HITACHI

GE Hitachi Nuclear Energy

James C. Kinsey
Vice President, ESBWR Licensing

PO Box 780 M/C A-55
Wilmington, NC 28402-0780
USA

T 910 675 5057
F 910 362 5057

MFN 06-364 Supplement 3

Docket No. 52-010

December 3, 2007

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

Subject: Response to Portion of NRC Request for Additional Information Letter No. 33 - Containment Systems - RAI Numbers 6.2-59 S01, 6.2-60 S01, and 6.2-62 S02

Enclosure 1 contains the GE Hitachi Nuclear Energy (GEH) response to the subject NRC RAIs originally transmitted via the Reference 1 letter and supplemented by NRC requests for clarification in Reference 2 (RAI Numbers 6.2-59 S01 and 6.2-60 S01) and Reference 3 (RAI Number 6.2-62 S02).

If you have any questions or require additional information, please contact me.

Sincerely,

Kathy Sedney for

James C. Kinsey
Vice President, ESBWR Licensing

D068
NRO

References:

1. MFN 06-167, Letter from U.S. Nuclear Regulatory Commission to David Hinds, *Request for Additional Information Letter No. 33 Related to ESBWR Design Certification Application*, June 1, 2006
2. E-Mail from Shawn Williams, U.S. Nuclear Regulatory Commission, to George Wadkins, GE Hitachi Nuclear Energy, dated May 22, 2007 (ADAMS Accession Number ML071430342)
3. E-Mail from Shawn Williams, U.S. Nuclear Regulatory Commission, to George Wadkins, GE Hitachi Nuclear Energy, dated July 27, 2007 (ADAMS Accession Number ML072080190)

Enclosure:

1. MFN 06-364 Supplement 3 - Response to Portion of NRC Request for Additional Information Letter No. 33 - Related to ESBWR Design Certification Application - Containment Systems – RAI Numbers 6.2-59 S01, 6.2-60 S01, and 6.2-62 S02

cc:	AE Cubbage	USNRC (with enclosures)
	GB Stramback	GEH/San Jose (with enclosures)
	RE Brown	GEH/Wilmington (with enclosures)
	eDRF	RAI 6.2-59 S01: 0000-0075-1304
		RAI 6.2-60 S01: 0000-0076-0611
		RAI 6.2-62 S02: 0000-0076-3320

Enclosure 1

MFN 06-364 Supplement 3

Response to Portion of NRC Request for

Additional Information Letter No. 33

Related to ESBWR Design Certification Application

Containment Systems

RAI Numbers 6.2-59 S01, 6.2-60 S01, and 6.2-62 S02

NRC RAI 6.2-59 S01:

In response to RAI 6.2-59, GENE provides input error corrections and model enhancement for the approved TRACG model. Please include this information in a topical report, for e.g., a supplement to NEDC-33083P-A, "TRACG Application for ESBWR," March 2005.

GEH Response:

The response to RAI 6.2-59 (MFN 06-364, October 3, 2006) provides input error corrections and model enhancement for the approved TRACG model (Reference 1).

The model enhancement discussed in the response to RAI 6.2-59 is the wetwell gas stratification model. This wetwell gas stratification model has been documented and discussed in Section 3.3.1.1.2 in Reference 1. The improved nodalization in the DCD analyses (i.e., one additional axial level added at an elevation near the top of wetwell) has been documented as Item #20, in Table 6.2-6a, DCD Tier 2, Revision 4.

The input error corrections are the time step size sensitivity, vacuum breaker flow area, Standby Liquid Control System (SLCS) flow input table, and axial power input for the part-length rod. Sensitivity studies were performed and the impact of these input errors on the key output parameters was small (Response to RAI 6.2-59, MFN 06-364, October 3, 2006). The large negative loss coefficient at the top horizontal vent exit to reduce the high vent flow oscillations was later removed from the input decks for both the Emergency Core Cooling System (ECCS)/loss-of-coolant accident (LOCA) and Containment/LOCA analyses (for analyses performed after DCD Tier 2, Revision 3).

Reference:

(1) GE Nuclear Energy, "TRACG Application for ESBWR," NEDC-33083P-A, Class III, (Proprietary), March 2005, and NEDO-33083-A, Class I (non-proprietary), October 2005.

DCD Impact:

No DCD changes will be made in response to this RAI.

NRC RAI 6.2-60 S01:

In response to RAI 6.2-60, GENE updated DCD, Tier 2, Revision 3, Section 6.2.1.3 to state that "[c]ontainment design basis calculations are performed for a spectrum of possible pipe break sizes and locations to assure that the worst case has been identified." In response to part (B) of RAI 6.2-60, GENE stated the TRACG results regarding break sizes will be incorporated into the DCD.

A. *Explain whether you considered different locations of breaks in addition to the difference elevations of breaks as discussed in response to part (c) of RAI 6.2-60.*

B. *Incorporate the response to part (c) of RAI 6.2-60 into the DCD.*

GEH Response:

A. Containment design basis calculations are performed for a spectrum of four double-ended guillotine pipe break sizes and four different locations to assure that the worst case has been identified. The four locations include the Gravity-Driven Cooling System (GDCS) line, reactor vessel bottom drain line, feedwater line, and main steam line.

B. The response to part (B) and part (C) of RAI 6.2-60 will be incorporated in the next revision to DCD Tier 2. DCD Tier 2, Subsection 6.2.1.3 will be revised to include a reference to the detailed information to be incorporated in the next revision to DCD Tier 2, Chapter 6, as a new Appendix 6F.

DCD Impact:

DCD Tier 2, Subsection 6.2.1.3, second paragraph, will be revised, and a new DCD Tier 2, Appendix 6F, will be provided, as shown in the attached markup.

6.2.1.3 Mass and Energy Release Analyses for Postulated Loss-of-Coolant Accidents

[DCD Tier 2, Subsection 6.2.1.3, Second Paragraph]

In meeting the requirements of GDC 50 the following criteria, which pertain to the mass and energy analyses, are used.

- Sources of Energy
  - The sources of stored and generated energy that are considered in analyses of LOCAs include reactor power, decay heat, stored energy in the core and stored energy in the reactor coolant system metal, including the reactor vessel and reactor vessel internals;
  - Calculations of the energy available for release from the above sources are done in general accordance with the requirements of 10 CFR 50, Appendix K, paragraph I.A. However, additional conservatism is included to maximize the energy release to the containment during the blowdown and reflood phases of a LOCA; and
  - The requirements of paragraph I.B in Appendix K, concerning the prediction of fuel cladding swelling and rupture are not considered, to maximize the energy available for release from the core to the containment.
- Break Size and Location
  - The choice of break locations and types is discussed in Subsection 6.2.1.1.3;
  - Of several breaks postulated on the basis stated above, the break selected as the reference case yields the highest containment pressure consistent with the criteria for establishing the break location and area; and
  - Containment design basis calculations are performed for a spectrum of ~~possible~~ four double-ended guillotine pipe break sizes and locations to assure that the worst case has been identified. These calculations are described in Appendix 6F.
- Calculations

Following the procedure, documented in Reference 6.2-1, calculations of the mass and energy release rates for a LOCA are performed in a manner that conservatively establishes the containment internal design pressure (that is, maximizes the post-accident containment pressure).

6F. BREAK SPECTRUMS OF BREAK SIZES AND BREAK ELEVATIONS

6F1. Break Spectrum of Break Sizes

Parametric cases were performed prior to DCD Revision 2 with different break areas (40%, 60%, 80% and 100% of the double-ended guillotine (DEG) break area) for the feedwater line break and the main steam line break.

Main Steam Line Break - Parametric Study on the Break Areas

The base case (MSL-8F_1DPV-72) considers a single failure of one depressurization valve (DPV) and nominal conditions (Table 6.2-6), and assumes 100% DEG break.

Parametric cases were performed with the break area varied (for both ends of the break pipe) from 100% break area to 40%, 60%, and 80%, respectively. The peak drywell (DW) pressures for these cases are summarized in Table 6F1-1. The transient DW pressures of these cases are compared and shown in Figure 6F1-1.

The peak DW pressures for these cases occur near the end of the calculation (72 hours), and the case with larger break area calculates slightly higher peak DW pressure. The base case (100% break area) calculates a peak DW pressure of 323 kPa. It should be noted that the corresponding margin to the design pressure of 413.7 kPa (60 psia) is 29%.

The long-term non-condensable (NC) gas distribution depends on the NC gas circulation in the DW annulus, which affects the initial removal and subsequent removals of the NC gases returning to the DW due to the vacuum breaker openings. The NC gas circulation pattern depends on the DW annulus geometry, and the strength and location of the steam source. To cap the effect and the uncertainty of the NC gas distribution on the DW pressure, sensitivity cases were performed by changing the location of the steam source. Results of these sensitivities are discussed in Section 6F2.

Feedwater Line Break - Parametric Study on the Break Areas

The base case (FWL-8D_1SRV-72) considers a single failure of one safety relief valve (SRV) and nominal conditions (Table 6.2-6), and assumes 100% DEG break.

Parametric cases were performed with the break area varied (for both ends of the break pipe) from 100% break area to 40%, 60%, and 80%, respectively. The peak DW pressures for these cases are summarized in Table 6F1-2. The transient DW pressures of these cases are compared and shown in Figures 6F1-2a and 6F1-2b.

The peak DW pressure for these cases occurs at around 78 seconds, shortly after the DPV opening. The change in peak DW pressure is less than 0.3 kPa (< 0.1% of the peak value) when the break size changes from 100% to 40%.

6F2. Break Spectrum of Break Elevations

Sensitivity cases were performed with different break discharge location to the DW annulus for the main steam line break. The break elevation varies between the reactor pressure vessel (RPV) bottom elevation and the main steam line (MSL) elevation.

Main Steam Line Break - Parametric Study on the Break Discharge Elevation

The base case (MSL-8F_1DPV-72) considers a single failure of one DPV and nominal conditions (Table 6.2-6), and assumes 100% DEG break.

In the base case (MSL-8F_1DPV-72), the broken MSL from the RPV discharges steam into the DW at Level 34 (DCD Tier 2 prior to DCD Revision 2, Figure 6.2-7, TRACG nodalization). Sensitivity cases were performed with different elevation for the break pipe discharge location. For the sensitivity cases, the break location changes from Level 34 to Level 23 (RPV bottom), to Level 25, and to Level 31. The peak DW pressures for these cases are summarized in Table 6F2-1. The transient DW pressures of these cases are compared and shown in Figure 6F2-1.

Figure 6F2-1 shows that the base case with highest break location (MSL-8F_1DPV-72) generates the limiting DW pressure. The peak DW pressure from this case also bounds those from cases with different break area discussed in Section 6F1. It should be noted that, for this limiting case, the margin to the design pressure of 413.7 kPa (60 psia) is 29%, at the end of the 72 hours transient.

Many parameters affect the gas circulation pattern, which in turn affect the NC gas distribution and the containment pressure. The limiting long-term containment pressure is determined by performing parametric cases with a different elevation for the break pipe discharge location.

Table 6F1-1.

Summary of Peak DW Pressures for the MSL Break Area Study

Case ID	DEG Break Size	Max. DW Pressure (kPa)	Time at Max. (hr)
MSL-8F_1DPV-72 (Base Case)	100%	323.0	71.7
MSL-8F_1DPVP8-72	80%	316.8	71.9
MSL-8F_1DPVP6-72	60%	312.7	71.6
MSL-8F_1DPVP4-72	40%	312.1	71.4

Table 6F1-2.

Summary of Peak DW Pressures for the FWL Break Area Study

Case ID	DEG Break Size	Max. DW Pressure (kPa)	Time at Max. (sec)
FWL-8D_1SRV-72 (Base Case)	100%	306.2	77.8
FWL-8D_1SRVP8-72	80%	306.0	77.9
FWL-8D_1SRVP6-72	60%	305.9	78.2
FWL-8D_1SRVP4-72	40%	305.9	78.3

Table 6F2-1.

Summary of Peak DW Pressures for the MSL Break Elevation Study

Case ID	Break Location* in DW	Max. DW Pressure (kPa)	Time at Max. (hr)
MSL-8F_1DPV-72 (Base Case)	Level 34	323.0	71.7
MSL-8F_1DPVL31-72	Level 31	316.3	71.5
MSL-8F_1DPVL25-72	Level 25	316.4	71.2
MSL-8F_1DPVL23-72	Level 23	314.6	71.3

* DCD Revision 3, Figure 6.2-7, TRACG nodalization

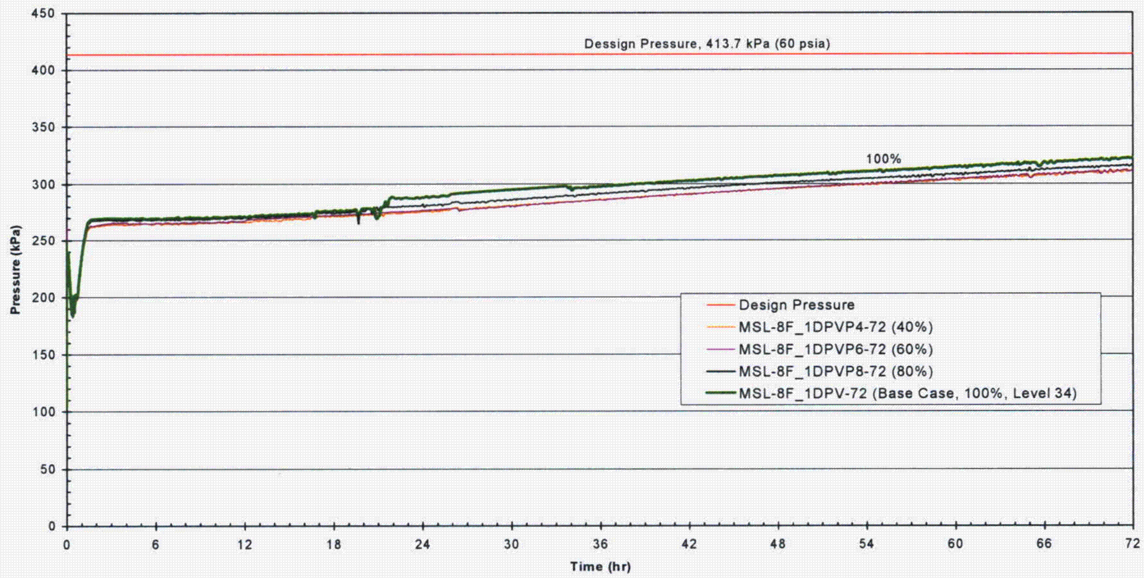

Figure 6F1-1 MSLB - Effect of Break Areas on Transient DW Pressures

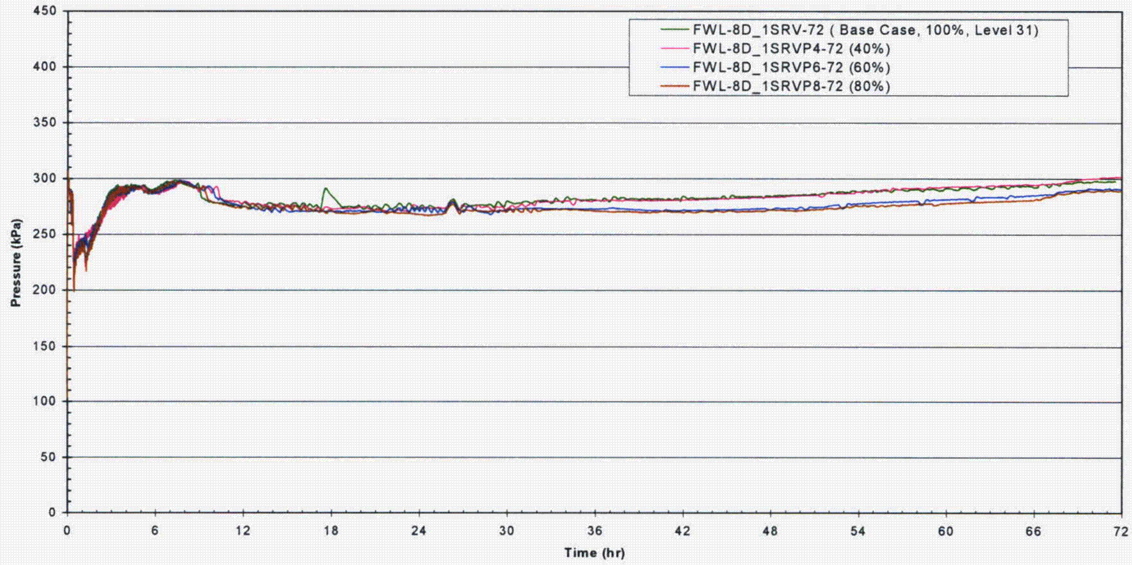

Figure 6F1-2a Feedwater Line Break - Parametric Study on the Break Areas (0-72 hrs)

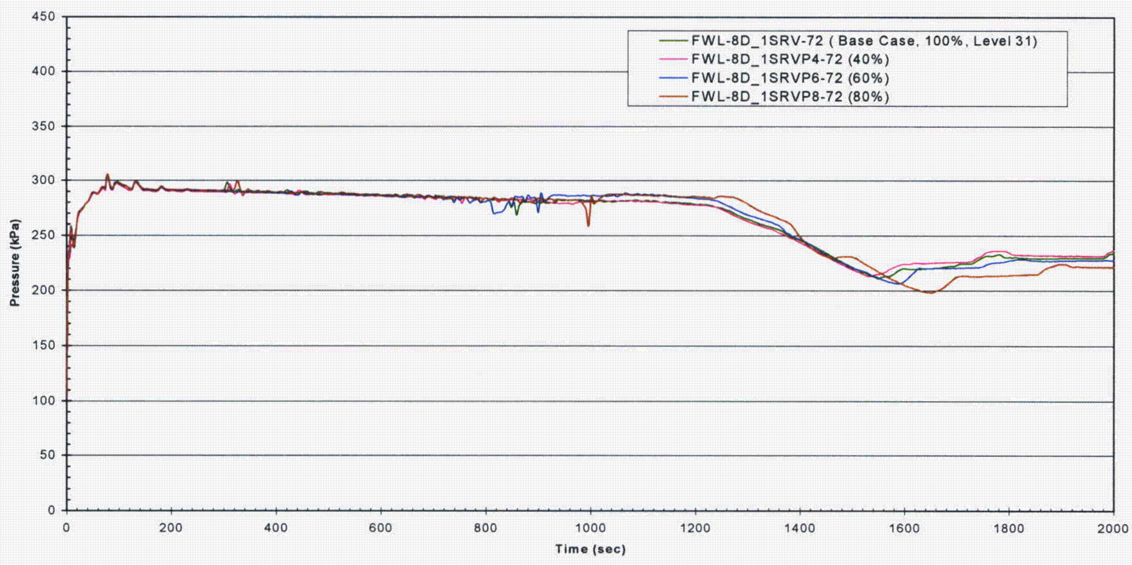

Figure 6F1-2b Feedwater Line Break - Parametric Study on the Break Areas (0-2000 sec)

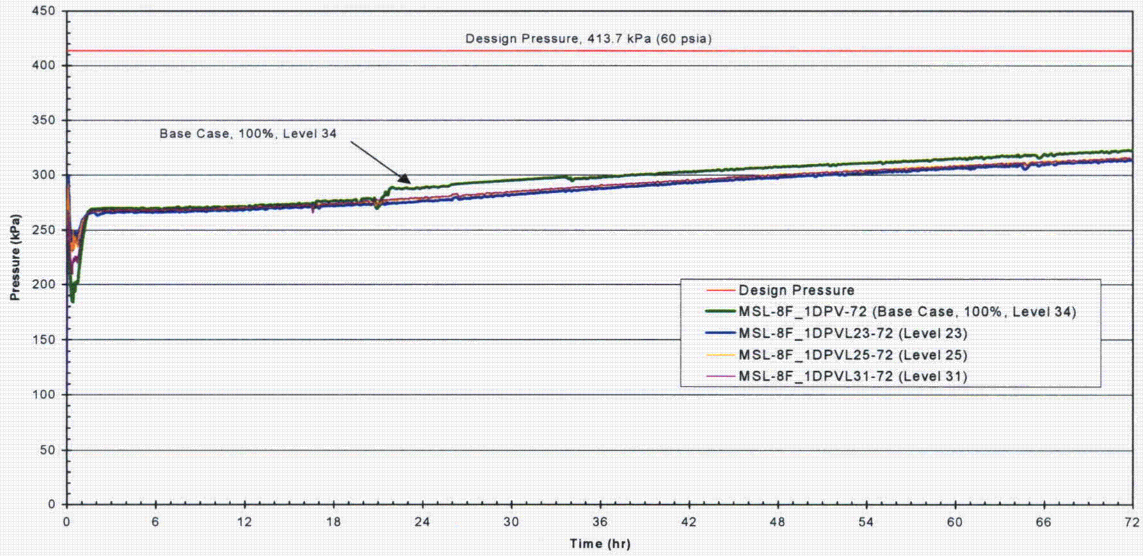

Figure 6F2-1 Main Steam Line Break - Parametric Study on the Break Elevation

NRC RAI 6.2-62 S02:

The Staff accepts GE's response except for the last statement, "No DCD changes will be made in response to this RAI." Please revise the DCD to include the information provided in this response including the graphs and tables.

GEH Response:

DCD Tier 2, Subsection 6.2.1.1.3.1 will be revised to include a summary of the requested information, with the detailed information to be incorporated in the next revision to DCD Tier 2, Chapter 6, as a new Appendix 6D.

DCD Impact:

DCD Tier 2, Subsection 6.2.1.1.3.1, first paragraph, will be revised, and a new DCD Tier 2, Appendix 6D, will be provided, as shown in the attached markup.

6.2.1.1.3.1 Feedwater Line Break – Nominal Analysis

[DCD Tier 2, Subsection 6.2.1.1.3.1, First Paragraph]

This analysis initializes the RPV and containment at the base conditions shown in the Nominal Value column of Table 6.2-6. Figure 6.2-6 and 6.2-7 show the TRACG nodalization of the RPV and the containment. Its fundamental structure is an axisymmetric "VSSL" component with 42 axial levels and eight radial rings. The inner 4 rings in the first 21 axial levels represent the RPV; the outer 4 rings in these levels are not utilized in the calculations. Axial levels 22 to 35 represent the DW, suppression pool, WW, and GDCS pools (Figure 6.2-7). Axial levels 36 to 42 represent the IC/PCC pool, expansion pools, and the Dryer/Separator Storage pool. Figure 6.2-8 shows the nodalization for the steam line system, including the SRVs and DPVs. Figure 6.2-8a shows the nodalization for the ESBWR isolation condenser system. Figure 6.2-8b shows the nodalization of the ESBWR feedwater line system. Appendix 6D provides a detailed description of the passive heat sinks within containment as per Regulatory Guide 1.70. Appendix 6A, Figure 6A-1 shows the TRACG nodalization of the drywell/wetwell walls as passive heat sinks.

6D. CONTAINMENT PASSIVE HEAT SINK DETAILS

Table 6-11, Item A, in Regulatory Guide (RG) 1.70, requests a listing of all structures, components, and equipment used as passive heat sinks according to RG 1.70 Table 6-4A. The ESBWR containment was conservatively modeled in TRACG by excluding all piping, equipment and miscellaneous structures. The TRACG containment volume was reduced by 1% to account for piping, equipment and miscellaneous structures. The passive heat sinks that were modeled for the ESBWR are shown in Table 6D-1.

Table 6-11B in RG 1.70 requests detailed passive heat sink data. The information to be provided and the format are given in RG 1.70 Tables 6-4B, 6-4C, and 6-4D. The containment drywell and wetwell inner and outer walls in TRACG are modeled as one-dimensional heat slabs. The modeling of these heat slabs is listed in Table 6D-2.

The thermophysical properties of the drywell and wetwell walls are listed in Table 6D-3. RG 1.70 Table 6-11C requests a graphical display of the condensing heat transfer coefficients as functions of time for the design basis accident. TRACG output data containing heat transfer coefficients are available to compare with NRC code calculations of heat transfer coefficients.

The passive heat sinks that are modeled in the ESBWR are the drywell/wetwell inner and outer walls. The walls are modeled as double-sided, one-dimensional heat slabs, which conduct heat in the radial direction from a TRACG cell to the next cell radially outward, or to an ambient temperature. The heat slabs are located between Levels 25 and 31 of the containment. The drywell/wetwell inner wall connects Ring 6 to Ring 7. The drywell/wetwell outer wall connects Ring 8 to the reactor building ambient conditions. The ambient temperature of the reactor building is 308°K. Appendix 6A, Figure 6A-1, illustrates the nodalization of the drywell/wetwell inner and outer walls. Table 6D-4 provides a further breakdown of the heat transfer area by level.

Table 6D-1.

Listing of Passive Heat Sinks

Item 7 from RG 1.70, Table 6-4A	Internal Separation Walls and Floors (Drywell/Wetwell)

Note: No other types of heat sinks listed in RG 1.70, Table 6-4A are modeled.

Table 6D-2.

Modeling of Passive Heat Sinks

Passive Heat Sink	Material	Material Thickness, m (ft)	Thickness Group	Surface Area, m^2 (ft^2)
Drywell/Wetwell Inner Wall	Concrete	0.6 (1.968)	b	900 (9687.5)
Drywell/Wetwell Outer Wall	Concrete	2.0 (6.561)	b	1386 (14918.8)

Table 6D-3.

Thermophysical Properties of Passive Heat Sink Materials

Material	Density kg/m^3	Specific Heat J/kg-K	Thermal Conductivity W/m-K
Concrete	2322.6767	879.228	1.3845872

Table 6D-4.

Total Heat Transfer Area by Containment Level

Level	Drywell/Wetwell Inner Wall m²	Drywell/Wetwell Outer Wall m²
25	73.81	201.31
26	56.81	154.94
27	62.2	169.65
28	29.03	79.17
29	214.94	226.19
30	287.58	429.77
31	175.62	124.41
Total Area	899.99	1385.44