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MFN 08-341

Docket No. 52-010

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U.S. Nuclear Regulatory Commission
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Subject: Response to Portion of NRC Request for Additional Information Letter No. 110 Related to ESBWR Design Certification Application - Containment Systems - RAI Number 6.2-180

Enclosure 1 contains the GE Hitachi Nuclear Energy (GEH) response to the subject NRC RAI transmitted via the Reference 1 letter. DCD Markups related to this response are provided in Enclosure 2.

Verified DCD changes associated with this RAI response are identified in the enclosed DCD markups by enclosing the text within a black box. The marked-up pages may contain unverified changes in addition to the verified changes resulting from this RAI response. Other changes shown in the markup(s) may not be fully developed and approved for inclusion in DCD Revision 5.

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

Sincerely,

James C. Kinsey

James C. Kinsey
Vice President, ESBWR Licensing

D068
NRO

Reference:

1. MFN 07-510, Letter from U.S. Nuclear Regulatory Commission to Robert E. Brown, *Request for Additional Information Letter No. 110 Related to ESBWR Design Certification Application*, September 19, 2007

Enclosures:

1. MFN 08-341 - Response to Portion of NRC Request for Additional Information Letter No. 110 Related to ESBWR Design Certification Application - Containment Systems - RAI Number 6.2-180
2. MFN 08-341 - Response to Portion of NRC Request for Additional Information Letter No. 110 Related to ESBWR Design Certification Application - Containment Systems - RAI Number 6.2-180 - DCD Markups

cc:	AE Cubbage	USNRC (with enclosures)
	DH Hinds	GEH/Wilmington (with enclosures)
	GB Stramback	GEH/San Jose (with enclosures)
	RE Brown	GEH/Wilmington (with enclosures)
	eDRF	0000-0078-3785

Enclosure 1

MFN 08-341

Response to Portion of NRC Request for Additional Information Letter No. 110 Related to ESBWR Design Certification Application

Containment Systems

RAI Number 6.2-180

NRC RAI 6.2-180:

DCD, Tier 2, Revision 3, Section 6.2.1.1.2 states that "[f]or a postulated DBA, the calculated maximum DW temperature and absolute pressure remain below their design values, shown in Table 6.2-1." However, DCD Table 6.2-5 shows that the "short-term" drywell temperature exceeding the design value for the following accidents: (1) main steam line break and bottom head drain line based on standard TRACG evaluation model and (2) feedwater line break for based on both standard TRACG evaluation model and bounding values model. This table also shows that the short-term peak drywell temperature predicted for the main steam line break based on the standard TRACG evaluation model (174.7°C (346.8°F)) is more limiting than that based on the bounding values model (170.8°C (339.6°F)).

A. *Please correct or explain this apparent discrepancy in the DCD. Also, please explain why it is acceptable for the drywell temperature to exceed its design value.*

B. *Please explain why the short-term peak drywell temperature for the main steam line break predicted by the standard TRACG evaluation model is more limiting than that based on the bounding values model.*

GEH Response:

A. The temperatures shown on DCD Tier 2, Revision 3, Table 6.2-5 represent the temperature at a single location in the TRACG evaluation. The selected location is not representative of the bulk drywell temperature used in the structural analysis. As a result, local temperatures near the break location can exceed the bulk design temperature used to determine the global structural response. DCD Tier 2, Subsection 6.2.1.1.2 will be revised to state that the bulk temperature remains below the design value. DCD Tier 2, Table 6.2-5 and Table 3H-8 values will be replaced with the bulk temperature rather than those of a single location, because this represents the temperature that is used in the structural analysis.

   For some cases, at about 1000 seconds into the transient, the drywell bulk temperature is much lower than the peak design temperature but slightly higher than the temperature transient profile considered in the structural analysis. The higher temperatures last for no more than 800 seconds. No impact on structural design is expected, and the effect on the structural analysis documented in DCD Tier 2, Appendix 3G will be taken into account in the detailed design stage.

B. The bounding main steam line break analysis is that which produces the greatest pressure within the containment, not necessarily the highest temperature. In the ESBWR, the peak containment pressure is obtained after several hours, but the peak containment temperature occurs within the first 5 seconds due to heating from the steam discharge. The initial conditions leading to the higher long-term containment pressure increase result in lower initial flow out the break, and thus lower initial increase in containment pressure and temperature.

DCD Impact:

DCD Tier 2, Appendix 3G; Table 3H-8; Subsection 6.2.1.1.2; Figures 6.2-9b1, 6.2-9b2, 6.2-9b3, 6.2-10b1, 6.2-10b2, 6.2-10b3, 6.2-11b1, 6.2-11b2, 6.2-11b3, 6.2-12b1, 6.2-12b2, 6.2-12b3, 6.2-13b1, 6.2-13b2, 6.2-13b3, 6.2-14b1, 6.2-14b2, and 6.2-14b3; and Appendix 6E will be revised as shown in the attached markup.

Enclosure 2

MFN 08-341

Response to Portion of NRC Request for Additional Information Letter No. 110 Related to ESBWR Design Certification Application

Containment Systems

RAI Number 6.2-180

DCD Markups

Verified DCD changes associated with this RAI response are identified in the enclosed DCD markups by enclosing the text within a black box. The marked-up pages may contain unverified changes in addition to the verified changes resulting from this RAI response. Other changes shown in the markup(s) may not be fully developed and approved for inclusion in DCD Revision 5.

3G. DESIGN DETAILS AND EVALUATION RESULTS OF SEISMIC CATEGORY I STRUCTURES

This appendix presents the structural design and analysis for the Reactor Building, Control Building, Fuel Building and Fire Water Service Complex of the ESBWR Standard Plant. It addresses all applicable items included in Appendix C to United States Nuclear Regulatory Commision (USNRC) Standard Review Plan, NUREG-0800, Subsection 3.8.4. Drawings depicted in the Design Control Document (DCD) are not used for construction. Construction drawings meet the technical licensing commitments made in the DCD but are issued under different contractual/industrial rules than the DCD drawings~~.~~ and reflect detailed design configuration. The final design details, and hence final component stresses may be different from those reported here but they will meet the structural acceptance criteria presented in Section 3.8.

3G.1 REACTOR BUILDING

The Reactor Building (RB) encloses the concrete containment and its internal systems, structures, and components. In addition, the RB contains the Isolation Condenser/Passive Containment Cooling (IC/PCC) pools and the services pools for storage of Dryer/Separator on the top of the concrete containment.

3G.1.1 Objective and Scope

The objective of this subsection is to document the structural design details, inputs and analytical results from the analysis of the ESBWR main building structures encased in the Reactor Building. The scope includes the design and analysis of the structure for normal, severe environmental, extreme environmental, and abnormal loads.

3G.1.2 Conclusions

The following are the major summary conclusions on the design and analysis of the Reactor Building, the concrete containment and the containment internal structures.

- Based on the results of finite element analyses performed in accordance with the design conditions identified in Subsections 3G.1.3 and 3G.1.5, stresses and/or strains in concrete, reinforcement, liner and containment internal structures are less than the allowable stresses and/or strains per the applicable regulations, codes or standards listed in Section 3.8.
- The factors of safety against floatation, sliding, and overturning of the structure under various loading combinations are higher than the required minimum.
- The thickness of the roof slabs and exterior walls are more than the minimum required to preclude penetration, perforation or spalling resulting from impact of design basis tornado missiles.

Table 3H-8

Thermodynamic Environment Conditions Inside Containment Vessel for Accident Conditions

Plant Zone			
(a-1 & a-2)	Upper and lower drywell [1] (Figure 6.2-1)	Time [2] Bulk Temp. °C (°F) Press. kPag (psig) Humidity %	0 h – 72 h. 178 (352) Max. 310 (45) Max. Steam
(a-3)	Wetwell (Figure 6.2-1)	Time [2] Bulk Temp. °C (°F) Press. kPag (psig) Humidity %	0 h – 72 h 135 (275) Max. 310 (45) Max. 100

Notes:

(1) For a pipe failure inside the containment vessel, water accumulates in the lower drywell. The amount depends upon the break location.

(2) Time denotes the time after the occurrence of LOCA.

- The containment structure shall withstand coincident fluid jet forces associated with the flow from the postulated rupture of any pipe within the containment.
- The containment structure shall accommodate flooding to a sufficient depth above the active fuel to maintain core cooling and to permit safe removal of the fuel assemblies from the reactor core after the postulated DBA.
- The containment structure shall be protected from or designed to withstand hypothetical missiles from internal sources and uncontrolled motion of broken pipes, which could endanger the integrity of the containment.
- The containment structure shall direct the high energy blowdown fluids from postulated LOCA pipe ruptures in the DW to the pressure suppression pool and to the Passive Containment Cooling System (PCCS).
- The containment system shall allow for periodic tests at the calculated peak or reduced test pressure to measure the leakage from individual penetrations, isolation valves and the integrated leakage rate from the containment structure to confirm the leak-tight integrity of the containment.
- The Containment Inerting System establishes and maintains the containment atmosphere to $\leq 3\%$ by volume oxygen during normal operating conditions to ensure inert atmosphere operation.
- PCCS shall remove post-LOCA decay heat from the containment for a minimum of 72 hours, without operator action, to maintain containment pressure and temperature within design limits.

6.2.1.1.2 Design Features

The containment structure is a reinforced concrete cylindrical structure, which encloses the Reactor Pressure Vessel (RPV) and its related systems and components. Key containment components and design features are exhibited in Figures 6.2-1 through 6.2-5. The containment structure has an internal steel liner providing the leak-tight containment boundary. The containment is divided into a DW region and a WW region with interconnecting vent system. The functions of these regions are as follows:

- The DW region is a leak-tight gas space, surrounding the RPV and reactor coolant pressure boundary, which provides containment of radioactive fission products, steam, and water released by a LOCA, prior to directing them to the suppression pool via the DW/WW Vent System. A relatively small quantity of DW steam is also directed to the PCCS during the LOCA blowdown.
- The WW region consists of the suppression pool and the gas space above it. The suppression pool is a large body of water to absorb energy by condensing steam from SRV discharges and pipe break accidents. The pool is an additional source of reactor water makeup and serves as a reactor heat sink. The flow path to the WW is designed to entrain radioactive materials by routing fluids through the suppression pool during and following a LOCA. The gas space above the suppression pool is leak-tight and sized to collect and retain the DW gases following a pipe break in the DW, without exceeding the containment design pressure.

The DW/WW Vent System directs LOCA blowdown flow from the DW into the suppression pool.

The containment structure consists of the following major structural components: RPV support structure (pedestal), diaphragm floor separating DW and WW, suppression pool floor slab, containment cylindrical outer wall, cylindrical vent wall, containment top slab, and DW head. The containment cylindrical outer wall extends below the suppression pool floor slab to the common basemat. This extension is not part of containment boundary, however, it supports the upper containment cylinder. The reinforced concrete basemat foundation supports the entire containment system and extends to support the RB surrounding the containment. The refueling bellows seal extends from the lower flange of the reactor vessel to the interior of the reactor cavity. This extension is also not part of the containment boundary, however, it provides a Seismic Category I seal between the upper drywell and reactor well during a refueling outage.

The design parameters of the containment and the major components of the containment system are given in Tables 6.2-1 through 6.2-4. A detailed discussion of their structural design bases is given in Section 3.8.

Drywell

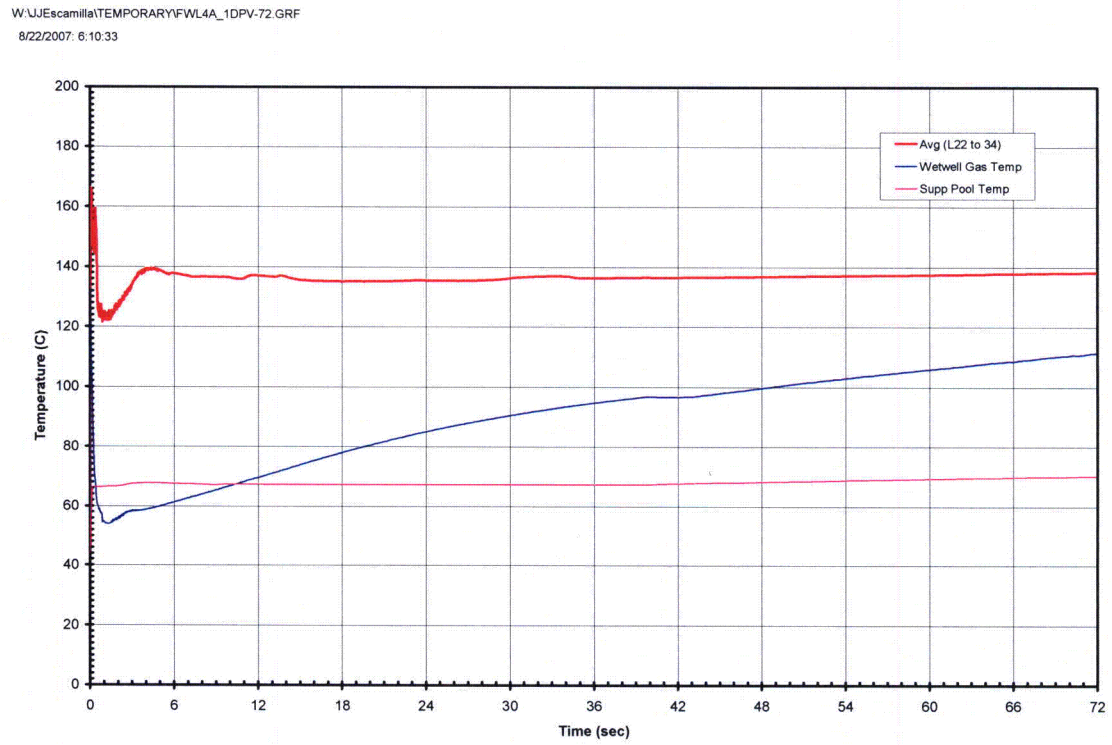
The DW (Figure 6.2-1) comprises two volumes: (1) an upper DW volume surrounding the upper portion of the RPV and housing the main steam and feedwater piping, GDCS pools and piping, PCCS piping, ICS piping, SRVs and piping, Depressurization Valves (DPVs) and piping, DW coolers and piping, and other miscellaneous systems; and (2) a lower DW volume below the RPV support structure housing the lower portion of the RPV, fine motion control rod drives, other miscellaneous systems and equipment below the RPV, and vessel bottom drain piping.

The upper DW is a cylindrical, reinforced concrete structure with a removable steel head and a diaphragm floor constructed of steel girders with concrete fill. The RPV support structure separates the lower DW from the upper DW. There is an open communication path between the two DW volumes via upper DW to lower DW connecting vents, built into the RPV support structure. Penetrations through the liner for the DW head, equipment hatches, personnel locks, piping, electrical and instrumentation lines are provided with seals and leak-tight connections.

The DW is designed to withstand the pressure and temperature transients associated with the rupture of any primary system pipe inside the DW, and also the negative differential pressures associated with containment depressurization events, when the steam in the DW is condensed by the PCCS, the GDCS, the FAPCS, and cold water cascading from the break following post-LOCA flooding of the RPV.

For a postulated DBA, the calculated maximum bulk DW temperature and absolute pressure in Table 6.2-5 remain below their design values, shown in Table 6.2-1.

Three ~~V~~vacuum breakers are provided between the DW and WW. The vacuum breaker is a process-actuated valve, similar to a check valve, see Figure 6.2-28. The purpose of the DW-to-WW vacuum breaker system is to protect the integrity of the diaphragm floor slab and vent wall between the DW and the WW, and the DW structure and liner, and to prevent back-flooding of the suppression pool water into the DW. The vacuum breaker is provided with redundant proximity sensors to detect its closed position. One out of the three vacuum breakers are required to perform vacuum relief function. The third vacuum breaker provides redundancy



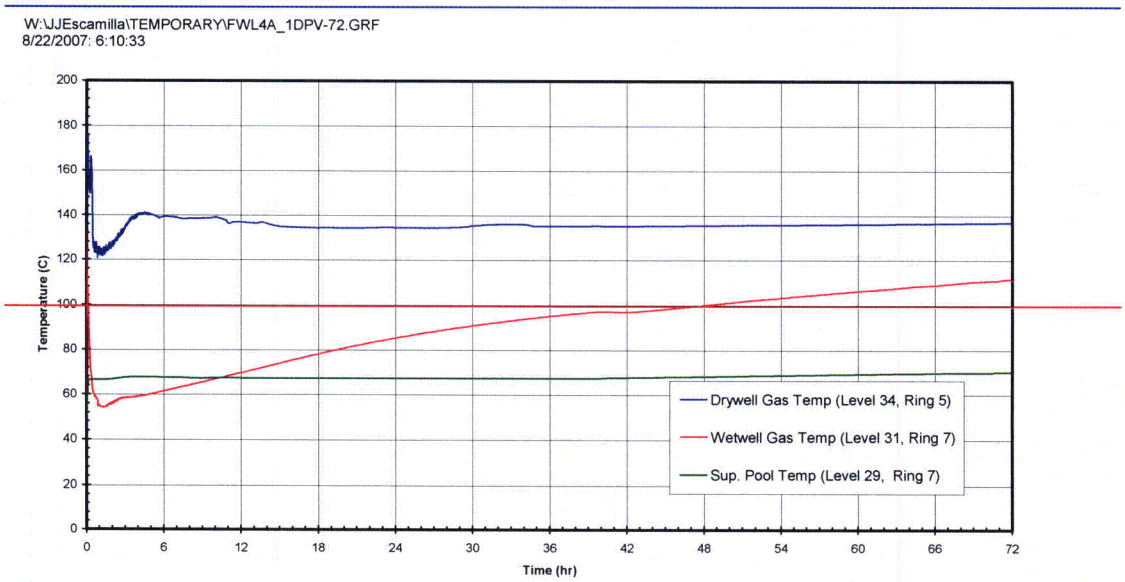
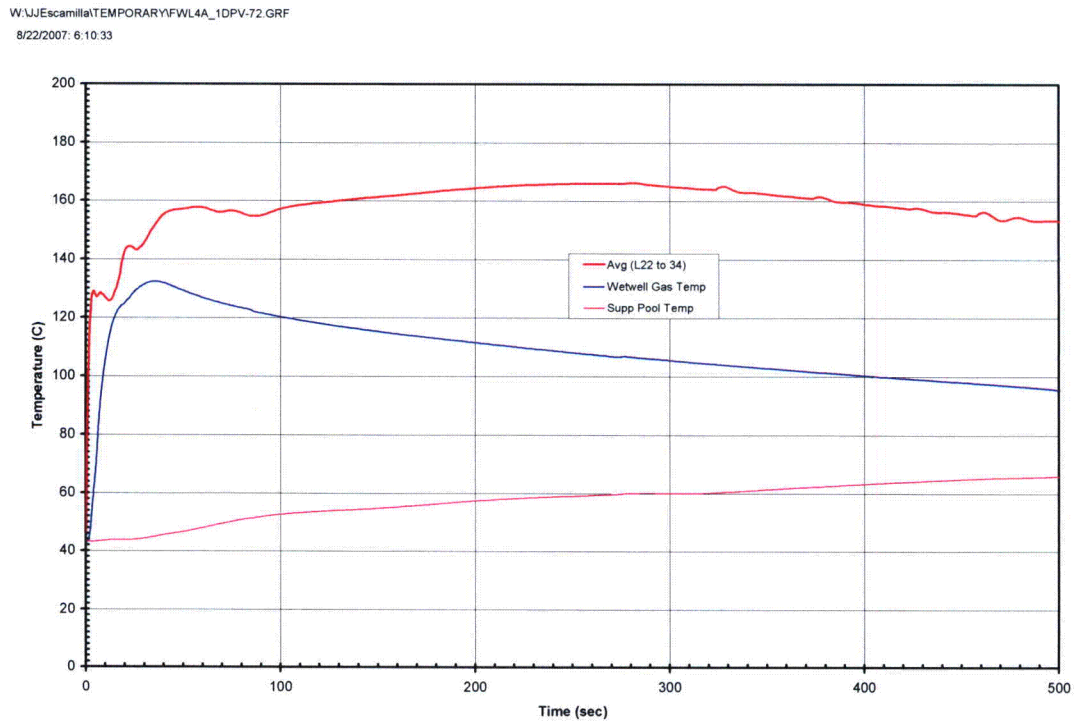


Figure 6.2-9b1. Feedwater Line Break (Nominal Case) – Containment Temperatures (72hrs)



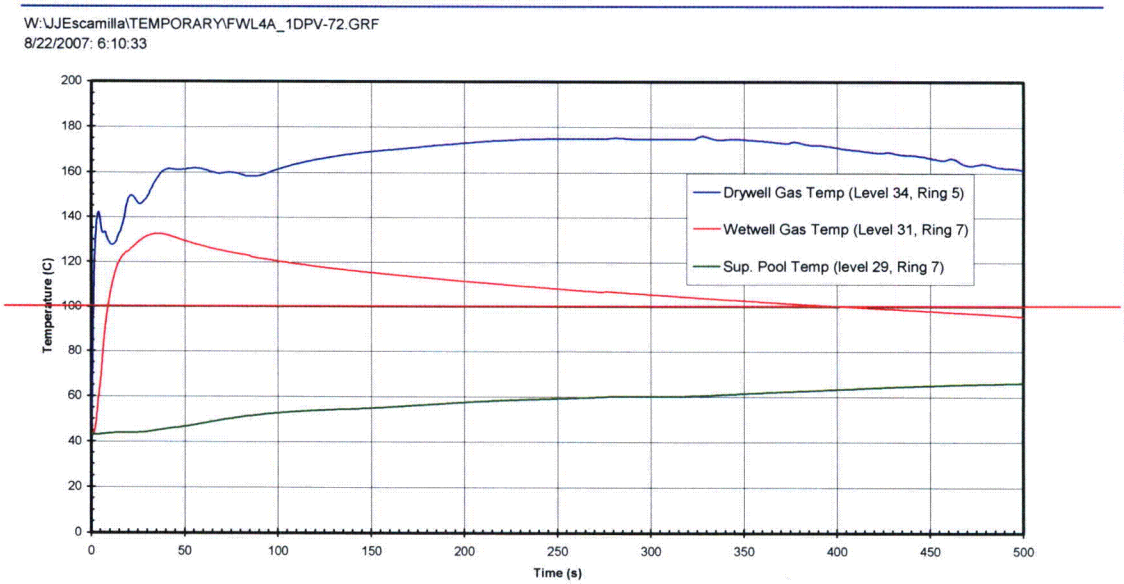
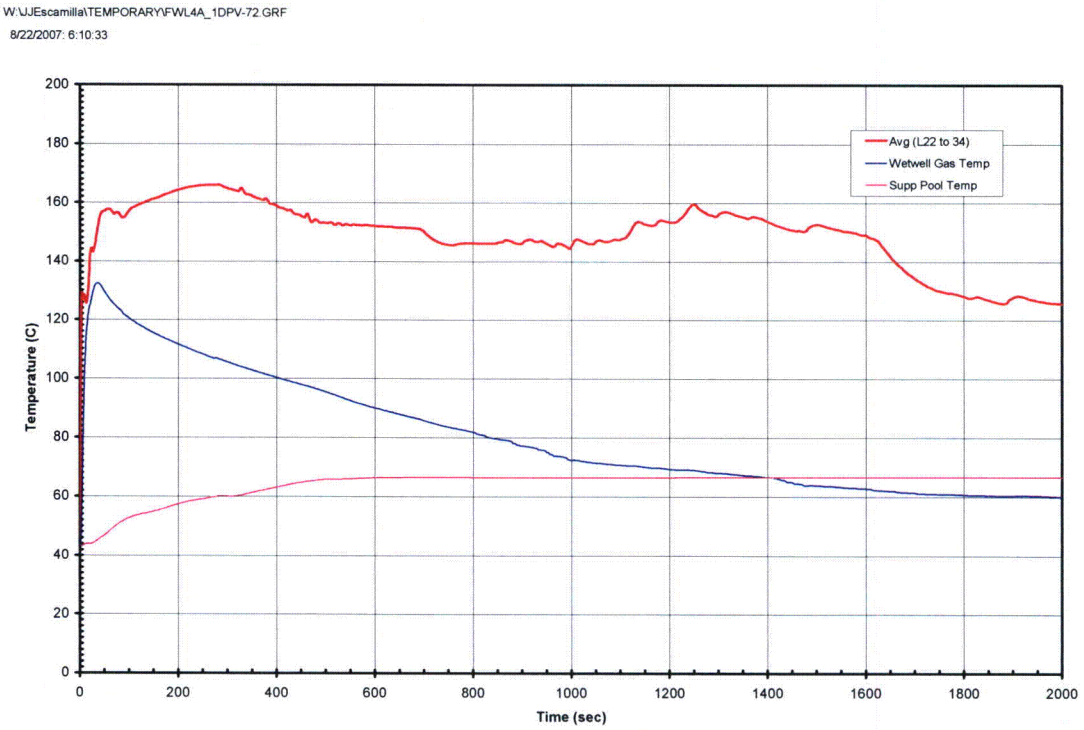


Figure 6.2-9b2. Feedwater Line Break (Nominal Case) – Containment Temperatures (500 s)



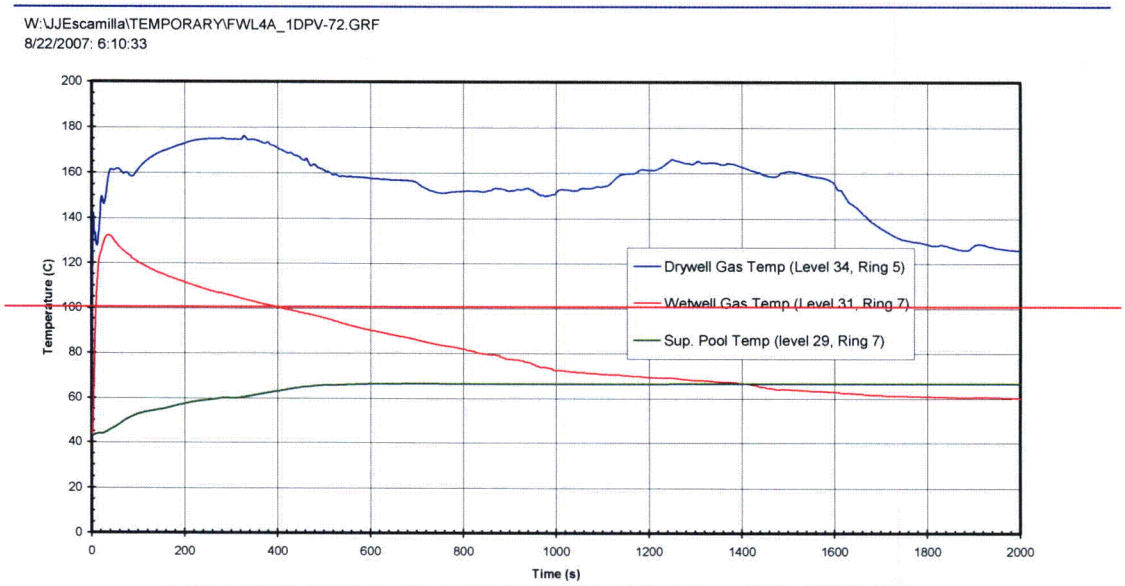
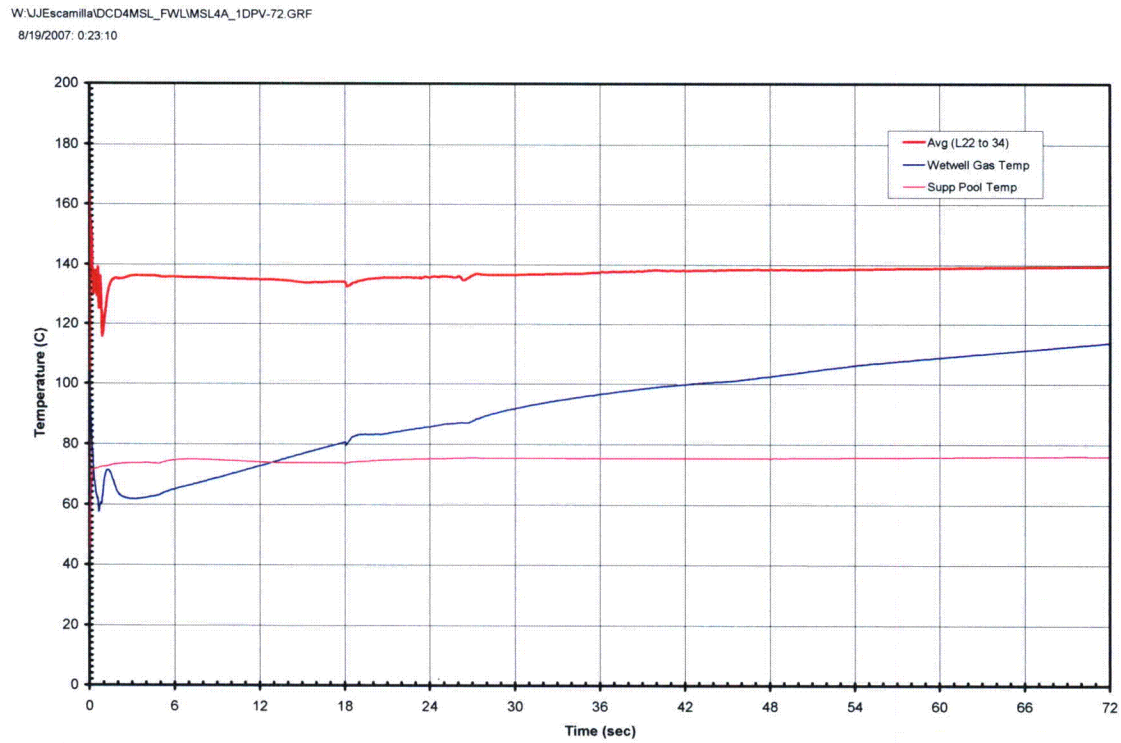


Figure 6.2-9b3. Feedwater Line Break (Nominal Case) – Containment Temperatures (2000 s)



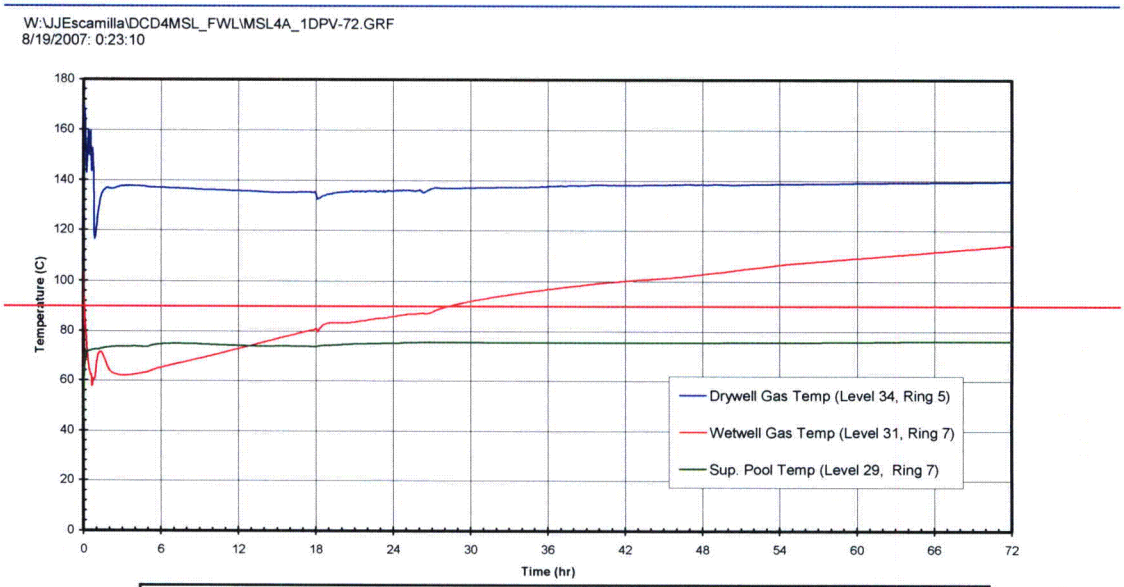
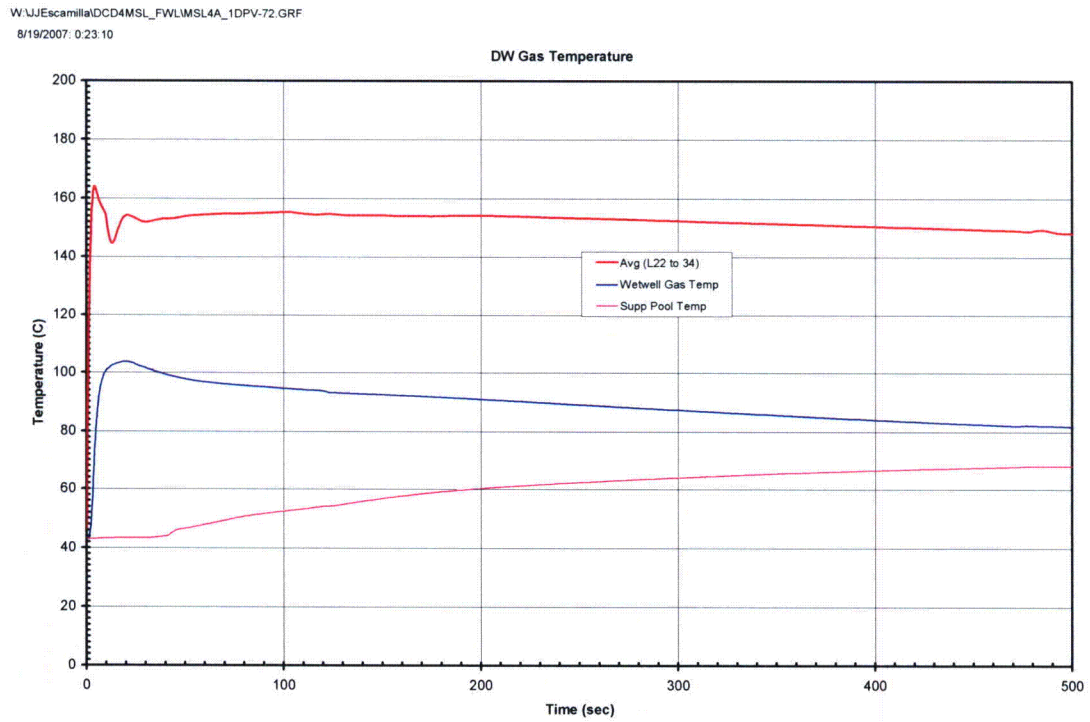


Figure 6.2-10b1. Main Steam Line Break (Nominal Case) – Containment Temperatures (72 hrs)



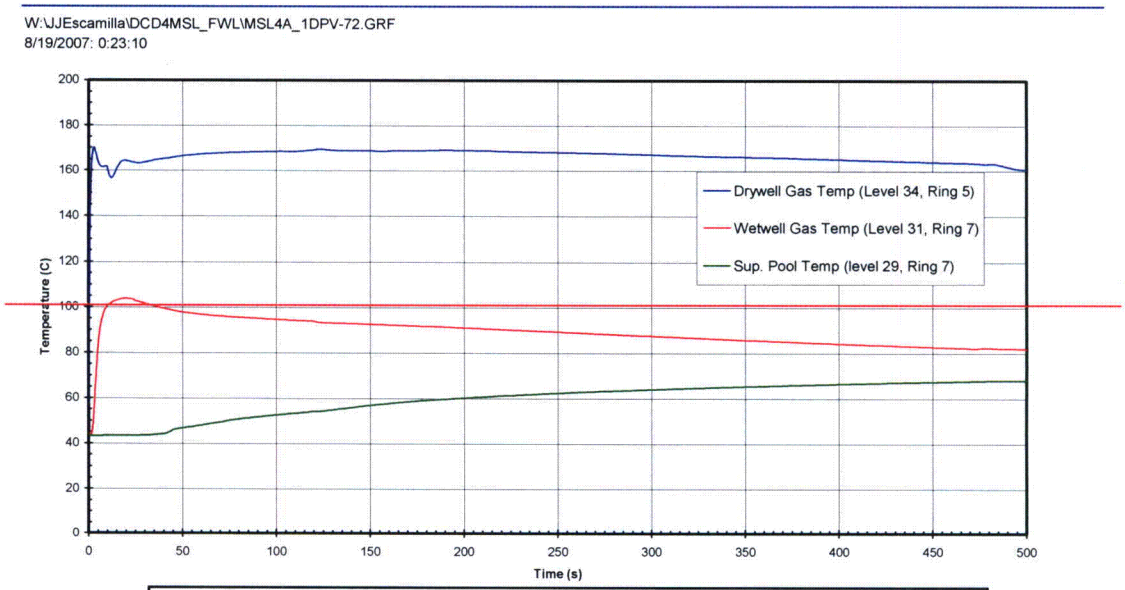
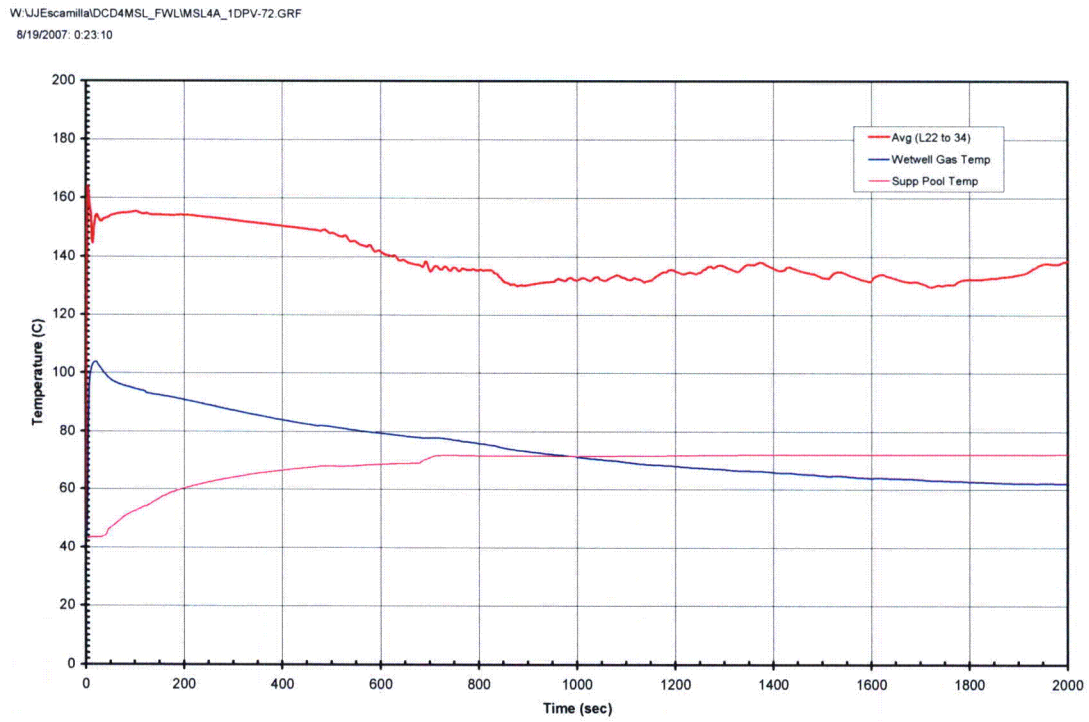


Figure 6.2-10b2. Main Steam Line Break (Nominal Case) – Containment Temperatures (500 s)



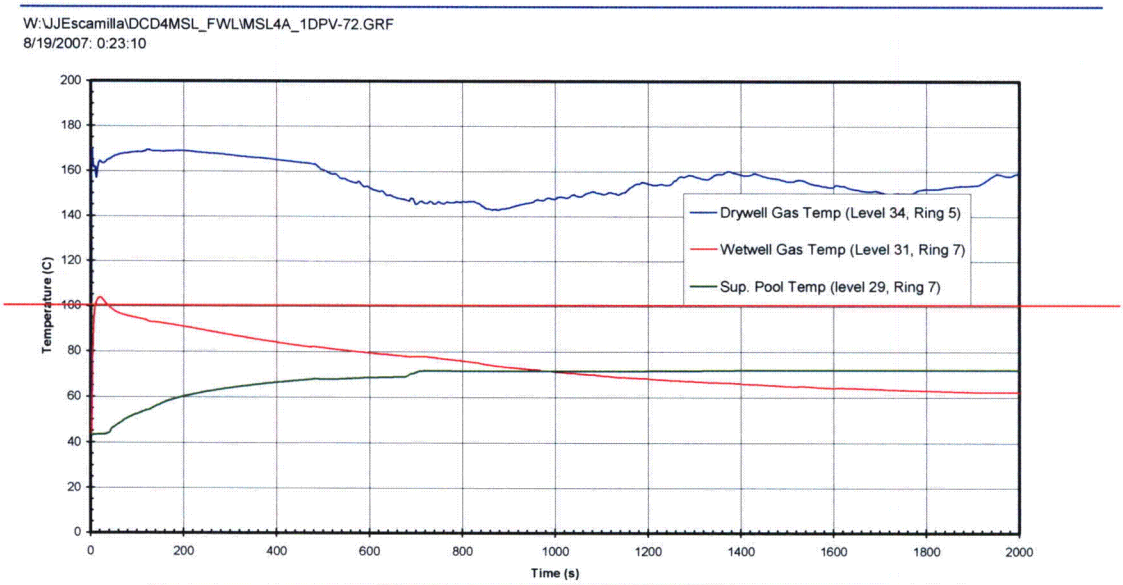
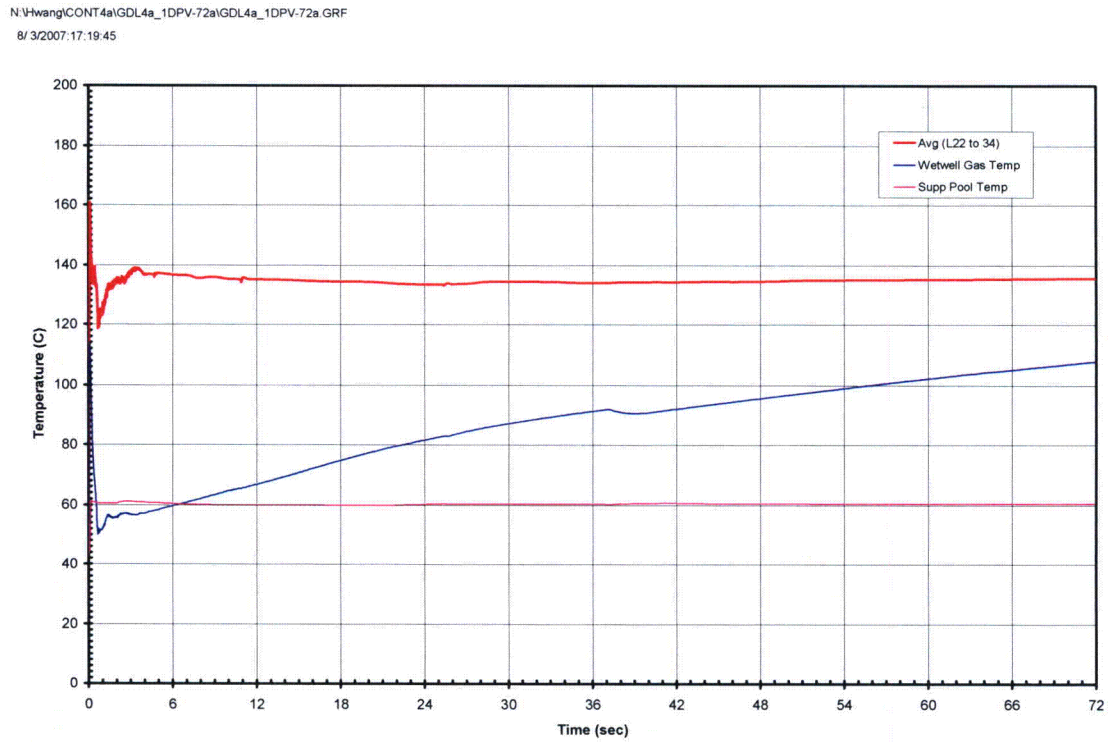


Figure 6.2-10b3. Main Steam Line Break (Nominal Case) – Containment Temperatures (2000 s)



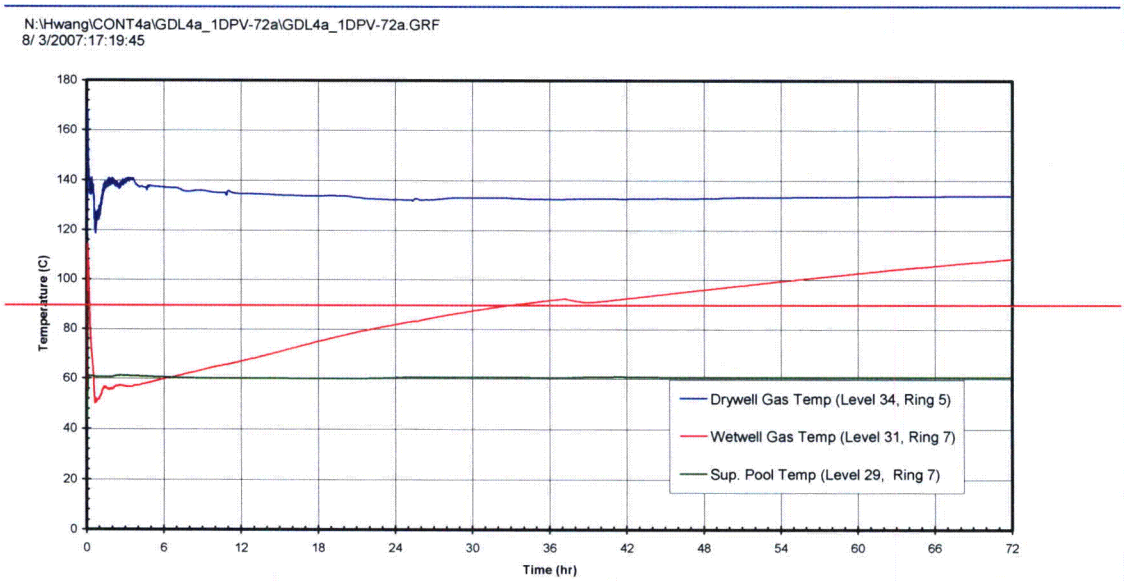
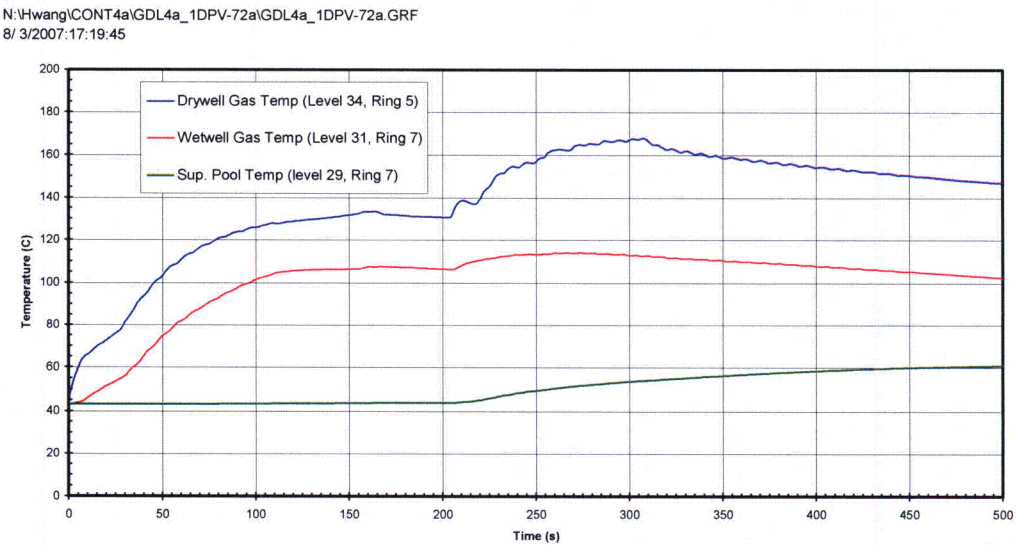


Figure 6.2-11b1. GDCS Line Break (Nominal Case) – Containment Temperatures (72 hrs)



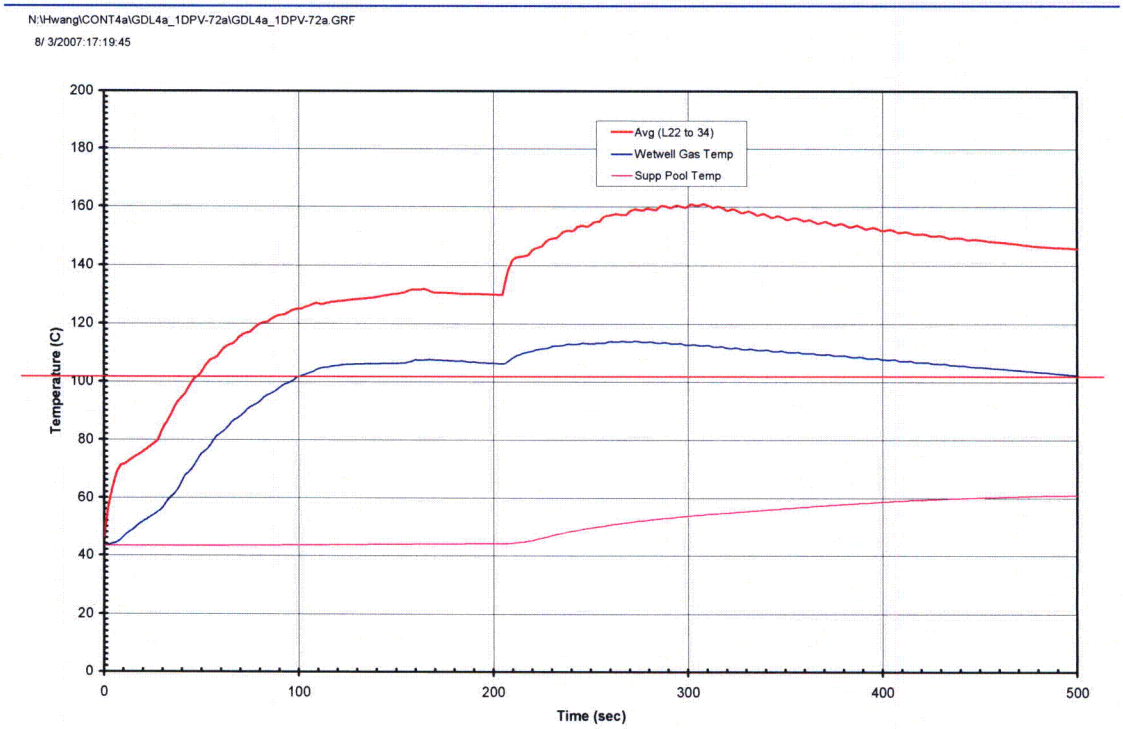
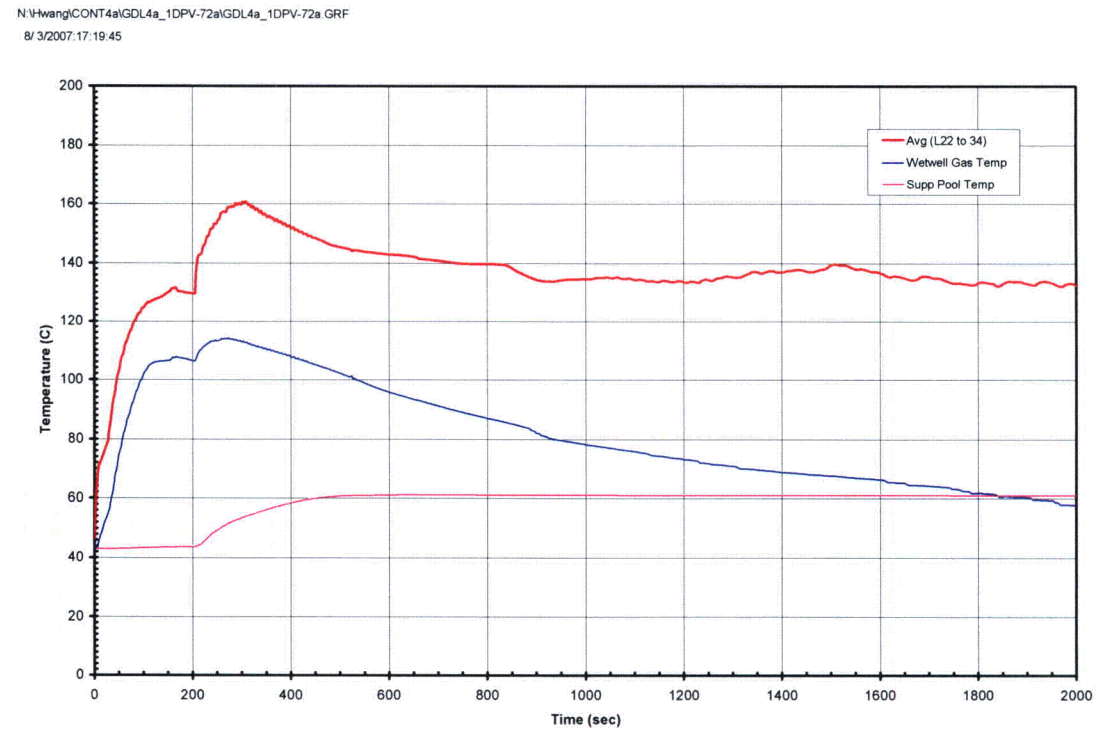


Figure 6.2-11b2. GDCS Line Break (Nominal Case) – Containment Temperatures (500 s)



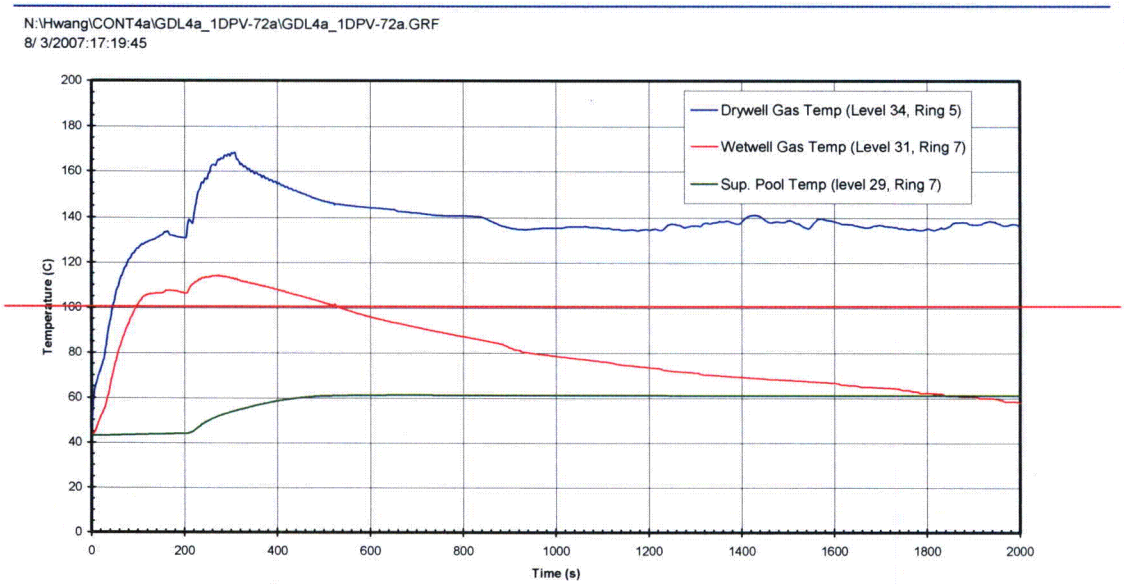
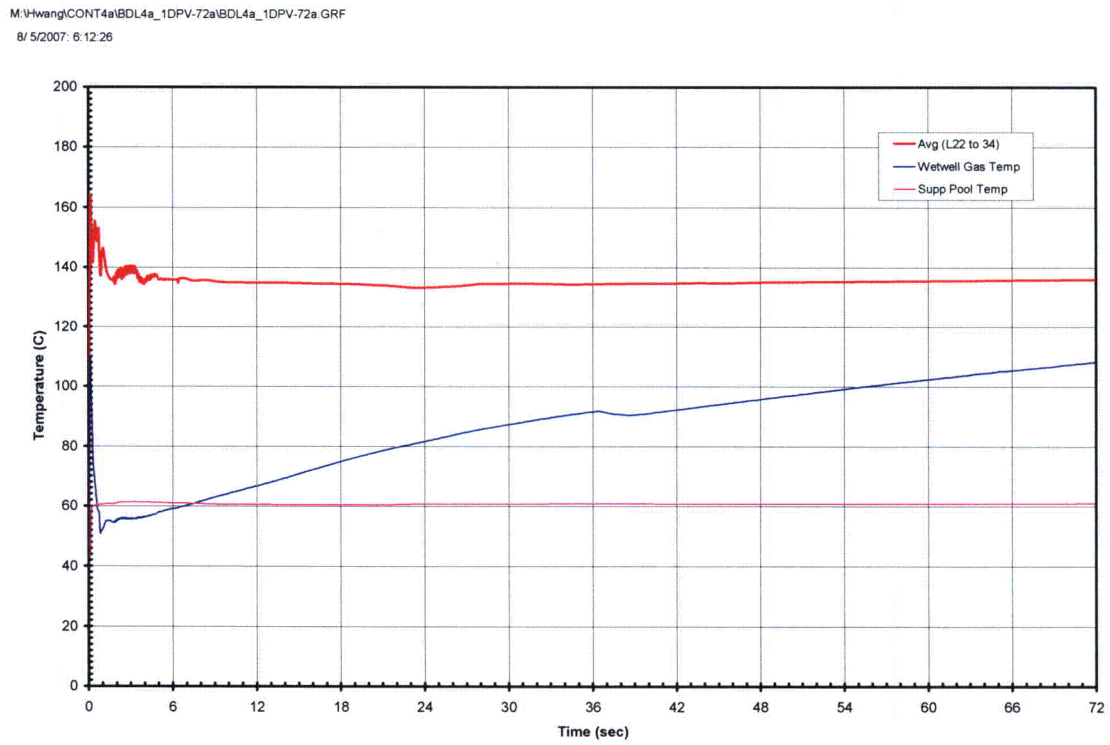


Figure 6.2-11b3. GDCS Line Break (Nominal Case) – Containment Temperatures (2000 s)



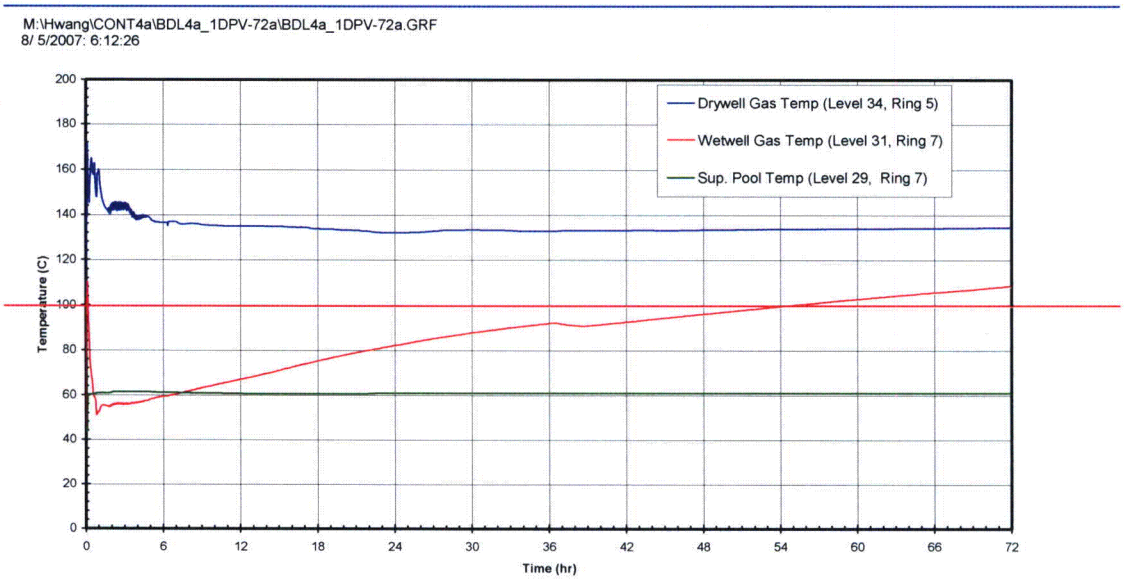
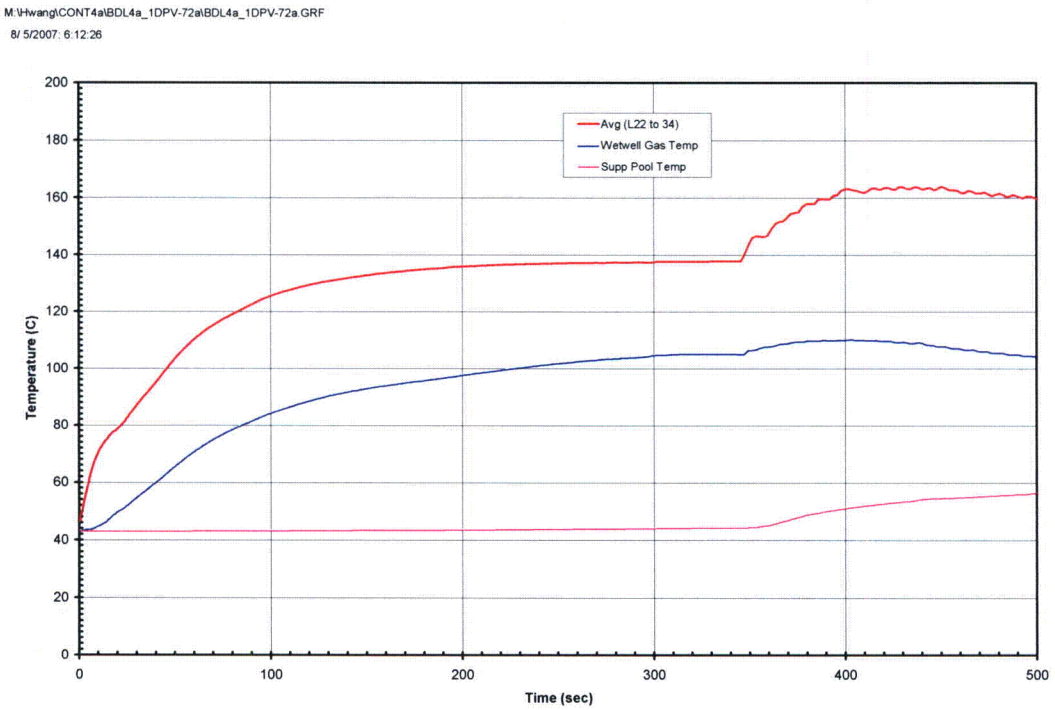


Figure 6.2-12b1. Bottom Drain Line Break (Nominal Case) – Containment Temperatures (72 hrs)



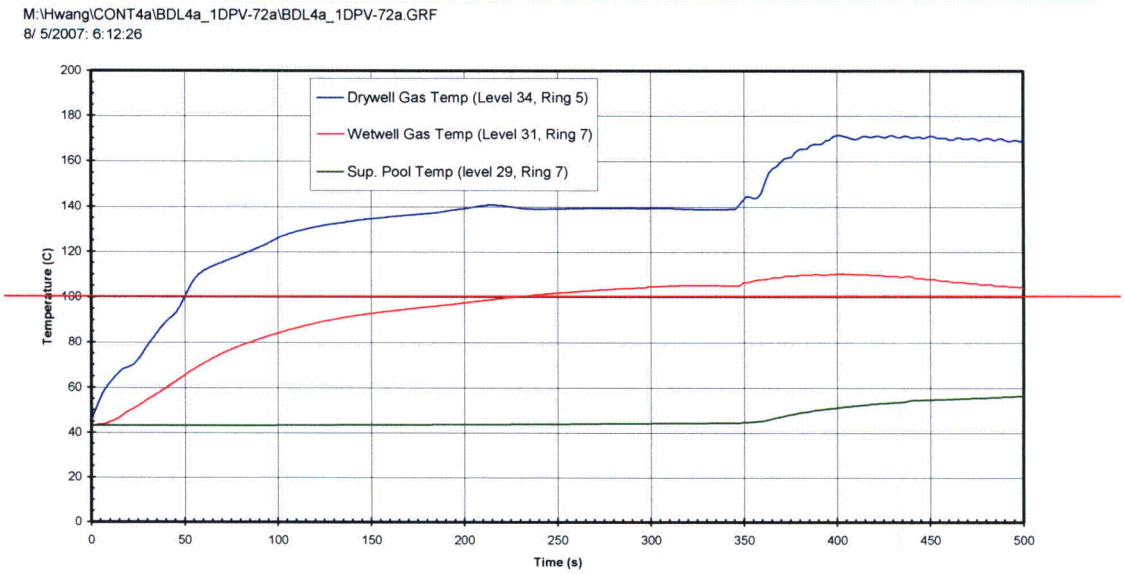
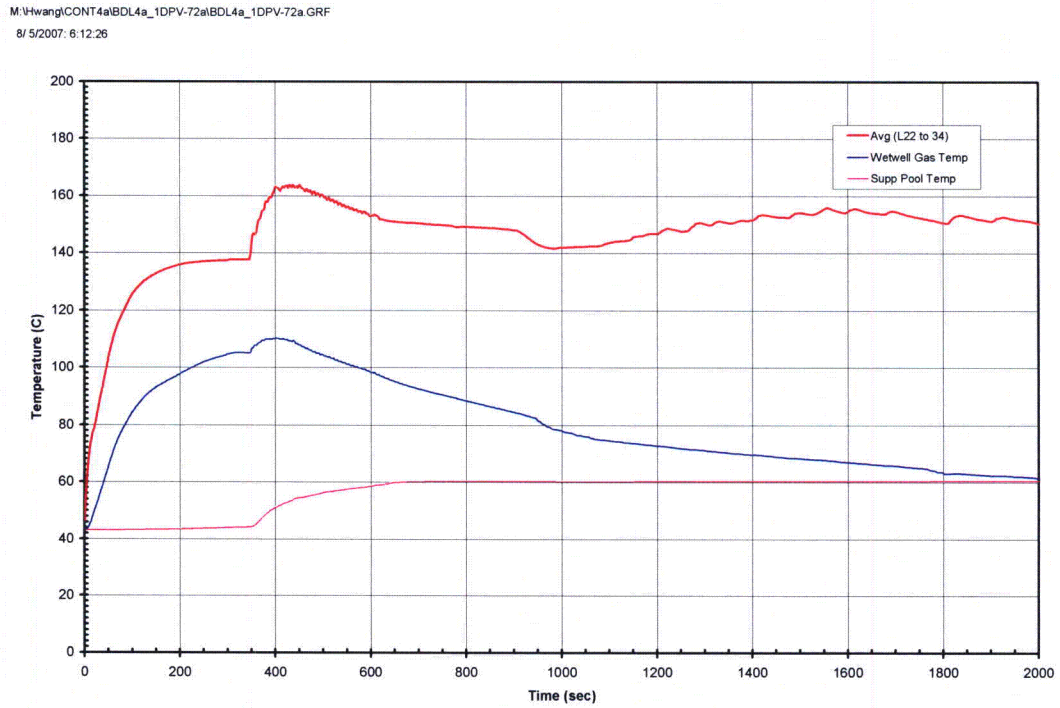


Figure 6.2-12b2. Bottom Drain Line Break (Nominal Case) – Containment Temperatures (500 s)



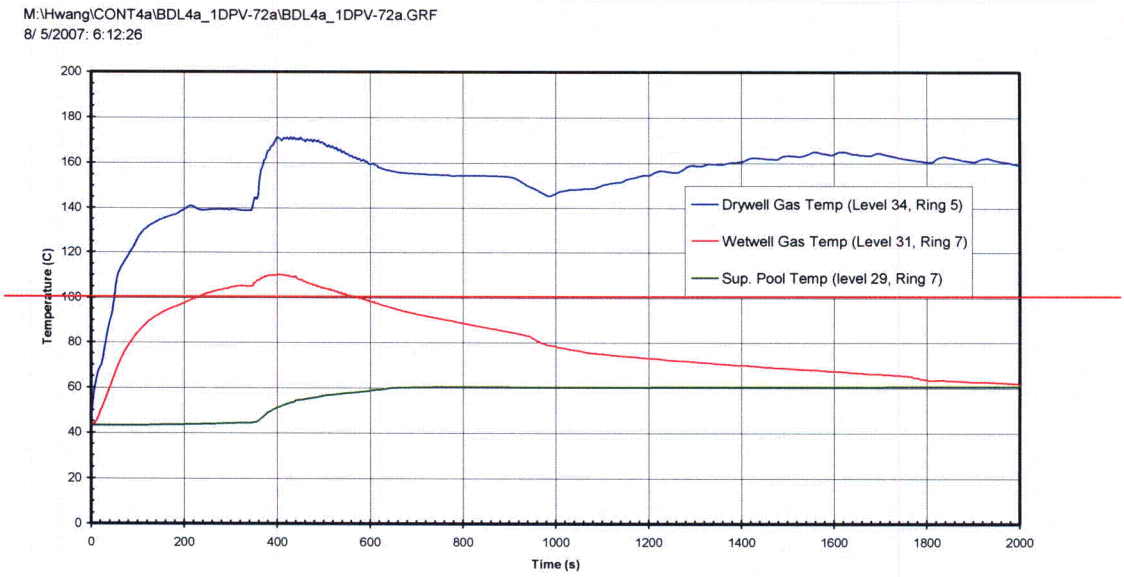


Figure 6.2-12b3. Bottom Drain Line Break (Nominal Case) – Containment Temperatures (2000 s)

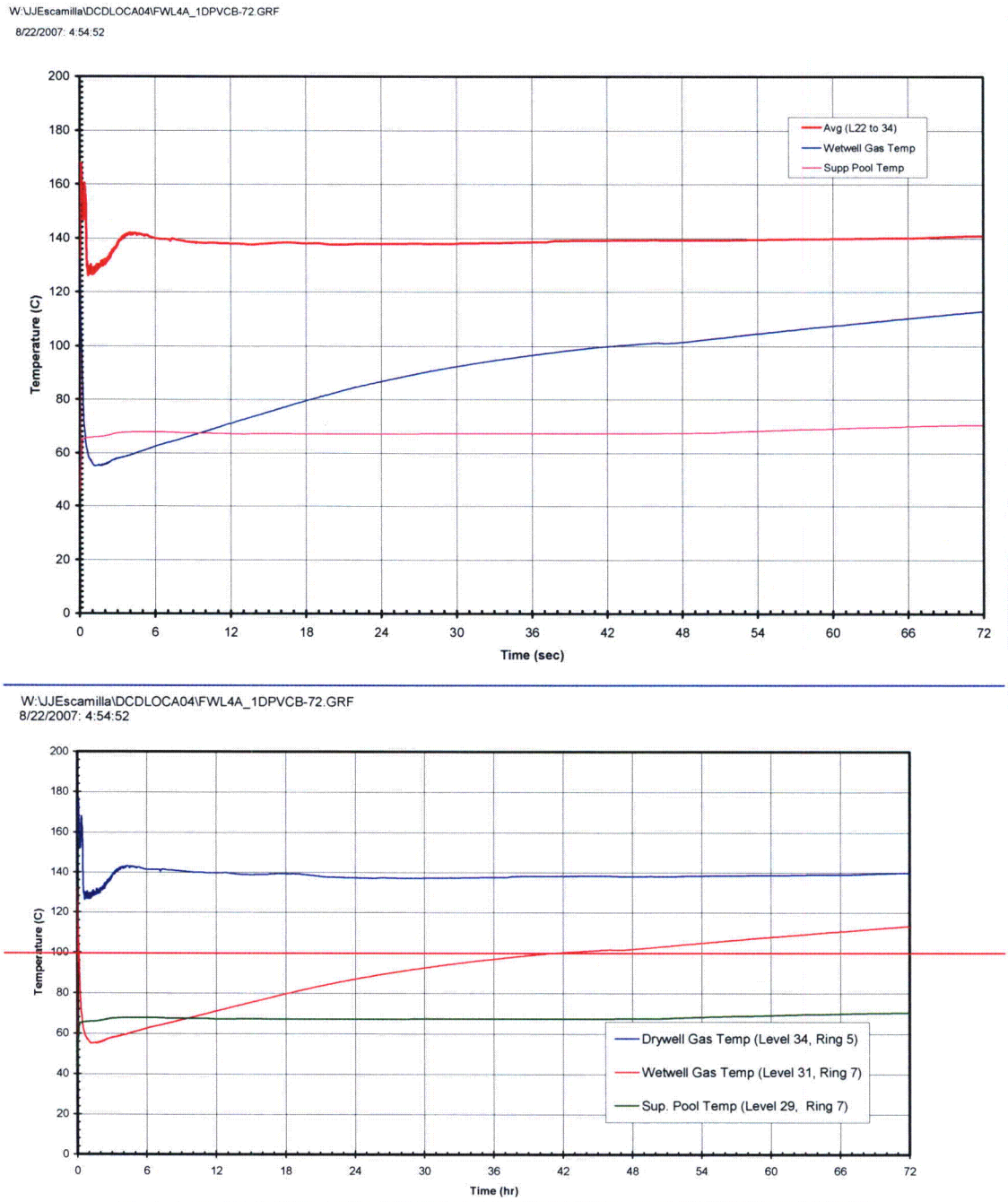
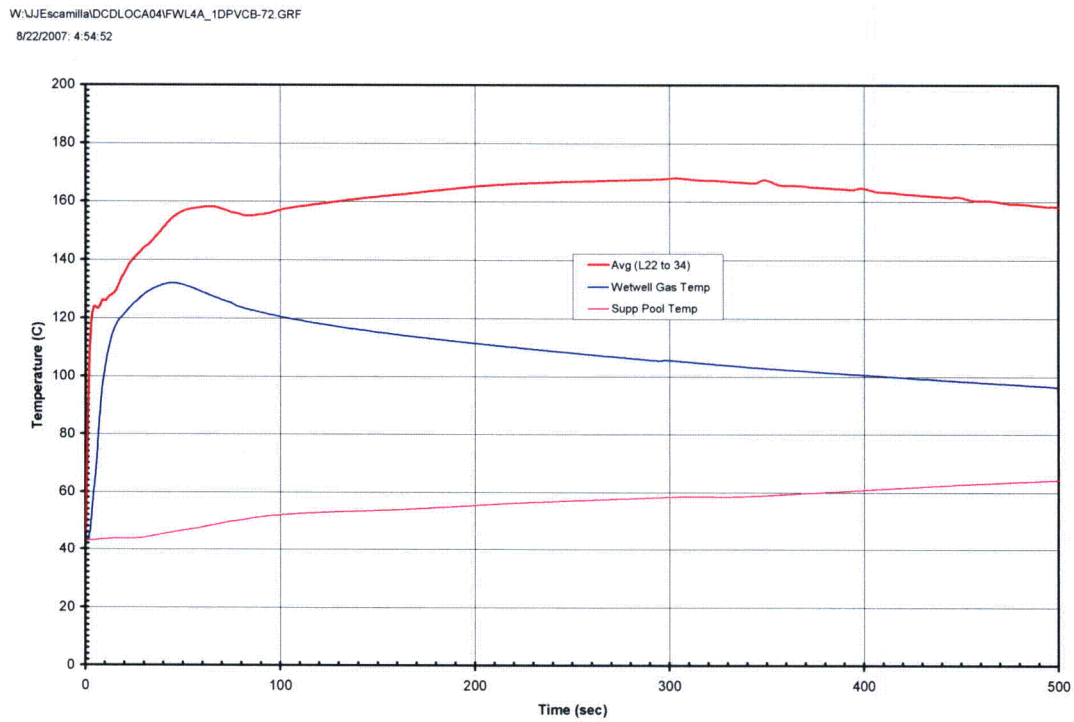


Figure 6.2-13b1. Feedwater Line Break (Bounding Case) – Containment Temperatures (72 hrs)



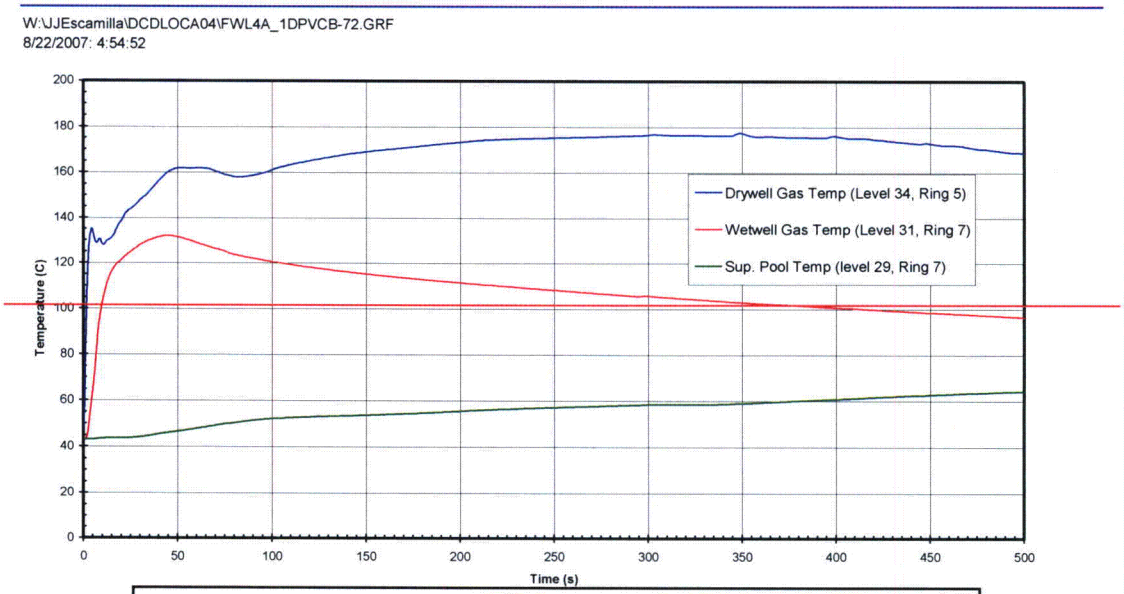
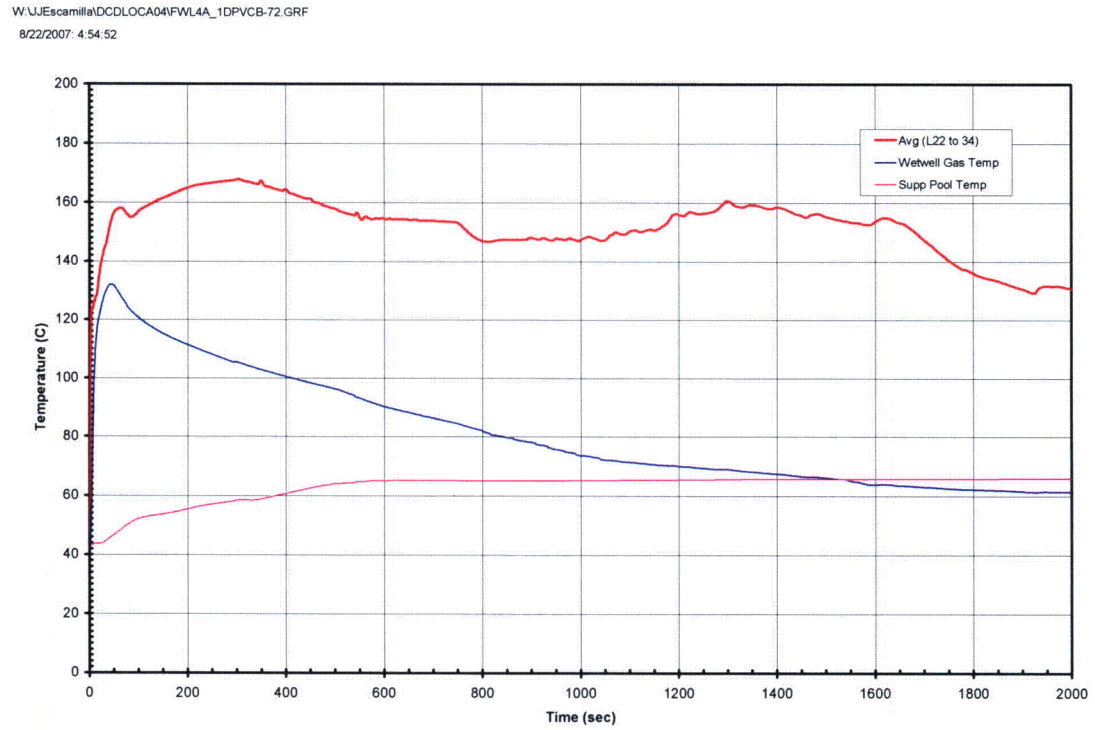


Figure 6.2-13b2. Feedwater Line Break (Bounding Case) – Containment Temperatures (500 s)



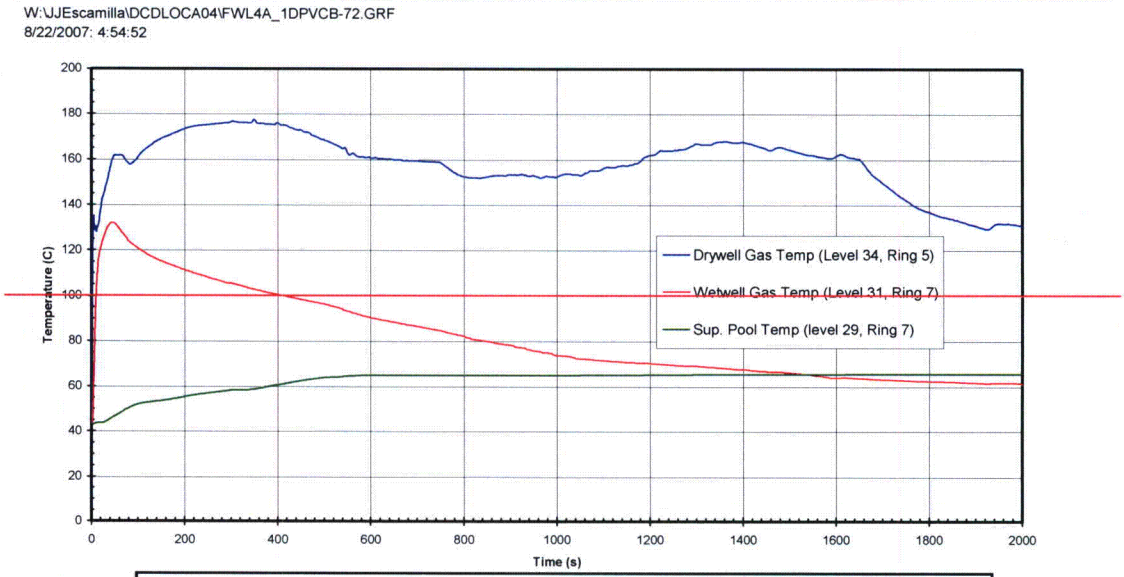


Figure 6.2-13b3. Feedwater Line Break (Bounding Case) – Containment Temperatures (2000 s)

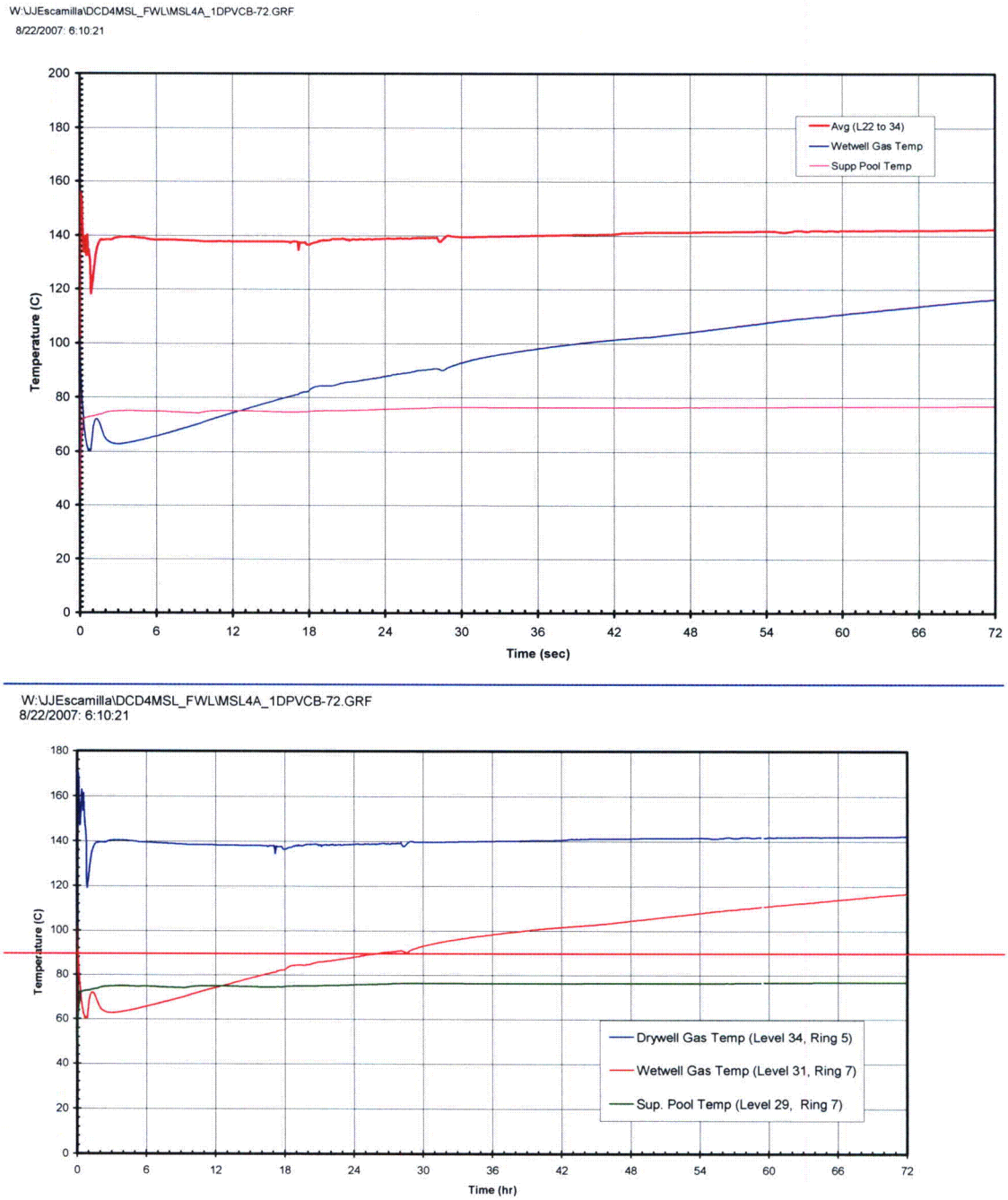
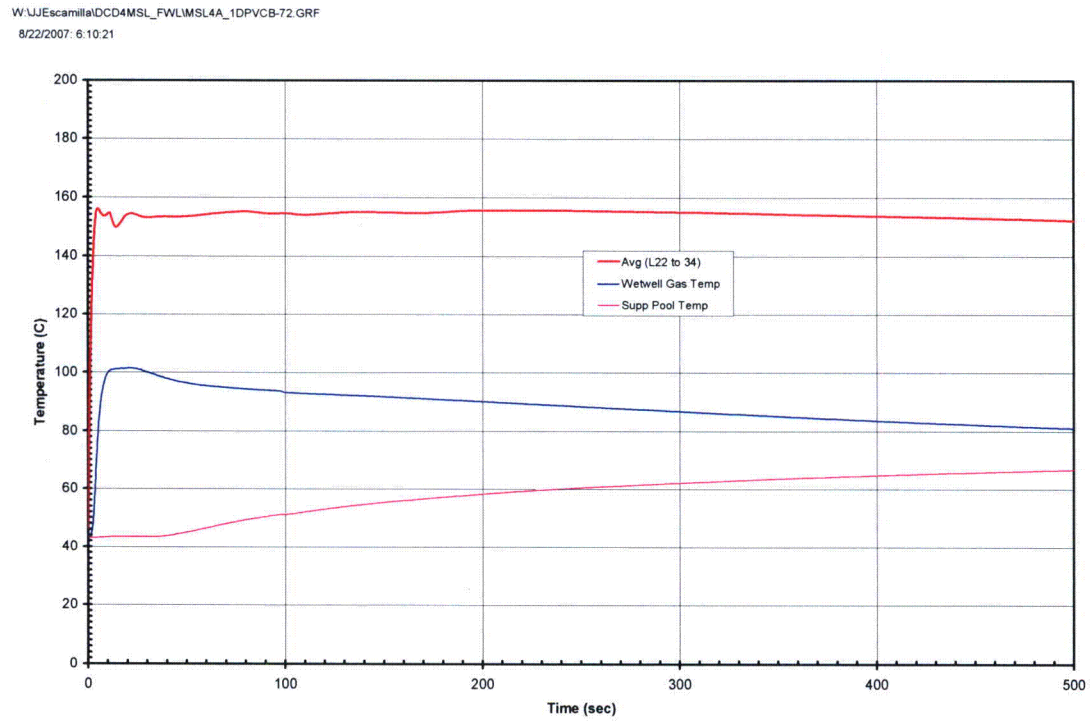


Figure 6.2-14b1. Main Steam Line Break (Bounding Case) – Containment Temperatures (72 hrs)



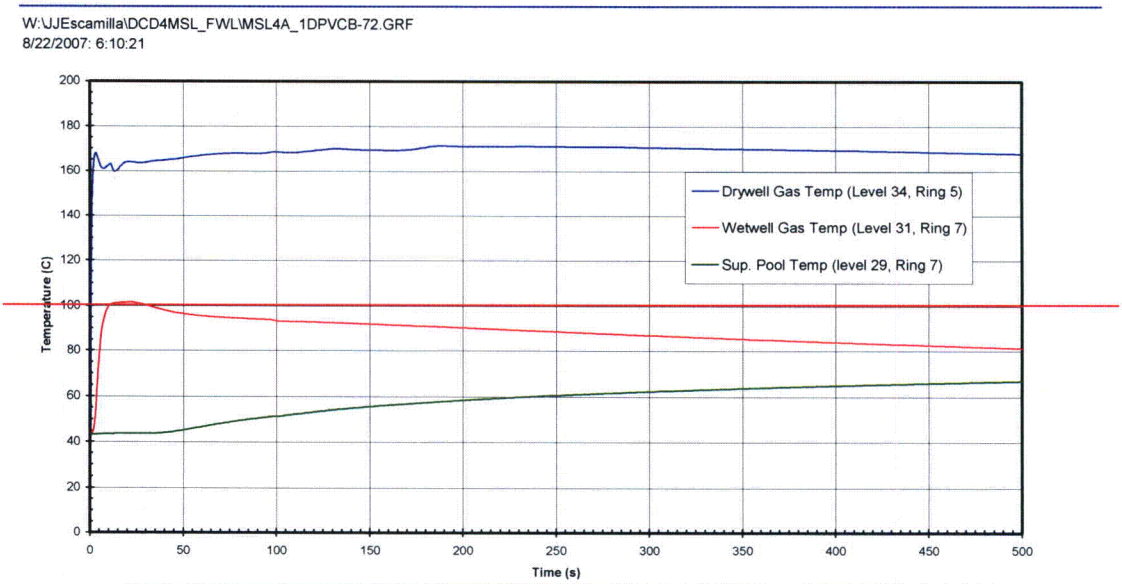
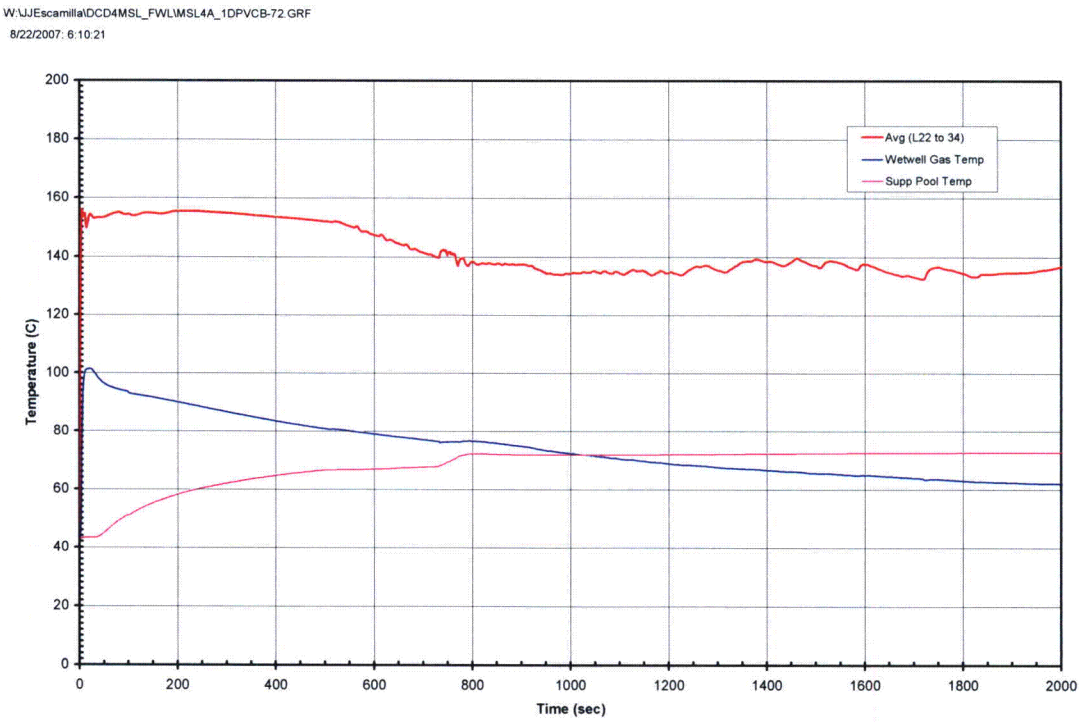


Figure 6.2-14b2. Main Steam Line Break (Bounding Case) – Containment Temperatures (500 s)



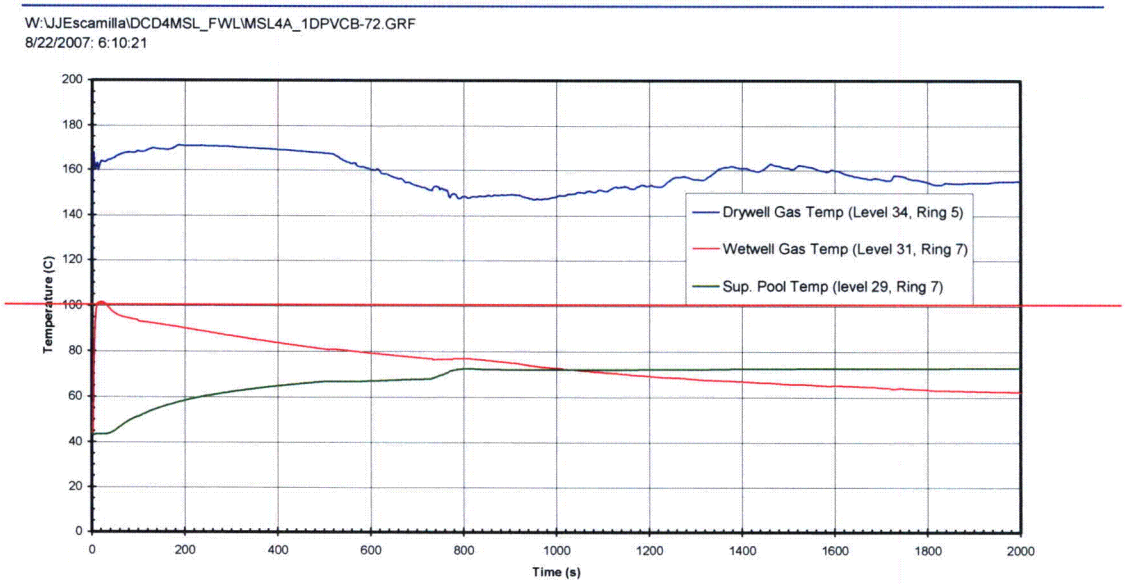


Figure 6.2-14b3. Main Steam Line Break (Bounding Case) – Containment Temperatures (2000 s)

6E. TRACG LOCA CONTAINMENT RESPONSE ANALYSIS

The chronology of progressions of Feedwater Line Break (FWLB) LOCA (Bounding Case) and Main Steam Line Break (MSLB) LOCA (Bounding Case) containment responses are discussed in detail in this appendix.

6E.1 Feedwater Line Break LOCA (Bounding Case)

Following the postulated FWLB LOCA, the Drywell pressure increases rapidly leading to the clearing of the Passive Containment Cooling System (PCCS) and main vents. The Drywell pressure increase is terminated at around 70 seconds (Figure 6.2-13a2), when most of the non-condensable (NC) gases in the Drywell annulus have been purged into the Wetwell (Figure 6.2-13d3).

As shown in Figure 6.2-13a2, the peak Drywell pressure is approximately 318 kPa (46.1 psia) at 347 seconds prior to the Gravity-Driven Cooling System (GDCS) flow initiation, shortly after the opening of the depressurization valves (DPVs) (Table 6.2-7d). This peak pressure is below the design pressure of 413.7 kPa (60 psia) with a large margin. The GDCS flow initiates at approximately 513 seconds. The subcooled GDCS water continues flowing into the vessel, reduces the steaming from the reactor pressure vessel (RPV) and the Drywell pressure. At approximately 800 seconds, the Drywell pressure drops below the Wetwell pressure, causing the opening of the vacuum breakers and allowing some NC gases to flow back into the Drywell. Consequently, the system pressures drop to a value of approximately 260 kPa (Figure 6.2-13a3).

Figures 6.2-13d1 through 6.2-13d3 show the NC gas pressures in the Drywell annulus, the Drywell head gas space, and the GDCS pool gas space. Figure 6.2-13d3 shows that most of the NC gases in the Drywell annulus are purged into the Wetwell within 100 seconds. At around 800 seconds, some NC gases flows back to the Drywell annulus (Figure 6.2-13d3) after the opening of the vacuum breakers.

Subsequently, decay heat overcomes the subcooling in the GDCS water and steaming resumes (at ~1800 seconds, Figure 6.2-13a3). The resumption of RPV steaming causes the Drywell pressure to increase again starting from 1800 seconds. The Drywell pressure reaches the long-term peak of 339 kPa (49 psia) at 72 hours (Figure 6.2-13a1).

After 1800 seconds, the Drywell pressure is higher than the Wetwell pressure. The PCCS moves the steam and NC gas mixture from the Drywell and purges the NC gases into the Wetwell. Most of the NC gases that returned to the Drywell annulus because of vacuum breaker openings are purged back into the Wetwell in approximately 3 hours (Figure 6.2-13d1).

Figure 6.2-13c1 compares the total heat removal by the PCCS with the decay heat. After the first 6 hours, the PCCS is able to remove all the decay heat with some margin to spare. From this point on, all the decay heat generated by the core is transferred to the Isolation Condenser (IC)/Passive Containment Cooling (PCC) pools, which are located outside of the containment.

Figures 6.2-13b1 through 6.2-13b3 show the Drywell gas temperature, Wetwell gas temperature, and suppression pool surface temperature. As shown in Figure 6.2-13b2, the bulk Drywell temperature reaches a maximum of 166.3°C at approximately 350 seconds. By the end of 72 hours, the Drywell temperature is maintained at 138.7°C, well below the Drywell design temperature of 171°C (340°F). While the Wetwell gas temperature remains below the Wetwell temperature design limit of 121.1°C (250°F) in 72 hours.

Figure 6E-1 compares the water levels in the Drywell annulus and suppression pool. The Drywell annulus water level rises due to the break flow discharges from the RPV and the broken feedwater piping. In approximately 10 hours, the Drywell annulus water level reaches the quasi-equilibrium elevation of 9 meters. At this elevation, the Drywell annulus water level is approximately 3 meters below the spillover holes. The hot water in the Drywell annulus will remain in the Drywell and not enter into the bottom of the suppression pool via the spillover holes.

Figure 6E-2 shows the GDCS pool levels. The water level drops below L33 (Figure 6.2-7) in approximately 4 hours, after the initiation of GDCS flow. This creates a bottom layer of NC gas space, which is approximately 6 meters below the connection pipes. The NC gas masses stored in the top two levels (L34 and L35) are purged to minimal values in a few hours, through the connection pipes.

6E.2 Main Steam Line Break LOCA (Bounding Case)

Following the postulated LOCA, the Drywell pressure increased rapidly leading to clearing of the PCCS and main vents. The containment pressure responses of the Drywell, Wetwell, and RPV are shown in Figures 6.2-14a1 through 6.2-14a3. During the initial blowdown phase, the NC gases in the upper regions of the Drywell are cleared rapidly to the Wetwell through the main vents, as shown by the Drywell and GDCS gas pressures in Figures 6.2-14d2 and 6.2-14d3. The Drywell pressure starts to turn around and decrease at approximately 930 seconds after the GDCS flow initiates, as shown in Figure 6.2-14a3. The GDCS flow fills the reactor vessel to the elevation of the break, and then spills into the Drywell. The heat transfer process during the GDCS cooling period is characterized by condensation of steam in the vessel and Drywell. At around 550 seconds, vacuum breaker openings occur as the steam production drops off, returning some of NC gases back to the Drywell. Subsequently, decay heat overcomes the subcooling of the GDCS water and steaming resumes. The Drywell pressure rises until flow is established through the PCCS, where condensate from the PCCS is recycled back into the vessel through the GDCS pool in the Drywell. It takes approximately 18 hours for the NC gases to be cleared from the GDCS gas space and 23 hours from the Drywell as shown in Figure 6.2-14d1. The long-term (72 hours) response of the containment pressure is shown in Figure 6.2-14a1. The Drywell pressure levels off at around 4 hours and increases in a gradual pace. The peak Drywell pressure reaches 384 kPa (55.8 psia) at the end of 72 hours, but remains below the design pressure of 414 kPa (60 psia).

Due to the presence NC gases, the effectiveness of condensation heat transfer in the PCCS is impaired for the first 28 hours, as shown by the PCCS heat removal versus decay heat in Figure 6.2-14c1. The spike in the heat removal for the time between 17 and 28 hours indicate

the period where the PCCS heat removal temporarily exceeds the decay heat, resulting in a drop in the Drywell pressure below the Wetwell pressure. This leads to a temporary cessation of steam flow to the PCCS. Subsequently the vacuum breakers open, some NC gases are returned to the Drywell, and the PCCS heat removal rate increases as the steam flow from the Drywell resumes. For the first 28 hours, the decay heat exceeds the PCCS heat removal capacity. Then after the NC gases are purged from the GDCS and Drywell gas spaces, the PCCS heat removal capacity matches the decay heat. However, it still cannot remove radiation heating generated in the core. Hence, both the Drywell and Wetwell pressures continue to increase continually over the 72-hour period.

Figures 6.2-14b1 through 6.2-14b3 show the Drywell, Wetwell and suppression pool gas space temperature responses at various elevations. Initially, all elevations heat up due to the main vent and break discharge. After the main vents close, only the upper levels are impacted by the PCCS vent discharge. The bulk Drywell gas temperature peaks at 156.2°C immediately following the blowdown, which is slightly below the Drywell design temperature of 171°C (340°F), due to adiabatic compression by the steam discharging from the break. In the long term (72 hours), the temperature remains below 171°C. While the Wetwell gas temperature remains below the Wetwell temperature design limit of 121.1°C (250°F) in 72 hours.

Figure 6E-3 compares the water levels in the Drywell annulus and suppression pool. The Drywell annulus water level rises due to the break flow discharges from the main steam line broken piping. The Drywell annulus water level will remain in the lower Drywell and not enter into the bottom of the suppression pool via the spillover holes.

For MSLB, the GDCS pool level (Figure 6E-4) drops to the elevation of the DPVs and stays above L33 (Figure 6.2-7). As no gas mass is stored below L33, gas masses stored in the top two levels (L34 and L35) are purged to minimal values in a few hours through the connection pipes. Approximately 60% (or 1200 m^3) of the initial GDCS cold water 46.1°C (115°F) remains inside the GDCS pools. The hot PCCS condensate mixes with the cold GDCS pool water. The temperature of the GDCS water injected into the RPV is lower than that of the FWLB. A portion of the decay heat is used to heat up the incoming colder GDCS water, which is shown in Figure 6.2-14c3 as the difference between the decay heat and the total PCCS condensation power. It takes approximately 60 hours to bring the mixture temperature to approximately 95°C.

Both the initial blowdown and the ensuing suppression pool heat up are larger in the MSLB than the FWLB, which leads to higher pool water temperature and higher suppression pool water level and smaller Wetwell gas space volume. All these effects result in higher Drywell pressure in the MSLB than FWLB.