains 1 &4 from those of 2 &3. Which of the four CCW trains listed is used in the long-term GOTHIC peak containment pressure analyses?

b. FSAR Table 6.2.1-8, the peak pressure row indicates units of "psia" but the figures (6.2.1-18 and 6.2.1-20) infer the pressures should be "psig;" please correct the apparent inconsistencies.

c. FSAR Tables 6.2.1-7 and 6.2.1-8, specifically Cases 14D, 28 and 31, the IRWST initial temperatures are not 122F as are the other cases, please explain the basis for these specified temperatures of 170F and 248F.

Response to Question 6.2.1-06a:

The long-term GOTHIC model includes only trains 1 and 4. This results in conservative heat exchanger performance because of the smaller heat transfer surface area as compared to trains 2 and 3.

Response to Question 6.2.1-06b:

The peak pressure values currently specified in FSAR Table 6.2.1-8 are units of psig, although the table row label indicates units of psia. The peak pressure values in the table will be changed to units of psia as shown in the enclosed markup.

Response to Question 6.2.1-06c:

The different in-containment refueling water storage tank (IRWST) temperatures used in Cases 14D, 28, and 31 are for the sensitivity study on the emergency core cooling system (ECCS) water source temperature.

In the RELAP5 model, a constant value was assigned as ECCS source temperature during the transient. In the U.S. EPR design, the ECCS pumps water from the IRWST, which has a variable temperature during the transient because of the high energy break flow discharged from the break into containment.

The constant IRWST temperature of 248°F used in Case 14D and Case 28 is the design temperature of the sump suction line. The constant IRWST temperature of 170°F used in Case 31 bounds the maximum medium head safety injection and low head safety injection mass weighted injection temperature during the transient.

FSAR Impact:

FSAR, Tier 2, Table 6.2.1-8 will be revised as described in the response and indicated on the enclosed markup.

Question 6.2.1-07:

(FSAR Section 6.2.1) In order to facilitate the review, the NRC staff needs certain design information as soon as possible. These include: (a) The design of rupture foils and convection foils; (b) details on the modeling of the containment in the multi-node GOTHIC calculations; and (c) detailed results for one of the multi-node GOTHIC calculations. (See detail below.)

a. Foils are installed in a steel framework. Does the framework separate the foil into many foils each of which has to rupture individually? What is the size of the individual foils? What is the total surface area of the foils? What is the available flow area once the foils rupture? Justify this flow area. What materials are the foils made of? What is the thickness of the foils? What is the weight of the foils per square foot? The above questions apply to both rupture foils and convection foils. In addition, how many fusible links are on the frame of a convection foil? Where are the links located?

b. Please provide a simplified sketch of the containment. Show internal walls, major components (steam generators, tanks, and so on) and the location and size of all mixing dampers, rupture foils and convection foils.

c. Provide the noding diagram that was used in the multi-node GOTHIC calculations (pages 47-50 of the U.S.-EPR Design Certification Acceptance Review presentation by AREVA of January 29, 2008). Provide the input data used in these calculations: volumes, elevations, cross sections, flow path dimensions, heat transfer surfaces. What was the break location and break size selected for the above referenced multi-node GOTHIC calculations?

d. Provide for one of the multi-node calculations (LB LOCA cold leg break if available) sufficient details of the results to permit visualization of flow patterns in the containment as well as heat transfer to the various heat sinks. Results should be given as a function of time for the duration of the accident. Please include flow in each flow path; content, temperature and pressure of each node; surface temperatures of significant heat sinks and heat transfer to each significant heat sink.

Response to Question 6.2.1-07a:

The rupture and convection foils are installed in a steel framework above each steam generator and are separated into individual foils according to the layout provided in Figure 6.2.1-07-1. The types and sizes of the individual foils are listed in Table 6.2.1-07-1 including the flow areas.

The rupture and convection foils are made of austenitic steel with an intermediate layer of plastic to establish the compartmental atmospheric seal during normal plant operation. In addition, the convection foils include a thermo lock, which is a fusible link made of brass and solder with a pre-defined melting point.

The thickness and dimensions of the foils will be designed to meet the functional requirements. Once the thickness is defined, the weight per square foot can be determined. The design of the foils will meet the following functional requirements:

- Rupture foils open on differential pressure > 50 mbar.
- Convection foils open on differential pressure > 50 mbar OR temperature > 80–85°C.

There is one fusible link (thermo-lock) on each convection foil.

The general arrangement between bursting element and fusible link is shown in a profile view of a convection foil as shown in Figure 6.2.1-07-2 in both the opened (vertical) and closed (horizontal) positions.

Response to Question 6.2.1-07b:

A simplified sketch of the containment is shown in Figure 6.2.1-07-3. The sketch shows the internal walls, major equipment, mixing damper locations, rupture foils, convection foils, and heat transfer parameters of interest.

The symbols in the sketch are defined as follows:

Q = heat transfer,
Γ = condensation,
ρ = density,
T = temperature, and
m = mass flow.

The subscripts in the sketch identify regions of the containment and are defined as follows:

A = heavy floor,
B = lower equipment room,
C = IRWST,
D = upper equipment room,
E = upper accessible area, and
E' = lower accessible area.

Response to Question 6.2.1-07c:

The geometric and hydraulic data used to construct the single node and multi-node GOTHIC models are provided in Table 6.2.1-07-2. The passive heat sink data are provided in FSAR Table 6.2.1-5.

The Reactor Building rooms identified in Table 6.2.1-07-2 were combined to obtain the lumped GOTHIC nodes. The vent paths connecting the Reactor Building rooms identified in Table 6.2.1-07-3 were combined to obtain the flow paths between the GOTHIC nodes.

The noding diagram that was used in the multi-node GOTHIC calculations and the postulated break sizes and locations will be provided as part of the response to RAI 6.2.1-07 part d.

Reactor Building rooms are shown in FSAR Figure 3.8-1 through Figure 3.8-13.

Response to Question 6.2.1-07d:

The Technical Report (TR) described in the response to Question 6.2.1-01 will provide details of the requested multi-node calculation. The results will be provided as a function of time for the duration of the accident, and will include flow in each flow path, content, temperature and pressure of each node, surface temperatures of significant heat sinks and heat transfer to each significant heat sink. The TR will be provided by January 28, 2009.

AREVA NP Inc. plans to have progressed sufficiently in the development of the TR by the end of June 2008 to support further discussions with the NRC staff.

FSAR Impact:

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

Table 6.2.1-07-1—Dimensions of Rupture and Convection Foils

Type	Cross-sectional Flow Area	Quantity (Per SG-Pressure Equalization Ceiling)
I	0.3721 m²	26
II	0.2867 m²	24
III	0.1508 m²	6
IV	0.1251 m²	2
V	Blind sheets (do not open)	36

Table 6.2.1-07-2—Containment Sub-Compartments – Nominal Free Volume Data

Room Name (30UJA)	Description	From Elev. ft	To Elev. ft	Free Volume ft^3
Spreading Rooms				
04-002	Spreading area	-20.67	-12.14	17,655.37
07-017	Venting area for spreading area	-12.14	20.47	2401.13
In-containment Refueling Water Storage Tank				
04-003	In-containment refueling water storage tank	-22.47	0.00	53,637.01
Reactor Pressure Vessel Pit				
11-001	Reactor pressure vessel pit	-20.67	24.41	3001.412
Components				
04-005	Compartment for flooding device	-21.16	-7.55	441.3842
04-006	Compartment for flooding device	-21.16	-7.55	441.3842
07-020	KTC floor drain 1	-7.55	2.95	3495.763
07-021	KTA heat exchanger	-7.55	2.95	1221.751
07-022	KTA pumps	-7.55	2.95	3142.655
07-023	KTD floor drain 2	-7.55	2.95	2789.548
07-024	KTA KTB tanks	-7.55	2.95	3319.209
07-026	KBA cooler	-7.55	2.95	1765.537
07-027	KBA cooler	-7.55	2.95	1800.847
07-028	KBA valves	-7.55	2.95	1553.672
07-029	KBA valves	-7.55	2.95	1377.119
11-022	KBA valves	4.92	20.47	2012.712
11-023	KBA valves	4.92	16.08	3495.763
11-024	KBA regenerative HX	4.92	15.26	2199.859
Steam Generator Blowdown (LCQ) HX Etc.				
04-004	KT sump	-20.18	-7.55	321.3277
07-018	Steam generator blowdown (LCQ) HX	-7.55	2.95	12,040.96
07-019	KT sump	-7.55	2.95	275.4237
Lower Annulus Rooms L1&2				
07-014	[ ]	[ ]	[ ]	22,951.98
Lower Annulus Rooms L3&4				
07-015	[ ]	[ ]	[ ]	22,951.98
Hot Piping				
07-016	[ ]	[ ]	[ ]	22810.73
Access				
07-013	[ ]	[ ]	[ ]	9745.763
11-020	[ ]	[ ]	[ ]	9357.345

Table 6.2.1-07-2—Containment Sub-Compartments – Nominal Free Volume Data

Room Name (30UJA)	Description	From Elev. ft	To Elev. ft	Free Volume ft^3
Elevator				
04-012	Elevator	[ ]	[ ]	233.0508
07-012	Elevator	[ ]	[ ]	628.5311
11-012	Elevator	[ ]	[ ]	603.8136
15-012	Elevator	[ ]	[ ]	586.1582
18-012	Elevator	[ ]	[ ]	843.9266
23-012	Elevator	[ ]	[ ]	942.7966
29-012	Elevator	[ ]	[ ]	759.1808
34-012	Elevator	[ ]	[ ]	677.9661
Surge Line, Below				
11-018	Steam generator blowdown (LCQ) flash tank	4.92	19.95	9110.169
11-019	JEG Pressurizer relief tank	4.92	19.95	4908.192
15-018	Spray lines	21.92	31.17	4731.638
15-019	Surge line	21.92	31.17	5568.503
18-018	Spray valves	32.15	47.31	8757.062
18-019	Surge line	32.15	49.11	8580.508
Pressurizer				
23-019	Pressurizer	49.11	65.29	6762.006
23-041	Instrument measuring table	49.11	61.35	2612.994
29-019	Pressurizer	67.91	81.56	5473.164
34-019	Pressurizer safety relief valves	83.20	91.86	4519.774
Lower Equipment Rooms L1				
11-002	Reactor coolant pump loop 1	4.92	15.22	6899.718
11-003	Steam generator loop 1	4.92	16.90	7768.362
15-002	Reactor coolant pump loop 1	15.22	24.08	3548.729
15-003	Steam generator loop 1	15.22	30.77	7980.226
Lower Equipment Rooms L2				
11-004	Steam generator loop 2	4.92	16.90	7768.362
11-005	Reactor coolant pump loop 2	4.92	15.22	5840.395
15-004	Steam generator loop 2	15.22	30.77	7980.226
15-005	Reactor coolant pump loop 2	15.22	24.08	3283.898
Lower Equipment Rooms L3				
11-006	Reactor coolant pump loop 3	4.92	15.22	5840.395
11-007	Steam generator loop 3	4.92	16.90	7768.362
15-006	Reactor coolant pump loop 3	15.22	24.08	3283.898
15-007	Steam generator loop 3	15.22	30.77	7980.226
Lower Equipment Rooms L4				
11-008	Steam generator loop 4	4.92	16.90	7768.362
11-009	Reactor coolant pump loop 4	4.92	15.22	6899.718

Table 6.2.1-07-2—Containment Sub-Compartments – Nominal Free Volume Data

Room Name (30UJA)	Description	From Elev. ft	To Elev. ft	Free Volume ft^3
15-008	Steam generator loop 4	15.22	30.77	7980.226
15-009	Reactor coolant pump loop 4	15.22	24.08	3495.763
Middle Annulus Rooms L1&2				
11-013	Annular space loop1	4.92	16.90	10,911.02
11-014	Annular space loop2	4.92	16.90	16,207.63
11-021	FAL valves	4.92	15.26	1447.74
11-025	JND-JNG valves loop 1	4.92	15.26	1945.621
11-026	JND-JNG valves loop 2	4.92	15.26	1677.26
11-031	Access loops 1 & 2	4.92	12.17	741.5254
15-013	Annular space accumulator tank loop1	16.90	28.54	15,148.31
15-014	Annular space accumulator tank loop2	16.90	28.54	17,584.75
15-020	[ ]	[ ]	[ ]	2330.508
15-021	[ ]	[ ]	[ ]	9427.966
15-025	FAL valves	16.90	26.90	1666.667
18-013	Annular space accumulator tank loop1	28.54	45.28	27,259.89
18-014	Annular space accumulator tank loop2	28.54	45.28	27,718.93
23-013	Annular space accumulator tank loop1	45.28	63.98	23,658.19
23-014	Annular space accumulator tank loop2	45.28	63.98	24,392.66
23-042	Instrument measuring cabinet	49.11	61.35	4590.395
Middle Annulus Rooms L3&4				
11-015	Annular space loop 3	4.92	16.90	17,196.33
11-016	Annular space loop 4	4.92	16.90	11,970.34
11-027	JND-JNG valves loop 3	4.92	15.26	1677.26
11-028	JND-JNG valves loop 4	4.92	15.26	1945.621
11-032	Access loops 3 & 4	4.92	12.17	741.5254
15-015	Annular space accumulator tank loop3	16.90	28.54	18,820.62
15-016	Annular space accumulator tank loop4	16.90	28.54	18,926.55
18-015	Annular space accumulator tank loop3	28.54	45.28	28,707.63
18-016	Annular space accumulator tank loop4	28.54	45.28	29,378.53
23-015	Annular space accumulator tank loop3	45.28	63.98	25,105.93
23-016	Annular space accumulator tank loop4	45.28	63.98	26,087.57
Staircase (South)				
11-010	Staircase (south)	[ ]	]	1550.141
15-010	Staircase (south)	[ ]	]	1504.237
18-010	Staircase (south)	[ ]	]	2161.017
23-010	Staircase (south)	[ ]	]	1998.588
29-023	[ ]	[ ]	]	7097.458
Staircase (North)				
15-011	Staircase (north)	[ ]	[ ]	2182.203
18-011	Staircase (north)	[ ]	[ ]	3135.593

Table 6.2.1-07-2—Containment Sub-Compartments – Nominal Free Volume Data

Room Name (30UJA)	Description	From Elev. ft	To Elev. ft	Free Volume ft^3
23-011	Staircase (north)	[ ]	[ ]	3495.763
29-011	Staircase (north)	[ ]	[ ]	2814.266
34-011	Staircase (north)	[ ]	[ ]	1200.565
	Reactor Cavity			
15-001	Reactor cavity	24.41	60.53	29,943.5
15-017	Core internals storage	24.41	60.53	21,751.41
15-024	[ ]	[ ]	[ ]	455.5085
	Middle Equipment Rooms L1			
18-002	Reactor coolant pump loop 1	24.08	45.28	9004.237
18-003	Steam generator loop 1	30.77	45.28	8792.373
23-002	Reactor coolant pump loop 1	45.28	60.70	6754.944
23-003	Steam generator loop 1	45.28	64.80	7824.859
23-017	KLA 6 (containment ventilation)	45.28	61.35	6355.932
23-020	FAL (fuel pool purification)valves	45.28	60.70	1991.525
	Middle Equipment Rooms L2			
18-004	Steam generator loop 2	30.77	45.28	8792.373
18-005	Reactor coolant pump loop 2	24.08	45.28	8474.576
23-004	Steam generator loop 2	45.28	64.80	11,052.26
23-005	Reactor coolant pump loop 2	45.28	63.98	11,052.26
	Middle Equipment Rooms L3			
18-006	Reactor coolant pump loop 3	24.08	45.28	8474.576
18-007	Steam generator loop 3	30.77	45.28	8792.373
23-006	Reactor coolant pump loop 3	45.28	63.98	7824.859
23-007	Steam generator loop 3	45.28	64.80	11,052.26
	Middle Equipment Rooms L4			
18-008	Steam generator loop 4	30.77	45.28	8792.373
18-009	Reactor coolant pump loop 4	24.08	45.28	7627.119
23-008	Steam generator loop 4	45.28	64.80	11,052.26
23-009	Reactor coolant pump loop 4	45.28	60.70	6754.944
23-018	KLA 6 (containment ventilation)	45.28	61.35	6355.932
23-031	FAL skimming pump	45.28	60.70	1468.927
	Upper Equipment Rooms L1&2			
29-003	Steam generator loop 1	64.80	79.07	5261.299
29-004	Steam generator loop 2	64.80	79.07	5261.299
29-005	Reactor coolant pump loop 2	63.98	79.07	4237.288
34-003	Steam generator loop 1	79.07	103.51	9869.35
34-004	Steam generator loop 2	79.07	103.51	9869.35
34-005	Reactor coolant pump loop 2	79.07	91.86	3407.486

Table 6.2.1-07-2—Containment Sub-Compartments – Nominal Free Volume Data

Room Name (30UJA)	Description	From Elev. ft	To Elev. ft	Free Volume ft^3
Upper Equipment Rooms L3&4				
29-006	Reactor coolant pump loop 3	63.98	79.07	4237.288
29-007	Steam generator loop 3	64.80	79.07	5261.299
29-008	Steam generator loop 4	64.80	79.07	5261.299
34-006	Reactor coolant pump loop 3	79.07	91.86	3407.486
34-007	Steam generator loop 3	79.07	103.51	9869.35
34-008	Steam generator loop 4	79.07	103.51	9869.35
Upper Annulus Rooms L1&2				
29-014	Annular space (240 deg–0 deg)	63.98	77.43	31,250
29-018	Operating floor access	63.98	75.79	6850.282
34-014	Annular space (240 deg–0 deg)	79.07	93.50	34,957.63
34-018	Reactor pressure vessel closure head storage area	79.07	93.50	9004.237
Upper Annulus Rooms L3&4				
29-015	Annular space (0 deg–120 deg)	63.98	77.43	25,070.62
29-022	KLA (containment ventilation) equipment	63.98	77.43	8121.469
34-015	Annular space (0 deg–120 deg)	79.07	93.50	24,540.96
34-022	KLA (containment ventilation) equipment	79.07	91.86	7697.74
Lower and Upper Dome L1, L2, L3 & L4				
15-023	Instrumentation lance storage	20.67	63.98	278.9548
29-013	Set down area operating floor	63.98	93.50	96,186.44
29-016	[ ]	[ ]	[ ]	106,991.5
34-104	Steam generator L1&2 above rupture disks and dampers	103.51	113.02	11,052.26
34-108	Steam generator L3&4 above rupture disks and dampers	103.51	113.02	11,052.26
40-001	Dome	93.50	188.68	1,390,890

Table 6.2.1-07-3—Containment Sub-Compartments – Flow Path Data

Table 6.2.1-07-3—Containment Sub-Compartments – Flow Path Data

Table 6.2.1-07-3—Containment Sub-Compartments – Flow Path Data

Table 6.2.1-07-3—Containment Sub-Compartments – Flow Path Data

Table 6.2.1-07-3—Containment Sub-Compartments – Flow Path Data

Table 6.2.1-07-3—Containment Sub-Compartments – Flow Path Data

Table 6.2.1-07-3—Containment Sub-Compartments – Flow Path Data

Table 6.2.1-07-3—Containment Sub-Compartments – Flow Path Data

Table 6.2.1-07-3—Containment Sub-Compartments – Flow Path Data

Table 6.2.1-07-3—Containment Sub-Compartments – Flow Path Data

Table 6.2.1-07-3—Containment Sub-Compartments – Flow Path Data

Figure 6.2.1-07-1—Overhead View of Rupture and Convection Foils

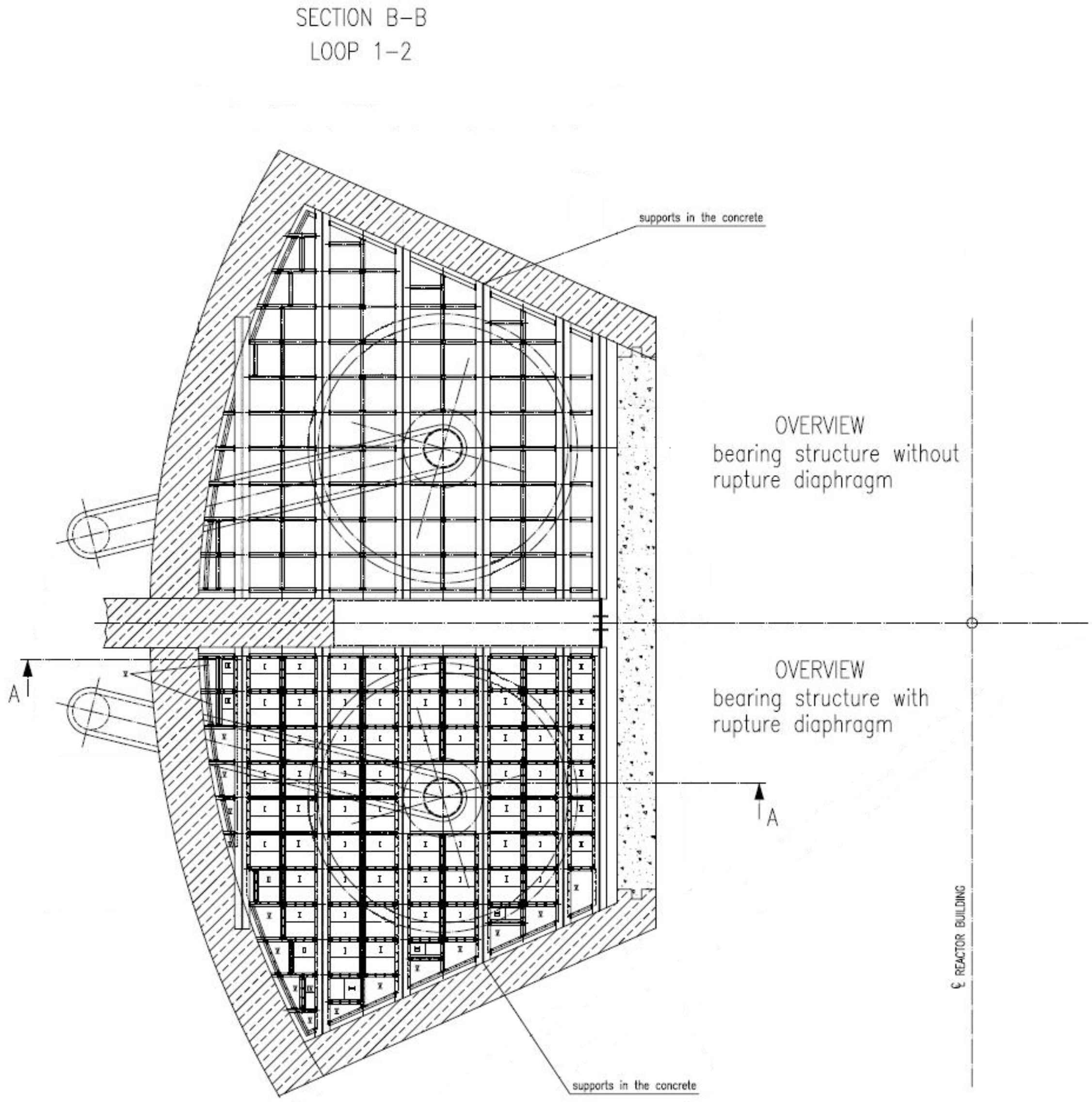


Figure 6.2.1-07-2—Side View of Convection Foil

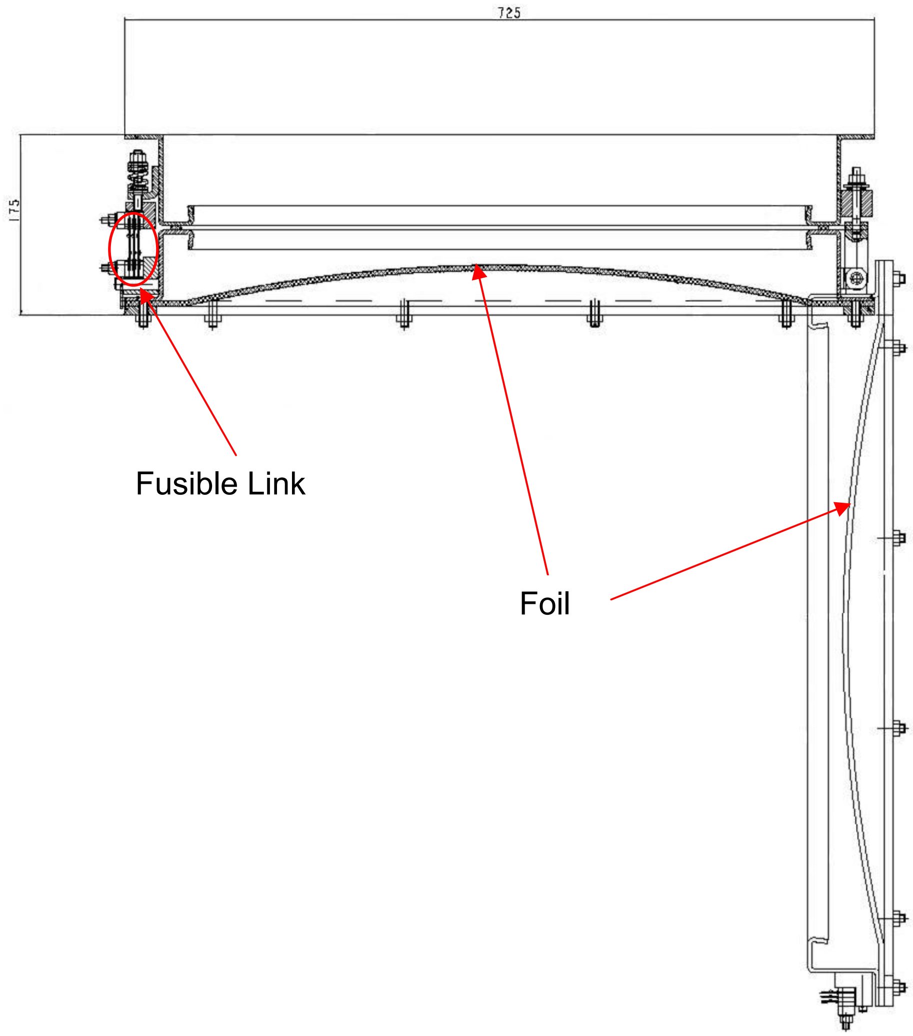
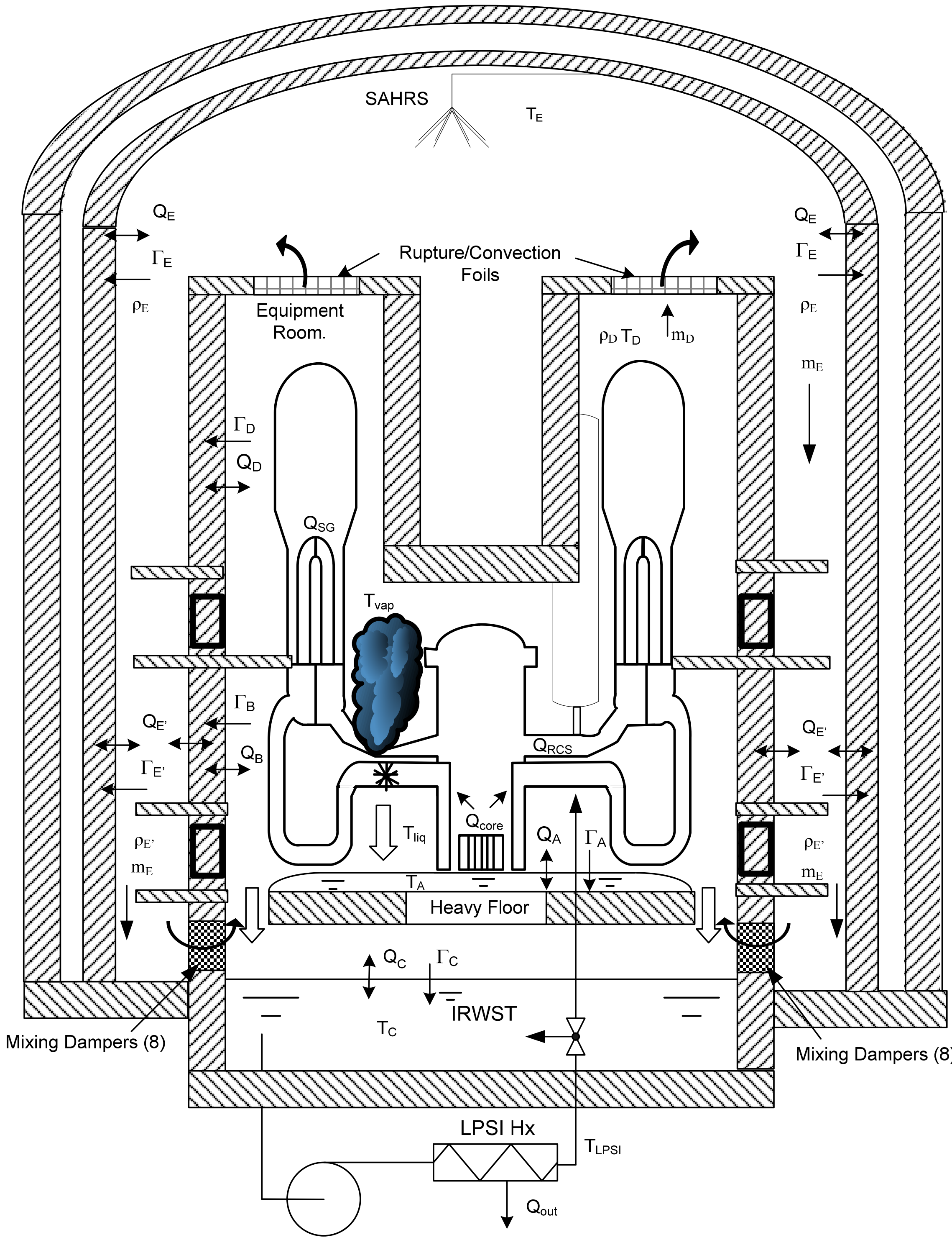


Figure 6.2.1-07-3—Locations of Mixing Damper, Rupture Foils, and Convection Foils



Question 6.2.1-08:

FSAR Section 6.2.1.2.3 states that the Homogeneous Equilibrium Model (HEM) is utilized in several of the cases to calculate the mass flux into subcompartments following a postulated piping break. The HEM is generally not acceptable for the calculation of liquid and two-phase critical flow from a piping break because it under predicts critical mass flow rates. Please provide subcompartment analyses using break flow models which are conservative for the prediction of critical flow.

Response to Question 6.2.1-08:

Of the eleven critical rooms analyzed for sub-compartment pressurization, critical flow rates for eight rooms were determined using the HEM. The remaining three rooms were analyzed using the system analysis code CRAFT2 which uses the Moody model in the saturated region and the Modified Zoludek in the subcooled region.

Commonly used alternatives to HEM are:

- Moody model in the two-phase and steam regions.
- Henry-Fauske model in the subcooled and saturated water regions.

The Moody and Henry Fauske critical flow models have tendencies to generally predict higher flow rates than the HEM. Figure 6.2.1-08-1 shows a comparison of predictions of HEM, Moody and Henry-Fauske models at 2200 psia as well as approximate conditions of two representative critical rooms. Figure 6.2.1-08-1 also shows that the difference between the critical flow models is most pronounced in the transition from subcooled to saturated conditions and gradually vanishes at either extreme (i.e., under highly subcooled water or superheated steam conditions). Unlike the Moody or Henry-Fauske models, the HEM covers the entire range of fluid states from subcooled water to superheated steam conditions. The tendency observed in Figure 6.2.1-08-1 is also observed at other pressures as illustrated in Figure 6.2.1-08-2, which shows a comparison of the critical flow models at 1200 psia.

Both the Moody and Henry-Fauske models are appropriate for determining the discharge of large vessels through short nozzles or orifices where slip and phase non-equilibrium effects come into play. For discharges through long pipes, the HEM offers a more realistic approach because with increasing travel distance, flow approaches homogeneous equilibrium conditions. This is confirmed in an EPRI study of critical flow models which concludes that for L/D>1.5, HEM produces good agreement with test data [Ref. 1]. Similarly, in a topical book co-authored by F. J. Moody, it is stated that, “The Moody model predicts equilibrium flowrates somewhat higher than data representative of [boiling water reactor] BWR blowdowns…" The author further states that the homogeneous equilibrium model agrees quite well with the available data [Ref. 2]. This confirmation applies to pressurized water reactors because BWR conditions also exist in pressurized water reactor systems.

Regardless of the appropriateness of HEM, the pressure response has been evaluated by considering the rooms in three distinct categories: rooms analyzed by CRAFT2, rooms with negligible pressure rise, and rooms with small pressure rise. The evaluation for each category is as follows:

Rooms Analyzed with CRAFT2 (Rooms 15-006, 23-004, 29-004)

These rooms were analyzed with the system analysis code CRAFT2 which uses Moody in the two-phase and steam regions, and Modified Zoludek in the subcooled and saturated water regions. Accordingly, they are excluded from further justification.

Rooms with Negligible Pressure Rise (Rooms 11-004, 11-006, 11-027, 15-004, 18-004, 34-004)

These rooms experience a subcompartment pressure rise of less than 1.4 psi. With the exception of rooms 18-004 and 34-004, all of these rooms have high energy lines under subcooled water conditions. Therefore, their associated critical flow rates would increase negligibly by using Henry-Fauske instead of HEM. Room 18-004 contains a high energy line with saturated water that will see a significant increase in its critical flow using Henry-Fauske instead of HEM. Room 34-004 contains a saturated steam line and would exhibit a nearly negligible increase in critical flowrate.

Regardless of the increase in critical flowrate, the impact on subcompartment pressure rise is negligible. Because the room pressurization is primarily due to inertia effects, an adjusted pressure rise can be estimated by linear interpolation of pressure as a function of flowrate. The change in critical flow for these rooms still predicts a pressure rise of less than 1.4 psi. Therefore, it can be concluded that the analytical results and conclusions for these rooms are both reasonable and acceptable.

Rooms with Small Pressure Rise (Rooms 29-019, 34-019)

These rooms contain high energy lines carrying saturated steam and experience a pressure rise of roughly 4 psi. The increase in critical flowrate in going from the HEM to the Moody model is about 5%. With such a small increase in the critical flowrate, the adjusted pressure rise is still in the vicinity of 4 psi, representing an increase of roughly 0.2 psi from the HEM to the Moody model. Because this is a small increase and because these rooms are constructed similar to Room 29-019, which is qualified for a pressure rise of 16 psi, it can be concluded that the existing analytical results and conclusions for these rooms are both reasonable and acceptable.

References

1. EPRI NP-2192, “Critical Flow Data Review and Analysis,” Electric Power Research Institute, January 1982.
2. Lahey, R. T. and F. J. Moody, “The Thermal-Hydraulics of a Boiling Water Nuclear Reactor,” American Nuclear Society, 1979.

FSAR Impact:

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

Figure 6.2.1-08-1—Comparison of Critical Flow Models at 2200 psia

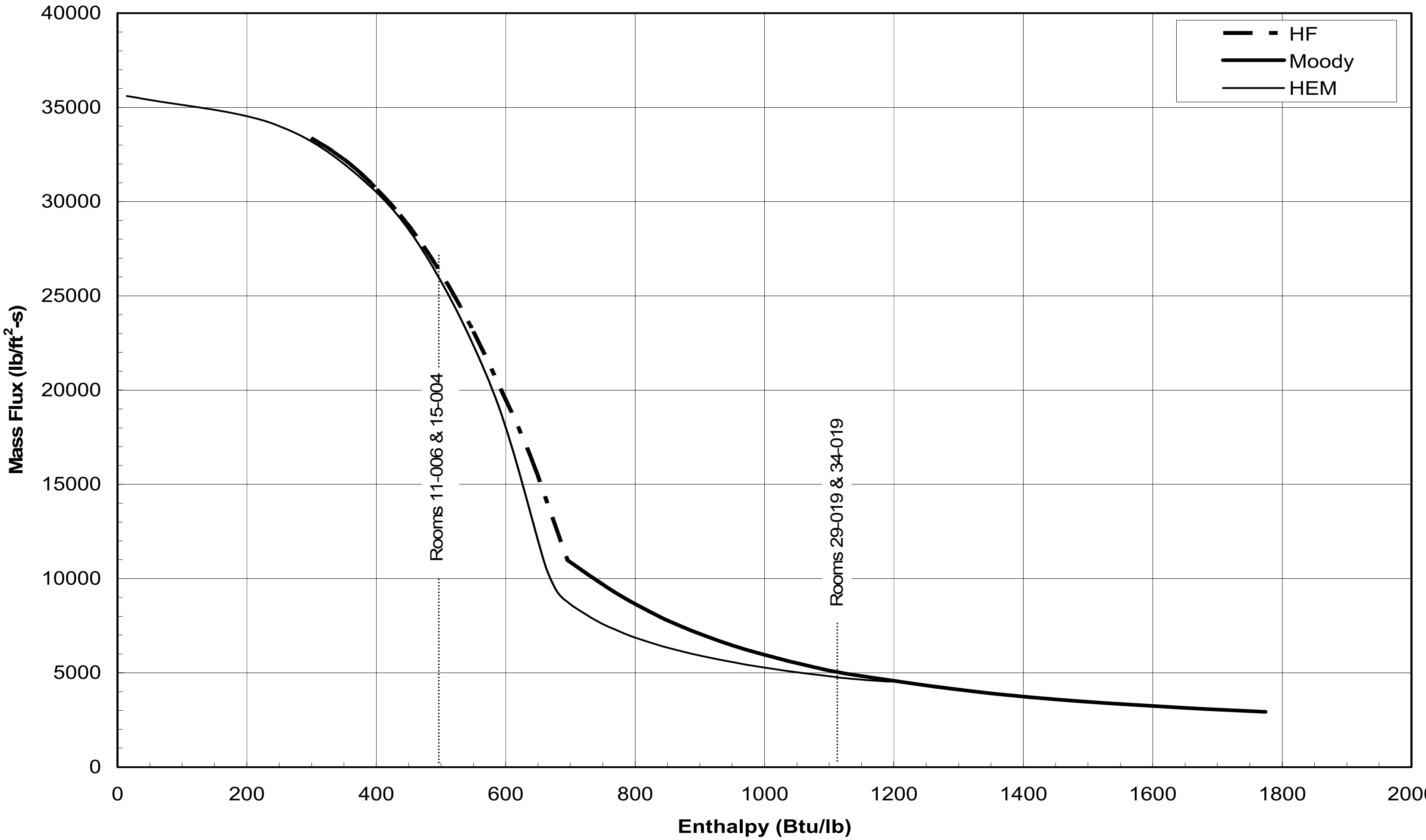
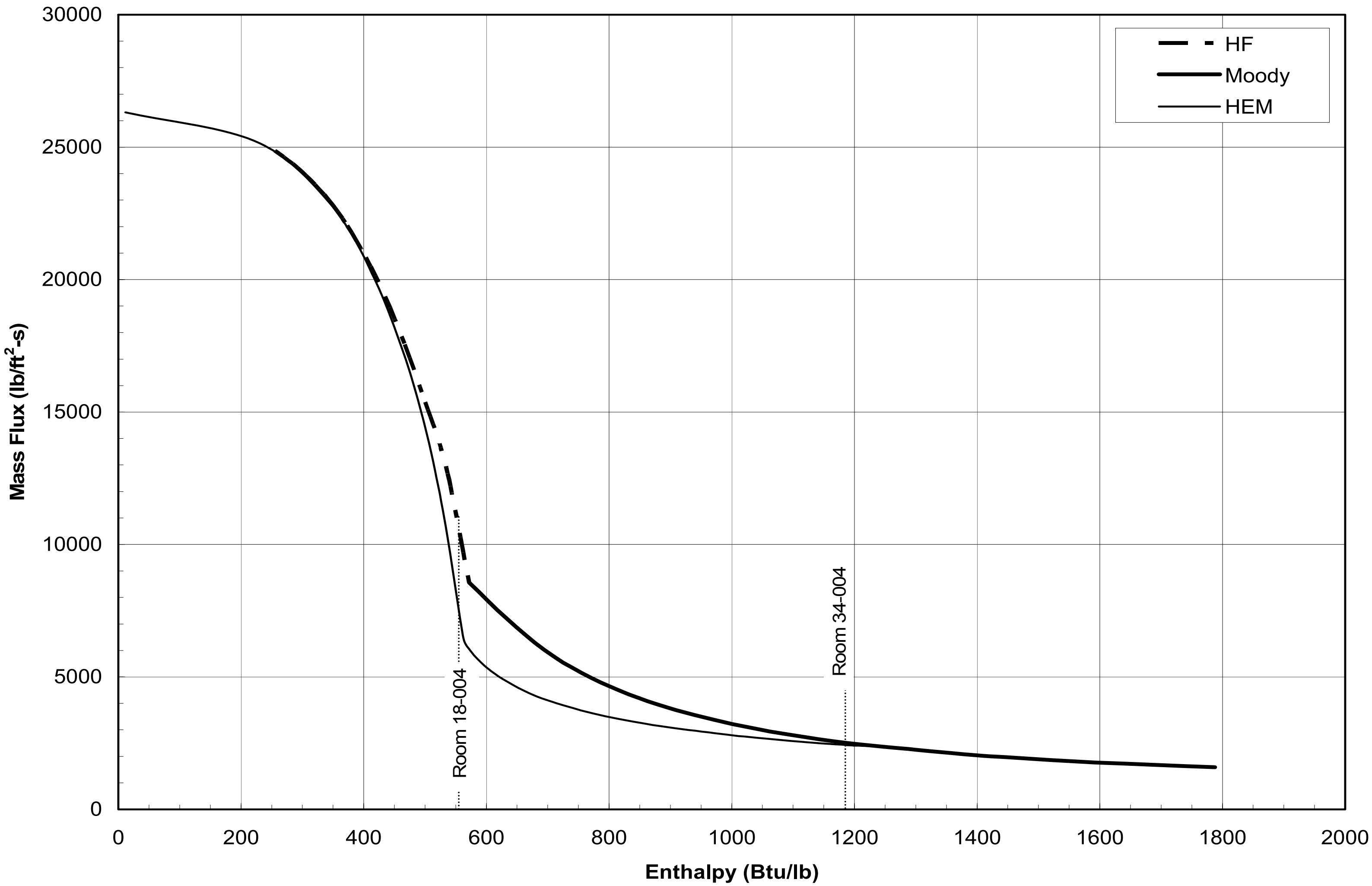


Figure 6.2.1-08-2—Comparison of Critical Flow Models at 1200 psia



Question 6.2.1-09:

(FSAR Section 6.2.1.5) For the evaluation used to determine minimum containment pressure for ECCS analysis, please provide the following information:

a. The mass and energy release input that was utilized to produce the containment pressure as a function of time in Figure 15.6-50. Include the spilled ECCS water and the nitrogen accumulator gas release.

b. Describe and justify assumptions made for the mixing between containment steam and the spilled ECCS water.

c. FSAR Section 15.6.5.1.2 indicates that a condensing heat transfer coefficient of 1.7 times that predicted by the Uchida correlation is used. Provide a comparison of the containment pressure result during the core reflood period between this assumption and that produced by the heat transfer model recommended in BTP 6.2 which is 4 times the Tagami correlation followed by 1.2 times the Uchida correlation. Provide this justification specifically for the US-EPR analysis given in Figure 15.6-50.

d. FSAR Section 6.2.1.5.3 states that heat transfer between the IRWST water and the containment vapor space is not (Error in FSAR) considered in the analysis. Rewrite paragraph and include how this heat transfer path is considered in a conservative manner for minimum containment pressure analysis.

Response to Question 6.2.1-09a:

The total mass and energy release input that was utilized to produce the containment pressure as a function of time in FSAR Figure 15.6-50 is provided in Table 6.2.1-09-1 The spilled ECCS water and the nitrogen accumulator gas release are included in the total mass and energy release as shown in the spreadsheet.

Response to Question 6.2.1-09b:

The containment model assumes thermodynamic equilibrium between the spilled ECCS water and containment steam. Consequently, the spilled ECCS water and containment steam are well mixed, which is expected as the large break LOCA progresses through blowdown and re-flood. Although the mixing between containment steam and spilled ECCS water is not a dominant phenomenon in determining the peak cladding temperature (PCT), it is conservative and bounding to assume thermodynamic equilibrium between the two phases. This assumption conservatively reduces containment pressure, which penalizes clad temperatures by increasing reactor coolant system voiding and break flow.

Response to Question 6.2.1-09c:

The realistic large-break LOCA (RLBLOCA) methodology described in the Topical Report, ANP-10278P, “U.S. EPR Realistic Large Break Loss of Coolant Accident Topical Report,” incorporates a best estimate containment pressure calculation based on heat transfer coefficients of 1.0 times Tagami and 1.2 times Uchida. Because the ICECON containment pressure calculation is performed concurrent with the S-RELAP5 blowdown calculation, it is not possible to use Tagami directly. The Tagami correlation requires advance knowledge of total energy transferred to the containment during blowdown and the time of end of blowdown. Figure 3-1 of ANP-10278P shows the process by which AREVA NP Inc. validates that the use of the Uchida correlation alone with a multiplier of 1.7 produces a pressure response that conservatively bounds the best estimate Tagami-Uchida formulation (1.0 times Tagami and 1.2 times Uchida).

Figure 6.2.1-09.1 compares these containment pressure responses for the case requested (FSAR Figure 15.6-50).

Figure 6.2.1-09.1 includes the corresponding pressure response calculated using 10 CFR 50, Appendix K rules (i.e., 4.0 times Tagami and 1.2 times Uchida).

Response to Question 6.2.1-09d:

The statement in FSAR Section 6.2.1.5.3 is incorrect, as noted in the question. The sentence “Heat transfer between the IRWST water and the containment vapor space is not considered in the analysis” will be replaced as indicated in the enclosed markup.

FSAR Impact:

FSAR, Tier 2, Section 6.2.1.5.3 will be revised as described in the response and indicated on the enclosed markup.

Table 6.2.1-09-1—Total Mass and Energy Release Input

Time, s	(Btu/s) <HFLOW>	(lbm/s) <BFLOW>
0.00	0	0
0.50	45360747	79958.28
1.00	43720108	76951.86
1.50	40566276	71133.87
2.00	37325196	65138.89
2.50	34186465	59368.05
3.00	31831708	54996.61
3.50	29251083	50300
4.00	27060927	46244.79
4.50	25027126	42396.95
5.00	22068743	36826.84
5.50	19136266	31557.06
6.00	18254631	29672.61
6.50	17510913	28068.67
7.00	16921404	26778.86
7.50	16281760	25393.21
8.00	15830925	24427.11
8.50	15508931	23795.8
9.00	15220552	23225.64

Table 6.2.1-09-1—Total Mass and Energy Release Input

Time, s	(Btu/s) <HFLOW>	(lbm/s) <BFLOW>
9.50	14937449	22666.56
10.00	14608662	21974.36
10.50	14250451	21211.11
11.00	13911056	20538.78
11.50	13530989	19710.26
12.00	13146051	18857.84
12.50	12788961	18137.22
13.00	12405975	17208.97
13.50	12029868	16313.16
14.00	11637157	15317.77
14.50	11265829	14326.63
15.00	10909531	13356.27
15.50	10529868	12514.36
16.00	10171872	12836.66
16.50	9593496	12295.83
17.00	9241732	12266.51
17.50	8945455	11940.07
18.00	8609935	11458.56
18.50	8201596	10510.84
19.00	7767492	9944.102
19.50	7178054	9360.553
20.00	6657798	8320.561
20.50	6207992	9155.84
21.00	5967179	9669.508
21.50	5665944	9766.291
22.00	5297438	9656.632
22.50	4762748	8936.203
23.00	4275287	8310.041
23.50	3892447	8024.504
24.00	3533384	7665.587
24.50	3175839	7262.815
25.00	2994190	7322.083
25.50	3026574	8356.756
26.00	2440227	7253.753
26.50	2293935	7200.881
27.00	3150827	11696.73
27.50	2888522	11231.21
28.00	2536994	9187.221
28.50	2428082	9819.08

Table 6.2.1-09-1—Total Mass and Energy Release Input

Time, s	(Btu/s) <HFLOW>	(lbm/s) <BFLOW>
29.00	1961004	8222.916
29.50	1771966	7786.387
30.00	1517304	7111.244
30.50	1336069	6657.291
31.00	1123216	5793.873
31.50	457565.8	3246.613
32.00	598106.1	4196.357
32.50	501344.4	3966.794
33.00	301567.6	2864.405
33.50	203415.4	2267.516
34.00	188649.8	2227.902
34.50	297719.6	2349.922
35.00	227621	2242.67
35.50	225167.5	2229.378
36.00	204786.2	2186.162
36.50	246294.5	2222.718
37.00	307533.8	2284.649
37.50	351050	2412.339
38.00	935554.5	5799.372
38.50	970732.4	6107.72
39.00	1135943	7485.929
39.50	918075.9	6311.237
40.00	950955.2	6706.57
40.50	754528.8	5491.769
41.00	596290.1	4427.492
41.50	486814.6	3613.919
42.00	397923.5	3053.952
42.50	459778.9	3363.372
43.00	573649.1	4158.575
43.50	856811.2	6189.687
44.00	1120619	8157.259
44.50	1057636	7717.34
45.00	1139453	8447.892
45.50	717514.1	5285.534
46.00	700461.1	5219.483
46.50	715471.3	5412.145
47.00	535958.1	4074.626
47.50	461023.7	3499.933
48.00	554236.3	4062.803

Table 6.2.1-09-1—Total Mass and Energy Release Input

Time, s	(Btu/s) <HFLOW>	(lbm/s) <BFLOW>
48.50	819180.2	6025.825
49.00	1145428	8383.744
49.50	1053414	7611.88
50.00	1109249	7944.943
50.50	1273004	9183.812
51.00	1044547	7454.506
51.50	1004201	7234.373
52.00	783596.7	5682.025
52.50	644311.9	4751.984
53.00	543607.3	4049.373
53.50	481791.8	3581.59
54.00	519429.4	3795.157
54.50	656514.5	4715.363
55.00	991529.2	6986.139
55.50	1109661	7786.977
56.00	1050539	7294.509
56.50	1237467	8685.254
57.00	997026	6979.979
57.50	933333.9	6552.768
58.00	936589.7	6580.666
58.50	810838.4	5697.275
59.00	714801	5056.031
59.50	592175.9	4214.319
60.00	543112.8	3875.503
60.50	505234.5	3605.308
61.00	565919.8	4022.4
61.50	530251.1	3778.112
62.00	724685.7	5052.716
62.50	853023.7	5874.485
63.00	837511.7	5749.536
63.49	1124569	7737.804
63.99	1424664	9791.33
64.49	1418605	9742.913
64.99	1470894	9916.971
65.49	1077593	6335.384
65.99	643213.3	2628.988
66.49	831328.9	4648.865
66.99	923483.8	5291.246
67.49	707862.5	2790.571

Table 6.2.1-09-1—Total Mass and Energy Release Input

Time, s	(Btu/s) <HFLOW>	(lbm/s) <BFLOW>
67.99	947595.4	4392.581
68.49	1239852	5694.658
68.99	1028931	4040.311
69.48	1001874	3464.646
69.98	1203150	4632.88
70.48	1200095	3769.397
70.98	1278637	3592.977
71.48	1252368	3590.75
71.98	1060120	2258.314
72.48	1090084	2206.838
72.98	1021986	1945.721
73.48	992456.6	1905.987
73.98	981811.7	1855.043
74.48	965923.6	1807.743
74.98	939408.5	1777.842
75.48	902420.7	1697.801
75.97	893271.4	1452.936
76.47	881531.3	1405.112
76.97	837802.1	1483.587
77.47	804757.5	1873.067
77.97	782776.4	1490.69
78.47	649667.3	1100.464
78.97	751453.5	1476.474
79.47	736994.5	1412.708
79.97	717084.3	1434.882
80.47	633622.8	1398.322
80.97	670479.3	1696.929
81.47	564820.8	1135.366
81.96	569505.5	1110.342
82.46	595701.3	1149.281
82.96	580676.7	1028.778
83.46	624119.3	1653.272
83.96	481679.2	1111.731
84.46	497896.9	829.3213
84.96	484838.5	834.1344
85.46	496761.9	956.754
85.96	425129.8	1001.219
86.46	418262.6	798.0462
86.96	416099.9	710.9605

Table 6.2.1-09-1—Total Mass and Energy Release Input

Time, s	(Btu/s) <HFLOW>	(lbm/s) <BFLOW>
87.46	425891.1	705.8352
87.96	409486.9	700.7247
88.45	398614.6	696.0653
88.95	388769.9	671.4886
89.45	375938.9	641.1912
89.95	362575.7	623.6384
90.45	358761.7	620.5811
90.95	358965.7	618.3748
91.45	353758.9	612.9991
91.95	353633.9	614.2716
92.45	366779	626.7433
92.95	359551.7	618.1738
93.45	349197.4	608.2011
93.95	341711.8	600.6025
94.44	331803.8	589.4436
94.94	321209.2	579.4755
95.44	331225.7	592.2112
95.94	323791	579.9542
96.44	316393	576.3155
96.94	349236.5	610.4535
97.44	333153.1	592.3053
97.94	326065.3	586.7927
98.44	323226.5	603.2555
98.94	333828.2	601.6689
99.44	335956.8	609.9893
99.94	329045.2	610.788
100.44	357269.8	729.967
100.93	449169.4	1060.911
101.43	394315.4	973.9308
101.93	491018.8	1438.623
102.43	408098.5	977.2027
102.93	340530.5	671.3989
103.43	321427.9	580.4753
103.93	321146.2	562.8738
104.43	312397	548.1218
104.93	308446.4	542.0078
105.43	344468.7	571.2733
105.93	324723.4	549.0128
106.43	309144.1	539.7338

Table 6.2.1-09-1—Total Mass and Energy Release Input

Time, s	(Btu/s) <HFLOW>	(lbm/s) <BFLOW>
106.92	309704.8	546.2018
107.42	299744.8	538.0428
107.92	286337.5	527.5321
108.42	284295.9	530.3774
108.92	274554.1	522.1173
109.42	284032.5	531.2051
109.92	287734.1	533.699
110.42	304097.8	547.8436
110.92	308116.6	551.5951
111.42	309930.5	553.1233
111.92	306485.7	550.1848
112.42	315051.8	559.0221
112.92	306547.1	554.1663
113.41	299168.2	550.7096
113.91	299266.3	554.2507
114.41	321159.8	576.2623
114.91	330353	582.4452
115.41	320004.2	573.9642
115.91	309887.4	566.9151
116.41	307398	568.0083
116.91	304818.4	567.9522
117.41	296329.7	560.6897
117.91	284877.4	551.3446
118.41	277947.1	548.2768
118.91	274275.9	545.7031
119.40	282934.5	559.7067
119.90	294772.6	565.6052
120.40	298599.1	566.5831
120.90	310724.8	572.4679
121.40	306048.8	567.5559
121.90	297152.5	562.208
122.40	290223	560.5948
122.90	288460.6	566.0727
123.40	282740.1	560.7657
123.90	330018.5	610.6159
124.40	311540.3	588.4912
124.90	303595.4	581.5423
125.40	301079.8	580.116
125.89	298421	581.8754

Table 6.2.1-09-1—Total Mass and Energy Release Input

Time, s	(Btu/s) <HFLOW>	(lbm/s) <BFLOW>
126.39	296939.6	583.3413
126.89	291027.5	579.9263
127.39	292490.5	586.1934
127.89	309383.6	600.9183
128.39	301204.5	591.9927
128.89	296403.4	587.6046
129.39	292774.5	586.3432
129.89	289169.6	586.3284
130.39	288014.8	586.3771
130.89	287374.7	586.1808
131.39	284316.9	583.2621
131.88	285666.8	587.9967
132.38	283423.9	568.7085
132.88	282199.7	549.3566
133.38	274830.7	534.1469
133.88	275598.2	535.2195
134.38	272997.2	543.1308
134.88	271077.9	551.3265
135.38	269079.9	553.7688
135.88	267806.6	550.3281
136.38	266016.3	547.4384
136.88	262752.2	544.5905
137.38	262268.4	547.4306
137.88	259064.2	548.3202
138.37	259392.7	551.7796
138.87	261068.9	553.2908
139.37	259673.2	551.9994
139.87	255691.3	548.7924
140.37	255592.3	549.0871
140.87	252137.4	547.8615
141.37	254457.1	552.5846
141.87	258154.4	555.6761
142.37	259047.9	557.5757
142.87	255241.5	553.5398
143.37	253730.1	551.3768
143.87	252461.6	551.4327
144.36	255533.7	555.2912
144.86	254999.7	557.4919
145.36	256780.5	559.9739

Table 6.2.1-09-1—Total Mass and Energy Release Input

Time, s	(Btu/s) <HFLOW>	(lbm/s) <BFLOW>
145.86	257562.7	561.7727
146.36	256917.1	559.6707
146.86	257790	561.0206
147.36	255576.8	559.2014
147.86	254615.7	560.1754
148.36	253960.2	560.5727
148.86	254552.3	562.1494
149.36	254477.3	562.8265
149.86	253542.4	561.6831
150.36	252856.6	560.9702
150.85	255597.2	564.6434
151.35	258486.5	566.6078
151.85	258287.3	567.05
152.35	257387.8	565.0259
152.85	256521.9	566.3003
153.35	252406	563.1241
153.85	252058.9	563.3166
154.35	249398.1	562.2113
154.85	252238.3	566.8981
155.35	245596.6	547.3456
155.85	244192	545.7689
156.35	248623	568.2708
156.84	249336.9	573.7845
157.34	260784	642.4537
157.84	269780.3	685.6745
158.34	272016	712.3367
158.84	269706.2	709.1302
159.34	271629	695.0633
159.84	270831.7	677.3798
160.00	269980.6	673.1631
160.50	264550.1	656.5573
161.01	266989.8	649.7394
161.51	263978.6	640.8227
162.02	263884.3	633.9134
162.52	266805.9	600.7989
163.02	262001.7	588.8399
163.53	257803.4	567.5701
164.03	254605.7	562.5307
164.54	250068.4	553.5318

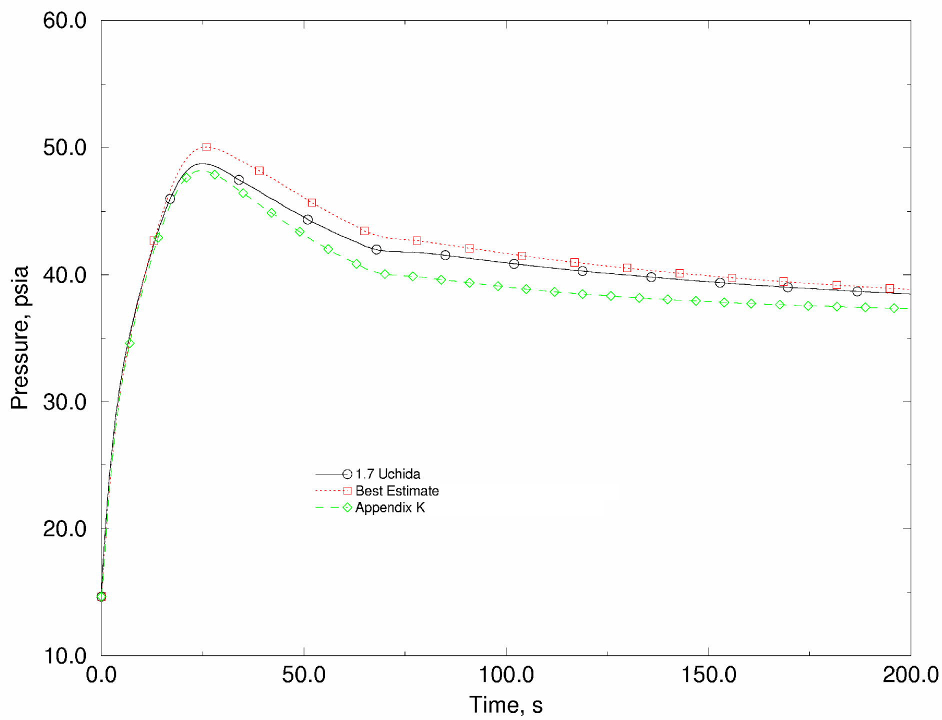
Table 6.2.1-09-1—Total Mass and Energy Release Input

Time, s	(Btu/s) <HFLOW>	(lbm/s) <BFLOW>
165.04	250305.6	553.8793
165.54	247537.7	550.1778
166.05	246232.4	547.8034
166.55	243184.5	543.2625
167.06	240723.8	541.0978
167.56	241184.5	539.8837
168.06	237713	535.318
168.57	239102.2	536.3103
169.07	242156.4	535.4554
169.58	244695.3	538.5901
170.08	248792.2	539.8138
170.58	252304	540.8019
171.09	246298.2	536.0193
171.59	242784.5	533.5589
172.10	238605.9	533.1742
172.60	238532.9	537.0002
173.10	240293.2	535.1809
173.61	239325.6	525.4067
174.11	236411.2	511.4512
174.62	234539.7	492.1516
175.12	232744.6	470.7186
175.62	233636.3	459.7402
176.13	231736.8	470.5145
176.63	232509.3	489.3869
177.14	231232	504.3904
177.64	232493.3	510.4346
178.14	231632.6	507.5275
178.65	230934.5	505.7307
179.15	231649.2	510.2572
179.66	233417	519.9175
180.16	235848.5	528.9299
180.66	235556.8	534.5842
181.17	236323	537.6279
181.67	235487.3	538.9727
182.18	236076.8	542.8549
182.68	238942.4	548.8564
183.18	239358.2	562.7546
183.69	242418.2	597.792
184.19	253037	636.1759

Table 6.2.1-09-1—Total Mass and Energy Release Input

Time, s	(Btu/s) <HFLOW>	(lbm/s) <BFLOW>
184.70	255048.7	664.2909
185.20	254696	670.429
185.70	252227.1	660.7477
186.21	249302.2	644.1462
186.71	246963.5	631.7017
187.22	244347.3	621.4756
187.72	248886	601.1444
188.22	245749.4	588.1517
188.73	242315	567.4781
189.23	241207.3	565.4551
189.74	238317.7	562.6784
190.24	237959.4	564.4491
190.74	239120.7	559.6783
191.25	236585.4	551.6777
191.75	236184.8	542.0155
192.26	233839.5	529.3838
192.76	229337.2	514.5755
193.26	227074.4	510.7707
193.77	225901.5	514.1438
194.27	227902.3	523.6717
194.78	237413	536.0621
195.28	240845.9	541.189
195.78	240937.2	538.0149
196.29	238890.1	535.6489
196.79	234660	533.1336
197.30	235174	539.7335
197.80	234711.7	542.6897
198.30	237221.4	549.0992
198.81	238045.9	549.0382
199.31	238686.5	548.1236
199.82	240486.8	549.7926
200.00	235018.5	546.7758

Figure 6.2.1-09-1—Comparison of Containment Pressure Histories for Case Identified in FSAR Figure 15.6-50



Question 6.2.1-10:

FSAR Section 6.2.3 describes the US-EPR Reactor Shield Building (RSB). Provide the evaluation for the RSB post LOCA pressure and temperature following design basis conditions. Provide the graphical results for the limiting case of the atmospheric pressure and temperature and the reactor containment building (RCB) wall temperature. The evaluation should include heat transfer from the RCB, the equipment heat loads within the RSB and the decrease in the RSB volume as the result of thermal and mechanical expansion of the RCB.

Response to Question 6.2.1-10:

An evaluation was performed to determine the thermal hydraulic response of the annulus between the RSB and the RCB following a design basis LOCA. This analysis is discussed in FSAR Section 6.2.3.3 and Table 6.2.3-2. Specifically, the analysis takes into account heat transfer from the RCB by conservatively assuming an infinite heat transfer coefficient. The concrete wall of the RCB is ≈3 feet thick; therefore, the actual RCB conditions and heat transfer from the RCB are unimportant in this evaluation. During a postulated LOCA, heat loads generated within the annulus are negligible; therefore, there is no need to account for them in the evaluation. The decrease in the annulus volume because of thermal and mechanical expansion of the RCB is conservatively modeled by step-reducing the annulus volume at the start of the event.

Figures 6.2.1-10-1 through 6.2.1-10-4 present the annulus pressure and temperature and RCB wall temperature for the limiting case. The analysis covers the first 24 hours after the postulated accident. Figure 6.2.1-10-1 shows the short term response of the annulus pressure at the lowest containment elevation. The annulus pressure initially increases before the annulus ventilation system (AVS) is activated, 60 seconds after the start of the postulated accident, and gradually decreases after the AVS is activated. The results indicate that the AVS draws down the annulus to -0.25 inches water gauge in 305 seconds and to -2.5 inches water gauge in 565 seconds. The long term annulus pressure response, shown in Figure 6.2.1-10-2, demonstrates the capability of the AVS to maintain the annulus at design subatmospheric conditions.

Figure 6.2.1-10-3 shows the annulus temperature response. The continuously increasing trend is because of the conservative fluid temperature used for the in-leakage. The RCB wall temperature at three instants in time (0.1 second, 1000 seconds and 86,400 seconds) is shown in Figure 6.2.1-10-4. The distance is measured from the center line of the RCB. The results demonstrate that the RCB thermal conditions do not penetrate the RCB wall until ≈24 hours into the event. The primary reason is the thick RCB concrete wall. Therefore, the assumptions regarding RCB conditions and heat transfer from the RCB are inconsequential to this evaluation.

FSAR Impact:

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

Figure 6.2.1-10-1—Short Term Annulus Sub-Pressure Response, Limiting Case

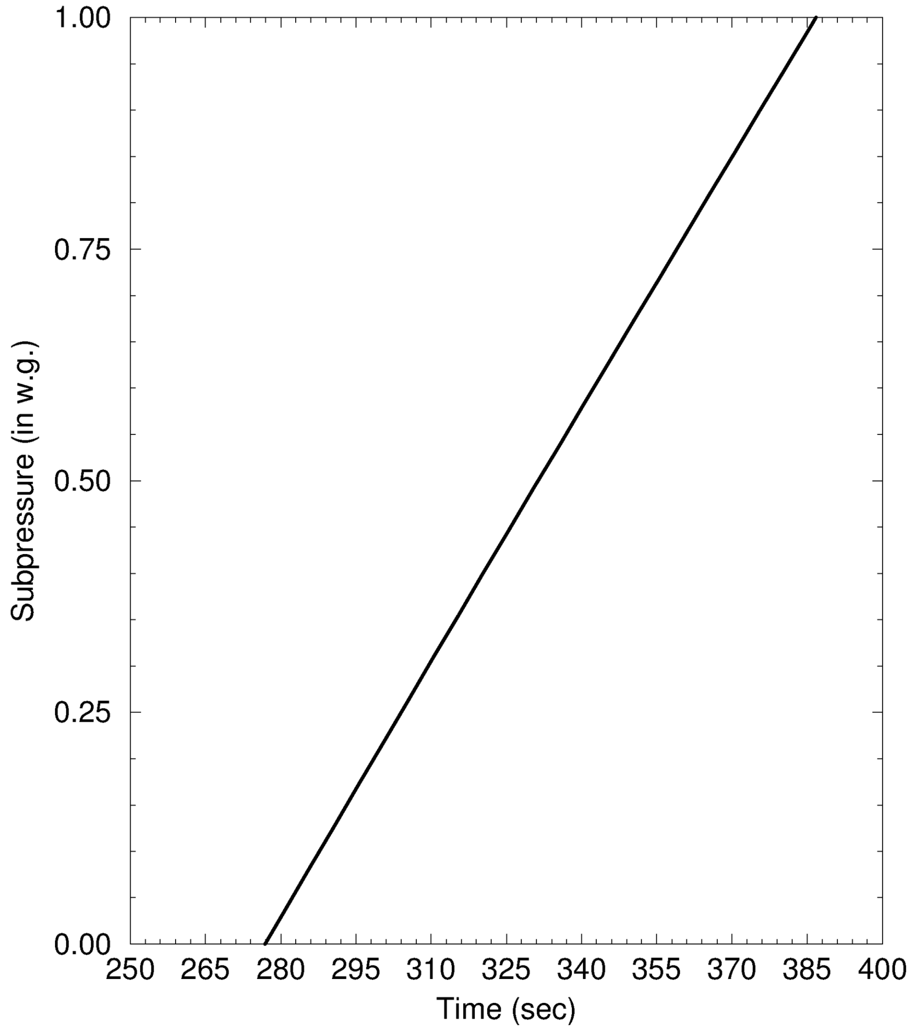


Figure 6.2.1-10-2—Long Term Annulus Pressure Response, Limiting Case

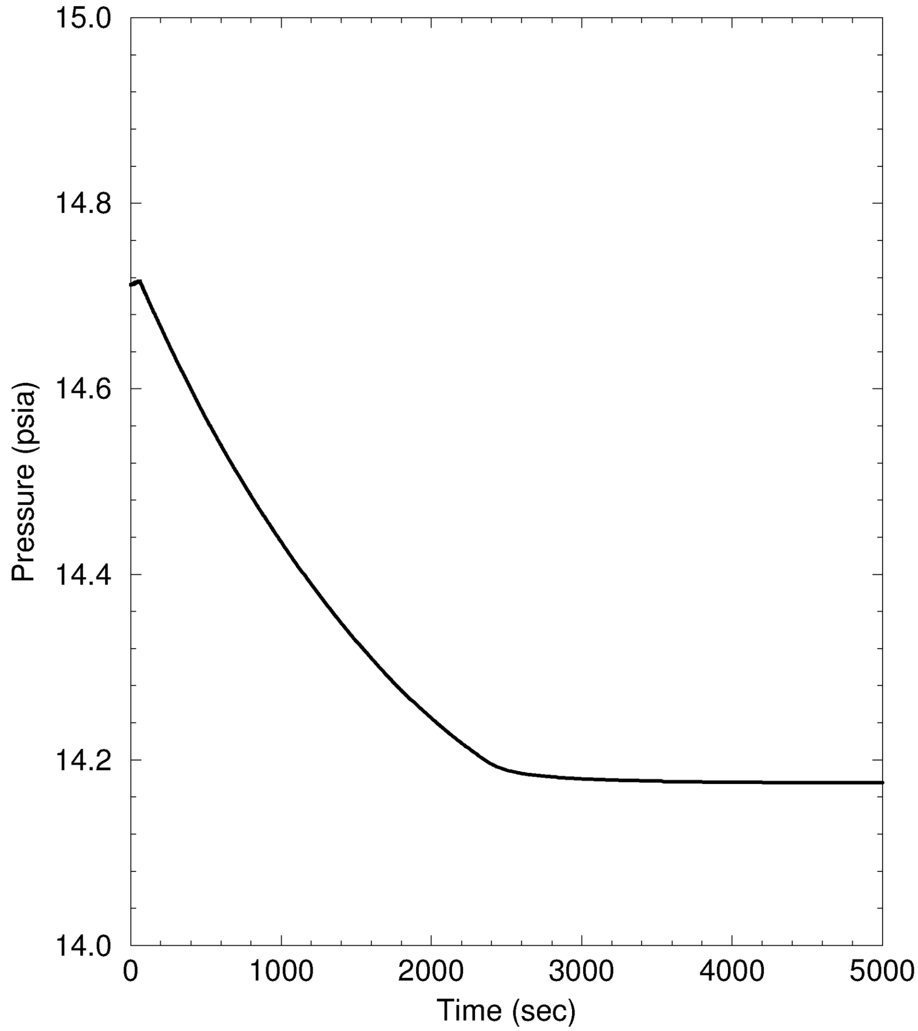


Figure 6.2.1-10-3—Annulus Temperature Response, Limiting Case

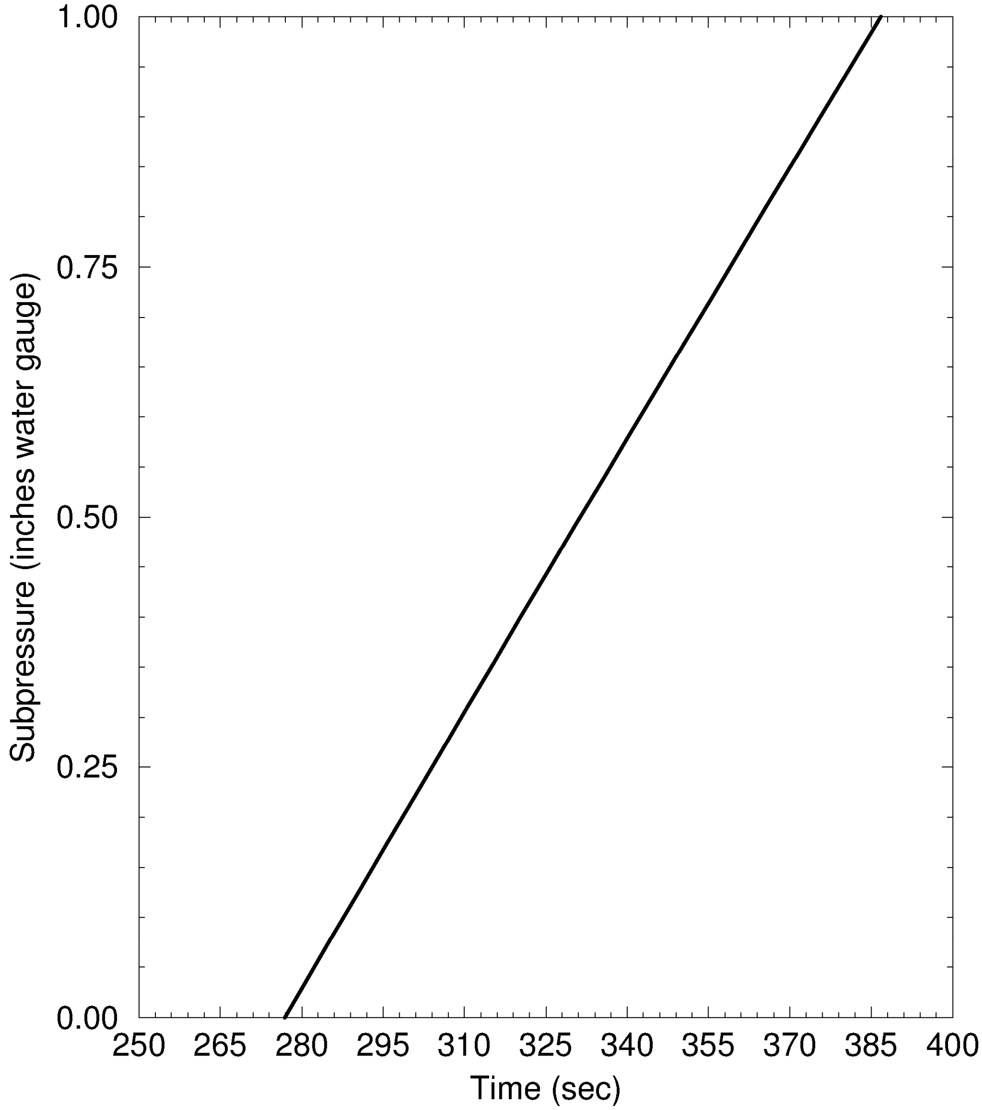


Figure 6.2.1-10-4—RCB Wall Temperature, Limiting Case