Containment Loading Resulting from Severe Accident Analyses

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Abstract

Analyses of severe accident loading of principal PWR and BWR containment types are summarized. The analyses assume that ultimate loading of a containment will result in general yield or penetration failure. The relative resistance to such accidents of the large dry containments is substantiated for PWRs. And the relative resistance of the BWR Mark III containments to suppression pool bypass compared to the Mark I containment is developed.

1. Introduction

Nuclear power plants are provided with containments to prevent release of radionuclides to the environment. The design basis accident is the double ended primary coolant pipe rupture and the consequent release of primary coolant. The resultant structural design provides a containment strong enough, based on ASME Code Design, to withstand the pipe rupture accident over the lifetime of the plant with due allowance for construction errors, material variations, corrosion and certain degradations. Thus the following analyses of beyond the design basis accidents frequently result in containment loads beyond the containment design basis but attempt to determine nominal ultimate loads which could be sustained only once or sustained for a period of time that delays and reduces the radioactive source at release. The following analyses are applied to pressurized water reactors (PWR) and boiling water reactors (BWR). The containment for PWRs are in three categories: dry atmospheric, subatomspheric and ice condenser.

2. Pressurized Water Reactor Containment

Dry atmospheric containments are usually steel-lined cylindrical hemispherical dome concrete structures that are relatively open to permit rapid mixing of gases throughout their free volumes (up to 3,600,00 ft^3). The initial pressure during reactor operation is near atmospheric. All dry atmospheric PWR containments have containment sprays for removing heat and radionuclides from their atmospheres. They also have fan coolers for control of the atmosphere temperature. The overflow of water into the reactor cavity from accumulation of water in the sump and on the containment floor varies with the specific design.

Subatmospheric containments are similar steel-lined cylindrical hemispherical domed concrete structures. Subatmospheric containments are smaller than dry atmospheric containments (1,800,000 ft^3) and the initial pressure during operation is about 10 psia. Subatmospheric containments have sprays for removing heat and radionuclides from the atmosphere.

Ice-condenser containments are the smallest with total free volumes of about 1,200,000 ft^3. There are containment sprays in the upper compartments and about 2,450,000 lbs. of ice to remove heat and radionuclides from the containment atmosphere during an accident. Deliberate ignition systems provide hydrogen control. Water may overflow into the reactor cavity.

A loss of coolant accident (LOCA) for a large dry containment will be assumed to be compounded by emergency core cooling (ECC) failure and failure of all containment sprays and fan coolers. Water is injected from the refueling water storage tank and the solid line in Figure 1 shows a stable containment pressure condition for more than 10 hours. If an arbitrary and limited quantity of water from the RWST is injected, that necessary to saturate containment atmosphere at failure pressure, the minimum best estimate time-to-failure of 570 minutes is obtained. Similarly compounded failures for a small break LOCA result in the pressure-time characteristics of Figure 2, with the "base" case showing relative stability with containment spray only and requiring more than 10 hours to reach failure pressure without even containment spray availability.

In the event that hot core debris is ejected from the pressure vessel and quenched in water present in the reactor cavity, adiabatic conditions may be assumed for rapid, less than one minute, quenches. Thus, the available core debris thermal energy and the resultant amount of water vaporized determine adiabatic pressure rises of 25 to 30 psi for large dry containments. If the limited reactor coolant remaining after core uncovery is assumed to flash to containment and is added to the above quench steam loading an additional 15 psi is possible. These two combined effects are not sufficient to challenge the integrity of large dry or subatmospheric containments.

The potential for loading the containment by hydrogen combustion is summarized in Figure 3. Steam inerting results when the steam mol fraction exceeds the range of 0.5 to 0.6. Preburn pressure parameters of 20, 30 and 40 psia are provided. Utilizing 100 percent clad oxidation and the maximum possible hydrogen release, a significant margin exists to containment failure. Thus the integrity of large dry containments is not threatened prior to breaching of the primary system. A similar result is obtained for subatmospheric containments.

Experimental tests have been conducted at 1/20 and 1/10 linear scales of high pressure ejection of core debris onto both dry and wet reactor cavity simulations. Interaction of such high pressure ejections with water may cause sufficient fragmentation to create rapid direct heating of the containment atmosphere to cause 2.5 to 3.0 times as large a pressure rise as produced by steam from quenching 100 percent core debris. Thus one half of the energy to steam quench and one half to direct heating could cause a 56 psi pressure rise in addition to the pre breach containment pressure. Transmittal of so much of the melt energy by direct heating must be considered unlikely but the 1/10 scale high pressure ejection test is being modified to provide a quantitative measurement of direct heating.

A dominant sequence severe accident is a transient involving the loss of main and auxiliary feedwater and AC power. Figure 4 shows steam generator dryout at about 90 min. and containment pressure increase over the next 40 min. as the safety relief valves discharge into containment. As the core melts (140 to 180 min.) the containment pressure decreases sightly as some steam condenses while the core absorbs the decay heat. Failure of the bottom head and evaporation of the reactor cavity liquid and formation of noncondensable gases cause the pressure spike. Heat transfers to passive heat sinks decrease the pressure, but gradual build-up of pressure occurs over a period of 10 hours to several days. Reestablishing a heat sink (sprays, fan coolers etc.) can readily result in a managed accident.

The calculated pressure response for an ice-condenser containment for TMLB' accident scenario is shown in Figure 5. For this scenario with a total loss of AC power neither the fans nor sprays are operable. Also, because the ignitors will not operate without AC power, the hydrogen ignition threshold was arbitrarily set to 12 percent. Ignition may be a random event in such a case, depending on electrostatic discharge, an operable piece of electrical equipment, or the discharge of the molten core debris following vessel breach. The large pressure spike is due to a burn that begins in the dome with 12 percent hydrogen concentration. Ignition is in the dome because the upper plenum was rendered inert due to oxygen depletion by a previous burn. The pressure rise was high because of a high baseline pressure, 12 percent ignition limits, and the presence of large quantities of hydrogen in the upper plenum which burn after free oxygen from the dome is pushed back into the upper plenum.

3. Boiling Water Reactor Containments

Boiling water reactors (BWR) have three types of containments; Mark I, II and III. Figure 6 shows the Mark I containment with the relatively small drywell (159,000 ft^3) connected to a wetwell pressure suppression pool. A large standby gas treatment system maintains a small negative pressure in the secondary containment and discharges through HEPA and charcoal filters via the plant stack. The design basis LOCA discharge is condensed in the pressure suppression pool and the pool is cooled.

BWR severe accidents involve one of two categories: 1) inadequate cooling of the pressure suppression pool and steady evaporative steaming fails the relatively small containment by overpressurization or 2) loss of reactor vessel injection, boiloff of the vessel water inventory with consequent core uncovery, core melt, melt-thorugh of the vessel bottom head and melt deposition onto the drywell floor.

For Category 1 accidents with catastrophic containment failure, one operable injection system provides successful accident management with continued core cooling. If all injection is lost at the time of sudden containment failure, the core is uncovered about one hour later later as a result of boil-off and the accident proceeds to melt-through. If the containment fails by penetration leakage sufficient to stabilize the containment pressure, the effluent from the drywell is exhausted to the atmosphere by the standby gas treatment system from the secondary containment. If one or more water injection system continues to operate, stable management of the accident is assured.

For Category 2 accidents, the containment is intact at the time the vessel bottom head fails. The melt would quickly fail the 2.3 in steel drywell liner and probably the secondary containment. In the event that the melt did not reach the steel drywell liner, the pressure suppression pool would have no effect on the large quantities of noncondensable gases. These include hydrogen from the metal-water reaction and gases from the corium-concrete reaction on the dry well floor. The increasing gas pressure would eventually fail the containment. The structural strength of the containment could be reduced, particularly with respect to penetrations, since containment temperatures in the range of 800 to 1500°F are calculated depending on various assumptions. The Mark II containment is functionally similar to Mark I and has similar serious limitations for severe accidents.

The Mark III containment is presented in Figure 7. The drywell is almost twice as large as in Mark I and is enclosed by the primary containment providing five times the free volume of the Mark I containment. The primary containment is not inerted and may be occupied during operation. A large filtered recirculation system is provided with a smaller standby gas treatment system. All discharge from the drywell must pass through the pressure suppression pool.

The Category 1 class of severe accidents resulting in evaporative steaming of the suppression pool and eventual containment failure is much less critical for Mark III. The structural ultimate failure pressure has been calculated to be about 100 psig, about six times the design pressure. Many hours and perhaps days would be available before containment failure. Thus an interruption of suppression pool cooling could be restored.

For Category 2 accidents, hydrogen generated within the core as cooling is lost enters the pressure suppression pool. Hydrogen flames would accelerate the failure of the containment but the chief concern is more rapid failure of penetrations. This concern is currently being evaluated.

After core melt and bottom head failure the molten corium attacks the concrete pedestal supporting the reactor vessel. Improved corium-concrete interaction codes and experimental results are currently being employed to make more precise computations of the time required for radial and vertical attack on the concrete and ultimate failure of the pedestal. The gases generated in the corium-concrete reaction pass into the outer containment and both heat and pressurize the outer containment. However the increased size of the Mark III containment provides a much larger time to failure than other BWR designs and all such effluent must continue to pass through the pressure suppression pool.

4. References

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5. K. D. Bergeron, D. C. Williams and T. Zimmerman, "CONTAIN Calculations of Containment Loadings of Dry PWR's, "Second Workshop on Containment Integrity, Arlington, VA., June, 1984.

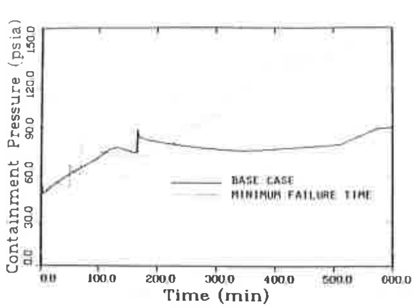

FIGURE 1. CONTAINMENT LOADING; DOUBLE-ENDED COLD LEG BREAK

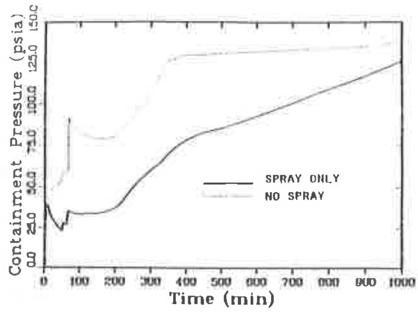

FIGURE 2. CONTAINMENT LOADING; SMALL BREAK LOCA

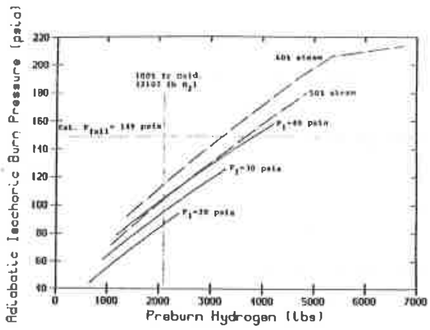

FIGURE 3. ADIABATIC BURN PRESSURES FOR COMPLETE DEFLAGRATIONS IGNITED AT SATURATED CONDITIONS.

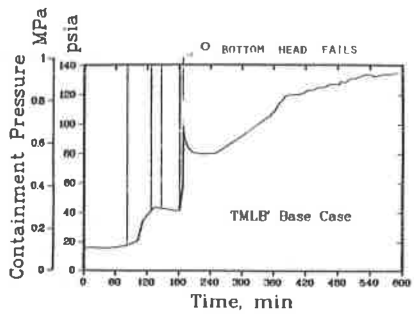

FIGURE 4. CONTAINMENT BUILDING INTERNAL PRESSURE VERSUS TIME FOR TMLB'

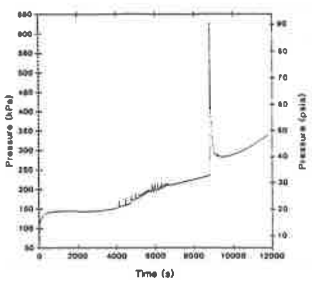

Figure 5. Pressure Response in Dome

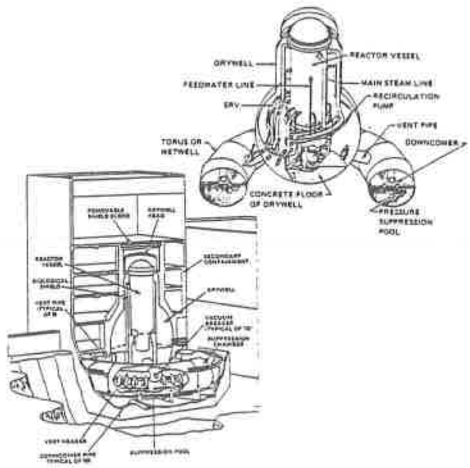

FIGURE 6. THE BWR MARK I CONTAINMENT SYSTEM.

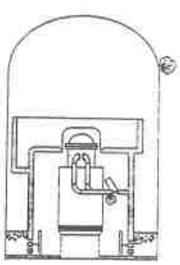

FIGURE 7. SCHEMATIC OF MARK III CONTAINMENT WITH LARGE LOCA IN DRYWELL AND RELEASE THROUGH THE PRESSURE SUPPRESSION POOL