3.7 Special Considerations for BWR Facilities

Boiling water reactors have unique features that would cause their behavior under severe accident conditions to differ significantly from that expected for the pressurized water reactor design.[1,2] This section addresses several special considerations affecting BWR severe accident progression and mitigation. In this endeavor, many implications of the phenomena described in Sections 3.1 through 3.6 (such as zirconium oxidation) will be demonstrated by example. First, however, it is necessary to review some of the BWR features pertinent to severe accident considerations.

3.7.1 Pertinent BWR Features

An important distinction of the BWR design is that provisions are made for direct operator control of reactor vessel water level and pressure. Reactor vessel pressure control is normally accomplished rather simply by manually induced actuation of the vessel safety/relief valves (SRVs) or by operation of the reactor core isolation cooling (RCIC) system turbine or, for plants so equipped, the isolation condenser or high-pressure coolant injection system (HPCI) turbine. Each of these methods relies to some extent, however, upon the availability of DC power or control air, which may not be available under accident conditions. SRV considerations will be described in Section 3.7.2.5.

All BWR plant designs except Oyster Creek, Nine Mile Point 1, and Millstone 1 incorporate either the RCIC or HPCI steam turbine-driven reactor vessel injection system; the later BWR-3 and all BWR-4 plants have both. These systems can be used for reactor vessel pressure control when run continuously in the recirculation mode, pumping water from the condensate storage tank back to the condensate storage tank and periodically diverting a small portion of the flow into the reactor vessel as necessary to maintain the desired water level. The steam taken from the reactor vessel by the turbine is passed to the pressure suppression pool as turbine exhaust, which provides a slower rate of pool temperature increase than if the vessel pressure control were obtained by direct passage of steam from the vessel to the pool via the SRVs. Plants having both HPCI and RCIC systems can employ the HPCI turbine exclusively for pressure control while the RCIC system is used to maintain the reactor vessel water level. The HPCI turbine is larger than the RCIC turbine and, therefore, is more effective for pressure control. These systems require DC power for valve and turbine governor control, but have no requirement for control air.

All BWR-5 and -6 plants are equipped with an electric-motor-driven high-pressure core spray (HPCS) system rather than a turbine-driven HPCI system. The HPCS pump takes suction from the condensate storage tank and delivers flow into a sparger mounted within the core shroud. Spray nozzles mounted on the spargers are directed at the fuel bundles. As in the case of HPCI, the pressure suppression pool is an alternate source of water for the HPCS.

All BWR facilities employ the low-pressure coolant injection (LPCI) mode of the residual heat removal (RHR) system as the dominant operating mode and normal valve lineup configuration; the RHR system will automatically align to the LPCI mode whenever ECCS initiation signals such as low reactor vessel water level or high drywell pressure are sensed. LPCI flow is intended to restore and maintain the reactor vessel coolant inventory during a LOCA after the reactor vessel is depressurized, either by the leak itself or by opening of the SRVs.

All BWR facilities also employ a low-pressure core spray (LPCS) system, which takes suction on the pressure suppression pool and sprays water directly onto the upper ends of the fuel assemblies through nozzles mounted in sparger rings located within the shroud just above the reactor core. With the reactor vessel depressurized, the automatically-initiated LPCI and LPCS flows, which begin when the reactor vessel pressure-to-suppression pool pressure differential falls below about 2.00 MPa (290 psig), are large. As an example, for a 1065-MWe BWR-4 facility such as Browns Ferry or Peach Bottom, the combined flows would be more than 3.16 m^3/s (50,000 gal/min), which is sufficient to completely fill an intact reactor vessel in less than four minutes. It should be recalled that the amount of vessel injection necessary to remove decay heat (by boiling) is only about 0.013 m^3/s (200 gal/min).

Eight BWR design features have important implications with respect to differences (from PWR behavior) in the expected response of a BWR core under severe accident conditions. These are:

1. There is much more zirconium metal in a BWR core, which under similar conditions would increase the amount of energy released by oxidation and the production of hydrogen. Compared with a PWR of the same design power, a BWR typically contains about one and one-half times the mass of UO_2 and three times the mass of zirconium metal (counting both fuel rod cladding and channel box walls).
2. The BWR reactor vessel would be isolated under most severe accident conditions, due to closure of the main steam isolation valves (MSIVs). This tends to make the BWR severe accident sequence thermal hydraulic calculation simpler to perform, since natural circulation pathways through external loops such as hot legs and steam generators need not be considered.
3. Because of the marked reduction in the average radial power factor in the outer regions of the BWR core, degradation events would occur in the central core region long before similar events would take place in the peripheral regions.
4. SRV actuations would cause important pressure and water level fluctuations within the reactor vessel. Operator actions (mandated by emergency procedure guidelines) to depressurize the reactor vessel would lead to early (and total) uncovering of the BWR core.
5. Diversity of core structures (control blades, channel boxes, fuel rods) would lead to progressive downward relocation of different materials from the upper core region to the core plate.

6. With early material accumulation upon its upper surface, the fate of the BWR core plate determines whether the initial debris bed would form within the lower portion of the core or in the vessel lower plenum.

7. Because there are many more steel structures in the BWR lower plenum, BWR debris would have a much greater steel content.

8. There is a much larger volume of water (relative to the core structural volume) within the lower plenum beneath a BWR core. If conditions are favorable, debris relocating from the core region can be completely quenched - with sufficient water remaining to remove decay heat (by boiling) for several hours without makeup.

The importance of each of these items will be elucidated in the discussions of Sections 3.7.2 through 3.7.7. Item 3, however, deserves special amplification here. Figure 3.7-1 illustrates a typical division of a BWR core into radial zones for code computation purposes. This example is based upon the Browns Ferry Unit 1 core, which comprises 764 fuel assemblies. Since a symmetric core loading is maintained, the drawing shows just one-fourth (191 assemblies) of the core.

What should be noted from Figure 3.7-1 is that the outer 25.1% of the core (sum of volume fractions for zones 9 and 10) is characterized by average radial peaking factors of just 0.670 and 0.354. This dramatic falloff is illustrated by Figure 3.7-2, which also indicates the volume-averaged central region power factor (1.199) associated with a four-radial-zone code representation of the core. Because of the associated reductions in decay heating beyond the central region of the core, predicted severe accident events in the central region lead those in the peripheral regions by considerable periods of time. For example, formation and downward relocations of large amounts of debris are calculated for the central region before structural degradation is predicted to begin in the outermost core region.

3.7.2 Provisions for Reactor Vessel Depressurization

The BWR Owners Group Emergency Procedure Guidelines (EPGs)[3] require unequivocally that the operators act to manually depressurize the reactor vessel should the core become partially uncovered under conditions (such as station blackout) characterized by loss of injection capability. The operators would meet this requirement by use of the Automatic Depressurization System (ADS). The following discussions address why manual actuation of an "automatic" system is necessary, what is expected to be achieved by the rapid depressurization, the status of the core during the subsequent periods of structural degradation (if the accident is not terminated), and the importance of keeping the reactor vessel depressurized during the latter stages of the accident.

3.7.2.1 Why Manual Actuation is Necessary

The most direct means of BWR reactor vessel pressure control is by use of the SRVs, which require no outside energy source for operation as a safety valve but do require both control air and DC power when used as a remotely operated relief valve. This dependence upon the availability of control air and DC power pertains both to remote-manual opening of the valves by the control room operators and to the valve-opening logic of the ADS.

The purpose of the ADS is to rapidly depressurize the reactor vessel so that the low-pressure emergency core cooling systems (ECCS) can inject water to mitigate the consequences of a small or intermediate loss-of-coolant accident should the high-pressure systems prove inadequate. The number of ADS-associated SRVs is plant-specific; these valves are signaled to open automatically if required to provide reactor vessel depressurization in response to low reactor vessel water level caused by transients or small breaks. ADS initiation is by coincidence of low reactor vessel water level and high drywell pressure, provided that at least one of the low-pressure pumps is running. Recently, a bypass timer (typically 265 seconds) has been backfitted to the ADS logic to ensure automatic actuation of ADS on sustained low water level even if the high drywell pressure signal is not present.

There is, however, no timer bypass for the requirement that at least one of the low-pressure ECCS pumps (RHR or Core Spray) be running. (The actual signal is derived by sensing the pump discharge pressure.) This is reasonable, since a great deal of water is lost from the reactor vessel when the ADS is actuated and therefore it is prudent to require that a replacement water source be available. As explained in the next Section, however, it is desirable to actuate the ADS under certain severe accident situations even though there is no operating pump. Without the discharge pressure signal, the ADS must be actuated manually (operator pushbuttons).

(NOTE: Typically, the ADS timer is initiated when the reactor vessel water level is between two and three feet above the top of the core. Current EPGs direct that the operators prevent automatic actuation of the ADS and instead manually initiate this system when the water level reaches the top of active fuel. This intentional delay of ADS for cases when the low pressure pumps are running is a matter of controversy, and not all BWR facilities invoke this provision.)

3.7.2.2 Rapid Depressurization for Steam Cooling

For BWR accident sequences involving partial uncovering of the core, the EPGs provide that the operators must take action to initiate "steam cooling" which, for plants without isolation condensers, is accomplished by manually initiating the ADS. The purpose is to delay fuel heatup by cooling the uncovered upper regions of the core by a rapid flow of steam. Because the source of steam is the remaining inventory of water in the reactor vessel, the steam cooling maneuver can provide only a temporary respite.

In order to illustrate the effects of steam cooling, let us first consider a case in which this maneuver is *not* used. Figure 3.7-3 shows the calculated reactor vessel water level for a postulated loss of injection (caused by station blackout and failure of RCIC) at Grand Gulf. It should be noted that after falling below the top of active fuel (TAF), the calculated curve follows the exponentially decreasing water level predicted by Figure 3.2-1 until a downward deviation becomes apparent, near the bottom of active fuel (BAF). This deviation occurs because debris relocating from the upper, uncovered, region of the core is relocating downward into the water remaining in the lower portion of the core, accelerating its boiloff.

Let us now consider the same accident sequence, but with implementation of the steam cooling maneuver. Figure 3.7-4 shows the calculated reactor vessel pressure and water level when the ADS is actuated at about one-third core height (75 minutes after scram). At this time, there has been no degradation of the upper core. At Grand Gulf, 8 SRVs (of 20 total) are associated with the ADS. The vessel depressurizes quickly and the accompanying water loss due to flashing causes the water level to fall into the lower plenum, well below the BAF and the core plate. Subsequently, the flashing ceases, and the remaining water is significant for debris quenching.

The maximum fuel rod temperature in the central region of the core is plotted versus time in Figure 3.7-5, for both cases. The temperature escalations that occur after time 80 minutes for the case without ADS are caused by the energy releases associated with zirconium oxidation. (The dotted lines on this figure indicate the time at which the temperature increases above 1832°F [1273°K]). For the case with ADS actuation, the temperature decreases immediately after the valves are opened due to the effects of steam cooling. Subsequently, the temperature again increases, but the time at which the runaway zirconium oxidation temperature is reached has been delayed by about 15 minutes. The differences in hydrogen generation between the two cases during the period plotted are substantial, as can be appreciated by a comparison of the two subplots of Figure 3.7-6.

Table 3.7-1 displays the times associated with the major events of the accident sequence for both cases. When the ADS is actuated, core plate dryout follows immediately thereafter, and debris begins relocating from the upper core region at about time 110 minutes. Without ADS actuation, debris relocation begins about 23 minutes earlier and before core plate dryout, which is delayed until time 102.5 minutes.

It is instructive to consider why the first local core plate failure occurs *earlier* for the case with ADS actuation. Recall that the core plate is dry when debris relocation begins in this case, so that the hot debris falls directly on the plate surface. When the ADS is not actuated, the initial debris relocations fall into water overlying the core plate. This initial debris is quenched and forms a protective layer over the plate. Later, when water no longer remains over the plate, failure is delayed until the newly relocating debris has heated both the plate and the previously quenched debris.

It is important that the ADS be manually initiated at the proper time. Too soon means that reactor vessel water inventory will be lost without the compensatory benefits afforded by effective steam cooling. Too late means that a steam-rich atmosphere will exist during the onset of runaway metal-water reactions. By procedure, steam cooling is to be placed into effect when the "Minimum Zero-Injection RPV Water Level" is reached. In Revision 4 of the EPGs, this is defined as the lowest vessel level at which the average steam generation rate within the covered portion of the core is sufficient to prevent the maximum clad temperature in the uncovered region of the core from exceeding 1800°F (1255 K). This level is plant-specific; the basis for its determination and procedures for its calculation are described in Appendices to the EPGs.

3.7.2.3 Core Region Dry During Core Degradation

As explained in the previous section, the delay in the onset of core damage gained by use of the steam cooling maneuver is temporary. Nevertheless, staving off the onset of core degradation, which otherwise would begin at about 78 minutes, for an additional 15 minutes can be significant when trying to regain electrical power or implement other means of restoring reactor vessel injection capability. Even if such efforts are unsuccessful so that the accident sequence proceeds into core degradation, the steam cooling maneuver provides the benefit of assuring that the core region will be steam-starved when runaway metal-water reaction temperatures are reached.

When considering severe accident progression for BWRs, it is extremely important to recognize that when the specified procedures are followed, the core region would be dry during the period of core degradation. As illustrated in the water level plot included with Figure 3.7-4, execution of the steam cooling maneuver causes the water level to fall below the core plate. This plot represents the results calculated when the ADS valves are opened with the water level at about one-third core height, but the final level will fall below the core plate even if this maneuver is initiated with the water level near the top of the core (although the achieved fuel temperature decrease will be much less).

Figure 3.7-7 shows the water level relative to the core plate immediately after execution of the ADS maneuver. It should be noted that some water is trapped in the downcomer region surrounding the jet pumps. This occurs because the initial temperature of the water in the jet pump region is less than the temperature of the water in the core region. Hence, a lower proportion of the water in the downcomer region is flashed during the rapid vessel depressurization

3.7.2.4 Threat of Reactor Vessel Repressurization

The motivation for keeping the reactor vessel depressurized under severe accident conditions is, first, that the capacity for quenching of the debris relocating from the core region into the lower plenum is enhanced and, second, that relocation of molten debris into the relatively small BWR drywell would then be, should bottom head penetration failures occur, by gravity-induced flow and not by rapid vessel blowdown. For the less probable case that penetration failures do not occur, so that the vessel bottom head ultimately undergoes gross failure by creep rupture, the time of failure would be delayed by several hours. Keeping the reactor vessel depressurized eliminates direct heating concerns and greatly reduces the initial challenge to the integrity of the primary containment.

The chief threats that the reactor vessel may be pressurized at the time of bottom head failure arise from two considerations, one derived from the potential for equipment failure and the other derived from the possibility of operator error. The question of equipment failure is chiefly associated with the long-term station blackout accident sequence, for which injection capability is maintained until the unit batteries are lost. With loss of the batteries, the ability to operate the SRVs manually is also lost. Because multiple SRVs are installed and operation of any one valve is sufficient for depressurization under severe accident conditions, improved reliability of SRV operation can be attained simply by ensuring that the small amount of DC power and control air necessary for opening will be available (to at least one valve) when required. Several BWR utilities have taken steps toward this end such as provision of a dedicated small DC diesel generator and backup compressed nitrogen supply bottles.

The impact of operator error upon vessel depressurization as typically represented in probabilistic risk assessments is direct. It is postulated that the operators fail to take the required action to manually depressurize the reactor vessel. Although the assumed probability for such failure, say 0.001, may seem low, typical core melt frequencies are much lower – on the order of 10^{-5}. Accordingly, it is important to recognize that such assumptions concerning operator error, while seeming reasonable and conservative, may lead to the unrealistic conclusions that BWR core melt, should it happen, would always occur in a pressurized vessel and that there is no point in providing equipment upgrades such as a dedicated DC generator since the operators would not use them anyway.

3.7.2.5 Notes Concerning SRV Operation

Any serious attempt to study and comprehend the probable course of an unmitigated severe accident sequence at a BWR facility *must* include development of a thorough understanding of the operation of the installed SRVs under abnormal conditions of reactor vessel and containment pressure. The pertinent characteristics of the more common SRV designs are described in the following paragraphs. The reader should particularly note that control air pressure sufficient for valve operation under normal conditions may not be adequate if the reactor vessel is depressurized and the containment pressure is elevated.

All SRVs are located between the reactor vessel and the inboard main steam isolation valves (MSIVs) on horizontal runs of the main steam lines within the drywell. The discharge from each valve is piped to the pressure suppression pool, with the line terminating well below the pool surface, so that the steam is subject to condensation in the pool. The number of SRVs varies from plant to plant (e.g., 11 at Limerick; 24 at Nine Mile Point 2), as do the rated relief valve flows.

Some operating BWRs are equipped with three-stage Target Rock valves, which have exhibited a greater tendency to stick open in the past than have other types of valves. Many BWR utilities, however, have replaced the original three-stage valves with the newer two-stage Target Rock valves (Figure 3.7-8). Some operating BWRs are equipped with Dresser electromatic relief valves. BWR-5 and BWR-6 plants are equipped with Crosby and Dikkers dual function SRVs (Figure 3.7-9).

The differences in SRV operation in the automatic and remote-manual or ADS modes can be demonstrated with reference to the two-stage Target Rock design shown in Figure 3.7-8. During normal reactor operation, a small piston orifice serves to equalize the steam pressure above and below the main valve piston, and the main valve disk remains seated. The reactor vessel pressure (valve inlet pressure) is ported via the pilot sensing port to tend to push the pilot valve to the right. When the reactor vessel pressure exceeds the setpoint established by the setpoint adjustment spring, the pilot valve is moved to the right, the stabilizer disk is seated, and the volume above the main valve piston is vented to the valve outlet via the main valve piston vent. The sudden pressure differential causes the main valve piston to lift, opening the valve.

For the remote-manual or ADS modes, the SRV opening is initiated by control air, which is admitted via a DC solenoid-operated valve (not shown) to the air inlet at the right of the setpoint adjustment spring. The control air moves the valve actuator to the right (against drywell pressure), which compresses the setpoint adjustment spring and pulls the pilot valve open, seating the stabilizer disk and venting the space above the main valve piston. Because the control air pressure and the reactor vessel pressure work in tandem to move the pilot valve to the right, the amount of control air pressure required to open the SRV will depend upon the reactor vessel-to-drywell pressure differential. Also, because the control air acts to move the air actuator against drywell pressure, the required control air pressure will increase with drywell pressure.

(It should be noted that the three-stage Target Rock valves behave differently with respect to the effect of the reactor vessel-to-drywell pressure differential. A good description of the operation of this older valve design is available in Reference 4.)

The spring-loaded direct-acting SRV shown in Figure 3.7-9 is opened in the spring mode of operation by direct action of the reactor vessel pressure against the disk, which will pop open when the valve inlet pressure exceeds the setpoint value. In the power-actuated mode, a pneumatic piston within the air cylinder moves a mechanical linkage to compress the spring and open the valve. As in the case of the two-stage Target Rock valve, the control air is provided via DC solenoid-operated valves, and the air pressure required for valve opening decreases with reactor vessel pressure and increases with drywell pressure.

All SRVs associated with the ADS are fitted with pneumatic accumulators (located within the drywell) to ensure that these valves can be opened and held open for some (plant-specific) period following failure of the drywell control air system. For severe accident considerations, it is important to recall that remote operation of the SRVs is possible only as long as DC power remains available and the pneumatic supply pressure exceeds the containment pressure by some minimum amount.

3.7.3 Recriticality Concerns

The progression of damage and structural relocation of the various components (control blades, channel boxes, fuel rods) of a BWR core during an unmitigated severe accident sequence will be discussed in detail in Section 3.7.4. There it will be shown that the first structures to melt and relocate downward are the control blades. Here we pause to consider severe accident sequences that have the potential for early termination, i.e., accident sequences for which the core structure sustains significant damage but reactor vessel injection capability is restored while the major portion of the fuel remains above the core plate.

If significant control blade melting and relocation were to occur during a period of temporary core uncovering, then criticality would follow restoration of reactor vessel injection capability if the core were rapidly recovered with cold unborated water using the high-capacity low-pressure injection systems.[5] Obviously, a neutron poison should be introduced into the reactor vessel for reactivity control under these circumstances, but question arises as to how best to do this. The normal means of adding boron to the reactor vessel is by injection with the standby liquid control system (SLCS). Although this system is designed to inject sufficient neutron-absorbing sodium pentaborate solution into the reactor vessel to shut down the reactor from full power (independent of any control rod motion) and to maintain the reactor subcritical during cooldown to ambient conditions, the SLCS is not intended to provide a backup for the rapid shutdown normally achieved by scram.

As indicated in Figure 3.7-10, the basic SLCS comprises a heated storage tank, two 100% capacity positive displacement pumps, and, as the only barrier to injection into the reactor vessel, two explosive squib valves. In most of the current BWR facilities, the sodium pentaborate solution enters the reactor vessel via a single vertical sparger located at one side of the lower plenum just below the core plate. However, so as to improve the mixing and diffusion of the injected solution (which has a specific gravity of about 1.3) throughout the core region, some BWR facilities have been modified to provide a third displacement pump and to permit the injected solution to enter the reactor vessel via the core spray line and sparger.

For the purpose of reducing the time required for reactor shutdown for the ATWS accident sequence, the NRC has issued a Final Rule[6] requiring that the SLCS injection be at a rate *equivalent* to 86 gal/min (0.0054 m^3/s) of 13 wt.% sodium pentaborate solution, the boron being in its natural state with 19.8 at.% of the boron-10 isotope. With this increased injection rate, sufficient boron for hot shutdown can be pumped into the reactor vessel in about 20 minutes, and for cold shutdown in about 48 minutes. It requires approximately an hour to inject the entire contents of the tank.

The operators would have no direct means of knowing whether significant control blade relocation had occurred. Thus, there is a strong potential for surprise should, for example, a station blackout accident sequence suddenly be converted into an uncontrolled criticality upon restoration of electrical power and reactor vessel injection capability. If the SLCS is used to inject sodium pentaborate at a relatively slow rate while the core is rapidly recovered with unborated water using the high-capacity, low-pressure injection systems, then criticality would occur and the core would remain critical until sufficient boron for shutdown (at the prevailing temperature) reached the core region. To avoid the possibility of temporary criticality, it would be desirable to inject effective quantities of boron along with the ECCS flow being used to recover the core. A strategy to accomplish this using only existing plant equipment but employing a different chemical form for the boron poison has been proposed.[7]

The only currently available information concerning the poison concentration required is derived from a recent Pacific Northwest Laboratory (PNL) study,[5] which indicates that much more boron would have to be injected than is available (as a solution of sodium pentaborate) in the SLCS. Furthermore, the dominant loss-of-injection accident sequence is station blackout, and without means for mechanical stirring or heating of the injection source, the ability to form the poison solution under accident conditions becomes of prime importance. Hence the need for the alternate chemical form.

The PNL study[5] provides the estimate that a boron-10 concentration of between 700 and 1000 ppm would be required within the vessel to preclude criticality once control blade melting had occurred. This is much greater than the concentration (about 225 ppm) attainable by injection of the entire contents of the SLCS tank.

At this point, it should be noted that the conclusions of the PNL study with respect to the boron concentrations required to preclude criticality are acknowledged by the authors of that study to be very conservative. Stated another way, in the many instances where it was necessary to make assumptions during the study, the assumed quantities were selected in a manner that tends to increase reactivity (promote criticality). As an example, debris particles are assumed to exist in the form of spheres. As discussed in the following paragraphs, the resulting high boron concentration requirement makes development of a practical coping strategy difficult.

One means to achieve such a high boron concentration would be to mix the powder directly with the water in the condensate storage tank during the blackout period and then, once electrical power is restored, to refill the reactor vessel by pumping the solution in a controlled manner using one of the low-pressure injection system pumps.

The condensate storage tank is an important source of water to the reactor vessel injection systems for each BWR unit. As indicated in Figure 3.7-11 (based upon the Browns Ferry arrangement), it is the normal suction source for the steam turbine-driven HPCI and RCIC systems and the alternate source for the electric motor-driven RHR and core spray pumps.

During normal reactor operation, the condensate storage tank provides makeup flow to the main condenser hotwells via an internal tank standpipe, as indicated on Figure 3.7-12. The purpose of the standpipe is to guarantee a reserve supply of water for the reactor vessel injection systems that take suction from the bottom of the tank. Any practical strategy for direct poisoning of the tank contents must include provision for partial draining to reduce the initial water volume, especially if boron-10 concentrations on the order of 700 ppm are to be established. The condensate storage tank can be gravity-drained through the standpipe to the main condenser hotwells under station blackout conditions.

Additional information concerning this example of a candidate accident management strategy and the characteristics of the alternate boron poison chemical form is available in Reference 7. It seems desirable that the very conservative estimates of the PNL study should now be replaced by more realistic estimates, which certainly would be expected to lower the target boron concentration from its present value of 700 ppm and thereby improve the practicality of such a strategy. (For example, Reference 8, which incorporates an assumption that three-fourths of the control blade B_4C remains in the core region, suggests that reflood water boron-10 concentrations as low as 200 ppm might be sufficient.) In the meantime, many of the BWR facilities have implemented accident management measures, on a voluntary basis, to provide backup capability for the SLCS. These backup strategies invoke such methods as modification of the HPCI or RCIC pump suction piping to permit connection to the SLCS tank, or poisoning of the condensate storage tank.

3.7.4 Eutectic Formation and Relocation Sequence for BWR Core Structures

This section addresses the progression of damage and structural relocation of BWR core components that would be expected to occur during an unmitigated severe accident sequence, i.e., an accident sequence for which reactor vessel injection capability is not restored. The BWR core is basically an assembly of unit cells, one of which is shown in the center drawing of Figure 3.7-13. As indicated, each unit cell comprises four fuel assemblies, each located in one of the four quadrants of a central control blade. Additional details concerning the fuel assembly and control blade internal compositions are shown in Figures 3.1-6 and 3.1-7, respectively.

As may be confirmed by an inspection of Figure 3.7-14, one-half of the channel box outer surfaces do not see an intervening control blade. This arrangement affects the local core heatup rates calculated for conditions in which a shutdown core (all control blades inserted) is postulated to be uncovered. Where the control blades exist, they serve as heat sinks for radiation from the adjacent channel box walls. Where there are no blades, the channel box walls radiate to each other.

Experiments to investigate the phenomena of core melt progression in prototypical BWR core geometries have been carried out in the Annular Core Research Reactor (ACRR) at Sandia National Laboratories (one BWR test) and at the CORA out-of-pile facility[9] at the Kernforschungszentrum Karlsruhe (KfK) in the former Federal Republic of Germany (six BWR tests). The first of these was the DF-4 experiment,[10] conducted within the ACRR in November 1986. The test apparatus, placed within the cylindrical region surrounded by the ACRR annulus, included a control blade arm, channel box walls, and 14 fresh fuel rods. The apparatus was dry, but the 20-inch (50-cm) long test section was supplied from below with a steam flow representative of BWR boiloff conditions.

When the DF-4 fuel rod cladding was heated beyond the runaway zirconium oxidation temperature, the energy release associated with oxidation accelerated the temperature escalation. Much of the clad melted at 2125 K (3365°F) and relocated downward; the remainder was converted to and remained in place as ZrO_2, which has a much higher melting point ([4900°F]2978 K).

The control blade in the DF-4 experiment melted earlier than expected and progressively and rapidly relocated downward. Subsequently, the reactor was shutdown to terminate power generation within the test assembly fuel rods before fuel melting could begin. In a post-test cross-section, the relocated control blade material was found in the form of an ingot at the very bottom of the test section, which was below the bottom of active fuel. Both the control blade and the channel box wall portions of the DF-4 test section were more than 90% destroyed due to melting and relocation during the experiment, but the fuel pellet stacks were predominantly still standing. Relocated cladding blocked the base of the fuel rod regions of the experiment.

Figure 3.7-15 illustrates the results of the DF-4 experiment, extrapolated to the same portion of the core that is represented in Figure 3.7-14. (Here the water rods, which were not included in the DF-4 experiment, have been assumed to relocate in the same time frame as the channel box walls.) The ramifications of these standing fuel pellet stacks in the absence of control blades with respect to the potential for criticality if water were to be introduced at this point in an actual accident sequence should be obvious.

The early control blade relocation observed in the DF-4 test was later determined to have been caused by a eutectic interaction between the control blade neutron absorber (B_4C powder) and the surrounding stainless steel of the blade structure. This occurred at a temperature well below the stainless steel melting point ([2600°F] 1700 K). This early B_4C-SS eutectic formation was also observed in the subsequent CORA BWR tests. The reaction proceeds rapidly when the local temperature increases above (2240°F) 1500 K, and sudden and complete liquefaction has been observed in CORA special-effects tests at (2372°F) 1573 K.[11] Post-test analyses[12] of the CORA BWR core melt experiments have found that a liquefaction temperature of (2250°F) 1505 K for the B_4C/SS combination best fits the observed structural failures and melt relocations.

The CORA experiments also demonstrated the formation of zirconium-based eutectics when the mixture formed from the destruction of the control blades at one axial level flows downward and comes into contact with the adjacent channel box wall at a lower axial level. Typically, this downward relocation of stainless steel and B_4C occurs in a series of rapidly repeated temporary freeze-remelt steps. Whenever the path of the downward-flowing liquid encounters a temporary blockage, some of the flow is diverted horizontally toward the blade tip and from there into the unbladed portion of the unit cell. In this manner, the liquid steel-B_4C mixture is spread through much of the lower portion of the unit cell.

When the gap between the control blade and the outer surface of the channel box wall becomes bridged by a semi-permanent blockage (low in the core), the continuously accumulating SS-B_4C liquid attacks the local channel box wall aggressively. The resulting zirconium-based eutectics are formed at the prevailing temperature of about 1523 K (2282°F). Thus, channel box wall failures follow soon (within minutes) after the onset of control blade failures.

To recap, based on the experimental record, structural damage within an uncovered BWR core is expected to be initiated when the temperature of the control blades in the upper regions of the core reaches about (2250°F) 1505 K. This is about (350°F) 195 K below the melting temperature of stainless steel. Within a few minutes thereafter, channel box wall damage would be initiated in the lower regions of the core, at local wall temperatures of about (2282°F) 1523 K. This is some (1080°F) 600 K below the melting temperature of zirconium. (It should be noted, however, that the destruction of the channel wall is not by melting, but rather by the process of dissolution by the liquid steel.) All of this structural damage occurs at temperatures far below the melting temperature [5400°F] (3011 K) of the UO_2 fuel.

As mentioned previously, it is important to note that the fuel rod pellet stacks, encased in thin ZrO_2 sheaths, continued to stand at the end of the DF-4 experiment. (CORA results are not germane to this question because these experiments were driven by electrically heated fuel rod simulators.) With the internal fission power heating, melting of the DF-4 cladding was initiated at the inner clad surface. This liquid zirconium then interacted with the outer surfaces of the UO_2 pellets to form a paste that, upon subsequent cooling, solidified in a manner that tends to glue the pellets together.

It is a valid question as to whether or not the fuel pellet stacks would continue to stand in an actual reactor accident. The DF-4 experiment employed fresh fuel, which had neither the local cracking nor the internal fission product inventory that would be present in actual fuel after long periods of power operation. Fission product release experiments with high-burnup fuel have demonstrated extensive fuel swelling and foaming.[13,14]

Finally, there remains the question as to the response of the hot fuel pellet stacks, should they remain standing, to the introduction of water. It is well known that hot cladding will shatter when thermally shocked, but the case of standing fuel has not been addressed for BWRs. Fuel collapse did occur in the upper portion of the TMI core when RCP 2B was temporarily restarted.

3.7.5 Potential Modes for Debris Movement Past the BWR Core Plate

The BWR core plate is located at the base of the core region within the lower portion of the core shroud, as shown in Figure 3.7-16. Although the core plate does not support the core, it separates the core region from the reactor vessel lower plenum and thus would serve as an impediment to the movement of core and structural debris into the lower plenum under severe accident conditions. In fact, the fate of the core plate is pivotal to the progression of a BWR severe accident; whether the core plate remains in place or fails and relocates will determine whether the debris bed comprising the materials accumulating below the active core forms over the plate surface or within the lower plenum.

3.7.5.1 Core Plate Structure

The core plate is basically a circular stainless steel plate strengthened by an underlying support structure. Figure 3.7-16 indicates the relative arrangement of the fuel assemblies, fuel support piece, control rod guide tube, control rod drive housing, and stub tube for one core unit cell. The primary functions of the BWR core plate are to laterally align the upper portions of the control rod guide tubes and to provide the partition that under normal operating conditions prevents flow from the lower plenum from entering directly into the core region. Instead, the flow enters holes (one for each fuel assembly) in the upper portion of each control rod guide tube and passes through the fuel support pieces into the fuel assemblies; one of these entrance holes is labeled "flow inlet into fuel bundle" on Figure 3.7-16.

To provide an illustrative example of the dimensions of the core plate, the following discussion is based upon the 251-in. ID BWR-4 reactor vessel installed at 1065-MWe facilities such as Peach Bottom or Browns Ferry. Each of the 185 control rod guide tubes supports four fuel assemblies via an orificed fuel support piece such as the one shown in Figure 3.7-17. The support piece rests within the upper portion of the control rod guide tube while the core plate provides an alignment pin to ensure proper placement of both the guide tube and support piece. The upper surface of the core plate is located about 23 cm (9 in.) below the bottom of active fuel within the fuel assemblies.

The core plate, which is 5 cm (2 in.) thick and weighs 9300 kg (20,500 lb), provides vertical support to only the 24 outermost fuel assemblies (of the 764 assemblies that make up the core). The support arrangement for one of these 24 peripheral assemblies is shown at the extreme right of Figure 3.7-17. In contrast to the four-lobed fuel support pieces used for the majority of the fuel assemblies, each peripheral fuel support carries only a single fuel assembly and is firmly seated within the core plate itself.

The stainless steel core plate surface resembles a perforated drum membrane, being penetrated by 185 large holes (28 cm [11 in.] ID) to accommodate the passage of the control rod guide tubes and 55 smaller holes (5 cm [2 in.] ID) for the in-core instrument guide tubes. The core plate is supported around its outer periphery, which is bolted to a ledge on the core shroud as indicated on Figure 3.7-18. Central support is limited to that provided by the stiffener plates and stiffener rods labeled "core plate support structure" on Figure 3.7-16.

An appreciation for the extent of the core plate, which has a diameter of about 4.9 m (16 ft.), can be gained from Figure 3.7-19. Under the conditions that would be imposed by a severe accident, the peripherally supported core plate is in position to assume the role of providing vertical support for the relocating core and structural debris that would accumulate over its upper surface. This would at least delay any major movement of debris into the lower plenum until local plate structural failures had opened the necessary pathways. Local plate structural failure (creep rupture) would be caused by the combined effects of an increasing weight of debris to be supported and a local loss of structural strength due to elevated plate temperature.

3.7.5.2 Accident Sequence Classification for Core Plate Considerations

The characteristics and rates of debris relocations from the active core region down onto the upper surface of the BWR core plate under severe accident conditions are accident-sequence dependent. The decay heat level at the onset of debris relocation, for example, depends upon the time at which the core becomes uncovered, which can vary from about 40 minutes (short-term station blackout) to more than 35 hours (loss of decay heat removal) after scram. The time required to boil away the water over the core is plant-specific, and depends strongly upon the decay heat level. Thus, it is necessary to determine how much this time is shortened and adjust procedures accordingly when power uprates are introduced.

From the standpoint of core plate response, BWR severe accident sequences can be broadly divided into two distinct sets: those sequences for which the core plate would be dry when debris relocation begins, and those for which the plate would be covered with water.

3.7.5.2.1 Dry Core Plate Accident Sequences

As explained in Section 3.7.2.3, almost all BWR severe accident sequences invoke procedural steps for manual actuation of the ADS when the core becomes partially uncovered. The attendant flashing of steam and high rate of flow through the open SRVs would cause rapid loss of reactor vessel water inventory and almost immediate core plate dryout. Heatup of the totally uncovered core would then lead to structural relocation of molten control blade (stainless steel/B_4C) and channel box (zirconium) materials. With the core plate dry, plate heatup and the potential for local temperatures conducive to creep rupture would begin immediately after the relocating metallic liquids reached the plate.

As local plate failures occur, overlying debris would fall into the lower plenum, contributing to the establishment of a debris bed there. (See Section 3.7.6.) On the other hand, for regions where the core plate remains intact, the debris bed would form in the region above the plate surface, with the oxides (UO_2 fuel pellets and ZrO_2 from the oxidized portion of the cladding) generally above the metals. This buildup of a debris bed above the plate surface is more likely for the case of the wet core plate, as will be described below.

Before leaving this discussion of the dry case, it is worthwhile to note that the core plate was not designed to constitute an impermeable partition. Leakage through the plate during reactor operation is intended to provide some cooling flow to the core interstitial region, to supplement the control rod drive hydraulic system flow. Because of the numerous original leakage pathways, it would not be necessary to have actual core plate failures in order for much of the metallic liquid reaching the plate surface to flow through, particularly in the vicinity of the incore instrument guide tube penetrations. (These hollow stainless steel tubes would be susceptible to melting long before the core plate itself.)

3.7.5.2.2 Wet Core Plate Accident Sequences

The intent of the BWR Owners Group Emergency Procedure Guidelines (EPGs)[3] regarding rapid reactor vessel depressurization when the core has been partially uncovered cannot be carried out in some BWR severe accident sequences because either control air or DC power, or both are not available at the time that it becomes necessary to open the SRVs. This is true, for example, for the long-term station blackout accident sequence, which is estimated by the NUREG-1150 study to constitute about 42% of the overall core damage risk (internal events) at Peach Bottom. For these accident sequences and in sequences involving failure to depressurize due to operator error, molten materials relocating downward from the uncovered upper portion of the core would freeze upon entering the two-phase (steam/water) region above the plate. The associated steam generation would cause a higher degree of metal-water reaction in the upper portion of the core and an accelerated core degradation rate in that vicinity.

For the wet case, much of the relocating metallic liquid would not reach the core plate, but rather would freeze at some elevation above the plate. Subsequent debris bed formation and melting above the plate surface would lead to a period within the accident sequence more like the Three Mile Island (PWR) experience. In other words, retention and buildup of a debris bed above the core plate is more likely for the wet case where core plate dryout, heatup, and structural failure are delayed because the plate is submerged in water during initial material melting and relocation.

Would the presence of water in the BWR core region during the initial stages of debris relocation from the upper core lead to formation of a bowl - like crucible containing oxidic melt as occurred at Three Mile Island? No experiment has directly addressed this possibility, but it seems very unlikely because of the open spaces between the fuel assemblies in the BWR core configuration and the different frothing heights among adjoining fuel bundles. The difference in frothing heights (due to different power densities) between adjacent bundles in the central region of a typical BWR core would exceed 0.30 m 1ft.); consequently, any metallic blockages would be discontinuous in the radial direction. (Contrary to first expectations, the freezing level for downward-flowing metallic liquids would be higher in the higher-power fuel assemblies.)

Core plate dryout would, of course, eventually occur for the case without vessel depressurization, when sufficient debris had been quenched to boil away all of the water initially above the plate. By this time, however, the core plate would be covered with a layer of previously quenched metallic material. Subsequent relocation of oxides onto the surface of this mostly metallic layer would induce heating of the plate and the accumulated overlying debris. The temperature increase of the plate would, however, be much slower than for the dry case, where molten metals interact directly with the plate surface.

3.7.5.3 Status of Experimental Findings

The current experimental foundation for understanding downward relocation of core debris under BWR severe accident conditions consists of the DF-4 experiment and the CORA BWR tests. Each of these experiments represents an uncovered length (control blade, channel box, and fuel rods) in the upper BWR core with steam flows (from below) typical of a wet core plate accident sequence. However, in each case, the steam flows were fed into the test apparatus from outside and there was no representation of water in the lower portion of the test section or of a core plate. These experiments all demonstrate rapid relocation of metals (control blade and channel box materials) to below the fueled region of the test assembly. These results confirm that relocation of metallic metals would occur first, so that debris initially reaching the core plate surface would consist entirely of metals, but provide no information concerning the core plate response.

Results of an experimental program[15] to establish the necessary information concerning the severe accident response of the BWR core plate were published in 1997. The XR2 test at Sandia National Laboratories examined the behavior of downward relocating molten metallic materials in the lower portion of a dry BWR core. The material composition and geometry of the XR2 test section was prototypic in both the axial and local radial directions. The simulated portion of the lower core included one-half meter of the fuel assembly, the nose piece, the core plate, the control blade and velocity limiter, and the fuel support structure with the coolant inlet nozzles. The imposed test conditions were calculated beforehand by the NRC-sponsored SCDAP/RELAP code, which itself includes models[16] based upon the results of the previous BWR DF-4 and CORA experiments. Specifically, these calculations provided the bases for the initial thermal state of the test assembly and for the timing and associated rates for the introduction of molten metals onto the test assembly from above.

To simulate downward draining control blade liquids reaching the lower core, control blade material was fed into the upper end of the test section over a period of 1000 seconds. Employing a specially designed wire guide and melter system, this feed material was delivered in a prescribed pattern over the test assembly control blade at a controlled, constant rate. Subsequently, molten Zircaloy was introduced as appropriate to represent the calculated degradation of the upper fuel rod cladding and channel box walls.

As reported in Reference 15, the fuel assemblies were severely degraded during the test. The channel box walls were destroyed by the aggressive eutectic-forming action of the molten control blade material, and the fuel rods were stripped of cladding. The x-ray imaging system showed the forming of temporary blockages and pools early during the test, but the liquids were able to break free so that large masses of molten materials would drain suddenly to the lower reaches of the test assembly. Thermocouple responses also indicated the effects of large-scale and sudden melt relocations to regions beneath the core plate.

Subsequent to the test, more that half of the feed material and material that melted in the upper portion of the test assembly was found below the core plate elevation. Some of this was located in the inlet nozzle, some had come to rest on the control blade velocity limiter, but most was found in the lower catch basin (which simulates the vessel lower plenum). Thus, the results of the XR-2 experiment demonstrate that the downward flowing metallic liquids do not freeze to form permanent blockages above the core plate, but rather continue to the plate surface and beyond to the lower plenum through existing pathways in the lower core structures and core plate.

It is important to recognize that the XR2 experiment was not intended to address core plate failure directly, but rather to provide the temperature response of the core plate for use in separate creep rupture calculations. The experiment did not represent the true radial extent of the core plate or the magnitude of the imposed load that would exist at the time that creep rupture would be expected to occur in the actual case. In actuality, the core plate is a perforated disk (with underbracing) that is 4.9 m (16 ft.) in diameter and is supported around the edge. Ultimately (after the period of initial metals pour addressed by the experiment), the central portion of this disk would be loaded by some 200,000 kg of fuel debris from the upper central core. This is when gross core plate failure would be expected to occur, and with the plate temperatures (in excess of [1700 °F] 1200 K) demonstrated in this experiment, such failure seems likely.

3.7.6 Severe Accident Events in the BWR Lower Plenum

As explained in the previous section, it is expected that metallic liquids relocating downward through a dry BWR core would to a large extent pass through existing core plate pathways into the reactor vessel lower plenum. Subsequently, following collapse of the central fuel pellet stacks and local failures of the core plate boundary, oxidic debris would enter and begin to accumulate within the lower plenum. While much of this entering oxide would be in the solid phase as UO_2 fuel pellet fragments, code predictions indicate that a significant portion would be in the liquid phase in the form of a UO_2-ZrO_2 eutectic mixture. At this point, it is important to consider the extent to which the falling debris would interact with the lower plenum water.

3.7.6.1 Debris Interactions With Lower Plenum Water

Fortunately, several experiments[17,18] have been conducted to examine the behavior of corium streams falling through water. In general, these have been special effects tests employing the actual materials of interest at the temperatures of interest. For example, Reference 17 reports the results of six tests in which a corium composition (by mass) of 60% UO_2, 16% ZrO_2, and 24% stainless steel (SS) at an initial temperature of (5072°F) 3073 K was poured at atmospheric pressure into water pools of depth about one meter. No steam explosions occurred, and the fraction of the pour quenched during the fall varied from 55-72% for subcooled pools to 33-45% for initially saturated pools. Very little (about 1%) of the steel dropped into the subcooled pools was oxidized during the fall, but for the saturated pools, up to 35% of the available steel was oxidized. The corium not quenched during the fall was subsequently quenched while resting at the bottom of the pool, where a debris bed of solidified particles was formed. There was no ablation of the pool floor.

When contemplating the corium pours that might be generated in hypothetical BWR accidents, it is necessary to consider the effects associated with the presence of zirconium metal as a constituent. In spite of the difficulties associated with the handling of liquid zirconium, the experiments conducted to date in the corium pour test series reported in Reference 18 have included one UO_2-ZrO_2-Zr pour, the results of which may be compared with the results of three similar tests carried out in the same apparatus with UO_2-ZrO_2 pours. No steam explosions occurred in these tests, which were carried out at 5.0 MPa (725 psia) with saturated water pools varying in depth from 1-2 meters. All of these pours were subject to significant breakup and quenching during the melt fall through the water.

For the Reference 18 tests with UO_2-ZrO_2, a portion (from 1/6 to 1/3) of the corium mass did not break up during the fall, but rather reached the pool floor while still molten (and subsequently quenched there). For the test with UO_2-ZrO_2-Zr, however, full oxidation of the zirconium and complete breakup of the melt occurred during the fall. Apparently, the energy release associated with the metal oxidation during the fall had the effect of promoting spreading (and quenching) of the accompanying oxide portion of the pour.

Future corium-water interaction tests are expected to provide additional insights. In the meantime, the available evidence from the special effects experiments[17,18] supports a contention that in the actual case, the relocating debris would quench in the lower plenum water.

For BWRs, the argument that the falling masses of hot debris would be quenched in the reactor vessel lower plenum is buttressed by the extent of the stainless steel structures located there and the large surrounding volume of water. For the Browns Ferry/Peach Bottom example, there are 185 control rod guide tubes of (11 in.) 28 cm outer diameter on a (12 in.) 30.5 cm pitch in the vessel lower plenum; within each unit cell, any free-falling debris must pass through a (50 in.^{2}) 316 cm^2 opening (see Figure 3.7-20) that is (12 ft.) 3.7 m in length (see Figure 3.7-21). Thus, any rapid passage of large, coherent, molten masses through the lower plenum water is precluded, and a large amount of energy would be transferred from the debris to the relatively cold steel with which it came into contact. These considerations, plus the initial presence of sufficient water in the lower plenum to completely quench more than one entire core, leads to the conclusion that the relocating debris would be quenched. The associated steam generation would be relieved via the SRVs, so that vessel integrity would not be threatened.

3.7.6.2 Events After Lower Plenum Dryout

In accordance with the BWR core material damage and relocation sequence described in Section 3.7.5, it is expected that the composition of the quenched debris bed that accumulates in the lower plenum would vary with height. Lowermost in the bed would be the debris first relocated into the lower plenum. This normally would comprise mostly metallic debris (control blades, channel boxes, candled clad and dissolved fuel) that had either passed through the intact core plate, had accumulated on the plate surface before local plate failure, or had subsequently relocated downward within the same local region before fuel pellet stack collapse.

Higher, within the middle region of the bed would be the collapsed fuel and ZrO_2 from the central region of the core. The initial leakages of metallic liquids through the core plate and the subsequent plate structural failures would cause temporary bursts of steaming as the relocating metallic debris was quenched; however, with the collapse of the central core fuel pellet stacks, a constant heat source (the decay heat associated with the pellets) would be introduced to the lower plenum reservoir, initiating a continuous boiloff of the remaining water.

After lower plenum dryout, the debris bed temperature would increase, causing thermal attack and failure of the control rod guide tube and instrument tube structures, which the debris would completely surround to a depth of about (10 ft.) 3 m. Since the control rod drive mechanism assemblies and the control rod guide tubes support the core, the remaining standing outer radial regions of the core would be expected to collapse into the lower plenum when these support columns fail. Thus, the uppermost portion of the completed lower plenum debris bed should primarily consist of the collapsed metallic and fuel materials from the relatively undamaged outer regions of the core. The stainless steel of the control rod guide tubes and mechanism assemblies, the instrument tubes, and other lower plenum structures would be subsumed into the surrounding debris as it became molten.

3.7.7 BWR Bottom Head Failure Modes

Given that the lower portion of the debris bed would be comprised almost entirely of metallic materials while UO_2 pellets constituted more than half of the central bed, then the central region would heat up much more rapidly after lower plenum dryout, and heat transfer within the bed would be toward the vessel wall. As the temperature of the bed increased, materials in the central region would begin to melt, migrate to cooler regions within the bed, freeze, and subsequently melt again. Eventually, temperatures near the wall would be sufficient to threaten its integrity. Failure of the vessel bottom head wall as a pressure boundary might occur as a result of penetration failures, or by creep-rupture failure of the wall itself. Both of these potential failure modes are discussed below.

3.7.7.1 Failure of the Bottom Head Penetrations

There are more than 200 bottom head penetrations in a BWR reactor vessel of the size employed at Browns Ferry or Peach Bottom, where there are 185 control rod drive (CRD) mechanism assembly penetrations, 55 instrument guide tube penetrations, and a (2 in.) 5.1 cm drain line penetration near the low point of the bottom head. The inner surface of the bottom head is clad with Inconel (thickness [0.125 in.] 0.32 cm), while the penetrations are stainless steel. Cross-sections of the CRD mechanism assembly and instrument tube penetrations and their weldments are illustrated in Figure 3.7-22. Each CRD mechanism assembly is held in place by an Inconel-to-stainless steel weld located at the upper end of the Inconel stub tube, as shown in Figure 3.7-23. (These are the welds that support the weight of the BWR core.) The location of one of the instrument guide tube welds at the inner surface of the bottom head wall is shown in Figure 3.7-24.

It is important to note that the CRD mechanism assembly welds are located about (4 in.) 10 cm above the vessel wall and thus would lie within the confines of the lower plenum debris bed. As a consequence, these welds would be expected to reach elevated temperatures before the instrument tube welds at the vessel wall. Failure, if it occurred, would be by creep rupture, which would be promoted at much lower temperatures if the reactor vessel were pressurized.

Although bottom head pressure boundary failures should occur first at the upper stub tube welds, this type of penetration failure is less important from the standpoint of potential for debris release from the reactor vessel than are instrument tube failures. This is true because BWRs are required to have an auxiliary support structure beneath the vessel bottom head that would limit the downward movement of any control rod mechanism assembly to about 3 cm (1 in.) in the event of failure of its stub tube weld. (This requirement does not derive from severe accident considerations, but rather from a need to guard against the sudden expulsion of a control blade from the core during critical operation at very low power.) Since the thickness of the vessel bottom head in the region of the penetrations is 21.4 cm (8 7/16 in.), this limited downward movement could not open a significant pathway through the vessel wall even if the CRD mechanism assembly were melted within the debris bed. This is not true for the instrument guide tubes, for which there is no provision to limit their downward movement.

For an unmitigated accident sequence, temperatures at the inner surface of the bottom head wall would eventually become sufficiently high to cause failures of the welds that hold the instrument tubes in place. However, it is probable that a different mode of failure for the instrument guide tubes would occur first. As illustrated in Figure 3.7-25, this potential early failure mode for the instrument guide tubes involves melting of the portions of these guide tubes within the central region of the debris bed. Then, when the sequence of melting, downward movement, and freezing processes for the metals had progressed to the point that molten debris liquids were standing in the central region of the bed, these liquids could spill into the failed instrument tubes and pour through the vessel wall. In order to complete the pathway to the containment, however, it is necessary that the entry of molten debris liquids into the interior of these tubes induce tube wall failures *external* to the bottom head wall.

Would movement of molten debris liquids through an instrument tube result in tube failure outside the vessel wall? This question has been extensively studied.[7,19] An important measure of the vulnerability of the tube wall is the ratio of the volume within the tube available for occupation by melt to the volume of the adjacent wall. For BWR instrument guide tubes, this ratio is 1.40, whereas for PWR guide tubes, this ratio varies from 0.06 to 0.52. Thus, BWRs are more susceptible to debris release by penetration failures than are PWRs. Analyses[19] indicate, however, that the BWR instrument guide tube vulnerability is limited to cases where the vessel pressure is greater than (290 psia) 2 MPa, and molten oxidic (ceramic) melt enters the tube. (Metallic liquids entering the tube and passing through the vessel wall would be expected to freeze while the tube wall remains intact.) If the reactor vessel is depressurized, then calculations indicate that the only possible penetration failure involves the entrance of ceramic liquids into the vessel drain.

Given the presence of internal debris liquids, then the BWR vessel drain offers the highest potential for the opening of an escape route through the vessel wall, as evidenced by its melt volume-to-wall volume ratio of 1.57, which is the largest value of this ratio for any type of penetration. However, it is important to recognize that the pathway by which molten debris would enter the vessel drain is different than that for the instrument guide tubes. The vessel drain is located at the bottom of the lower plenum, offset slightly (about [6 in.] 15 cm) from the point of vessel zero. Therefore, metallic particles quenched while falling through the lower plenum water during the initial stages of debris movement through the core plate would collect in the drain. After lower plenum dryout, metallic liquids forming within the debris bed and moving downward would enter the remaining voids within the vessel drain. On the other hand, molten material would flow laterally within the bed to enter the failure location of an instrument tube only after the voids in the lower portions of the bed had been filled with liquid, to the level of the failure location. The upshot is that the vessel drain would probably be filled earlier by a lower-melting-temperature metallic mixture; the instrument guide tubes would probably be filled later by a higher-melting-temperature oxidic mixture.

To summarize, BWR penetration failures by internal flow are unlikely, but the possibility cannot be excluded. If not previously filled by frozen metals, the instrument tubes would be expected to fail ex-vessel if attacked from within by ceramic melt at a pressure of at least 2 MPa (290 psia). While the presence of ceramic melt within the vessel drain could cause failure of the wall with zero pressure differential, it is very probable that the drain would be filled by metallic debris long before molten ceramics became available in the vicinity. Since there are 55 instrument guide tubes and just one drain, it seems that between the two possibilities, opening of an instrument tube pathway is more probable.

The experimental evidence with respect to bottom head failure as discussed in Section 3.5.7 indicates that the presence of penetrations does accelerate bottom head failure, not by internal sneak pathways through the vessel wall, but rather external to the penetrations. These failures are characterized by cracking of and leakage through the circumferential welds that originally held the penetrations within the surrounding vessel wall. Once initiated, this leakage rapidly increases as the diameter of the vessel wall through-holes provided for the penetrations increases by as much as a factor of two.

If penetration failures did occur, then a leak path from the vessel to the containment atmosphere would be created. Subsequently, the vessel gaseous content would blow down if the reactor vessel were at pressure or, if the vessel were depressurized, would slowly leak out as the gas temperature in the vessel increased and the water in the vessel downcomer region boiled away. The leak path for the steam generated from the water surrounding the jet pumps would be up through the downcomer region, down through the core region, and out through the debris bed. Steam passing through the debris would react with any unoxidized metals in its path, greatly augmenting the local bed heatup rate and promoting local melting. The liquids would flow from the vessel as they are created by melting within the bed. Stated another way, the rate of release of debris liquids from the vessel would be controlled by the rate of debris melting.

3.7.7.2 Gross Bottom Head Failure

If penetration failures do occur, the downward flow of molten liquids around the instrument tube and control rod guide tube locations would induce ablation of the surrounding solid debris (in lower portions of the bed) and the vessel wall. Wall ablation can significantly increase the size of the effective flow pathways through the wall; however, as discussed in the previous Section, the rate at which liquids are released from the vessel after the initial penetration failures and blowdown would be controlled by the rate at which the solid debris is melting. Eventually, wall ablation would lead to gross failure of the wall, such that all remaining debris (comprised primarily of solids) would be released from the vessel.

For cases in which penetration failures did not occur, heat transfer from the central portion of the bed after lower plenum dryout would eventually increase the temperature of the bottom head wall to the point of failure by creep rupture. (This has been observed experimentally for vessel bottom heads without penetrations as discussed in Section 3.5.7.) However, about 95% of the vessel wall stress under normal operating conditions is due to the internal vessel pressure, and, as described in Section 3.7.2, emergency procedures direct the control room operators to manually depressurize the reactor vessel at an early phase of any severe accident sequence (when the core is partially uncovered). The wall stress after lower plenum dryout with the reactor vessel depressurized and taking into account the weights of both the debris resting on the bottom head and the bottom head itself would be less than 3% of the normal operating value.[7] At this low stress level, failure by creep rupture would occur only at wall temperatures approaching the melting temperature of the carbon steel. Calculations based on short term station blackout without penetration failures typically predict that wall temperatures of this magnitude would be reached about six hours after lower plenum dryout.

3.7.7.3 Effectiveness of External Water Cooling

It is important to note that containment flooding to above the level of the core is currently incorporated within the BWR Owners Group EPGs[3] as an alternative method for providing a water source into the reactor vessel in the event of LOCA (the water would flow into the vessel from the containment through the break). Here we will discuss whether or not containment flooding, if successfully carried out, might be effective in preventing the release of molten materials from the reactor vessel for the risk-dominant non-LOCA accident sequences such as station blackout. The practical difficulties associated with attempting to inject large amounts of water into containment under severe accident conditions will be addressed in Section 4.7. Heat transfer from the instrument guide tube and vessel drain outer surfaces would be greatly enhanced by the presence of water because the heat transfer mode would be shifted from natural convection of air to nucleate or film boiling of water. HEATING code calculations[7] have demonstrated that the effectiveness of the water cooling is such that the submerged instrument guide tubes or vessel drain would be expected to survive filling with any possible category of molten debris, even if the vessel remains pressurized.

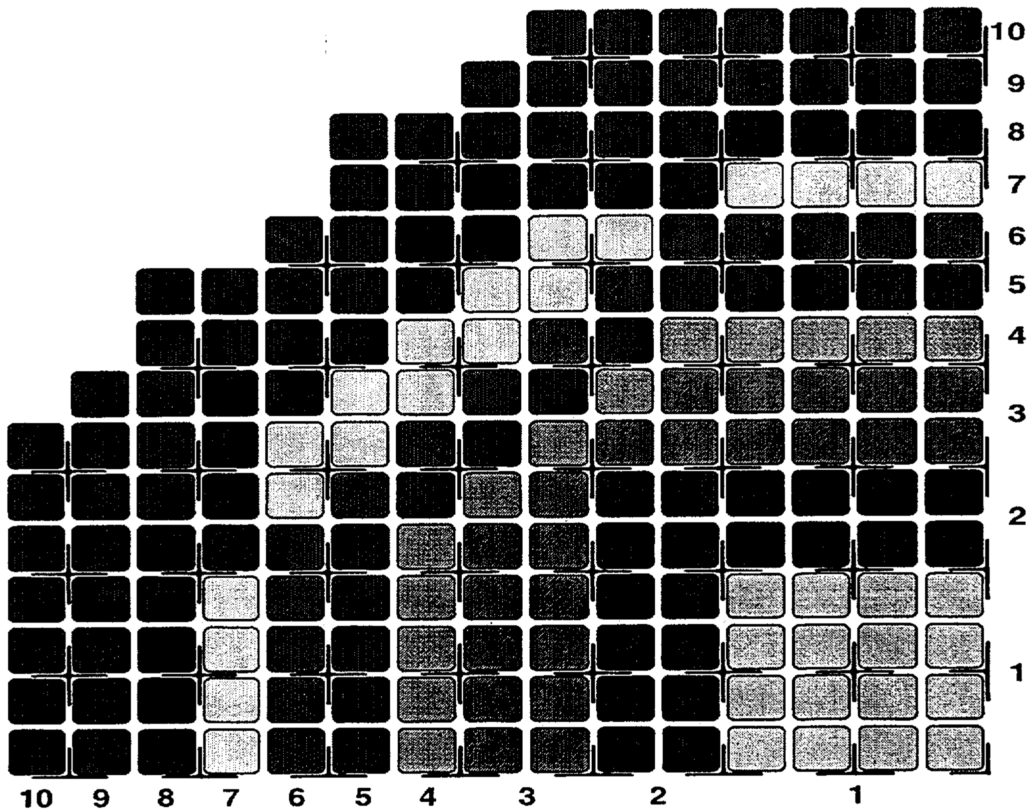
If the containment were flooded with water, a portion of the drywell atmosphere would be trapped within the reactor vessel support skirt, as illustrated in Figure 3.7-26. The fraction of the bottom head surface area beneath the skirt that would be submerged in water could be increased by providing a vent pathway such as by drilling several small holes in the vessel skirt, just below the vessel attachment weld. However, from the standpoint of regulatory requirements, the NUREG-1150 core melt frequency estimates, cost-benefit analysis, and the desire to minimize radiation exposure to personnel, this is clearly not a practical proposal for existing BWR facilities.

Analyses[7] have shown that the existence of a trapped gas pocket beneath the vessel skirt attachment would ultimately prove fatal to the integrity of the bottom head wall. Figure 3.7-27 illustrates the expected condition of the debris at the time of wall failure. The central portion of the bed is a liquid slurry while a crust (thicker at the bottom) adheres to the wall. Calculations indicate that the presence of water as shown in Figures 3.7-26 and -27 would serve to delay bottom head creep rupture from about 10 hours after scram if the bottom head is dry to about 13 hours after scram.

It is instructive to briefly consider a case in which the bottom head is completely submerged as shown in Figure 3.7-28. Here none of the local bottom head temperatures would ever become high enough to threaten failure by creep rupture. However, upward radiative heat transfer within the vessel from the debris surface would eventually melt all of the upper internal structures; the ensuing stainless steel liquid would be added to the central debris pool. After exhaustion of the stainless steel, the only remaining internal heat sink above the debris surface would be the carbon steel of the vessel wall. All portions of the wall cooled by water on their outer surfaces would remain intact, but unless the water height within the drywell extended well above the surface of the debris pool, upper portions of the vessel wall with exteriors exposed to the drywell atmosphere would ultimately reach failure temperatures. Figure 3.7-28 indicates the minimum water height required to prevent melting of the wall inner surface as determined by an actual calculation[7] based upon the Peach Bottom or Browns Ferry configuration.

Table 3.7-1 Vessel depressurization at one-third core height postpones the predicted core degradation events for short term blackout

Event	Time (min)	
Swollen Water Level Falls Below Top of Core	40.7	40.7
ADS Actuated	–	75.0
Core Plate Dryout	–	75.6
Begin Relocation of Core Debris	87.4	110.2
Core Plate Dryout	102.5	–
First Local Core Plate Failure	116.7	111.6



RADIAL ZONE	NO. OF BUNDLES IN 1/4 CORE	POWER RANGE	MEAN PEAKING FACTOR	VOLUME FRACTION
1	16	0.97 – 1.27	1.115	0.08377
2	20	1.06 – 1.32	1.160	0.10471
3	22	1.10 – 1.34	1.233	0.11518
4	13	1.23 – 1.36	1.309	0.06806
5	14	1.10 – 1.33	1.194	0.07330
6	15	1.15 – 1.37	1.258	0.07853
7	19	1.10 – 1.23	1.155	0.09948
8	24	0.92 – 1.13	1.000	0.12565
9	23	0.55 – 0.85	0.670	0.12042
10	25	0.23 – 0.47	0.354	0.13089
	191			

Figure 3.7-1 Definition of radial zones for Browns Ferry unit 1 cycle 6 core

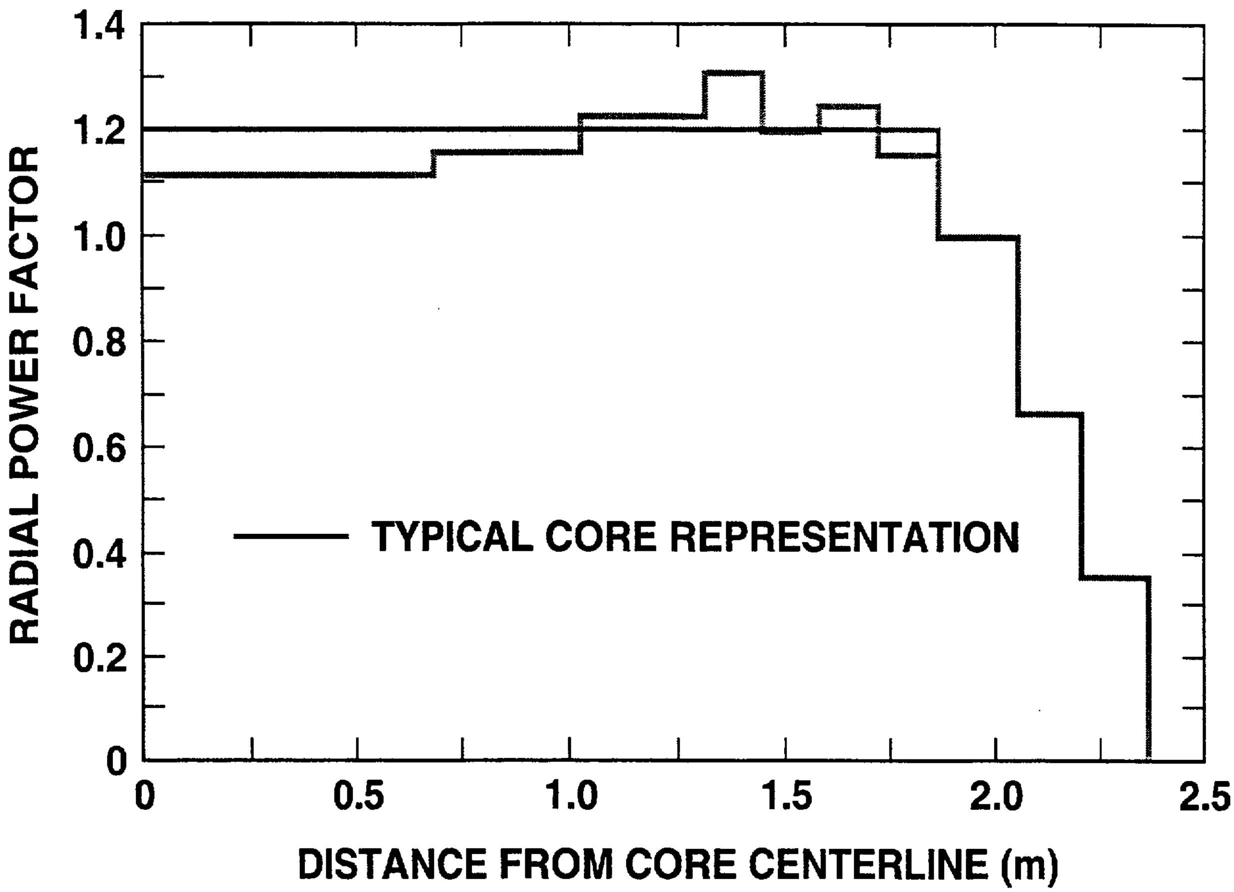


Figure 3.7-2 The progression of severe structural damage in the outer core would significantly lag events in the central core

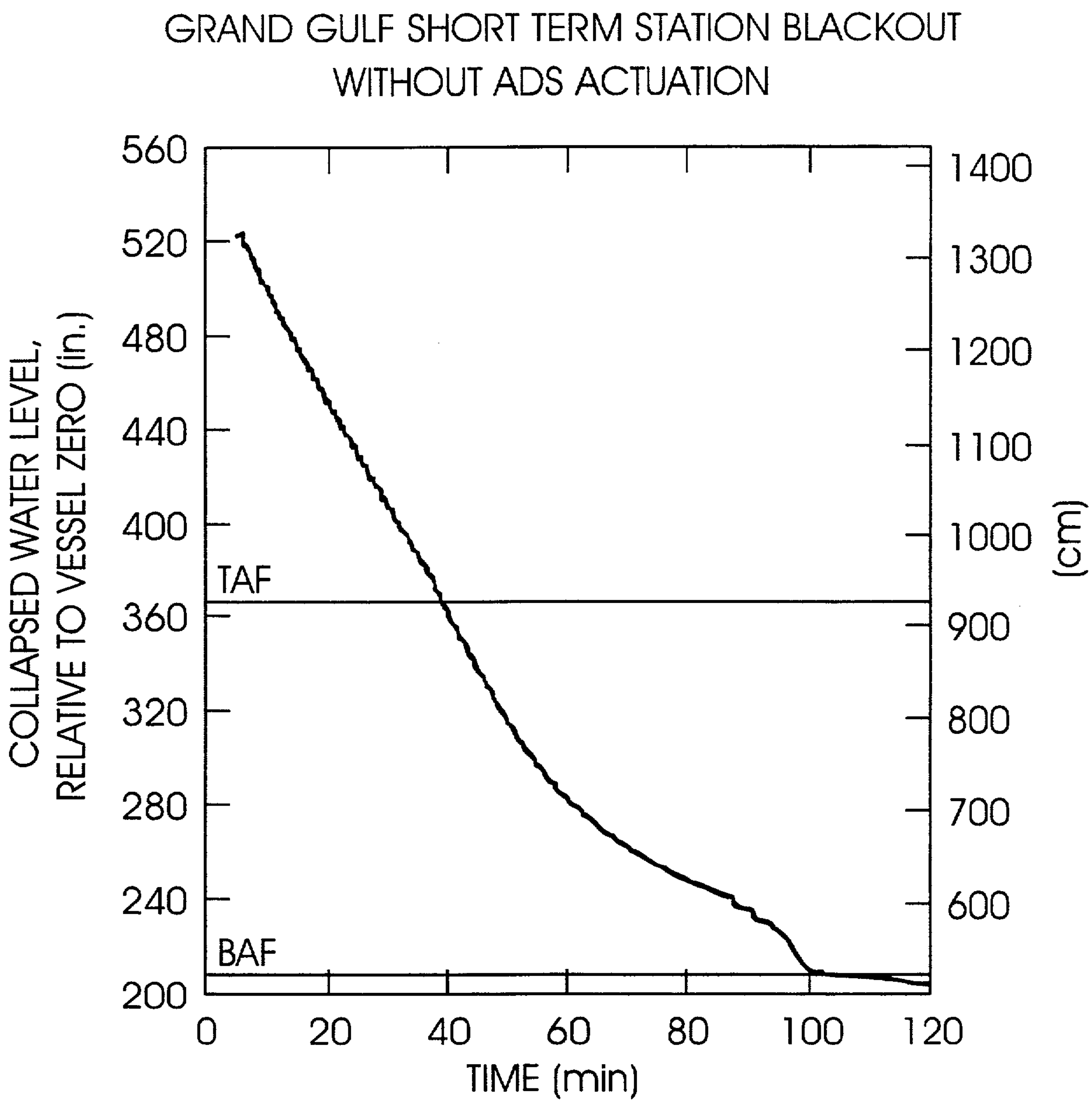


Figure 3.7-3 **If the reactor vessel remains pressurized, relocating core debris falls into water above the core plate**

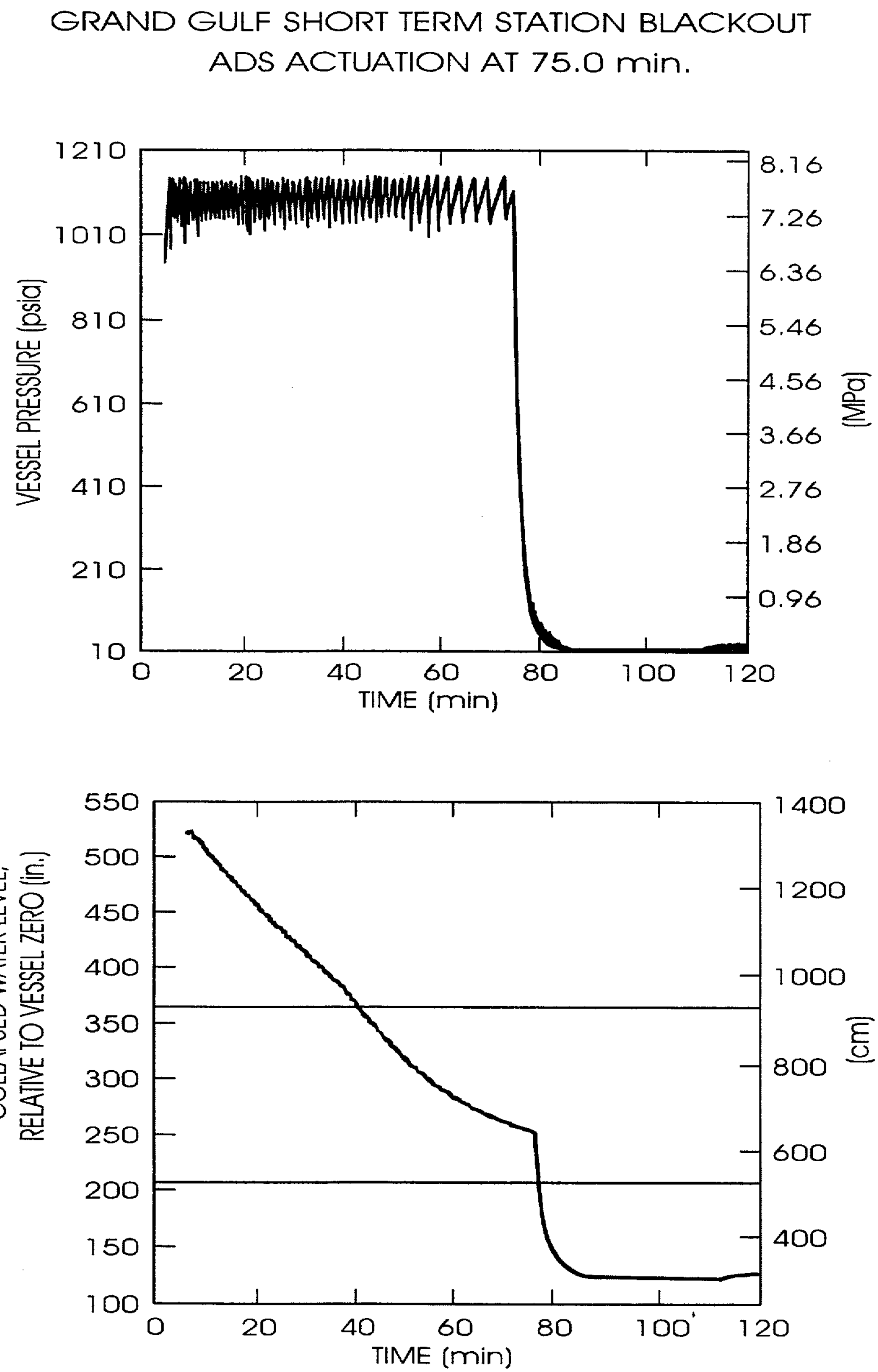


Figure 3.7-4 Effects of manual actuation of ADS at about one-third core height

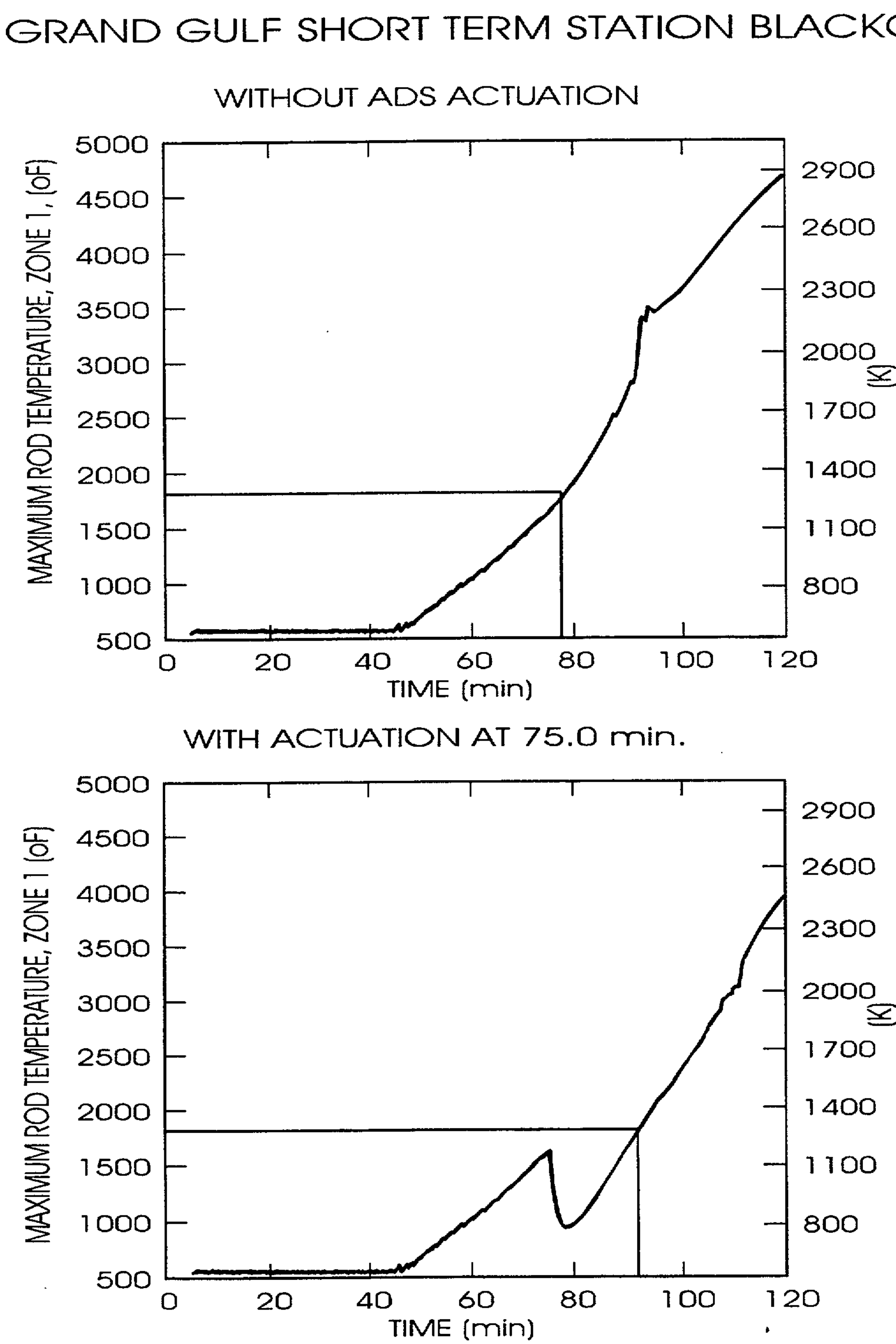


Figure 3.7-5 **Vessel depressurization at one-third core height provides steam cooling that temporarily reverses core heatup**

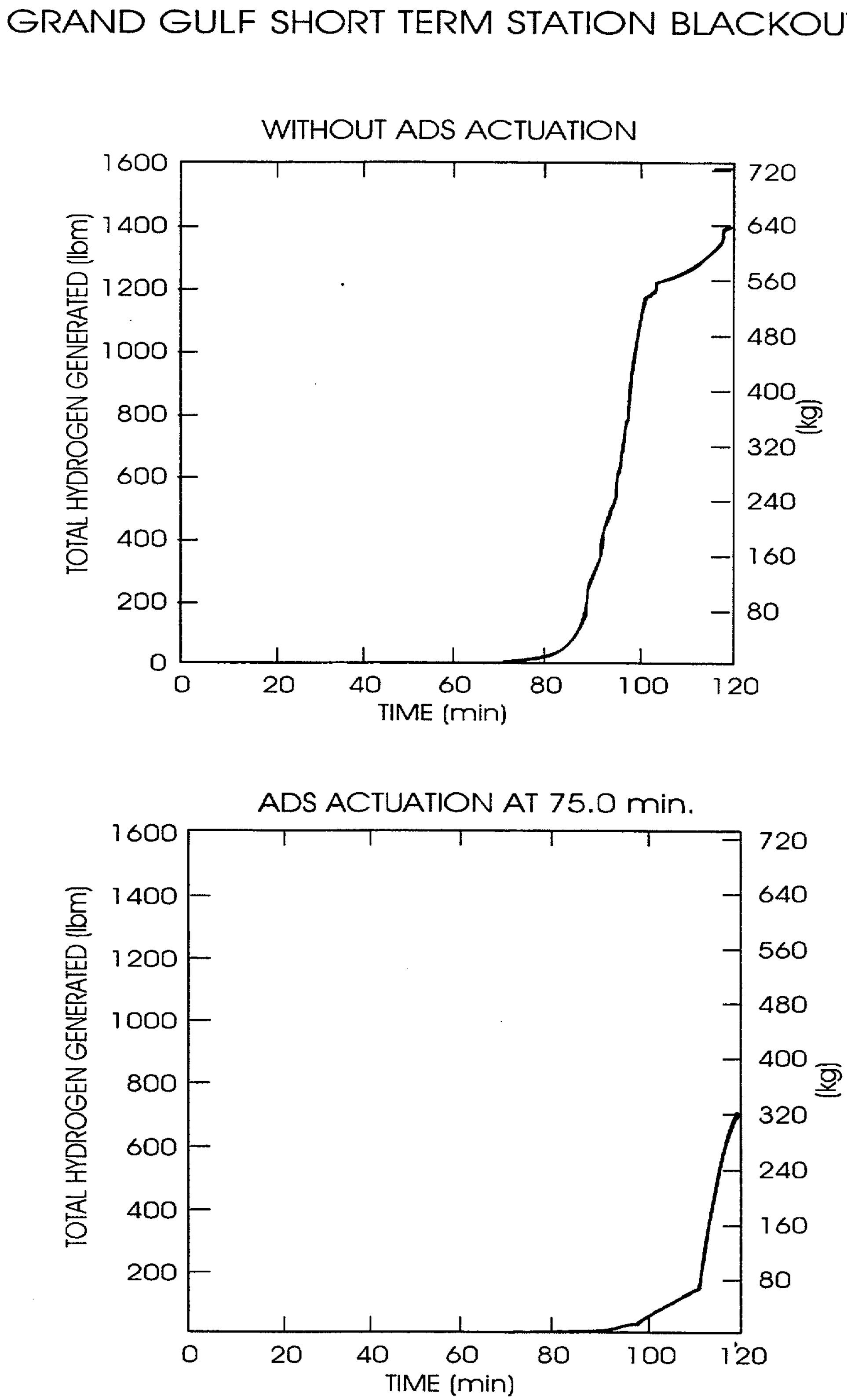


Figure 3.7-6 **Vessel depressurization at one-third core height delays hydrogen release**

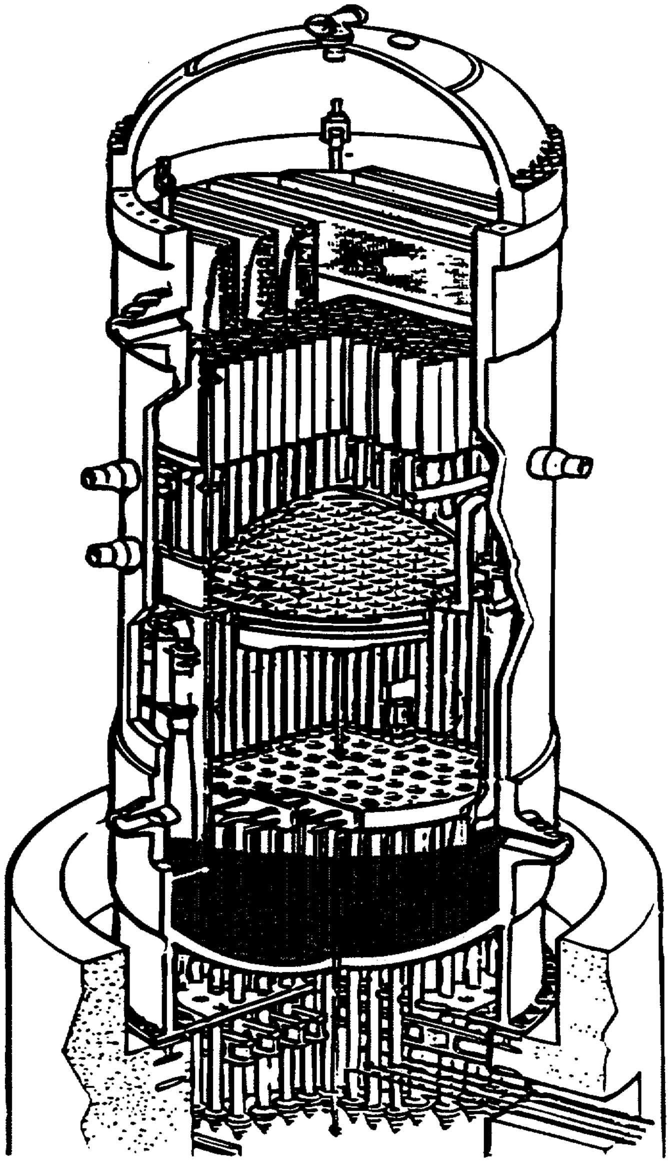


Figure 3.7-7 Regional above core plate would be dry during structural degradation

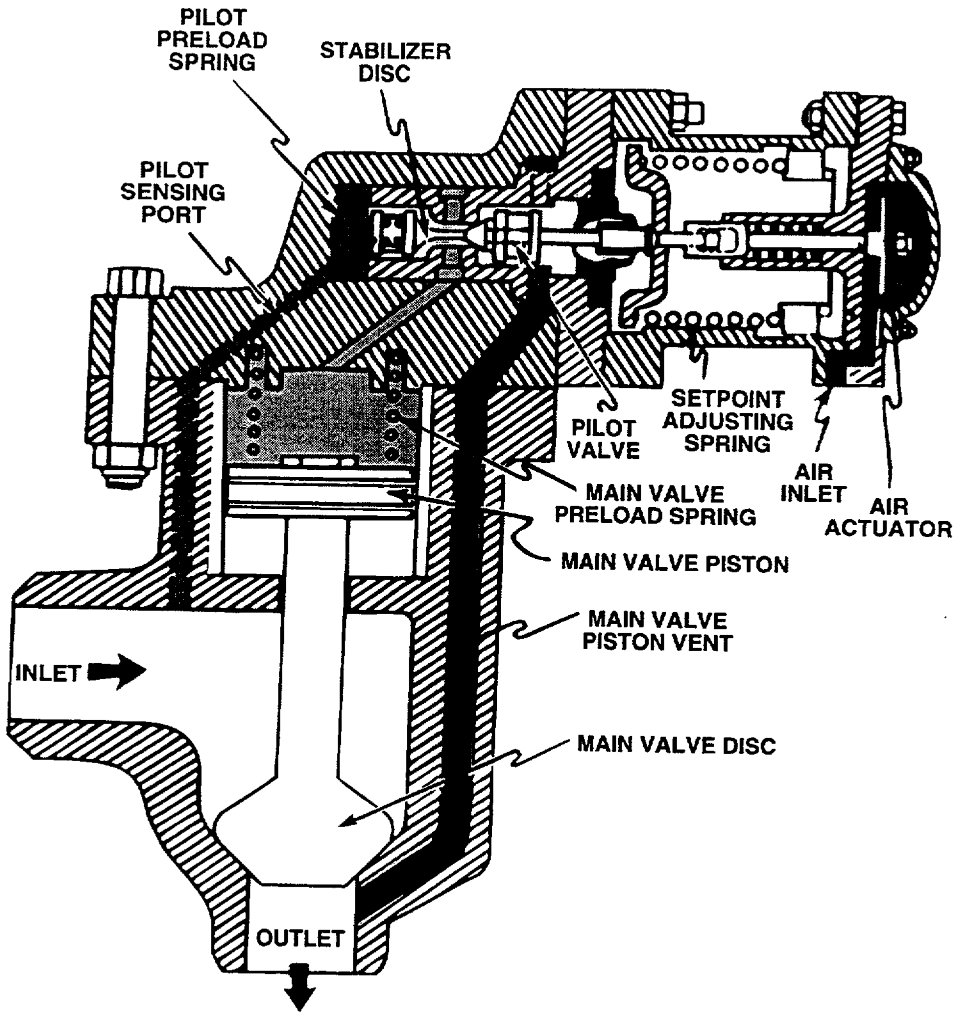


Figure 3.7-8 For the two-stage target rock SRV, control air and system pressure act in concert to position the pilot valve.

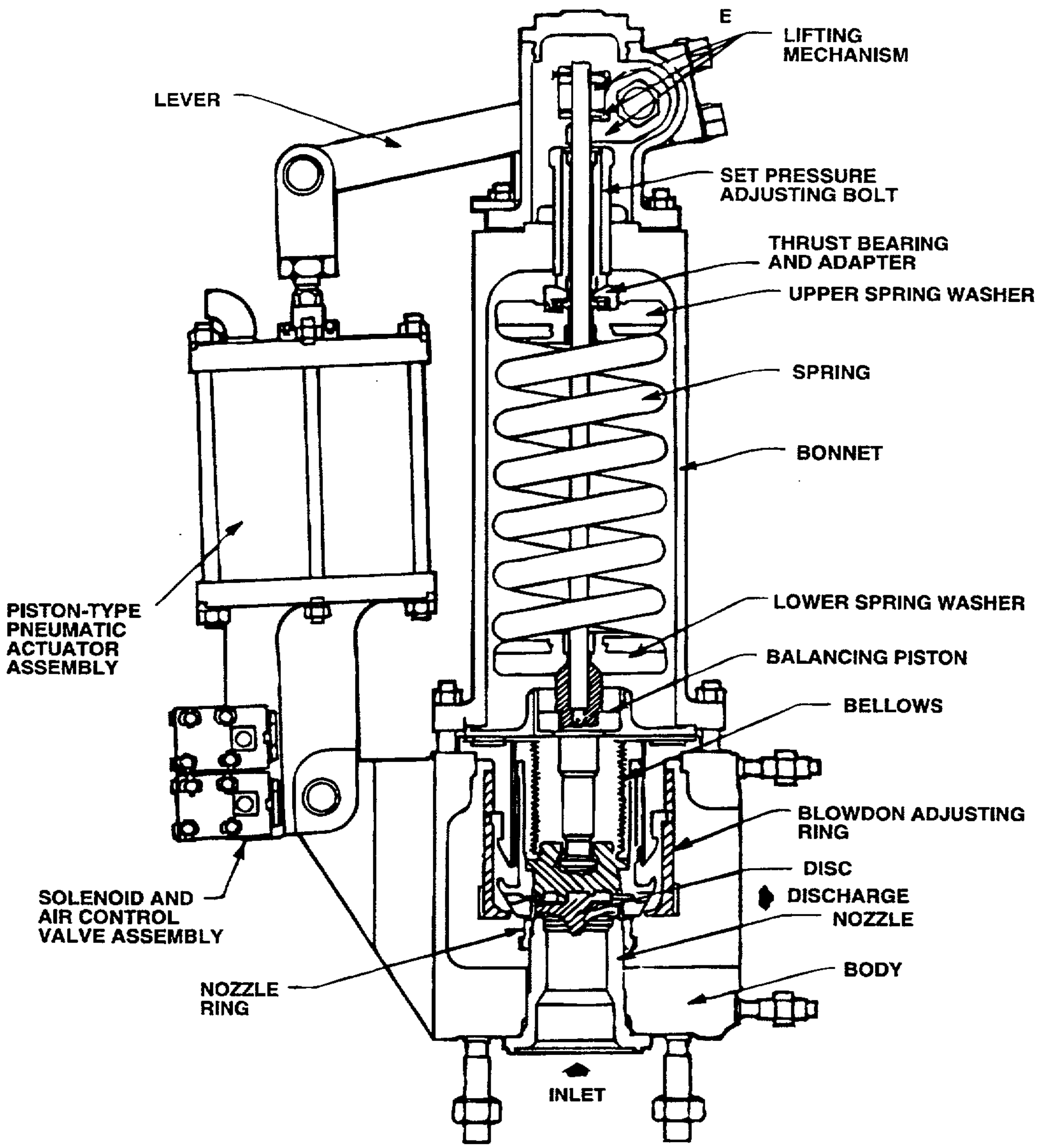


Figure 3.7-9 For the Crosby SRV, control air opens the main valve

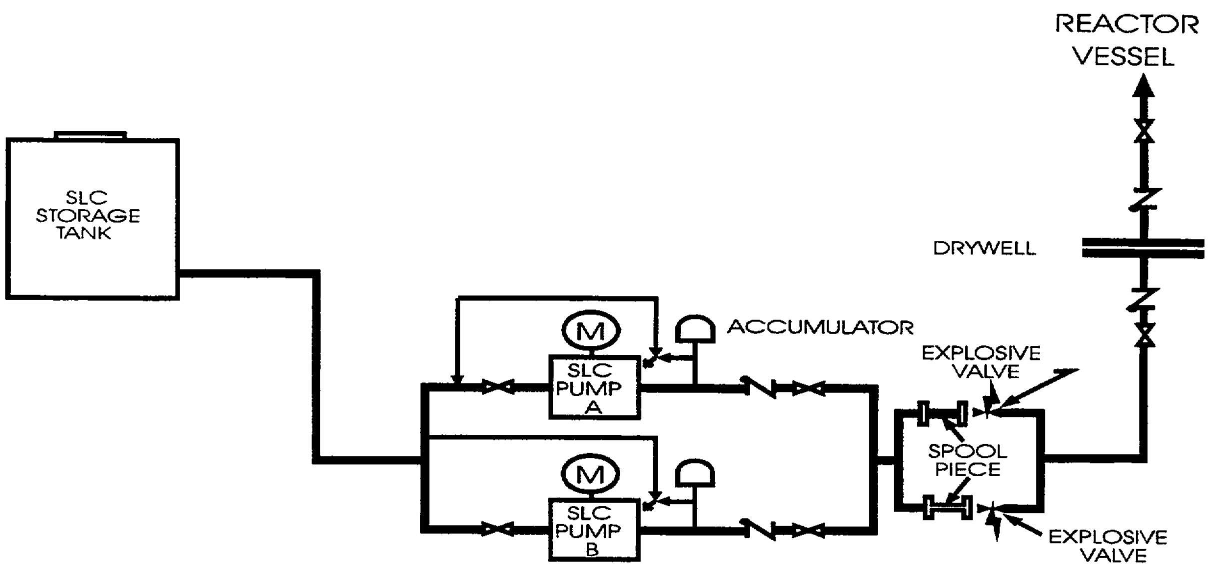


Figure 3.7-10 Abbreviated schematic of a typical BWR SLCS

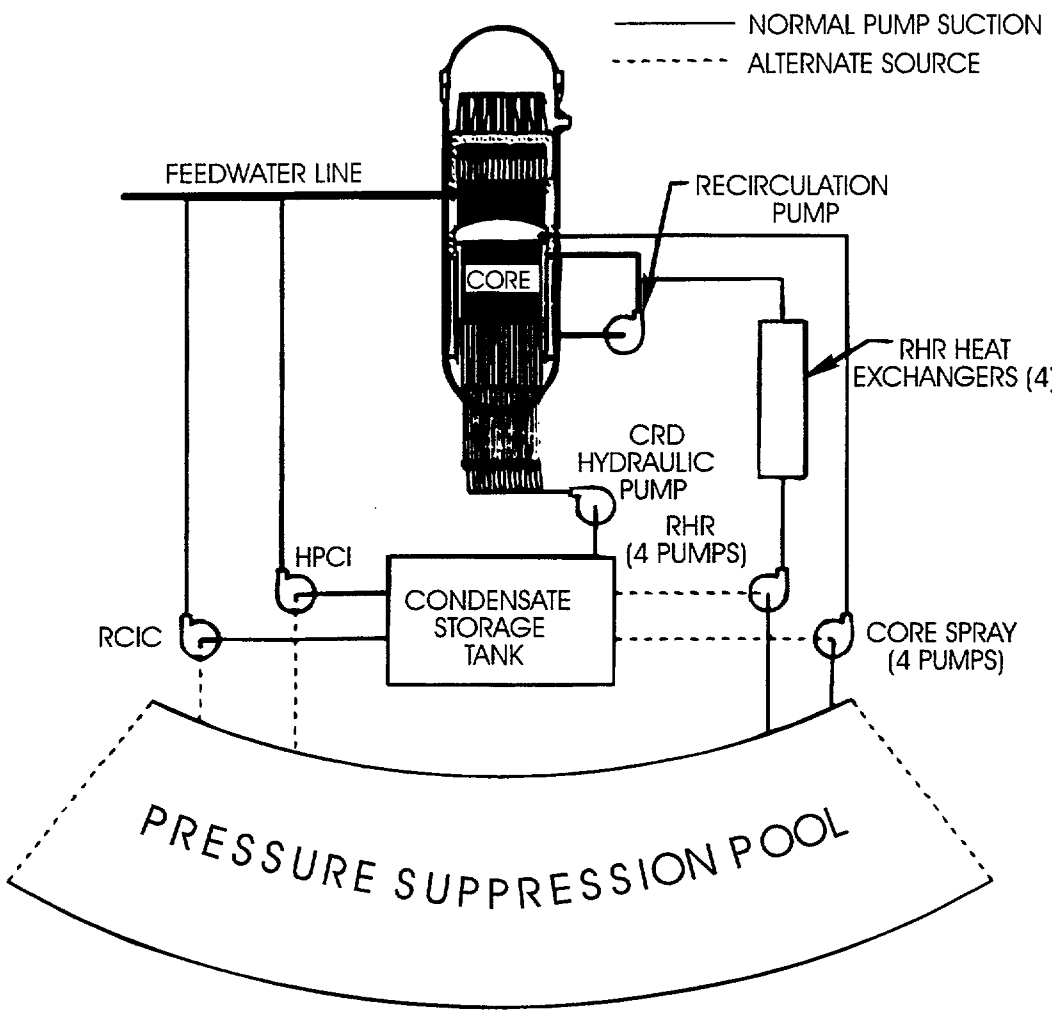


Figure 3.7-11 The condensate storage tank is an important source of water during accident sequences other than LBLOCA

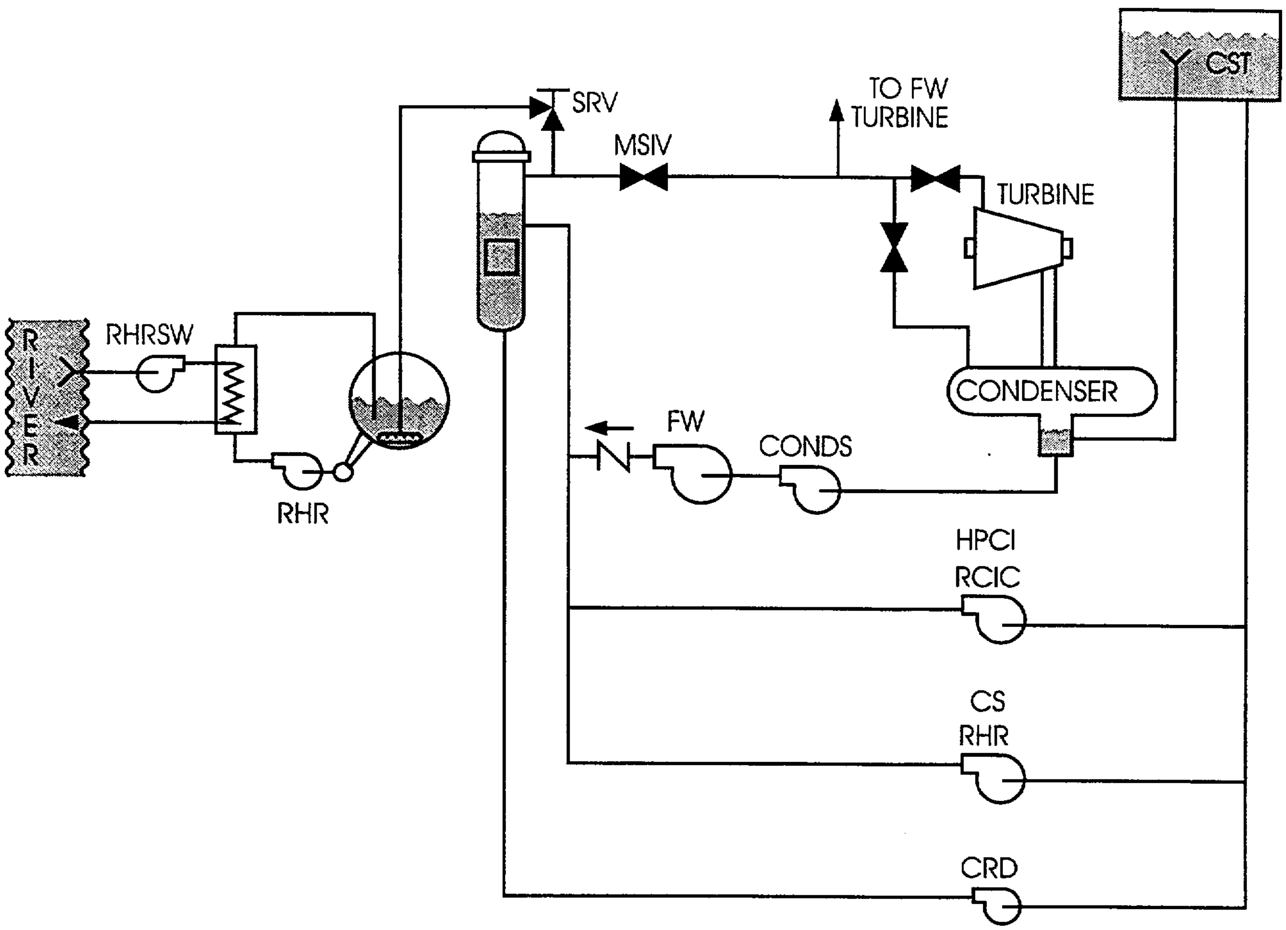


Figure 3.7-12 **The condensate storage tank can be drained to the main condenser hotwells, leaving sufficient water volume for the reactor vessel injection**

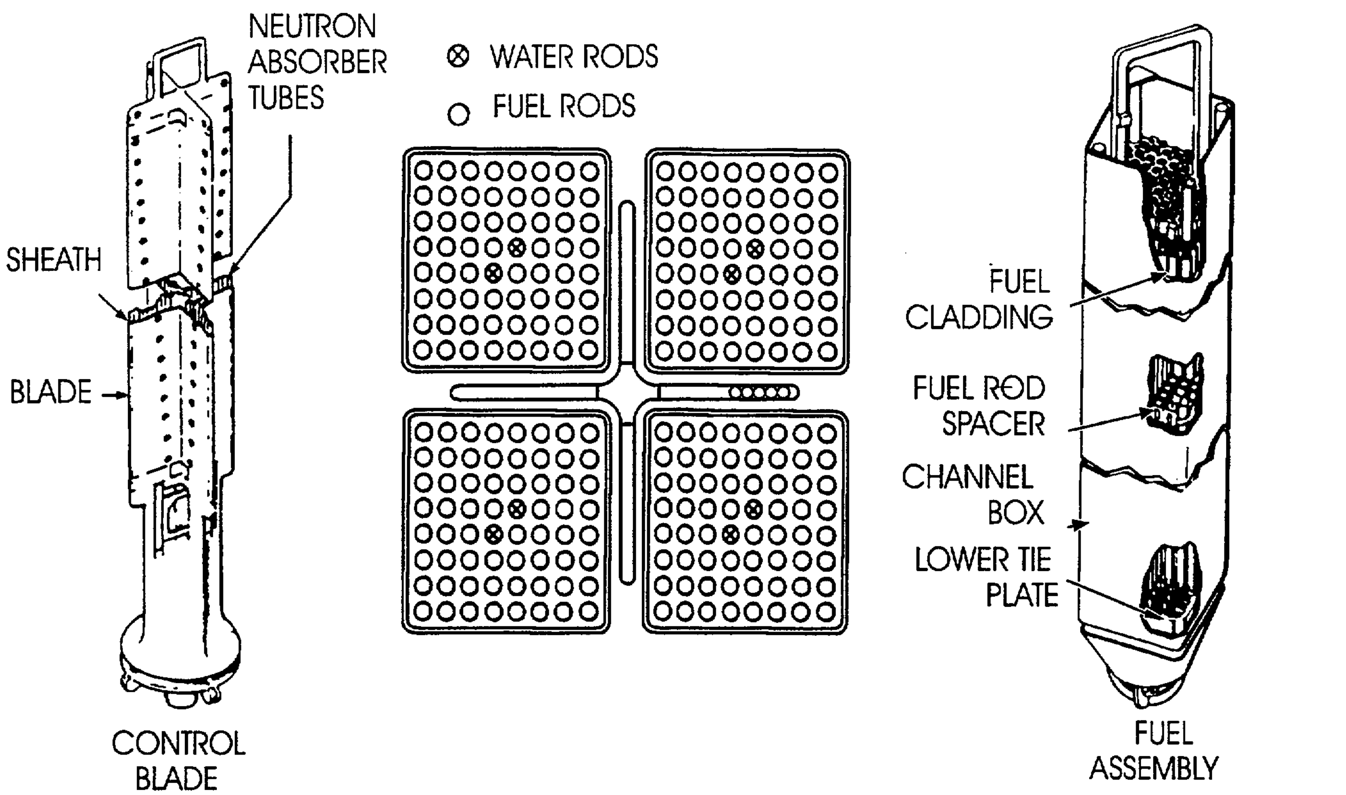


Figure 3.7-13 **The BWR control blades are inserted into the interstitial region between fuel assemblies in the core.**

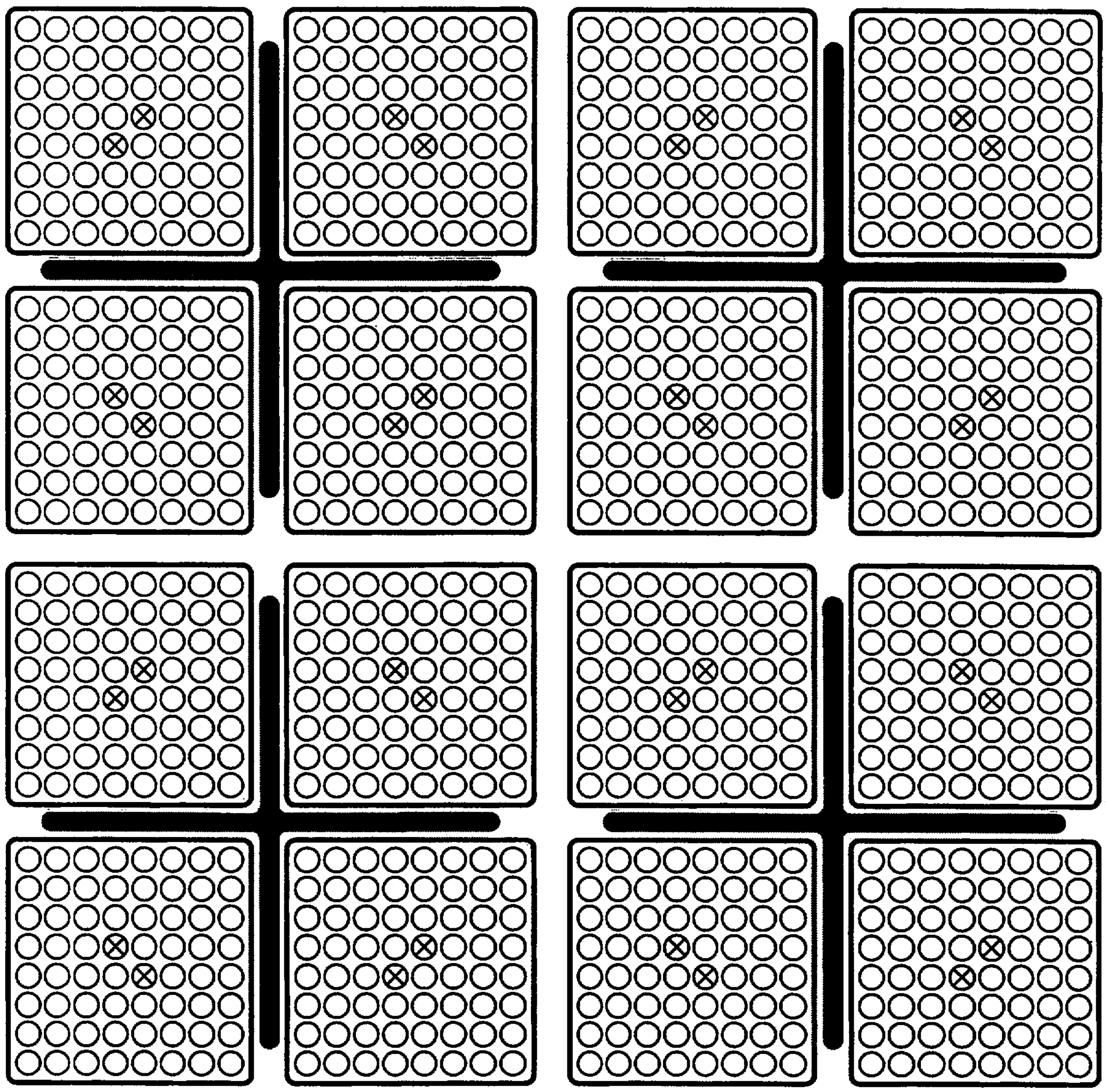


Figure 3.7-14 One-half of the channel box outer surfaces do not see an intervening control blade

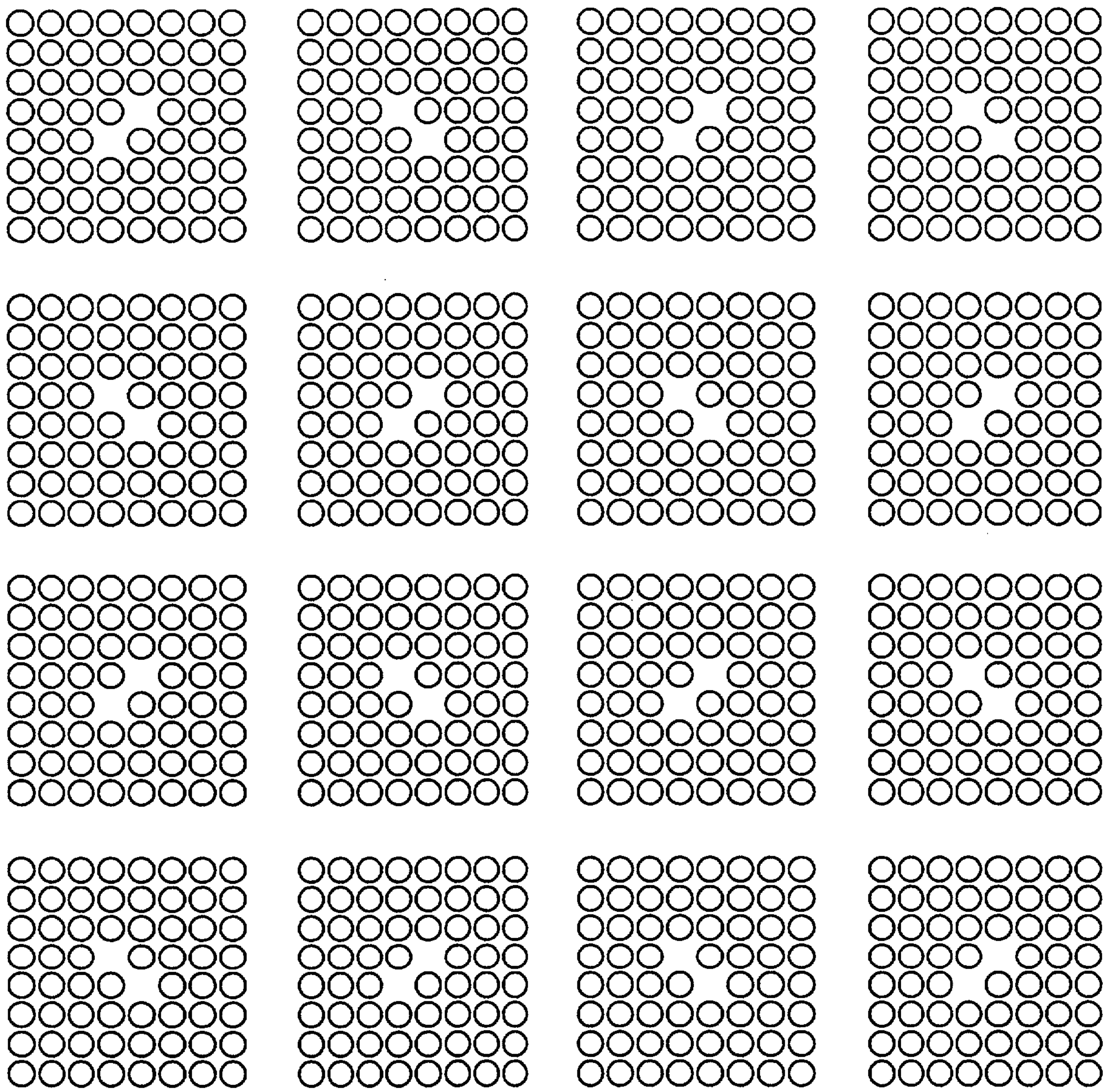


Figure 3.7-15 Relocation of control blades and channel box walls leaves on UO_2 pellets encased in thin $Zr0_2$ sheaths

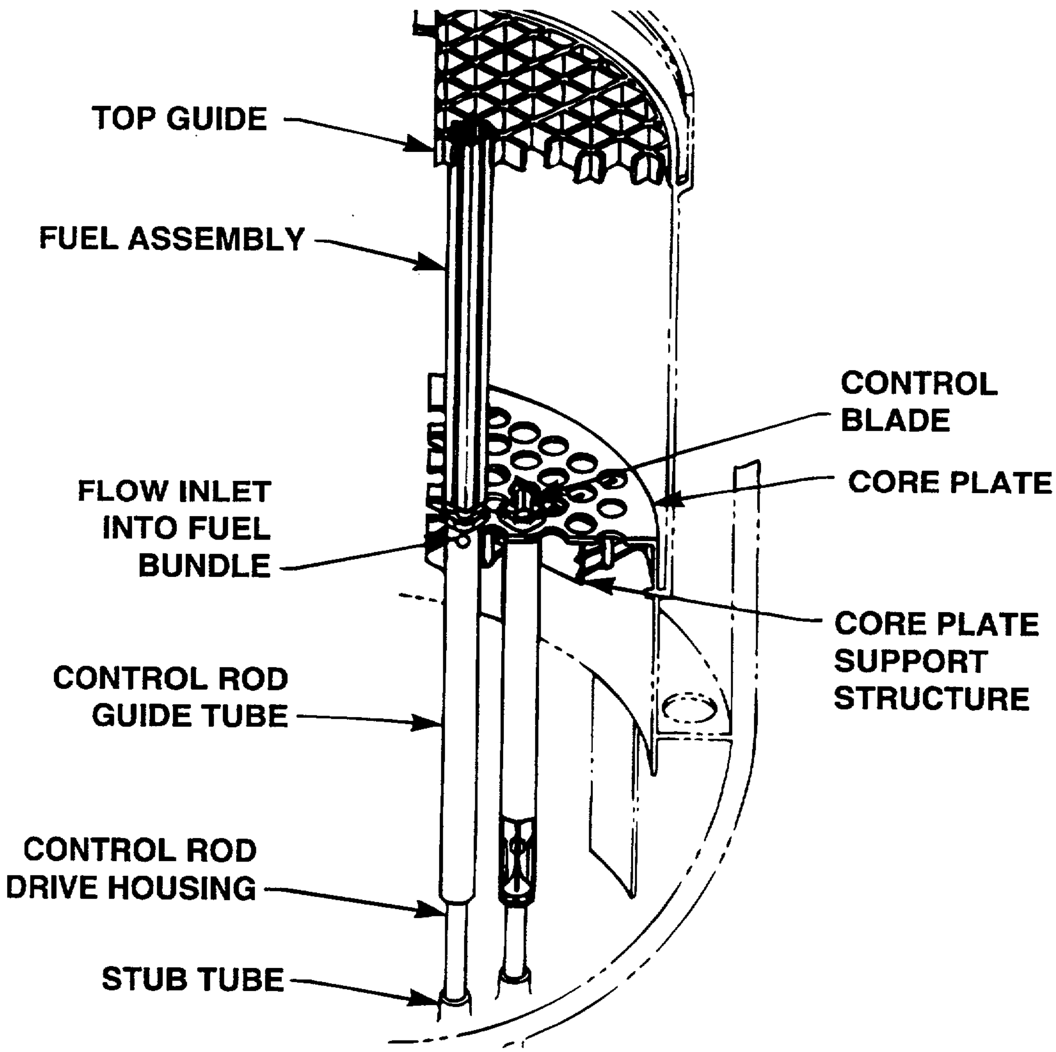


Figure 3.7-16 **The BWR core plate separates the core region from the reactor vessel lower plenum but does not support the core**

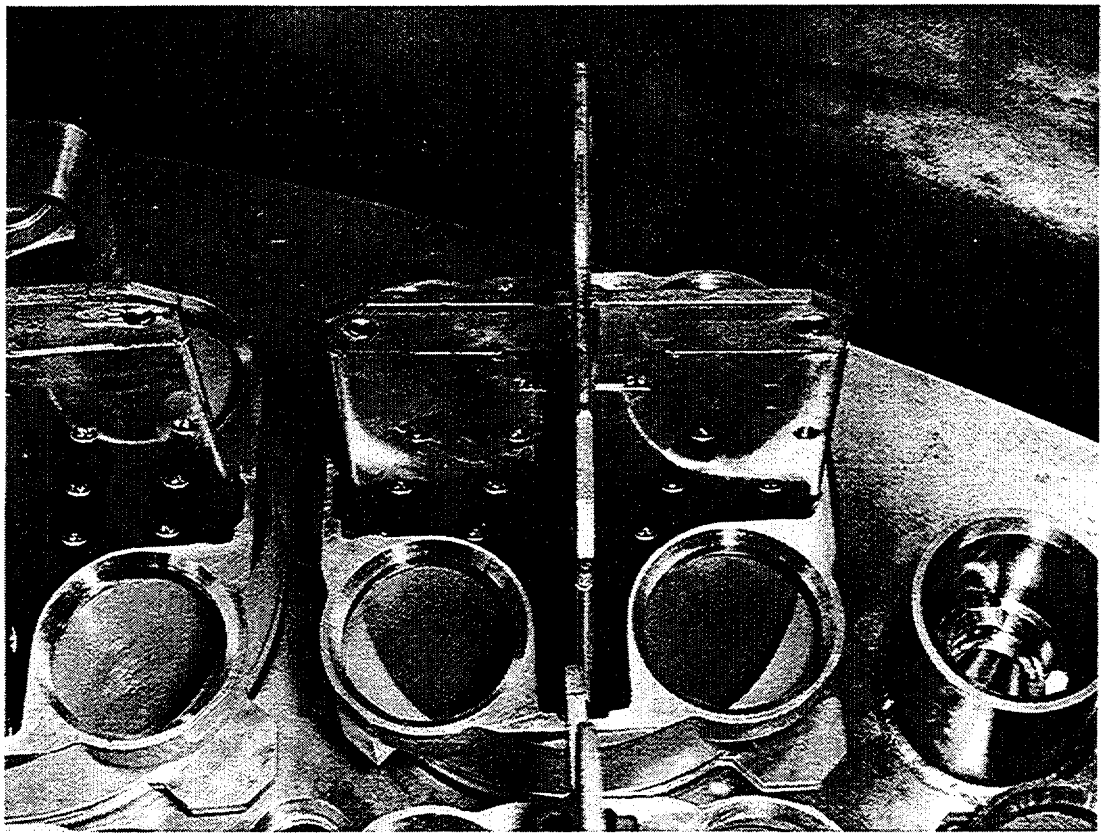


Figure 3.7-17 Control blade tip emerging from fuel support structure near core plate edge at Peach Bottom

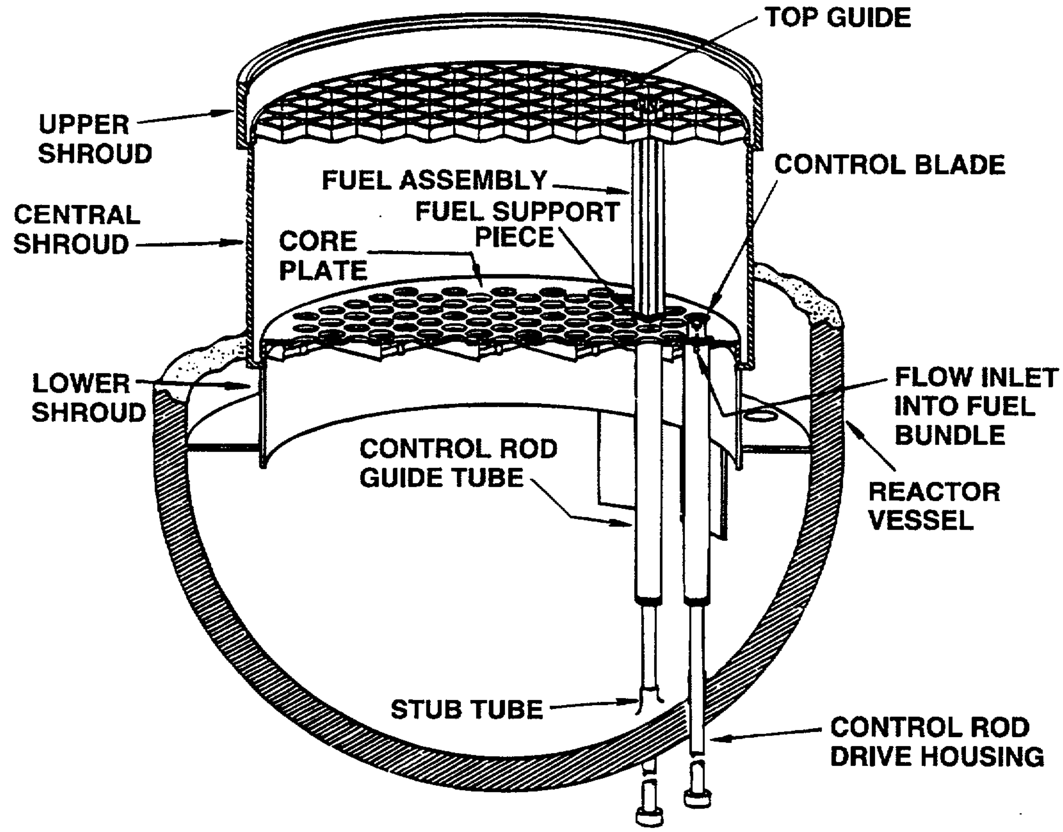


Figure 3.7-18 Material relocating from the core region would enter the reactor vessel lower plenum

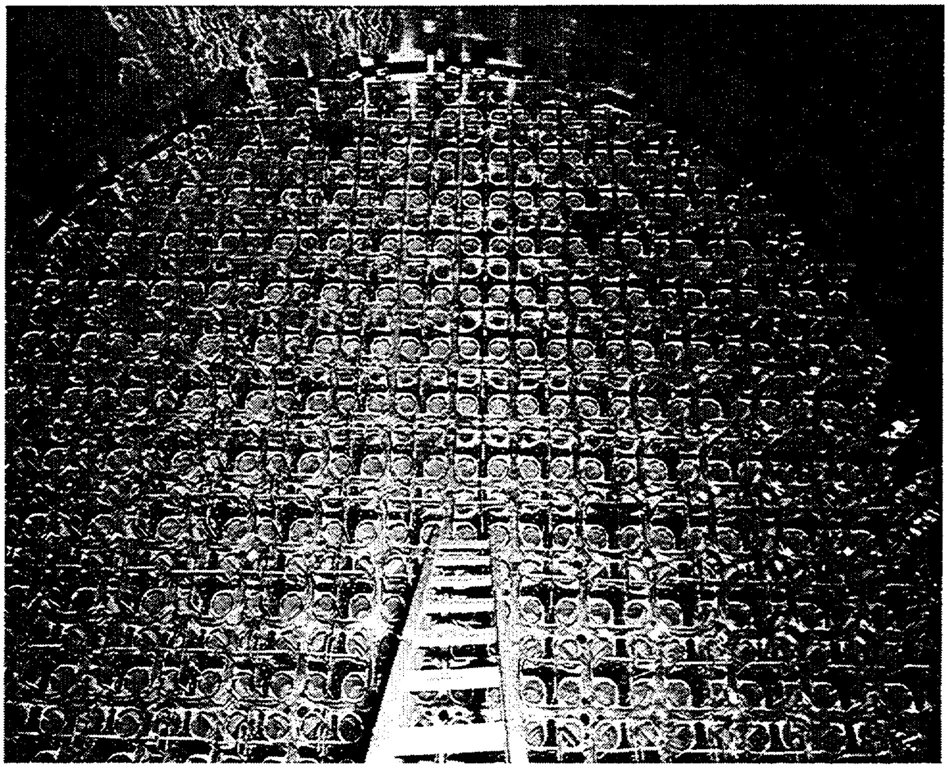


Figure 3.7-19 View of core plate with fuel support structures in place at Peach Bottom

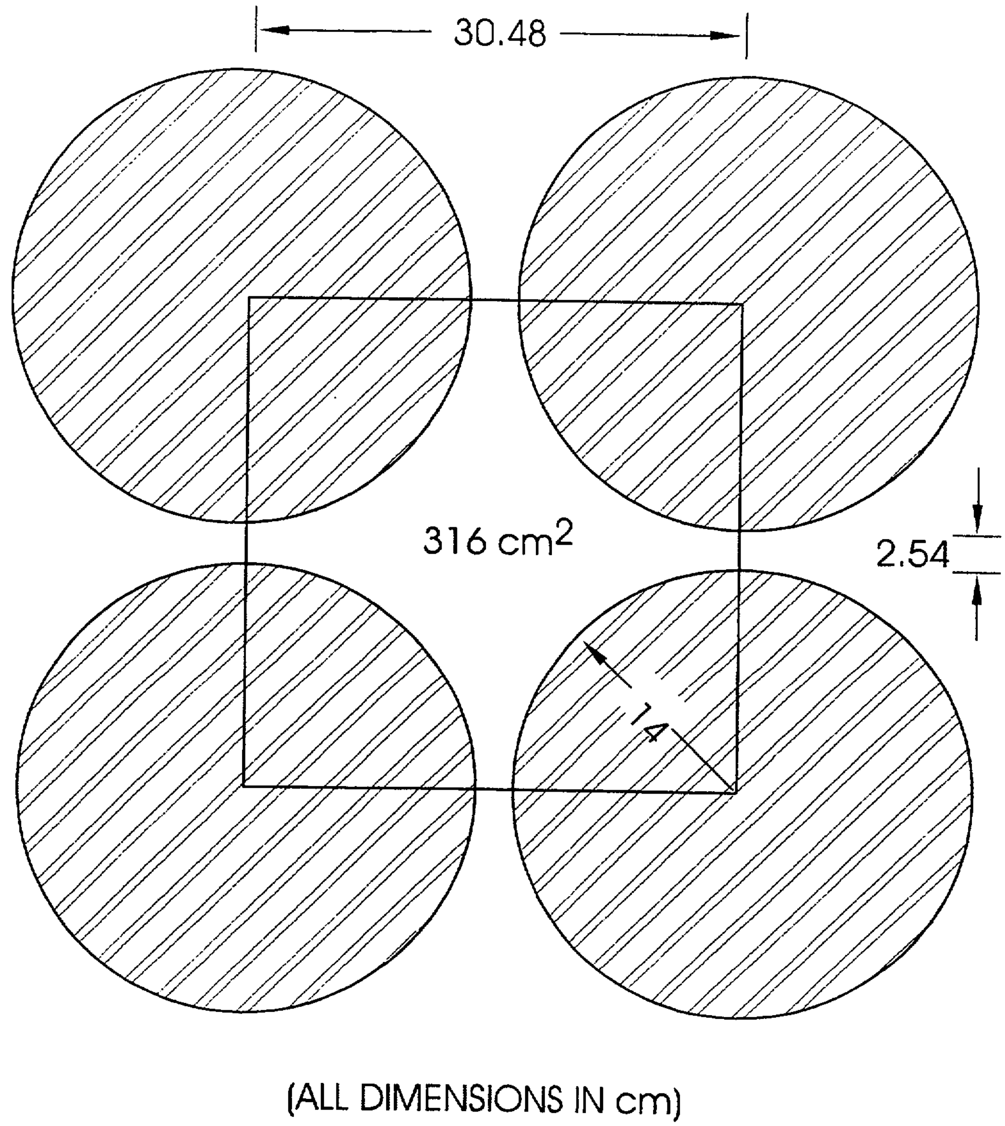


Figure 3.7-20 Two-thirds of the area beneath the BWR core is blocked by the control rod guide tube

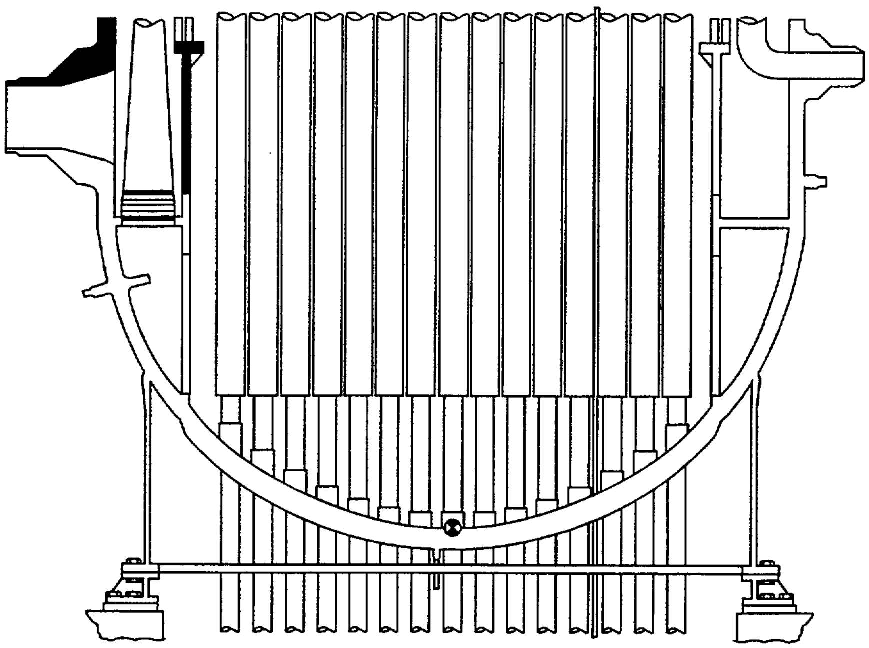


Figure 3.7-21 Code models specific to the BWR lower plenum and bottom currently exist

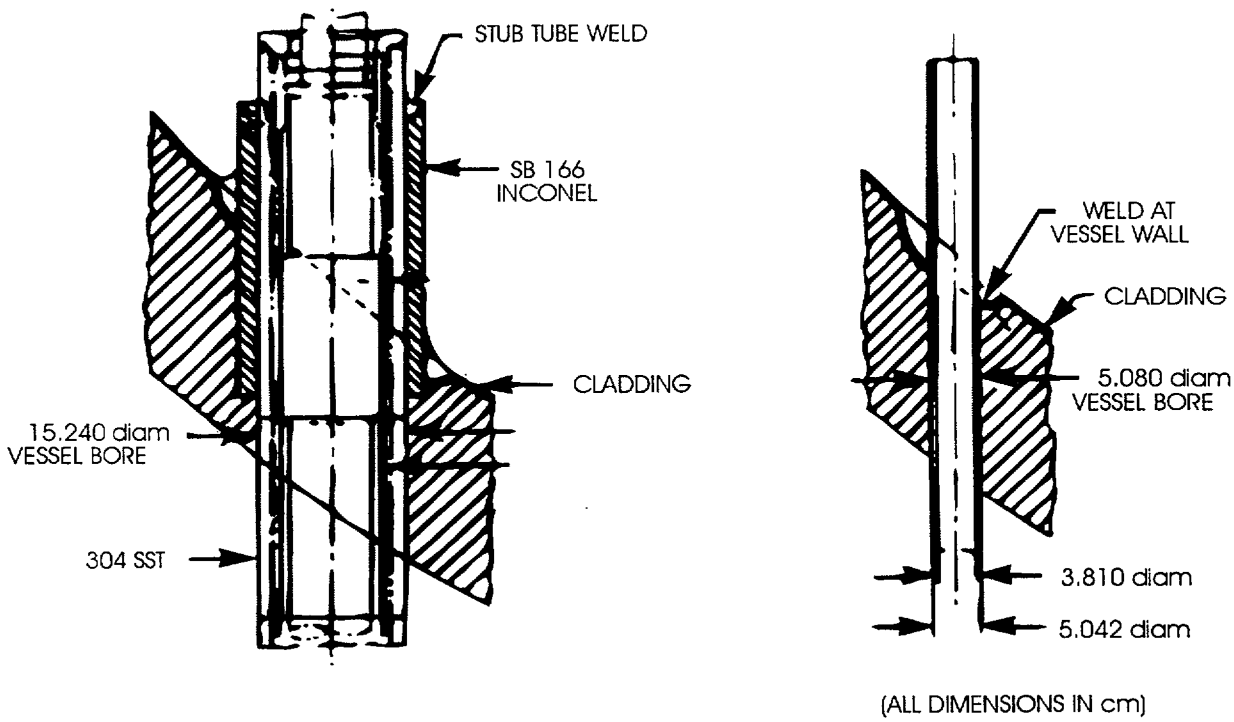


Figure 3.7-22 The BWR control rod drive mechanism assemblies are held in place by upper stub welds; the incore instrument tubes are supported by welds at the vessel wall.

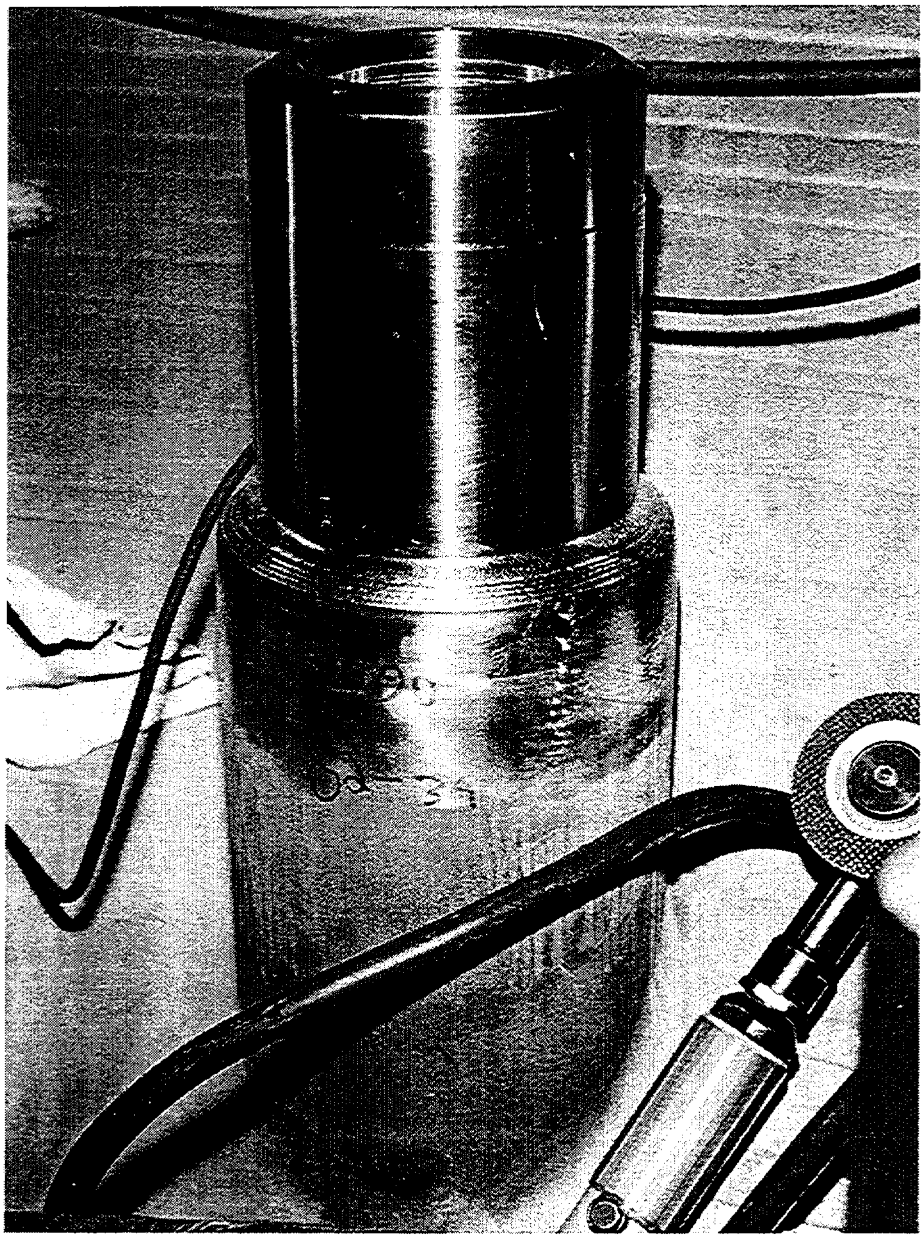


Figure 3.7-23 Weld holding control rod drive housing in place within stub tube at Peach Bottom

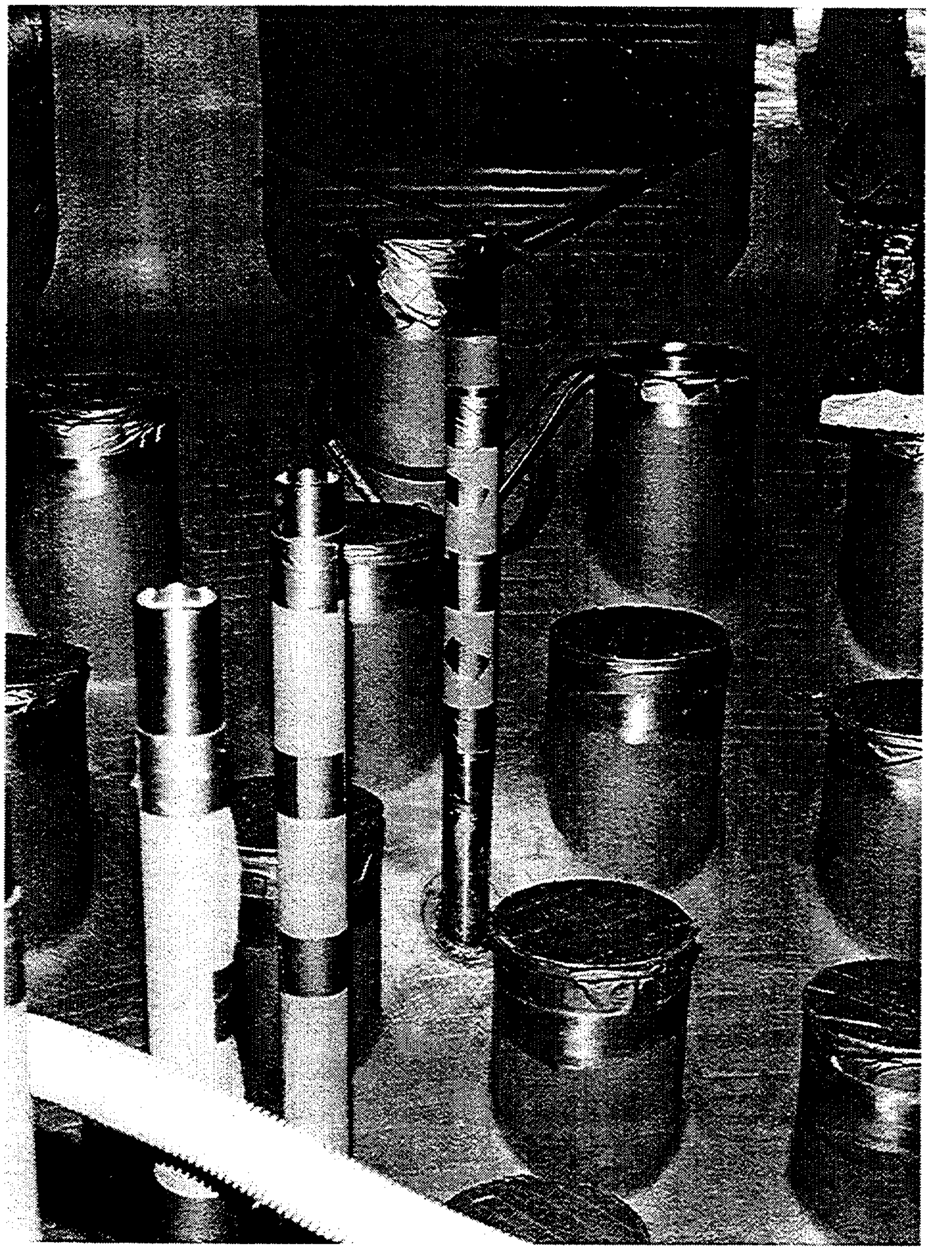


Figure 3.7-24 Instrument guide tube weld location at inner surface of vessel wall at Peach Bottom

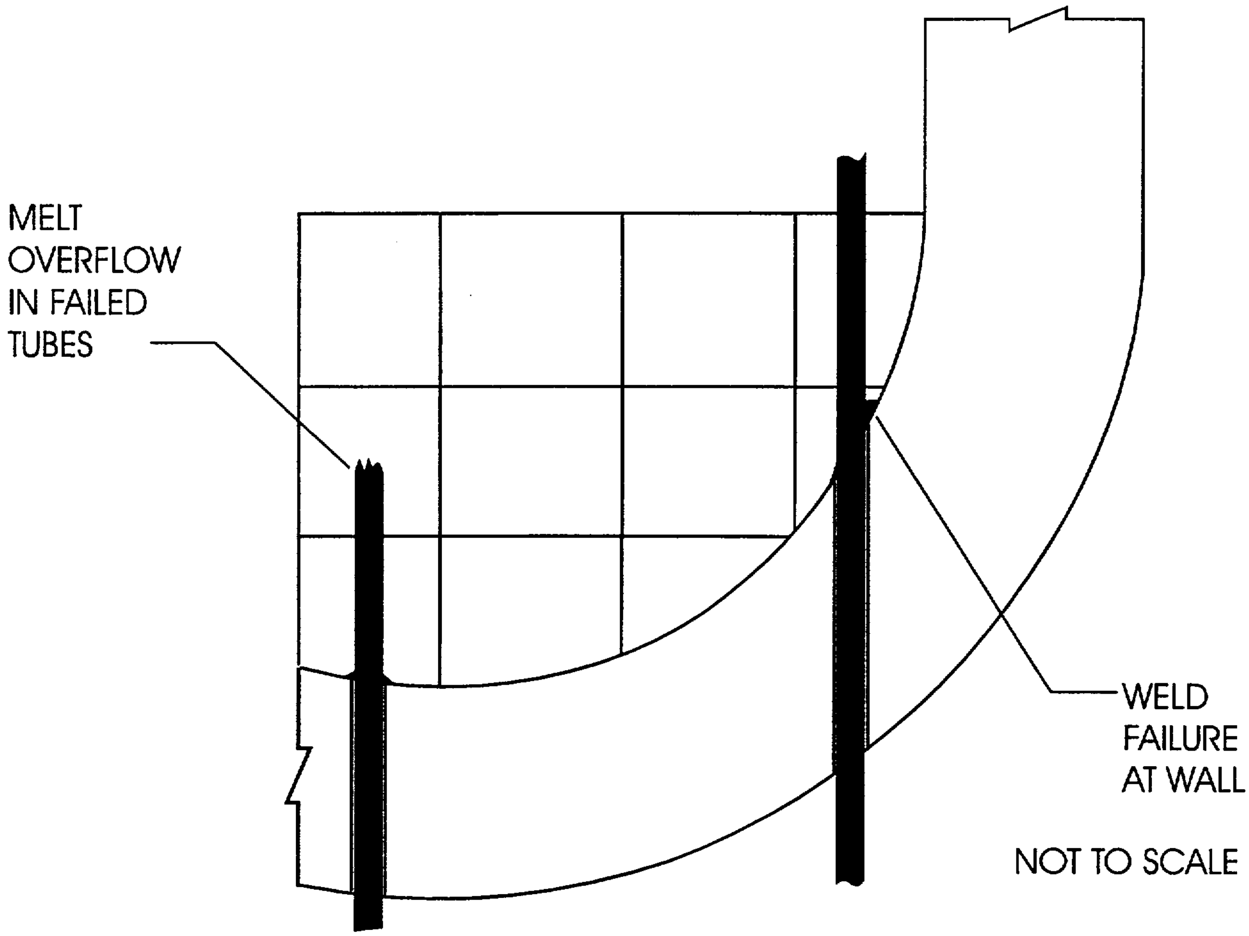


Figure 3.7-25 Instrument tube failure by creep-rupture of welds and by melt overflow can be represented

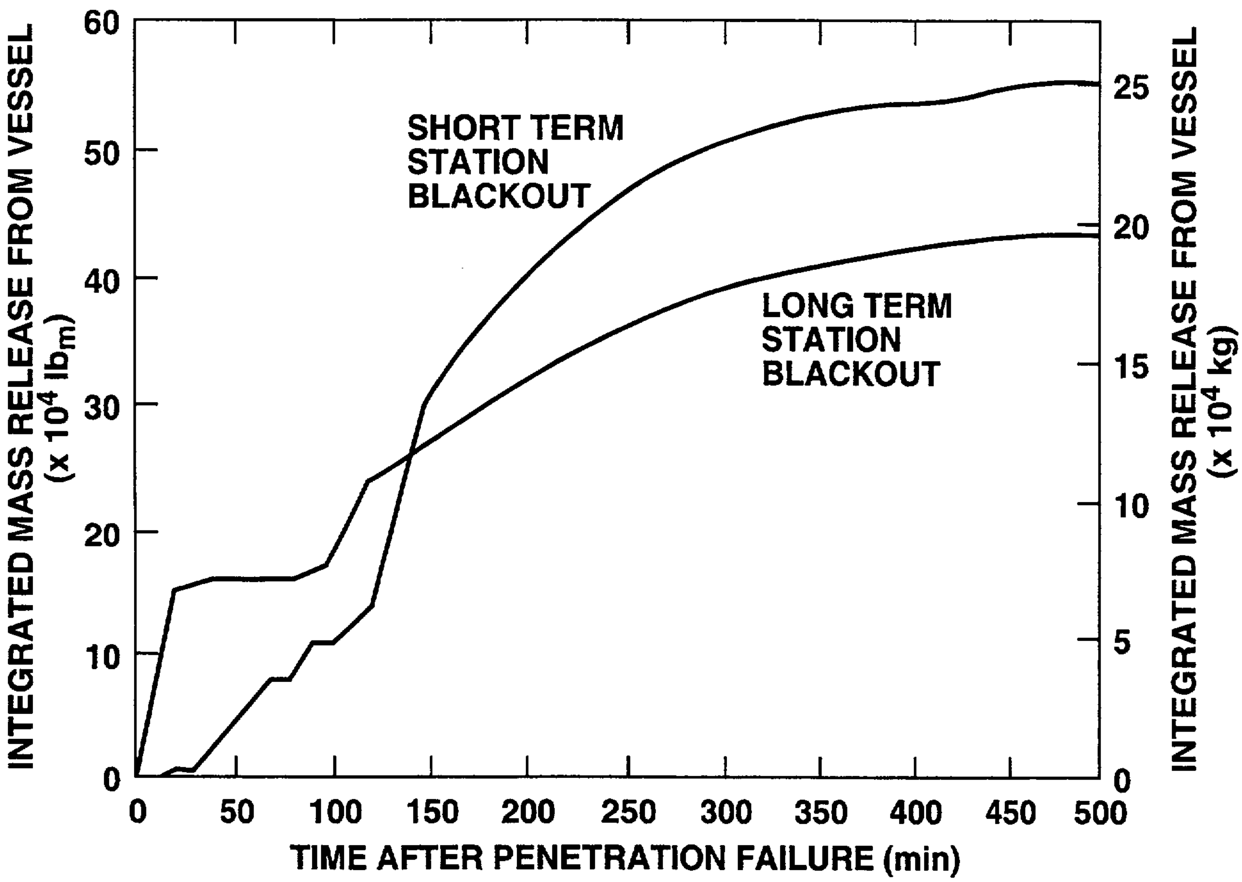


Figure 3.7-26 Zirconium oxidation accelerates the initial debris release rate for pressurized accident sequences

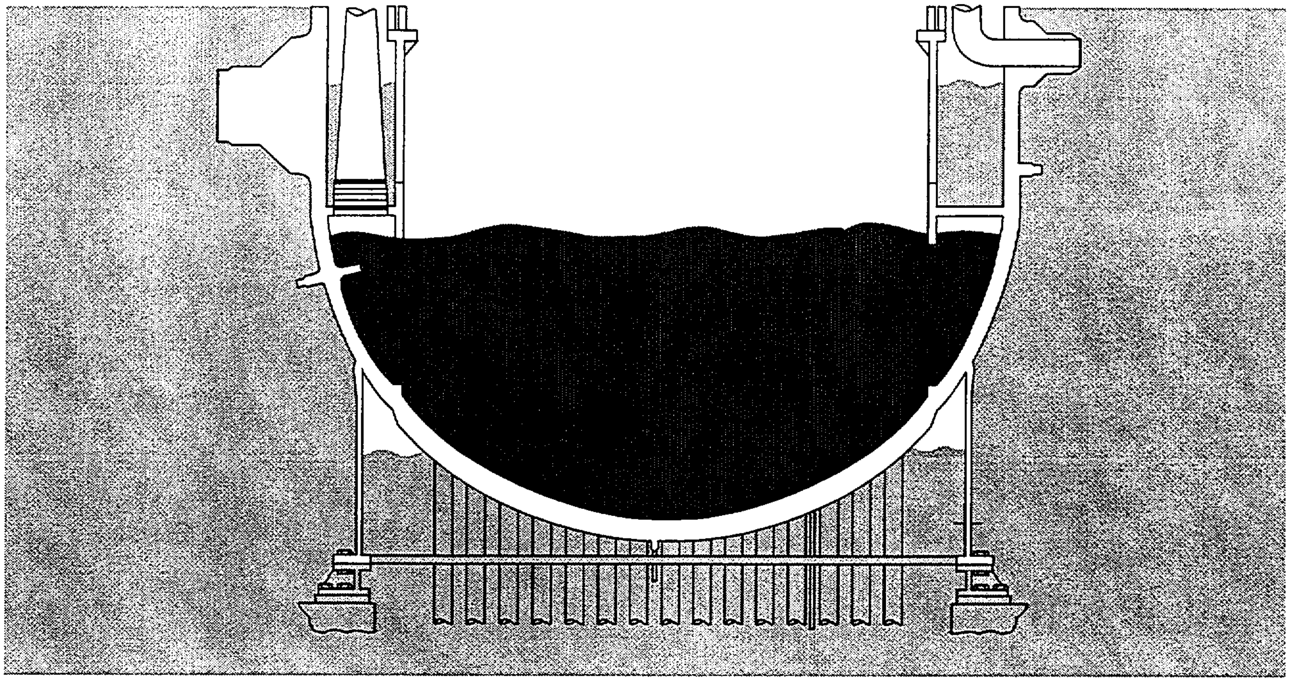


Figure 3.7-27 Atmosphere trapping within the reactor vessel support skirt could limit water contact with the wall

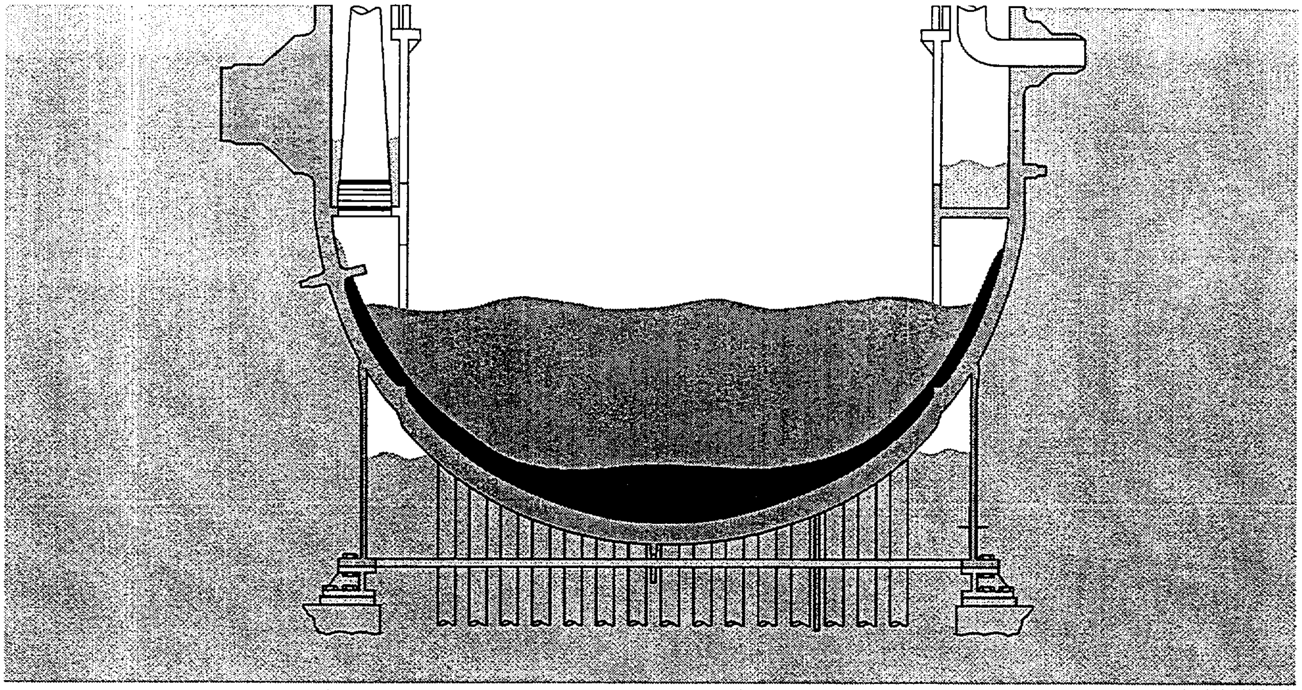


Figure 3.7-28 Delayed wall creep rupture would occur in the vicinity of the gas pocket

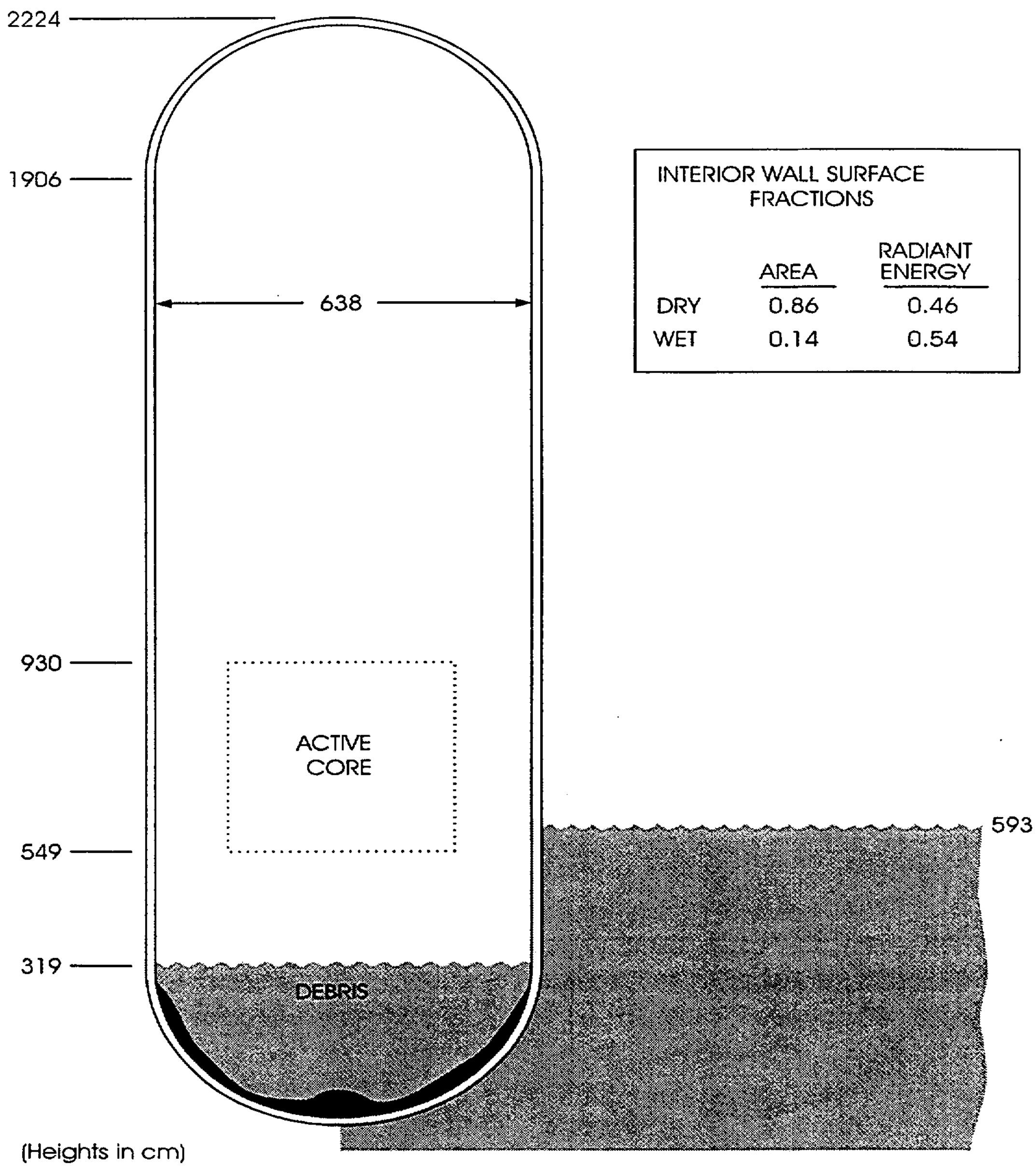


Figure 3.7-29 Cooling of upper vessel wall would be necessary after internal vessel structures have melted

References for Section 3.7

1. S. R. Greene, "Realistic Simulation of Severe Accidents in BWRs - Computer Modeling Requirements," NUREG/CR-2940, ORNL/TM-8517, Oak Ridge National Laboratory, April 1984.

2. L. J. Ott, "Advanced Severe Accident Models for BWR Applications," Nuclear *Engineering Design*, 115, pp. 289-303, 1989.

3. General Electric Company, "BWR Owner's Group Emergency Procedure Guidelines," Revision 4, NEDO-31331, March 1987.

4. Tennessee Valley Authority, "Browns Ferry Final Safety Analysis Report," Volume 2, Section 4.4.5.

5. W. B. Scott, et al., "Recriticality in a BWR Following a Core Damage Event," NUREG/CR-5653, PNL-7476, Pacific Northwest Laboratory, December 1990.

6. U.S. Nuclear Regulatory Commission, "Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants, Final Rule," July 1984.

7. S. A. Hodge, J. C. Cleveland, T. S. Kress, and M. Petek, "Identification and Assessment of BWR In-Vessel Strategies," NUREG/CR-5869, ORNL/TM-12080, Oak Ridge National Laboratory, October 1992.

8. B. I. Shamoun and R. J. Witt, "Parametric Study of Recriticality in a Boiling Water Reactor Severe Accident," *Nuclear Technology*, 107, pp. 227-235, August 1994.

9. S. Hagen and K. Hain, "Out-of-Pile Bundle Experiments on Severe Fuel Damage," CORA Program, KfK 3677, Forschungszentrum Karlsruhe (FZK), (formerly Kernforschungszentrum Karlsruhe [KfK]), 1986.

10. R. O. Gauntt, R. D. Gasser, and L. J. Ott, "The DF-4 BWR Control Blade/Channel Box Fuel Damage Experiment," NUREG/CR-4671, SAND86-1443, Sandia National Laboratories, November 1989.

11. P. Hofmann, M.Markiewicz, and J. Spino, "Reaction Behaviour of B_4C Absorber Material with Stainless Steel and Zircaloy in Severe LWR Accidents," KfK 4598, Forschungszentrum Karlsruhe (FZK), (formerly Kernforschungszentrum Karlsruhe [KfK]), July 1989.

12. L. J. Ott, "Posttest Analysis of the CORA-28 Preoxidized BWR Experiment," ORNL/NRC/LTR/-93/26, Letter Report, Oak Ridge National Laboratory, September 1993.

13. M. D. Allen, H. W. Stockman, K. O. Reil, and J. W. Fisk, "Fission Product Release and Fuel Behavior of Irradiated Light Water Reactor Fuel Under Severe Accident Conditions - The ACRR ST-1 Experiment," NUREG/CR-5345, SAND89-0308, Sandia National Laboratories, November 1991.

14. A. J. Grimley, "A Thermodynamic Model of Fuel Disruption in ST-1," NUREG/CR-5312, SAND-3324, Sandia National Laboratories, February 1991.

15. R. O. Gauntt and L. L. Humphries, "Final Results of the XR2-1 BWR Metallic Melt Relocation Experiment," NUREG/CR-6527, SAND97-1039, Sandia National Laboratories, 1997.

16. F. P. Griffin, "BWR Control Blade/Channel Box Interaction and Melt Relocation Models for SCDAP," Revision 3, ORNL/NRC/LTR-92/12/R3, Letter Report, Oak Ridge National Laboratory, March 1995.

17. B. W. Spencer, et al., "Fragmentation and Quench Behavior of Corium Melt Streams in Water," NUREG/CR-6133, ANL-93/32, Argonne National Laboratory, February 1994.

18. D. Magallon, et al., "Status of theFARO/KROTOS Melt-Coolant Interactions Tests," *Proceedings of the Twenty-Third Water Reactor Safety Information Meeting*, NUREG/CP-0149, Vol 2, pp. 157-185, European Commission, Joint Research Centre, Safety Technology Institute, 1995.

19. J. L. Rempe, et al., "Light Water Reactor Lower Head Failure Analysis," NUREG/CR-5642, EGG-2618, Idaho National Engineering Laboratory, October 1993.

4.0 Accident Progression In The Containment

4.0.1 Introduction

As discussed in Chapter 1, containments began to evolve when designers realized that remote siting would not be practical in all cases. The first containments were provided for the Knolls Atomic Power Laboratory and Shippingport experimental reactors in order to allow them to be sited in more populated areas. Containments for large power reactors evolved during the 1960s, representing a key element of the defense-in-depth strategy. In the event of a design-basis accident, containments are designed to minimize leakage and keep offsite doses well below the 10 CFR 100 limits.

Two basic strategies are used in U.S. containments. The passive pressure suppression approach, used in all General Electric Boiling Water Reactors (BWRs) and Westinghouse Pressurized Water Reactor (PWR) Ice Condenser Containments, involves the use of an energy absorbing medium to absorb most of the energy released during a design-basis loss-of-coolant accident (LOCA). For BWRs the medium is water contained in a suppression pool, and for ice condenser containments, the medium consists of numerous columns of ice. The second approach, used in most PWRs, is simply to design a large, strong volume to receive the energy. All containments also contain active cooling systems, such as sprays and fan coolers, to provide additional cooling and pressure suppression during a design-basis accident. These active systems do not act quickly enough to affect the initial blowdown during a large-break LOCA, but limit further pressure increases and are also beneficial during slower developing accidents.

Containments are designed to cope with the accidents specified in Chapter 15 of the Safety Analysis Report, as discussed in Section 2.1. Design-basis accidents resulting to the greatest predicted containment loads are generally initiated by large reactor coolant or main steam pipe breaks. As described in Section 4.1, containments are designed to survive such accidents with considerable margin.

The China Syndrome and the Reactor Safety Study began to cast doubt on the ability of containments to survive all possible accidents, and it became clear that risk to the public is usually dominated by those accidents in which the containment fails or is bypassed. In a severe accident, there are sources of energy and phenomena that can cause a greater threat to containment than the design-basis LOCA. The hydrogen burn at Three Mile Island highlighted the potential threats from severe accident phenomena, even though the containment survived that particular event. The remainder of this chapter describes different containment designs and the potential threats to those designs.

4.0.2 Learning Objectives for Chapter 4

At the end of this chapter, the student should be able to:

1. Describe the six basic containment types and associated engineered safety features.

2. Identify which containment types are less susceptible to isolation failures.

3. Contrast the potential failure mechanisms for steel and concrete containments.

4. Describe the following causes of containment failure. For each cause, indicate when failure could occur.

    a. direct containment heating
    b. fuel-coolant interactions
    c. liner meltthrough
    d. combustion
    e. long-term overpressure

5. Describe a BWR accident scenario in which venting of a Mark I or Mark II containment might be appropriate.

6. List at least one concern regarding the containment if AC power is restored late in a station blackout accident.

7. Explain the different hydrogen control measures used in BWR Mark I, II, and III and PWR ice condenser containment designs.

8. Characterize the usefulness of hydrogen recombiners during severe accidents.

4.1 Containment Characteristics and Design Bases

4.1.1 Containment Types

There are six basic containment types used for U.S. Light Water Reactors (LWRs). Four of those designs primarily use the passive pressure suppression concept, and two rely primarily on large, strong volumes. All of these containments are constructed of either steel or concrete with a steel liner for leak tightness. Except for Big Rock Point, BWR designs, which have evolved from the Mark I to the Mark III design, all use a pressure suppression pool. A few Westinghouse PWRs have ice-condenser (pressure suppression) containments, but most PWRs have large dry containments or a subatmospheric variation of the large dry containment. Table 4.1-1 lists the number of containments of each type.[1] Figure 4.1-1 shows a comparison of the containment volumes and design pressures for typical containments.[2] The design pressures for containments are based on a very conservative design process. If all isolation features work properly, it is likely that containments will not fail until the design pressures have been greatly exceeded. Figure 4.1-2 compares the design pressures with realistic estimates of ultimate failure pressures for six typical containments.[3,4]

The next six subsections describe the six containment types in more detail. It is important to note that there are plant-specific variations within each containment type, and these discussions do not delineate all of these design differences.

4.1.1.1 Large Dry Containments

A typical large dry containment is shown in Figure 4.1-3. A large dry containment is designed to contain the blowdown mass and energy from a large break LOCA, assuming any single active failure in the containment heat removal systems. These systems may include containment sprays and/or fan coolers, depending on the particular design. Large dry containments can be of either concrete or steel construction. Concrete containments have steel liners to assure leak tightness. Large dry (and all other) containments have a large, thick basemat that provides seismic capability, supports the structures, and may serve to contain molten material during a severe accident.

During an accident, most of the water introduced into containment through a pipe break or relief valves collects in the sump. The water can include the initial reactor coolant inventory plus additional sources injected into the reactor coolant system. Water may enter containment as vapor, liquid, or a two phase mixture. The liquid portion drains quickly into the sump and the vapor portion may condense (on structures or containment spray drops or coolers) and then drain into the sump. Once water storage tanks have been depleted, water in the sump is recirculated to the vessel and/or the containment sprays using recirculation systems to provide long-term heat removal. It is important that the sumps be kept clear of debris that could inhibit this recirculation. Large dry containments are not as susceptible to hydrogen combustion as other, smaller containments. No systems are provided for short term hydrogen control during a severe accident (see Section 4.6). However, hydrogen recombiners are provided to allow long-term hydrogen control.

4.1.1.2 Subatmospheric Containments

Subatmospheric containments are very similar to large dry containments, as shown in Figure 4.1-4. The major difference is that the containment is maintained at a negative pressure (~5 psi or 35 kPa) with respect to the outside atmosphere. This negative pressure means that leakage during normal operation is into the containment rather than to the atmosphere. Further, this negative pressure provides some additional margin for response to design basis accidents, and therefore the design pressure and/or volume can be reduced accordingly. Keeping the containment at a subatmospheric pressure also means that any significant containment leaks will be readily detected, when maintaining the negative pressure becomes more difficult.

4.1.1.3 Ice Condenser Containments

Figure 4.1-5 shows the layout of an ice condenser containment and Figure 4.1-6 shows the ice condenser in more detail. Ice condenser containments are constructed of either concrete or steel. Ice condenser containments are the only PWR containments that rely primarily on passive pressure suppression. The containment consists of an upper and a lower compartment connected through an ice bed. In the event of a design-basis LOCA, steam flows from the break, into the lower compartment, and up into the ice beds where most of the steam is condensed. Return air fans maintain a forced circulation from the upper to lower compartments, enhancing flow through the ice beds. One-way doors are present at the entrance and exit of the ice bed region. These doors open upon slight pressure from the lower compartment, but close if air flow occurs in the reverse direction.

The ice beds are more than adequate to limit the peak pressure from a design-basis LOCA. However, in a long-term accident, the ice will eventually melt and containment heat removal will be required. Thus, containment sprays are provided in the upper compartment of the containment. Water from the sprays drains through sump drain lines down into the lower compartment sump, where it can be recirculated for long-term heat removal. It is noteworthy that, because of the melting ice, there will be more water in the lower compartment during many accidents than would be present in a large dry containment. The effect of this additional water upon severe accident phenomena will be discussed in later sections.

Because of their smaller volume, ice condenser containments are more susceptible to combustion events than large dry containments. In fact, a combustion event involving the same quantity of hydrogen that was burned at TMI-2 might have led to containment failure in an ice condenser containment. Therefore, specific hydrogen control requirements have been placed on ice condenser containments. These requirements are examined in Section 4.6.

4.1.1.4 BWR Mark I Containments

Mark I containments are provided for most of the older BWR plants, 24 in number. The Mark I is a pressure suppression containment, which allows the containment to be smaller in volume. The basic design is shown in Figure 4.1-7. The containment is divided into the drywell containing the reactor vessel and the wetwell (torus) containing the suppression pool. The containment may be constructed of either concrete or steel. The water in the suppression pool acts as an energy absorbing medium in the event of an accident. If a LOCA occurs, steam flows from the drywell through a set of vent lines and downcomers into the suppression pool, where the steam is condensed. Steam can also be released from the reactor vessel through the safety relief valves and associated piping directly into the suppression pool. In the event that the pressure in the wetwell exceeds the pressure in the drywell, vacuum breakers are provided that equalize the pressure.

The water in the suppression pool can be recycled through the core cooling systems, much the same as sump water is recycled in a PWR. Long-term containment heat removal can be provided by sprays or suppression pool cooling systems either of which can be aligned with appropriate heat exchangers. In addition, Mark I containments are equipped with lines connected to the wetwell that can be used to vent the containment if the pressure becomes too high. As will be discussed later, the particular venting strategy chosen can significantly impact the course of an accident.

Because of the small volume of the Mark I containment, hydrogen control measures are required. In this case, the drywell is inerted with nitrogen during most of the operating cycle to preclude the possibility of combustion. More details on hydrogen concerns for Mark I BWRs are contained in Section 4.6.

4.1.1.5 BWR Mark II Containments

Mark II containments are similar in concept to Mark I containments. Figure 4.1-8 shows a Mark II containment. The suppression pool design is simplified, and the entire containment structure is more unified. Instead of the complicated torus design included in the Mark I containment, the suppression pool simply sits in the wetwell region below the drywell. Containment heat removal systems (sprays and suppression pool cooling) and nitrogen inerting strategies are the same as for the Mark I containments. Containment venting can also be performed in a similar fashion to the Mark I containments.

4.1.1.6 BWR Mark III Containments

While the Mark II design represented an evolution of the Mark I design, the Mark III design introduced major changes. A typical Mark III containment is shown in Figure 4.1-9. Mark III containments can be free-standing steel or steel-lined concrete. These containments have a drywell that functions much as the older designs, but have a larger surrounding containment that includes the wetwell. In the Mark III design, the suppression pool is located in an annular region outside the drywell.

The suppression pool function is essentially the same as in the older designs. In this case, if there is a LOCA in the drywell, then steam will flow through horizontal vents to the suppression pool where the steam will be condensed. It is possible for the blowdown to cause the suppression pool to slosh over the weir wall and partially fill the drywell. In order to assure that adequate water is available in the suppression pool, allowing for recirculation, evaporation, and sloshing, water can be added to the suppression pool from the upper pool above the drywell.

If the pressure in the outer containment exceeds the pressure in the drywell, then vacuum breakers open to equalize the pressure. Long-term containment heat removal can be accomplished with suppression pool cooling or by containment sprays (with appropriate circulation of the water through heat exchangers) in the outer containment.

An important asset of the Mark III design is construction of the outer containment around the drywell, effectively providing a double layer of protection. If containment failure were to occur, in many cases the outer containment would fail first, leaving the drywell and suppression pool intact. Any subsequent fission product releases would still be scrubbed as they passed through the suppression pool, greatly reducing the source term. Thus, the only accidents (other than bypass sequences) likely to produce large source terms must involve failure of the outer containment plus either loss of the suppression pool or failure of the drywell. Further, the containment sprays can be used to remove fission products and reduce the source term.

The Mark III design is an intermediate-sized containment, much like the ice condenser containment. It is large enough that inerting is not required for hydrogen control, but still small enough that some hydrogen control measures are needed. Those measures are discussed in later sections.

4.1.2 Containment Design Criteria

Section 2.1 provided a discussion of design-basis accidents, as included in Chapter 15 of the Safety Analysis Report (SAR). For containments, the design must preclude exceeding of the 10 CFR 100 dose guidelines, given the most limiting accident evaluated in Chapter 15. Specifically, the requirements of 10 CFR 50, Appendix A, General Design Criterion 50 state:

> *The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-off-coolant accident.*[5]

It is interesting to note that, while the criterion indicates *any* loss-of-coolant accident, only those LOCAs considered in Chapter 15 of the SAR are actually considered. For example, the containments are not specifically designed for Reactor Vessel Rupture or Steam Generator Rupture. Generally, one of the most limiting Chapter 15 accidents is the large break LOCA. The large break LOCA tends to produce both higher pressures and more fission products in containment than the other Chapter 15 accidents. Main Steam Line Breaks tend to produce the highest temperatures in containment and determine the temperature design limits.

Section 2.1 discusses the calculations involved in analyzing a Chapter 15 accident, including the significant conservatisms. Figures 4.1-10, 4.1-11, and 4.1-12 depict containment pressure, temperature and energy balance results for PWR design-basis LOCAs in a large dry containment. Figure 4.1-10 shows the calculated containment pressures resulting from a spectrum of postulated reactor coolant system pipe breaks. For this set of calculations the maximum containment pressure of 50.21 psig (346 kPa) occurs for an 8 ft^2 (0.74 m^2) reactor coolant pump discharge line break. Figures 4.1-11 and 4.1-12 provide more detail for this particular accident. In this accident, the blowdown takes approximately 25 sec. Despite the fact that the blowdown occurs with no containment cooling systems operating, the peak pressure does not occur during this period. The reflooding of the core, which includes core flood tank injection at 15.3 sec. and emergency core cooling at 26 sec., generates additional steam which continues to pressurize containment until about 918 sec., when the peak pressure is reached. In this calculation, which can vary for other plants, a containment cooler is started at 43 sec. and the sprays are started at 67 sec., providing some positive reduction in the peak pressure. After 918 sec., the pressure declines, and recirculation cooling from the sump is established at 3500 sec.

While the large break LOCA presents the most significant design-basis accident pressure challenge for containment designers, there are other types of loads that must be considered in the design.[6] These loads include:

1. temperature transients and gradients
2. safe shutdown earthquake loads
3. internal and external missiles
4. mechanical loads from pipe rupture
5. external pressures
6. winds and tornadoes.

Section 2.1.4 described the design basis for seismic and other external events. Thermal transients and gradients could conceivably lead to stresses and cracks or tears in the containment. Missiles can come from many sources, including control rod ejection, shrapnel from a failed pipe, or aircraft impact. When a pipe ruptures, the resulting forces on the piping could cause failure at the point where the piping penetrates the containment. External pressures (and buoyant forces) can result due to external increases in barometric pressure or internal drops in pressure resulting from internal cooling or inadvertent spray operation.

In practice, it is impossible to design and construct a perfect containment, that is, one that has zero leakage over the range of postulated accident conditions. Therefore, nonzero design leakage rates are established that are intended to be as low as can be reasonably achieved and that will keep the offsite exposures below the dose guidelines established in 10 CFR 100.[7] These design leakage rates can be site- and plant-specific because the offsite doses are affected by the site geometry and the local meteorology, as well as the reactor type. However, some plants simply use standard technical specifications that are more stringent than a site-specific analysis would allow.

Leakage from a containment structure can occur due to failure of the containment structure, failure of penetrations through the structure, and failure of isolation valves. Penetrations through the containment structures include piping penetrations, electrical penetrations, hatches and airlocks. Isolation valves are provided on all pipes and ducts that penetrate the containment. Normally, two isolation valves are provided for each line, with the isolation valves consisting of locked closed or automatic isolation valves. Requirements for these isolation valves are contained in 10 CFR 50, Appendix A, General Design Criteria 54 through 57.[8]

Containment leakage rates are determined in the SAR and Technical Specifications. Criteria for testing containment leakage are set forth in 10 CFR 50, Appendix J.[9] This appendix became effective in 1973. Its purpose is to implement, in part, 10 CFR 50, Appendix A, General Design Criteria 16 which mandates "an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment ..." for postulated accidents. Until 1995 this appendix specified prescriptive containment leakage-testing requirements, including the types of tests required, how they should be conducted, and the reporting requirements. Effective October 26, 1995, an amendment to Appendix J was issued which added an alternative method, based on performance and risk, for meeting the containment leakage rate test requirements.[10] This method is designated as Option B; the original Appendix J method is referred to as Option A. Guidance for the implementation of Option B is provided in NRC Regulatory Guide 1.163.[11] With either option, three types of tests are generally performed to assure leakage remains within design limits:

1. Type A tests - tests of the overall integrated leakage rate,

2. Type B tests - tests to detect local leaks around containment penetrations, and

3. Type C tests - tests to measure containment isolation valve leakage rates.

Table 4.1-2 provides examples of design leakage rates that correspond to Option A of Appendix J. The higher allowed leakage rates for the pressure suppression containments is a result of their smaller volumes. Although the criteria are somewhat different, the allowable leakage permitted under Option B of Appendix J is 1.0 wt.%/day. This reflects the NRC conclusion that a small additional leakage would have no significant impact on safety.

Assuring that the design leakage rates are met is a complex process involving a variety of tests, some of which are difficult to perform. Based on an analysis of past containment tests, Option B of Appendix J was developed to simplify the testing process, primarily by allowing the testing frequency to be reduced if the results of past tests demonstrate high containment performance and therefore low risk of significant leakage. Table 4.1-3 provides a comparison of the test frequency requirements for the two Appendix J options. This table illustrates the advantage that Option B provides if the containment leakage requirements are consistently met. However, in recognition of the costs and effort required to implement a new testing process, licensees can continue to test under Option A; conversion to Option B is voluntary.

The amount of leakage from a containment is a function of the length of time that the containment remains pressurized. Further, there are some postulated accidents in which energy may be added to containment for many hours of even days. Therefore, the NRC has established requirements for containment heat removal. These requirements are contained in 10 CFR 50, Appendix A, Criterion 38.[12] Containment heat removal systems may involve sprays, fan coolers, suppression pool cooling, or emergency core cooling recirculation cooling and must meet the single failure criterion.

4.1.3 Containment Failure Modes

In the event that a containment does fail, the manner in which it fails can have a significant impact on offsite releases. If a containment leaks slowly, then large fractions of the radionuclides may still be retained inside the containment or surrounding buildings, depending on where the leak occurs. Retention can result from gravitational settling of radioactive aerosols inside the containment or surrounding buildings or from sprays or other systems removing the radionuclides from the containment atmosphere. The effectiveness of these processes depends upon the residence time of the radionuclides in containment. Conversely, a large rupture of the containment can lead to rapid transport of radionuclides to the environment with minimal retention.

The containment failure mode that occurs depends upon the containment design and the particular phenomena that cause the failure. Particular severe accident phenomena (including those beyond the design-basis) will be discussed in later sections; however, the challenges that they produce include:

1. overpressure
2. dynamic pressures (shock waves)
3. internal missiles
4. external missiles
5. meltthrough
6. bypass.

Overpressure can theoretically lead to either leakage or large rupture in any type of containment. Overpressure can result from several different causes, as discussed in later sections. However, it is important to recognize that pressure transients following reactor vessel blowdown under severe accident conditions can be more severe than those normally considered in the analyses of design-basis accidents. This is true because of the initial containment pressure increases caused by hydrogen generation and containment heating during the early phases (prior to reactor vessel failure) of a severe accident. As a containment is pressurized, it begins to deform. High temperatures exacerbate the problem. These deformities can lead to leakage around penetrations in the containment or to tearing of the steel liner (in concrete containments). Based on recent studies, leakage is considered the more likely outcome for concrete containments.[13] The concrete structure is unlikely to rupture as a result of pressure challenges (even if the steel liner tears), but rather is more likely to crack. Steel containments are susceptible to rupture in the event that the penetrations do not leak and the containment continues to pressurize. Given sufficient pressure, a crack in a steel containment can propagate catastrophically. Generally, assuming that early penetration leakage does not occur, steel containments have a larger margin between the design and ultimate failure pressures than concrete containments.

Shock waves and missiles can potentially cause large holes in the containment. However, the containments are designed for the most credible external missiles, such as tornado-driven missiles, and some types of internal missiles, such as control rod ejections. Missiles or shock waves resulting from hydrogen detonations or steam explosions are a possible threat that will be discussed in more detail later.

There are two basic types of meltthrough to consider. First is the possibility of basemat meltthrough (the China Syndrome). In this case, following vessel failure the molten material melts through the basemat over a period of hours or days and vents the containment through the surrounding soil and can release substantial amounts of contaminated water. This failure mode is not generally catastrophic, because of the long time available for emergency response actions and the possibility of some retention in the soil. The second type of meltthrough is most applicable to Mark I BWR containments. In this case, molten material can exit the area beneath the reactor and flow across the floor, directly contacting the steel liner and causing it to fail. This type of failure, which is addressed in more detail in Section 4.7, can happen much more quickly than basemat meltthrough and can lead to more serious consequences. A similar scenario may be possible for PWR ice condenser containments, if debris is blown out of the reactor cavity near the seal table.

There are two other types of containment failure that can lead to severe consequences: (1) containment bypass and (2) isolation failure. Containment bypass involves failure of the reactor coolant system boundary in such a manner that a path is created to the outside without going through containment.

Bypass involves failures in the reactor coolant pressure boundary separating high pressure and low pressure systems. Normally, this involves the failure of at least two valves. For example, the valves separating the primary system from the Residual Heat Removal (RHR) system may fail, thus putting high pressure into the RHR system. Because the RHR system is normally constructed with low pressure piping and components, it may fail outside containment, providing a direct path from the core to the outside. Such sequences are usually referred to as interfacing systems LOCAs. In PWRs, steam generator tube ruptures provide an additional possibility of containment bypass. Primary system pressure will lift the relief valves on the secondary side, with the potential for stuck-open valves to provide the path to the atmosphere.

Containment isolation failure involves failure of the containment isolation function as a result of containment isolation valve failures or other openings in the containment boundary external to the reactor coolant system. These failures may be the result of preexisting leaks or the failure of isolation valves to close upon demand. The failures are more related to system and procedural malfunctions, rather than severe accident phenomena. In this case, the containment has no chance to function and fission products have a direct path outside to the atmosphere. Isolation failures are less likely in Mark I and II BWRs because of their inerted containments that make large leaks easily detected. Similarly, isolation failures are unlikely in PWR subatmospheric containments.

It should be recognized that plant-specific containment design differences abound, many with important ramifications with respect to plant response under severe accident conditions. This is true even for supposedly "sister" plants such as the Browns Ferry and Peach Bottom plants with Mark I containments. A good example is provided by Figure 4.1-13, which affords a subjective comparison of the bolting arrangements used at these two plants for the respective drywell head closure flanges. In this case, the Browns Ferry arrangement is less prone to flange separation by bolt elongation at elevated temperatures. This difference may be important for severe accident sequences involving high drywell temperatures (see Section 4.2), since the integrity of the silicon seals has been demonstrated[14] to degrade significantly at temperatures in excess of about 600 K (620 °F). The point here is that the failure pressure can vary with temperature and can be affected by seemingly unimportant design differences.

Table 4.1-1 Number of U.S. containments of each type

Containment Type	Number
PWR Large Dry	59*
PWR Subatmospheric	7
PWR Ice Condenser	9
BWR Mark I	24
BWR Mark II	8
BWR Mark III	4

*Includes Big Rock Point, which is a BWR I.

Table 4.1-2 Examples of design leakage rates (integrated leakage)*

Plant	Containment Type	Peak Design-Basis Accident Pressure psig	(kPa)	Maximum Allowable Leakage (wt.%/day)
Peach Bottom	BWR Mark I	49.1	(339)	0.5
LaSalle	BWR Mark II	39.6	(273)	0.635
Grand Gulf	BWR Mark III	11.5	(79)	0.437
Sequoyah	PWR Ice Condenser	12	(83)	0.25
Surry	Subatmospheric	45	(310)	0.1
Zion	Large Dry	47	(324)	0.1

*data taken from the following reports:

Integrated Leak Rate Test Report for Peach Bottom Unit 3, March 18, 1992.
Integrated Leak Rate Test Report for LaSalle Unit 1, March 12, 1992.
Integrated Leak Rate Test Report for Grand Gulf Unit 1, August 4, 1989.
Integrated Leak Rate Test Report for Sequoyah Unit 2, February 19, 1985.
Integrated Leak Rate Test Report for Surry Unit 2, September 3, 1991.
Integrated Leak Rate Test Report for Zion Unit 1, July 5, 1988.

Table 4.1-3 10 CFR 50 Appendix J test frequency requirements

Test Type	Option A	Option B
A	1. Preoperational leakage rate test. 2. Three tests during each 10 year service period, at approximately equal intervals. 3. If any periodic test fails, the schedule for subsequent tests will be reviewed and approved by the Commission. A summary report must be provided to the NRC. 4. If two consecutive tests fail, a test is required at each refueling or every 18 months, whichever comes first. 2. Applies after two consecutive successful tests.	1. Preoperational leakage rate test. 2. Test within 48 months and then at periodic interval (maximum of 10 years) based on past leakage performance of containment. 3. If test fails, successful test required within 48 months before returning to extended test interval. 4. Visual inspection of containment interior and exterior required prior to each test. Perform two additional inspections at refueling outages prior to the next test if the interval has been extended to 10 years. 5. Test results which exceed performance criteria may require reporting to NRC (e.g., License Event Report [LER]).
B	1. Except for airlocks: (a) Tests required during shutdown for refueling, or other convenient interval, but at least every 2 years. (b) Frequency of tests reduced for penetrations employing a continuous leakage monitoring system. 2. Airlocks: Tests required prior to initial fuel loading and at six-month intervals thereafter. 3. If any periodic test fails, a summary report must be provided to the NRC.	1. Test prior to initial criticality. 2. Test within 30 months and then at periodic interval based on safety significance and past performance of each non-airlock boundary. 3. If test fails, successful test required within 30 months before returning to extended test interval. 4. Tests required at least every 10 years. The maximum interval for airlocks is 30 months. 5. Test results which exceed performance criteria may require reporting to NRC (e.g., LER).
C	1. Tests required during shutdown for refueling, but at least every 2 years. 2. If any periodic test fails, a summary report must be provided to the NRC.	1. Test prior to initial criticality 2. Test within 30 months and then at periodic interval based on safety significance and past performance of each isolation valve. 3. If test fails, successful test required within 30 months before returning to extended test interval. 4. Tests required at least every 60 months. The maximum interval for certain valves (e.g., main steam valve in BWRs) is 30 months. 5. Test results which exceed performance criteria may require reporting to NRC (e.g., LER).

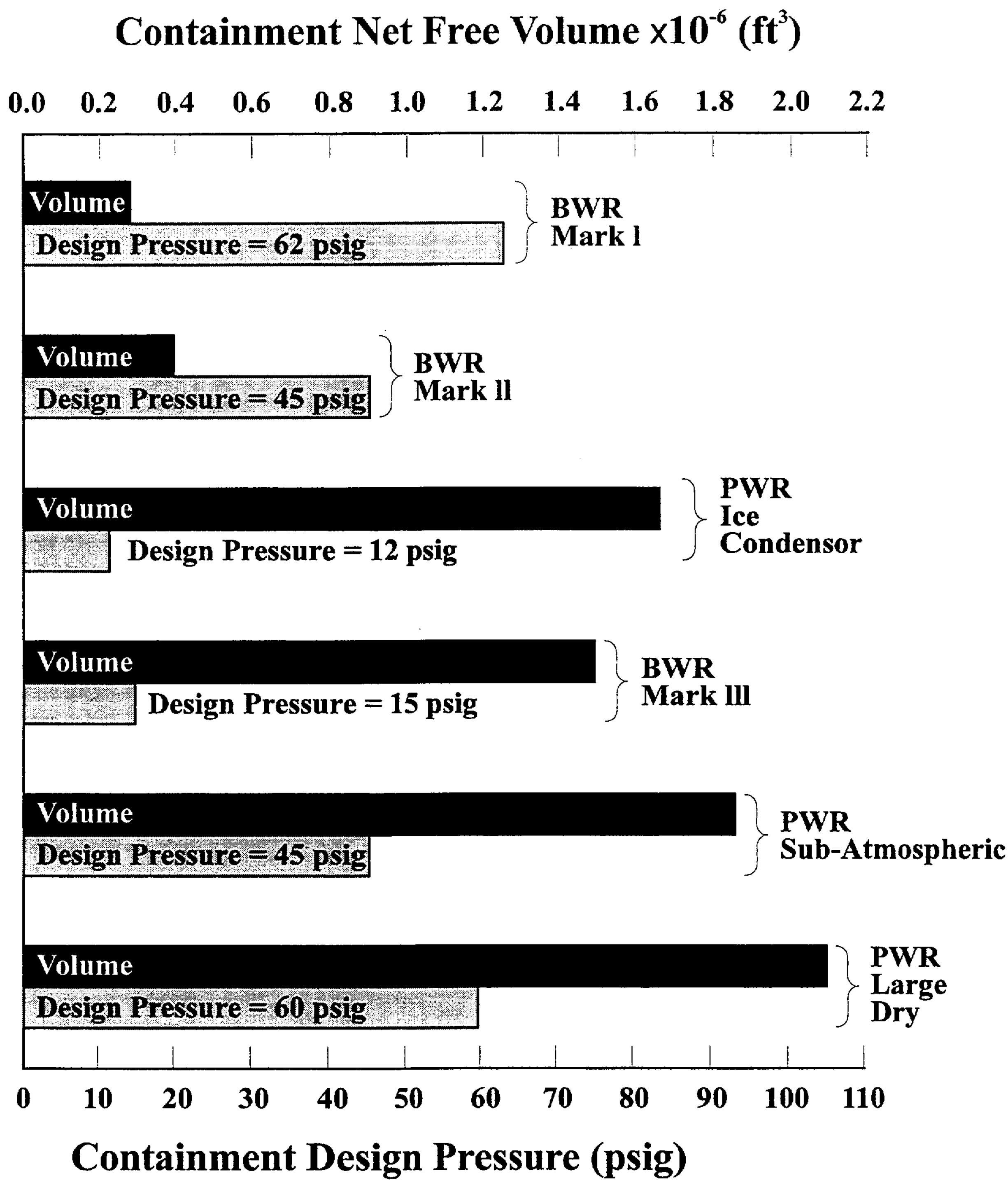


Figure 4.1-1 Typical containment volumes and design pressure (psig)

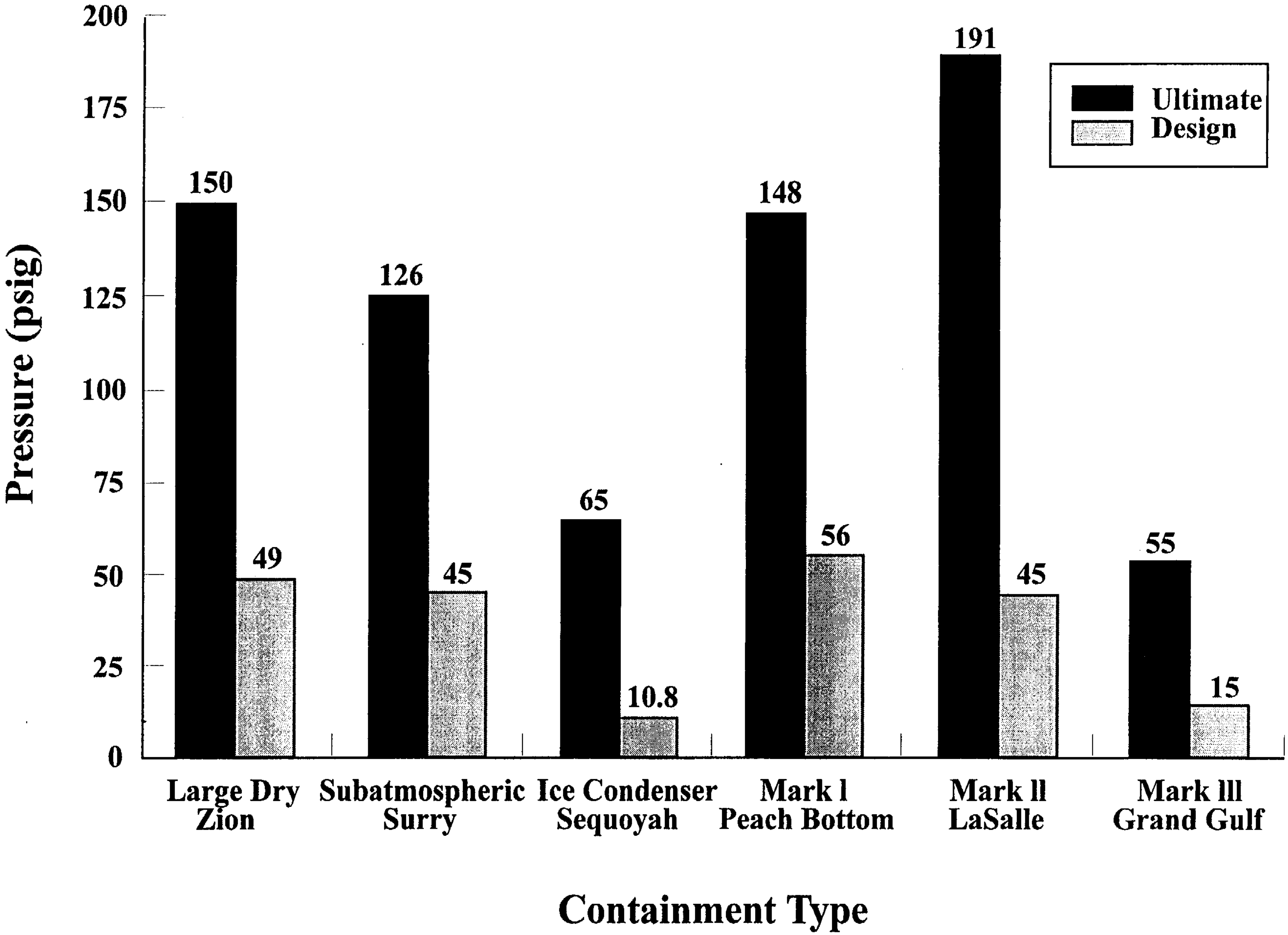


Figure 4.1-2 Comparison of design pressure and ultimate failure pressure

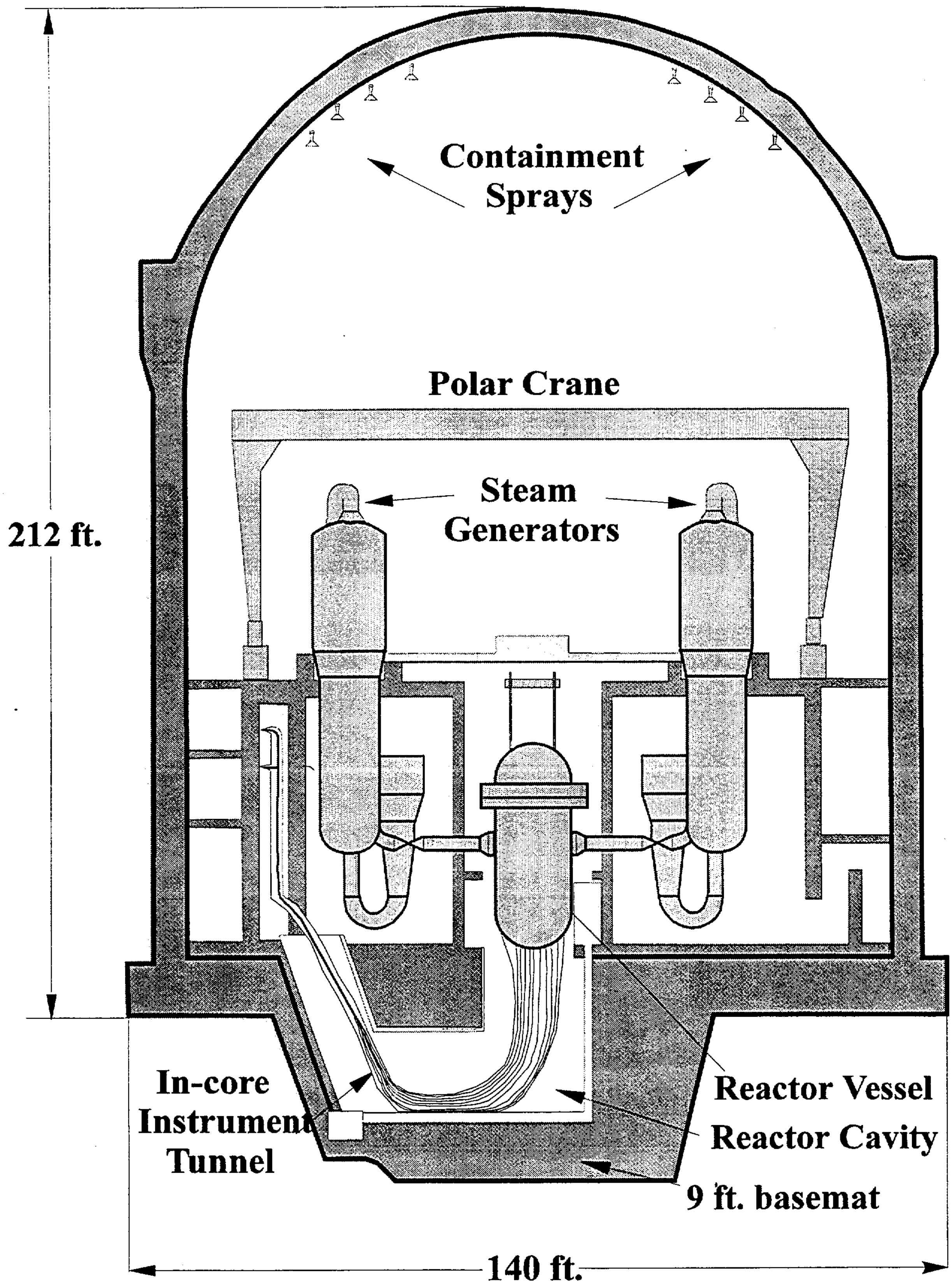


Figure 4.1-3 Typical large dry containment

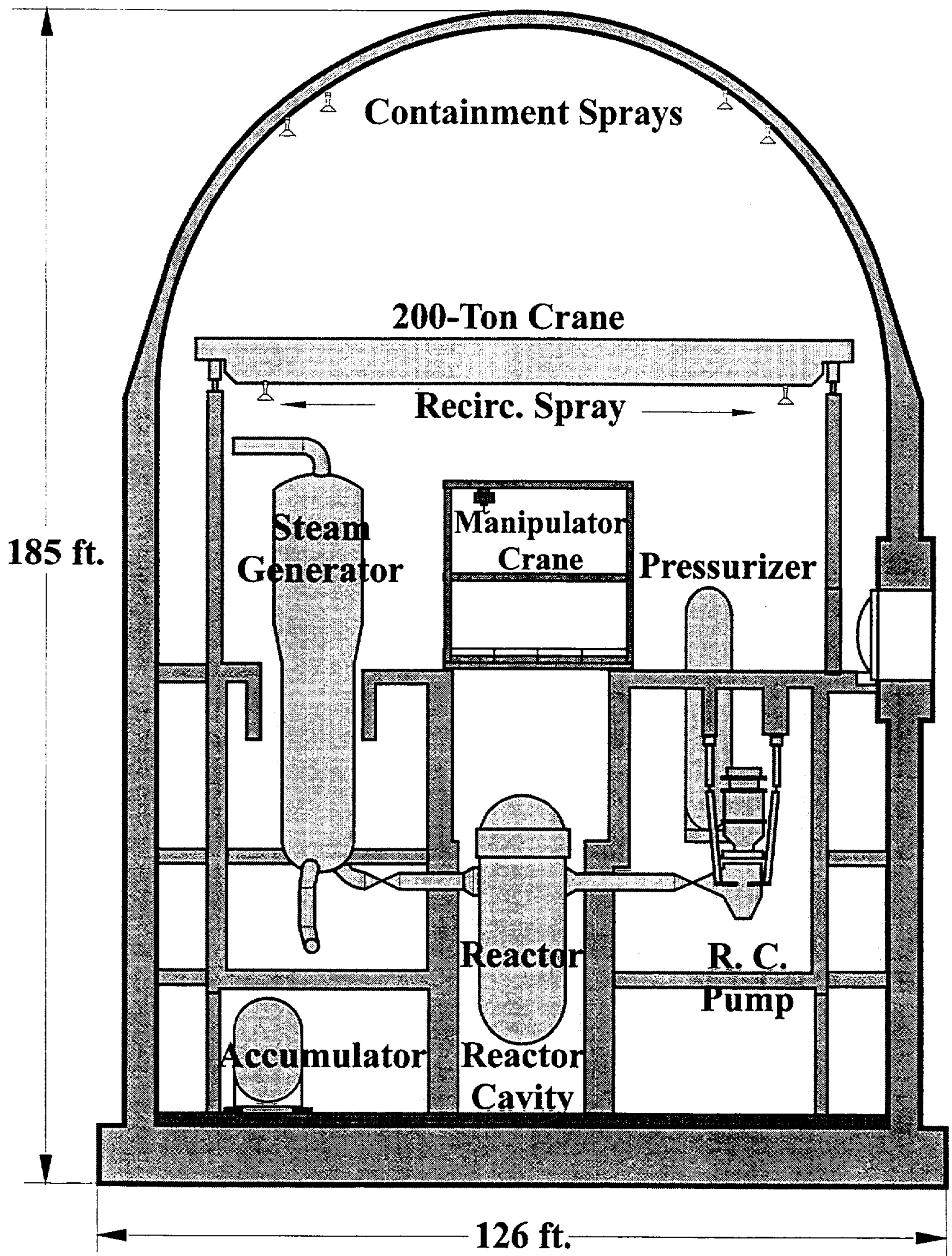


Figure 4.1-4 Typical subatmospheric containment

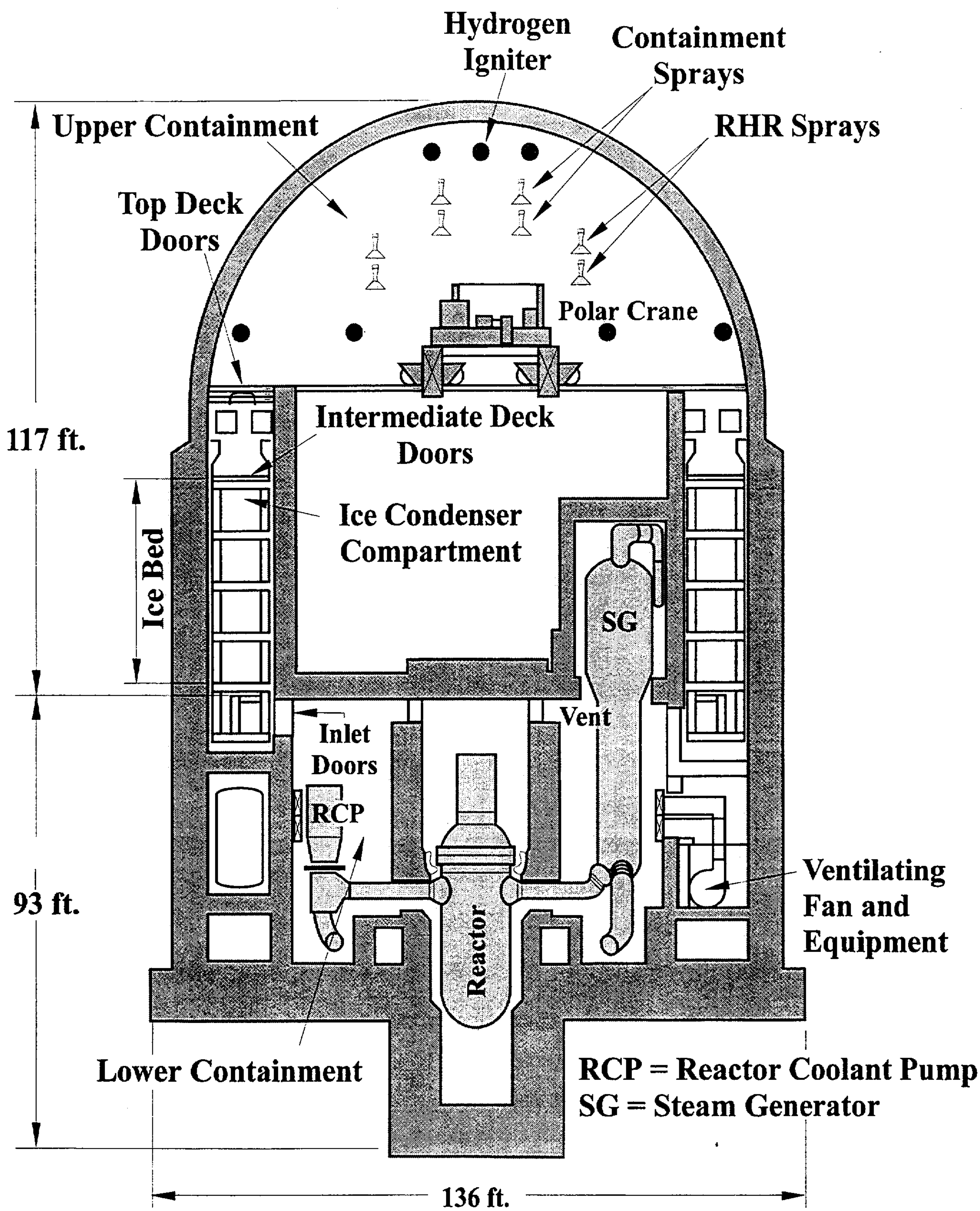


Figure 4.1-5 Typical ice condenser containment

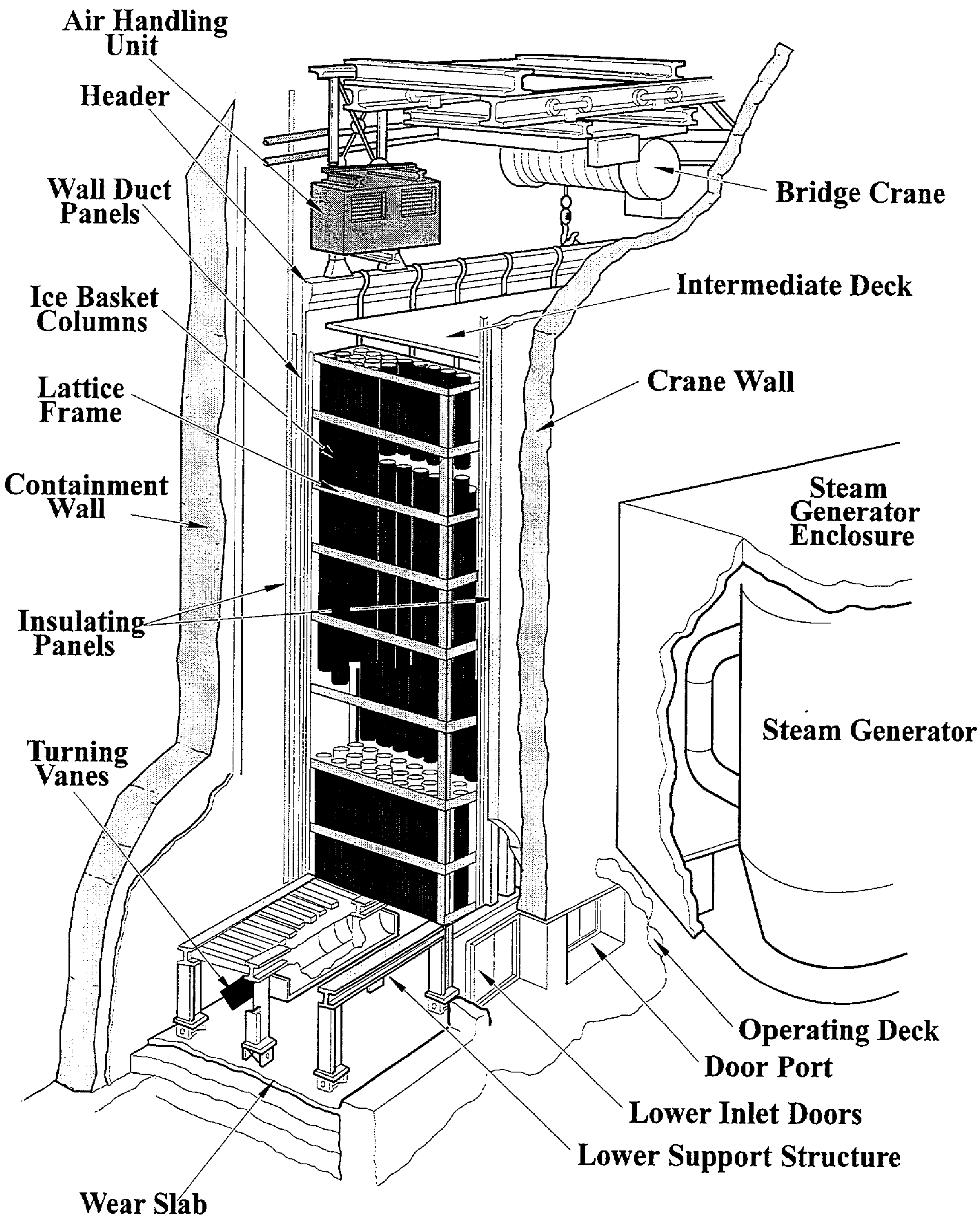


Figure 4.1-6 Ice condenser cutaway

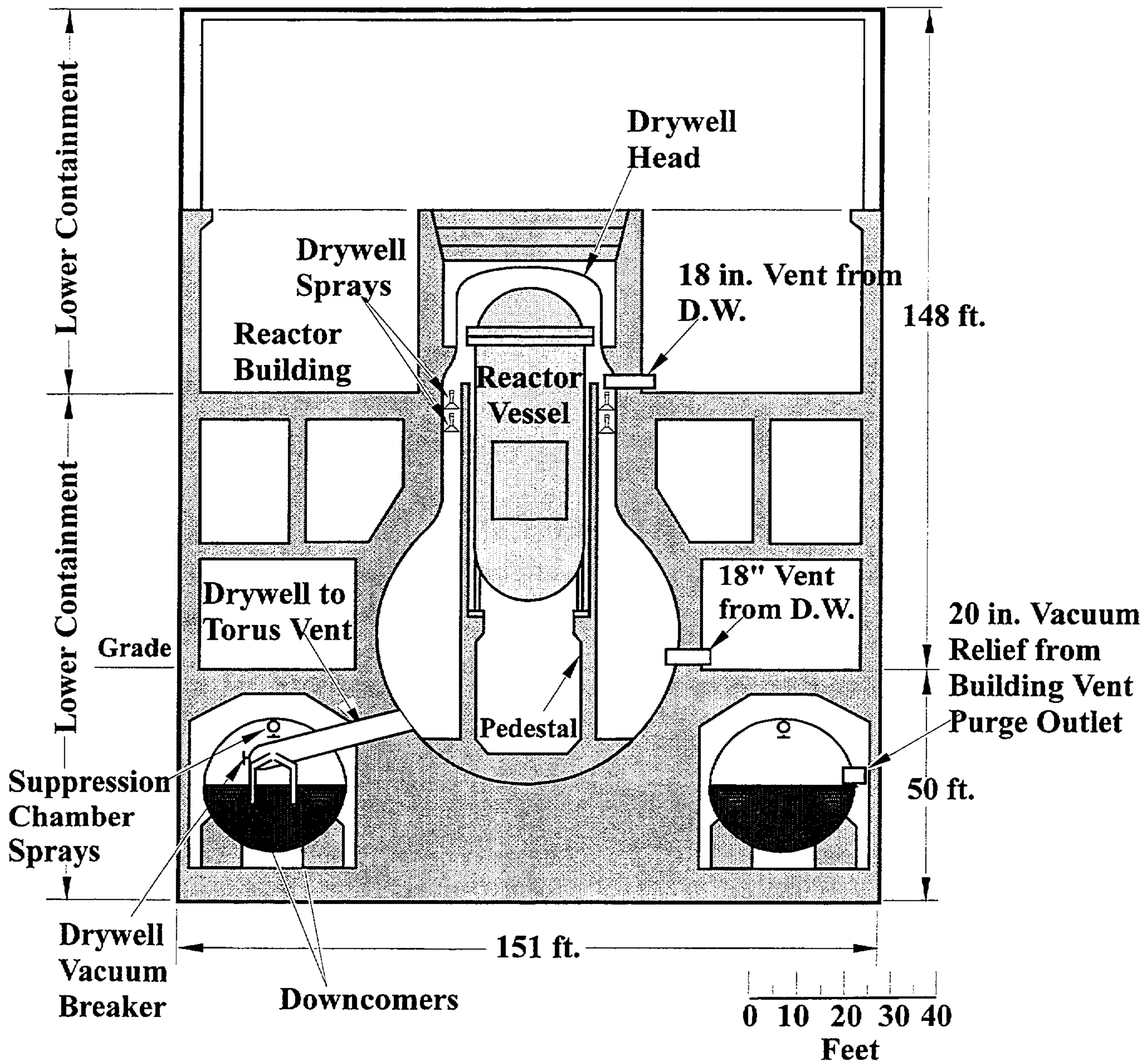


Figure 4.1-7 Typical BWR Mark I containment

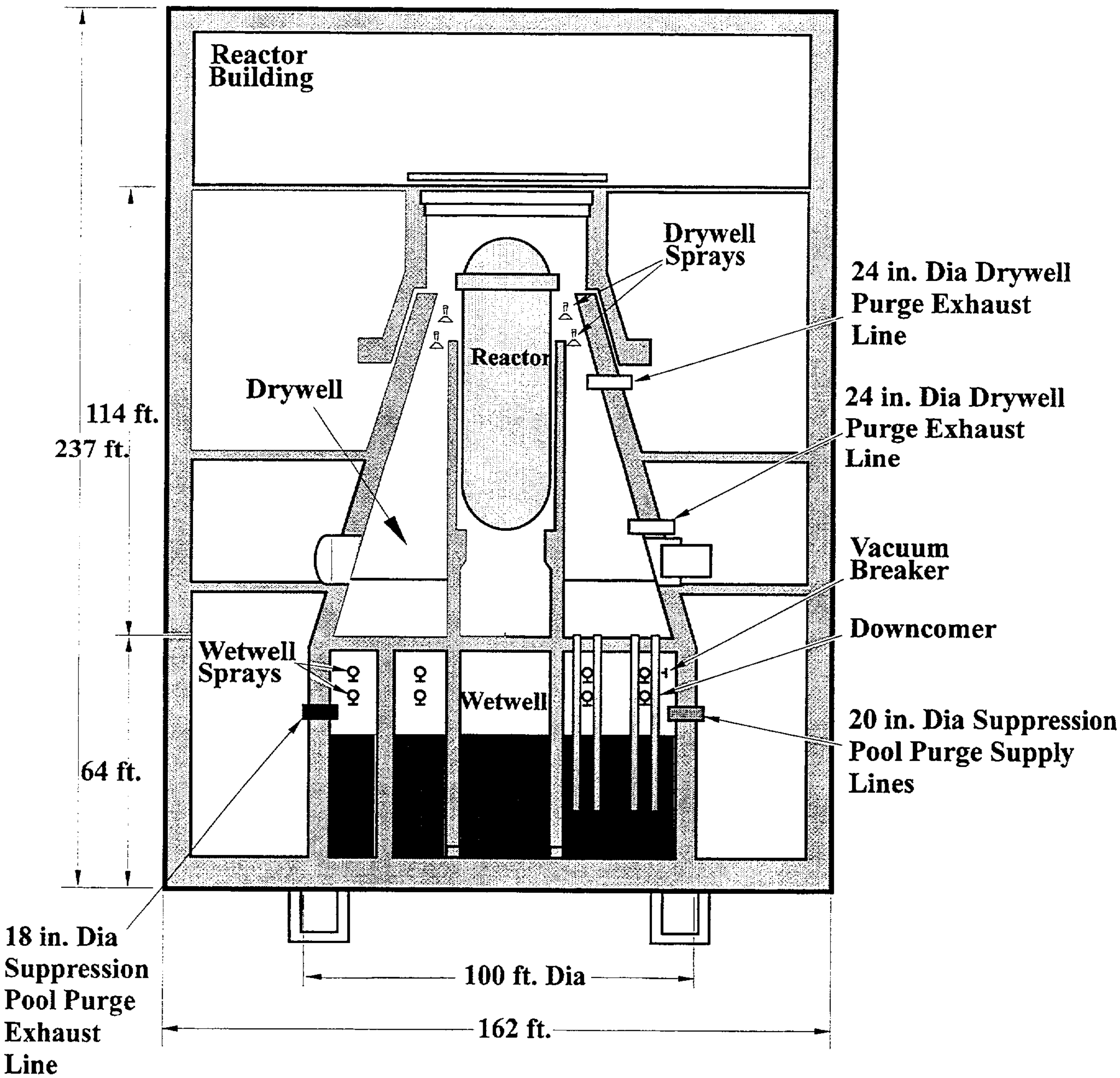


Figure 4.1-8 Typical BWR Mark II containment

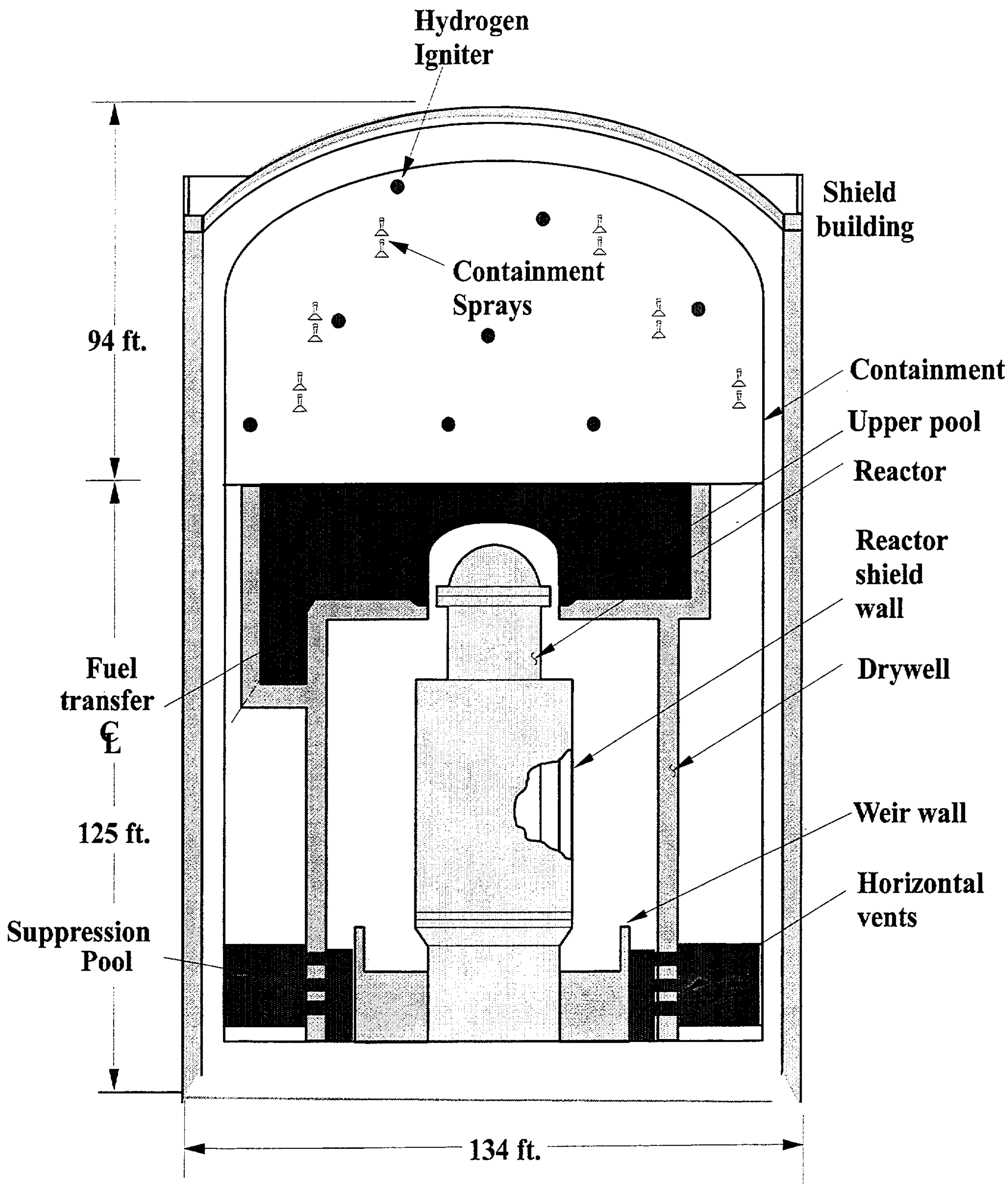


Figure 4.1-9 Typical BWR Mark III containment

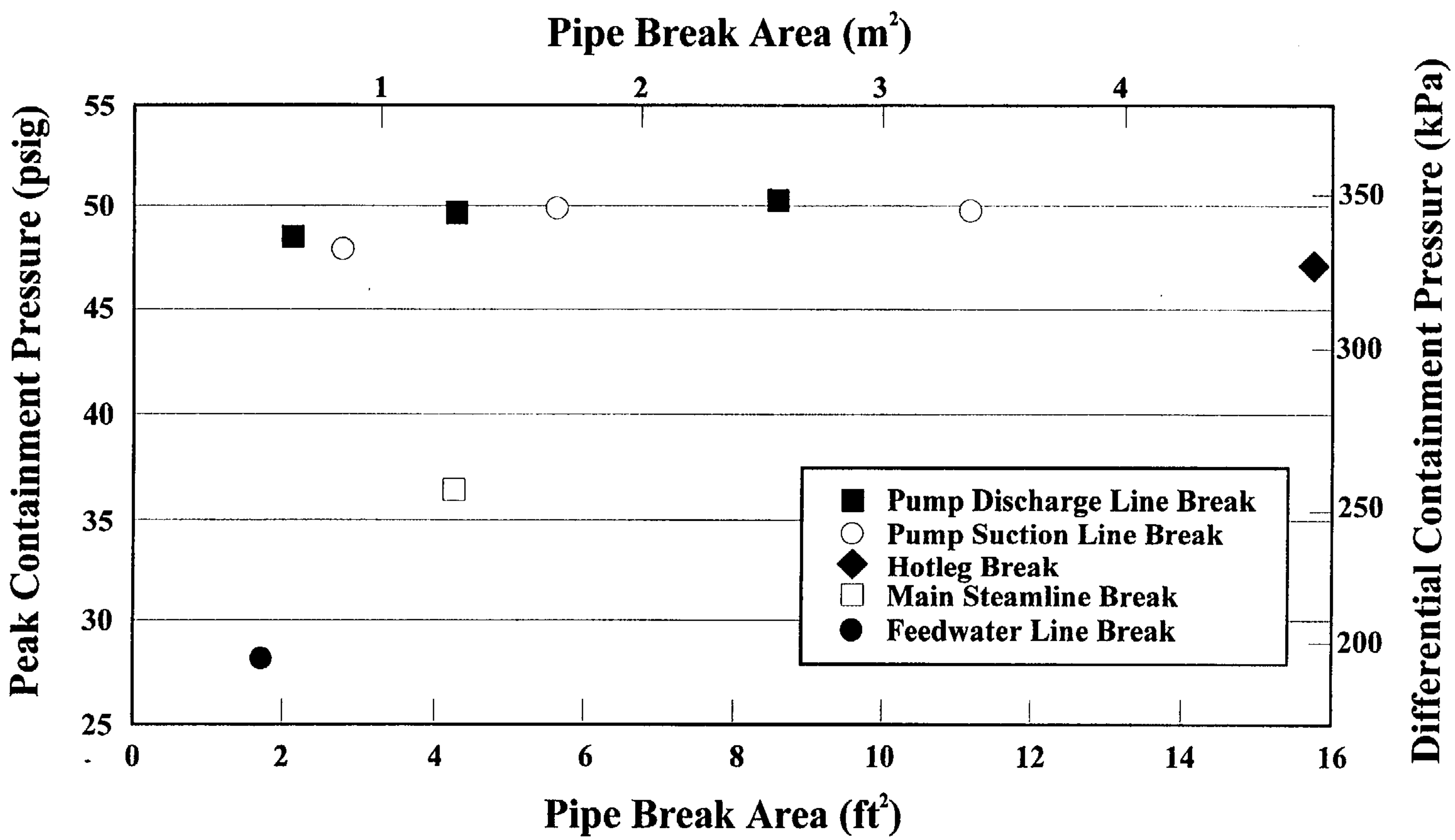


Figure 4.1-10 Peak containment pressure for one PWR

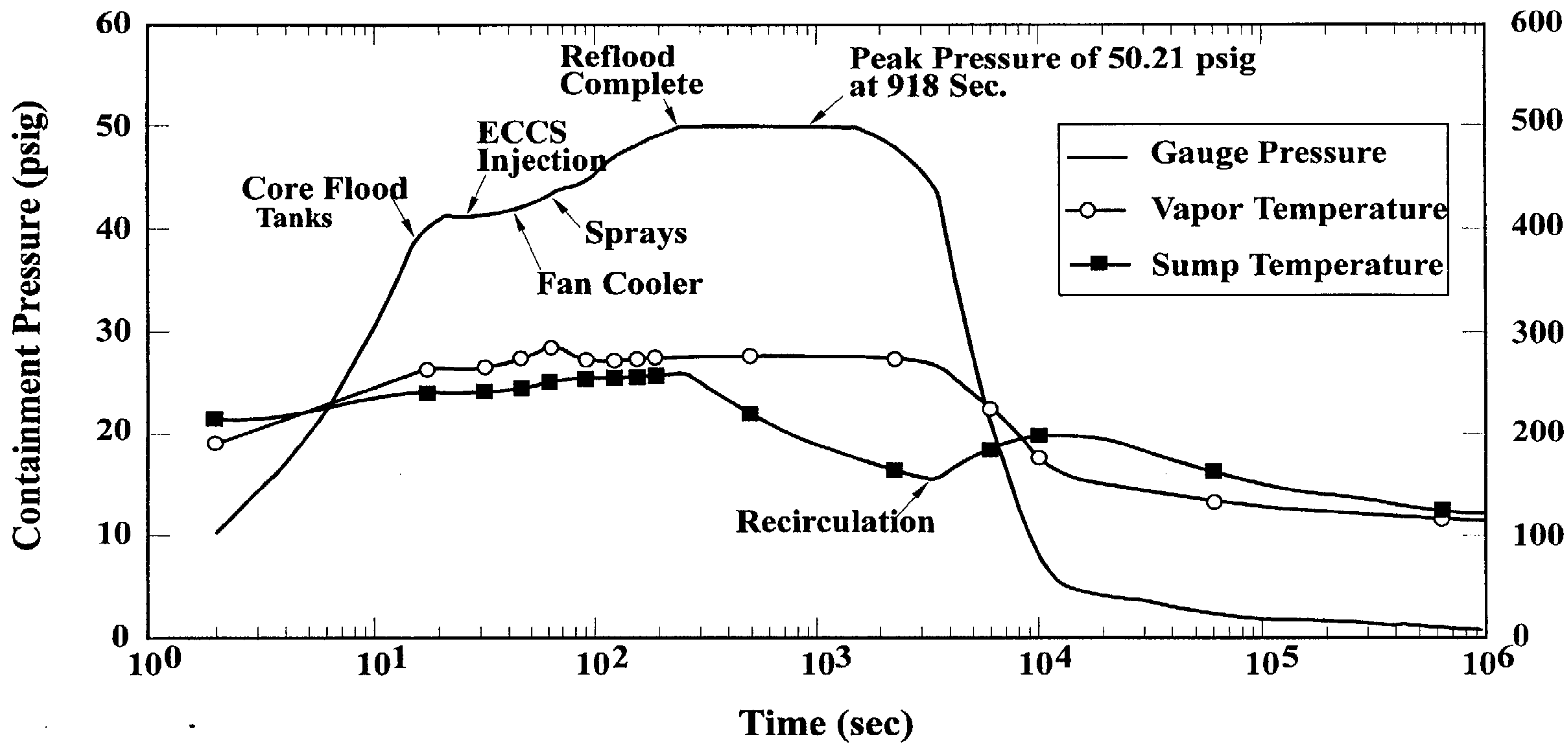


Figure 4.1-11 Containment pressure-temperature response for 8.55 ft² pump discharge break

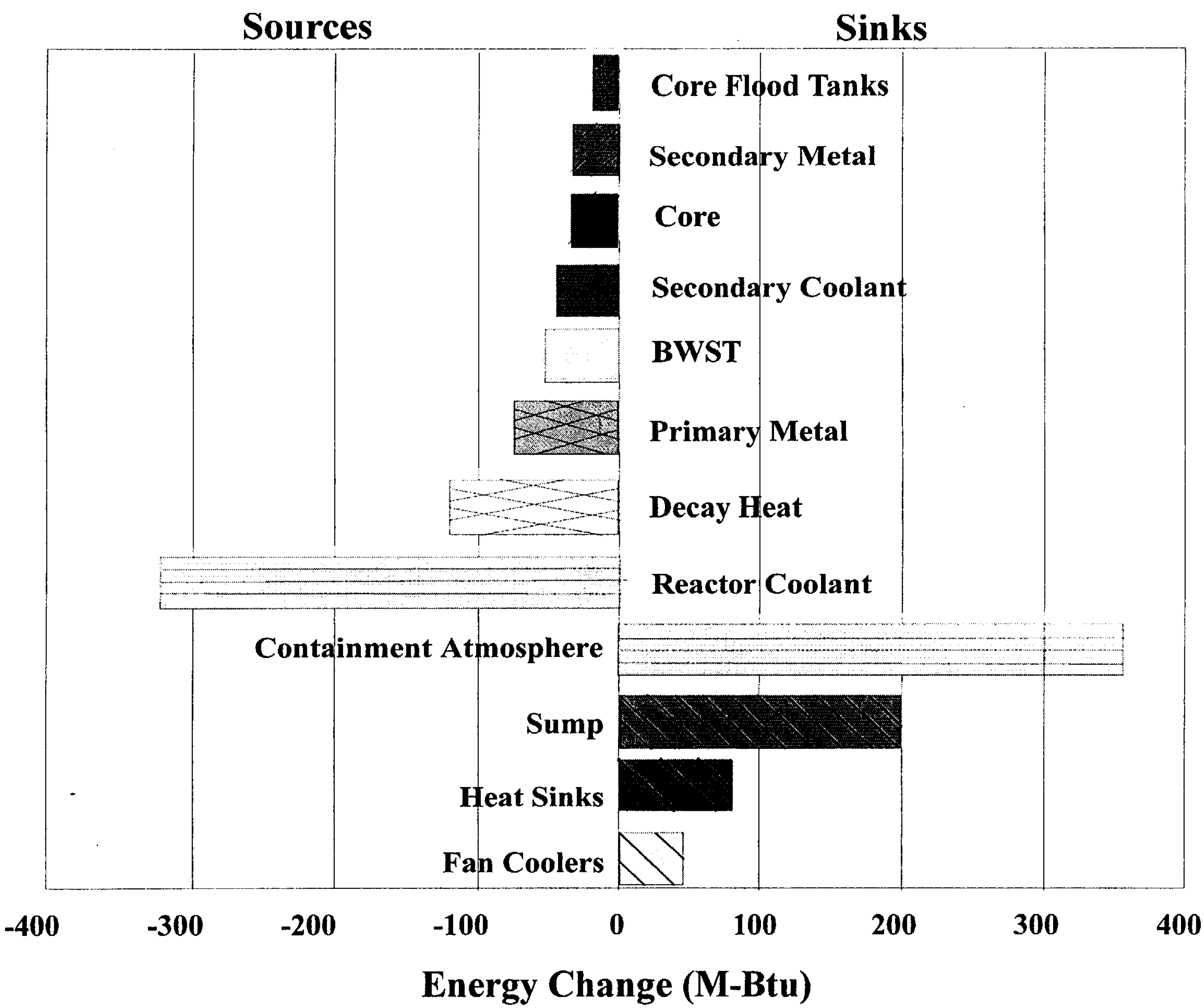


Figure 4.1-12 Energy changes up to time of peak containment pressures for one PWR

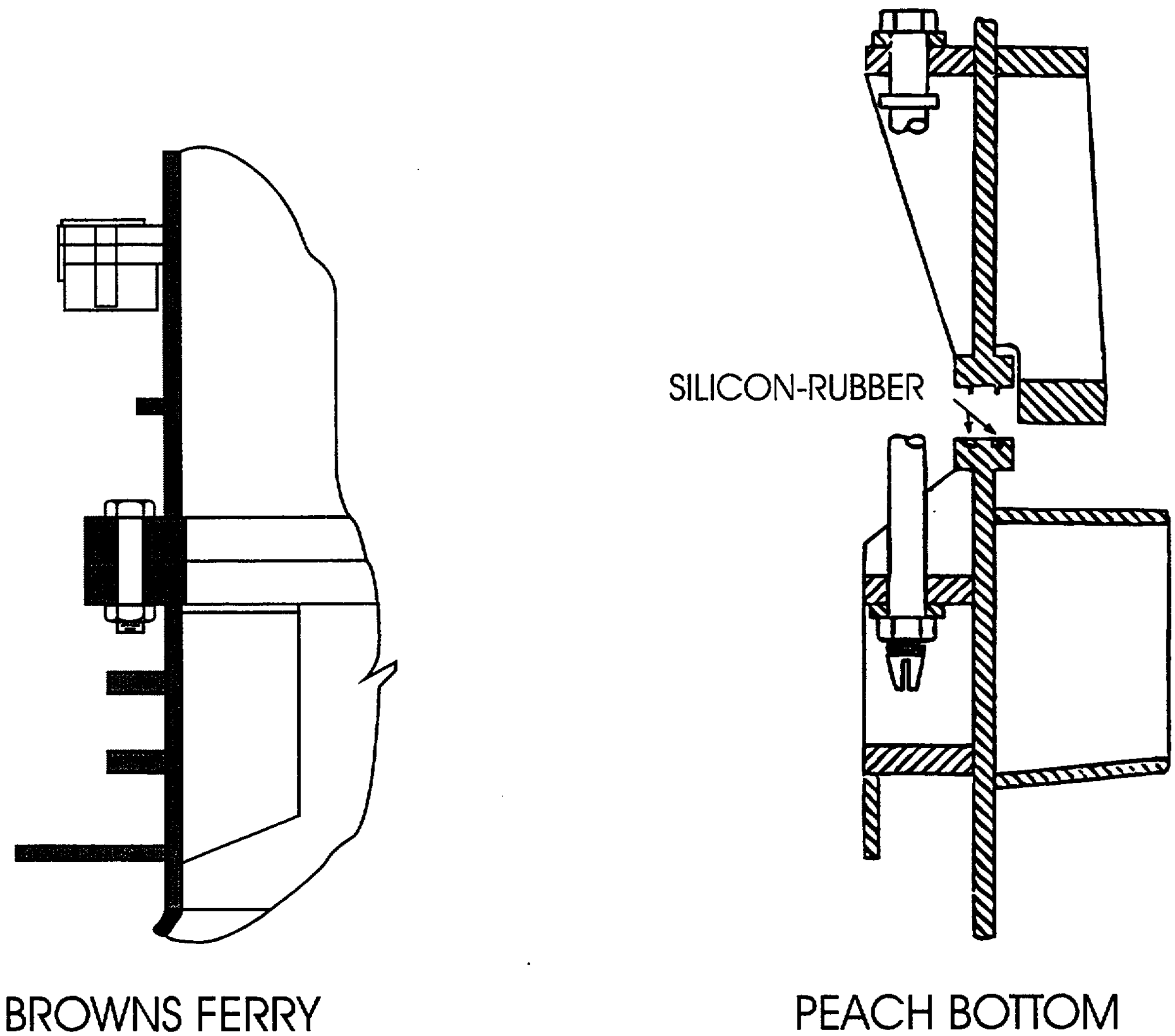


Figure 4.1-13 Different bolting arrangements on drywell head closure flange for Browns Ferry and Peach Bottom

References for Section 4.1

1. U.S. Nuclear Regulatory Commission, World Wide Web Site, http://www.nrc.gov, May 10, 1996.

2. A. L. Camp, et al., "Light Water Reactor Hydrogen Manual," NUREG/CR-2726, SAND82-1137, Sandia National Laboratories, August 1983.

3. U. S. Nuclear Regulatory Commission, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150, December 1990.

4. T. D. Brown, et al., "Integrated Risk Assessment for the LaSalle Unit 2 Nuclear Power Plant: Phenomenology and Risk Uncertainty Evaluation Program (PRUEP)," NUREG/CR-5305, SAND90-2765, Sandia National Laboratories, 1992.

5. *U.S.Code of Federal Regulations,* Title 10, Part 50, Appendix A, General Design Criterion 50, U.S. Government Printing Office, January 1, 1991.

6. Kenneth C. Lish, "Nuclear Power Plant Systems and Equipment," Industrial Press Inc., 200 Madison Avenue, New York, New York, 1972.

7. *U.S. Code of Federal Regulations,* Title 10, Part 100, U.S. Government Printing Office, January 1, 1991.

8. *U.S. Code of Federal Regulations,* Title 10, Part 50, Appendix A, U.S. Government Printing Office, January 1, 1991.

9. *U.S. Code of Federal Regulations,* Title 10, Part 50, Appendix J, U.S. Government Printing Office, January 1, 1991.

10. *U.S. Code of Federal Regulations,* Title 10, Part 50, Appendix A, General Design Criterion 38, U.S. Government Printing Office, January 1, 1991.

11. U.S. Nuclear Regulatory Commission, "Performance-Based Containment Leak-Test Program," NRC Regulatory Guide 1.163, September 1995.

12. *U.S. Code of Federal Regulations*, Title 10, Part 50, Appendix A, General Design Criterion 38, U.S. Government Printing Office, January 1, 1991.

13. R. J. Breeding, et al., "Evaluation of Severe Accident Risks: Quantification of Major Input Parameters," NUREG/CR-4551, Vol. 2, December 1986.

14. D. A. Brinson and G. H. Graves, "Evaluation of Seals for Mechanical Penetrations of Containment Buildings," NUREG/CR-5096, SAND88-7016, Sandia National Laboratories, August 1988.

4.2 Containment Response to Beyond-Design-Basis Accidents

As discussed in Section 4.1, containments are not likely to fail from the loads resulting from design-basis accidents. In fact, there are very large margins between the pressures resulting from design-basis accidents and predicted ultimate failure pressures. However, the China Syndrome and the Reactor Safety Study made it clear that more severe challenges to containment were possible.[1,2] In fact, it appeared that public risk was probably dominated by accidents in which substantial core damage occurred and the containment failed or was bypassed. The TMI-2 accident further emphasized the importance of phenomena, such as hydrogen combustion, that could accompany severe accidents. This section provides some general perspectives on the vulnerabilities of containments to severe accident phenomena. Later sections will describe key severe accident phenomena in more detail.

4.2.1 Containment Challenges and Timing of Events

Challenges to containments can occur during four time regimes:

1. at the start of the accident,
2. prior to reactor vessel failure,
3. at or soon after reactor vessel failure, or
4. long after reactor vessel failure.

Table 4.2-1 summarizes the time regimes and their associated containment challenges. Isolation failures and bypass occurring at the start of the accident were addressed in Section 4.1. The three other time regimes are discussed below.

Prior to vessel failure, there are three types of containment pressure loads that can occur. The first type of load includes the pressure loads that result from the initial reactor coolant system blowdown and subsequent steam and hydrogen releases due to reflooding. For design-basis accidents, these loads are not a threat; however, containments are not designed to withstand the loads that may occur during some severe accidents resulting from the rupture of a reactor vessel or steam generator. As of early 1996, there have been no definitive studies concerning the likelihood of containment failure from such events; fortunately, the frequency of such events is estimated to be very small.

As a related matter, it should be noted that the occurrence of high containment pressure signals does not necessarily mean that a LOCA has occurred. Loss of drywell coolers and the concomitant increase in drywell temperature will cause the drywell pressure in the relatively small Mark I and Mark II containments to exceed the alarm setpoints (about 1.7 psig or 12 kPa) within a few minutes. Although this temperature-induced pressure increase has no implications with respect to the integrity of the drywell boundary, its effect in generating one of the LOCA accident signatures illustrates a potential for operator confusion under conditions such as those accompanying Station Blackout.

A second type of load that can occur prior to vessel breach involves the failure of containment heat removal systems to cope with the ongoing mass and energy additions to the containment even though core cooling is successful. This problem can occur in many ATWS sequences or in LOCAs or transients in which containment heat removal systems fail. In the latter cases, the design pressure may be exceeded early, but the ultimate failure pressure would not be reached for many hours or even days. In fact, some containments may not fail at all, if the heat losses through the structure can eventually match the decreasing decay heat load. If the containment does fail, then there is the potential for the loss of core cooling as a result of several phenomena, including:

1. loss of net positive suction head (NPSH) to pumps that are recirculating water from a heated sump or suppression pool,

2. failure of piping as a result of the containment failure, or

3. failure of core cooling system components located in the reactor building of a Mark I or Mark II BWR when steam enters the surrounding reactor building following containment failure.

If core damage results from one of these phenomena, then the accident will proceed in a containment that is already failed.

The third phenomena that can cause failure prior to vessel breach is hydrogen combustion. Hydrogen will be generated during the core heatup and meltdown phase due to zirconium-steam reactions. If a significant amount of this hydrogen is released through relief valves (as at TMI-2) or through a pipe break, then combustion prior to vessel breach can threaten the containment. Hydrogen combustion is discussed in more detail later in this chapter.

The second time phase of interest, and the one that is often most threatening to containment, is the phase that occurs at or soon after vessel breach. When vessel breach occurs, there are several phenomena that can ensue, sometimes acting simultaneously. Those phenomena include:

1. steam spike,
2. steam explosion,
3. direct containment heating,
4. hydrogen combustion,
5. containment shell meltthrough,
6. downcomer failure (Mark II BWR).

Steam spikes or explosions can occur if there is water in the reactor cavity or pedestal region below the reactor vessel. In-vessel steam explosions and alpha mode failures were addressed in Chapter 3. Water may be present below the vessel as a result of leakage from the reactor coolant system, the operation of containment sprays, or melted ice in an ice condenser containment. By themselves, steam spikes are unlikely to threaten containment, unless the containment is already substantially pressurized. The amount of mass and energy added to the containment atmosphere is determined by the amount of water converted to steam as the melt is quenched in the water. If a steam explosion occurs, then shock waves may cause damage to the containment structure or the vessel supports. If the vessel supports fail and the vessel moves significantly, then containment failure may result around the piping penetrations. In some BWRs, steam explosions could lead to suppression pool bypass, possibly resulting in eventual overpressurization of the containment. Steam explosions are discussed more in Section 4.3.

Direct Containment Heating (DCH) involves the ejection of the melt from the vessel at high pressure, thus spraying the molten material into containment. With the melt broken up into small particles, rapid heat transfer to the containment atmosphere can occur, most likely accompanied by the chemical energy associated with oxidation of metals in the melt. This "direct heating" has the potential to transfer more energy to the containment atmosphere than a steam spike and provides a more significant threat to containment. DCH is discussed more in Section 4.5.

When the reactor vessel fails, any hydrogen contained in the reactor coolant system will be released to containment, and additional hydrogen may be generated as a result of chemical reactions accompanying steam spikes, steam explosions, or direct containment heating. This hydrogen may burn immediately if sufficient oxygen is present, particularly if the molten material provides an ignition source or the hydrogen is already at very high temperatures. Hydrogen combustion at vessel breach may directly threaten containment or may threaten containment in combination with one or more of the other phenomena that can occur.

A phenomenon of importance primarily for Mark I BWRs is shell (liner) meltthrough. At vessel breach, the molten material may flow out of the pedestal region, across the drywell floor and then directly contact the steel liner, causing failure. The likelihood of this event and potential means for its mitigation are discussed in more detail in Section 4.7.

A phenomenon of importance for Mark II BWRs is downcomer failure. While Mark II designs vary significantly, there is often the potential for molten material to flow across the floor and into the downcomers. This molten material may directly fail the downcomer or, possibly, lead to a steam explosion that fails the downcomer. Downcomer failure does not lead to immediate containment failure; however, the suppression pool is bypassed, thus negating its heat removal and fission product scrubbing capabilities.

The third time phase of interest is the late phase, hours or more after vessel failure. The late phase threats consist primarily of high temperature, overpressure, basemat meltthrough, and hydrogen burns. High temperature and long term overpressure can result if containment heat removal systems are inoperative. In a BWR, high drywell temperatures can result even if the suppression pool cooling systems are working. With most of the core materials now present in the containment, the decay heat must be removed somehow to prevent temperature and pressure buildup. High temperatures can result in weakened structures that may leak more than expected or fail at pressures lower than the expected ultimate failure pressure. The problem is exacerbated by noncondensible gases that can be generated by core-concrete interactions. These noncondensible gases contribute to the overall pressure.

Basemat meltthrough is a long term result of core-concrete interactions. These interactions can generate hydrogen and other non-condensible gases, generate copious amounts of radioactive and nonradioactive aerosols, and eventually fail the basemat. Core-concrete interactions will be discussed in more detail in Section 4.4.

Hydrogen burns can also occur during the late phase. In some cases this may involve hydrogen that was present previously, but did not burn due to the lack of an ignition source or an excess of steam in the atmosphere. If steam is removed late in an accident, for example, due to recovery of sprays, a gaseous mixture that was inert may become flammable. Another factor affecting hydrogen burns is the amount of flammable gases (hydrogen and carbon monoxide) being generated from core-concrete interactions. These additional gases can lead to burning late in an accident.

Section 4.2.1 has summarized the time phases of an accident and the phenomena that occur during those phases. Section 4.2.2 will now discuss estimates of containment failure probabilities as a result of those particular phenomena.

4.2.2 Implications of Containment Failure

The significance of containment failure depends upon the particular accident sequence, the mode of containment failure and the timing of radioactive releases. Chapter 5 addresses the importance of the timing of releases relative to warning times and evacuation speeds. The importance of accident sequence type and containment failure mechanisms are discussed briefly below.

Containment failure can only represent a significant concern if radionuclides are released from the fuel and the reactor coolant system. If fuel melting does not occur and only the activity in the reactor coolant and the radioactive gases in the fuel pins (gap release) are released, then the consequences will be minimal even if containment failure occurs.

If fuel melting does occur and a significant amount of radionuclides is released to containment, then the timing and mode of containment failure are critical factors in determining the offsite consequences. Generally, the most severe failure modes are ones that occur early in time (before or during reactor vessel failure) so that there is little settling or other retention of radionuclides in the containment. Radionuclides can be retained in containment in a number of ways:

1. scrubbing in suppression pools,

2. scrubbing by containment sprays,

3. retention in an ice condenser,

4. gravitational settling and other natural processes,

5. trapping along tortuous release paths.

Most of these retention mechanisms are affected by the time available for the mechanism to work. Small containment leaks allow more time for settling and scrubbing by sprays. Therefore, ruptures are more likely to lead to severe consequences than leaks. If the radionuclides can be mostly retained until after evacuation occurs, then many of the health effects can be substantially reduced. Also, failures that lead into surrounding buildings allow further opportunities for retention.

Chapter 5 will discuss the offsite consequences of particular accident types in more detail. However, the importance of containment failure can be summarized by stating that the worst failures are failures (or bypasses) that occur early and allow rapid, unscrubbed transit of radionuclides out of the containment.

4.2.3 Likelihood of Containment Failure During Severe Accidents

The most comprehensive study of containment failure probabilities is contained in the NUREG-1150 documents.[3] In addition, the industry has performed individual plant examinations (IPEs) assessing the performance of containments in severe accidents. Seventy-five IPE submittals covering 108 units are included in the discussions below.[4] Despite the fact that severe accidents provide challenges beyond the design-basis, NUREG-1150 (and the IPE studies) show that containments have the capacity to withstand many of these accidents. This capability is a result of the very conservative design process that provides substantial margin with respect to less severe design-basis accidents.

The likelihood of actual containment failure is usually considered according to one of two measures: conditional containment failure probability (CCFP) and containment failure frequency (CFF). Both of these measures depend upon several factors, including the particular containment design and accident sequence. The conditional containment failure probability is the probability of containment failure given an accident. The containment failure frequency is the frequency per reactor year of accidents involving containment failure. These quantities are determined from:

$$CCFP = \sum_{i=1}^{n} \frac{S_i}{CDF} C_i$$

$$CFF = \sum_{i=1}^{n} S_i C_i$$

where

CCFP is the conditional containment failure probability,

CFF is the containment failure frequency,

CDF is the total core damage frequency,

S_i is the frequency of accident sequence i,

C_i is the conditional probability of containment failure given accident sequence i,

and

n is the total number of accident sequences.

Because S_i and C_i depend on the particular accident sequences (which vary considerably among the plants), both CCFP and CFF can be significantly different for two plants with identical containments.

Most of the time, values for CCFP and CFF are calculated for early failures and bypass events. Failures happening many hours after core damage normally contribute much less to public risk. Figure 4.2-1 shows the relative probability of different containment failure modes (CCFP), given a core damage accident, for the five plants evaluated in NUREG-1150. In this figure, early failures include failures that occur before, at, or soon after vessel breach. Note that many of the outcomes at Grand Gulf, which has a Mark III containment, involve failure of the outer containment with the drywell and suppression pool remaining intact. Therefore, the containment failures for Grand Gulf do not all lead to significant radiological releases.

With the caveat noted above for Grand Gulf, the failures that most impact public risk are the early failures and the bypass events. Figure 4.2-2 shows the CFFs of such events for the five NUREG-1150 plants and the IPEs. Figure 4.2-3 shows the CCFP for the same plants. These figures, which consider only internally initiated accidents (the IPEs include internal flooding), account for the variation in accident frequency and type. As noted in Chapter 2, Grand Gulf has a substantially lower core damage frequency than Sequoyah, and this is reflected in a lower containment failure frequency, even though Grand Gulf has a higher probability of early failure given an accident (CCFP).

Because of the risk importance of early releases, the phenomena, mechanism, and accident scenarios that can lead to such releases are of particular interest. These involve early structural failure of the containment, containment bypass, containment isolation failures and, for some BWR plants, deliberate venting of the containment.

As a group, the large dry PWR containments analyzed in NUREG-1150 and the IPEs have significantly smaller conditional probabilities of early structural failure (given core melt) than the BWR pressure suppression containments analyzed. Nonetheless, containment bypass and isolation failures are generally more significant for the PWR containments. As seen in Figure 4.2-2, however, these general trends are often not true for individual IPEs because of the considerable range in the results. For instance, conditional probabilities for both early and late containment failure for a number of large dry PWR containments were higher than those reported for some of the BWR pressure suppression containments.

The results for BWRs, grouped by containment type, follow expected trends and indicate that, in general, Mark I containments are more likely to fail during a severe accident than the later Mark II and Mark III designs. However, the ranges of predicted failure probabilities are quite high for all BWR containment designs and there is significant overlap of the results, given core damage. A large variability also exists in the contributions of the different failure modes for each BWR containment group. However, plants in all three BWR containment groups found a significant probability of early or late structural failure, given core damage.

The Commission has previously considered a subsidiary safety goal involving the frequency of containment failures that are accompanied by large releases of radioactivity. The goal was tentatively set at 10^{-6} per reactor year. However, the commission abandoned this goal due to difficulties in achieving consensus on an appropriate definition of a large release. An alternate subsidiary safety goal of an average CCFP of 0.1 is still being considered. It is not clear whether the CCFP goal should include all containment failures or only early failures and bypass. Figure 4.2-3 shows that few plants meet this goal if all failures are considered, and most of the BWRs do not meet the goal even for early failures and bypass. No new specific actions to deal with BWR containments, based on IPE findings, were being planned as of mid 1997. Low core-damage frequencies for BWRs provide partial justification for a lack of action.

More recently, the staff has been considering a return to a large release goal. As discussed in Section 2.6, a large early release frequency of 10^{-5} per year has been proposed as part of the risk-informed regulation initiative. Note that, if this approach is implemented, there will not be a direct measure of containment performance as part of the decision criteria. That is, a plant could meet the numerical goal with a low core-damage frequency and a poor containment. However, qualitative consideration of defense-in-depth is intended to assure that containment performance is not neglected entirely.

4.2.4 Containment Venting Strategies

Containments are somewhat unusual in that they are pressure vessels without safety relief valves. Thus, if containment heat removal is lost, there is no designed-in feature to prevent structural failure. Most containments have penetrations that could conceivably be used to vent the containment and relieve pressure. These penetrations include the lines used for leak rate testing, among others. However, most plants do not have procedures for venting during an accident. There are several reasons for this, including the belief that it is unnecessary, the requirements for AC power for valves, the desire to avoid guaranteed release of radioactive materials, and the potential hazards to personnel involved in the venting process.

Recently, utilities with BWR Mark I and II containments have included venting in their emergency procedures. Venting can be particularly valuable for accident sequences involving the long-term loss of containment heat removal in Mark I and II BWRs. In these sequences, often referred to as TW sequences, core cooling is initially successful. However, the loss of containment heat removal leads ultimately to containment failure. After containment failure, the core cooling systems may fail as a result of the loss of net positive suction head or from the harsh environments due to steam in the reactor building. In some cases, core cooling may fail even before the containment fails. For some BWR plants, high containment pressure can cause the Automatic Depressurization System (ADS) valves to close, leading to the loss of low pressure injection systems. In others, the reactor core isolation cooling system will fail due to high turbine exhaust backpressure. Venting can prevent these problems.

The particular venting procedures vary widely from plant to plant, but include use of leak rate testing lines and lines to the standby gas treatment systems. These plants generally have several possible lines that can be used, ranging in size from two inches to two feet in diameter. Generally, the venting is effective only for long-term loss of containment heat removal sequences. Venting can not occur fast enough to relieve pressure rises from energetic events, such as steam explosions or hydrogen burns. Venting is generally not possible during station blackout, due to the requirements for AC power to open the vents and is not adequate to handle the steaming rate from an Anticipated Transients Without Scram (ATWS) event.

As discussed in Section 4.1, vent lines from the containment are available in Mark I and II BWRs. Venting is possible from either the wetwell or drywell; however, venting from the wetwell is advantageous, because any radionuclide releases can still be scrubbed through the suppression pool. Thus, such venting is more attractive for BWRs than for other designs. A possible negative effect is that venting may lead to a saturated suppression pool, causing loss of net positive suction head to some pumps. At some plants, the procedures call for cycling the vent valves to prevent this loss of net positive suction head.

At some plants venting occurs through strong piping. However, in others the venting may involve ductwork and relatively weak gas flow paths. If venting occurs at high containment pressure, this ductwork will fail, releasing steam and possibly hydrogen and noble gases into the reactor building. These gases may lead to failure of safety equipment in the reactor building and exacerbate the accident. As a result of these concerns, the NRC has reached agreement with owners of Mark I containments to develop procedures for venting only through hardened piping to alleviate this concern.[5]

A final note concerns venting as it relates to emergency response. Current procedures for venting do not attempt to coordinate venting strategies with orders to evacuate. Venting at the wrong time, particularly from the drywell, could conceivably lead to significant releases at the time when the public is moving out onto the roads and is most vulnerable. However, when appropriately used, venting can be an effective measure to release gases from containment via the most desirable pathway if releases are inevitable. Stated another way, early releases of steam, hydrogen, and heated atmosphere and perhaps fission product noble gases may very well preclude a later uncontrolled and high pressure-driven release of atmosphere charged with radioactive aerosols.

The remaining sections in Chapter 4 discuss some of the specific phenomena that can challenge containments during a severe accident.

Table 4.2-1 Containment threats according to time regime

Time Regime	Challenge
Start of the Accident	Pre-existing Leak Containment Isolation Failure Containment Bypass
Prior to Vessel Breach	Reactor Coolant System Blowdown Insufficient Containment Heat Removal Hydrogen Combustion Delayed Bypass
At or Soon After Vessel Breach	Steam Spike Steam Explosion Combustion Direct Containment Heating Debris Contact with Containment
Late (> 2 Hours After Vessel Breach)	Failure of Containment Heat Removal Combustion Non-condensible gas generation Basemat Meltthrough

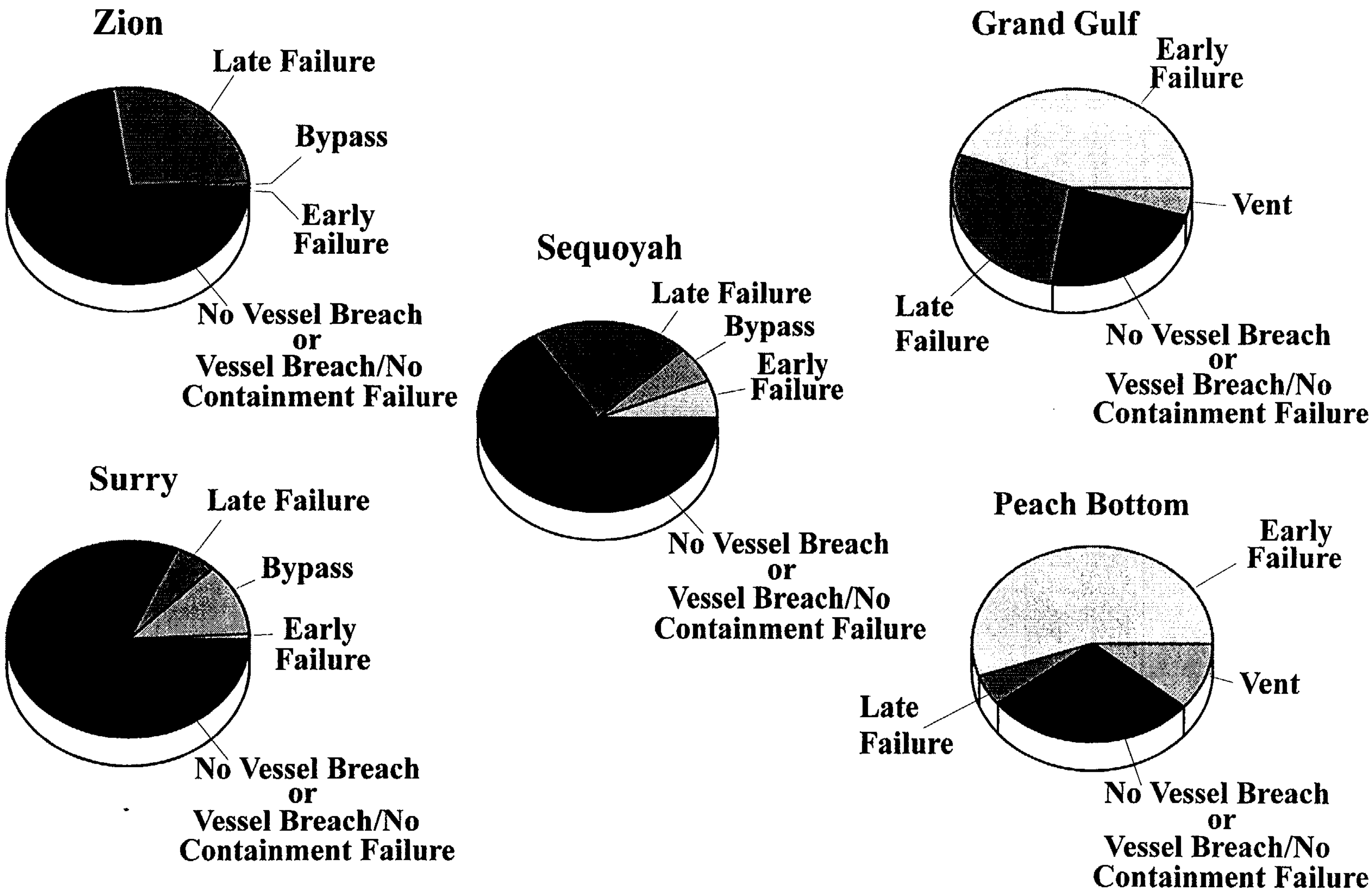


Figure 4.2-1 Relative probability of containment failure modes (internal events from NUREG-1150) given core damage

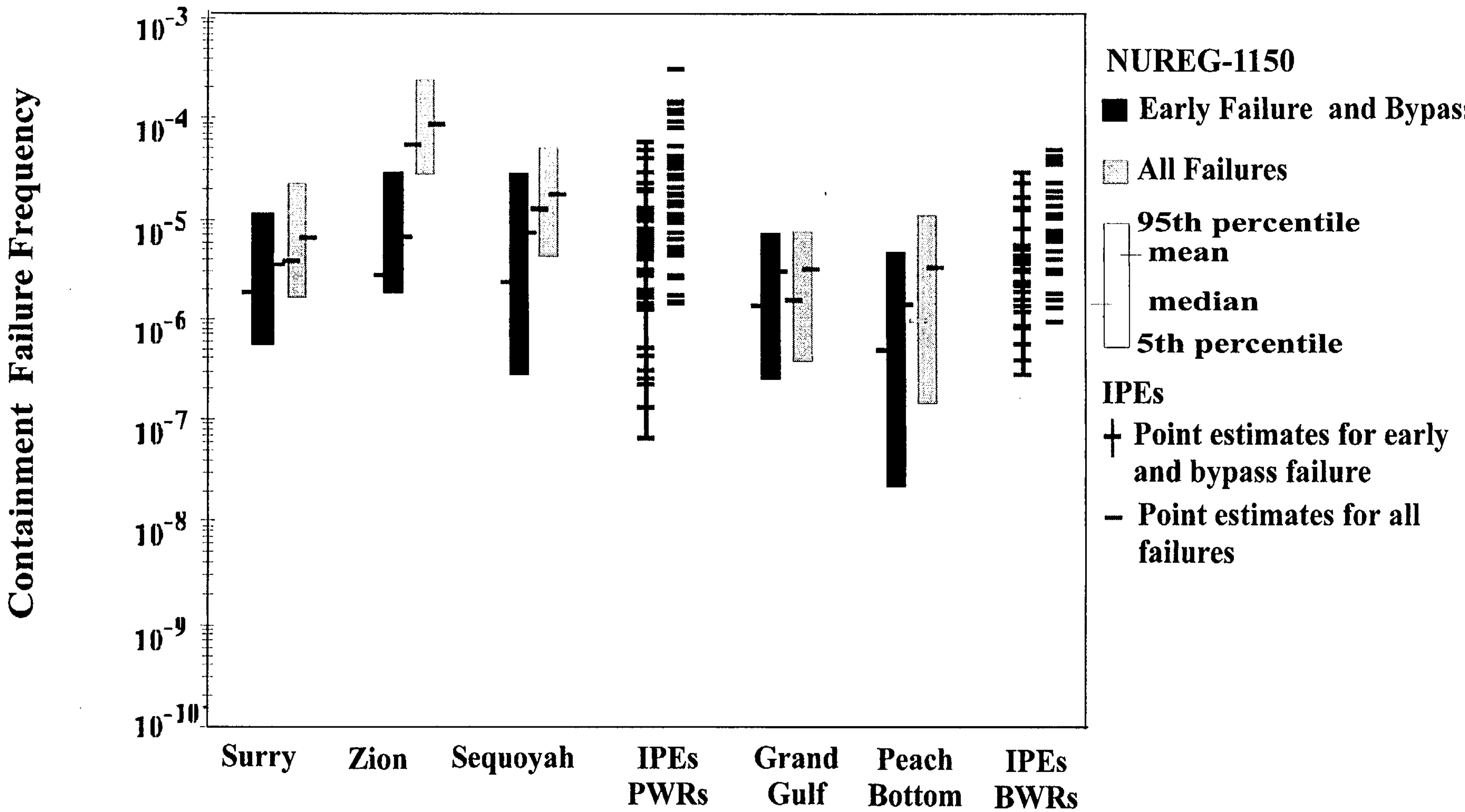


Figure 4.2-2 Containment failure frequency

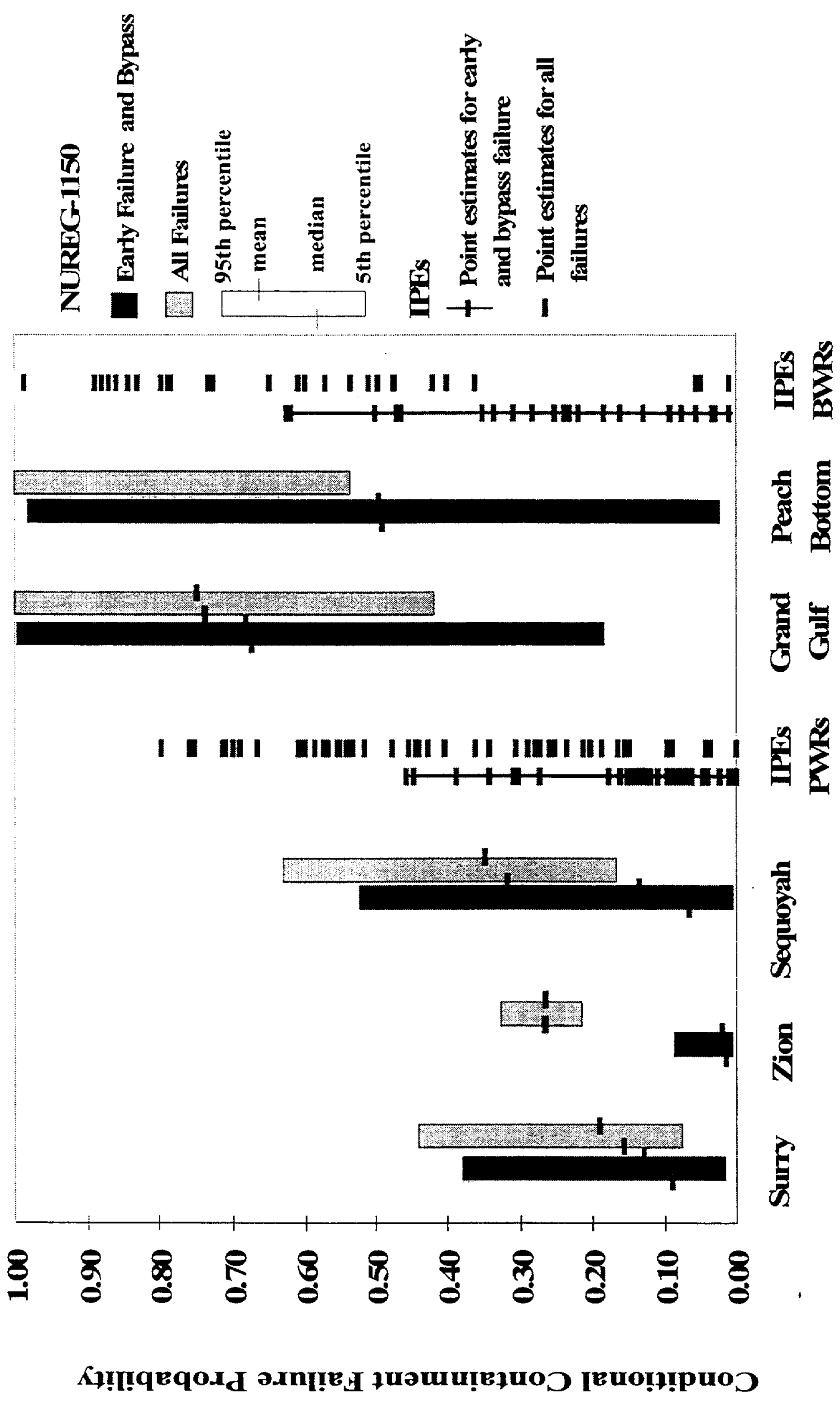


Figure 4.2-3 Conditional containment failure probability (internal events)

References for Section 4.2

1. David Okrent, "Nuclear Reactor Safety: On the History of the Regulatory Process," The University of Wisconsin Press, Madison, Wisconsin, pp. 99-101, 1981.

2. U.S. Nuclear Regulatory Commission, "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400, October 1975.

3. U.S. Nuclear Regulatory Commission, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150, December 1990.

4. U.S. Nuclear Regulatory Commission, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," NUREG-1560, Part 1, Volume 1, Draft for Comment, December 1997.

5. Generic letter 89-16, *Installation of a Hardened Wetwell Vent*, September 1, 1989.

4.3 Ex-Vessel Fuel-Coolant Interactions

For some plants and particular accident situations, water will be present underneath the reactor vessel when the molten material exits the vessel at the time of failure. In other cases, water may be added on top of the molten material subsequent to vessel failure. It is generally considered axiomatic that water addition is always a good thing in a reactor accident. While current guidance to operators is always to add water, it is important to note that there are several different possible outcomes when molten core debris contacts water, and only some of these outcomes are desirable:

1. The water may act to cool and quench (refreeze) the molten core debris, and may limit the spread of molten core across the containment floor.

2. The debris may form a molten pool under the water, probably with an overlying crust layer, and remain molten.

3. An energetic fuel-coolant interaction may occur.

Each of these possibilities is discussed in more detail below.

4.3.1 Quenching of Core Debris

Quenching and continued cooling of the core debris is generally the most desirable outcome. When the debris is solidified, the release of radioactive materials from the debris is effectively terminated. The most significant detrimental effect of quenching is the generation of large quantities of steam, which causes a pressure spike in the containment atmosphere. For the most part, a steam spike will not directly be a threat to the containment unless other phenomena occur simultaneously or the containment is already pressurized significantly prior to the steam spike.

Figure 4.3-1 depicts the quenching process for a PWR. The process involves energy transfer from the molten core debris to liquid water. The molten debris gives up latent heat of fusion plus sensible heat in cooling down to a near-equilibrium temperature. Oxidation energy will be involved if there are unoxidized metals present in the melt. The energy transferred to the water will heat the water to saturation and produce boiling sufficient to the available energy. The steam generated will then enter the containment atmosphere, causing a pressure increase. The speed of the quenching process depends upon how well the molten core mixes with water, the debris particle sizes, and the geometry of the mixture. The quenching process may be very rapid or take many minutes, depending upon these factors.

A calculation was performed for a station blackout sequence in the Zion large dry containment, considering the complete and rapid quenching of an entire molten core, along with 30% oxidation of the available metals.[1] This quenching process would yield approximately 268 Million Btu (283,000 MJ) of energy, and would produce a pressure spike of about 35 psig (240 kPa). Figure 4.3-2 shows the pressure in the Zion containment that could result from this accident sequence, assuming that the entire core is dropped into a reactor cavity full of water at about 14,000 seconds. The total containment pressure approaches 90 psig (620 kPa) as a result of the combined effects of prepressurization prior to vessel breach, vessel blowdown at vessel breach, and the 35 psi (241 kPa) pressure rise resulting from the quenching in the reactor cavity. Two different quenching times are shown in Figure 4.3-2, corresponding to one minute and one hour. Without operating containment heat removal systems, the two different times produce similar containment pressure rises. The longer time available for heat transfer to structures is somewhat offset by the continued addition of decay heat.

In reality, quenching the debris will usually result in pressures much less than those indicated in Figure 4.3-2. First, it is extremely unlikely that all of the core debris will be involved in one large steam spike. Most models of accident progression indicate that a significant fraction of the core will remain in the vessel and be released slowly over a long time period. Second, there must be sufficient water available to participate in the quenching process. In the example shown, there was a completely full reactor cavity. Even if sufficient water is initially present, some of the water may be blown out of the reactor cavity before it can contact the core debris, possibly resulting in debris that is not quenched.

Subatmospheric containments will respond to steam spikes in much the same manner as large dry containments. There is general agreement that other containment types are even less susceptible to steam spikes due to their pressure suppression design.[1] While not designed specifically for steam spikes at vessel breach, suppression pools and ice condensers can readily handle such loads, provided that the water or ice has not been depleted prior to the event. Note that, after the debris quenches, a continuing water supply and long-term heat removal are still necessary in most cases to remove the decay heat that can gradually pressurize the containment.

4.3.2 Non-Coolable Debris

There are some cases in which core debris may not quench or, if quenched, may subsequently form a rubble bed that is non-coolable. Cooling of core debris requires that the debris remain in contact with water, to allow boiling heat transfer to carry away the decay heat. Two mechanisms that can prevent this contact are debris bed dryout and crust formation. As discussed in Chapter 3, the vapor that flows up out of the debris bed can provide resistance to overlying and surrounding water that is needed to permeate the debris bed. If the resistance to water is sufficient, parts of the bed may dry out, leading to continued melting and possible core-concrete attack. Figure 4.3-3 depicts the mechanisms contributing to debris bed dryout.

As discussed in Chapter 3, the key factors affecting debris bed dryout are the particle sizes and the geometry (porosity) of the debris bed. Mixed particle sizes, particularly with smaller particles and deeper debris beds, tend to be less coolable than shallow debris beds composed of large particles. With smaller particles, the surface area for heat transfer is larger, and therefore, the vapor generation rates are increased relative to water ingress rates. Many particle sizes are possible during a severe accident, ranging from 0.01 inches (0.025 cm) or less up to inch size and larger. There is no one exact particle size that provides a threshold for coolability. However, particle sizes of a tenth of an inch (0.25 cm) and smaller are the ones most likely to be non-coolable. Such small particles can form during energetic melt ejection from the vessel or as a result of energetic fuel-coolant interactions (discussed in the next subsection).

In addition to debris bed dryout, there is a second possibility for non-coolable core debris. If a molten pool is contacted by an overlying water pool, a crust may form, preventing the further contact of water with the melt. In this case, core-concrete attack may continue unabated, as discussed in Section 4.4.

With non-coolable core debris, any boiling that does occur will not rapidly affect the containment pressure, and can generally be neglected, unless a sequence involves loss of all containment heat removal for many hours or even days. Because some of the decay heat goes into the core-concrete attack as opposed to the containment atmosphere, this case actually produces less of a long-term overpressure threat from steaming than the case where the debris is quenched. The threats from core-concrete attack and combustible (and other non-condensible) gas generation may more than offset the benefits of reduced steaming and are discussed in more detail in later sections.

4.3.3 Ex-Vessel Steam Explosions

The largest threat to containment resulting from the ex-vessel interaction of molten core debris and water is an energetic ex-vessel fuel-coolant interaction (steam explosion). An ex-vessel steam explosion is simply an extreme case of a steam spike where the quenching occurs explosively and produces dynamic as well as static pressures. An ex-vessel steam explosion can threaten the containment in several different ways, including:

1. generation of dynamic pressure loads (shock waves) that can fail the containment structure,

2. generation of pressures and shock waves that can fail vessel support structures, leading to movement of the vessel and failure of containment piping penetrations,

3. generation of energetic missiles that can be thrown into the containment, or

4. generation of pressures and shock waves that can fail the drywell floor of a BWR Mark II containment or the drywell wall of a Mark III containment.

Generally, the second and fourth threats above are the ones of most concern, and generally more so for BWRs (and a few PWRs) because of the confined pedestal region and the impact of pedestal failure on the containment. Section 4.3.4 discusses the design-specific aspects of ex-vessel steam explosions in more detail. As with in-vessel steam explosions, there are many factors that contribute to the magnitude of any ex-vessel steam explosion. These include:

1. the amount of water available to participate,

2. the composition of the melt, including the amount of unoxidized metals that may react during the explosion,

3. cavity or pedestal region geometry, insofar as it may lead to confinement of the explosion or focusing of shock waves,

4. transmission of shock waves through a water pool,

5. pouring rate and contact mode, i.e., water on corium, corium on water, or jet ejection into water, and

6. fraction of the core participating.

The physical processes involved in steam explosions were described in Chapter 3. Those processes are similar for ex-vessel steam explosions, except that some of the initial conditions are different. The ex-vessel case will always be at low pressure, no higher than the containment failure pressure. Steam explosions tend to be more likely at low pressure. Second, the geometry is different, involving varying degrees of confinement. Third, there are three contact modes to consider. The corium may pour from the vessel into a water pool or water may be added on top of corium, not unlike some in-vessel scenarios, or the corium may be ejected from the vessel as a high pressure jet into a water pool.

The latter case is unique to ex-vessel conditions and results when the vessel fails at high pressure. Experiments indicate that some steam explosions are likely under these conditions, but the magnitude is largely unknown. If the initial mass exiting the vessel reacts, it may blow the water out of the cavity or pedestal region, resulting in less reaction of the later material. Because the jet is not all released instantaneously, it is likely that a fairly small fraction of the core will participate. However, significant challenges to containment and vessel supports are still possible, particularly if oxidation accompanies the explosion.

One potential benefit of an ex-vessel steam explosion is that the core debris may be dispersed in the containment, reducing the concerns of core-concrete attack, and possibly making the debris more coolable. On the other hand, the benefit of such an event depends on exactly where the debris ends up and the continuing availability of long-term containment heat removal, and the impact on fission product releases.

As noted in Chapter 3, rapid quenching of core debris, explosively or otherwise, can result in significant oxidation of any metals contained in the core debris. Hydrogen generated as a result of this oxidation can present a significant threat that will be discussed in later sections.

4.3.4 Containment Design Considerations

As noted above, there are many features that can impact the importance of ex-vessel fuel-coolant interactions. First and foremost, the presence of water is necessary for a fuel-coolant interaction to occur. In some scenarios, particularly for large dry PWR containments, the reactor cavity will be dry or nearly so. Generally, for large quantities of water to be present in the reactor cavity, the containment sprays must have operated or large quantities of water must have flowed out through a break in the reactor coolant system. Then, if the sump and floor design allows, some of this water will overflow into the reactor cavity. Ice condenser containments are more likely to contain water in the reactor cavity due to the melting of ice combined with other sources. In fact, ice condenser containments can be deeply flooded in the lower compartment, mitigating fission product releases, but also providing a transmission medium for shock waves.

In BWR containments, water is likely to be present under the vessel for most LOCAs. Transient sequences may have a relatively dry pedestal region if the drywell sprays have not been used, and there has not been significant prior leakage. Mark III containments are the most likely to have large amounts of water under the vessel as a result of water spilling over the weir wall from the suppression pool. However, all three BWR containment types are susceptible to failure of the vessel supports, with relatively small amounts of water present. Figures 4.3-4, 4.3-5, and 4.3-6 depict typical pedestal regions for BWRs and point out some of the important vulnerabilities. As noted earlier, the Mark II containments are also susceptible to failure of the floor separating the drywell from the wetwell. Another factor for Mark II containments, resulting from the considerable design variation among the Mark II containments, is the possibility of corium flowing down the downcomers into the suppression pool, failing the downcomers with a steam explosion or as a result of meltthrough, thus leading to suppression pool bypass. One Mark II containment has downcomers located directly below the vessel, guaranteeing some flow into the downcomers.

For both BWRs and PWRs, if water is not present prior to vessel failure, then water may be pumped into the reactor coolant system at a later time and flow through the failed vessel onto the melt.

The relative containment failure probabilities from ex-vessel fuel-coolant interactions were assessed for the six containment types in the NUREG-1150 and LaSalle studies.[2,3] These studies indicate that containment failure is very unlikely for the three PWRs examined. For the three BWRs, drywell failures from steam explosions contribute noticeably to the overall containment failure probabilities, particularly for the Mark I and Mark II designs. In contrast, most of the individual plant examinations (IPEs) performed by the utilities found ex-vessel fuel-coolant interactions to be unimportant contributors to the likelihood of containment failure. A few of the IPEs used NUREG-1150 as the basis for their analysis and produced similar results.

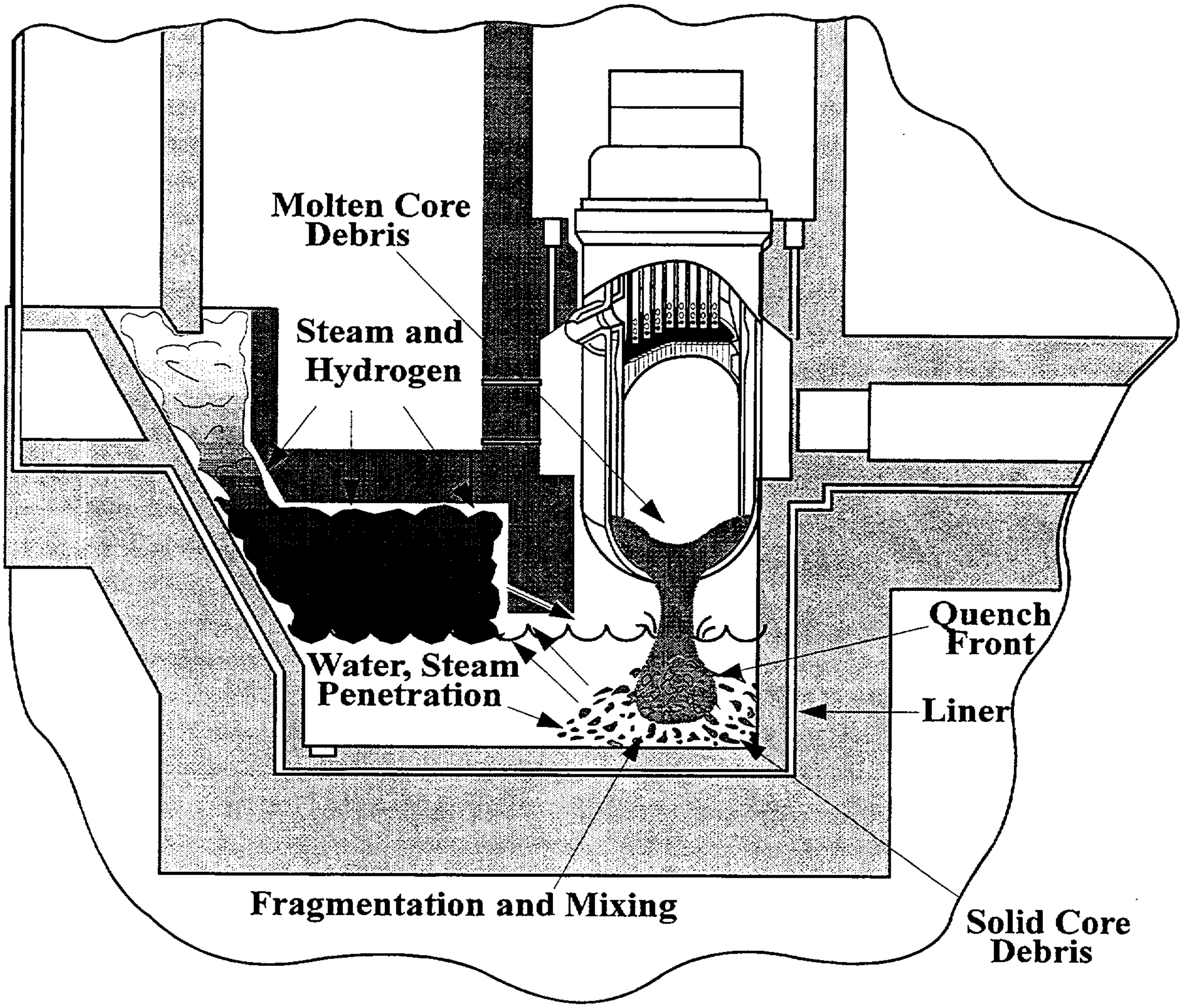


Figure 4.3-1 Molten core quenching process

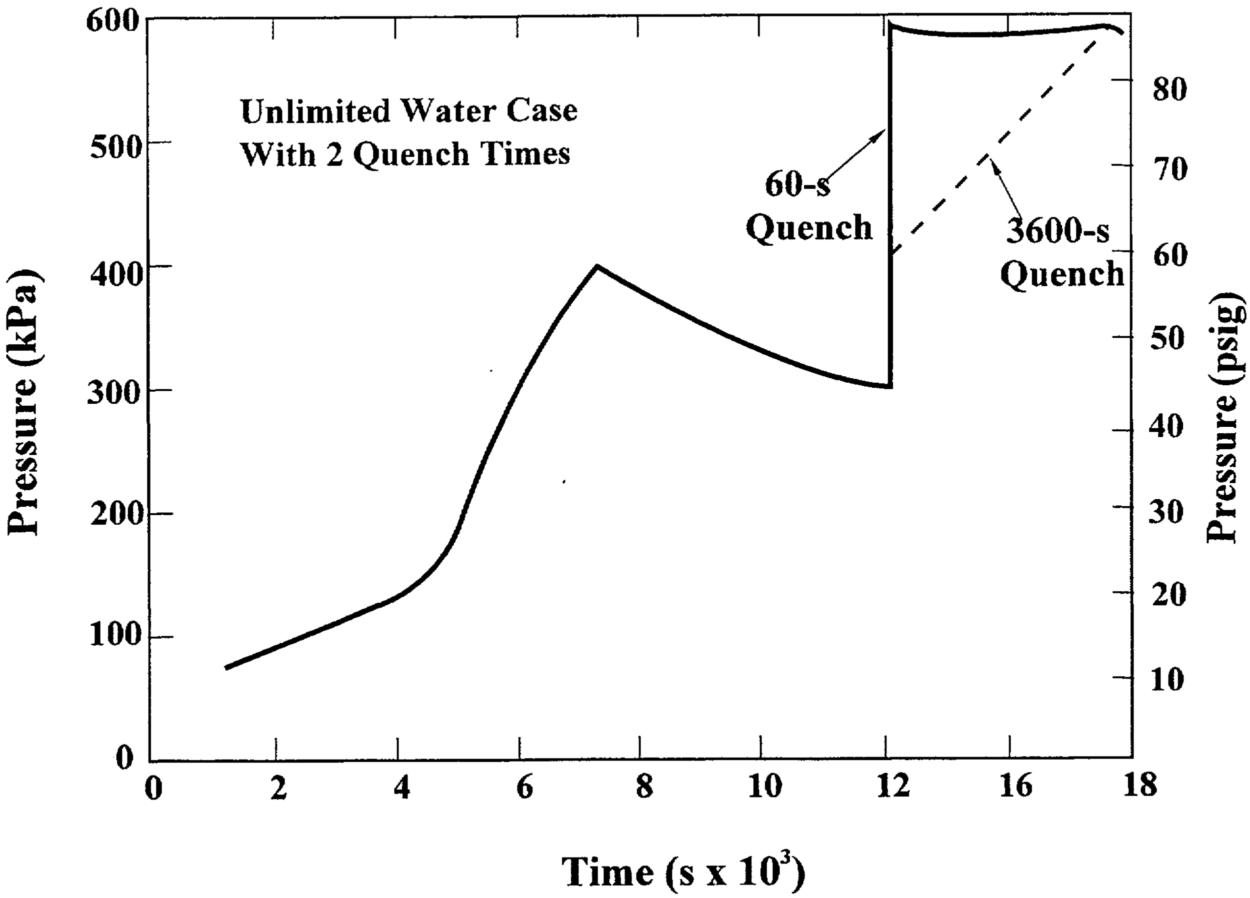


Figure 4.3-2 Containment pressure versus time for Zion station blackout sequence

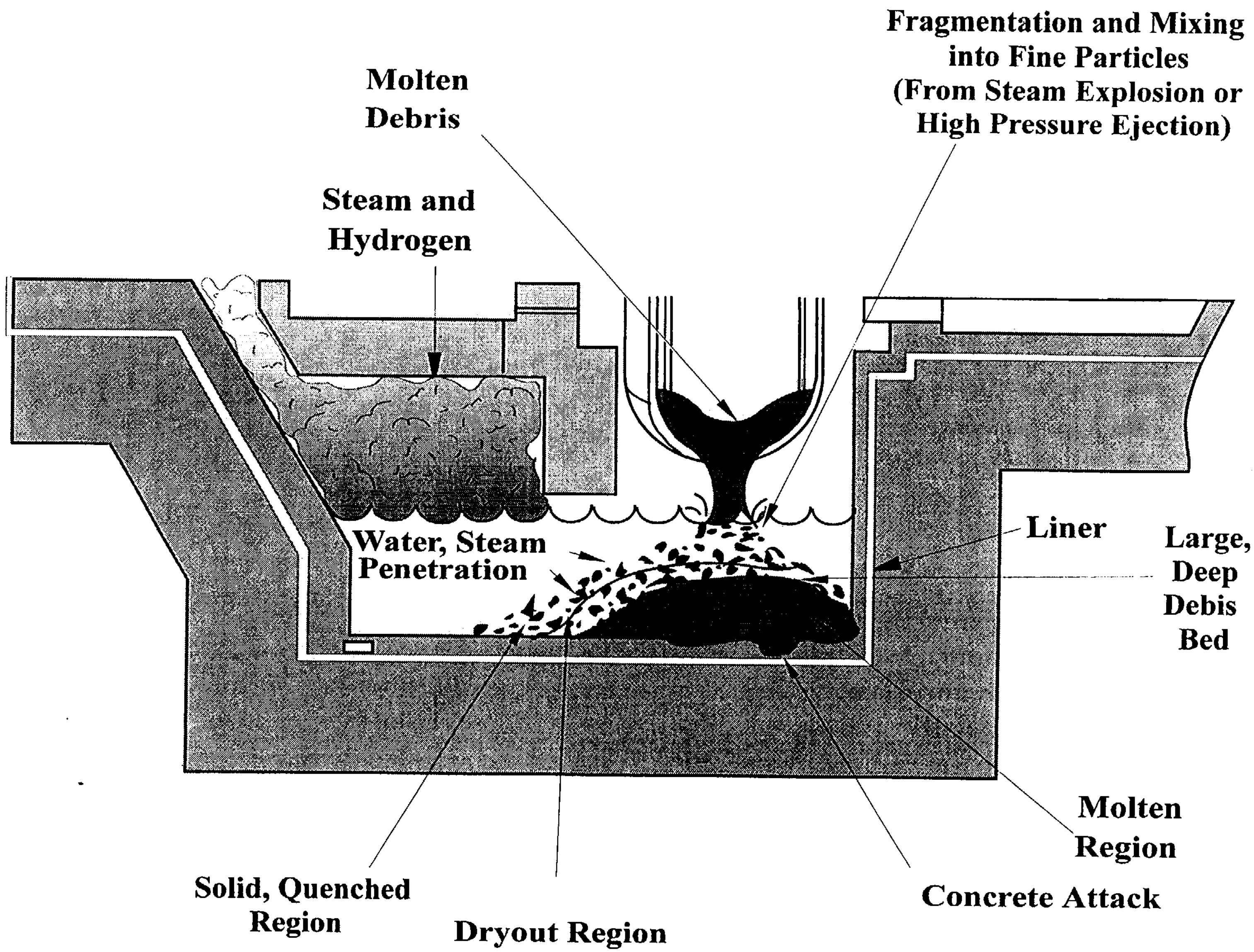


Figure 4.3-3 Non-coolable debris bed

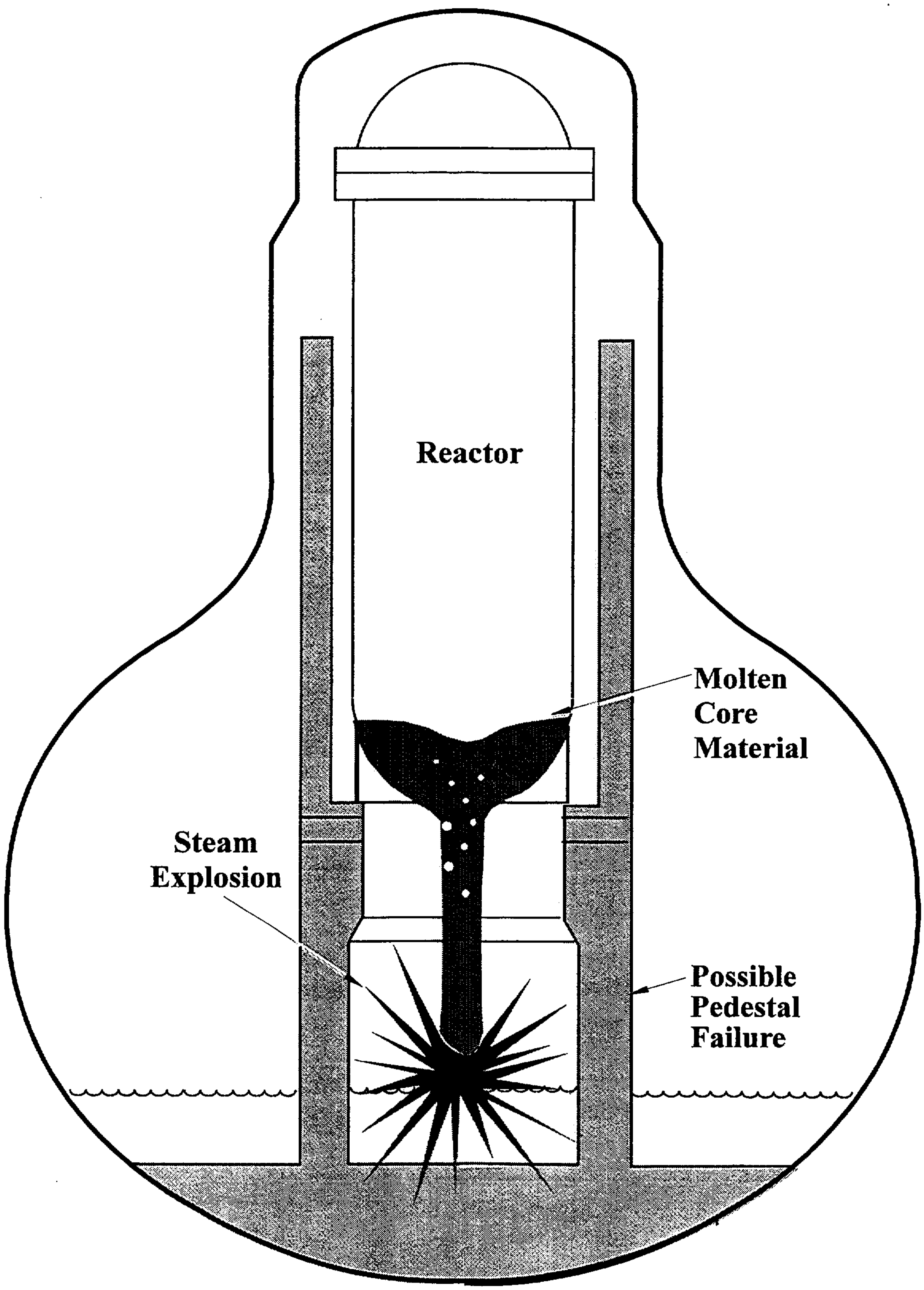
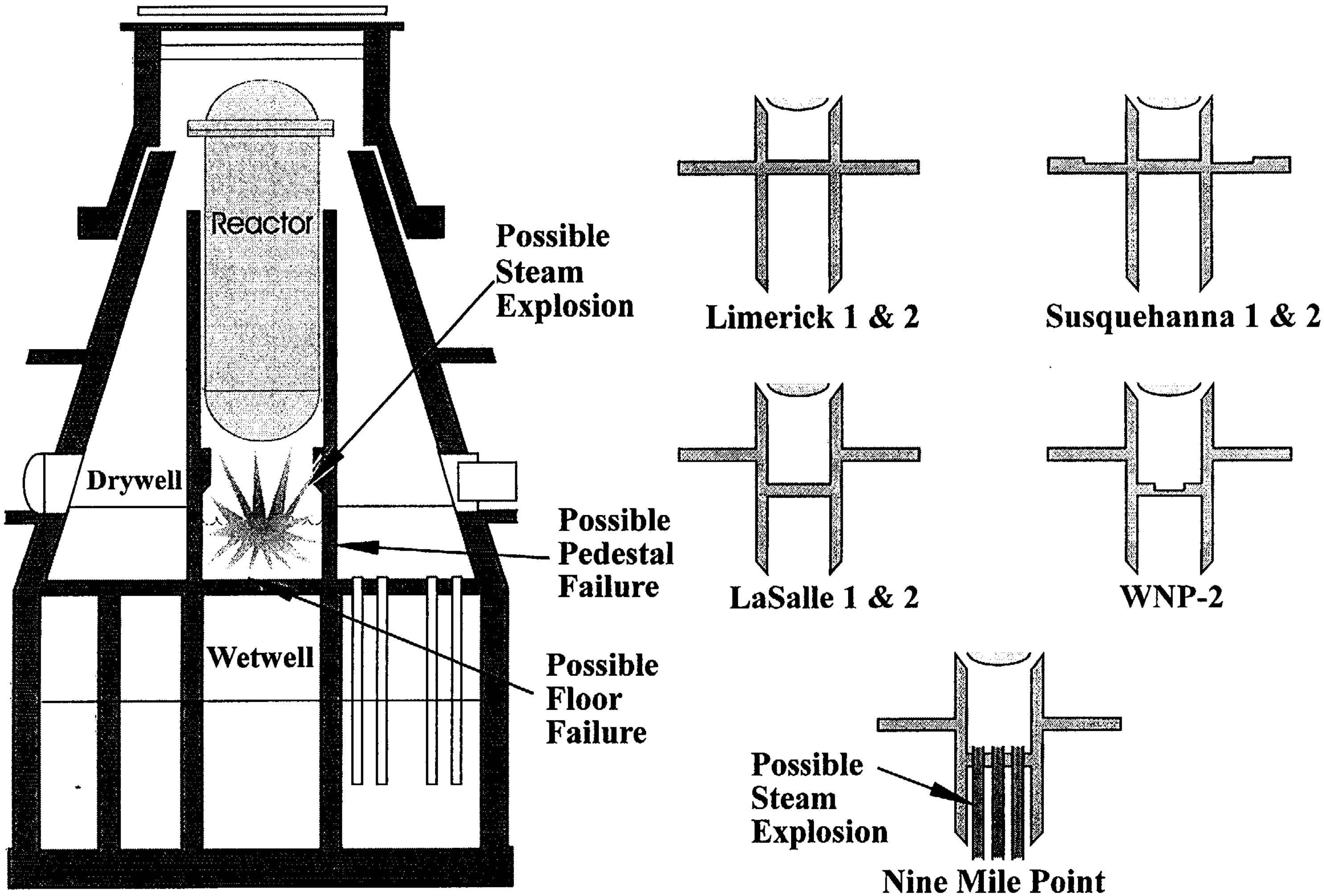


Figure 4.3-4 BWR Mark I containment pedestal region



Mark II Containment

Mark II Cavity Variations

Figure 4.3-5 BWR Mark II containment pedestal region

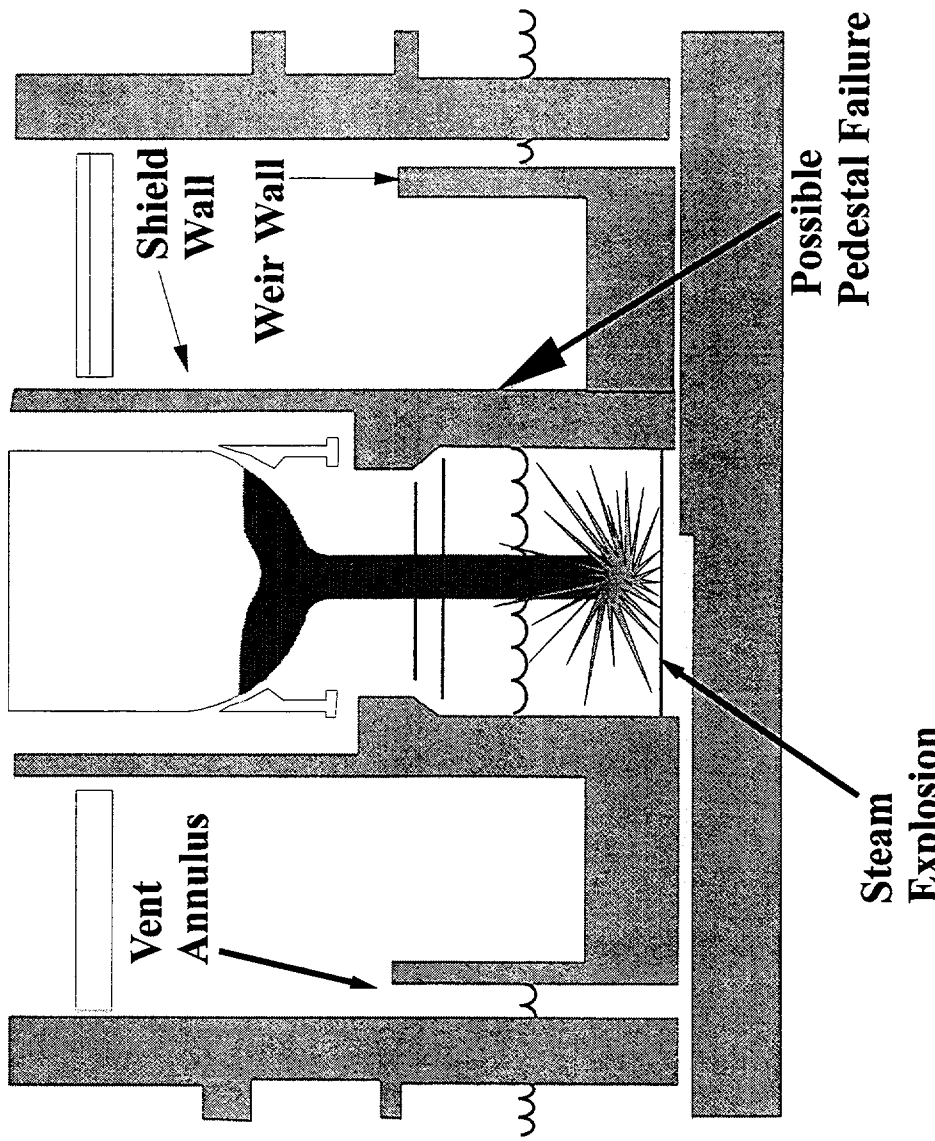


Figure 4.3-6 BWR Mark III pedestal region

References for Section 4.3

1. U.S. Nuclear Regulatory Commission, "Estimates of Early Containment Loads from Core Melt Accidents," NUREG-1079, Draft Report for Comment, December 1985.

2. U.S. Nuclear Regulatory Commission, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150, December 1990.

3. T. D. Brown, et al., "Integrated Risk Assessment for the LaSalle Unit 2 Nuclear Power Plant: Phenomenology and Risk Uncertainty Evaluation Program (PRUEP)," NUREG/CR-5305, SAND90-2765, Sandia National Laboratories, 1992.

4.4 Core-Concrete Interactions

If molten core material falls into the reactor cavity or pedestal region and is not blown out due to high pressure melt ejection or ex-vessel steam explosions, then Core-Concrete Interactions (CCIs) are possible. The possibility of CCIs leading to basemat meltthrough and containment failure was proposed by Brookhaven National Laboratory in reference to the China Syndrome.[1] Numerous studies and experimental programs have since verified that basemat meltthrough is possible, although there are still significant uncertainties. Research has indicated that CCIs can also have other important effects in accidents, even when the basemat remains intact. In particular, combustible gas generation can occur and large quantities of aerosols can be generated, thus affecting the source term if the containment fails. In the subsections below, these topics are discussed in more detail.[2]

4.4.1 Concrete Attack

The most obvious concern about CCIs is the compromising of the containment structure. In addition to basemat meltthrough, CCIs can lead to failure of vessel supports and other local structures that can indirectly lead to containment failure. The ensuing discussions of concrete attack are intended to include all of these possibilities.

Most concrete used in reactor applications is either limestone, basaltic or a combination of limestone and common sand. Table 4.4-1 gives typical compositions for these three types of concrete.[3] As shown in Figure 4.4-1, the attack of concrete by corium is driven largely by thermal processes. Decay heat and some heat from chemical reactions (which may dominate for short periods of time) are generated in the molten pool and may be transferred to the top surface of the pool or to the surrounding concrete. Under most circumstances, the heat flux to the concrete is sufficient to decompose it, releasing gases and melting the residual materials which are primarily oxides and metal reinforcing bars. The melted materials are added to the molten pool, thus diluting it, increasing its surface area, and reducing the volumetric heat generation rate. In time, heat transfer out the top of the molten pool and through the surrounding concrete may be sufficient to remove the generated heat and the temperature will decline to the point at which the CCI is terminated. Typical CCIs can penetrate concrete at the rate of several inches (tens of cm) per hour. Whether or not the CCI is terminated prior to basemat meltthrough is determined by many factors, including:

1. type of concrete and aggregate used in the structure,
2. basemat thickness,
3. cavity size and geometry,
4. melt mass in the cavity,
5. melt composition, and
6. presence of overlying water.

As noted in Section 4.3, the presence of an overlying water pool does not guarantee that the debris will be coolable. A crust may form over the melt, and heat transfer may be insufficient to remove all the decay heat from the melt. A number of experiments have shown minimal effect of water on concrete ablation rates.[4] However, an overlying water can reduce fission product releases even if it does not cool the debris.

As its temperature increases, concrete begins to fail (lose its structural integrity) even before gross melting of its constituents occurs. The loss of structural integrity accompanies the release of water and carbon dioxide from the concrete in three phases:[5]

1. release of molecular and physically entrapped water between 86 and 446 °F (30 and 230 °C),

2. release of water chemically constituted as hydroxides between 662 and 932 °F (350 and 500 °C), and

3. release of carbon dioxide from the aggregate and the cementitious phases between 1112 and 1832 °F (600 and 1000 °C).

The point at which concrete loses its integrity varies with the type of concrete, but generally occurs as hydraulic bonds are eliminated and well before the carbon dioxide is released. Typical concrete contains about 4 to 9 wt.% water and 0 to 45 wt.% carbon dioxide. Loss of structural integrity is particularly important when considering the possible impact of CCIs upon vessel supports in BWRs.

Figure 4.4-2 is an example calculation of concrete attack in the LaSalle BWR Mark II containment.[6] The concrete at LaSalle is a mixture of limestone and common sand. In general, limestone concrete will ablate more rapidly than basaltic concrete. An important aspect of containment failure due to concrete attack is that, even if it occurs, one would expect that many hours would be available to initiate emergency response plans, including evacuation and sheltering, so that offsite health effects can be minimized.

4.4.2 Gas Generation

CCIs result in the generation of large amounts of gases, some of which are combustible. Combustible gases are generated indirectly in a CCI. As shown in Figure 4.4-3, water and carbon dioxide are released from the concrete. When these gases rise through the melt they can react with unoxidized metals to produce metal oxides and the combustible gases hydrogen and carbon monoxide. As a result of complex reactions within the melt, the actual concentrations of hydrogen and carbon monoxide in the gases exiting the melt can vary significantly. It is likely that the flow of gases up through the melt will be nonuniform and that the melt itself will consist of layers of varying metallic content.

The total amount of combustible gas that can be formed as a result of CCIs is limited primarily by the amount of metallic constituents present in the melt, although some other reactions are possible that can slightly increase this quantity. The molten pool in the reactor cavity may contain large amounts of steel from the reactor vessel, below vessel structures, containment liner plate, concrete reinforcing bars, and other structures. As a result, the total quantity of combustible gas released from core concrete interactions can exceed that produced by 100% oxidation of all available zirconium, which is normally the limit for in-vessel hydrogen production.

Figure 4.4-4 shows examples of amounts of various gases that can be generated during core-concrete interactions[4]. Note that much more gas (primarily CO and CO_2) is produced by limestone concrete than basaltic concrete. In any case, it is not inconceivable that a few thousand pounds (or kilograms) of combustible gases could be generated from CCIs.[5]

As the combustible gas exits the top of the melt, there are several possibilities. First, if there is an overlying water pool, the gases will cool before they pass into the containment atmosphere. Second, if there is no overlying water pool, the gases may spontaneously ignite above the molten corium. This spontaneous ignition requires high temperatures (supplied by the molten pool) and the presence of oxygen. Oxygen in the cavity will be rapidly depleted unless flow paths exist to circulate oxygen from the rest of containment. Spontaneous ignition can not occur in Mark I and II BWRs, which have inert containments. Combustion effects will be discussed in more detail in Section 4.6.

For Mark I and II containments, despite their inerted condition, gases from CCIs can still represent a concern. Because these gases are noncondensible, they can lead to significant pressure buildup that can not be removed using sprays or suppression pool cooling. This is why venting may ultimately be required to prevent long-term overpressure from these gases.

4.4.3 Aerosol Generation

In the absence of an overlying water layer, Core-Concrete Interactions produce dense clouds of aerosols. Two processes produce such aerosols. First, volatile and semi-volatile chemicals species including many fission products can be present in hot gases that bubble up through the melt. As these species emerge into and mix with the cooler atmosphere above the melt, they condense and become aerosol particles. Second, when gas bubbles emerge and burst at the surface of the melt, aerosols containing less volatile species are formed and entrained in the flowing gases. An overlying water pool can effectively remove most of the aerosols generated in CCIs, particularly the less volatile species.

In a severe accident, the radioactive and nonradioactive aerosols released from CCIs can significantly impact the concentrations of radionuclides in the containment atmosphere and, given containment failure, the quantities of radionuclides released to the environment. In general, generation of radioactive aerosols will increase the releases to the environment. In fact, the largest release fractions postulated in risk assessments are generally for accident scenarios involving Core-Concrete Interactions with no overlying water pool in the reactor cavity. However, the generation of large quantities of non-radioactive aerosols can accelerate the agglomeration and gravitational settling of radioactive aerosols in the containment. Thus, if containment failure is delayed long enough, aerosol generation can actually reduce releases to the environment. Large quantities of aerosols, radioactive or not, have the potential to plug air filters that are not designed for such loadings.

Figure 4.4-5 shows example VANESA calculations of aerosol generation rates as a result of CCIs at three plants and for three different accident scenarios.[7] The wide variations result from differences in melt composition and concrete type. These calculations do not account for any overlying water pools. This figure indicates the tremendous mass of material that can be suspended in the containment in the form of aerosols. Table 4.4-2 and Figures 4.4-6 and 4.4-7 indicate the types of materials that can be contained in the aerosols. Most of the mass is made up of concrete materials, such as CaO and SiO_2. However, Table 4.4-2 and Figure 4.4-6 also show that significant fractions of fission products are released during CCIs.

Table 4.4-1 Typical chemical compositions of concrete (wt.%)

Oxide	Basaltic Concrete	Limestone Concrete	Limestone/ Common Sand Concrete
SiO_2	54.73	3.60	35.70
CaO	8.80	45.40	31.20
Al_2O_3	8.30	1.60	3.60
MgO	6.20	5.67	0.48
Fe_2O_3	6.25	1.20	1.44
K_2O	5.38	0.68	1.22
TiO_2	1.05	0.12	0.18
Na_2O	1.80	0.08	0.82
MnO	-	0.01	0.03
Cr_2O_3	-	0.004	0.014
H_2O	5.00	4.10	4.80
CO_2	1.50	35.70	22.00

Table 4.4-2 Core-concrete release for Peach Bottom station blackout sequence

Species	Released Mass (kg) (1 kg = 2.2lb.)	Release Fraction[(1)]
Fission Products		
I+Br	1.8	1.0
Cs+Rb	27	1.0
Te+Sb	14	0.64
Sr	53	0.84
Mo	5.0×10^{-4}	2.0×10^{-6}
Ru[(2)]	3.0×10^{-4}	9.0×10^{-7}
La[(3)]	33	3.9×10^{-2}
Nb	4.3	1.0[(4)]
Ce+Np+Pu	90	9.0×10^{-2}
Ba	64	0.62
Steel		
Fe[(5)]	1234	1.3×10^{-2}[(6)]
Cr	6.6×10^{-2}	8.10^{-6}
Ni	29	6.2×10^{-3}
Mn	89	0.50
Zircaloy		
Zr[(7)]	0.55	8.0×10^{-6}
Sn	46	5.0×10^{-2}
Control Material		
Gd	17	5.8×10^{-2}
Fuel		
U	23	2.0×10^{-4}
Concrete [(6)]		
Ca0	1988	2.9×10^{-2}
Al_20^3	339	0.14
Na_20	82	0.74
K_20	656	0.64
$Si0_2$	1124	0.21

(1) Based on melt inventory at start of core-concrete interaction.
(2) Includes Tc, Rh, and Pd.
(3) Includes Y, Zr(fp), Pr, Nd, Pm, Eu, and Sm.
(4) Quantitative release is calculated because of the assumed oxide chemical form, which is under review.
(5) Includes Fe from concrete and reinforcing bars.
(6) Release fraction based on the amount of concrete and reinforcing bars incorporated into the molten pool.
(7) Structural Zr only.

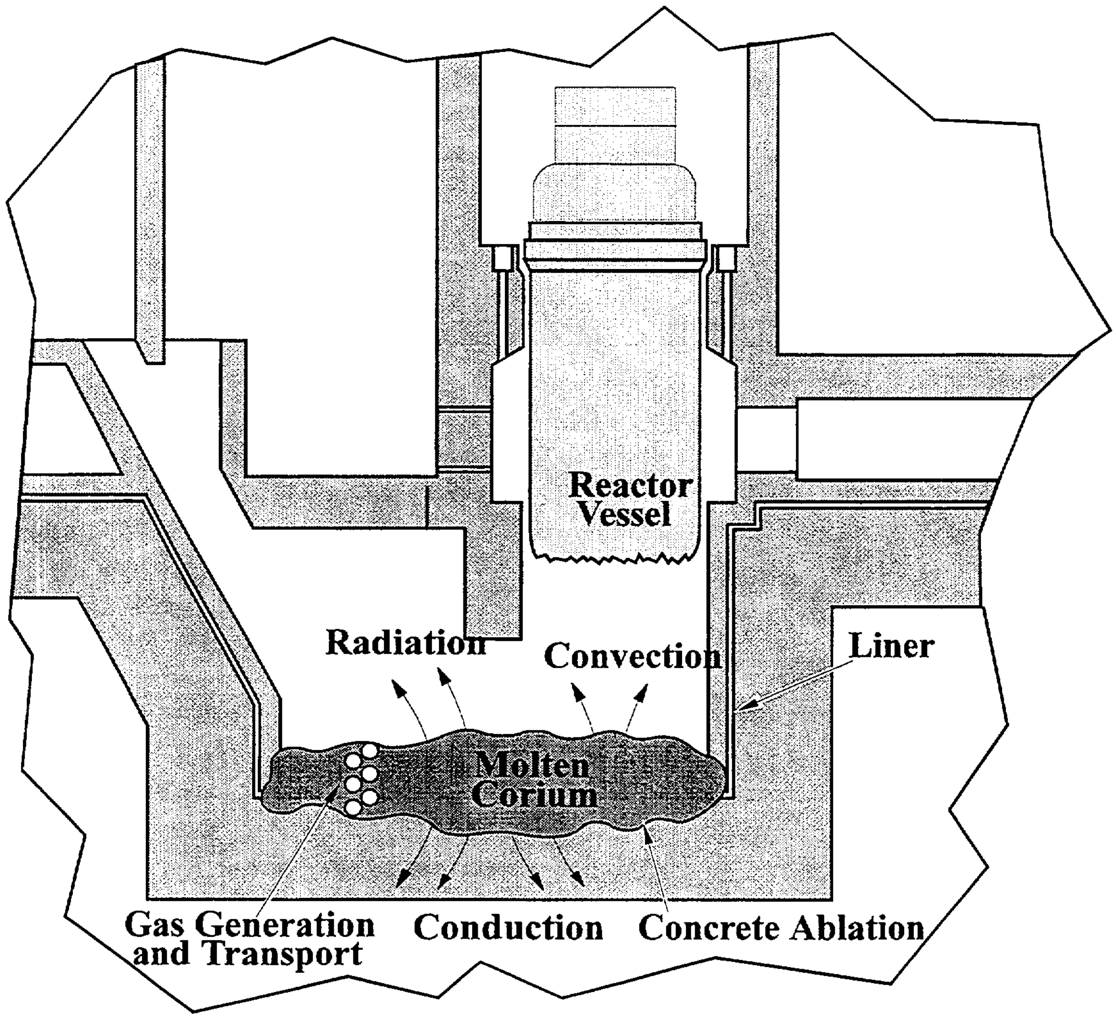


Figure 4.4-1 Thermal aspects of core-concrete interactions

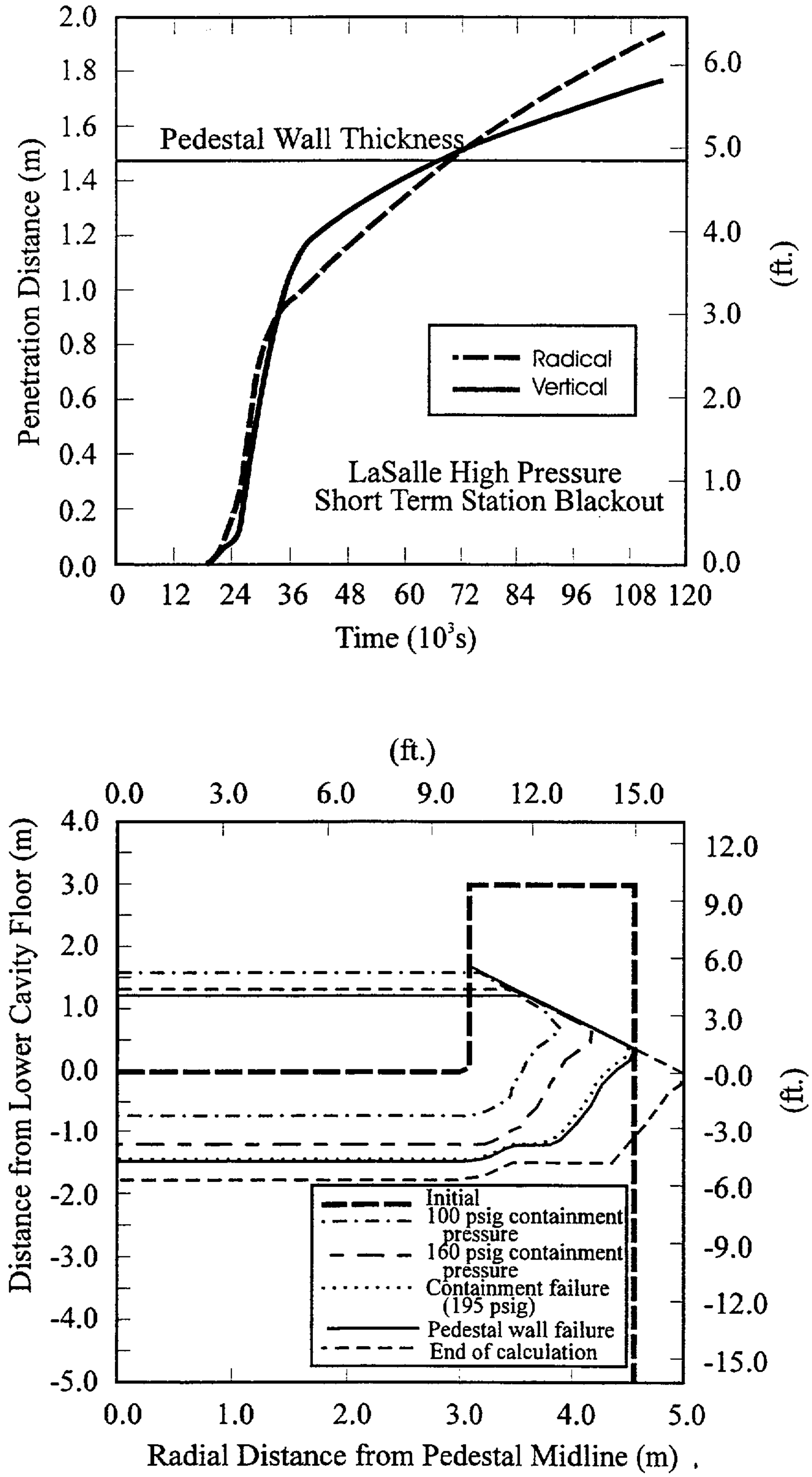


Figure 4.4-2 Calculations of concrete attack in a BWR Mark II containment during a station blackout sequence

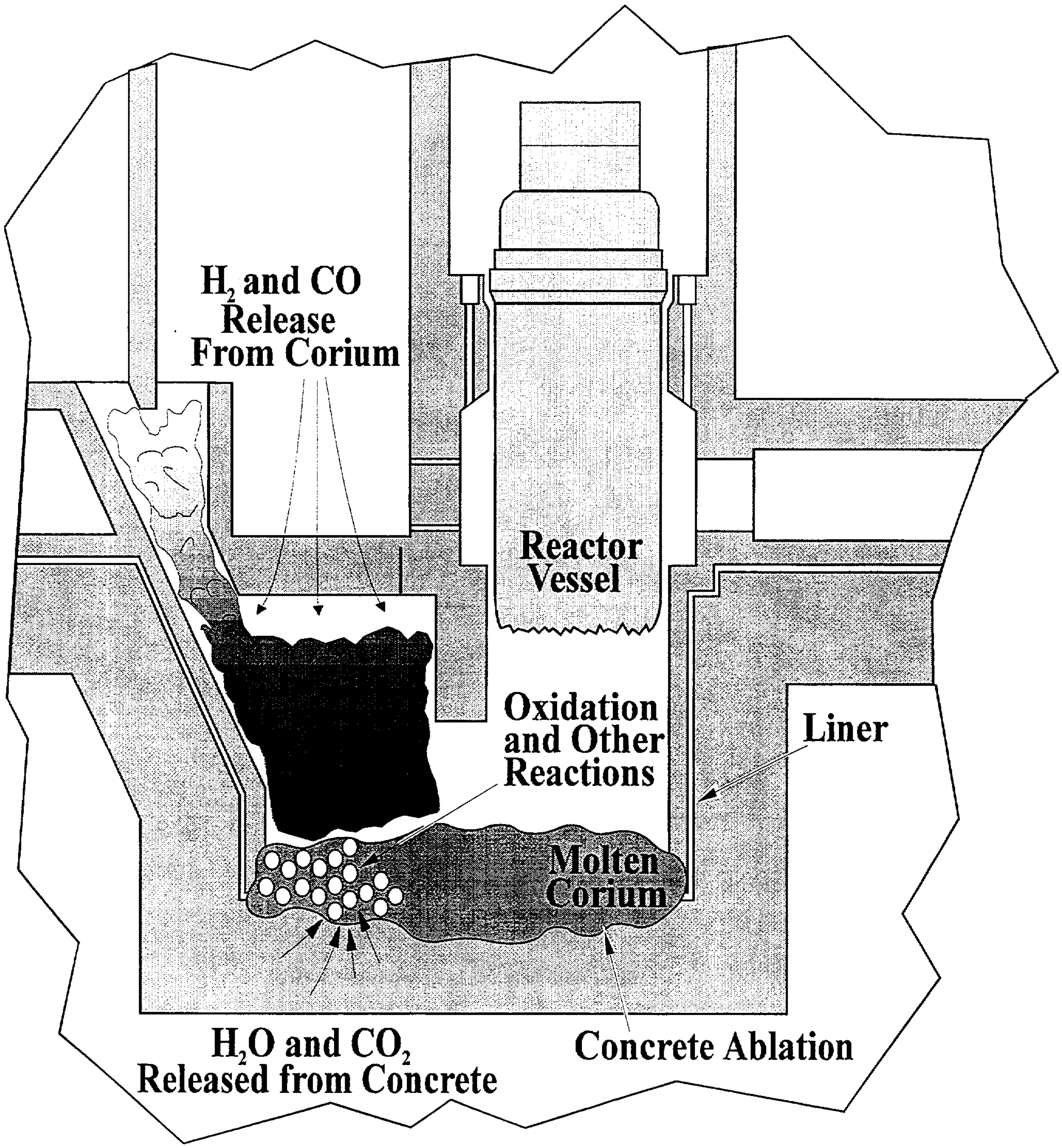
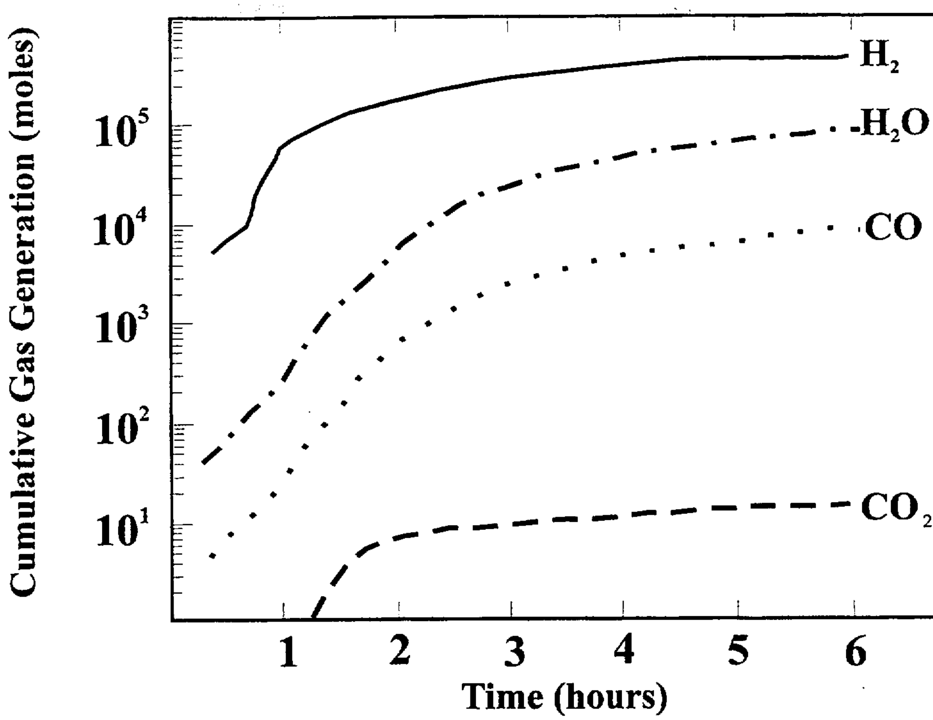


Figure 4.4-3 Combustible gas generation during CCIs

Basaltic Concrete



Limestone Concrete

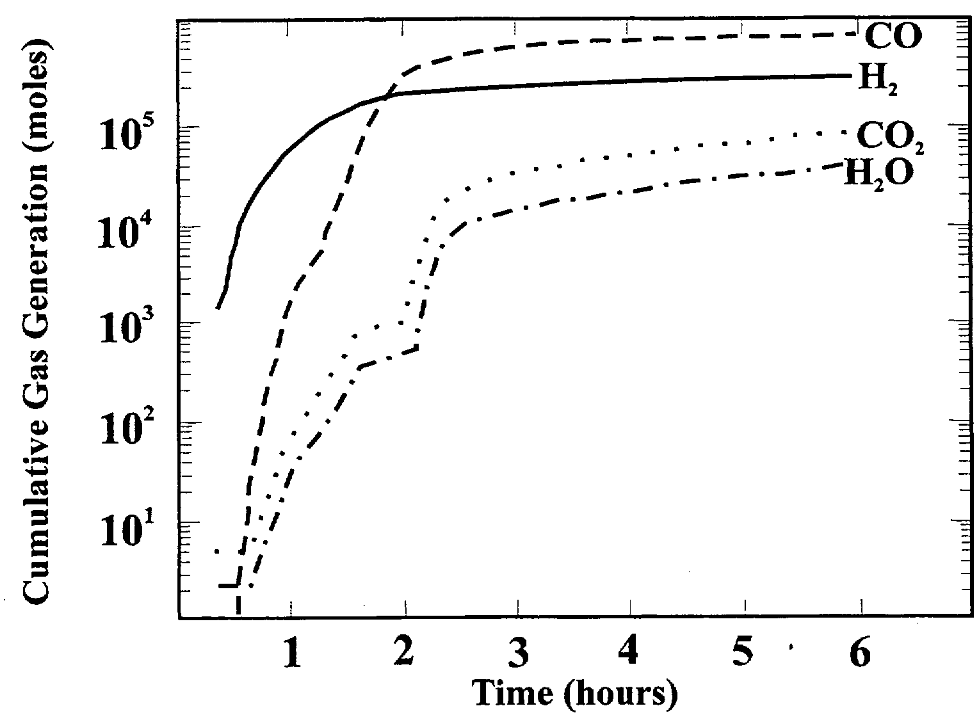


Figure 4.4-4 Example amounts of various gases that can be generated during core-concrete interactions

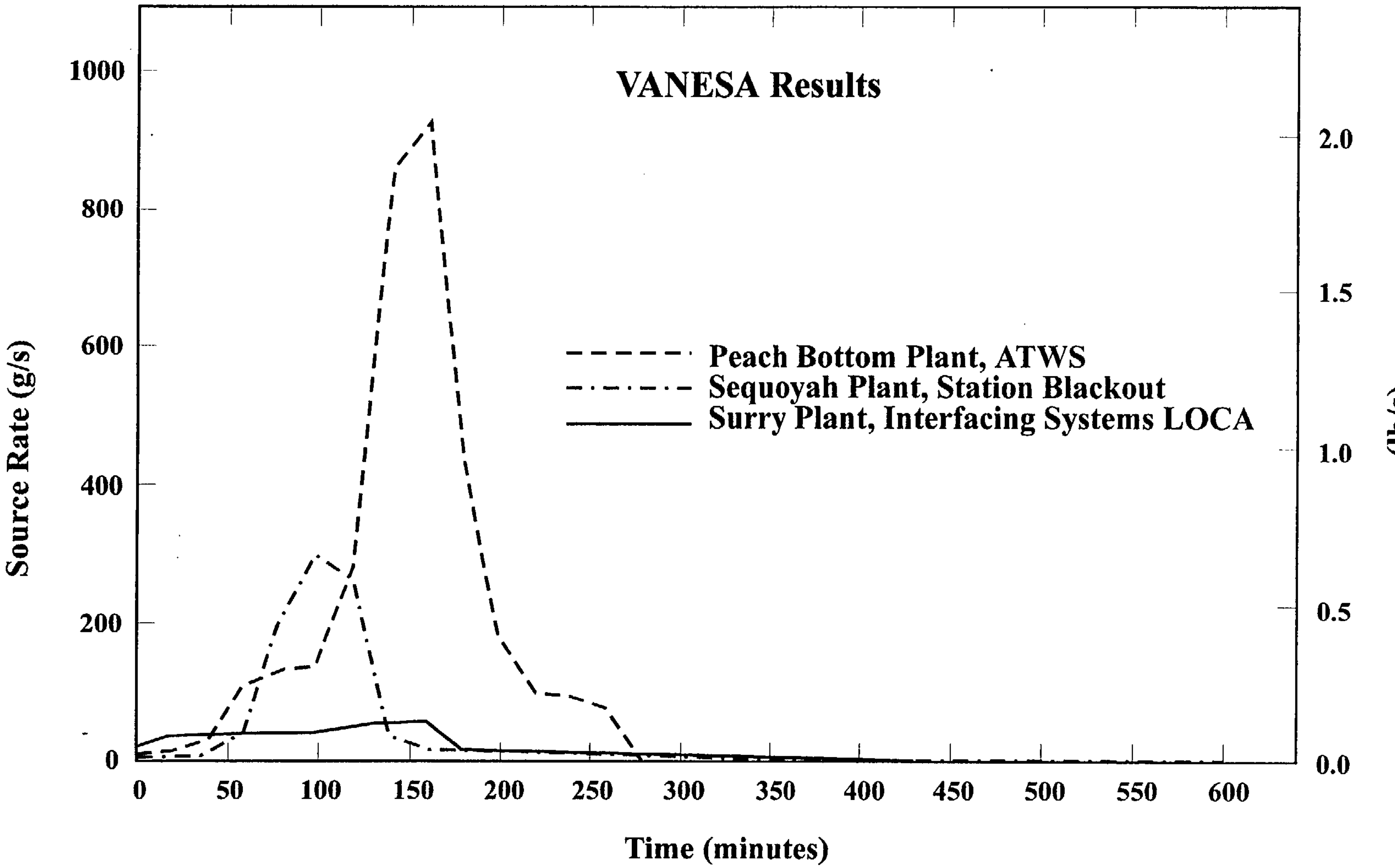


Figure 4.4-5 VANESA calculations of aerosol generation rates

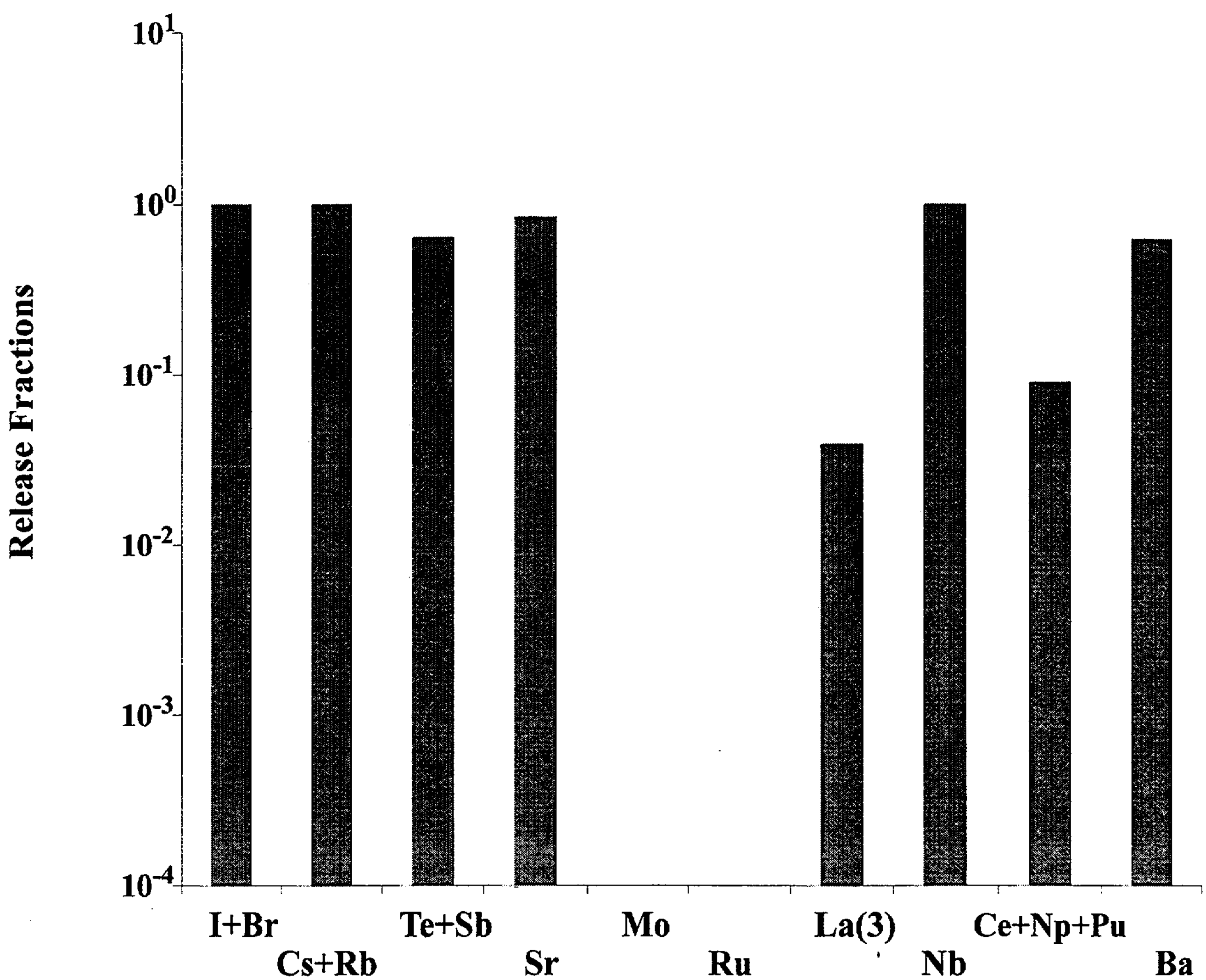


Figure 4.4-6 Peach Bottom station blackout, fission products released to drywell from core-concrete interactions

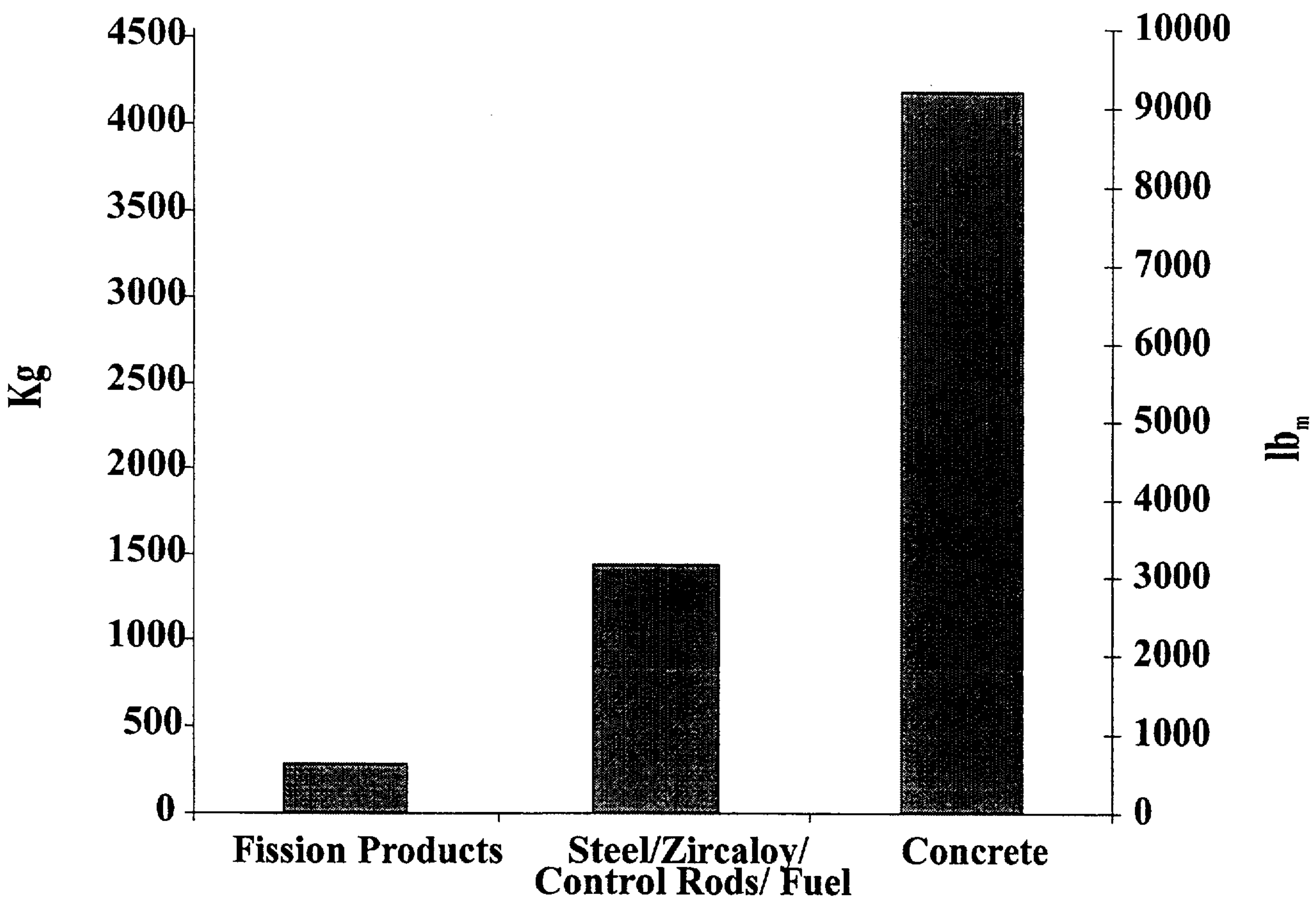


Figure 4.4-7 Peach Bottom station blackout, masses released to drywell from core-concrete interactions

References for Section 4.4

1. ACRS Subcommittee Meeting Minutes, June 3, 1966, reproduced in Okrent, pp. 99-101.

2. R. K. Cole, Jr., D. P. Kelly, and M. A. Ellis, "CORCON-Mod2: A Computer Program for Analysis of Molten-Core Concrete Interactions," NUREG/CR-3920, SAND84-1246, Sandia National Laboratories, Albuquerque, NM, August 1984.

3. E. R. Copus and D. R. Bradley, "Interaction of Hot Solid Core Debris with Concrete," NUREG/CR-4558, SAND85-1739, Sandia National Laboratories, June 1986.

4. Reactor Safety Research Semiannual Report, July-December 1986, Volume 36, NUREG/CR-4805, SAND86-2752, Sandia National Laboratories, November 1987. (Note: an error was found on the first graph of the original figure. The labels for the CO and Co_s plots were reversed.)

5. A. L. Camp, et al., "Light Water Reactor Hydrogen Manual," NUREG/CR-2726, SAND82-1137, Sandia National Laboratories, Albuquerque, New Mexico, August 1983.

6. C. J. Shaffer, L. A. Miller, and A. C. Payne, Jr. "Integrated Risk Assessment for the LaSalle Unit 2 Nuclear Power Plant: Phenomenology and risk Uncertainty Evaluation Program (PRUEP)," Volume 3: MELCOR Code Calculations, NUREG/CR-5305, 3 of 3, SAND90-2765, Sandia National Laboratories, Albuquerque, New Mexico, July 1992.

7. M. Silberberg, et al., "Reassessment of the Technical Bases for Estimating Source Terms," NUREG-0956, July 1986.

4.5 Direct Containment Heating

A severe accident may progress with either high or low pressure in the reactor coolant system up to the point of vessel breach. Chapters 2 and 3 discussed some of the accident scenarios that could involve high pressure at the time of vessel breach. When vessel failure occurs at a pressure of a few hundred psi (several hundred kPa) or more, the melt will be ejected as a jet into the reactor cavity. What happens next depends upon the reactor vessel pressure, the cavity and containment design, the presence of water in the cavity, the amount of melt ejected and other factors. One possibility, discussed earlier, is that a steam explosion will result in the reactor cavity, if sufficient water is available and the melt can mix with the water. Another possibility is that some of the melt will be fragmented by jet breakup and swept out of the cavity into the containment where it will heat the atmosphere (direct containment heating [DCH]). The latter process can lead to very rapid and efficient heat transfer to the atmosphere, possibly accompanied by oxidation reactions and hydrogen burning that further enhance the energy transfer. Energy transfer to the containment atmosphere, of course, leads to containment pressurization. The important phenomena are discussed in more detail below.

4.5.1 Ejection of Melt from the Vessel

The melt ejection process is depicted in Figure 4.5-1. When the vessel first fails, molten material will be ejected as a liquid stream (melt ejection phase). As the liquid corium level in the vessel drops, gas blowthrough will begin to occur, resulting in a two-phase mixture blowing down from the vessel. The noncoherence of the steam blowdown and melt ejection is predicted to have a large impact on the DCH loads, as discussed in Section 4.5.3. The high velocity expanding gas flow provides the motive force for entraining corium and ejecting it from the reactor cavity (gas blowdown phase).

Vessel failure may occur at a small opening, such as an instrument tube, or as a result of a larger rupture. The particular failure mode does not have a large impact during the melt ejection phase, but can be important for ex-vessel steam explosions or for the gas blowdown phase interactions discussed in Sections 4.5.2 and 4.5.3. The amount of material participating in an ex-vessel steam explosion and the nature of the explosion will be affected by the ejection rate (which depends on the opening size). Small amounts of molten material may result in small explosions that force water out of the cavity and preclude larger explosions. For Westinghouse PWR vessels, it has been estimated that the initial hole size could be approximately 1.3 ft. (0.4 m) in diameter for a thermally-induced rupture or about an inch (0.03 m) for ejection of an incore instrument tube.[1] However, for the latter case the hole is expected to rapidly ablate to about the size of a thermally-induced rupture, so that the overall effect of the small initial opening size is minimal.

Along with the hole size, the amount and composition of molten material in the lower plenum of the vessel is an important factor. In some scenarios, vessel failure may occur early, when only part of the core is molten. Core material that has not relocated to the lower plenum will not contribute significantly to the direct heating process. Figure 4.5-2 shows an example estimate of the amount of material that may be ejected for given core melt scenarios in PWRs.[2]

The melt composition is also important. Melts rich in metal will tend to result in higher DCH loads because of the energy released when the metal is oxidized, and because of the energy released by combustion of any hydrogen produced by the reaction of steam with the metal.

As part of the DCH resolution program, best estimate SCDAP/RELAP5 calculations were performed for representative reactors of all nuclear steam supply systems (PWRs) in the United States:[3,4*] Zion (Westinghouse 4-loop), Surry (Westinghouse 3-loop), Calvert Cliffs (Combustion Engineering), ANO-2 (Combustion Engineering), Oconee (Babcock & Wilcox lowered loop) and Davis-Besse (Babcock & Wilcox raised loop). In all of these calculations, the melt composition was predicted to be predominantly oxidic at the time of vessel breach. This is consistent with observations from the TMI accident. It may not be consistent with accidents that have less water available. Indeed, recent tests in the PHEBUS-FP tests raise questions about the accuracy of computer code calculations of the metallic fraction of the core melt at the time core debris relocates from the core region.

4.5.2 Interactions in the Reactor Cavity

When molten material is ejected into the reactor cavity at high pressure, there are a number of phenomena that are important to consider. The possibility of an ex-vessel steam explosion has already been mentioned. Additional phenomena that lead to fragmented debris that can be dispersed from the cavity are also important. These phenomena, depicted in Figure 4.5-3, include molten jet breakup, gas evolution and chemical reactions, and trapping of a portion of the jet before it can escape the cavity.

The presence of water in the reactor cavity could result in some quenched debris, thus partially mitigating the DCH threat.

Experimental evidence indicates that the presence of water in the reactor cavity can be detrimental. A jet of molten material entering a pool of water will often lead to a steam explosion.[**] With small levels of water, the experiments show that the initial contact with molten debris produces a steam explosion that blows the remaining water out of the cavity, ending immediate debris-water interactions. Experiments with high temperature melt injection into model reactor cavities filled completely with water have produced dramatic steam explosions of sufficient magnitude to threaten structural damage. In addition to potential steam explosions, water can also provide an additional source of hydrogen by chemically reacting with unoxidized metals in the molten debris. As the jet encounters water or steam (either from the blowthrough or as a result of water in the cavity), oxidation of any metals can occur, leading to rapid hydrogen production. Some experiments indicate that the gases exiting the reactor cavity can contain as much as 50% hydrogen during some phases of the blowdown.[**]

The sizes of particles produced by breakup of a jet of molten material can affect the heat transfer and chemical reaction rates (by determining the available surface area), as well as particle transport within containment. At high RCS pressures, the dispersed melt is highly fragmented (~1mm) with a broad distribution of particle sizes. Extensive (thermal and chemical) interactions (near equilibrium) of melt with blowdown steam can be expected during dispersal.

**Resolution of the Direct Containment Heating Issue for all Combustion Engineering Plants and Babcock & Wilcox Plants,* NUREG/CR-6475, In Preparation.

**Memo from Richard Griffith to R. G. Gido, Sandia National Laboratories, May 11, 1992.

Figure 4.5-4 shows some estimated mean particle sizes that can result for given conditions.

Evolution of gases dissolved in the melt can result in changes in the jet breakup, and can also significantly affect fission product releases. The melt breakup process is likely to release most of the volatile materials and also allow formation of numerous radioactive aerosols, although these processes are not well understood.

As the high-temperature jet passes through the cavity, melt is entrained and swept out into the containment. The debris dispersal is noncoherent with the RCS blowdown; that is, the melt is fully dispersed from the vessel and cavity long before blowdown is complete. Gases exiting the reactor cavity may have velocities of several hundred feet per second (hundreds of m/s) according to some estimates.[3] As the melt is swept along, some of it impinges upon the cavity floor or walls. Significant erosion of concrete is not expected to occur because the melt will mostly splash off. Additional metal may enter the DCH process through two processes: incore instrument tubes, which pass through many cavities, may be ablated and dispersed from the cavity with the melt; and RPV insulation in the annulus around the RPV may be ablated and dispersed with the melt.

As the jet passes through the cavity, corium will bounce off of the walls, perhaps multiple times, as it is carried along by the gases. Ultimately, depending on the driving pressure, some fraction of the melt will be retained in the cavity and not enter the main containment. Particles may be trapped under a seal table or any other obstruction in the path of the jet, as long as the jet does not cut through the obstruction. Locations where the flow sharply changes direction may also collect debris. However, for all PWR reactor cavities examined to date, experiments have shown nearly complete dispersal of the debris at RCS pressures of interest. At low pressures, some cavity designs retain debris, but with the low RCS pressures, the availability of steam is more limiting to DCH loads[1]. Note that any trapped material may result in subsequent core debris-concrete interactions within the reactor cavity.

Most reactor cavities can withstand the loads accompanying high pressure melt ejection (HPME). However, weaker cavities might be vulnerable to overpressure damage resulting from initial melt/water interactions (explosive or nonexplosive) or from high cavity pressures resulting from the dispersal process itself.

4.5.3 Energy Deposition and Pressure Rise in Containment

As core debris is swept out of the reactor cavity, it is transported throughout the containment. The degree to which the debris can be transported to the top of the containment affects the resulting pressure rise. In the lower regions of most PWR containments, the containment is highly subcompartmentalized. It is expected that significant quantities of the core debris will be trapped in these subcompartments before it can reach the upper regions of containment. This trapping may significantly reduce the predicted containment pressure rise. The reduction comes about because debris within subcompartments can thermally saturate the subcompartment atmosphere and consume any available oxidant without completely cooling or completely reacting before it is trapped.

Some containments have a fairly open path around the reactor vessel to the upper containment. Melt can be dispersed upwards from the cavity through the annulus around the RPV into the refueling canal and upper dome. This is the dominant dispersal path in some Combustion Engineering plants. These containments will not benefit as much from the effects of subcompartments.

Suspended debris particles can rapidly transfer their energy to the containment atmosphere. Because of the small particle sizes, the total surface area for heat transfer is quite large. The amount of thermal energy available in a molten core was discussed previously in Chapter 3. This thermal energy can be transferred to the containment atmosphere through radiative and convective heat transfer. This heat transfer will be very rapid, with much of it occurring in a matter of seconds if particles remain airborne and continue to encounter cool atmospheric gases. As the atmosphere heats, of course, the rates of heat transfer in the absence of exothermic reaction decrease substantially.

In addition to heat transfer, energy may be imparted to the containment atmosphere as a result of exothermic oxidation reactions involving metallic constituents in the core debris and either air or steam. The noncoherence of the steam blowdown and melt ejection limits the extent of these reactions, thereby reducing containment loads. The metal-steam reactions will result in the production of additional hydrogen. Hydrogen from these reactions plus hydrogen previously injected into containment may then burn, resulting in additional pressurization. The hot debris particles and the high temperatures of the exiting gases may lead to some hydrogen combustion even for mixtures outside the normal flammability limits (see Section 4.6).

Figure 4.5-5 shows examples from the NUREG-1150 study of the range of pressures considered possible for a DCH event in the Surry subatmospheric containment.[3] In that study, now believed to be conservative, the important factors were considered to be the vessel pressure, the presence of water in the cavity, the vessel hole size, the core fraction ejected, the amount of zirconium oxidation, and the operation of containment sprays.

In Figure 4.5-5, the dry cavity case (Case 1) results in higher pressures than the equivalent wet cavity case (Case 2). In these estimates, steam explosions resulting in dynamic pressures damaging the cavity or other parts of the containment were not considered. Without steam explosion damage, water was predicted to be beneficial, with the heat absorption outweighing any detrimental effects of hydrogen production.

4.5.4 Containment Failure Probabilities for DCH

It is clear that in extreme cases high pressure melt ejection and direct containment heating can produce pressures that threaten structural integrity of the containments. The issue becomes one, then, of the probability that such extreme loads will actually occur in an accident. Nearly all pressurized water reactor licensees have instituted measures in their emergency operating procedures to depressurize the reactor coolant systems to eliminate the driving force for high pressure melt ejection. Despite this, accidents can still be envisaged in which there is a failure to depressurize or there is incomplete depressurization. These accidents are not unlike those pressurized accidents found to be possible in boiling water reactors despite the availability of an automatic depressurization system. A variety of analyses have appeared that indicate natural processes during core degradation will lead to depressurization of the reactor coolant system. Typically these analyses show that the natural circulation of gases and vapors heat portions of the reactor coolant system such as the surge line or the nozzles for pipes attached to the reactor vessel that they fail by creep rupture before core debris can penetrate the reactor vessel. Comforting as these analyses may seem, the fact remains that no evidence of such heating of the reactor coolant system was encountered during the accident at Three Mile Island. It is difficult, then, to find a firm basis for discounting direct containment heating based solely on the low probability of it occurring.

Research sponsored by the NRC has focused on the probability of containment loads produced by high pressure melt ejection exceeding the structural capabilities of the containment. Three computer models of the phenomena associated with high pressure melt ejection have been developed - CLCH (Convection Limited Containment Heating), TCE (Two-cell Equilibrium), and the CONTAIN code that is a systems level model of containment response during severe reactor accidents. Experiments involving high pressure melt ejection into scaled models of reactor cavities have been used to develop and validate these models which can usually be run to give very similar results.

Extrapolation of the model predictions to the scale of reactor accidents has shown that the uncertain quantities that most influence the pressurization of containment following high pressure melt ejection are:

- melt mass expelled
- metal fraction in the expelled melt
- mass of water ejected
- coherence of melt and water dispersal from the reactor cavity

Investigators have used the SCDAP/RELAP5 model to analyze a variety of accidents particularly for the Zion reactor, which has a large, dry containment, and the Surry reactor, which has a subatmospheric containment to determine the ranges of values of the first three of these influential quantities. Uncertainty distributions have been developed for each of these based on the analyses. SCDAP/RELAP5 is NRC's best estimate code for analysis of core degradation. It must, however, be noted that recent tests conducted in the PHEBUS-FP program have called into question the accuracy with which SCDAP/RELAP5 predict both the relocation of molten core debris from the core region and the amounts of zirconium metal in this relocated core debris.

The coherence with which steam and molten core debris is dispersed from a reactor cavity has been derived from experiments using scale model reactor cavities. The derivation of the coherence ratio from these experiments that will be applicable at the full scale has not been without controversy. A model of the coherence ratio satisfactory to a panel of thermal hydraulic experts has been devised.

The uncertainty distributions for the melt mass and metal fraction in the melt have been used in the models to develop uncertainty distributions for the loads expected on containments during various accident scenarios involving various amounts of co-dispersed water. These uncertainty distributions for the loads have been compared to the uncertainty distributions for the structural capabilities of the containments at Zion and Surry. An example of such a comparison is shown in Figure 4.5-6. From these comparisons, it has been found that the conditional probability that direct containment heating will fail the Zion containment is less than 0.01.[4] Similarly low conditional containment failure probabilities have been found for Surry.[5]

The methodology developed for the Zion reactor and tested for the Surry reactor has been extrapolated to all Westinghouse reactor with large dry containments and subatmospheric containments.[6] It has been found for all these reactors that the conditional containment failure probability as a result of direct containment heating is less than 0.01.

The methodology has been extended to Combustion Engineering and Babcock and Wilcox reactors.[7] Combustion engineering plants are of particular concern because they typically have rather large annular gaps around the reactor vessel. Core debris expelled from the reactor vessel could disperse up through these gaps into the large volume of the reactor containment where heat transfer from the debris to the atmosphere could be most complete. Nevertheless, the conditional containment failure probabilities as a result of direct containment heating for all Combustion Engineering and Babcock and Wilcox plants were estimated to be less than 0.1 and in most cases less than 0.01.

Though they have automatic depressurization systems, boiling water reactors could be susceptible to high pressure melt ejection in some accidents. Hydrogen combustion in the drywells of inerted boiling water reactor containments will not be especially important, but pressurization as a result of heat transfer to the drywell atmosphere might be significant in these containments that are typically much smaller than pressurized water reactor containments. To date there has been no experimental investigations of high pressure melt ejection in boiling water reactors and only the most limited analytic studies.

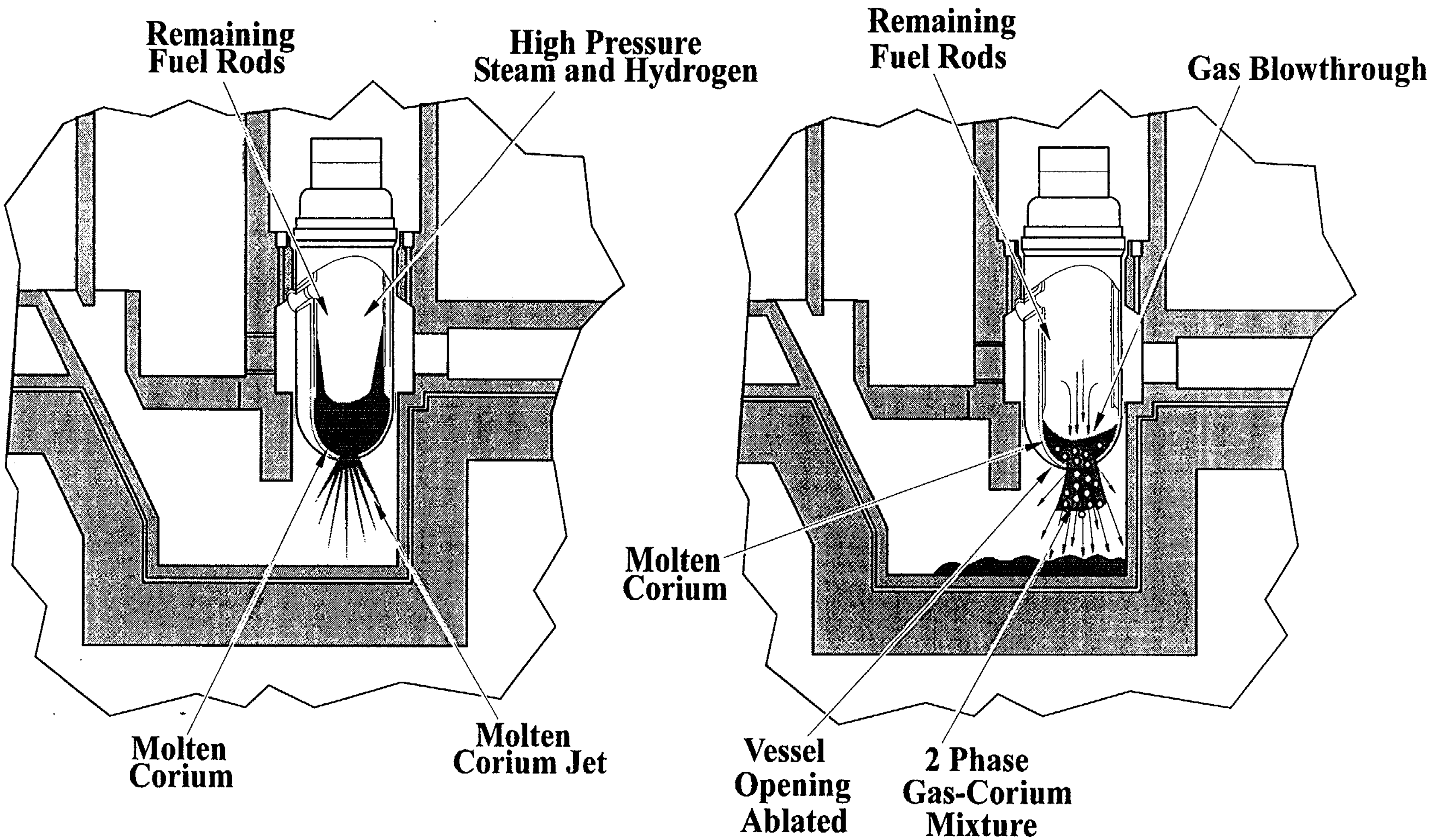


Figure 4.5-1 Melt ejection process

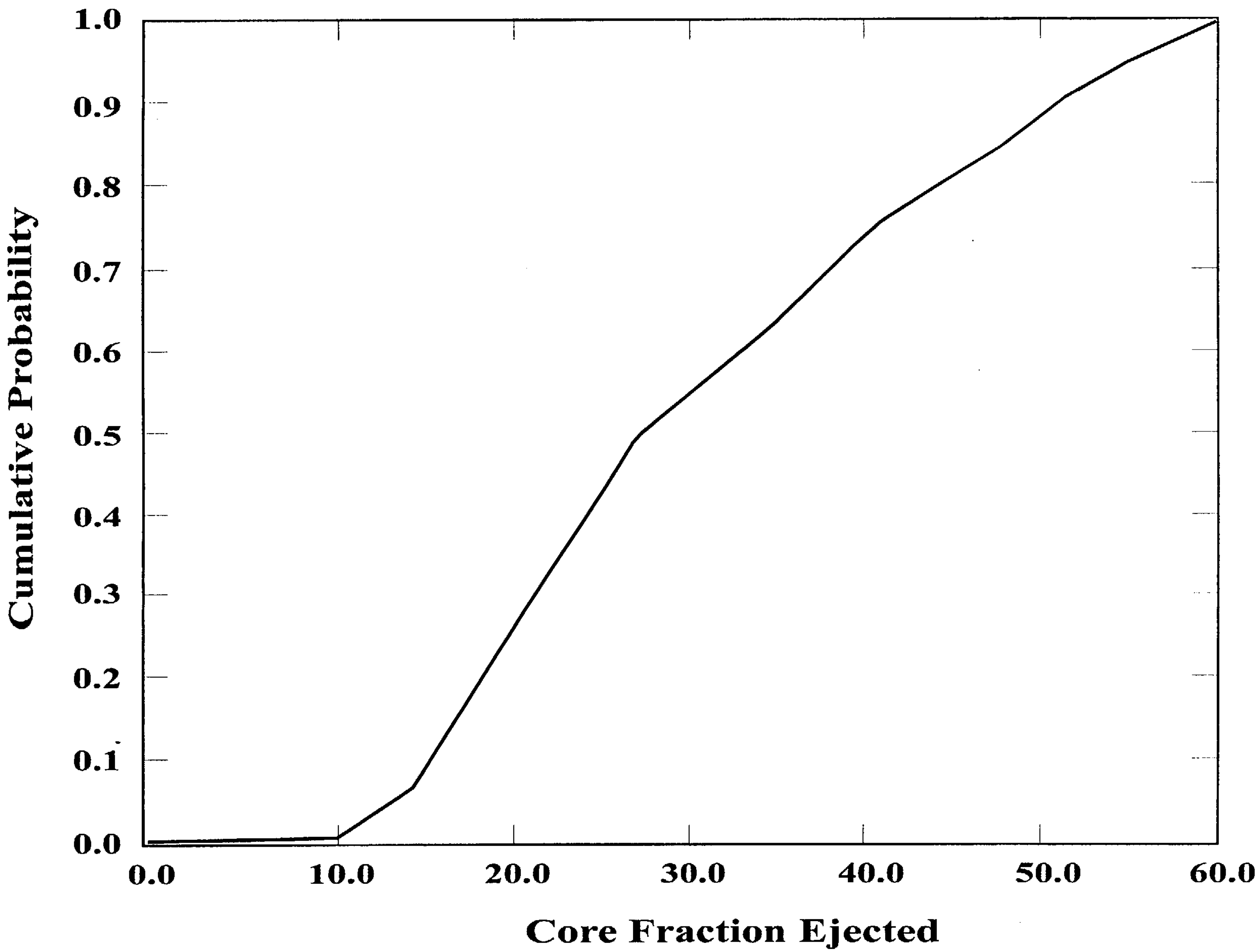


Figure 4.5-2 Distribution for fraction of core material ejected, PWR

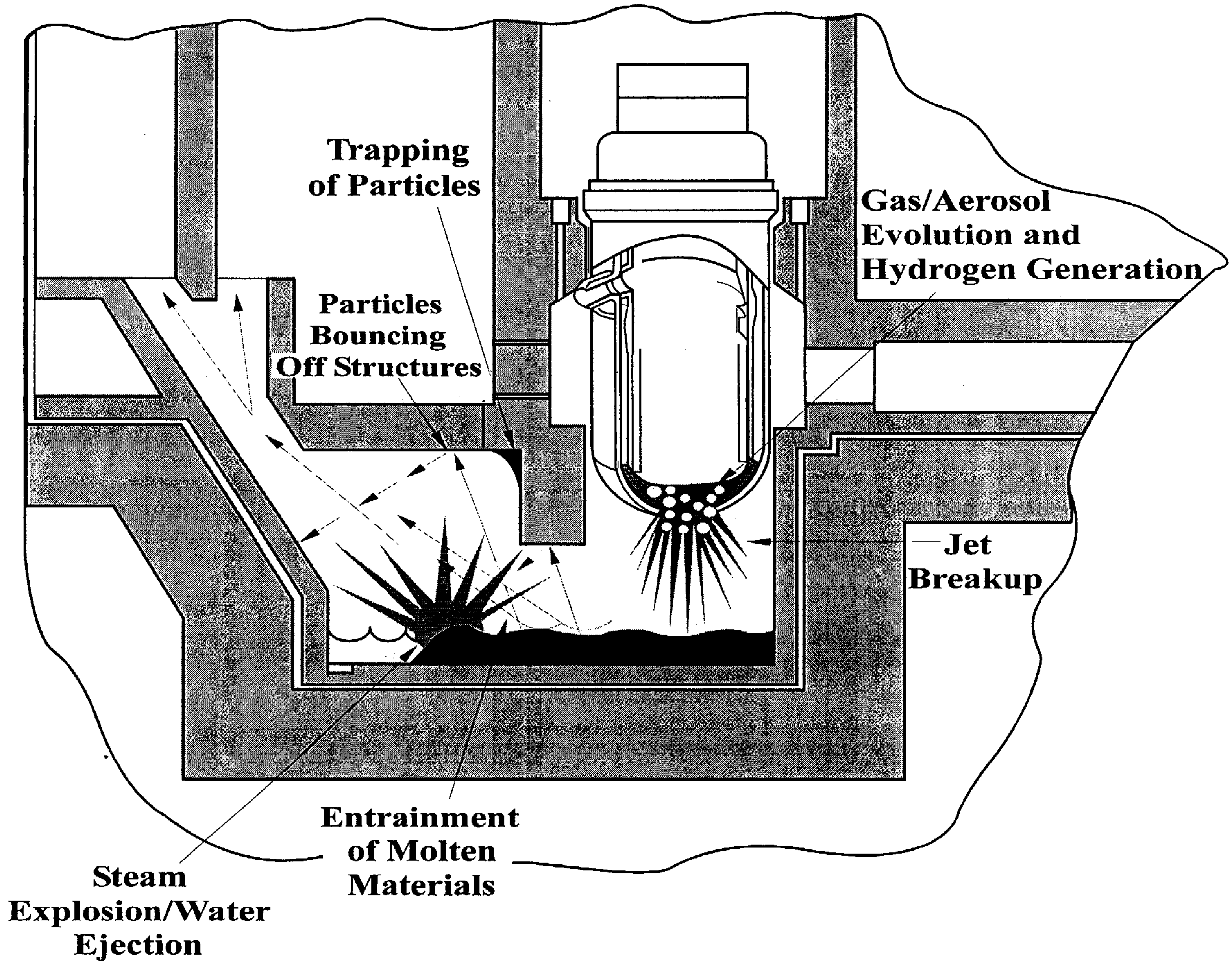


Figure 4.5-3 Reactor cavity interactions

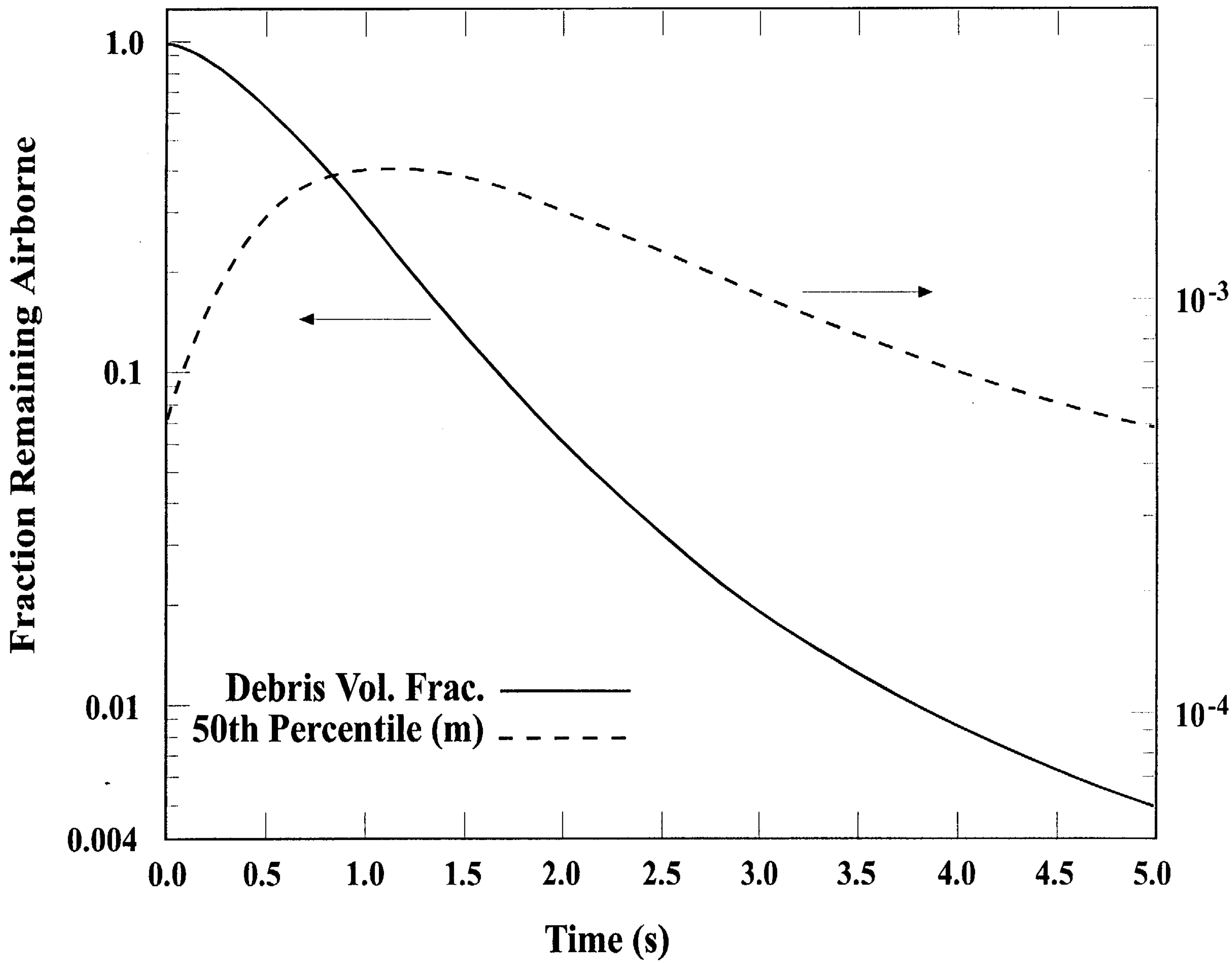
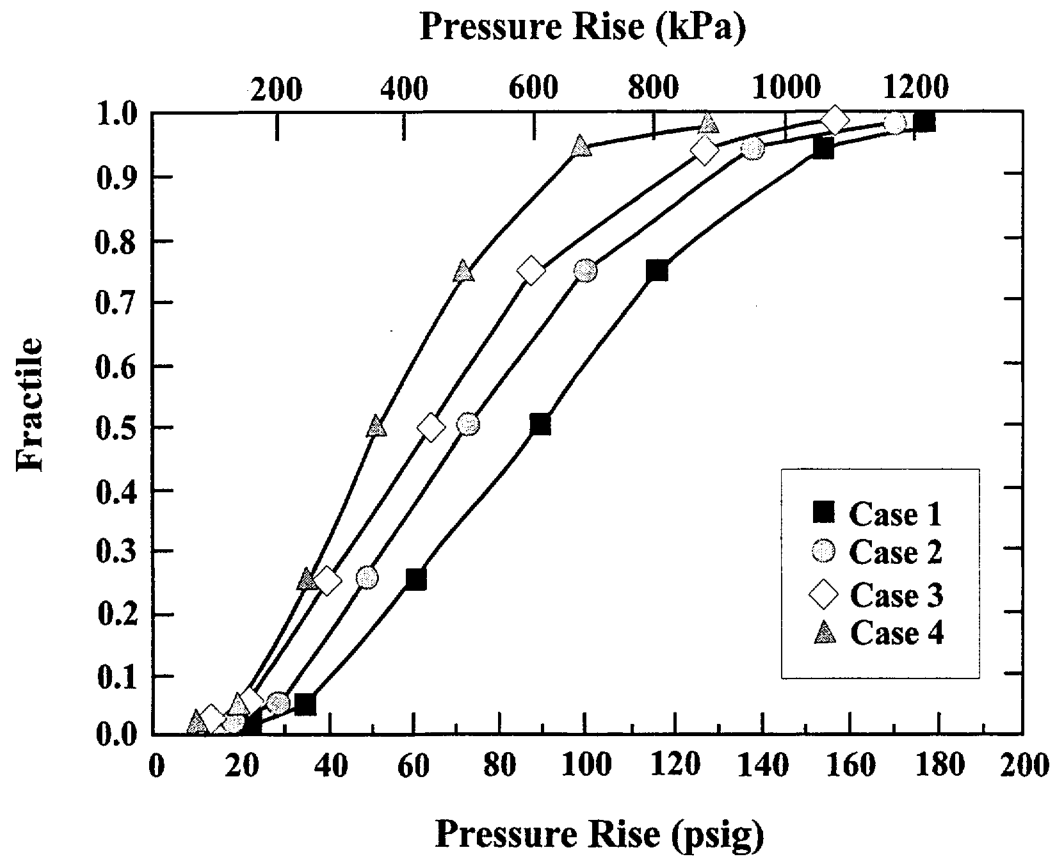


Figure 4.5-4 Estimated median particle size versus time



	RCS Pressure psi (MPa)	Vessel Hole Size ft^2 (m^2)	Cavity	Core Fraction Ejected
Case 1	2000-2500 (13.8-17.2)	21.5 (2)	Dry	40-60%
Case 2	2000-2500 (13.8-17.2)	21.5 (2)	Wet	40-60%
Case 3	500-1000 (3.4-6.9)	21.5 (2)	Wet	40-60%
Case 4	500-1000 (3.4-6.9)	1.1 (0.1)	Dry	20-40%

Figure 4.5-5 Example distributions for pressure rise at vessel breach, Surry

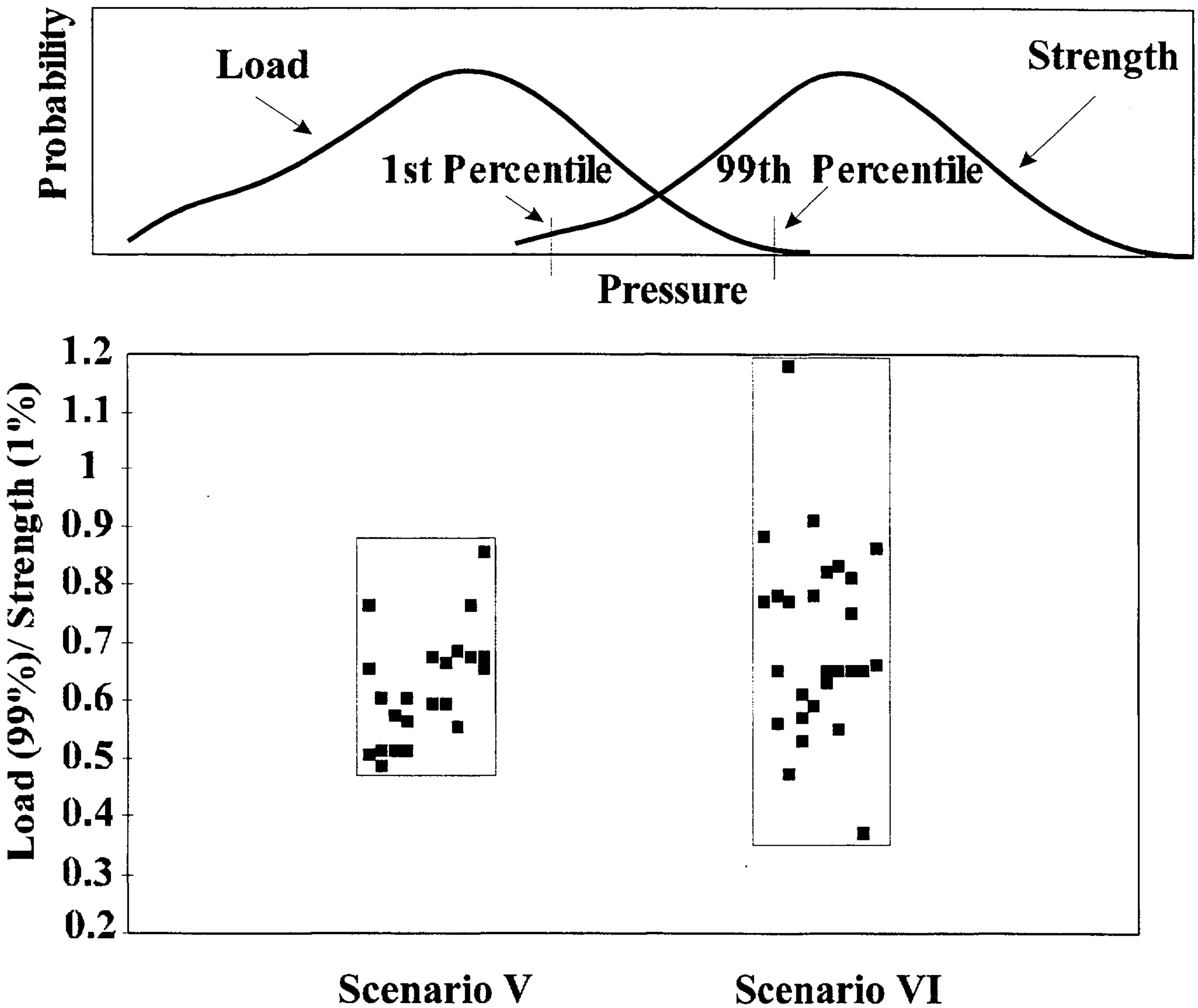


Figure 4.5-6 Large dry and subatmospheric containment results from DCH resolution effort

References for Section 4.5

1. M. M. Pilch, Michael D. Allen, and David C. Williams, "Heat Transfer During Direct Containment Heating," Academic Press, 1996.

2. F. T. Harper, et al., "Evaluation of Severe Accident Risks: Quantification of Major Input Parameters," NUREG/CR-4551, SAND86-1309, Vol. 2, Rev. 1, Part 1, Sandia National Laboratories, December 1990.

3. F. T. Harper, et al., Evaluation of Severe Accident Risks: Quantification of Major Input Parameters," NUREG\CR-4551, SAND86-1309, Vol. 2, Rev. 1, Part 2, Sandia National Laboratories, April 1991.

4. M.M. Pilch, et al., "The Probability of Containment Failure by Direct Containment Heating in Zion", NUREG/CR, SAND93-1535, Sandia National Laboratories, Albuquerque, NM, December, 1994

5. M.M. Pilch, et al., "The Probability of Containment Failure by Direct Containment Heating in Surry", NUREG/CR-6109, SAND93-2078, Sandia National Laboratories, Albuquerque, NM, May 1995

6. M. M. Pilch, et al., "Resolution of the Direct Containment Heating Issue for all Westinghouse Plants with Large Dry Containments or Subatmospheric Containments," NUREG/CR-6338, SAND95-2381, Sandia National Laboratories, February 1996.

7. M.M. Pilch, et al., "Resolution of the Direct Containment Heating Issue for Combustion Engineering Plants and Babcock and Wilcox Plants" NUREG/CR-6475, SAND97-0667, Sandia National Laboratories, Albuquerque, NM, November 1998.

4.6 Hydrogen Combustion

During the TMI-2 accident, hydrogen generated from in-vessel zirconium oxidation was released to the containment through the pressurizer relief valve. This hydrogen eventually ignited, resulting in a 28 psig (193 kPa) peak pressure in the containment. While this particular event did not threaten the TMI-2 containment, it raised awareness of the potential threats that might arise for other scenarios and for other containment types. The main concern over hydrogen combustion in nuclear reactor containments is that the high pressure generated might cause a breach of containment and a release of radioactivity. A second concern is that the resultant high temperature or pressure might damage important safety-related equipment. This section describes the physical mechanisms important to hydrogen combustion events, discusses the TMI-2 event in more detail, and describes the subsequent regulatory activities that have been taken to reduce the potential combustion threats. Much of the material in this section is excerpted from the *Light Water Reactor Hydrogen Manual.*[1]

4.6.1 Hydrogen Combustion Reaction

Combustion of hydrogen according to the reaction:

$$2H_2 + O_2 \dashrightarrow 2H_2O + \text{energy (heat)}$$

results in the release of about 5.2×10^4 Btu/lb-mol of hydrogen burned (57.8 kcal/gm-mole). Combustion waves are usually classified either as deflagrations or detonations. The term "explosion" usually refers to a detonation, but is somewhat ambiguous and should be avoided. Deflagrations are combustion waves in which unburned gases are heated by thermal conduction to temperatures high enough for chemical reaction to occur. Deflagrations normally travel subsonically and result in quasi-static (nearly steady state) loads on containment. Detonations are combustion waves in which heating of the unburned gases is due to compression from shock waves. Detonation waves travel supersonically and produce dynamic or impulsive loads on containment in addition to quasi-static loads. The pressure and temperature obtained from the complete combustion of hydrogen in air, adiabatically (without heat loss) and at constant volume, are shown in Figures 4.6-1 and 4.6-2. These figures show the ratio of initial to final pressures and final temperatures that could be expected for gas mixtures with low steam concentrations. Appendix A shows examples of pressure and temperature calculations for the types of air-steam-hydrogen mixtures that might occur in a reactor containment. In the following sections, the conditions necessary for combustion and the different combustion modes are discussed in detail.

4.6.2 Conditions Necessary for Combustion

Normally, for substantial combustion of hydrogen to take place, the gaseous mixture must be flammable, and an ignition source must be present. The special case of high temperature combustion is discussed later. For a flammable gas mixture, the flammability limits are defined as the limiting concentrations of fuel, at a given temperature and pressure, in which a flame can be propagated indefinitely. Limits for upward propagation of flames are wider than those for downward propagation. Limits for horizontal propagation are between those for upward and downward propagation.

The lower flammability limit is the minimum concentration of hydrogen required to propagate a flame, while the upper limit is the maximum concentration. At the lower limit, the hydrogen is in short supply and the oxygen is present in excess. At the upper limit of flammability for hydrogen in air, the oxygen is in short supply, about 5% oxygen by volume. The behavior of the upper limit of flammability of hydrogen with various mixtures such as air:steam is more easily understood if one considers it as the lower flammability limit of oxygen.

In large PWR containments we are usually interested in the lower limit of flammability, there being large amounts of oxygen present. In the much smaller BWR containments, particularly the inerted containments, we may be interested in the upper flammability limit.

For hydrogen:air mixtures, the flammability limits of Coward and Jones are still accepted.[2] Values for hydrogen flammability in air saturated with water vapor at room temperature and pressure are given in Table 4.6-1. These limits may vary slightly during accident conditions. There may be scale effects due to the large size of reactor containments as well as variations in flammability due to the ignition source strength.

In reactor accidents the conditions inside containment prior to hydrogen combustion may include elevated temperature, elevated pressure, and the presence of steam. The flammability limits widen with increasing temperature. For example, at 212°F (100 °C) the lower limit for downward propagation is approximately 8.8% (see Figure 4.6-3).

If the containment atmosphere is altered by the addition of carbon dioxide, steam, nitrogen, or other diluent, the lower flammability limit will increase slowly with additional diluent, while the upper flammability limit will drop more rapidly. With continued increase in diluent concentration, the two limits approach one another until they meet and the atmosphere is inerted. A flame cannot be propagated a significant distance for any fuel:air ratio in an inerted atmosphere. The addition of diluents has been proposed as a hydrogen mitigation strategy. Figure 4.6-4 shows the flammability limits with the addition of excess nitrogen or carbon dioxide. Note that for 75% additional nitrogen, the atmosphere is inert.[3,4] This corresponds to 5% oxygen at the limit of the flammable region, a value very close to that of the upper limit for hydrogen:air combustion. For carbon dioxide, the atmosphere is inerted when the carbon dioxide concentration is 60% or above, corresponding to 8% oxygen or less. The larger specific heat of carbon dioxide reduces the flame temperature and flame velocity; hence carbon dioxide suppresses flammability more than nitrogen. It requires about 60% steam to inert hydrogen:air:steam mixtures. The triangular diagram of Shapiro and Moffette indicates regions of flammability of hydrogen:air:steam mixtures.[4] It has been widely reproduced and appears as Figure 4.6-5.

Ignition of dry hydrogen:air mixtures, particularly when the mixtures are well within the flammability limits, can occur with a very small input of energy.[4] Common sources of ignition are sparks from electrical equipment and from the discharge of small static electric charges. The minimum energy required from a spark for ignition of a quiescent hydrogen:air mixture is of the order of 10^{-7} Btu (10^{-4}J) (a very weak spark). The ignition energy required as a function of hydrogen concentration is shown in Figure 4.6-6.[5] For a flammable mixture, the required ignition energy increases as the hydrogen concentration approaches the flammability limits. The addition of a diluent, such as steam, will increase the required ignition energy substantially. As mentioned previously, high energy ignition sources can cause mixtures outside the flammability limits to burn for some distance.

4.6.3 Deflagrations

Deflagrations are flames that generally travel at subsonic speeds relative to the unburned gas. Deflagrations propagate mainly by thermal conduction from the hot burned gas into the unburned gas, raising its temperature high enough for a rapid exothermic chemical reaction to take place. The propagation of a deflagration can be understood by examining the flammability limits discussed in the previous section. Consider a quiescent mixture of hydrogen:air. For hydrogen concentrations below about 4.1% there will be no significant propagation away from an ignition source. For hydrogen concentrations between 4.1 and 6.0%, there will be upward propagation from the ignition source. Hydrogen concentrations between 6.0 and 9.0% will produce both upward and horizontal propagation, and hydrogen concentrations above 9.0% will produce propagation in all directions, although the upward propagation may be faster than the downward propagation. Exact values for propagation limits will, of course, vary with temperature, pressure, and the presence of diluents. The degree of turbulence is also very important with turbulence tending to enhance combustion as long as the turbulence is not violent enough to "blow out" the flame.

It has been found in laboratory experiments that when hydrogen:air mixtures with hydrogen concentrations in the range 4-8% were ignited with a spark, some of the hydrogen was not burned.[6,7,8,9,10] The resultant pressure rise was below that predicted for complete combustion, as shown in Figure 4.6-7. Experimental results with a spark ignition source indicate that the completeness of combustion in quiescent mixtures increases with increasing hydrogen concentration, and is nearly complete at about 8-10% hydrogen. The range of incomplete combustion corresponds to the range in which the mixture is above the flammability limit for upward propagation, but below the flammability limit for downward propagation. As shown in Figure 4.6-7 for the "fans on" cases, turbulence and mixing of the gases can significantly increase the completeness of combustion. The additional variations in Figure 4.6-7 for mixtures below 8% tend to result from variations in the geometry and scale of the experiments.

Another important parameter when studying deflagrations is the flame speed. The flame speed determines how much time is available for heat transfer during a burn. Heat transfer results in pressures and temperatures below those predicted in Figures 4.6-1 and 4.6-2. The dominant heat transfer mechanisms are evaporation of containment sprays, radiation, and convection. Some plants also contain fan coolers. Normally, if the sprays are on, they will dominate the heat transfer process. Radiation heat transfer can also be important due to the high gas temperatures expected during a hydrogen burn. Convection may be less significant over the short time of a burn. One note is that the presence of sprays may significantly increase the flame speed due to the increased turbulence induced by the sprays. Typically, pressure rises above 80% of the adiabatic pressure rises are predicted for reasonable values of the flame speed, assuming complete combustion.

As shown in Figure 4.6-8, laminar burning velocities are quite slow. The laminar burning velocity (in a Lagrangian sense) denotes the speed of gases at a steady burner. Propagating laminar flames have flame speeds (in an Eulerian sense) which are 5-7 times faster due to volumetric expansion of the burned gases. The maximum laminar burning velocity of hydrogen:air mixtures is about 9.8 fps (3 m/s) near a concentration of about 42% hydrogen. The burning velocity becomes much smaller as the flammability limits are approached.

In a reactor containment, it is likely that a laminar deflagration will become turbulent. Turbulent flames can have average burning velocities 2 to 5 times the laminar burning velocity. Therefore, a hydrogen combustion event can occur in a containment in a matter of seconds, as opposed to the long times predicted by the laminar burning velocities. If the turbulent *flame speed* (laboratory system) becomes greater than about one-tenth the speed of sound (the speed of sound is approximately 1150 fps (350 m/s) in containment air), shock waves will be formed ahead of the flame front. In that case dynamic loads, in addition to static loads, will be imposed on the containment structure. The mechanisms leading to flame acceleration and detonation will be discussed in the next section.

4.6.4 Detonation of Hydrogen

A detonation is a combustion wave that travels at supersonic speeds relative to the unburned gas in front of it. For near stoichiometric hydrogen air mixtures this speed is about 6600 fps (2000 m/s) (see Figure 4.6-9). The compression of the unburned gas by shock waves in the detonation raises the gas temperature high enough to initiate rapid combustion.

We will attempt to answer as well as possible the following three questions:

1. Under what conditions is a hydrogen:air or hydrogen:air:steam detonation possible in containment?

2. If a detonation is possible, what is the likelihood that it will occur?

3. What pressure loads could a detonation cause?

We can answer the first question fairly well (at least with regard to hydrogen:air mixtures) and also the third question. The second question concerns the transition from deflagration to detonation and is still not completely understood after more than 50 years of investigation. We can say that, in most postulated reactor accident scenarios, deflagrations are much more likely than detonations.

4.6.4.1 Detonation Limits

Hydrogen:air mixtures near stoichiometric (about 29% hydrogen, two parts H_2 to one part O_2) are known to be detonable. Mixtures departing from stoichiometric, either in the hydrogen-lean or hydrogen-rich direction are increasingly more difficult to detonate. It has been observed that "detonation limits" are functions of geometry and scale, and not universal values at given mixture concentrations, temperatures and pressures.[11,12,13]

Our understanding of the possibility of sustaining a detonation in hydrogen:air mixtures, as well as other gas mixtures, has greatly increased within the last fifteen years. It has been found that a detonation wave is composed of unsteady oblique shock waves moving in an everchanging cellular structure (characterized by its transverse dimension), a "foamy" detonation front. Figure 4.6-10 shows the effect when a detonation passes by a smoked foil. The interacting shock waves form roughly diamond shape detonation “cells.”

The farther a mixture is from stoichiometric, and hence the less energetic the chemical reaction, the larger is the detonation cell size, λ. The cell width for hydrogen:air has been accurately measured over an extensive range of hydrogen:air ratios (see Figure 4.6-11).[13]

The knowledge of hydrogen:air cell size is valuable for evaluating detonation concerns in particular geometries. It is known that if a detonation is to propagate in a given geometry, there is a minimum size for which the detonation will propagate, related to the cell size. For smaller geometries, the detonation will fail. Figure 4.6-12 shows the relationship between various geometries and cell size. For example, at 16% hydrogen, the cell size is about 9.6 in. (24.5 cm). This means that a 16% hydrogen mixture detonation should be able to propagate down a tube 3.2 in. (8.2 cm) in diameter. The larger the tube diameter, the wider is the range of detonable hydrogen concentrations.

The detonability of a mixture is increased (cell size is decreased) with increasing temperature. For example, in a 17 in. (43 cm) tube at 68 °F (20 °C), a detonation can be propagated in a mixture with 11.7% hydrogen. At 212 °F (100 °C), the detonability limit changes to 9.5% hydrogen.[14]

The information provided above helps to answer the first question, "Under what conditions is a hydrogen:air detonation possible in containment?" The detonation limits are not fixed, but depend on the geometry and are wider for larger sizes and higher temperatures. The curve of cell size versus hydrogen fraction rises steeply on the hydrogen-lean side (see Figure 4.6-11). For the large geometrical scales in containments, detonations may propagate in leaner mixtures than has been demonstrated in small and medium scale experiments.

4.6.4.2 Transition to Detonation

A detonable mixture may only deflagrate (burn) and not detonate. Detonations can start directly by the use of a vigorous shock wave coming from a high explosive, strong spark, or laser. Approximately 0.035 oz. (1 gm) of tetryl explosive will initiate a spherical detonation of a stoichiometric hydrogen:air mixture. The increase in explosive charge required as the mixture departs from stoichiometric is roughly proportional to the increase in detonation cell size. Detonations can also start from deflagrations that accelerate to high speeds pushing shock waves ahead of the burn front until at some point shock heating is sufficient to initiate the detonation. Sources of such highly accelerated flames are high speed jets coming from semiconfined regions and flames passing through fields of obstacles.

Many obstacles that might potentially cause flame acceleration, such as pipes and pressure vessels, are present in the lower sections of most containments. Very fast burns may also occur due to the presence of a very intense ignition source, such as a jet of hot combustion products formed subsequent to ignition in some adjoining semi-confined volume.

Deflagration-to-detonation transition is probably the least understood aspect of detonation theory at this time. Measurements have been made of the distance required to have transition to detonation in smooth tubes. Distances many times the tube diameter have been required. If obstacles are inserted into the tube, the required distance to detonation is greatly reduced. The motion of the expanding gases around the obstacles leads to greatly increased flame front area, rapid flame acceleration and rapid transition to detonation. Confinement greatly promotes transition, but one cannot rule out transition to detonation in a containment if a detonable mixture of sufficient size is present. The second question, "If a detonation is possible, what is the likelihood that it will occur?"

therefore cannot be answered with certainty at present.

4.6.4.3 Detonation Pressures and Temperatures

For the purpose of studying the pressures and temperatures caused by a detonation, it is sufficient to ignore the detonation wave structure and consider it as a thin surface, a discontinuity. Chapman and Jouguet assumed that the detonation traveled at a speed that was sonic relative to the unburned gas. With this assumption one can compute a unique detonation speed for each hydrogen:air mixture, and find the corresponding temperature and pressure behind the detonation wave. The results are shown in Figures 4.6-13 and 4.6-14. It is an experimental fact that the measured speeds of detonations are approximately equal to the calculated Chapman-Jouguet values.

The burned gases behind a detonation are moving in the direction of the detonation. When a detonation hits a rigid wall, the gases must be brought to rest. This is accomplished by a reflected shock wave. We will consider only the case of a detonation wave striking a wall at normal incidence. The reflected shock wave further compresses the burned gas, increasing the detonation pressure by a factor of about 2.3. The pressures and temperatures predicted behind the normally reflected shock wave are also shown in Figures 4.6-13 and 4.6-14. In a containment one expects wave reflections from walls and obstacles to give rise to complex shock wave patterns. Wave interactions may lead to dissipation or, possibly, to wave focusing which can give rise to very high local peak pressures.

4.6.4.4 Local Detonations

In all the previous sections on detonations it has been assumed that the detonation is taking place in a homogeneous combustible mixture. Such detonations are global, traveling throughout the containment. With the exception of the strongest containments, containments will probably not be able to withstand the quasi-static pressure (adiabatic isochoric pressures) generated after the detonation, even without the additional dynamic loads due to detonation. It is therefore more appropriate to consider the effect of detonations when only a local portion of the containment atmosphere is detonable.

Consider a detonable cloud of hydrogen:air surrounded by air. As the detonation wave leaves the cloud, it will change into an expanding decaying shock wave. The shock wave intensity drops fairly rapidly if the shock wave expands spherically. Within a distance equal to 3 cloud radii, the shock wave pressure will drop to a value low enough to no longer threaten the containment structure. However, it has been found in detailed computer calculations that, because of the containment geometry, the shock waves may be focused in local regions, such as the top center of the containment dome, giving rise to large local peak pressures and impulses.[15,16] Local detonations may be dangerous in and near the detonable cloud, and may be dangerous at locations farther away if shock focusing effects are significant.

There are several locations to consider where high hydrogen concentrations are possible. These include:

1. near the hydrogen release point,

2. under ceilings or in the dome due to the rise and stratification of a low density plume, or

3. near steam removal locations such as ice condensers, suppression pools, and fan coolers.

A detonable mixture requires adequate hydrogen and oxygen, but not too much steam. Regions of stratification tend to be difficult to establish and maintain in a turbulent containment environment. Steam removal locations are generally a more significant concern for local detonations.

4.6.4.5 Missile Generation

Missiles may be generated when combustion (deflagration or detonation) occurs in a confined region or when a propagating combustion front produces dynamic pressure loads on equipment. Such missiles may pose a threat to the containment structure itself, as well as representing a potential threat to safety and control equipment. For instance, electrical cables may not be expected to withstand the impact of a door or metal box. The actual risk to plant safety posed by missiles generated from hydrogen combustion depends upon a number of independent factors and is very difficult to predict.

4.6.5 Continuous Combustion

The preceding discussions have dealt with the discrete combustion events associated with hydrogen:air:steam mixtures in containment. There are also mechanisms for continuous combustion that are possible in some containments and for certain accident scenarios. Hydrogen may enter the containment as part of a turbulent jet from a pipe break or relief valve or may enter as part of a buoyant plume from the top of a suppression pool or from core-concrete interactions. The hydrogen may be accompanied by large quantities of steam or, in the case of core-concrete interactions, carbon monoxide which is also flammable. The primary threat to nuclear power plants from continuous combustion is the temperature rise and the possible effect on equipment and structures. Pressure increases from continuous combustion will not generally threaten the containment.

Hydrogen that enters the containment may start to burn as a turbulent diffusion flame. A diffusion flame is one in which the burning rate is controlled by the rate of mixing of oxygen and fuel. The nature of the flame is determined by the Froude Number, which is the ratio of the momentum forces to the buoyant forces in the jet or plume. Figure 4.6-15 shows the types of flames that can occur for different source diameters and flow rates. For the hydrogen to burn, it is necessary that at some location the hydrogen:air:steam mixture be within flammability limits.

Combustion can begin either because of an outside ignition source, or because the mixture temperature is above the spontaneous ignition temperature. Shapiro and Moffette in 1952 presented experimental results on the spontaneous ignition temperature of hydrogen:air:steam mixtures (see Figure 4.6-16).[17] The spontaneous ignition temperature is in the range of 959-1076 °F (515-580 °C). Above this temperature, combustion can occur without external ignition sources such as electrical sparks. For example, continuous combustion may occur in a reactor cavity above CCI in a dry cavity. In this case, the combustion will be limited by the availability of oxygen. However, if any oxygen is present, hydrogen and carbon monoxide can react even if the mixture is not within normal flammability limits.

Turbulent jets, such as from a pipe break, tend to autoignite at higher temperatures than buoyant plumes. Experiments have shown that such jets can autoignite at temperatures above 1166 to 1346°F (630-730°C).[18] A stable flame will occur at a distance from the orifice such that the turbulent burning velocity is equal to the gas flow velocity. There is evidence to suggest that for a particular set of conditions (temperature, pressure, and composition), there is a minimum orifice diameter for flame stability.[19] This minimum diameter is typically on the order of a few hundredths of an inch (millimeters) or less, and therefore, all practical sized orifices will support a stable hydrogen flame. Turbulent jets of hydrogen can also accompany direct containment heating. Hydrogen may already be present in containment, with additional hydrogen coming from in-vessel and from oxidation reactions during the melt ejection process. The hot particles and high temperature gases will serve to ignite the hydrogen, resulting in an additional energy contribution to the direct containment heating process. As noted in Section 4.5, very rich mixtures of hydrogen may be found at the exit of a reactor cavity, raising the possibility of a detonation. However, in this case the mixture may be steam rich and oxygen starved near the release point.

4.6.6 Combustion at TMI-2

The TMI-2 accident was discussed at some length in Chapter 2. During the core heatup and degradation process, hydrogen was generated and released to containment through the pressurizer relief valve and the quench tank. Estimates of the total amount of hydrogen generated range from 594 to 814 lb_m (270 - 370 kg).[20] This amount of hydrogen corresponds to oxidation of about 40% of the zirconium in the core. Approximately 9 hours and 50 min. into the accident, a hydrogen deflagration occurred, resulting in a 28 psig peak pressure in containment (see Figure 4.6-17). The ignition source is not known, but could have been an electrical spark from a variety of sources.

The pressure rise observed at TMI-2 is consistent with the estimates of the generation and relatively complete combustion of between 7 and 8.2% hydrogen. The TMI-2 containment has a volume in excess of 2×10^6 ft^3 (5.7×10^4 m^3) and a failure pressure far in excess of 28 psig (193 kPa). However, BWR containments and PWR ice condenser containments are much smaller than TMI-2, and the same quantity of hydrogen could have resulted in a detonable mixture in those containments. The realization that hydrogen combustion could cause containment failure in smaller containments led to regulatory actions, as discussed in the following section.

4.6.7 Hydrogen Control Requirements

In general, there are very few regulations and guidelines dealing with beyond-design-basis accident phenomena in reactor containments. For example, there are no specific rules dealing with CCIs, ex-vessel steam explosions, or direct containment heating. Hydrogen control has been an exception to this approach, with significant regulations passed following the TMI-2 accident.

Limited hydrogen control was provided prior to TMI-2 in the form of hydrogen recombiners that could remove the small amounts of hydrogen that might be generated during a design-basis LOCA. However, these recombiners have virtually no value for the large quantities of hydrogen that could be generated during a severe accident. Therefore, the NRC took additional steps to protect the reactors considered most vulnerable to hydrogen combustion.

The hydrogen rule is contained in 10 CFR 50.44.[21] In 1981, the NRC ordered that all

BWRs with Mark I and Mark II containments be inerted during normal operation to preclude the possibility of combustion. These containments are small enough that relatively low levels of zirconium oxidation could produce detonable mixtures in containment. Although inerting will prevent combustion within the containment, hydrogen can enter the surrounding reactor building of a Mark I or II containment if the containment fails or is vented through structurally inadequate flow paths. This hydrogen can burn, presenting a thermal hazard for safety equipment located in those buildings.

BWR Mark III containments and PWR ice condenser containments were the object of long and controversial examination and are still being examined today. A variety of hydrogen control measures were considered by both the industry and the NRC. These measures included inerting, partial inerting, water fogs and foams, and deliberate ignition systems. Because of the need to enter containment for various operational activities and risks to personnel, the utilities opposed inerting approaches. Some other approaches, such as water fogs and foams, were not successfully demonstrated as practical prior to the decisions that were reached. Ultimately, the industry and NRC agreed on the deliberate ignition approach, even though other options are allowed under 10 CFR 50.44. The deliberate ignition approach is discussed in more detail below.

The acceptance of deliberate ignition as a viable strategy is based in part on a couple of controversial assumptions in the hydrogen rule. The TMI-2 accident did not result in vessel breach, and only about half of the available zirconium was oxidized. Therefore, the hydrogen rule was set up to address only degraded core accidents and not full scale melting and vessel breach. Consistent with the assumption that vessel breach does not occur, the limit of zirconium oxidation was set to 75% of the fuel cladding, not including channel boxes in BWRs. Greater amounts of hydrogen were not expected to be consistent with an accident in which most of the core did not melt or the vessel was not breached. Further, because the vessel is not breached, the release of hydrogen to containment was expected to occur over time periods of at least many minutes, if not longer. The large puff release that might accompany vessel breach, hot-leg rupture or ex-vessel steam explosions does not need to be considered in meeting the hydrogen rule. It is also interesting to note that, while the fuel damage is assumed to be arrested at some point, the reflooding process is assumed to not produce oxidation in excess of 75% and to not result in a large burst of hydrogen. Therefore, only a select subset of beyond-design-basis accidents is actually addressed.

Deliberate ignition is based on the premise that hydrogen can be burned off in small quantities as it enters the containment. Either numerous small deflagrations or continuous combustion may occur, resulting in minimal pressure rise in containment, although the temperature effects must be considered. If the containment is not steam-inerted, then lean mixtures will be combusted until either the hydrogen or oxygen is depleted. As shown in Figures 4.1-5 and 4.1-9, igniters are located throughout containment to assure that locally high concentrations of hydrogen are avoided. These igniters are typically glow plugs, requiring AC power to function.

There are some limitations and concerns associated with igniters. First, they require AC power and will not function during station blackout. Further, if the containment is filled with hydrogen and power is later restored, they could provide a distributed ignition source if the operators do not think to keep them turned off.

Second, there are two regions where higher than average hydrogen concentrations are possible. One is within an ice condenser and the other is above a Mark III suppression pool. In both cases, a steam-rich mixture may enter the condensing region, and the gas may emerge very hydrogen-rich. This is particularly true for rapid releases of hydrogen. A third concern relates to accidents more severe than degraded core accidents and to reflooding. Very rapid releases of hydrogen, such as associated with vessel breach or late reflooding, may overwhelm the igniters so that the effect is the same as for a large deflagration. A fourth possibility concerns sequences in which the containment sprays do not function and the containment becomes steam inert. If the hydrogen accumulates in the inert atmosphere, and the sprays are later recovered, combustion may occur when the containment deinerts. If the combustion occurs early in the deinerting process, while significant steam is still present, a fairly weak burn is expected.

Despite the concerns raised above, hydrogen igniters are expected to have a positive benefit in many accidents. However, persons responsible for managing accidents need to be aware of the possibilities and use the igniters appropriately.

No additional hydrogen controls have been required for large dry or subatmospheric containments. These containments are large enough and strong enough that deflagrations are not expected to threaten them, except in conjunction with other phenomena. Local detonations are possible, but not considered likely for many accidents.[22] Detonable mixtures involving most of the containment can not be achieved without complete oxidation of all zirconium, *plus* additional hydrogen generation from steel oxidation or core-concrete interactions. A large detonation would require all of this hydrogen to be generated, that none of it burn previously, and that the burn undergoes a transition to a detonation. This combination of events is considered unlikely.

4.6.8 Risk-Informed Changes to the Hydrogen Rule

As PRAs were performed throughout the 1980s and early 1990s, it became clear that hydrogen combustion was not a major threat to most large, dry and subatmospheric containments. Further, it was clear that hydrogen recombiners contributed very little to public safety, due to their limited capacity to mitigate large hydrogen releases from the reactor coolant system. In 1998, San Onofre requested an exemption to the requirement for hydrogen recombiners. This exemption was granted, thereby relieving the plant of the testing, maintenance, and technical specification burdens associated with their operation.

As the NRC moved forward with risk-informing the regulations, 10CFR50.44 became one of the first regulations addressed, due in part to industry requests. It is desirable to issue a rule change as opposed to granting individual exemption for many plants. Rulemaking is not yet complete for the new rule; however, the rule is likely to allow the elimination of hydrogen recombiners for large, dry and subatmospheric containments. For BWR Mark I and II containments, no major changes are anticipated.

Risk-informing the hydrogen rule for ice condenser and BWR Mark III containments is more complex. As noted earlier, these intermediate-sized containments have ac-powered igniter systems for hydrogen control. PRAs indicate that the risk from station blackout and similar accidents can be important. Therefore, it is possible that the risk-informed hydrogen rule could stipulate additional requirements for these two containment types. For example, DC backup power or other backup power sources for the igniters could be needed. Given that implementation of the risk-informed rule is voluntary, it is unlikely that any substantive changes will be made at these plants.

Table 4.6-1 Hydrogen flammability limits in steam-saturated air at room temperature

	Lower Limit Vol. % of Hydrogen	Upper Limit Vol. % of Hydrogen
Upward Propagation	4.1	74
Horizontal Propagation	6.0	74
Downward Propagation	9.0	74

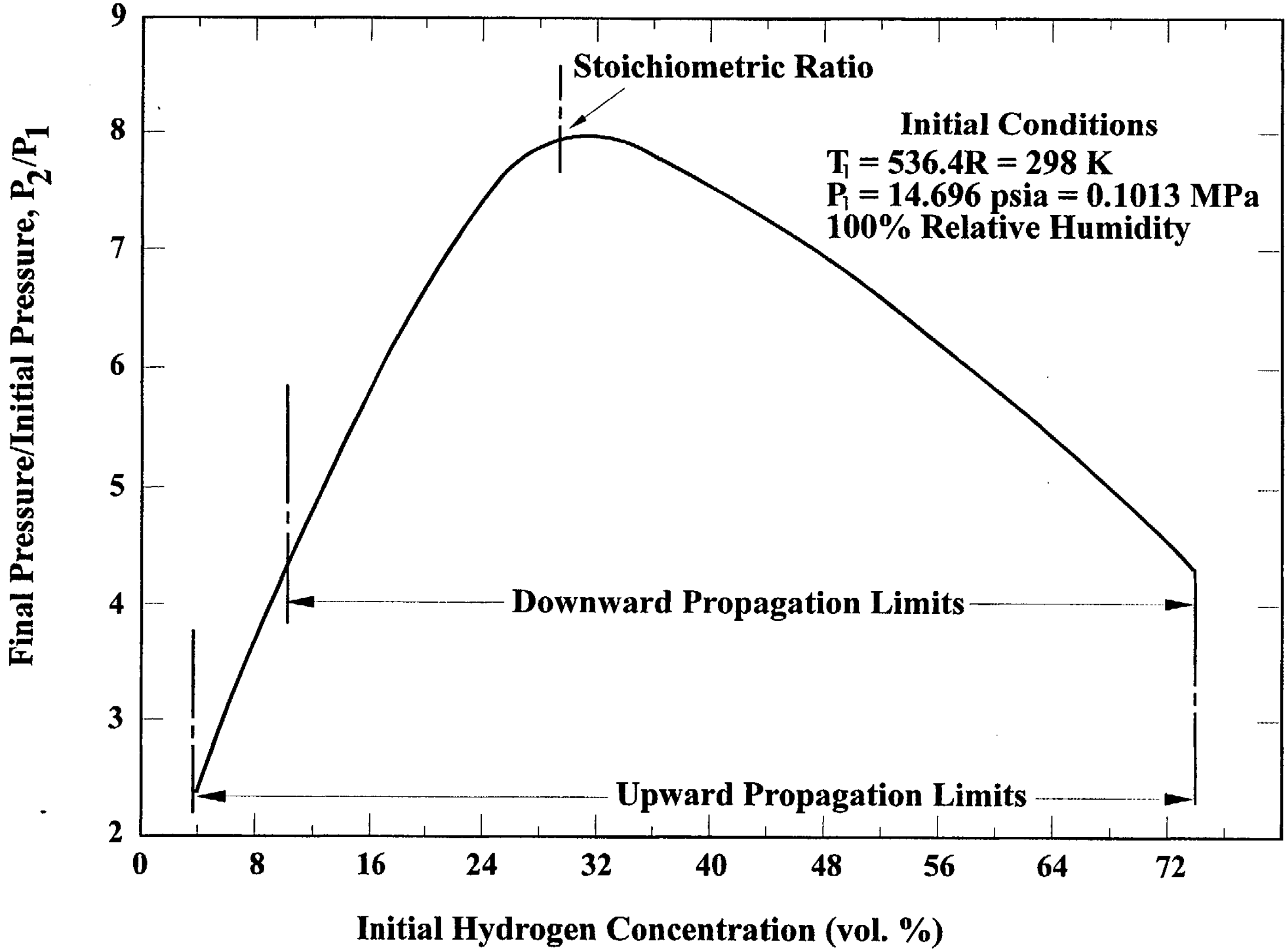


Figure 4.6-1 **Theoretical adiabatic, constant-volume combustion pressure for hydrogen : air mixtures**

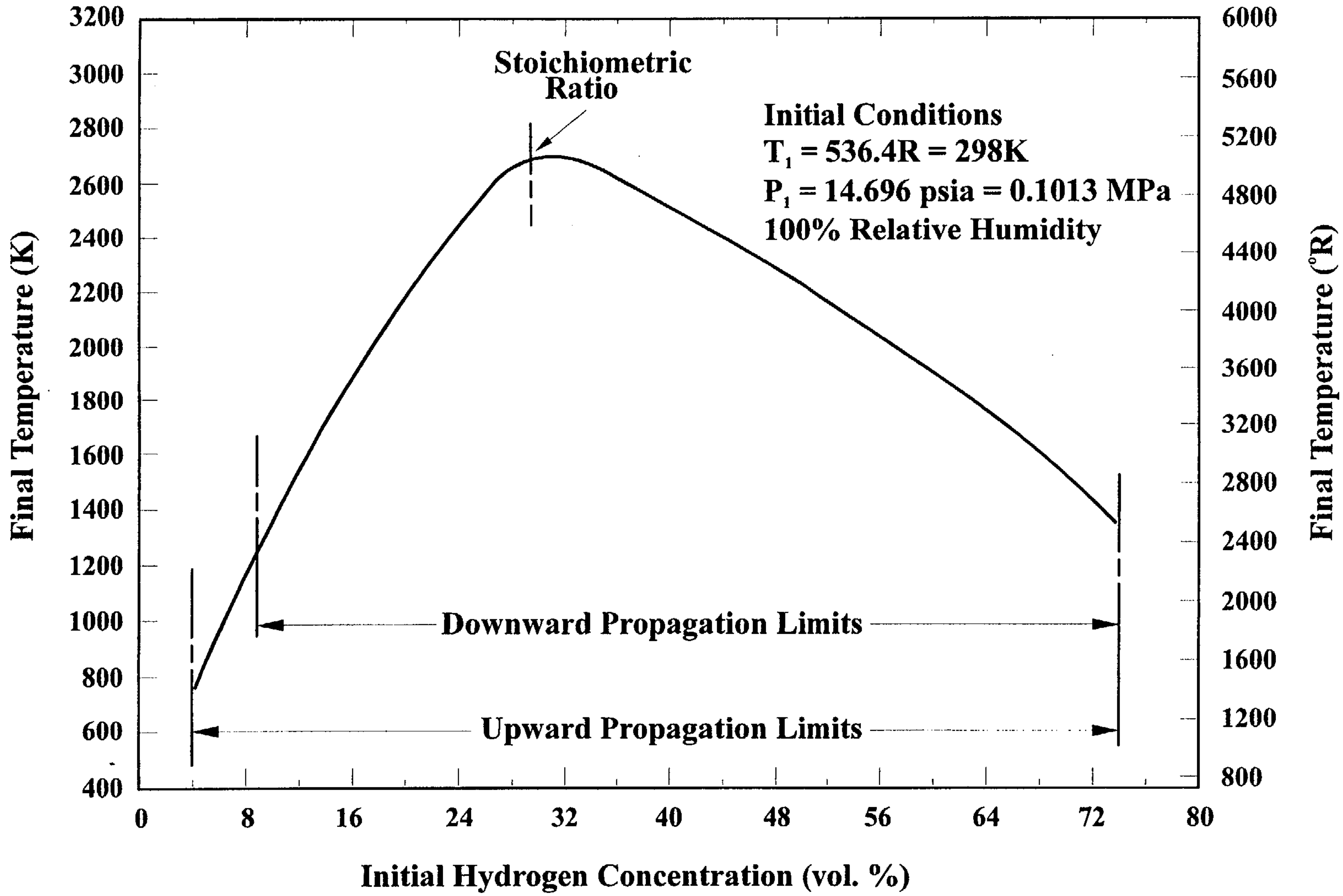


Figure 4.6-2 Theoretical adiabatic, constant-volume combustion temperature for hydrogen : air mixtures

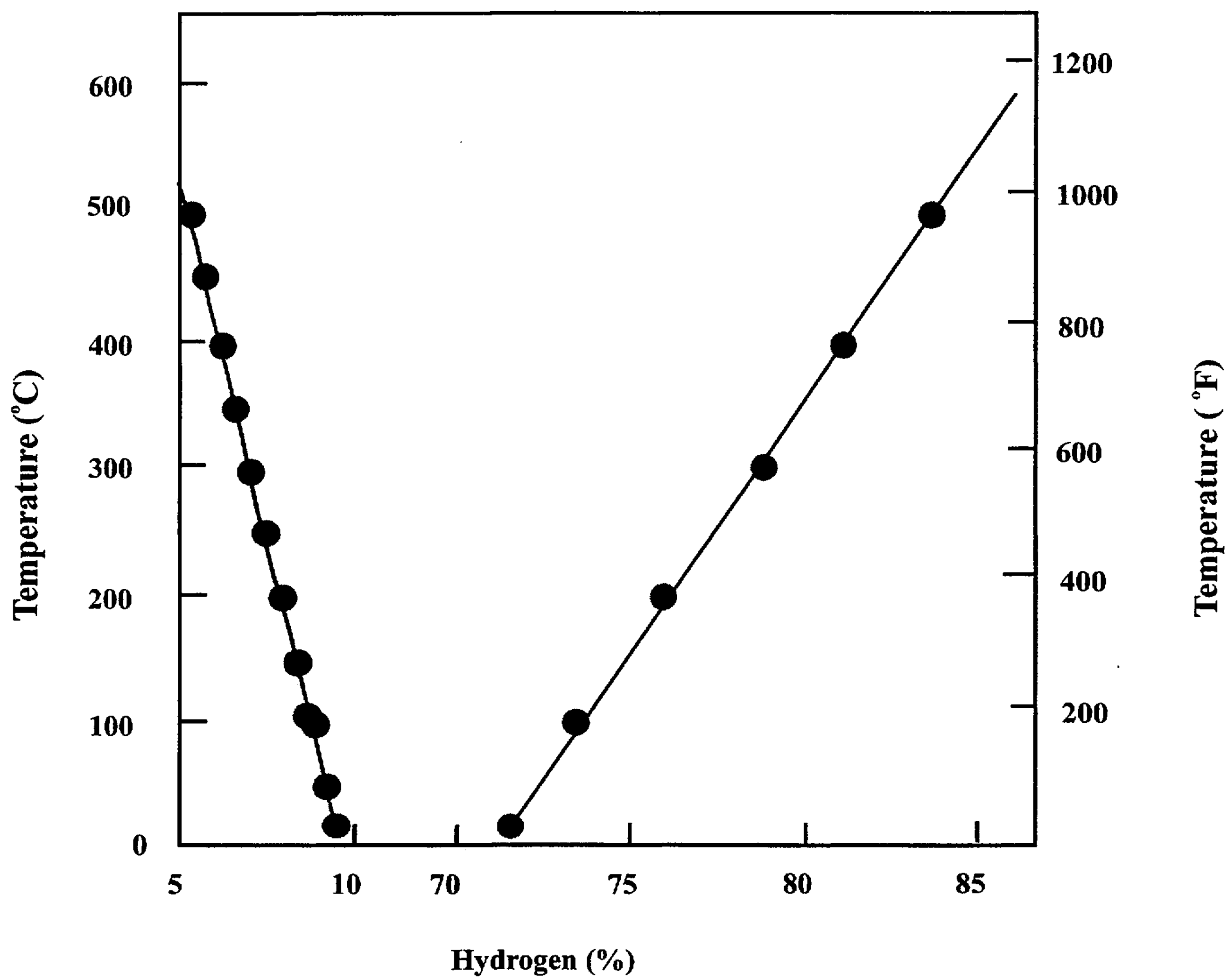


Figure 4.6-3 Effect of initial temperature on downward propagating flammability limits in hydrogen : air mixtures

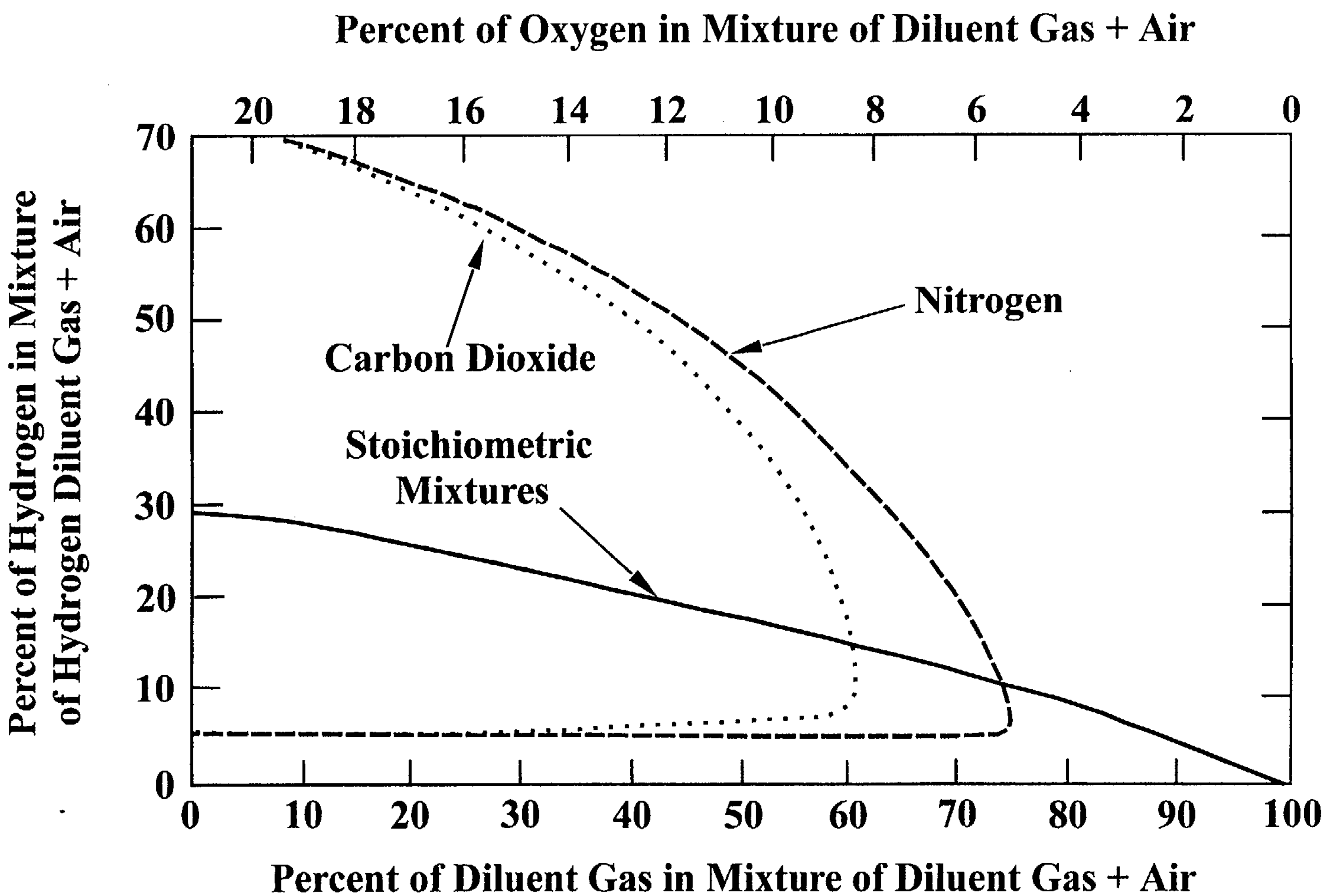


Figure 4.6-4 Flammability limits of hydrogen in air diluted with CO_2 and N_2

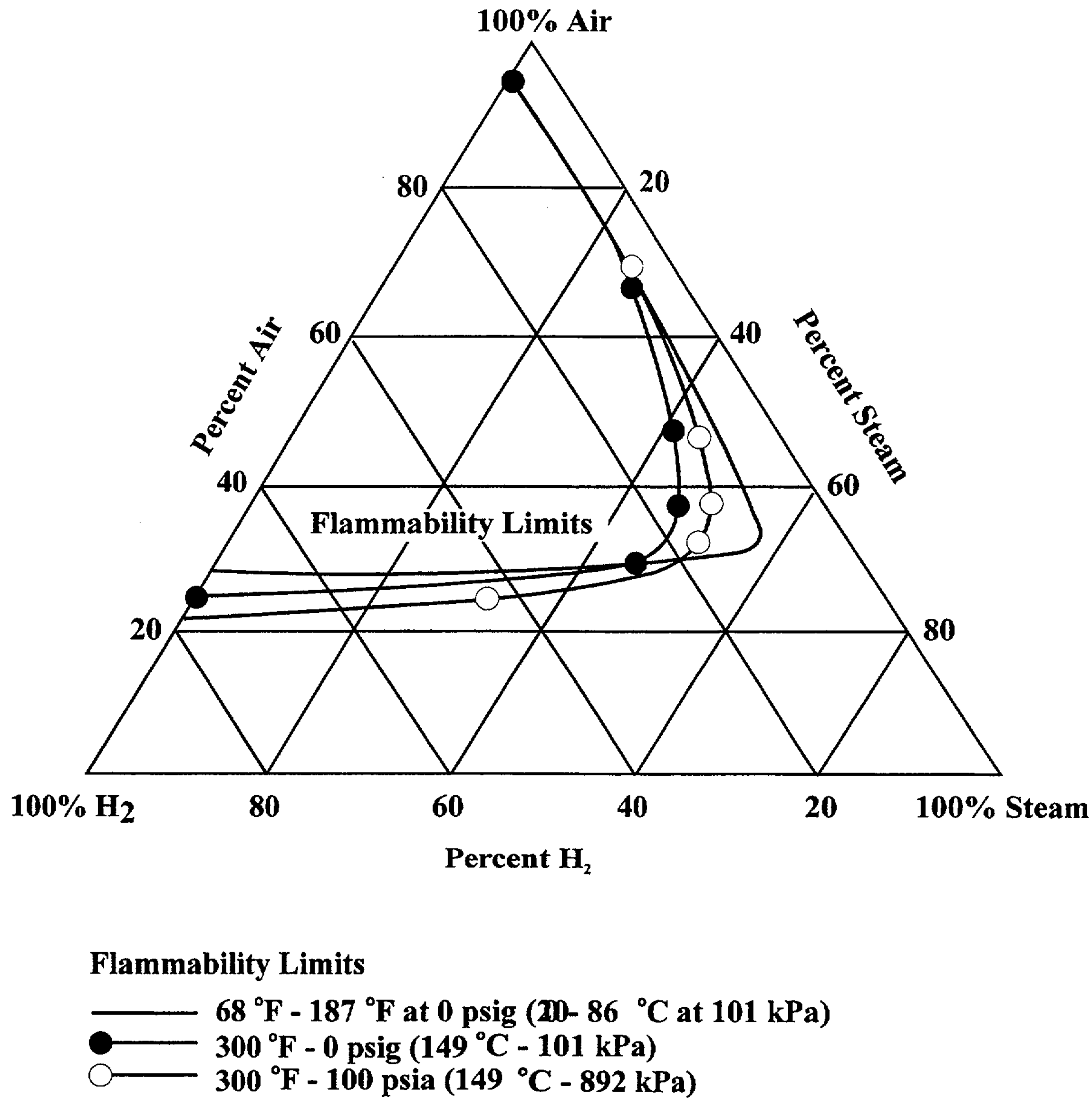


Figure 4.6-5 Flammability limits of hydrogen : air : steam mixtures

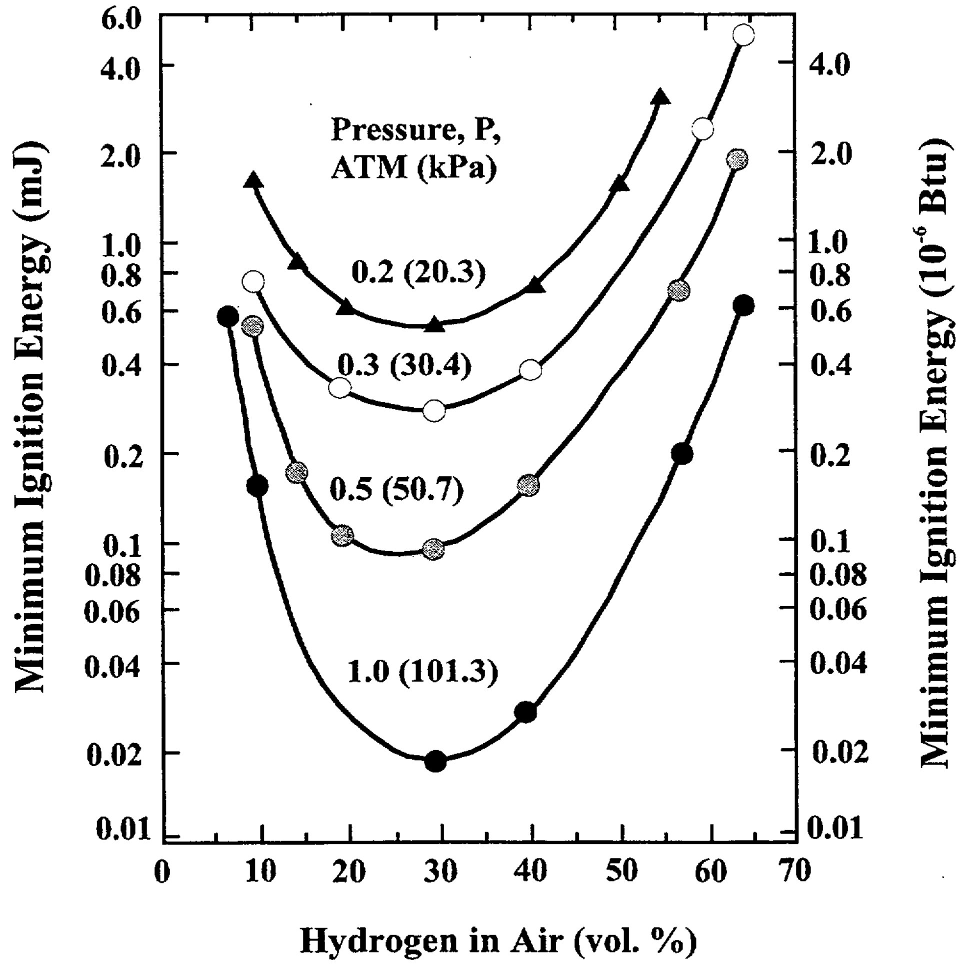


Figure 4.6-6 Spark ignition energies for dry hydrogen : air mixtures

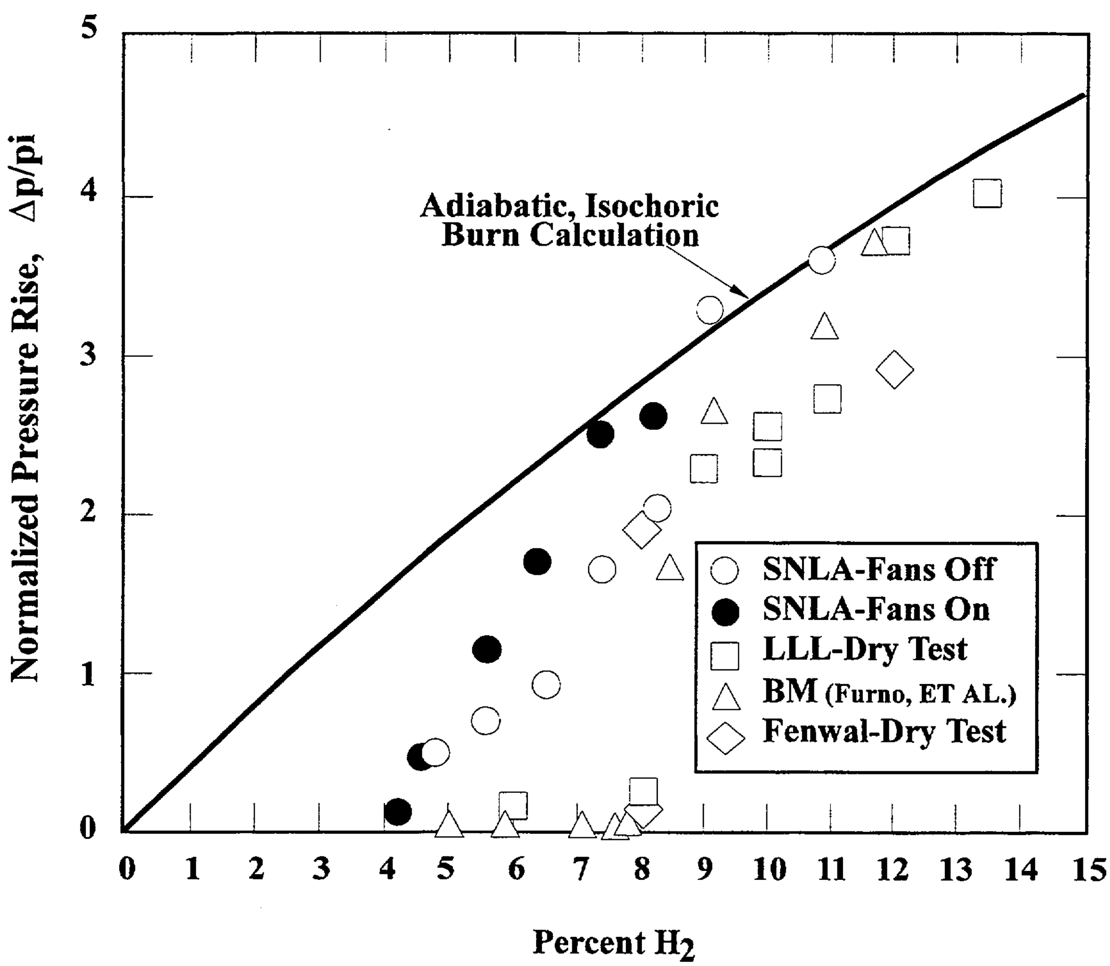


Figure 4.6-7 Normalized pressure rise versus hydrogen concentration

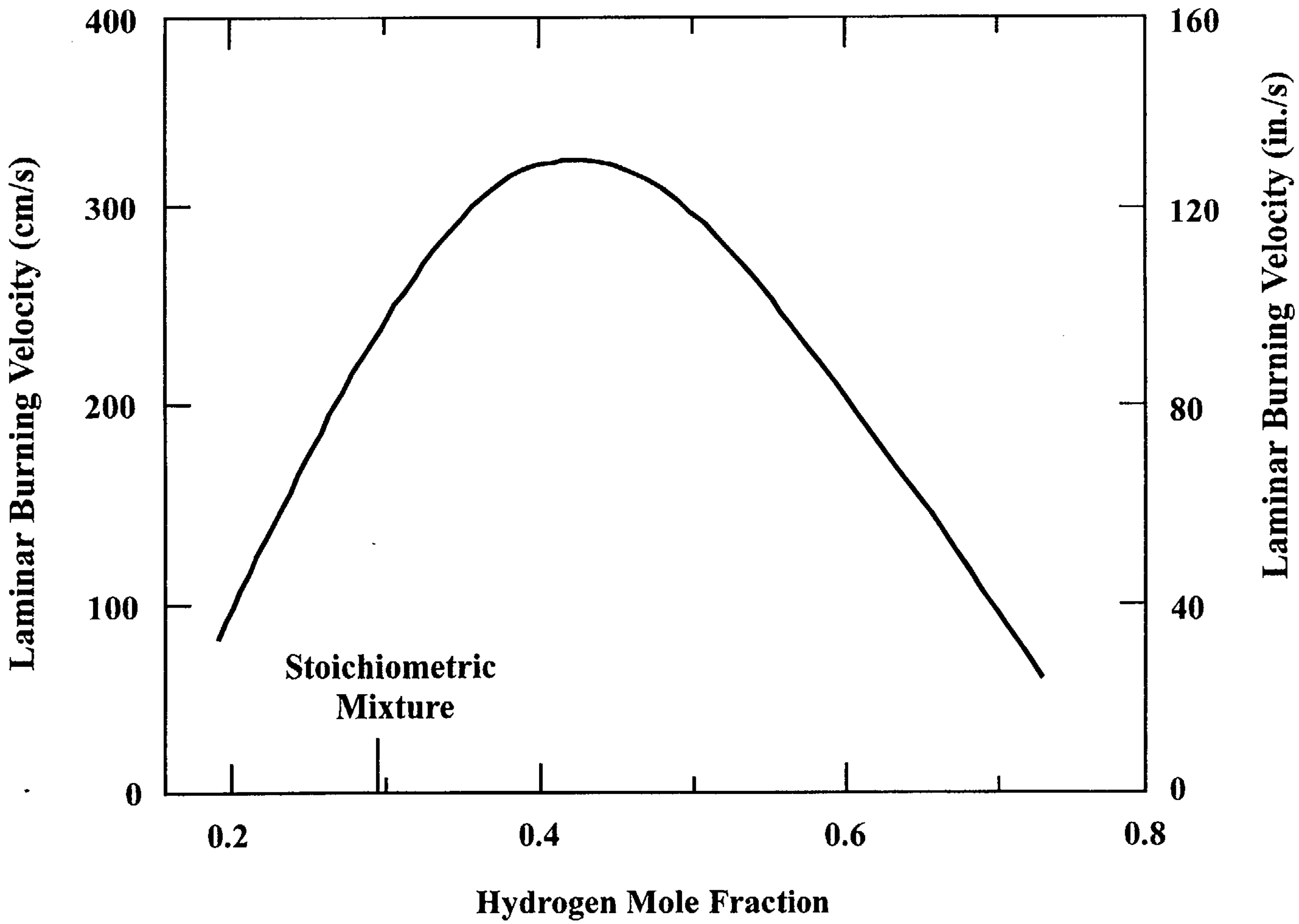


Figure 4.6-8 Laminar burning velocity of hydrogen : air mixture

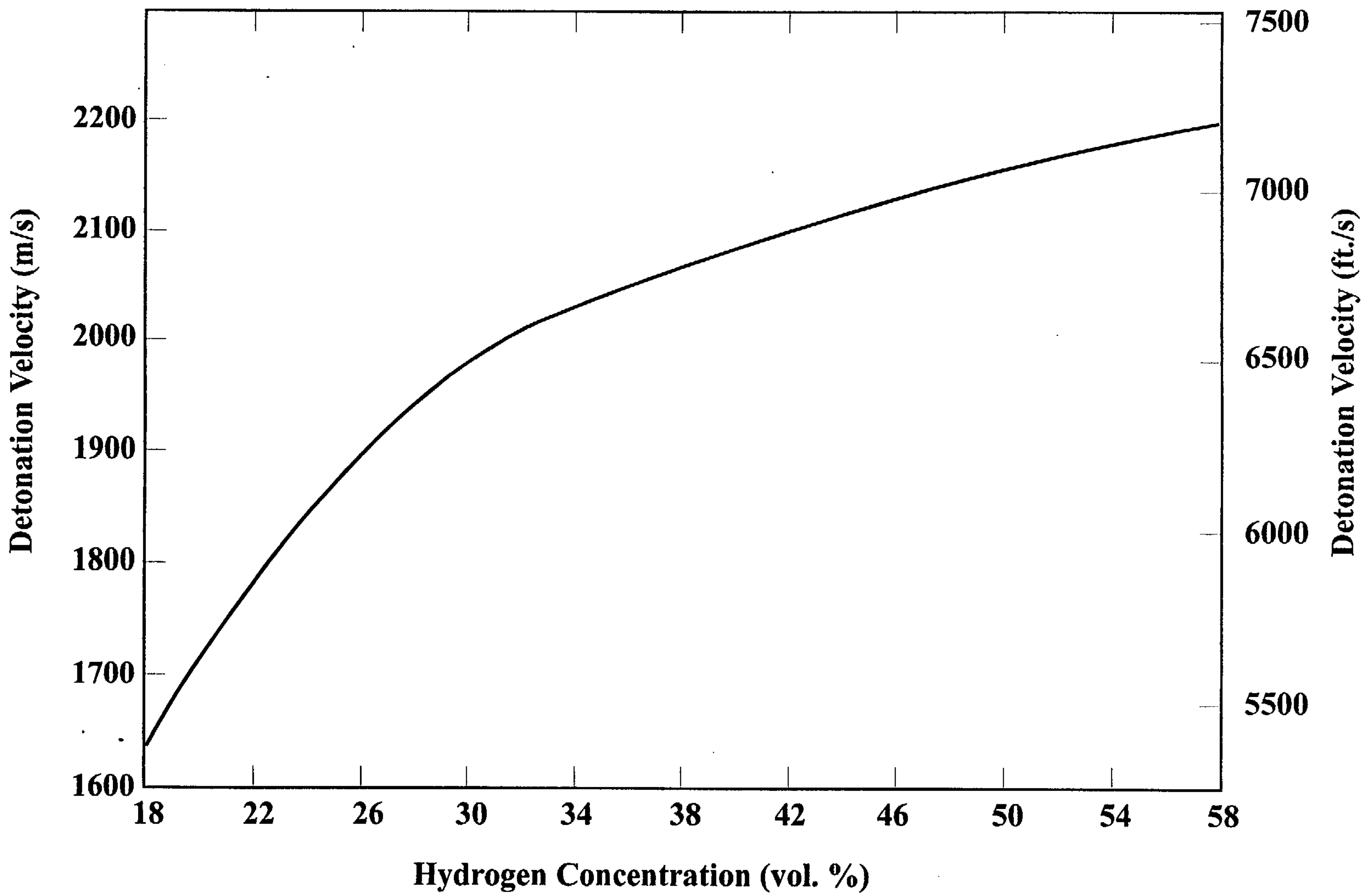


Figure 4.6-9 Theoretical detonation velocities for hydrogen : air mixture

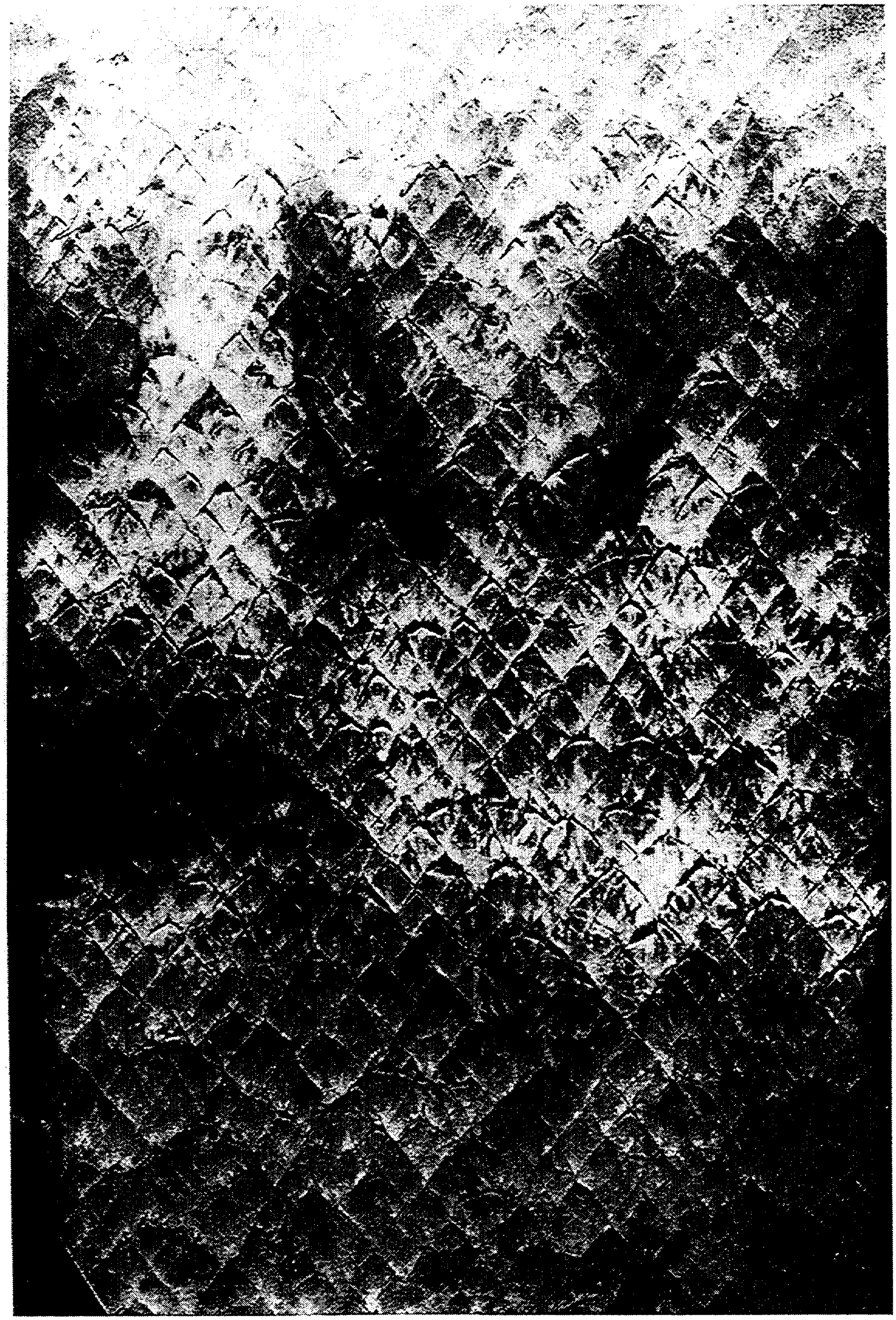


Figure 4.6-10 Hydrogen detonation cells

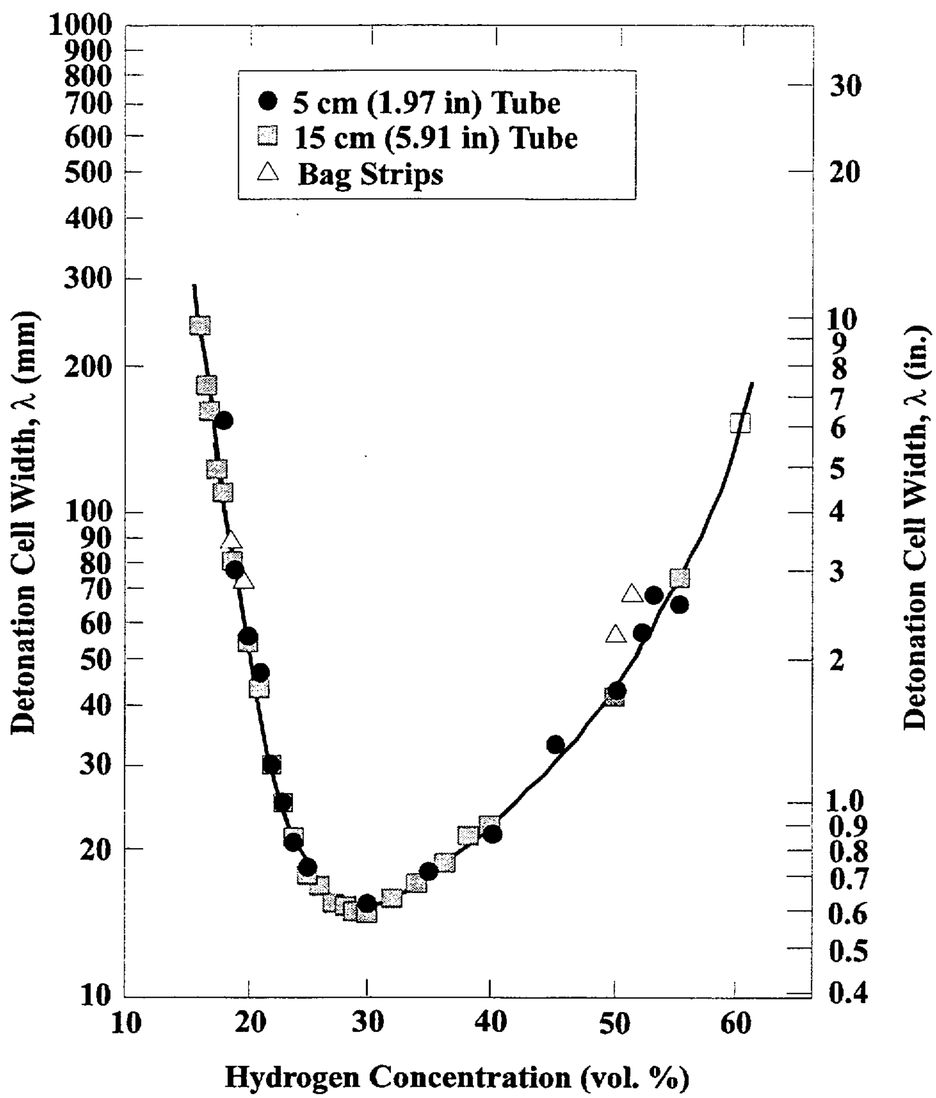


Figure 4.6-11 Measurement of detonation cell size for hydrogen : air mixtures at atmospheric pressure

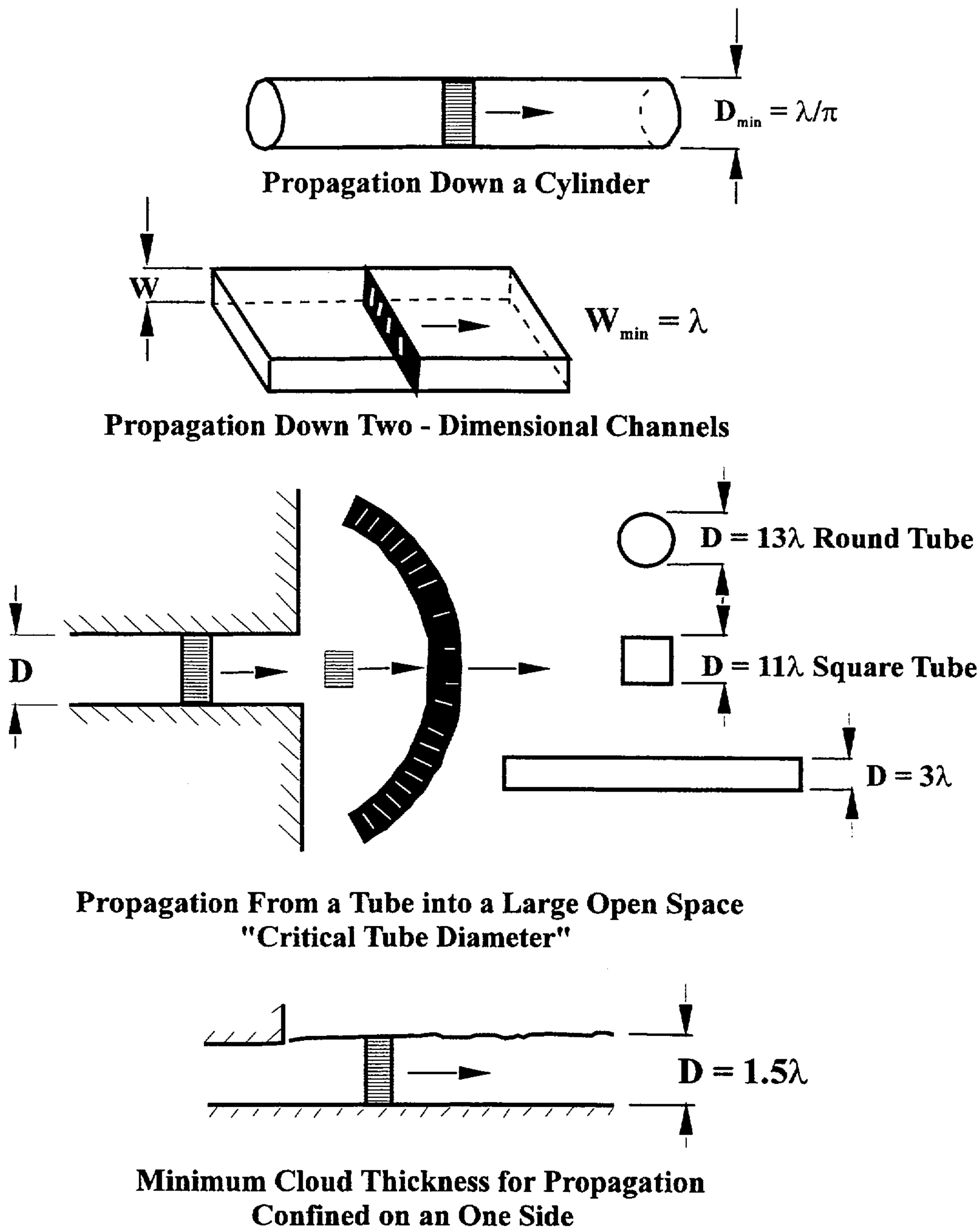


Figure 4.6-12 Dimensions required for detonation propagation in various geometries

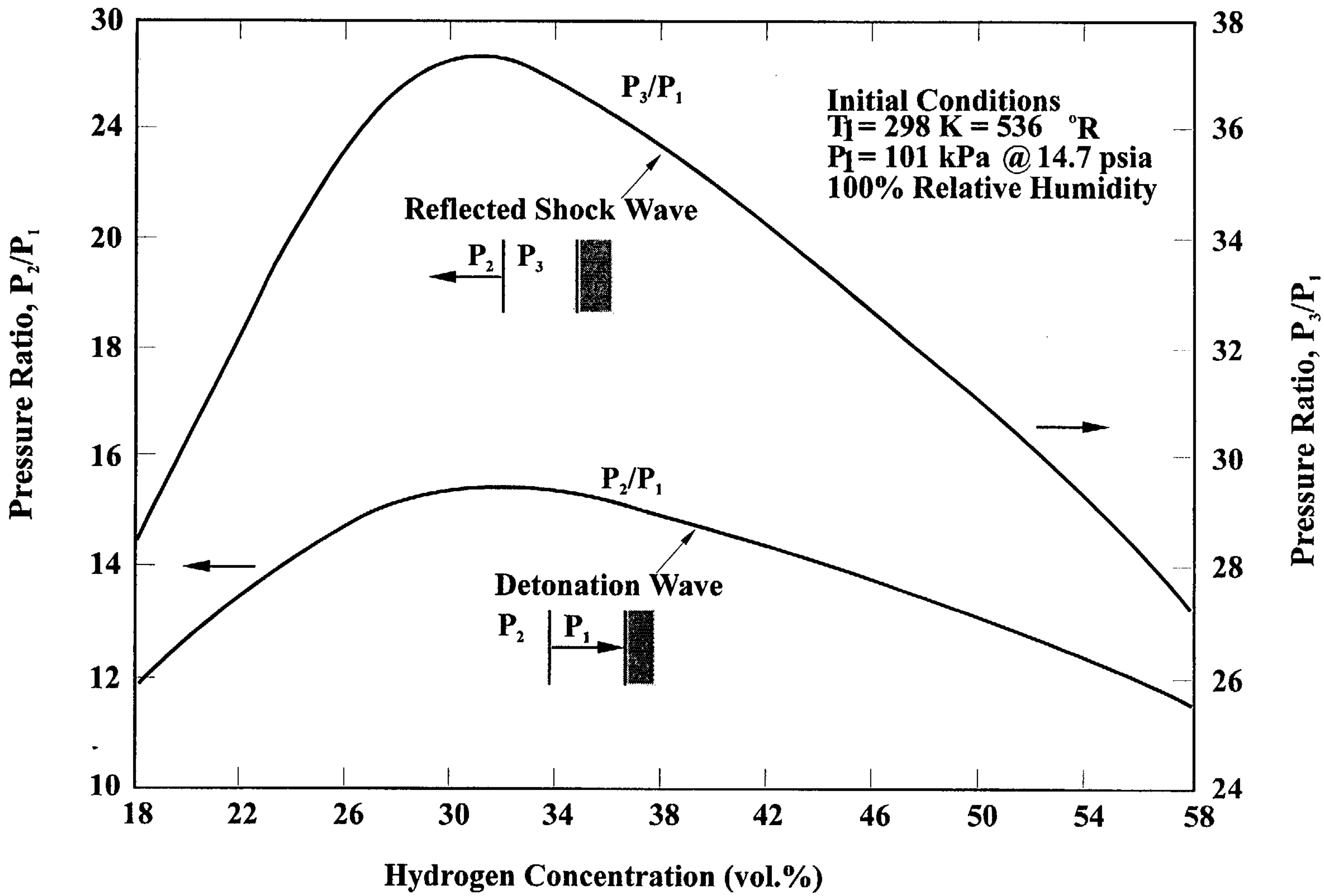


Figure 4.6-13 Theoretical detonation pressure and normally reflected pressure

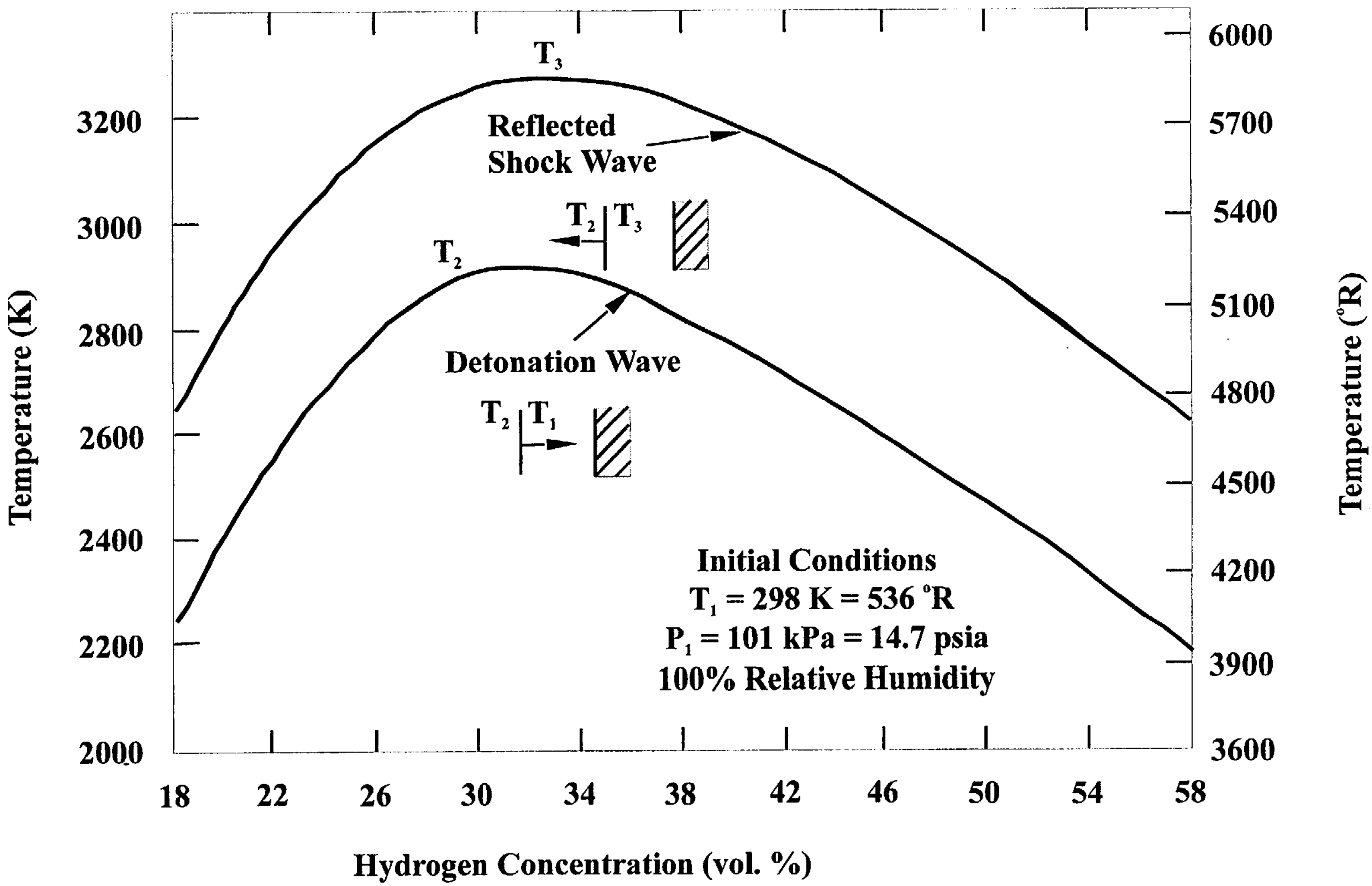


Figure 4.6-14 Theoretical detonation temperature and normally reflected detonation temperature

Flame Scaling

Structure Determined by Source Froude Number, F

$$F = \rho U^2 / g \Delta \rho D$$

- F >> 1 Jet-Like Flames
- F << 1 Plume-Like Flames

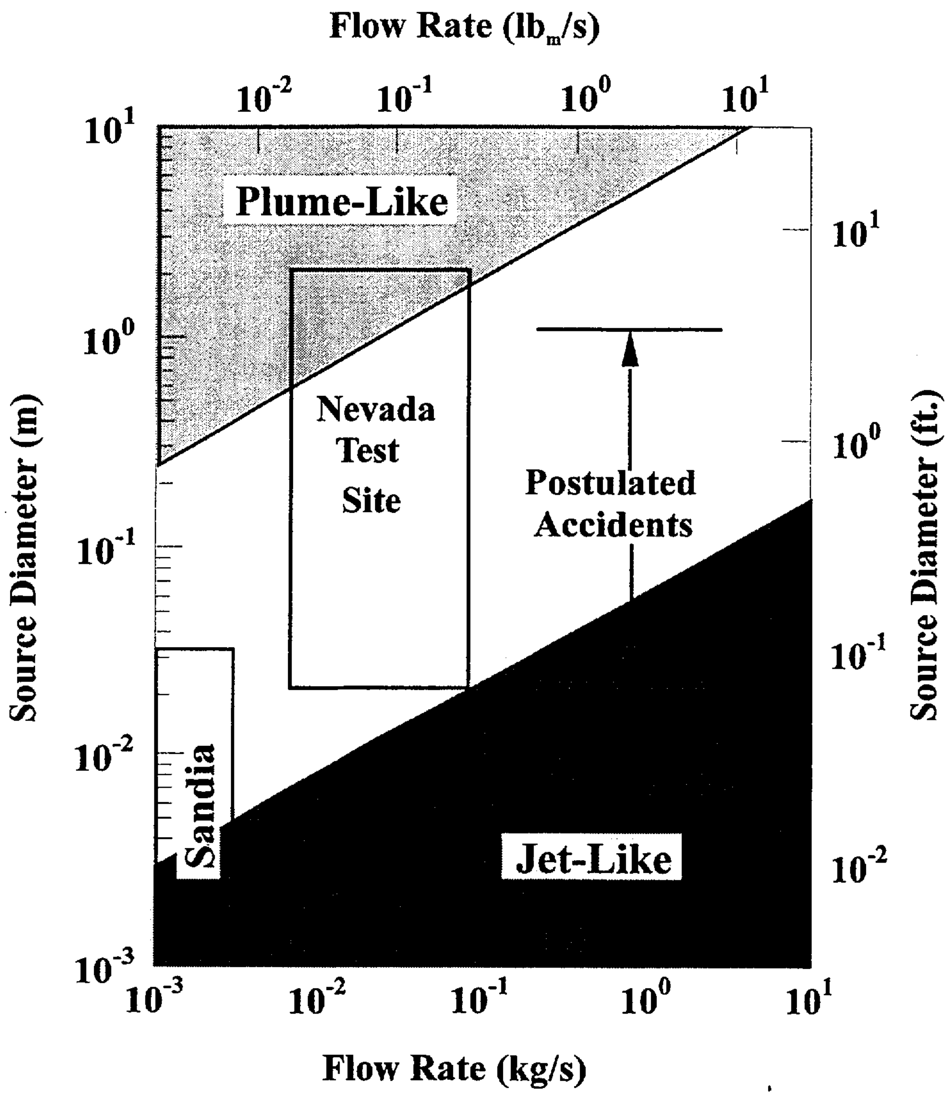


Figure 4.6-15 Flame structures for a range of geometries and flow rates

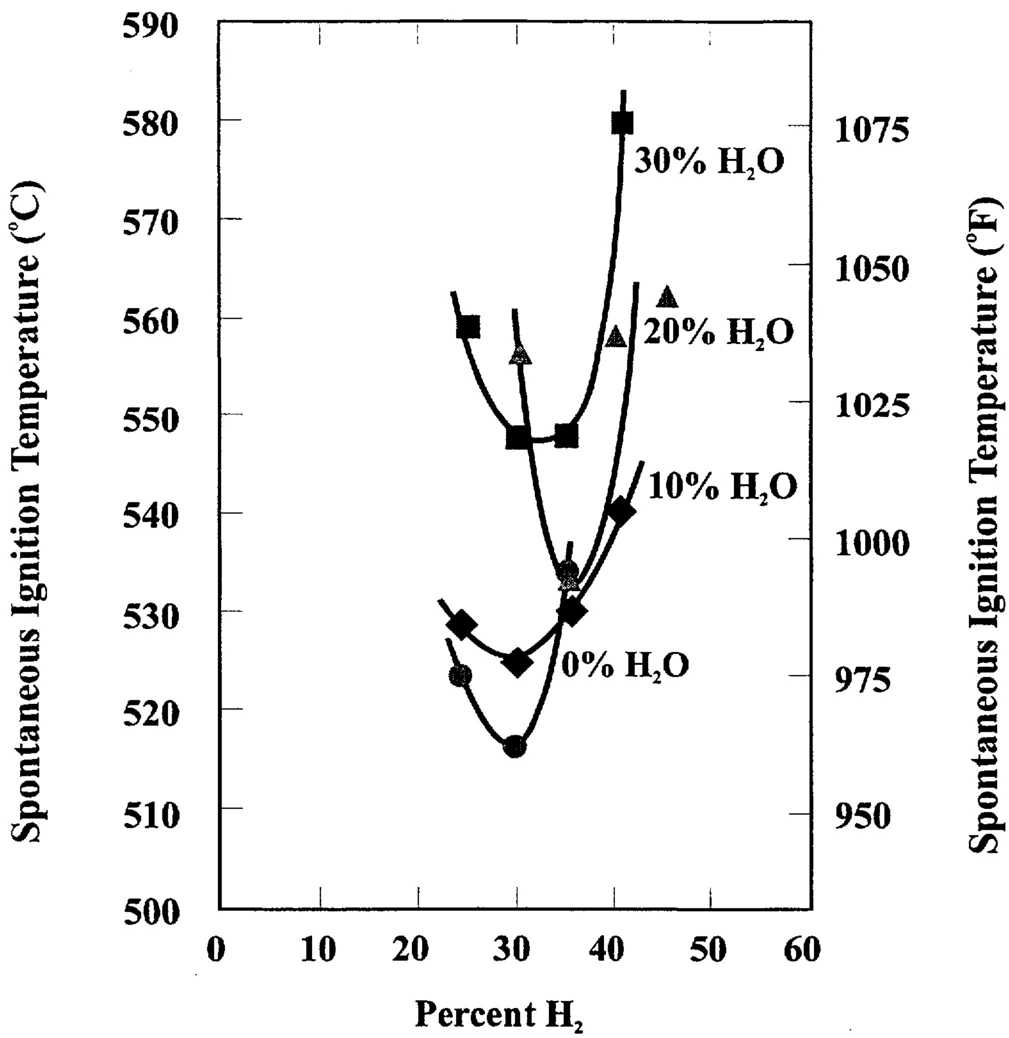


Figure 4.6-16 Minimum spontaneous ignition temperatures

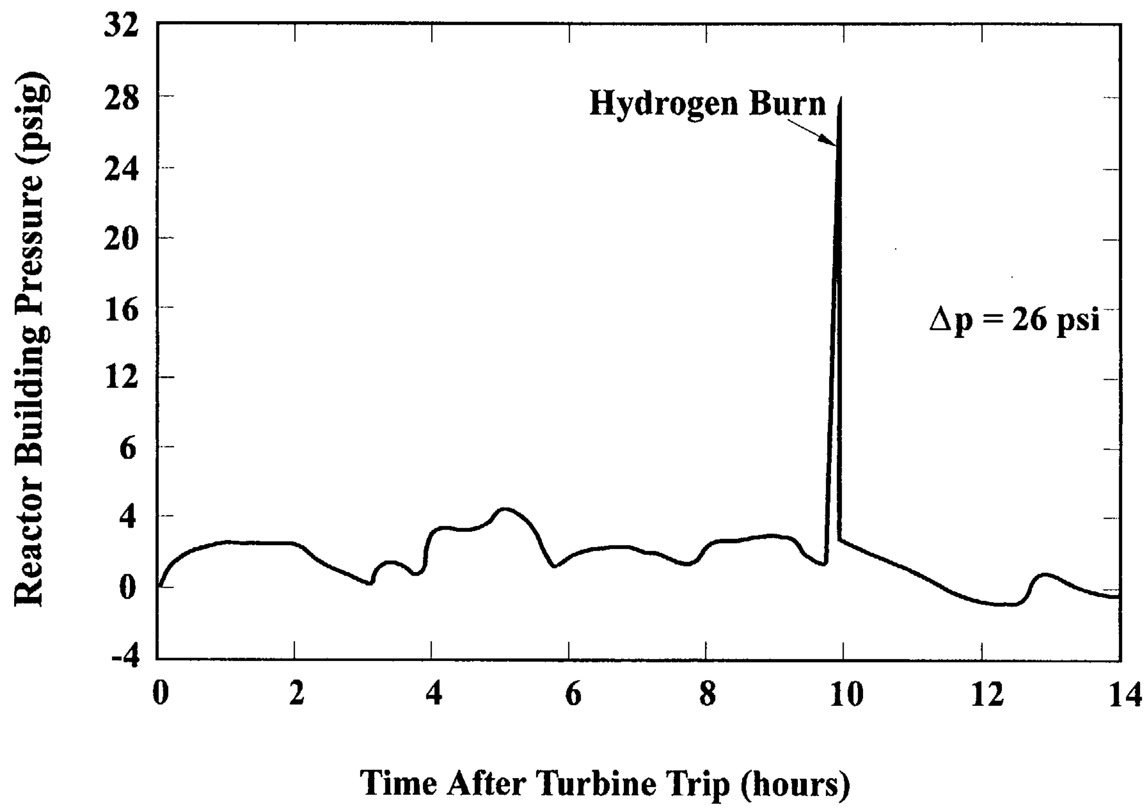


Figure 4.6-17 TMI-2 containment pressure versus time

References for Section 4.6

1. A. L. Camp, et al., "Light Water Reactor Hydrogen Manual," NUREG/CR-2726, SAND82-1137, Sandia National Laboratories, August 1983.

2. E. P. Coward and G. W. Jones, "Limits of Flammability of Gases and Vapors," Bulletin 503, Bureau of Mines, U. S. Department of Interior, 1952.

3. B. Bregeon, et al., "Near-Limit Downward Propagation of Hydrogen and Methane Flames in Oxygen-Nitrogen Mixtures," *Combustion & Flame*, 33:33-45, 1978.

4. Z. M. Shapiro and T. R. Moffette, "Hydrogen Flammability Data and Application to PWR Loss-of-Coolant Accident," WAPD-SC-545, Bettis Plant, September 1957.

5. I. L. Drell and F. E. Belles, "Survey of Hydrogen Combustion Properties," *NACA R 1383*, National Advisory Committee for Aeronautics, 1958.

6. A. L. Furno, E. B. Cook, J. M. Kuchta and D. S. Burgess, "Some Observations on Near-Limit Flames," *13 Symposium on Combustion*, Pittsburgh Combustion Institute, 593-599, 1971.

7. B. C. Slifer and T. G. Peterson, "Hydrogen Flammability and Burning Characteristic in BWR Containments," NEDO-10812, 73NED49, General Electric, April 1973.

8. M. P. Paulson and J. O. Bradfute, "Pressure and Temperature Transients Resulting from Postulated Hydrogen Fires in Mark III Containments," EI 75-4, Energy Inc., 1981.

9. M. Berman, "Light Water Reactor Safety Research Program Quarterly Report," January-March, 1981, NUREG/CR-2163,1 of 4, SAND81-1216,1 of 4, Sandia National Laboratories, Albuquerque, New Mexico, July 1981.

10. W. E. Lowry, et al., "Final Results of the Hydrogen Igniter Experimental Program," NUREG/CR-2486, UCRL-53036, Lawrence Livermore National Laboratory, February 1982.

11. M. Berman, "Light Water Reactor Safety Research Program Semiannual Report," April-September 1981, NUREG/CR-2481, SAND82-0006, Sandia National Laboratories, Albuquerque, New Mexico, February 1982.

12. Yu. M. Annushkin and E. D. Sverdlov, "Stability of Submerged Diffusion Flames in Subsonic and Underexpanded Supersonic Gas-Fuel Streams," Translation, Combustion, Explosions, and Shock Waves, *14*, 5, 1978.

13. C. M. Guirao, R. Knystautas, J. H. Lee, W. Benedick, and M. Berman, "Hydrogen-Air Detonations," presented at the 1982 Combustion (International) Symposium, Haifa, Israel, 1982.

14. D. W. Stamps and M. Berman, "A Critical Review of High-Temperature Hydrogen Combustion in Reactor Safety Applications," SAND88-0680C, International Conference on Thermal Reactor Safety, Avignon, France, October 1988.

15. R. K. Byers, "CSQ Calculations of H_2 Detonations in the Zion and Sequoyah Nuclear Plants," NUREG/CR-2385, SAND81-2216, Sandia National Laboratories, Albuquerque, New Mexico, July 1982.

16. M. A. Delichatsios, M. B. Genadry, and M. N. Fardis, "Calculations of Hydrogen Detonations in Nuclear Containments by the Random Choice Method," M.I.T., R81-30, December 1981.

17. Z. M. Shapiro and T. R. Moffette, "Hydrogen Flammability Data and Application to PWR Loss-of-Coolant Accident," September 1957.

18. J. E. Shepherd, "Hydrogen-Steam Jet-Flame Facility and Experiments," NUREG/CR-3638, SAND84-0060, Sandia National Laboratories, February 1985.

19. M. Berman, *Light Water Reactor Safety Research Program Semiannual Report, April-September 1981*, February 1982.

20. "Technical Aspects of Hydrogen Control and Combustion in Severe Light-Water Reactor Accidents," National Academy Press, Washington, D.C., 1987.

21. Title 10, *Code of Federal Regulations*, Part 50.44, U.S. Government Printing Office, January 1, 1991.

22. M. P. Sherman and M. Berman, "The Possibility of Local Detonations During Degraded-Core Accidents in the Bellefonte Nuclear Power Plant," NUREG/CR-4803, SAND86-1180, Sandia National Laboratories, January 1987.

4.7 BWR Mark I Liner Failure By Melt Attack

Because of its relatively small enclosed volume and drywell floor area, the BWR Mark I containment structural boundary is particularly vulnerable to failure by overpressure or by direct contact attack should molten core and structural debris leave the reactor vessel. Numerous analyses of the potential for early failure of the containment pressure boundary due to direct interaction with corium have been published, beginning with Reference 1. According to NUREG-1150 (Section 12.4, Perspectives), "At Peach Bottom, drywell meltthrough is the most important mode of containment failure."

In 1988, the NRC Office of Research began a dedicated major effort toward resolution of the Mark I liner failure issue. The approach involved an initial decomposition of the overall issue into considerations of melt release from the reactor vessel, melt spreading over the containment floor, and thermal loading of the drywell shell. The results of this effort, which is purported to be a "mechanistic treatment of the sequence of physical phenomena that lead to liner contact by corium debris, and their coupling through a probabilistic framework that allows representation of uncertainties,"[2] are documented in References 2 and 3.

The following subsections address the prospects for spreading of debris liquids to the steel drywell shell in a manner that would induce failure of that boundary, and the mitigative measures that might be taken. Following a description of the specific features of the Mark I containment that are relevant to this issue, the decomposition categories of melt release, melt spreading, and thermal attack are each discussed. Finally, an assessment of the mitigative effects of water and a summary of the current status of this issue are provided in Subsections 4.7.5 and 4.7.6, respectively.

4.7.1 Pertinent Features of the Mark I Containment Design

The typical BWR Mark I primary and secondary containment configuration is shown in Figure 4.7-1. With the sole exception of the two Brunswick units, which employ steel-lined reinforced concrete structures, all Mark I primary containments incorporate free standing steel structures. Thus, while "liner failure" is correct for identification of the direct corium attack issue with respect to the Brunswick units, it is a misnomer for the other 22 Mark I units, where "shell failure" would be more appropriate. Nevertheless, by historical repetition, "liner failure" has become the general appellation for this issue at all BWR facilities.

It should be recognized that plant-specific containment design differences abound, many with important ramifications with respect to plant response under severe accident conditions. This is true even for supposedly "sister" plants such as Browns Ferry and Peach Bottom.

With respect to the potential for emergent corium to come into contact with the carbon steel drywell shell at the level of the floor, it is important to first note that here also, any detailed analysis must take plant-specific differences into consideration. This discussion is based on the Peach Bottom/Browns Ferry configuration. Figure 4.7-2 provides a plan view of the intersection of the base of the reactor pedestal with the concrete floor. As indicated, there is a single doorway to direct any flow from the inpedestal region toward the opposite portion of the drywell wall. The distance from the point directly underneath the reactor vessel centerline to the closest point of the drywall wall is only about 7 m (22.85 ft.).

Figure 4.7-3 shows the placement of all reactor pedestal penetrations at Browns Ferry, and confirms that the doorway provides the only opening extending to the level of the floor. The location of the drywell sumps within the pedestal region is also shown in this figure.

The pedestal doorway and door at Peach Bottom are shown at the left side of Figure 4.7-4. This photograph also shows a portion of the carbon steel sump cover, which occupies a central rectangular section of the pedestal region floor. The existence of the sumps is a factor with respect to the potential for debris spreading since erosion of the thin sump cover would permit a significant fraction of the emergent debris to be retained in the sumps. The sump volume is about 5.7 m^3 (200 ft.^{3}) at both Browns Ferry and Peach Bottom.

Another pertinent plant-specific difference involves the entrances to the vent pipes, which lead to the pressure suppression pool as illustrated in Figure 4.7-1. As shown in Figure 4.7-5, the lower lip of the vent pipe shielded opening is located on the sloping drywell wall a short distance (about 0.61 m [2 ft.]) above the floor. At Peach Bottom, one of these vent pipes lies directly opposite the pedestal doorway. This is not the case at Browns Ferry, where this doorway faces a portion of the drywell wall midway between two of the vent pipe entrances.

4.7.2 Characteristics of Debris Pours From Vessel

Before undertaking to address debris flow through the pedestal doorway and spreading in the expedestal region, it is necessary to first select appropriate representative debris releases (rates, quantities, compositions, and temperatures) from the reactor vessel. Since the ultimate purpose of this selection process is for use in exploring means for mitigation, it is reasonable to exclude accident scenarios involving high-pressure melt ejection, for which failure of the shell integrity is virtually certain by means of DCH, liner melt, or overpressurization. (It is of course a goal of BWR accident management that high pressure severe accident sequences not occur. These have not been analyzed; see discussion in Section 3.7.2.)

The Mark I liner failure study[2,3] recognizes the existence of major uncertainties in the calculation of in-vessel core melt progression and therefore considers two debris pours, each associated with the predictions of a different severe accident code. It is important to recognize that this study is focused upon low-pressure accident sequences and the potential for "early" containment failure, that is, failure in conjunction with the initial release from the reactor vessel, before the aerosols have had time to settle. The two selected pours, which are described below, are intended to bound the spectrum of debris compositions that might reasonably be expected to be encountered within the realm of interest.

4.7.2.1 Scenario I: Large Initial Pour of Molten Oxides

The pour that Reference 2 analyzes as Scenario I derives from calculations performed by the industry-sponsored MAAP code[6]. In-vessel melt progression resembles the TMI event and is characterized by holdup of debris in the core region in the form of a large pool of molten oxides mixed with superheated metallic zirconium within a crucible-shaped metallic supporting crust. Eventually, a large portion of the molten pool breaks through the metallic crust, and enters the lower plenum while pushing the water away. There is little interaction between the flowing oxides and the water, or with the lower plenum stainless steel structures, except to cause immediate penetration failures. The liquid zirconium-oxide mixture then drains into the containment over a period of about five minutes.

The Mark I liner failure study[2] employs probability density functions for the pour characteristics, but it is appropriate to think of

Scenario I as an initial rapid release of 12.5 m^3 (441 $ft.^3$) of 20% zirconium metal in a UO_2-ZrO_2 ceramic mixture with negligible (50 K [90°F]) superheat. All water then drains from the vessel, and the subsequent debris release is relatively very slow, corresponding to the rate at which the debris remaining within the vessel melts. Specifically, the study considers an additional 7.5 m^3 (265 $ft.^3$) of the same debris composition to be released into the containment over the next 150 minutes.

4.7.2.2 Scenario II: Metallic Pour Followed by Release of Oxides

The Scenario II pour considered in Reference 2 derives from predictions of the BWRSAR code[7], which was sponsored at the time by the NRC. (Subsequently, many of the BWRSAR models have been made operational within MELCOR.) Here the core melt progression is that described in Sections 3.7.5 and 3.7.6 for the dry case. Basically, debris relocating into the lower plenum is quenched by the water there. The remaining water is boiled away under the impetus of decay heating and the lower plenum steel structures are subsumed into the surrounding debris. Penetration failures occur at a time when only liquid metals are present as the debris temperature increases after lower plenum dryout. Oxide melting (and release) follows the initial release of metals.

As noted in Section 3.7, detailed analyses[15] have indicated that bottom head release pathways initiated by debris flows internal to the penetration tubes are much less likely than previously thought. Instead, the experimental evidence (See Section 3.5.7) is that vessels equipped with bottom head penetrations would fail by flows external to the tubes and within the expanding vessel through-holes provided for and surrounding the penetrations. From the standpoint of debris pour characterization, however, there is little difference associated with the details of penetration failure. Thus, the Scenario II pour remains an appropriate representation of the debris releases associated with penetration failure.

For the Mark I liner failure study,[2] the Scenario II probability density functions are such that the initial pour can be approximated as 14 m^3 (494 $ft.^3$) of a mixture of stainless steel with 30% zirconium metal at 100 K (180°F) superheat. This initial pour, which occurs over a period of 20 minutes, is followed by a relatively slow release of 15 m^3 (530 $ft.^3$) of oxides (mixed with 15% zirconium metal) over a period of 100 minutes.

4.7.2.3 Accident Scenarios Not Represented

As mentioned previously, the Mark I liner failure study does not consider high-pressure accident sequences. Neither does it directly address accident sequences in which penetration failures are assumed not to occur. Debris pours are based upon decay heat rates prior to any increases due to power uprates.

Completion of the Mark I liner failure study involved a process by which a preliminary analysis was performed and subjected to extensive peer review. This first step is documented in Reference 2. Additional analyses were then performed to address the concerns identified by the peer review. This second round of analyses and the final conclusions of the study are documented in Reference 3.

One of the concerns addressed in the second round of analyses has to do with the amount of metals that might be mixed with the oxides in the Scenario I release. This is an important question because the magnitude of the material superheat that is carried with the spreading debris is paramount to the fate of the wall after contact. (Debris liquids have a higher effective conductivity for heat transfer into the wall surface.) Specifically, it was proposed that upward radiation from the molten pool held in the core region might induce melting of the upper reactor vessel internal structures so that the molten liquids, when released into the lower plenum, would include a large quantity of stainless steel. Dedicated calculations with the APRIL code,[8] however, subsequently showed that the composition of the Scenario I melt as originally conceived was indeed bounding for that scenario.

Although less probable, the accident sequence without penetration failure would produce a much larger included metals superheat than is represented by the Scenario I release. With the reactor vessel depressurized, creep rupture of the vessel wall would not occur until the wall was heated to near the carbon steel melting temperature. At this time, much of the central region of the lower plenum debris bed would be occupied by a slurry of solid oxides and superheated metals. But, if penetration failures did not occur and the bottom head failed by creep rupture, how would the contents of the lower plenum be released to the drywell?

It is, of course, unknown how the separation of the portion of the bottom head below the support skirt from the remainder of the vessel would progress. Would there be a complete break, or would one side of the bottom head sag? The available experimental evidence as obtained for the PWR bottom head configuration is discussed in Section 3.5.7, but BWR structure has important differences.

BWR vessels have a support skirt and, as shown in the lower part of Figure 3.7-7, there is a control rod drive housing support structure about 1 m (3 ft.) beneath the vessel that might interrupt the downward movement of the dislocated portion of the vessel bottom head. It is pointed out in Reference 2 that much of the lower plenum content could still be contained within the bottom head after its initial rupture, and that "Pending quantification of the merits of this new type of scenario, quantification of it is left for future study."

4.7.3 Debris Spreading Across The Drywell Floor

As described in the previous section, distinction between the two considered scenarios is that Scenario I involves a mostly oxidic melt with a high initial release rate whereas Scenario II involves an initially metallic melt with a relatively low release rate. The total release quantity for Scenario II is about twice that for Scenario I.

As considered in Reference 2, debris falling from the reactor vessel would first fill the drywell sumps, then would spill over the pedestal region floor. As sufficient height is accumulated over the floor, flow would begin through the pedestal doorway. Initial spreading as the flow enters the expedestal region would be slight, but after contact with the drywell wall, the flow would separate into two branches, each flowing along the wall in a nearly one-dimensional fashion. These two branches then meet at a position diametrically opposite to the doorway.

During the spreading process, the flowing debris radiates to the overlying atmosphere (or water) and transfers heat to and ablates the underlying concrete. Gases released from the concrete promote oxidation of the metals carried with the debris, and the associated energy release serves to increase the debris temperature. The purpose of the spreading calculations described in Reference 2 is to determine the parameters important to the question of survivability of the drywell wall. These are the depth of the debris immediately adjacent to the wall, the initial superheat of this debris, and the length of time that superheat is maintained.

For the second round of analysis, the peer review process recommended that the adequacy of the approach employed in the preparation of Reference 2 be checked by the application of other available analytical tools, specifically the

MELTSPREAD[9] and CORCON[10] codes. The MELTSPREAD code provides a mechanistic treatment of the basic processes involved in the spreading of debris over a steel or concrete substrate, including gravity-driven flow, melt freezing, immobilization and heatup, concrete decomposition and gas release, chemical oxidation of melt metallic constituents, enhancement of heat transfer by any overlying water, and spreading of new melt release over previously spread material. The CORCON code, which provides detailed modeling of core-concrete interaction phenomena, was invoked specifically to confirm the duration of superheat in the debris.

As described in Reference 3, the results of the independent MELTSPREAD and CORCON analyses support the contention that for the pours considered in Reference 2, the depth of debris at the wall, the initial superheat, and the duration of superheat employed for the shell failure analysis are appropriately conservative. In other words, the values used in considerations of heat transfer to the shell (discussed in the next section) are higher than those that would be produced in a best-estimate analysis.

4.7.4 Thermal Loading of the Shell

For the initial phase of the Mark I liner failure study, as documented in Reference 2, it was assumed that wall failure would occur if the local temperature reached 1773 K (2732 °F), which is tantamount to failure by melting. During the subsequent peer review, it was recognized that the drywell shell as installed at Peach Bottom and most other BWR Mark I facilities is susceptible to failure by creep rupture, which would occur at a much lower temperature .

The Peach Bottom drywell shell is encased in concrete below the level of the drywell floor. Above the floor, there is a 5.1 cm (2 in.) air gap between the outer surface of the shell and the surrounding concrete. Between these two regions, there is a sand-filled transition zone, which is intended to transmit any seismic loads from the primary containment evenly into the supporting concrete foundation. The arrangement of the shell, surrounding concrete, and sand-filled transition zone at the level of the floor is plant specific. At Peach Bottom, the shell is 3.2 cm (1.250 in.) thick and the top of the sand transition zone is at the level of the floor. (At Browns Ferry, the shell thickness is 2.9 cm [1.125 in.], and the sand-filled transition zone extends more than 10 cm [4 in.] above the floor level.)

The surface of the transition zone constrains the portion of the spherical shell just above it from moving radially outward against the surrounding concrete. Thus, the shell is subject to temperature-induced creep rupture at the locations where it would be heated by adjoining debris.

In response to the concern identified by the peer review in regard to the assumed shell failure criteria, the ANATECH Research Corporation was assigned to carry out a three-dimensional, finite element, structural analysis of the Mark I shell in localized contact with debris, the results of which are documented in Part V of Reference 3. The debris, which is of composition (oxidic) and depth (20 cm [8 in.]) corresponding to Scenario I, was assumed to be covered with water. Containment pressure was represented as remaining constant at 0.2 MPa (29 psia). Creep rupture was predicted to fail the shell at 1533 K (2300 °F), which is about 240 K (430 °F) lower than the carbon steel melting temperature. This temperature of 1533 K (2300 °F) was then adopted by the Mark I failure study for use as the best-estimate failure temperature for the drywell shell.

4.7.5 Mitigative Effects of Water

As described in Section 4.7.3, the most important parameters determining the amount of energy that would be transferred to the shell from the adjacent debris are the debris depth, the initial debris superheat, and the time duration that the debris remained superheated. If a significant quantity of water overlies the drywell floor at the time of initial debris release, then all three of these parameters would be affected favorably, from the standpoint of promoting the survival of the shell.

The depth of debris adjacent to the wall would be reduced, because more of the debris would freeze within the pedestal region, and less would reach the shell.

The largest beneficial effect would be in reducing the amount of superheat carried to the wall. Although the effectiveness of overlying water in cooling crusted debris is not well understood, insulating crusts would develop only after the superheat is lost. Since heat loss mechanisms to water from a superheated corium melt are straightforward and the associated heat transfer is large, it is pointed out in Reference 3 that the effectiveness of water in reducing superheat is not a matter of controversy.

With respect to the duration of superheat in the debris at the wall, the overlying water would play an important role by removing heat from the wetted shell just above the debris. In effect, the portion of the shell that is in contact with water above the debris acts as an efficient cooling fin. A cooling pathway is established from the debris into the shell and up through the fin to the overlying water. This accelerates the elimination of the debris superheat.

One cannot selectively consider only the benefits of the presence of water, however. Before recommending that provision be made to supply water to the Mark I drywell floor under severe accident conditions, it is first necessary to consider the potential for and effects of steam explosions. For the Mark I liner failure study, this consideration is given in Appendix A "The Occurrence and Role of Steam Explosions in a Mark-I Containment" to Part I of Reference 3. There it is argued that integrity-threatening steam explosions over the surface or at the leading edge of the spreading debris can be ruled out, and that in fact, small-scale steam explosions (fuel-coolant interactions) at these locations would be beneficial from the standpoint of promoting quenching. On the other hand, it is noted that there is a concern with respect to the potential for energetic steam explosions induced by the debris pouring within the pedestal region.

Appendix A to Reference 3 Part I also notes that the occurrence of an energetic steam explosion within the pedestal region would offer much more of a threat to the reactor pedestal than to the drywell shell. However, the argument is made that with a limited depth of water (about 30 cm [1 ft.]), an impulse to the pedestal of the magnitude necessary to cause failure cannot be delivered. (Note that the actual depth can vary from plant to plant.)

It is not yet possible to make definitive resolution as to whether or not it is worth risking the destructive potential of steam explosions in order to reap the beneficial aspects of water on the drywell floor. Phenomena associated with the introduction of corium into water have been studied in the series of experiments discussed in Section 3.7.6.1. At the very least, it can be concluded that energetic explosions with the oxidic components of core debris are very hard to generate.

4.7.6 Potential for Mark I Containment Failure

The Mark I liner failure study has considered the case of debris release via penetration failures from a depressurized Peach Bottom reactor vessel. The study concludes that there is a "virtual certainty" of shell failure if the containment floor is dry at the time of initial release, but that early shell failure is "physically unreasonable" if the drywell is flooded with water to the lower lip of the vent pipe openings, a depth of about 61 cm (2 ft.).

For its consideration of this issue, the Advisory Committee on Reactor Safeguards (ACRS) has stated that "Results of the severe accident research have shown that there is no threat of prompt containment failure posed by ... Mark I liner meltthrough. Research should continue to: ... determine the impact of ex-vessel steam explosions on the BWR containments."[11]

Apparently this statement is intended to include only cases with water overlying the drywell floor. What about the high-pressure case? The ACRS goes on to state "Additional assessment of DCH is needed for ... BWRs." Thus, it seems that the concern with respect to shell failure that might be caused by high-pressure melt ejection is effectively subsumed into the larger direct containment heating issue.

The reason that the presence of water to the level of the vent openings is considered to reliably preclude only "early" or "prompt" failure of the shell has to do with the formation of solid-base islands of quenched metallic debris in conjunction with the initial release. These could then serve as underwater causeways for transport of subsequent molten releases from the reactor vessel toward the drywell shell. If debris liquids can reach the shell at locations above the water surface, then the situation reverts to the dry case, for which shell failure is probable.

The current situation with respect to the Mark I liner failure analyses based on the Peach Bottom facility is summarized in Table 4.7-1. As indicated, high pressure cases and accident sequences involving creep rupture of the reactor vessel bottom head have not been analyzed. For the cases that have been studied, extension of the existing results to other Mark I facilities is discussed in the remainder of this section.

4.7.6.1 Extension to Other BWR Facilities

As is the case for all other aspects of severe accident research, plant-specific design features play an important role and must be taken into consideration when considering extrapolation of the Mark I study results, based on Peach Bottom, to other BWR facilities. To provide a feel for the extent of the design variations, there are seven different sizes of reactor vessels in U.S. Mark I containments, ranging from Duane Arnold (183 in. ID) to Peach Bottom (251 in. ID). At Duane Arnold, the radius to the shell at the level of the drywell floor is 6.34 m (20.8 ft.), as compared to 6.96 m (22.85 ft.) at Peach Bottom. On the other hand, the potential debris source is smaller at Duane Arnold, where the core comprises just 368 fuel assemblies, as opposed to the 764 assemblies at Peach Bottom.

One of the most important geometric parameters with respect to the shell failure issue is the height of the vent line entrance above the drywell floor. This height determines the maximum depth of water over the floor and, should debris enter a vent pipe, local failure would be virtually certain. As discussed in Part I of Reference 3, the location of the vent line openings is plant specific, but in general the shorter heights are associated with the facilities that have the smaller cores and hence the smaller potential debris pours.

Review by the NRC and its contractors[12,13] of the Individual Plant Examinations (IPEs) submitted by the various BWR facilities has confirmed the importance of the Mark I shell failure issue as previously identified in NUREG-1150. In fact, "Liner meltthrough was found to be the most important contributor to early containment failure for Mark I containments."[12] The IPEs also reveal, however, that in some Mark I containments, the sump and floor configuration is such as to automatically preclude shell failure. Specifically, the sumps at Monticello are large enough to contain all of the debris release from the vessel. At Oyster Creek, a concrete curb serves to prevent the debris from reaching the drywell shell.[12]

4.7.6.2 Drywell Flooding Capabilities

If the drywell floor is to be reliably flooded under severe accident conditions, then the necessary water would have to be capable of delivery into the containment in case of station blackout. The general concept is to vent the containment to permit use of low-pressure pumps and to inject the water via the existing drywell spray headers. To address station blackout concerns, it would be necessary to invoke new or upgraded independently powered pumping systems.

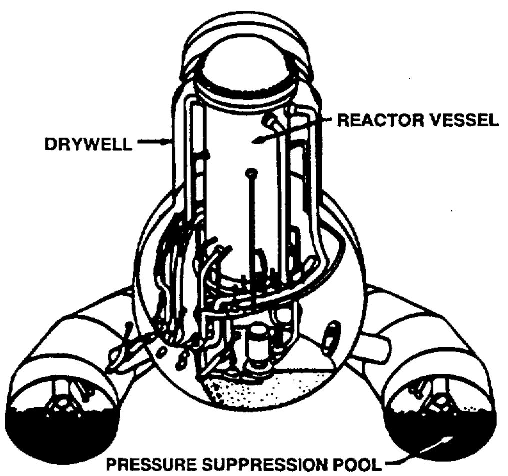
Going one step further, relatively minor modifications beyond the need for an independently powered dedicated pumping system might be employed to permit rapid filling of the wetwell, flooding of the vent pipes, and increase of the water level within the drywell to a height sufficient to cover the reactor vessel bottom head. If drywell flooding to this level could be achieved quickly enough, then the water in the drywell could provide two lines of defense against containment failure: first by serving to keep the debris within the reactor vessel as described in Section 3.7.7.3, and second by extending upward the protection of the drywell shell.

Provision of the necessary volume of water would require the availability of an independent containment flooding system of sufficient capacity to cover the reactor vessel bottom head before lower plenum dryout and the associated threat of penetration failures. This would in general require equipment modifications to existing plants, but similar modifications are required for flooding to protect the shell. In both cases, the drywell would have to be vented during the flooding process and beyond. The only additional requirement for the independently powered pumping systems necessary to deal with station blackout would be to increase their capacity. For two feet of water, 86 m^3 (22,800 gal.) would be required. Making allowance for the trapping of a portion of the containment atmosphere in the upper wetwell as indicated in Figure 4.7-6, about 5700 m^3 (1,500,000 gal.) would have to be added to the Peach Bottom containment in order to submerge the reactor vessel bottom head.

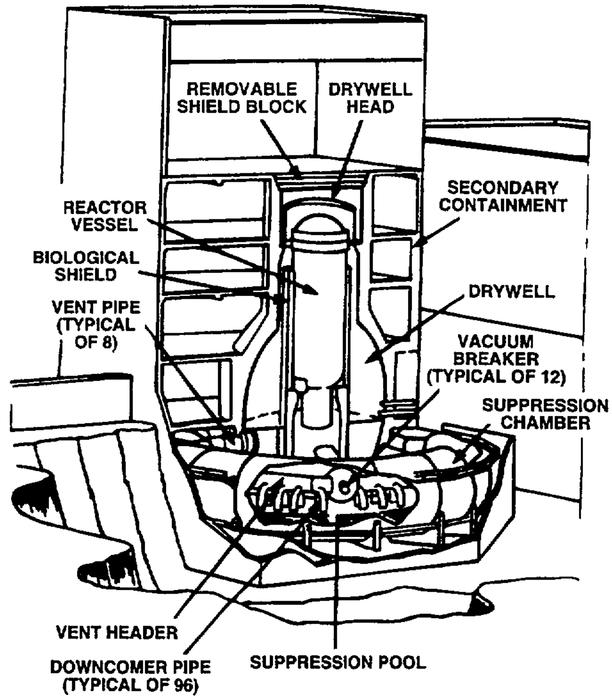
Detailed information concerning the prospects for a containment flooding strategy intended to maintaining the core debris within the reactor vessel is available in Reference 14. The BWR severe accident sequence leading most rapidly to the formation of a reactor vessel lower plenum debris bed is short-term station blackout, for which the vessel bottom head would have to be submerged in no more than 150 min. (2.5 hours) after the onset of core degradation. For Peach Bottom, this is equivalent to a required pumping capacity of 0.63 m^3/s (10,000 gal./min.).

Table 4.7-1 Mark I Liner Qualitative Failure Probabilities for Various Vessel Debris Release Modes

Reactor Vessel Depressurized?	Water Covers Drywell Floor?	Bottom Head Failure Mode?	Case Analyzed?	Estimated Probability of Prompt Liner Failure
No	Either Way	Any	No	High
Yes	No	Penetrations	Yes	High
		Creep Rupture	No	Medium-High
	Yes	Penetrations	Yes	Low
		Creep Rupture	No	Low-Medium



PRIMARY CONTAINMENT
(DRYWELL AND WETWELL)



SECONDARY CONTAINMENT

Figure 4.7-1 **The BWR Mark I containment design employs a small primary containment with a pressure suppression pool; secondary containment is provided by the surrounding structure**

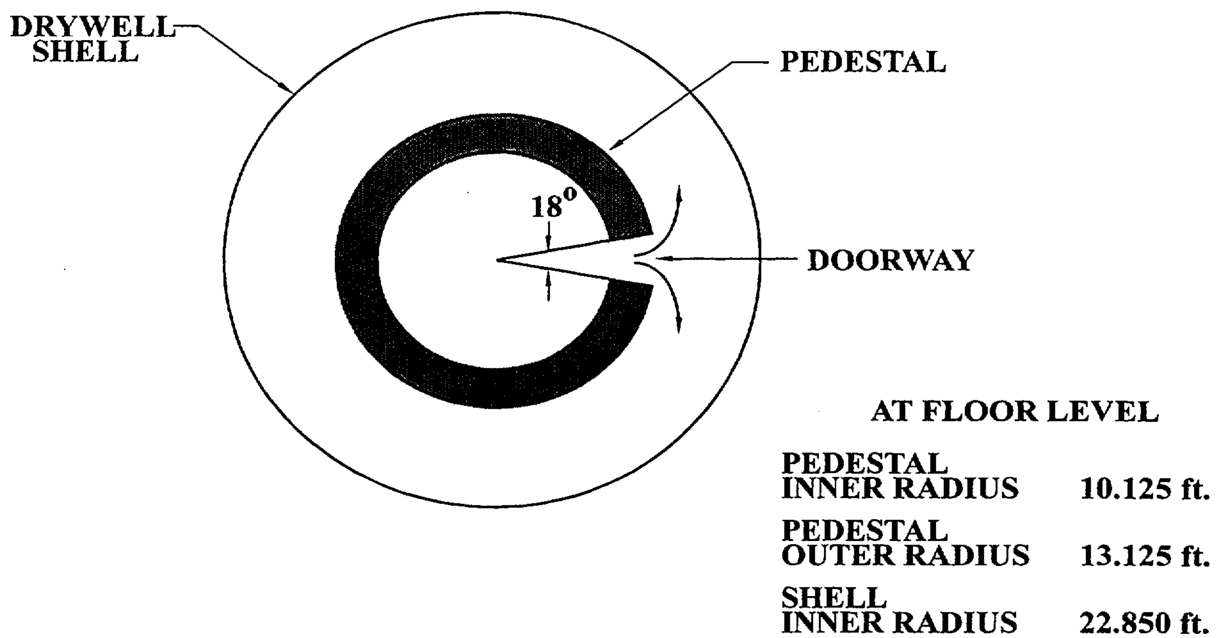


Figure 4.7-2 The Mark I drywell floor area is small and the drywell shell is within ten feet of the pedestal doorway

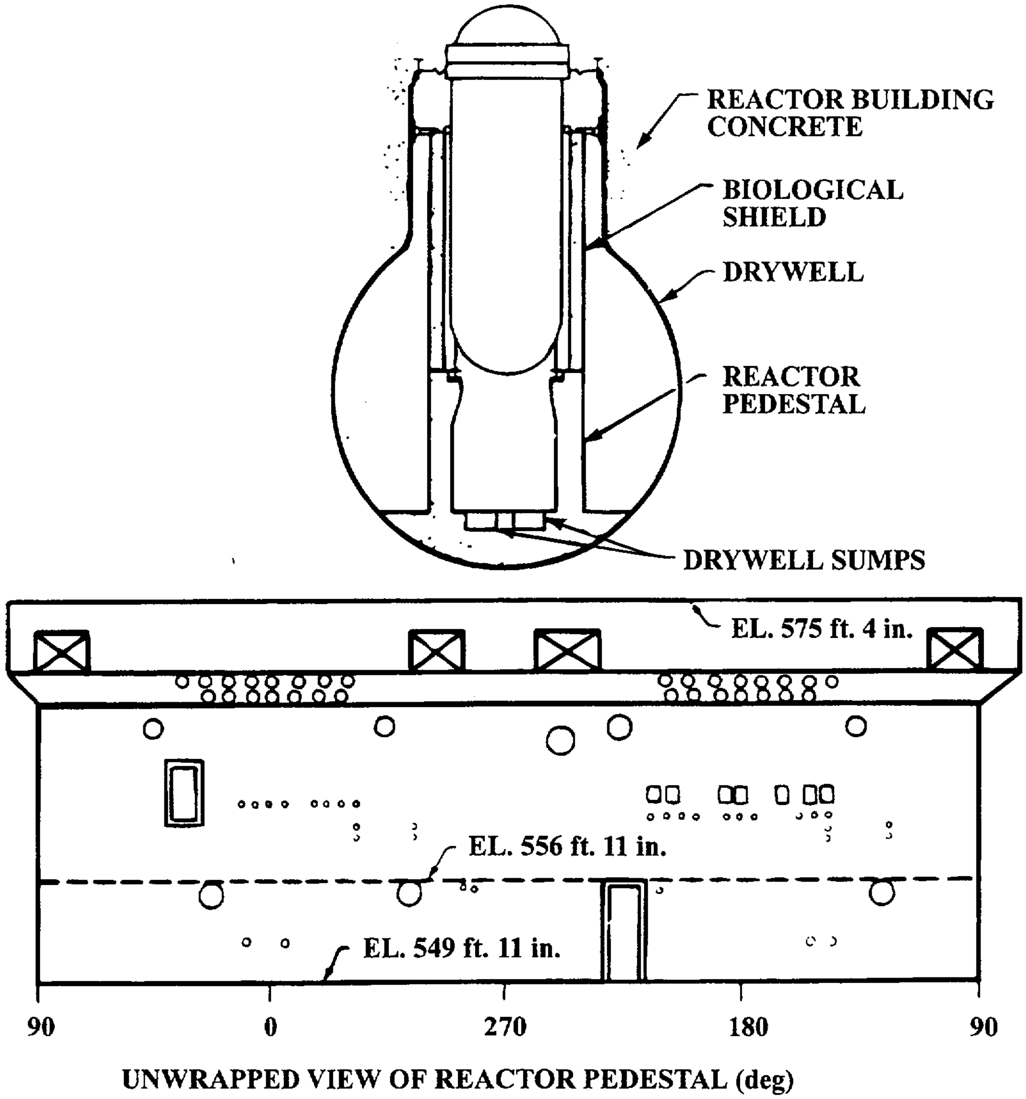


Figure 4.7-3 **Core debris released from the reactor vessel would spread over the BWR Mark I drywell floor, including the ex-pedestal region.**

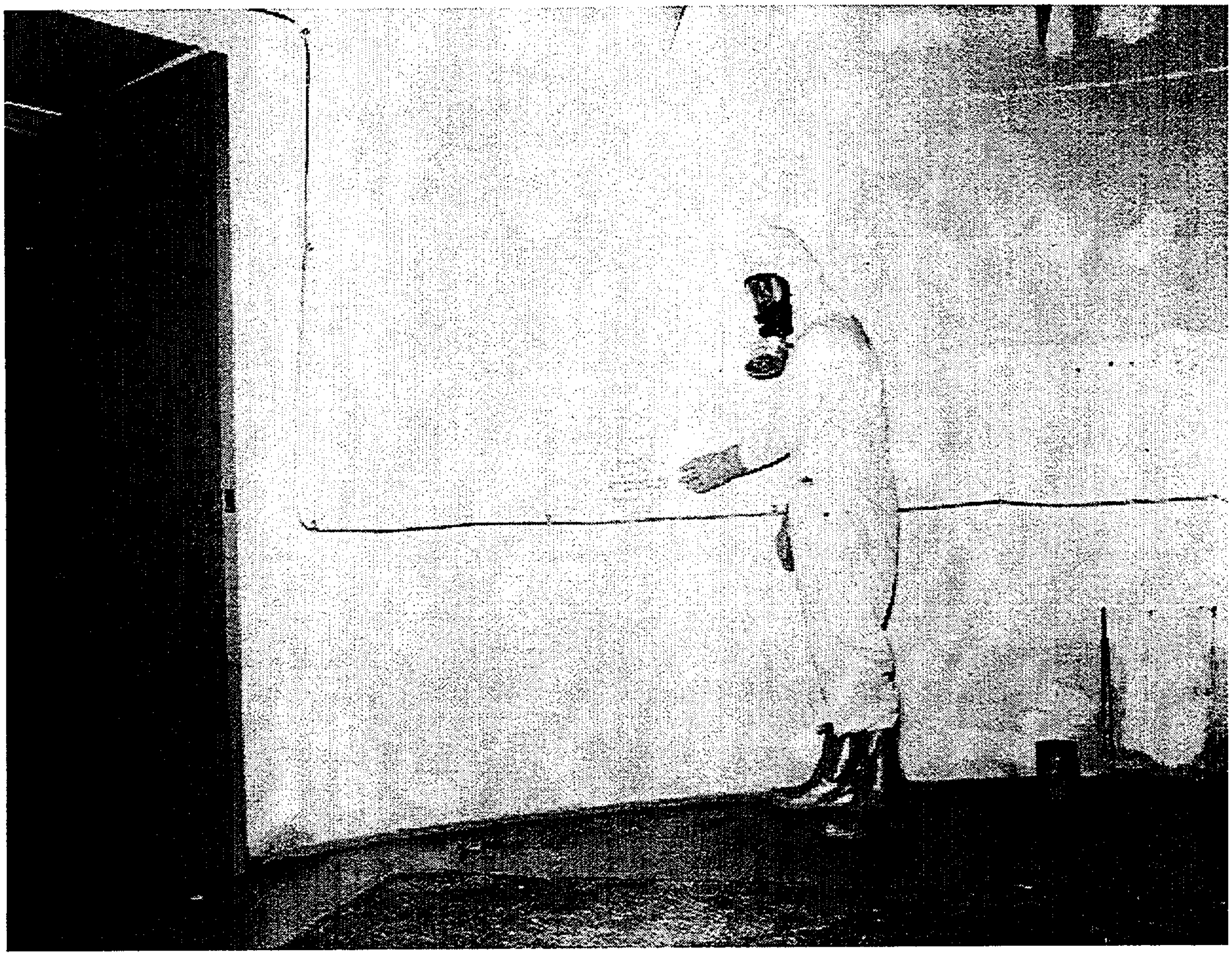


Figure 4.7-4 **Interior of reactor pedestal at Peach Bottom with partial view of doorway.**

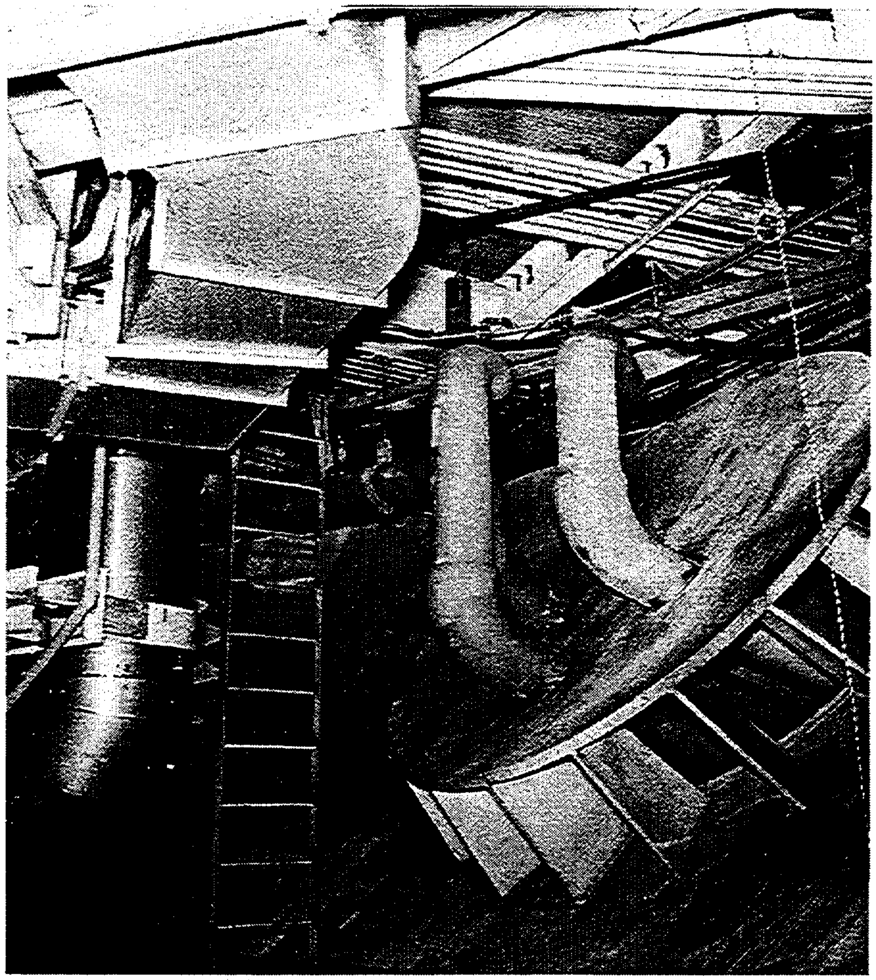


Figure 4.7-5 Shield over vent pipe entrance at Peach Bottom

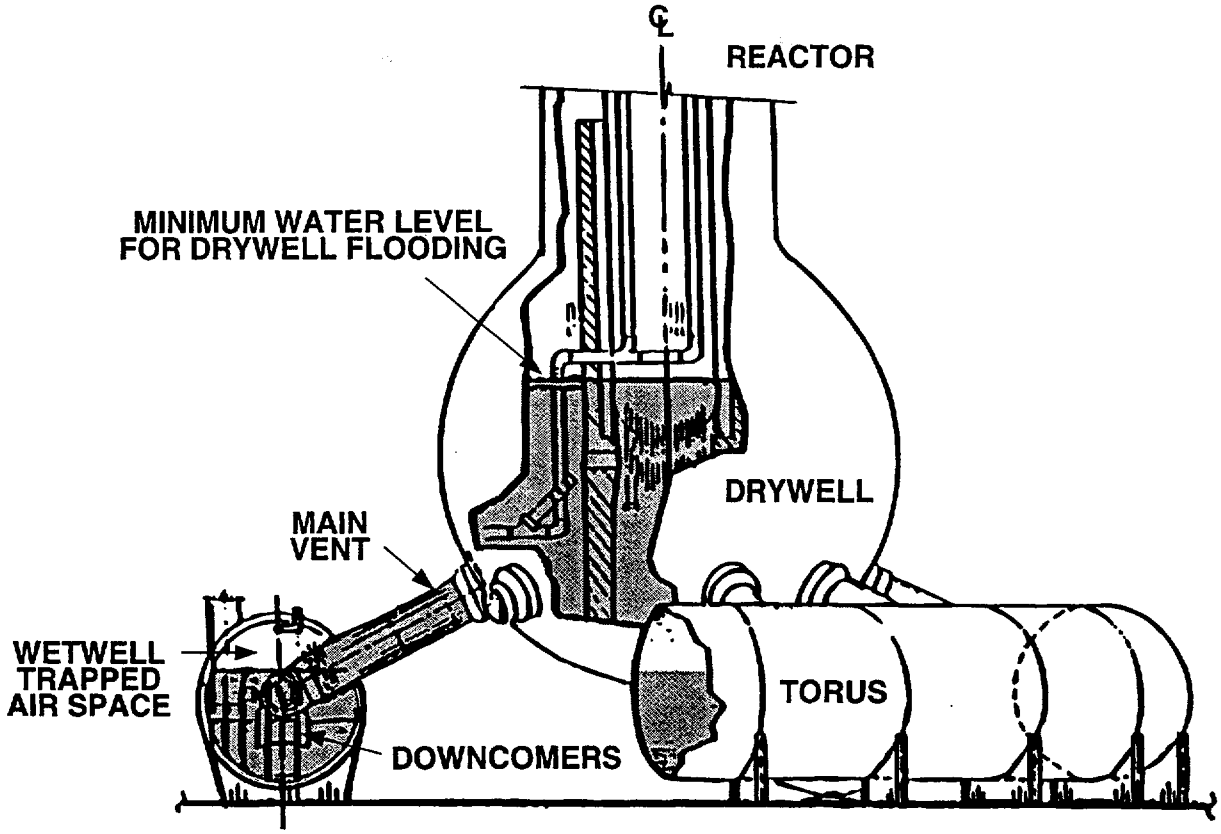


Figure 4.7-6 **Approximately 5700 m^3 (1.5x10^6) gallons would be required to cover the reactor vessel bottom head at the largest (1100MWe) BWR facilities**

References for Section 4.7

1. G. A. Greene, K. R. Perkins, and S. A. Hodge, "Mark I Containment Drywell, Impact of Core/Concrete Interactions on Containment Integrity and Failure of the Drywell Liner," *Source Term Evaluation for Accident Conditions,* paper IAEA, Vienna, Austria, October 1985.

2. T. G. Theofanous et al., "The Probability of Liner Failure in a Mark-I Containment,' NUREG/CR-5423, Center for Risk Studies and Safety, University of California, August 1991.

3. T. G. Theofanous, et al., "The Probability of Mark-I Containment Failure by Melt-Attack of the Liner," NUREG/CR-6025, University of California, Rensselaer Polytechnic Institute, Sandia National Laboratories, Argonne National Laboratory, ANATECH, Inc., November 1993.

4. D. A. Brinson and G. H. Graves, "Evaluation of Seals for Mechanical Penetrations of Containment Buildings," NUREG/CR-5096, SAND88-7016, Sandia National Laboratories, August 1988.

5. General Electric Company, "BWR Owner's Group Emergency Procedure Guidelines," Revision 4, NEDO-31331, March 1987.

6. R. E. Henry, et al., "Experiments Relating to Drywell Shell-Core Debris Interactions," *Proceedings of the Sixteenth Water Reactor Safety Information Meeting,* NUREG/CP-0097, Vol. 5, pp. 317-339, Fauske and Associates, October 1988.

7. S. A. Hodge and L. J. Ott, "BWRSAR Calculations of Reactor Vessel Debris Pours for Peach Bottom Short-Term Station Blackout," *Proceedings of the Sixteenth Water Reactor Safety Information Meeting,* NUREG/CP-0097, Vol. 5, pp. 377-401, Oak Ridge National Laboratory, October 1988.

8. S. H. Kim, et al., "The Development of APRIL.MOD2 - A Computer Code for Core Meltdown Accident Analysis of Boiling Water Nuclear Reactors," NUREG/CR-5157, Rensselaer Polytechnic Institute, 1988.

9. M. T. Farmer, J. J. Sienicki, and B. W. Spencer, "The MELTSPREAD-1 Computer Code for the Analysis of Transient Spreading in Containments," *Sixth Proceedings of Nuclear Thermal Hydraulics*, 1990 ANS Winter Meeting, November, 1990, p. 21, Argonne National Laboratory, 1990.

10. J. F. Muir et al., "CORCON-MOD 1: An Improved Model for Molten-Core/Concrete Interactions," NUREG/CR-2142, SAND80-2415, Sandia National Laboratories, July 1981.

11. Letter from the Advisory Committee on Reactor Safeguards to The Honorable Shirley Ann Jackson, Chairman, U. S. Nuclear Regulatory Commission, "Severe Accident Research," June 28, 1996.

12. C. C. Lin, J. R. Lehner, W. T. Pratt, and M. Drouin, "Severe Accident Progression Perspectives for Mark I Containments Based on the IPE Results," *Proceedings of the Twenty-Third Water Reactor Safety Information Meeting,* NUREG/CP-0149, Vol. 2, pp.

463-472, Brookhaven National Laboratory and U.S. Nuclear Regulatory Commission, October 1995.

13. W. T. Pratt, John Lehner, A. Camp, and E. Chow, "IPE Results as Compared with NUREG-1150," *Proceedings of the Twenty-Third Water Reactor Safety Information Meeting*, NUREG/CP-0149, Vol. 2, pp. 485-504, Brookhaven National Laboratory, Sandia National Laboratories, and U.S. Nuclear Regulatory Commission, October 1995.

14. S. A. Hodge, J. C. Cleveland, T. S. Kress, and M. Petek, "Identification and Assessment of BWR In-Vessel Strategies," NUREG/CR-5869, ORNL/TM-12080, Oak Ridge National Laboratory, October 1992.

15. J. L. Rempe et al., "Light Water Reactor Lower Head Failure Analysis." NUREG/CR-5642, EGG-2618, Idaho National Engineering Laboratory, October 1993.

Appendix 4A Example Calculation of Hydrogen Combustion Pressures and Temperatures

This appendix provides an approximate method for estimating hydrogen burn pressures and temperatures. The example is taken from Reference 1. With the aid of Figures 4A-1 and 4A-2, or 4A-3 and 4A-4, the pressure and temperature that would be caused by an adiabatic, constant-volume, complete combustion of a homogeneous hydrogen:air:steam mixture can be estimated. Figures 4A-1 and 4A-2 can be used for cases in which the steam mole fraction before the burn is small. This might be the case in the wetwell (or outer containment) of a Mark III BWR or the upper compartment of an ice condenser containment. Figures 4A-3 and 4A-4 are to be used when the conditions before the combustion are steam saturated. For initial temperatures not far above normal room temperature, the steam mole fraction is small even in a saturated atmosphere. In that case either set of figures could be used.

We will describe the procedure to be used in the computations in the next paragraph. For all the calculations absolute pressures and temperatures should be used.

Absolute Pressure = Gauge Pressure + Atmosphere Pressure (4A-1)

Typically, for normal atmosphere pressure,

Pressure (psia) = Pressure (psig) + 14.7 (4A-2)

or

Pressure (MPaa) = Pressure (MPag) + 0.101 (4A-3)

For temperature,

Temperature (Rankine) = Temperature (Fahrenheit) + 460 (4A-4)

Temperature (Kelvin) = Temperature (Celsius) + 273 (4A-5)

The subscripts A, S and H_2 refer to air, steam, and hydrogen. The analysis considers three times: t_0, the time at the start of the accident; t_1, the time just before the combustion; and t_2 the time just after the combustion. The object of the calculation is to determine $P(t_2)$ and $T(t_2)$, the pressure and temperature just after combustion. We will assume that conditions at time t_0 are known, and that sufficient information about conditions at time t_1 is known so that the unknown gas conditions at the time can be computed.

Consider the example when the conditions at the start of the accident are:

$P(t_0)$ = 14.7 psia (0.101 MPa)

$T(t_0)$ = 560°R (311 K)

Relative Humidity = 50%

Just before the combustion the temperature is 590°R (328 K), the air is saturated and a hydrogen detector measures 10 volume percent (mole fraction) hydrogen (see Table 4A-1).

For all three time periods, the total pressure is the sum of the partial pressures of air, hydrogen and steam,

$$P = P_A + P_S + P_{H_2} \qquad (4A\text{-}6)$$

Initially, there is no hydrogen, $P_{H_2}(t_0) = 0$. The saturation steam pressures are determined from "Steam Tables" found in thermodynamics textbooks or engineering handbooks. We have

$P_{SAT}(T_0) = P_{SAT}(560\ °R\ (311\ K)) = 0.95$ psia (0.0065 MPa) (4A-7)

$P_S(t_0)$ = relative humidity * $P_{SAT}(T_0) = 0.48$ psia (0.0033 MPa) (4A-8)

Therefore, the initial air partial pressure is

$P_A(t_0) = 14.7 - 0.5 = 14.2$ psia (0.098 MPa) (4A-9)

From steam tables we obtain, at t_1,

$P_S(t_1) = P_{SAT}(T_1) = 2.2$ psia (0.015 MPa) (4A-10)

The air partial pressure at t_1 is

$P_A(t_1) = (T_1/T_0)P_A(t_0) = 590/560) * 14.2 = 15.0$ psia (0.103 MPa) (4A-11)

the hydrogen mole fraction is

$$X_{H_2} = P_{H_2}/P \tag{4A-12}$$

which leads to

$$P_{H_2} = (P_A + P_S) * X_{H_2}/(1.0 - X_{H_2}) \tag{4A-13}$$

Hence

$$P_{H_2}(t_1) = 17.2 * 0.1/0.9 = 1.9\,psia\,(0.013\,MPa \tag{4A-14}$$

$P_1 = 17.2 + 1.9 = 19.1$ psia (0.131 MPa) (4A-15)

We now estimate the postburn conditions using Figures 4A-1 and 4A-2. Figure 4A-1 gives the final/initial pressure ratio for burns with a given set of initial conditions. However, the pressure ratio is insensitive to the initial pressure, and insensitive to small changes in initial temperature. The influence of initial steam mole fraction can be greater. The figures were computed using a humidity corresponding to a steam mole fraction of 3%. At 590°R (328 K) the steam mole fraction for 100% relative humidity will be higher, but will still be small enough to use Figures 4A-1 and 4A-2. From Figure 4A-1, we determine that $P(t_2)/P(t_1) = 4.2$, hence $P(t_2) = 4.2 * 19.1 = 80.2$ psia (0.55 MPa). An approximate final temperature can be estimated from Figure 4A-2 by adding to the temperature found from the figure the difference between $T(t_1)$ and 536°R (298 K).

$T(t_2) \approx 1230 + 30 = 1260$ K (2270°R) (4A-16)

When applicable, the use of Figures 4A-3 and 4A-4 is simpler than using Figures 4A-1 and 4A-2. These figures are applicable when the conditions at the start of the accident are near $P(t_0) = 1$ atm (0.101 MPa), $T(t_0) = 540$°R (300 K) and the conditions just before the combustion are steam saturated. It should be noted that the curves for constant $T(t_1)$ in the two figures correspond to varying pressure, $P(t_1)$, and varying steam mole fraction. At all points on the curves, the composition has been adjusted to saturation conditions. Much of the work in describing the conditions at time t_1 is not needed here because that information has been incorporated into the figures. For a temperature of 590°R (328 K), we determine that $P(t_2) = 4.9$ atm = 72.0 psia (0.50 MPa), and $T(t_2) = 2340$°R (1300 K).

The results of thermochemical calculations in a computer give values $P(t_2)$ = 74.4 psia (0.51 MPa), $T(t_2)$ = 2401 °R (1334 K). The difference between the results (summarized in Table 4A-1) gives an indication of the accuracy to be expected from the simple graphical methods.

If the pressure and temperature before the combustion are accurately measured and the hydrogen mole fraction measurement is absent or less accurate, the hydrogen mole fraction can be estimated (assuming saturation) from the relations,

$$P_{H_2} = P - P_A - P_S \tag{4A-17}$$

$$X_{H_2} = P_{H_2}/P \tag{4A-18}$$

Some hydrogen detectors may remove the water vapor content of the hydrogen:air:steam mixture. In this case the measured hydrogen mole fraction (of the dry hydrogen:air mixture) will be larger than the value in the original mixture. The correction required to recover the original value is

$$X_{H_2} = (1 - X_S)X_{H_2} \tag{4A-19}$$

where X_H is the hydrogen mole fraction in the dry hydrogen:air mixture and X_S is the steam mole fraction in the original hydrogen:air:steam mixture

$$X_S = P_S/P \tag{4A-20}$$

Table 4A-1 Computation of adiabatic, constant-volume pressure and temperature

	Time Before Accident (t_0)	Time Before Combustion (t_1)	Time After Combustion (t_2) Using Figures 4A-1 & 4A-2	Time After Combustion (t_2) Using Figures 4A-3 & 4A-4
Pressure - psia (MPa)	14.7 (0.101)*	19.1 (0.131)	80.2 (0.55)	72.0 (0.50)
Temperature - °R (K)	560 (311)*	590 (328)*	2270 (1260)	2340 (1300)
Hydrogen Mole Fraction	0.0*	0.1*		
Air Partial Pressure - psia (MPa)	14.2 (0.098)	15.0 (0.103)		
Steam Partial Pressure - psia (MPa)	0.48 (0.0033)	2.2 (0.015)		
Hydrogen Partial Pressure - psia (MPa)	0.0 (0.0)	1.90 (0.013)		

* Data directly from measured initial conditions

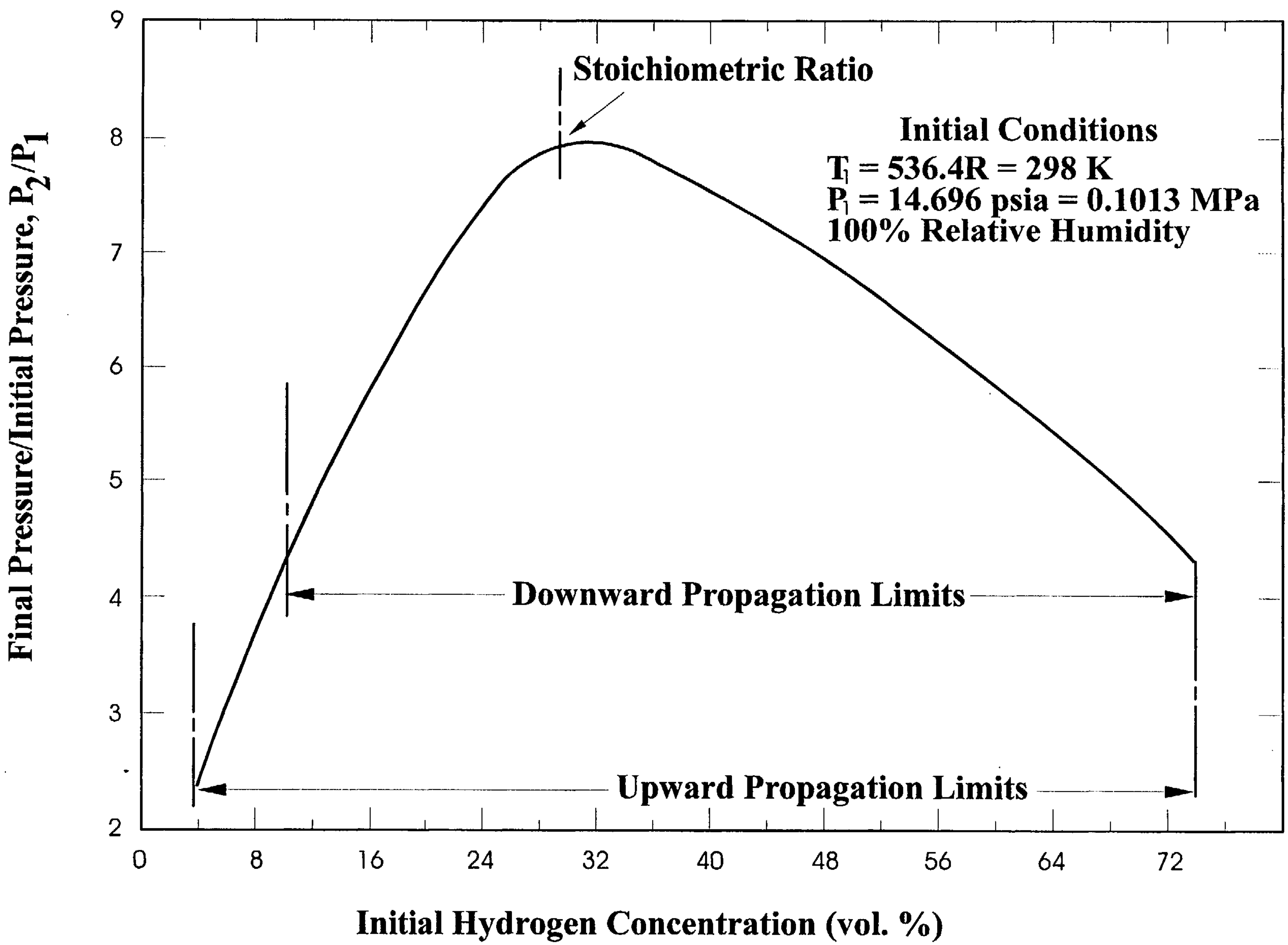


Figure 4A-1 **Theoretical adiabatic, constant-volume combustion pressure for hydrogen: air mixtures**

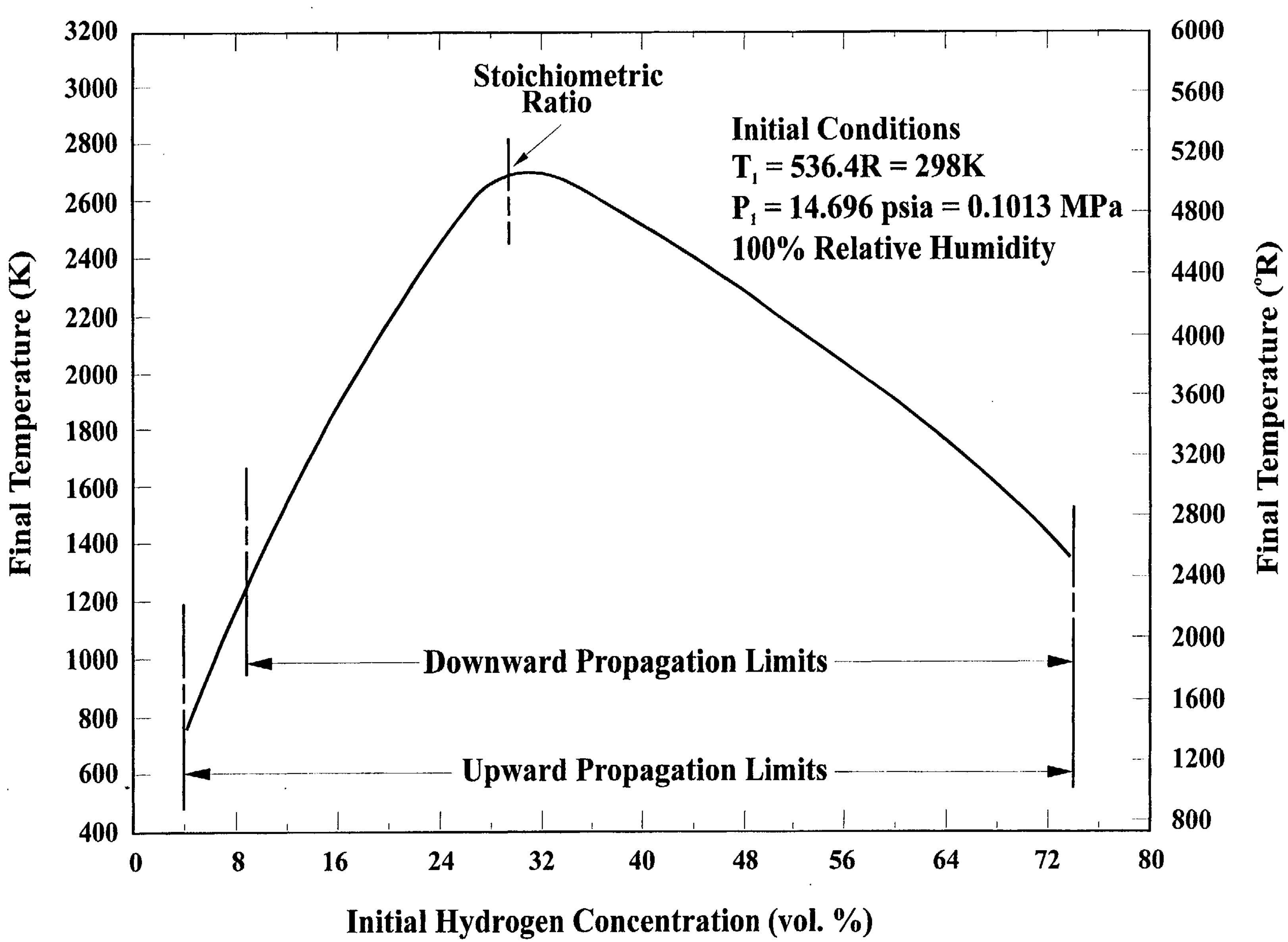


Figure 4A-2 Theoretical adiabatic, constant-volume combustion temperature for hydrogen: air mixtures

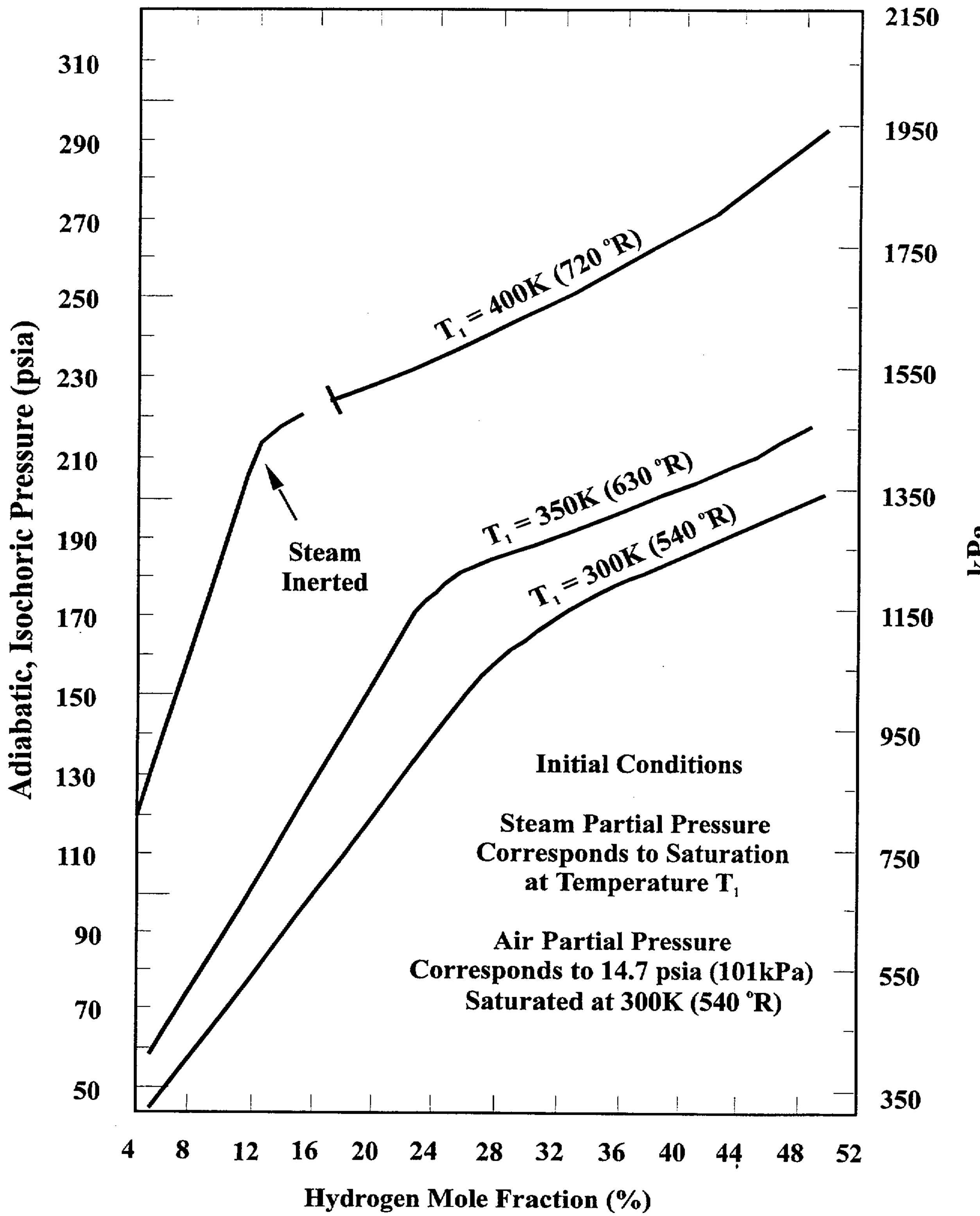


Figure 4A-3 Adiabatic, constant-volume combustion pressure for various containment initial conditions

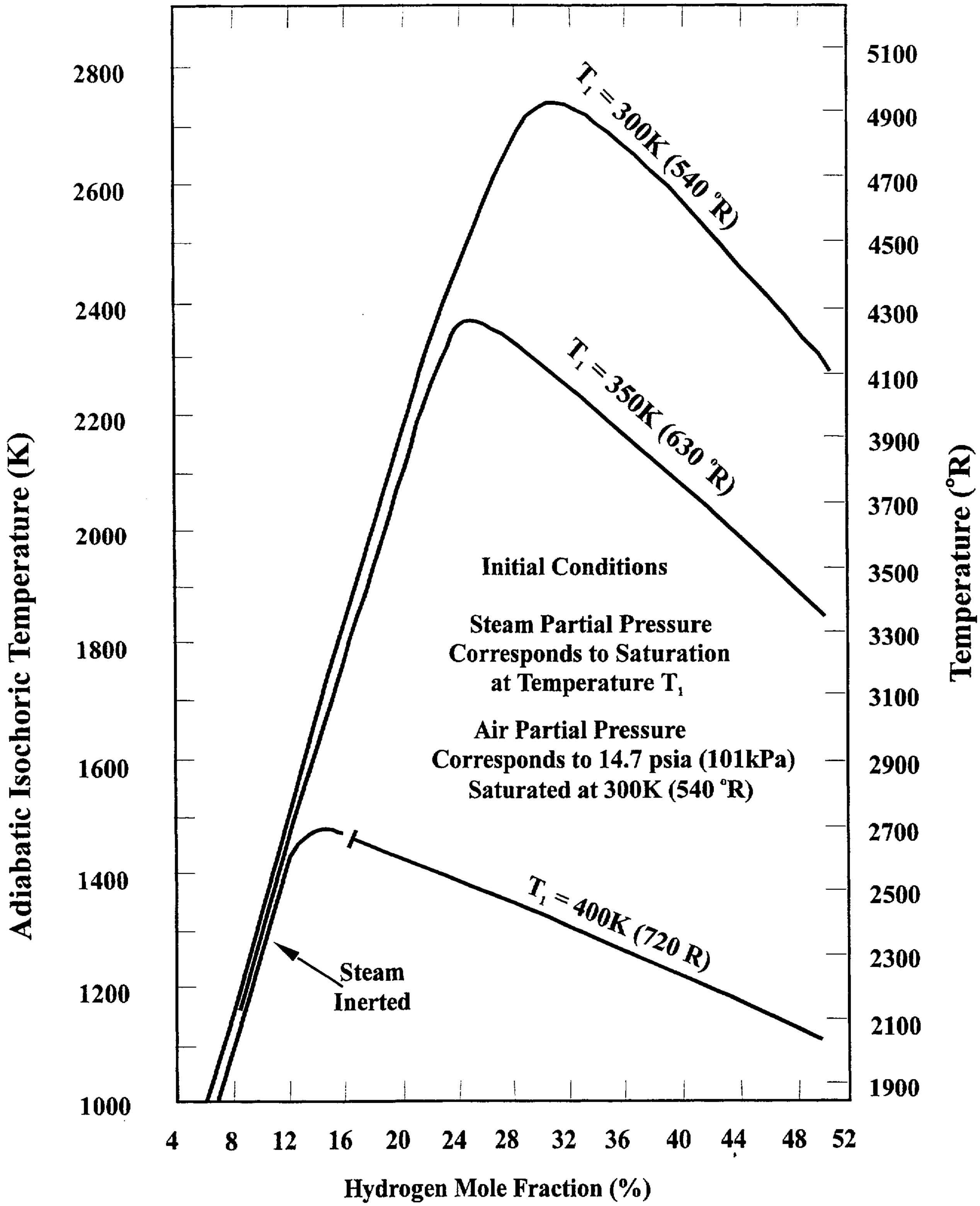


Figure 4A-4 Adiabatic, constant-volume combustion temperature for various containment initial conditions

5.0 Offsite Accident Impacts

5.0.1 Introduction

Chapters 1 through 4 emphasize the importance of the defense-in-depth philosophy for preventing severe accidents or containing radionuclide releases given such accidents. This chapter carries the defense-in-depth philosophy one step further. It discusses radionuclide releases that, however unlikely, could occur; the radiation doses and associated health effects that could result; actions that can be taken to protect the public; and the U.S. emergency planning process for implementing such protective actions.

5.0.2 Learning Objectives for Chapter 5

After completing this chapter, the student should be able to:

1. Describe the location in the plant of radioactive material that could potentially cause offsite injuries or fatalities and the extent of plant damage required for its release.

2. Describe the characteristics of the radioactive source term that have an important effect on offsite doses and the plant design features that could have a major impact on such characteristics.

3. Explain the impact of wind speed, stability class, radioactive decay, ground deposition, and rainfall on the rate of decrease in offsite dose versus distance.

4. Characterize our current ability to accurately project source term characteristics and offsite doses that could result from a severe core-damage accident.

5. Describe the roles and efficacy of evacuation, sheltering, ad hoc respiratory protection, and administration of stable iodine in protecting the public from potential nuclear power plant releases of radioactive materials.

6. Describe the Chernobyl source term and the actions taken to protect the public from this source term. Compare these with the current LWR source terms and NRC guidance on protective actions.

7. Indicate the primary responsibilities of the licensee, state and local agencies, and the NRC during a nuclear power plant emergency.

8. Describe plume exposure Emergency Planning Zone and the ingestion pathway Emergency Planning Zone.

9. Explain what Emergency Action Levels (EALs) are.

10. List the four classes of emergencies in order of increasing severity and indicate which require official notification and which require offsite protective actions.

11. Describe the functions of the Technical Support Center (TSC) and the Emergency Operations Facility (EOF) during a nuclear power plant emergency.

5.1 Source Terms

As indicated in Chapters 3 and 4, if the energy contained in the core of a nuclear power plant is not controlled, considerable damage can be done to the fuel, cladding, reactor vessel, and even the containment--the plant barriers that normally contain the core radionuclides. Even if the reactor is shut down, the substantial energy generated by the decay of fission products (decay heat) can lead to damage to these barriers. If sufficient quantities of radionuclides are released to the environment as a result of such damage, offsite health effects may result. This subsection discusses the quantities and characteristics of radionuclide releases to the environment (source terms). Transport of the released radionuclides in the environment and associated offsite doses and health effects are discussed in Section 5.2.

5.1.1 Radionuclide Inventories

The conventional unit that is used to quantify the radioactivity of a material is the curie (Ci). One curie of material undergoes radioactive decay at the rate of 3.7×10^{10} nuclear disintegrations per second, which is the radioactivity of one gram of pure radium. The corresponding Standard International (SI) unit of radioactivity is the becquerel (Bq). One becquerel is one nuclear disintegration per second, so 1 Ci = 3.7×10^{10} Bq.

Table 5.1-1 shows the principal components of the 5 billion or so curies of radioactive materials in the core of a 3300 MWt light water reactor 30 min after shutdown according to their relative volatilities.[1,2] Of the groups listed, radionuclides of the noble gases krypton (Kr) and xenon (Xe) are the most volatile and, consequently, the most likely to be released from the plant to the environment during an accident. Up to 100% of the noble gases could be released in severe accidents involving containment failure or bypass. Radioactive iodine and cesium, which rank second in volatility, could also be released in substantial quantities. Radioiodine can concentrate in the thyroid. As a result, small quantities of radioiodine can cause damage to the thyroid gland. Long-lived radioactive cesium is a potential source of long-term offsite dose (e.g. from Chernobyl).

Table 5.1-2 shows radionuclide inventories of the volatile noble gases and iodine in various plant systems.[2] Note that the vast majority of this volatile radioactive material is contained in the core. All other reactor systems contain less than one-half of 1% of the xenon, krypton, and iodine activity in the core. Because radioactive cesium is long-lived, the spent fuel pool can contain more than the core; however, the driving force (decay heat) for release is much larger in the core.

5.1.2 Source Term Characteristics

Radionuclides would be released to the environment as gases such as krypton # (Kr), xenon (Xe), and iodine (I_2); aerosol particles of water soluble substances such as cesium iodide (CsI), cesium hydroxide (CsOH), and strontium hydroxide ($Sr(OH)_2$); or slightly soluble oxides of tellurium (Te), ruthenium (Ru), and lanthanum (La). Generally, a major release (source term) from a nuclear power plant can be viewed as a cloud (called the plume) of radioactive gases, aerosol particles, and water vapor (mist). As indicated in Figure 5.1-1, the plume could be released continuously over a long time period, or it could be released as a very short puff. It could be released at ground level or higher.

Source terms are characterized by the fractions of the core inventory of radionuclides that are released to the environment and other attributes that can have an important effect on offsite doses and the numbers and types of offsite health effects. Such attributes include the start time and duration of the release, the size distribution of the aerosols released, the elevation of the release, and the energy released with the radioactive material. For example, if the plume is hot, rise due to buoyancy may loft the plume over people living near the plant thereby limiting the doses they receive. Also, if the release is slow (takes a long time), shifts in wind direction may cause more people to be exposed but may also cause the exposure to any stationary group to be reduced. Such effects are discussed further in Section 5.2.

The isotopic composition of a source term is important because it determines decay rates and thus radiation exposure rates. Rapidly decaying (halflives under a minute) nuclides deliver most of their dose quickly at short distances from their release point. Conversely, slowly decaying nuclides may deliver dose over many years out to great distances from their release point. The chemical and physical form of the released radioactive materials also influences offsite doses. For example, if only noble gases are released, deposition on the ground and incorporation into the food chain does not take place thereby eliminating several important long-term exposure pathways. Conversely, if the released radioactive material is all in the form of water insoluble particles too large to be respirable, lung exposure due to inhalation does not occur.

5.1.3 Magnitude of Release Required to Cause Offsite Health Effects

It is not obvious from examining the values of a source term's characteristics what the potential health impact would be to the public. It is, however, easy to demonstrate that the release of only a relatively small fraction of the core inventory to the environment could be significant. Table 5.1-3 shows that, under certain meteorological conditions, a prompt release of only 10^6 Ci of noble gases or 600 Ci of halogens could, based on current federal guidelines (discussed in Appendix 5A), result in doses sufficient to warrant taking actions to protect persons within a mile of the plant. Doses at least ten times higher than federal protective action guides would be required to induce early injuries or fatalities. For example, under the same poor meteorological conditions assumed above, Table 5.1-3 indicates that a release of 10^7 Ci of noble gases or 2.4×10^4 Ci of halogens could result in doses exceeding the thresholds for acute, radiation-induced injuries.

As indicated in Section 5.1.1, only the core, the spent-fuel storage pool, and the reactor coolant contain more than 10^6 Ci of noble gases or 6×10^2 Ci of halogens. Accidents not involving one of these three regions (e.g., gas-decay tank rupture) would not, therefore, result in doses requiring offsite protective actions. It should be emphasized, however, that one cannot project offsite doses and health effects based solely on the Curies of radionuclides released. For example, as indicated in Table 5.1-3, the average annual release of noble gases for a LWR is 1000 Ci, but this release occurs gradually during all kinds of weather conditions. It therefore results in very low offsite doses.

Figure 5.1-2 compares the average annual releases of noble gases and iodine from U.S. plants to the estimated releases that occurred as a result of the TMI-2 and Chernobyl accidents. Clearly the uncontained Chernobyl release, which is discussed in Section 5.1.7, was large by any measure.

The TMI-2 release, much of which occurred during controlled venting of the containment, was very modest in comparison. Nevertheless, as discussed in Sections 5.3.7 and 5.4, current guidelines would call for a precautionary evacuation for an accident like that at TMI-2.

In addition to core damage, an accident resulting in offsite injuries and/or fatalities would require a direct pathway to the atmosphere and a driving force (e.g., steam). The radioactive material released from the core would have to move through the reactor coolant system and containment without being significantly filtered or removed by other methods such as containment sprays, ice condensers, fan coolers, or suppression pools. Even if such engineered safety features failed, over time natural removal processes (e.g., gravitational settling) would remove most particulate fission products from the atmosphere of an intact containment. Therefore, if the containment holds for several hours and the containment sprays or other removal systems work, early injuries or fatalities would be unlikely.

Figure 5.1-3 uses an event tree to display the potential public health consequences due to core-damage accidents. Moving from left to right in the figure, yes/no answers to questions at the top result in a series of branches, possibly to offsite consequences. For example, if only the radioactive material contained in the fuel pins (gaps) is released with late containment failure, the offsite consequences would be small (branch 7). If all answers are yes, extremely severe offsite consequences are possible (branch 1). As Figure 5.1-3 indicates that offsite health effects are likely only if core melting and containment failure (or bypass) occur. The implications of this observation in deciding when to initiate offsite protective actions are discussed in Section 5.3.7 and 5.4.

5.1.4 Design Features That Impact Source Terms

In Chapter 4, performance of the containment was described with respect to the timing of containment failure and the magnitude of resulting leakage to the environment. Environmental source terms are, however, affected by more than just the mode and timing of containment failure. The following paragraphs describe the effect of different safety systems and plant features on the magnitude of source terms. A common measure of the capability of a system or feature to remove radioactive aerosols or vapors in the decontamination factor (DF), which is the ratio of the inlet concentration to the outlet concentrations. Table 5.1-4 provides a summary of the more significant DFs associated with the features that are discussed.

5.1.4.1 Suppression Pools

BWR suppression pools can be very effective in scrubbing (removing) radionuclides that accompany steam and noncondensible gases bubbling up through the pool. The pool water retains soluble vapors and aerosols but provides little attenuation of noble gas fission products. Although Regulatory Guide 1.3 suggests not allowing credit for fission product scrubbing by BWR suppression pools,[3] Standard Review Plan Section 6.5.5 was revised to allow such credit.[4] The Reactor Safety Study assumed a DF of 100 for sub-cooled pools and 1.0 for saturated pools.[5] NUREG-1150 calculations based on more sophisticated models indicate decontamination factors ranging from 1.2 to 4000 with a median value of about 80.[6]

Some of the most important radionuclides, such as isotopes of iodine and cesium, and tellurium, are primarily released from fuel while it is still in the reactor vessel.

Risk-dominant accident sequences in BWRs are typically initiated by transients rather than pipe breaks. In transients, the in-vessel release is directed to the suppression pool rather than being released to the drywell. As a result, the in-vessel release is subjected to scrubbing in the suppression pool, even if containment failure has already occurred. Because the early health effects are often caused by early releases of volatile radioactive materials, the suppression pool is one of the reasons the likelihood of early fatalities is low for the BWR designs analyzed in NUREG-1150.

If not bypassed, the suppression pools can also be effective in scrubbing ex-vessel releases. Suppression pool bypass places an upper limit on the suppression pool decontamination factor. For example, if as little as 1% of the flow bypasses the suppression pool, the effective DF factor must be less than 100.

Experiments have shown that solution pH is a major factor in determining the amount of molecular iodine (I_2) and organic iodine found in solution. Unless chemical additives are introduced to control it, the pH tends to be reduced by long-term radiolysis. This favors the recombination of I^- ions to form I_2. Increasing pool temperatures and especially pool boiling can, in turn, cause I_2 to move from the liquid to the vapor phase.[7] Hence, it is possible that suppression pools would scrub substantial amounts of iodine in the early phases of an accident, only to re-evolve it later as I_2. In NUREG-1150, such re-evolution of iodine was judged to be important in accident sequences where the containment failed and the suppression pool eventually boiled.

There is presently no requirement for pH control in BWR suppression pools. It may well be that additional materials likely to be in the suppression pool as a result of a severe accident (such as cesium borate, cesium hydroxide, and core-concrete decomposition products) would counteract any reduction in pH from radiolysis and would ensure that the pH level remained sufficiently high to preclude re-evolution of iodine. If credit is to be given for long-term retention of iodine in the suppression pool, maintenance of the pH at or above a level of 7 must be demonstrated.[8]

5.1.4.2 Drywell-Wetwell Configuration

Depending on the timing and location of containment failure, the suppression pool may also be effective in scrubbing the release occurring during core-concrete attack or re-evolved from the reactor coolant system after vessel failure. In the NUREG-1150 analyses for Peach Bottom (Mark I containment), containment failure was found likely to occur in the drywell early in the accident. Thus, in many scenarios the suppression pool was not effective in mitigating the delayed release of radioactive material.

The Mark III design has the advantage, relative to the Mark I and Mark II designs, that the wetwell boundary completely encloses the drywell, in effect providing a double barrier to radioactive material release. As long as the drywell remains intact, any release of radioactive material from the fuel is subject to decontamination by the suppression pool. With the Mark III drywell intact, the environmental source terms is reduced to a level at which early fatalities would not be expected to occur, even for early failure of the outer containment. However, for Grand Gulf (Mark III containment), drywell failure accompanies approximately one-half of the early containment failures in NUREG-1150, and the suppression pool is ineffective in mitigating ex-vessel releases for such scenarios.

5.1.4.3 Containment Sprays

Containment sprays can be effective in reducing airborne concentrations of radioactive aerosols and vapors. In the Surry (subatmospheric) and Zion (large dry) designs, approximately 20% of the NUREG-1150 core meltdown sequences were predicted to eventually result in delayed containment failure or basemat meltthrough. The effect of sprays, in those scenarios in which they are operational for an extended time, is to reduce the concentration of radioactive aerosols airborne in the containment to negligible levels in comparison with non-aerosol radionuclides (e.g., noble gases). Qualified sprays can reduce airborne aerosol activities by an order of magnitude in 15 to 20 minutes. For shorter periods of operation, sprays would be less effective but could still have a substantial mitigative effect on the release. Without sprays, an order of magnitude reduction in airborne aerosol activities would typically take about 10 hours.

The Sequoyah (ice condenser) design has containment sprays for the purpose of condensing steam that might bypass the ice bed, as well as for use after the ice has melted. The effects of the sprays and ice beds in removing radioactive material are not completely independent since they both tend to preferentially remove larger aerosols.

5.1.4.4 Ice Condenser

The ice beds in an ice condenser containment remove radioactive material from the air by processes that are very similar to those in the BWR pressure-suppression pools. The decontamination factor is very sensitive to the volume fraction of steam in the flowing gas, which in turn depends on whether the air-return fans are operational. For a typical case when the air-return fans are on, the magnitude of the decontamination factors was assessed to be in the range from 1.2 to 20, with a median value of 3. Thus, the effectiveness of the ice bed in mitigating the release of radioactive material is likely to be substantially less than for a BWR suppression pool.

5.1.4.5 Reactor Cavity Flooding

The configuration of a PWR reactor cavity or BWR pedestal region affects the likelihood of water accumulation and water depth below the reactor vessel. In some PWRs, water reaching the containment floor does not flow into the reactor cavity, and unless the spray system is operating, the cavity will be dry at vessel failure. In the Peach Bottom (Mark I) design, there is a maximum water depth of approximately 2 feet on the pedestal and drywell floor before water would overflow into the suppression pool via the downcomer. Other designs investigated such as Sequoyah and Zion have substantially greater potential for water accumulation in the pedestal or cavity region. In the Sequoyah design, the water depth could be as much as 40 feet.

If a coolable debris bed is formed in the cavity or pedestal, and makeup water is continuously supplied, core-concrete release of radioactive material would be avoided. Even if molten core-concrete interaction occurs, a continuous overlaying pool of water can substantially reduce the release of radioactive material to the containment in the same way suppression pools mitigate releases.

5.1.4.6 Building Retention

In NUREG-1150, radionuclide retention was evaluated for the Peach Bottom reactor building. (An evaluation was not made for the portion of the reactor building that surrounds the Grand Gulf containment, which was assessed to have little potential for retention.) The range of aerosol decontamination factors for the Peach Bottom reactor building subsequent to drywell rupture was 1.1 to 80, with a median value of 2.6. The location of drywell failure affects the potential for reactor building decontamination. Leakage past the drywell head to the refueling building was assumed to result in very little decontamination. Failure of the drywell by meltthrough resulted in a release that was subjected to a decontamination factor of 1.3 to 90 with a median value of 4.

In the NUREG-1150 analyses of PWR interfacing LOCA sequences, some retention of radionuclides was assumed in the auxiliary building (in addition to water pool decontamination for submerged releases). In the Sequoyah analyses, retention was enhanced by the actuation of the fire spray system.

5.1.4.7 BWR Containment Venting

In the Peach Bottom and Grand Gulf designs, procedures have been implemented to intentionally vent the containment to avoid overpressure failure. By venting from the wetwell air space (in Peach Bottom) and from the containment (in Grand Gulf) assurance is provided that, subsequent to core damage, the release of radionuclides through the vent line will have been subjected to decontamination by the suppression pool.

As discussed in Chapter 4, containment venting to the outside can substantially improve the likelihood of recovery from a loss of decay heat removal and, as a result, reduce the frequency of severe accidents. The results of NUREG-1150 indicate, however, only limited benefits in consequence mitigation for the existing venting procedures and hardware. Uncertainties regarding the decontamination factor for the suppression pool and the re-evolution of iodine from the suppression pool are quite broad. As a result, the consequences of a vented release are not necessarily minor. Furthermore, the effectiveness of venting may, for some plants, be limited by the high likelihood of early containment failure mechanisms that would bypass the vent.

5.1.5 Source Term Uncertainty

Clearly, the magnitude of the source term varies depending on whether or not containment fails, when it fails, where and how it fails, and the effectiveness of engineered safety features in mitigating the release. However, even if details regarding the nature of containment failure and engineered safety feature performance are known, the uncertainty in predicting severe accident phenomena is still large.

A major shortcoming of the 1975 Reactor Safety Study was its limited treatment of the uncertainties in severe accident source terms. In the intervening years, particularly subsequent to the Three Mile Island accident, major experimental and code development efforts have broadly explored severe accident behavior. In the comprehensive NUREG-1150 study, which was published in 1989, care was taken to assess and display the uncertainties associated with the analysis of accident source terms. Many of the severe accident issues that are now recognized as the greatest sources of uncertainty were completely unknown to the earlier Reactor Safety Study analysts.

In the 1975 Reactor Safety Study, source terms were developed for nine release categories ("PWR1" to "PWR9") for the Surry plant and five release categories for the Peach Bottom plant ("BWR1" to

"BWR5"). In NUREG-1150, source terms were developed for a much larger number of accident progression bins. For each accident progression bin, an estimate of the uncertainty in the release fractions for each of the elemental groups was obtained.

Figure 5.1-4 provides a comparison of an important large release category (PWR2) from the Reactor Safety Study with a comparable aggregation of accident progression bins (early containment failure, high reactor coolant system pressure) from the NUREG-1150. The Reactor Safety Study results in this case are clearly conservative when compared to the NUREG-1150 results. Figure 5.1-5 compares results for an isolation failure in the wetwell region from the Reactor Safety Study, release category BWR4, with the venting accident progression bin from NUREG-1150. The Reactor Safety Study results are very similar to the mean release terms for the venting bin, with the exception of the iodine group, which is higher because of the late release mechanisms (re-evolution from the suppression pool and the reactor vessel) considered in the NUREG-1150 study. Overall, the comparisons indicate that the source terms in the Reactor Safety Study are in some instances higher and in other instances lower than those in NUREG-1150. However, for the early containment failure scenarios that have the greatest impact on risk, the Reactor Safety Study source terms are larger than the mean values of the NUREG-1150 study and are typically at the upper bound of the uncertainty range.

5.1.6 Revised LWR Source Term

In 1962, (see Sections 1.2.6 and 2.1.2) the Atomic Energy Commission issued Technical information document TID-14844,[9] which postulated a release of fission products from the reactor vessel into containment to be used for calculating offsite doses in accordance with the reactor siting criteria of 10 CFR Part 100. The TID-14844 release was based on a postulated core melt accident and the 1962 understanding of fission product behavior. In addition to evaluations of site suitability and plant mitigation features such as containment sprays and filtration systems, the TID-14844 release has influenced post-accident radiation environments for which safety-related components are qualified, post-accident habitability requirements for the control room, post-accident sampling systems, and post-accident accessibility considerations.

For currently licensed plants, the characteristics of the fission product release from the core into the containment were derived from TID-14844 as set forth in Regulatory Guides 1.3 and 1.4.[3,10] The release consists of 100% of the core inventory of noble gases and 50% of the iodines (half of which are assumed to deposit on interior surfaces very rapidly). These values were based largely on experiments performed in the late 1950s involving heated irradiated UO_2 pellets. The TID-14844 release also includes 1% of the remaining solid fission products, but these were dropped from consideration in Regulatory Guides 1.3 and 1.4. A 1% release of solid fission products is considered in certain areas such as equipment qualification.

In Regulatory Guides 1.3 and 1.4, the release to containment is assumed to occur instantaneously (with the initial blowdown of the reactor vessel in a LOCA). Further, the iodine chemical form is assumed to be 91% elemental (I_2), 5% particulate, and 4% organic. Organic iodine is not readily removed by containment sprays or filter systems. These assumptions have significantly affected the design of engineering safety features, particularly containment isolation valve closure times.

Recently, the NRC developed revised accident source terms based on careful evaluation of NUREG-1150 accident scenarios and associated source term estimates. This evaluation is documented in NUREG/CR-5747,[11] and the resulting revised source terms are published in NUREG-1465.[8] NUREG-1465 source terms were developed for regulatory application to future LWRs but may also be used to evaluate changes to regulatory requirements for existing plants.

Tables 5.1-5 and 5.1-6 present the NUREG-1465 release fractions, which are intended to be representative or typical (rather than conservative or bounding) values, except for the initial appearance of fission products from the failed fuel. This so-called gap release is set by the design basis initiator that leads to earliest cladding failure.

In contrast to the instantaneous releases postulated in Regulatory Guides 1.3 and 1.4, the NUREG-1465 releases are distributed in time to reflect the degree of fuel melting and relocation, reactor pressure vessel integrity, and, as applicable, attack upon concrete below the reactor cavity by molten core materials. The timing aspects are typical of a low pressure core-melt scenario, except that the onset of the release of gap activity is based upon the earliest calculated time of fuel rod failure under accident conditions.

NUREG-1465 also considers the chemical form of iodine in the containment and concludes that no more than 3% of the airborne iodine would be converted to organic form. When pH is controlled at values of 7 or greater within the containment, elemental iodine is assumed to comprise no more than 5% of the total iodine released to containment, and organic iodine no more than 0.15 percent (3 percent of 5 percent).

5.1.7 The Chernobyl Source Term

The initial release of radioactive materials from Chernobyl 4 reactor resulted from the explosions that destroyed the reactor core on April 26, 1986. Releases continued over a relatively long period of time and occurred in several stages, each of which differed in radionuclide composition and intensity. Figure 5.1-6 shows the estimated time-dependance of the release for the first 10 days following the accident when a stream of hot air carried particulates, noble gases and volatile radionuclides from the destroyed reactor up to the atmosphere[12].

From April 26 until May 1 the release rate decreased, perhaps under the influence of the measures undertaken to extinguish the burning graphite and cover the core, although there is no consensus on this point. During this stage the release consisted of finely dispersed fuel particles entrained by the escaping hot air and graphite combustion products. The radionuclide composition during this stage was similar to that of the fuel.

From May 1 through May 5, there was a rapid increase in the release rate, at first dominated by volatile radionuclides (especially iodine isotopes), after which the composition again became similar to that of the fuel. The release during this stage was attributed to heating of the fuel to over 2000°C from the residual heat. The released radionuclides were associated with aerosols of fuel and graphite combustion products.

On May 6, there was an abrupt decrease in the release rate perhaps due to chemical interactions of the radionuclides with the materials introduced into the core. A complete explanation of the sudden decrease is not available. Measurable releases of activity continued during the remainder of May. For example, there was a significant peak 20 to 21 days after the beginning of the accident, and a less pronounced peak in the 25- to 30-day period.[13]

Table 5.1-7 summarizes the quantitative estimates of the magnitude of the releases from the Chernobyl-4 reactor. Based on various data, including measurements of Cs-137 in fuel masses inside the reactor sarcophagus and in the global fallout, the total core cesium released to the environment is estimated to be 33%±10%, or 2.3±0.7 MCi.[14] The relative constancy of the ratio of I-131 and Cs-137 activities in fallout observed in the majority of European countries indicates that 50% to 60% of the initial I-131 inventory or 40 to 50 MCi of I-131 was released. Accounting for radioactive decay in the reactor in the period of the release, this corresponds to a total activity of 30 to 35 MCi released into the atmosphere. The ratio of Iodine-131 released as gases to that released as aerosols was approximately 3:7 over the first two weeks after the accident, but, by the third week, the ratio changed to approximately 10:1.[13]

Comparing the Chernobyl release fractions in Table 5.1-7 with the revised LWR source terms in Tables 5.1-5 and 5.1-6 shows the volatile release fractions (noble gases, iodine, and cesium) are similar in magnitude. On the other hand, releases of nonvolatile species at Chernobyl resulted from the initial explosions and the ensuing fires. The amount of fuel released is estimated to be 3.5%±0.5% of the fuel inventory in the core.[12,14] This far exceeds the NUREG-1465 release fractions for nonvolatile species. It should also be emphasized that the NUREG-1465 releases are to the containment, not directly to the atmosphere as occurred at Chernobyl.

5.1.8 On-Line Source Term Monitoring

As indicated in Section 5.1.5, it is not possible to predict with certainty the source term that would result from a given plant damage state. What, then, is the feasibility of on-line monitoring to measure source term characteristics during an accident?

For accidents where the total release is through a monitored pathway (e.g., the stack), it may be possible to adequately characterize the release. As part of the upgrades that followed the TMI-2 accident, on-line radiation monitors capable of measuring the noble gases released through plant vents were installed. Noble gases are not considered as great a threat to the public as the halogens and other chemical groups listed in Table 5.1-1. The presence of halogens and other chemical groups must, in general, be determined through analysis of samples taken during the release. Unfortunately, this can require several hours. Nevertheless, at a minimum, the magnitude of a stack release can generally be estimated if the monitors stay on scale.

By their very nature, however, releases resulting in offsite dose high enough to cause early health effects most likely could not be characterized by existing effluent monitors. A release resulting in early offsite health effects (death and injuries) would have to be fast, direct, and unfiltered. Most important are potential releases due to major containment failure. As illustrated in Figure 5.1-7, such a release would be through an unmonitored path to the atmosphere. Effluent-monitoring systems located in routinely monitored release paths (e.g., stacks) would not be able to assess the extent and the characteristics of such a release.

Table 5.1-1 Radioactive materials in a large [3300-MWt] light water reactor core grouped by relative volatility

Group	Isotope	Half-life	Core Inventory (Ci)	Group Total (Ci)
	Kr-85	10.72 y	6.69E+05	
Noble Gases	Kr-85m	4.48 h	3.13E+07	3.84E+08
	Kr-87	1.27 h	5.72E+07	
	Kr-88	2.54 h	7.74E+07	
	Xe-133	5.245 d	1.83E+08	
	Xe-135	9.09 h	3.44E+07	
	I-131	8.04 d	8.66E+07	
Halogens	I-132	2.30 h	1.28E+08	7.71E+08
	I-133	20.8 h	1.83E+08	
	I-134	52.6 m	2.01E+08	
	I-135	6.61 h	1.73E+08	
	Cs-134	2.062 y	1.17E+07	
Alkali Metals	Cs-136	13.16 d	3.56E+06	2.18E+07
	Cs-137	30.17 y	6.53E+06	
	Rb-86	18.66 d	5.10E+04	
	Sb-127	3.85 d	7.53E+06	
Tellerium Group	Sb-129	4.40 h	2.67E+07	2.13E+08
	Te-127	9.35 h	7.28E+06	
	Te-127m	109 d	9.63E+05	
	Te-129	1.16 h	2.50E+07	
	Te-129m	33.6 d	6.60E+06	
	Te-131m	30 h	1.26E+07	
	Te-132	3.26 d	1.26E+08	
	Ba-139	1.396 h	1.70E+08	
Barium, Strontium	Ba-140	12.746 d	1.68E+08	6.95E+08
	Sr-89	50.52 d	9.70E+07	
	Sr-90	29.1 y	5.24E+06	
	Sr-91	9.5 h	1.25E+08	
	Sr-92	2.71 h	1.30E+08	
	Co-58	70.88 d	8.71E+05	
Noble Metals	Co-60	5.271 d	6.66E+05	5.94E+08
	Mo-99	2.7476 d	1.65E+08	
	Rh-105	35.4 h	5.53E+07	
	Ru-103	39.27 d	1.23E+08	
	Ru-105	4.44 h	7.98E+07	
	Ru-106	1.02 y	2.79E+07	
	Tc-99m	6.01 h	1.42E+08	
	Am-241	432.7 y	3.13E+03	
Lanthanides	Cm-242	162.8 d	1.20E+06	1.54E+09
	Cm-244	18.1 y	7.02E+04	
	La-140	1.678 d	1.72E+08	
	La-141	3.90 h	1.57E+08	
	La-142	1.54 h	1.52E+08	
	Nb-95	34.97 d	1.41E+08	
	Nd-147	10.98 d	6.52E+07	
	Pr-143	13.57 d	1.46E+08	
	Y-90	2.67 d	5.62E+06	
	Y-91	58.5 d	1.18E+08	
	Y-92	3.54 d	1.30E+08	
	Y-93	10.2 h	1.47E+08	
	Zr-95	64.02 d	1.49E+08	
	Zr-97	16.8 h	1.56E+08	
	Ce-141	32.50 d	1.53E+08	
Cerium Group	Ce-143	1.38 d	1.48E+08	2.15E+09
	Ce-144	284.6 d	9.20E+07	
	Np-239	2.355 d	1.75E+09	
	Pu-238	87.7 y	9.90E+04	
	Pu-239	24100 y	2.23E+04	
	Pu-240	6560 y	2.82E+04	
	Pu-241	14.4 y	4.74E+06	

Table 5.1-2 Typical inventories of noble gases and iodine in reactor systems

Location	Inventory (Ci)	
	Noble gases (Xe, Kr)	Iodine (I)
Reactor core total	4.0E+8	7.5E+8
Reactor core gap[a]	3.0E+7	1.4E+7
Spent fuel storage pool	1.0E+6	5.0E+5[b]
Primary coolant[c]	1.0E+4	6.0E+2[c]
Pressurized Water Reactor--other systems		
Waste gas storage tank	1.0E+5	1
Boiling Water Reactor--other systems	1.0E+4[d]	25[d]
Steam line	5.0E+3	0.25
Waste gas treatment system		
Shipping cask	1.0E+4	1

[a]Gap between UO_2 fuel and Zircaloy cladding.
[b]One-third of the core is 30 days old; the rest is 1 year old.
[c]Nominal value, iodine levels can much higher or lower (factor of 10) depending on fuel leakage.
[d]Ci/hr (circulating).

Table 5.1-3 Illustrative noble gas and halogen releases

	Activity Released (Ci)	Fraction of 3000 MWt Core Inventory[a]
Noble Gases (Krypton and Xenon)		
Average annual release for a LWR[b]	$1x10^3$	$3x10^{-6}$
Prompt release resulting in 5 rem committed stomach dose equivalent[c] to unprotected individual one mile downwind under poor meteorological conditions[d]	$1x10^6$	$3x10^{-2}$
Prompt release resulting in 50 rem acute stomach dose[e] to unprotected individual one mile downwind under poor meteorological conditions.[d]	$1x10^7$	$3x10^{-2}$
Halogens (Iodine)		
Average annual release for a LWR[b]	0.13	$1x10^{-10}$
Prompt release resulting in 5 rem acute thyroid dose[c] to unprotected infant one mile downwind under poor meteorological conditions[d]	$6x10^2$	$1x10^{-6}$
Prompt release resulting in 200 rem thyroid dose[f] to unprotected infant one mile downwind under poor meteorological conditions[d]	$2.4x10^4$	$3x10^{-5}$

[a]See Table 5.1-1
[b]Predominately I-131
[c]The smallest dose for which evacuation would be considered based of Environmental Protection Agency protective action guides discussed in Appendix 5A
[d]Ground level, non-buoyant release, stability class F, 1 m/s wind speed, calculations using USNRC MACCS 1.5.11.1 computer code
[e]Threshold for radiation-induced prodromal vomiting
[f]Threshold for radiation-induced thyroiditis

Table 5.1-4 Decontamination factors associated with various design features

Design Feature	Decontamination Factor
Containment Sprays	~10 for aerosols in 10-20 minutes versus ~10 for aerosols in 10 hours without sprays
Ice Condensers	1.2 to 20 while ice is present
Reactor building surrounding a BWR Mark I containment	1.1 to 80
Suppression pools and overlying water layers	2 to 4000 before re-evolution

Table 5.1-5 NUREG-1465 BWR releases into containments*

	Gap**	Early In-Vessel	Ex-Vessel	Late In-Vessel
Duration (Hours)	0.5	1.5	3.0	10.0
Noble Gases: Xe, Kr	0.05	0.95	0	0
Halogens: I, Br	0.05	0.25	0.30	0.01
Alkali Metals: Cs, Rb	0.05	0.20	0.35	0.01
Tellurium group: Te, Sb, Se	0	0.05	0.25	0.005
Barium, strontium: Ba, Sr	0	0.02	0.1	0
Noble Metals: Ru, Rh, Pd, Mo, Tc, Co	0	0.0025	0.0025	0
Lanthanides: La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am	0	0.0002	0.005	0
Cerium Group: Ce, Pu, Np	0	0.0005	0.005	0

* Values shown are fractions of core inventory.
** Gap release is 3 percent if long-term fuel cooling is maintained.

Table 5.1-6 NUREG-1465 PWR releases into containments*

	Gap**	Early In-Vessel	Ex-Vessel	Late In-Vessel
Duration (Hours)	0.5	1.3	2.0	10.0
Noble Gases: Xe, Kr	0.05	0.95	0	0
Halogens: I, Br	0.05	0.35	0.25	0.01
Alkali Metals: Cs, Rb	0.05	0.25	0.35	0.01
Tellurium group: Te, Sb, Se	0	0.05	0.25	0.005
Barium, strontium: Ba, Sr	0	0.02	0.1	0
Noble Metals: Ru, Rh, Pd, Mo, Tc, Co	0	0.0025	0.0025	0
Lanthanides: La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am	0	0.0002	0.005	0
Cerium Group: Ce, Pu, Np	0	0.0005	0.005	0

* Values shown are fractions of core inventory.
** Gap release is 3 percent if long-term fuel cooling is maintained.

Table 5.1-7 Estimated releases from Chernobyl-4 accident[14]

Noble Gases	100%	190±20	MCi
^{131}I	55±5%	45±5	MCi
^{137}Cs	33%±10%	2.3±0.7	MCi
^{90}Sr, ^{90}Y	4%	2.8±0.8	MCi
Fuel	3.5±0.5%		

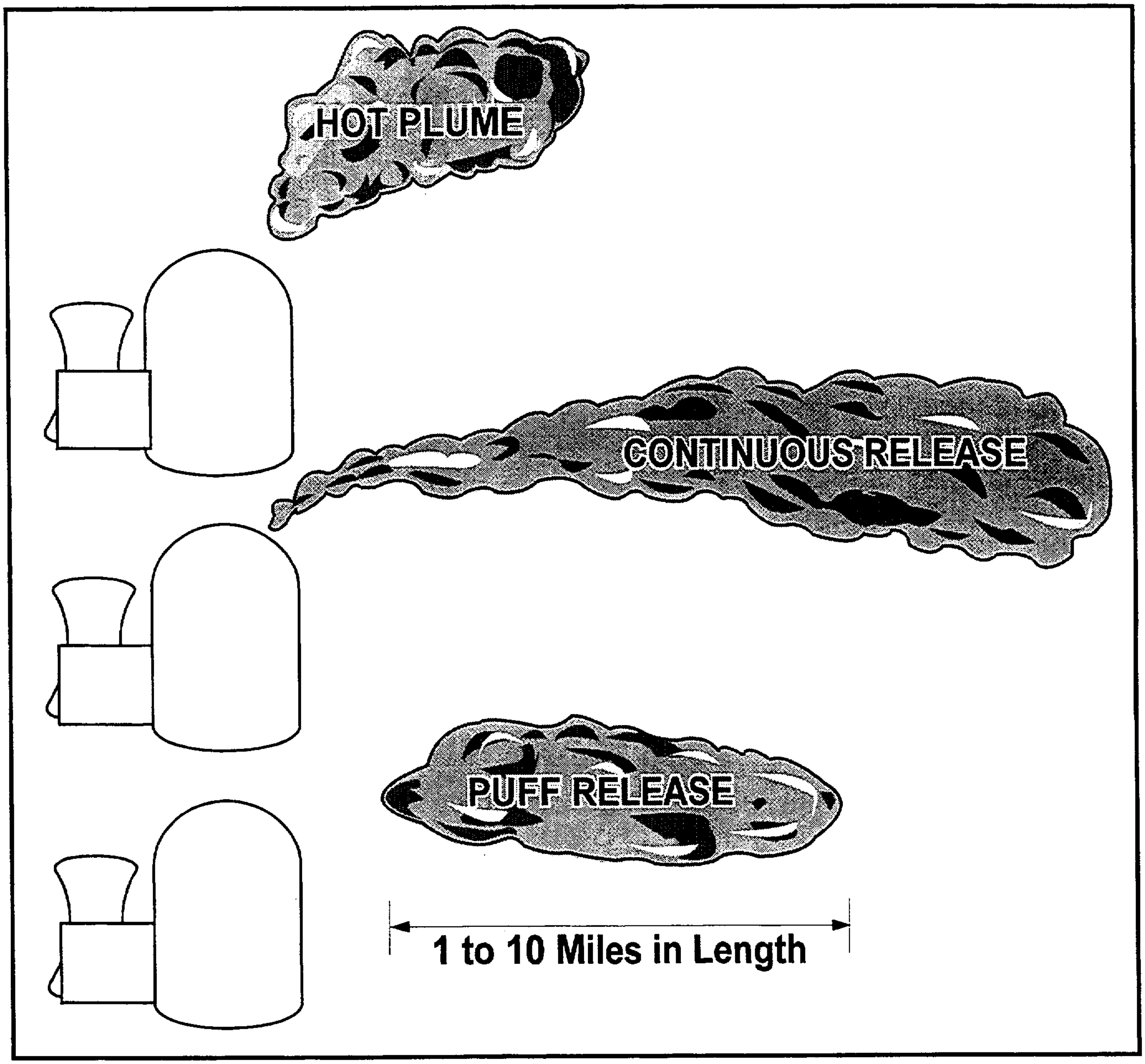


Figure 5.1-1 Examples of plume types

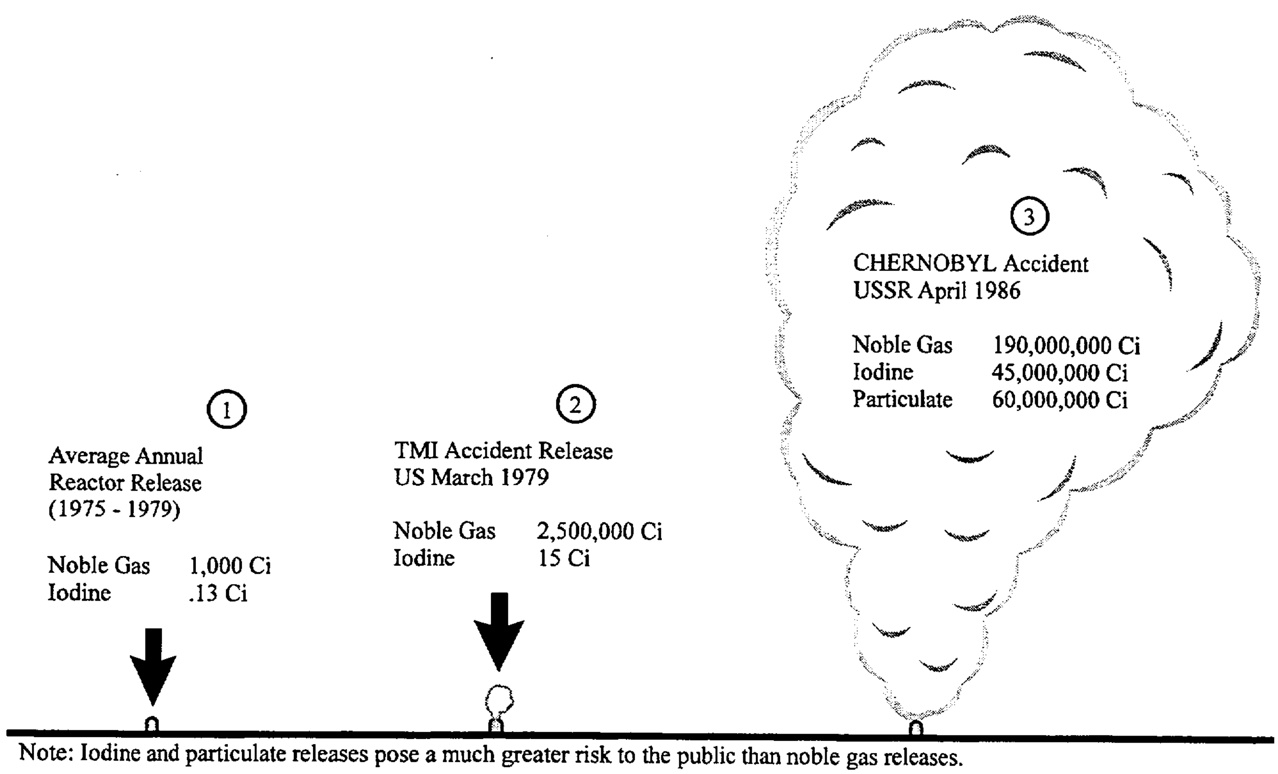


Figure 5.1-2 Putting radiation releases (curies-Ci) in perspective for the public

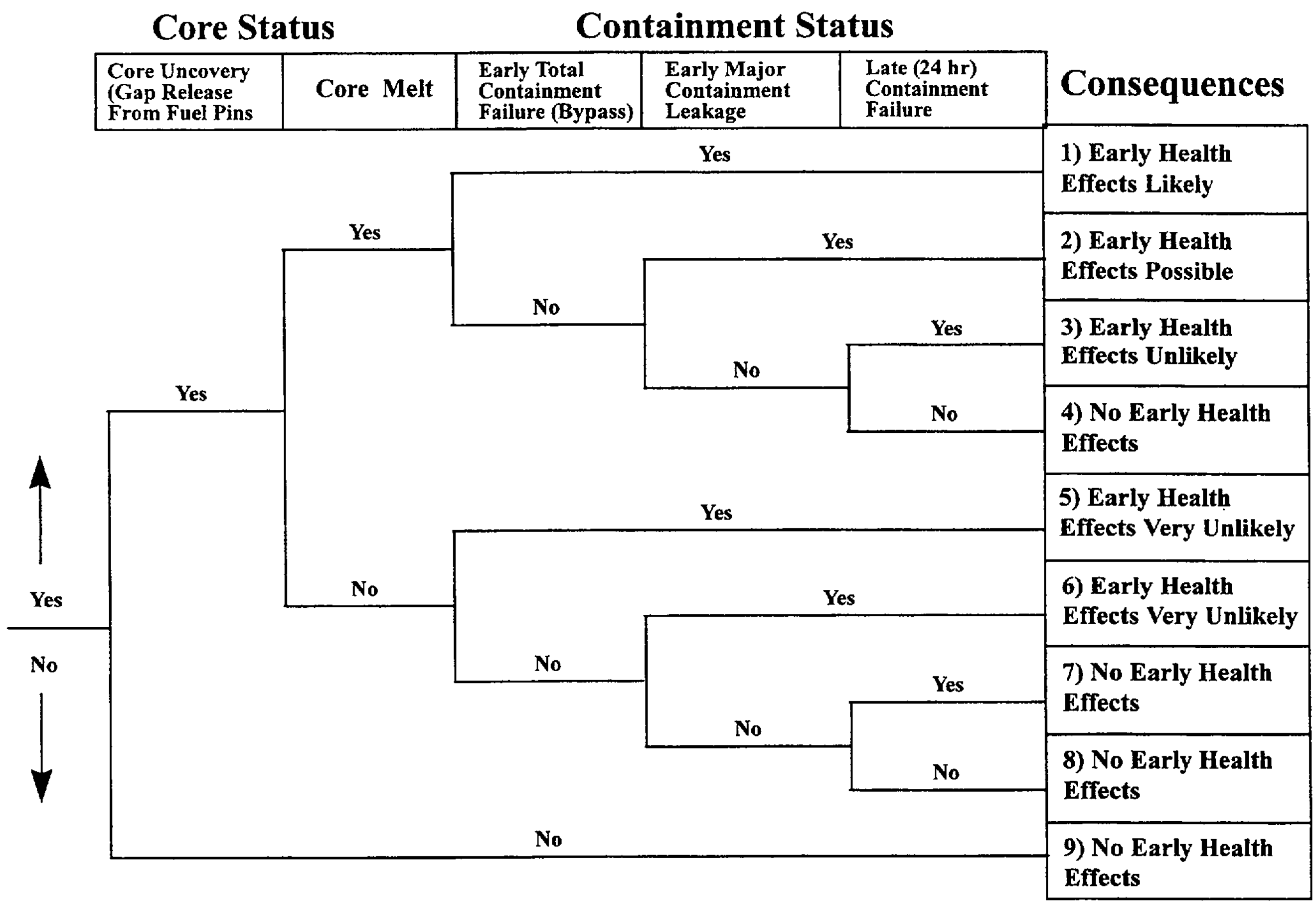


Figure 5.1-3 Event tree for severe accident consequences

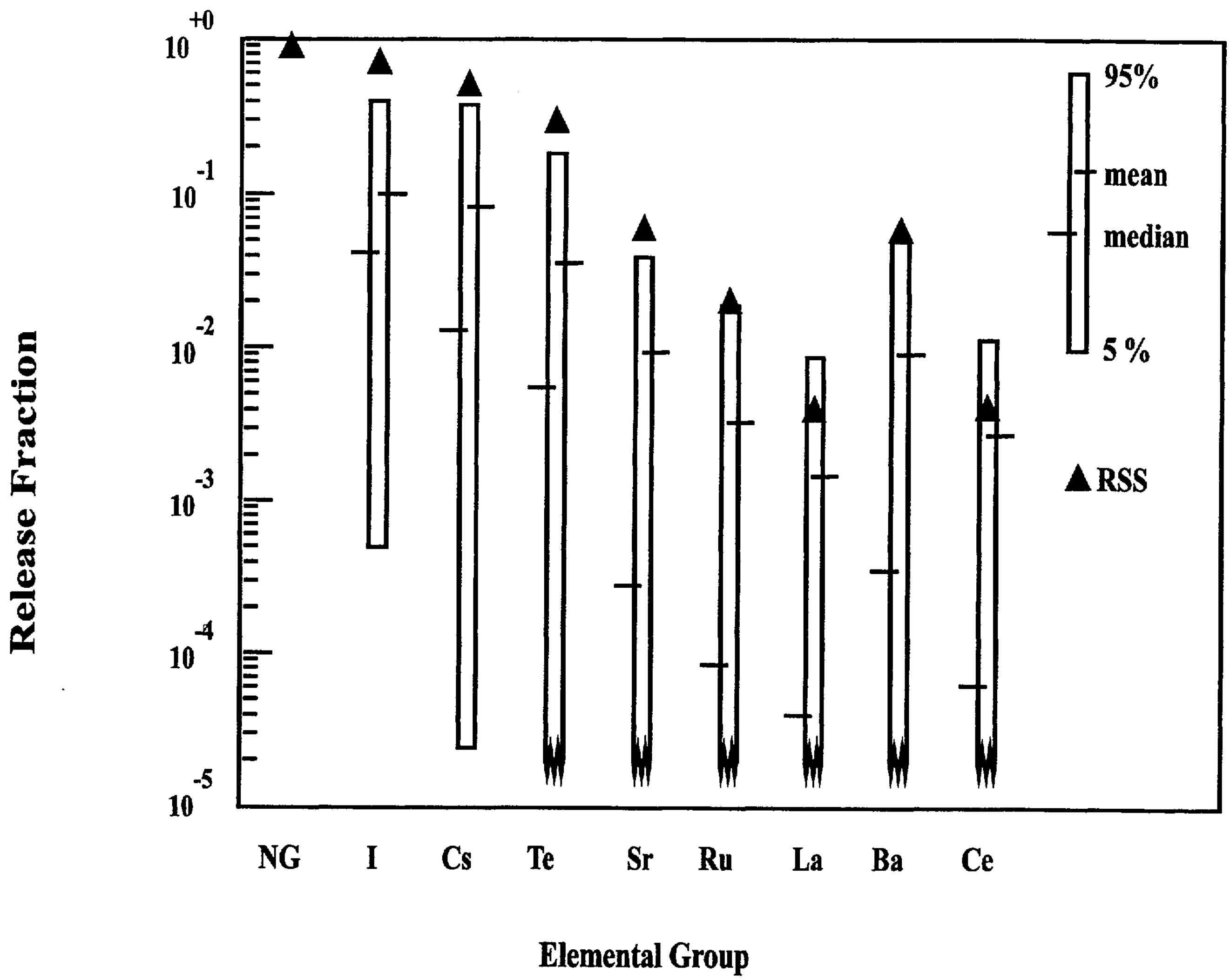


Figure 5.1-4 Comparison of NUREG-1150 source terms with Reactor Safety Study (Surry) bin PWR2

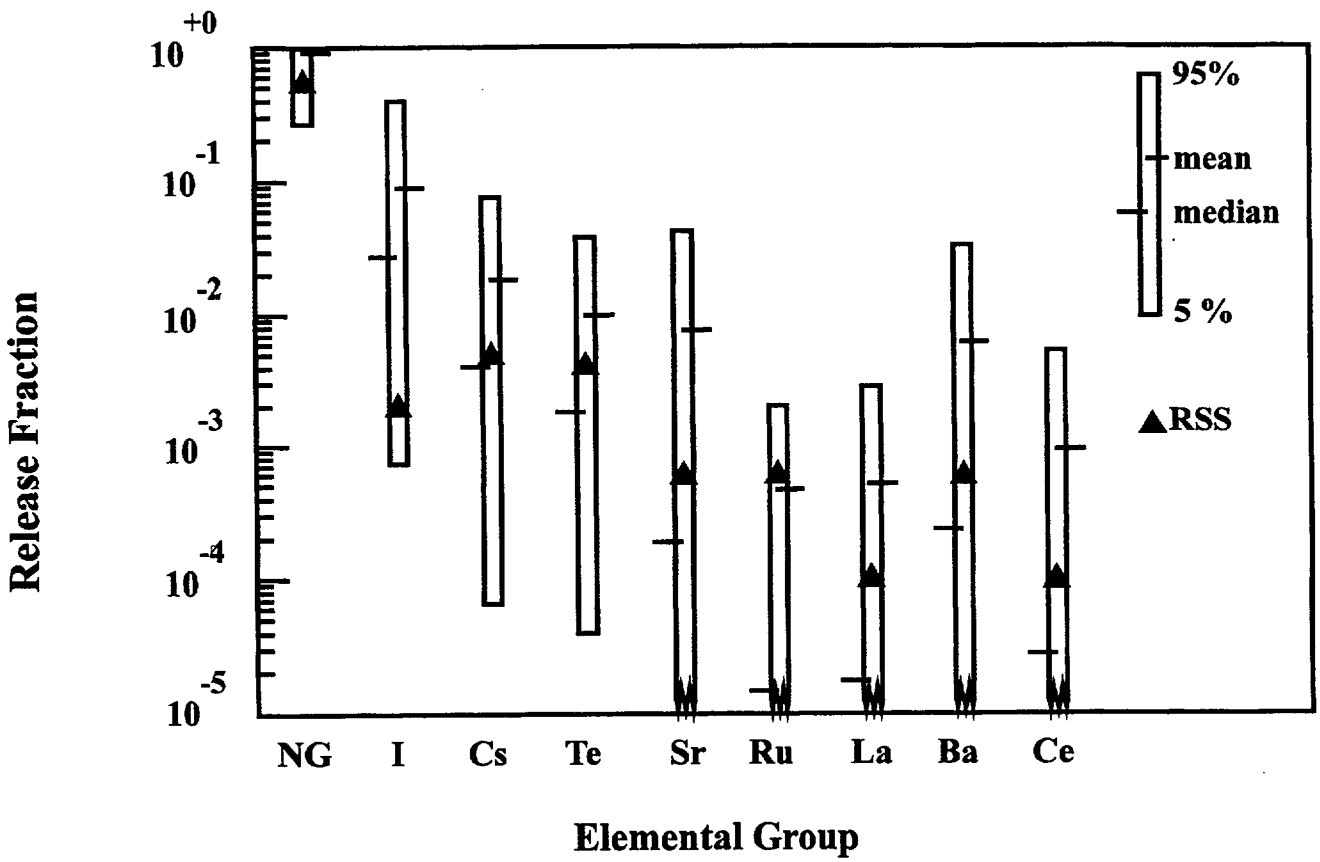


Figure 5.1-5 Comparison of NUREG-1150 source terms with Reactor Safety Study (Peach Bottom) bin BWR4

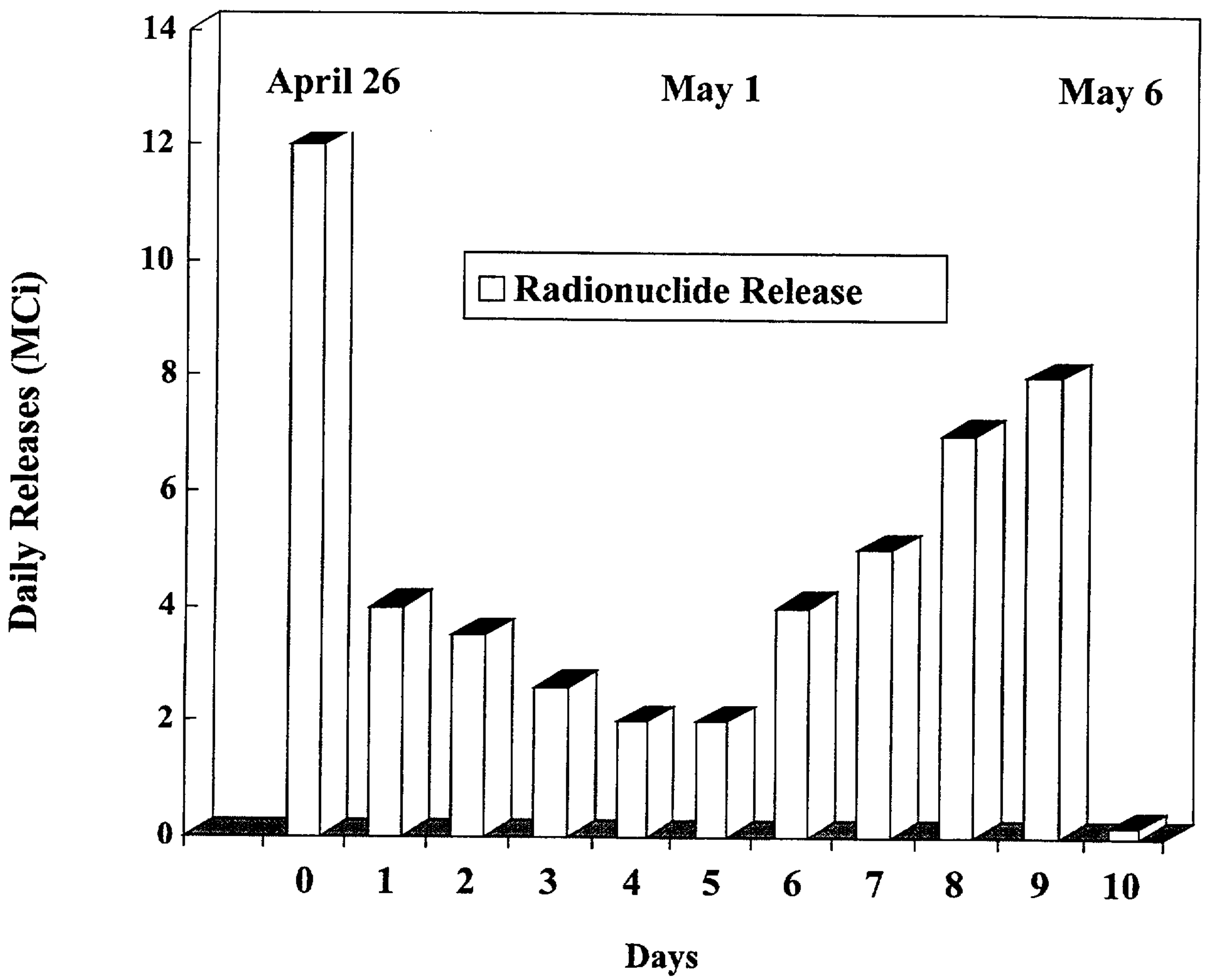


Figure 5.1-6 Release of radionuclides during the active stage of the Chernobyl accident[1]

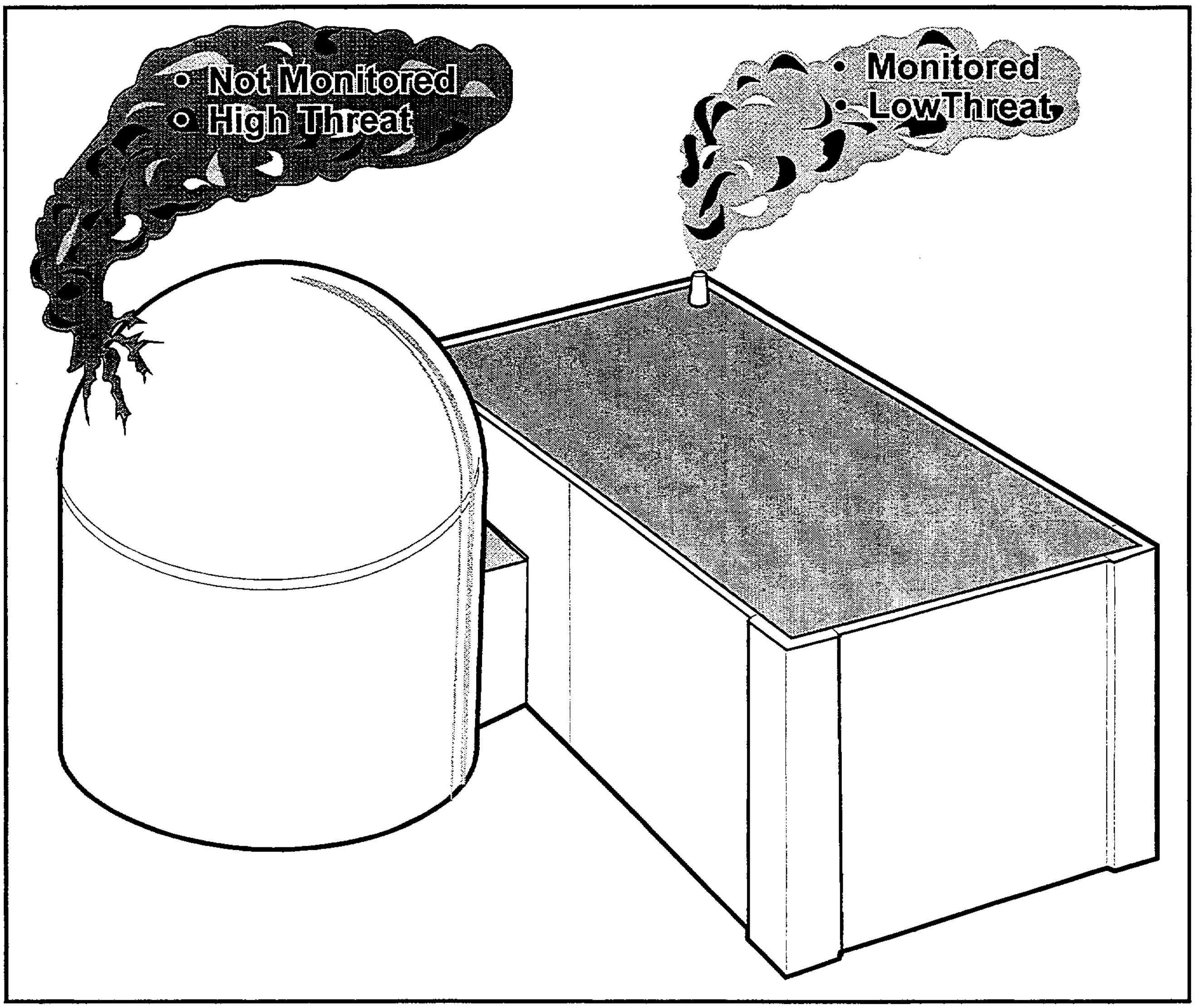


Figure 5.1-7 Types of release

References for Section 5.1

1. D. J. Alpert, et al., "Relative Importance of Individual Elements to Reactor Accident Consequences Assuming Equal Release Fractions," NUREG/CR-4467, SAND85-2575, March 1986.

2. T. J. McKenna, et al., "Pilot Program: NRC Severe Reactor Accident Incident Response Training Manual, Vol. 2, Severe Reactor Accident Overview," NUREG-1210, February 1987.

3. U.S. Nuclear Regulatory Commission, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors," Regulatory Guide 1.3, Revision 2, June 1974.

4. U.S. Nuclear Regulatory Commission, "Pressure Suppression Pool as a Fission Product Cleanup System," Standard Review Plan, Section 6.5.5, NUREG-0800, December 1988.

5. U.S. Nuclear Regulatory Commission, "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400, NUREG-75/014, December 1975.

6. U.S. Nuclear Regulatory Commission, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150, December 1990.

7. E.C. Beahm, C.F. Weber, and T.S. Kress, "Iodine Chemical Forms in LWR Severe Accidents," NUREG/CR-5732, ORNL/TM-11861, prepared for NRC by Oak Ridge National Laboratory, April 1992.

8. L. Soffer, S. B. Burson, C. M. Ferrell, R. Y. Lee, and J. N. Ridgely, "Accident Source Terms for Light-Water Nuclear Power Plants," NUREG-1465, February 1995.

9. J. J. DiNunno, et al., "Calculation of Distance Factors for Power and Test Reactor Sites," Technical Information Document (TID)-14844, U.S. Atomic Energy Commission, 1962.

10. U.S Nuclear Regulatory Commission, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors," Regulatory Guide 1.4, Revision 2, June 1974.

11. H. P. Nourbakhsh, "Estimates of Radionuclide Release Characteristics into Containment Under Severe Accidents," NUREG/CR-5747 (BNL-NUREG-52289, Brookhaven National Laboratory, November 1993.

12. USSR State Committee on the Utilization of Atomic Energy. The accident at Chernobyl and its consequences. Summary report on the post-accident review meeting on the Chernobyl accident. Vienna: International Atomic Energy Agency, IAEA Safety Series N 75-INSAG-1; 1986.

13. Yu. P. Buzulukov and Yu. L. Dobrynin, "Release of Radionuclides During the Chernobyl Accident," in *The Chernobyl Papers, Volume I, Doses to the Soviet Population and Early Health Effects Studies*, Steven

E. Merwin and Michail I. Balonav editors, Research Enterprises, Richland, Washington, 1993.

14. S. T. Belyaev, A. A. Borovoy, and V. F. Demin, et al., "The Chernobyl Source Term, In: *Proceedings of a seminar on comparative assessment of the environmental impact of radionuclides released during three major nuclear accidents,* Kyshtym, Windscale, Chernobyl. Luxembourg: EUR; Report EUR 13574; 1991: 71-91.