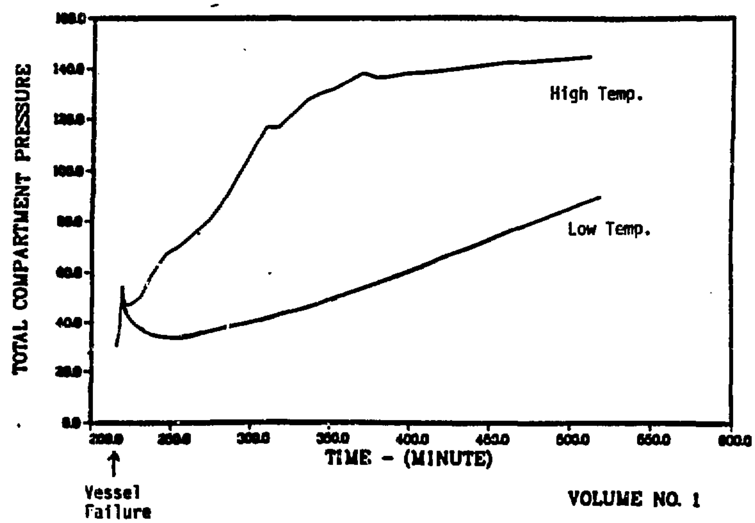

Figure 3 Calculated pressure response with and without leakage in the drywell for a Class I loss-of-make-up accident in a BWR Mark I containment for an assumed high temperature debris at vessel failure based on CORCON gas generation rates.

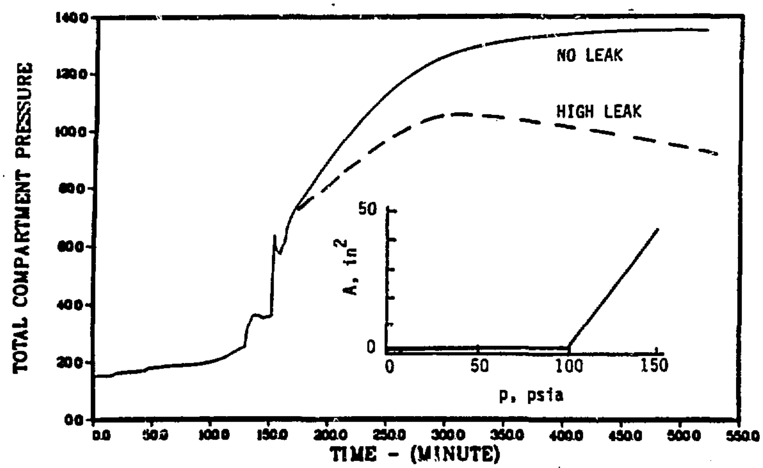

Figure 4 Calculated pressure response with and without leakage in the drywell for a Class I loss-of-make-up accident in a BWR Mark II containment for Case 5.

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CONTAINMENT PERFORMANCE FOR CORE MELT ACCIDENTS IN BWRs WITH MARK I AND MARK II CONTAINMENTS*

K. R. Perkins, J. W. Yang, G. A. Greene,
W. T. Pratt, and C. Hofmayer
Department of Nuclear Energy
Brookhaven National Laboratory
Upton, New York 11973

MASTER

ABSTRACT

Most previous risk assessment studies have assumed catastrophic failure of containments for severe accidents which are predicted to exceed the containment yield stress. This investigation analyzes the progression of a severe accident in order to develop realistic containment temperature and pressure loading, utilizes models for containment leakage estimates for the various loading histories, and assesses the expected failure modes and timing of releases for core melt accidents in Boiling Water Reactors (BWRs) with Mark I and Mark II containments. The results of the investigation indicate that leakage through the seal on the drywell head may be sufficient to prevent catastrophic failure of the containments for a wide range of hypothetical core melt scenarios. In addition, the investigation has indicated the potential for a previously unidentified failure mode (containment liner melt-through) for Mark I containments in which a large fraction of the core is released from the vessel in a molten state.

INTRODUCTION

The Reactor Safety Study (1) analysis of the Mark I BWR considered the γ-mode of containment failure as the dominant overpressure failure mode. The γ-mode is defined as overpressure failure of the drywell liner resulting in release of fission products and aerosols directly into the reactor building. The failure pressure for this event has been estimated (1) at 1.2 MPa for Peach Bottom. The nearly identical containment at Browns Ferry was estimated (2) to fail at .91 MPa using a somewhat more conservative failure criteria.

*Work performed under the auspices of the U.S. Nuclear Regulatory Commission.

However, recent results from the Severe Accident Sequence Assessment (SASA) program analyses (3,4,5) of a Mark I BWR have indicated that high temperatures in the drywell during ex-vessel core/concrete interactions may result in containment failure due to seal degradation prior to gross failure due to overpressurization. Virtually all risk assessments performed to date have used a threshold model, which defines a threshold pressure, with some associated uncertainty, at which the containment will suffer a loss of holding capability resulting in significant release of fission product inventory. If the containment pressure loading is calculated to be below the threshold pressure, the containment is considered to be intact and the off-site consequences are therefore predicted to be quite low. At the request of the NRC Severe Accident Research Plan (SARP) Senior Review Group, a Containment Performance Working Group (CPWG) was established to develop containment leakage models for use in severe accident source term work. These leakage models will serve to help quantify leakage areas as a function of containment pressure and temperature loading, for various containment types. The leakage models can then be incorporated into existing containment computer codes to permit an assessment of the effects of leakage on containment behavior for severe accidents. Specifically, containment leakage as a function of pressure and temperature loading and the impact of containment pressure relief (due to leakage) on the mode and timing of containment failure can be estimated. These results can then be used as input to a radiological consequence analysis. Preliminary results of the CPWG have been incorporated into a draft report which presents approximate leakage models for six containment types (6).

The Containment Loads Working Group (CLWG), which was also formed by the NRC, has attempted to define potential containment loads during core meltdown accidents for a range of containment designs. This paper utilizes the work performed by BNL in support of the CLWG for application to BWRs with Mark I and Mark II containments in order to estimate containment loading for the containment performance calculations. The CLWG used the concept of "standard problems" to focus on important issues regarding containment loads. After considering various possible failure modes, the CLWG concluded that the issue of temperature loading is the major concern of the BWR Mark I and Mark II containments.

The accident sequence defined as the standard problem was a transient event with a failure of all coolant make-up (a TQUV sequence using WASH-1400 Nomenclature). For the TQUV event, it is assumed that all reactor vessel injection capability is lost at the time of a reactor trip from 100% power. Because of mass loss out the safety relief valves (SRV's) and the lack of coolant injection, the core eventually becomes uncovered. When the reactor vessel bottom head fails, the corium falls onto the dry concrete floor of the drywell and the corium/concrete reaction begins. Steam and noncondensable gases are released from the concrete. The previously unoxidized zirconium will react with the steam and CO_2 released from concrete decomposition. The drywell pressure and temperature will increase beyond the design values.

Our investigation has indicated that another mode of drywell failure must be considered in addition to the gross overpressure failure and the leak-before-failure modes. This additional mode of failure is local ablation of the steel drywell liner due to contact with the molten corium. Since pathways through the obstructions on the drywell floor are available, molten core debris can flow outward from the pedestal region and contact the drywell liner (see Figure 1). As long as the corium is at a temperature greater than the steel melting temperature, it will present a threat to the containment integrity due to local melt-through. Should this occur in a Mark I containment, a large flow path to the reactor building and standby gas treatment system, bypassing the wetwell, will be available for blowdown of the high temperature concrete decomposition gases from the ex-vessel core/concrete interaction, aerosols, and fission products. Although the gap between the drywell liner and the concrete may be filled with fiberglass and polyester foam (see Figure 1), at high temperatures and pressures they cannot be relied on to provide significant resistance to the release once the liner has failed.

For the Mark II containment, there is still a possibility of the core debris contacting and melting the liner. However, the liner is immediately adjacent to the concrete wall and thus no significant leakage could pass through the hole in the liner until the concrete also failed.

ANALYTICAL METHODS

The present results use the MARCH 1.1B (10) for the Mark I calculations, MARCH 1.1 (11) for the Mark II calculations, and CORCON-MOD 1 (12) computer codes. The MARCH 1.1B (10) computer code developed at ORNL has been used for application to severe accidents in BWR Mark I plants. MARCH utilizes the INTER (13) code as a subroutine to model corium/concrete interactions. Murfin (13) stressed that the model represented a preliminary qualitative description of the major core/concrete interaction phenomena and he indicated that the applicability to interactions with large oxide fractions was questionable. An improved core/concrete interaction model, CORCON-MOD 1 (12) has been developed by Sandia National Laboratories. For the BNL calculations, the initial conditions for core/concrete interactions obtained from the sample problem were input to the CORCON-MOD 1 code. The output from CORCON-MOD 1 involving water, hydrogen, carbon dioxide, and carbon monoxide generation was then input to the MARCH analyses, which bypassed the INTER subroutine. Details of the core/concrete reaction results are provided in the Containment Loading report (7).

For the separate effects calculation in which the molten debris is assumed to contact the liner wall, the heat transfer coefficient from CORCON was then input into the calculational procedures for the transient heat-up of the steel liner and the steel liner ablation calculations. The heat transfer from the molten corium to the steel liner was modeled as one-dimensional transient convective heat transfer with sensible and latent heat transfer. The transient heat-up of the liner from its initial temperature to the steel melting temperature was calculated as

$$(\rho c)_{steel} V \frac{dT_{steel}}{dt} = h_i (T_i - T_{steel}) A$$

subject to the initial condition

$$T_{steel} (t=0) = T_0 = 300 \text{ K}$$

where ρ is the steel density, T_i is the corium temperature, h_i is the heat transfer coefficient from the corium to the wall, c is the specific heat, V is the liner volume, and A is the contact area of the liner with the molten core debris. Note that V/A is the liner thickness, δ.

Once the liner is calculated to have heated to its melting temperature of 1750 K, the rate of melting of the steel liner is calculated until the calculational procedure is terminated. The melt rate of the liner is calculated as follows:

$$\rho_{steel} h_{fs,steel} \frac{d\delta}{dt} = h_i(T_i - T_{ablate})$$

subject to the initial condition

$$\delta(t = t_0) = 3 \text{ cm}$$

where h_{fs} is the latent heat of the steel, T_{ablate} is the steel ablation temperature, and t_0 is the time at the start of the ablation calculation.

The calculation proceeds until one of three criteria are satisfied. First, the calculation is terminated when the thickness of steel ablated exceeds the initial liner thickness. This time, t_{ablate}, indicates the containment failure time at which point fission products and aerosols would flow into the gap between the liner and shield wall, eventually finding their way into the reactor building. The second criterion which will terminate the calculation is if the downward erosion depth into the concrete exceeds the depth of the corium against the steel liner. Once the erosion depth exceeds the corium pool depth, it is assumed that contact of the corium with the steel is ended, and the threat to the liner is over. If the liner is not penetrated at this time, it is not estimated to fail by melt-through. The third criterion for termination of the calculation is if the calculated corium-steel interfacial temperature falls below the steel melting temperature. Once this occurs, melting of the liner ends, and failure by melt-through is avoided.

RESULTS FOR THE MARK I CONTAINMENT

For the TQUV sequence, the modes and timing of containment failure are intimately related to the temperature and quantity of corium exiting the primary system. There is a large uncertainty as to the condition and location of the core debris after vessel failure, but for the purposes of this investigation, it is assumed that a large fraction of the fuel (80%) along with all the zirconium and most of the lower head steel (63,000 Kg) is distributed on the drywell floor.

If the core debris is retained within the Mark I pedestal area, the debris pool would be 85 cm deep. With gas fluidization (bubbling) from corium/concrete interactions (CCI), the pool depth will be even greater. It is clear that such a deep pool will remain molten and could spread through the two pedestal access doors into the ex-pedestal (annular) space. An even spreading over the whole available area would produce a pool 22 cm deep (collapsed level). This is still a rather deep layer, but based on the scoping estimates of heat losses for the Mark I design, it appears that spreading over the entire drywell floor is unlikely. The base case therefore assumes that the corium will cover 50 percent of the drywell floor. The present results use MARCH 1.1B (10) for in-vessel calculations and CORCON-MOD 1 (12) for core/concrete interaction calculations. The calculations neglect the effect of the transient spreading of the corium. A sensitivity study of various input parameters is provided in the Containment Loads report (7), and comparisons to other analytical methods are given in the Standard Problem consensus report (14). For the present results, we consider only two extreme cases to illustrate the effect of containment leakage estimates on the containment response.

There are two major variations of this base case: A high temperature case (at the fuel melting point) and a low temperature case (at the melting point of steel). It is assumed that for the low temperature case, the core debris could not flow and would remain confined within the pedestal wall. Conversely, the high temperature case is expected to spread rapidly into the annular space surrounding the pedestal. These two cases, illustrated in Figure 2, produce dramatically different results, but most of this difference is due to the debris temperature difference and not to geometric differences.

Note that for the high temperature case, the ultimate capacity of the containment (.91 MPa or 132 psia) is reached within 2 hours after vessel failure, although no failure is modeled in this calculation. However, our investigation also indicates that the seal on the upper head could begin to leak at .69 MPa. This leakage area is modeled to increase linearly with pressure up to 90 cm^2 (14 in^2) at .91 MPa. When this leakage model is included in the calculation, even the limiting high temperature debris case is prevented from reaching the ultimate capacity of the containment as shown in Figure 3. The leakage itself will release fission products to the reactor building, but catastrophic overpressure failure is averted and the release may be mitigated by the reactor building standby gas treatment system (SGTS), if it is available. Note that the low temperature debris case (also shown in Figure 2) does not reach .69 MPa during the five-hour simulation. Even for the low temperature case, the leakage initiation pressure (.69 MPa) will be reached, eventually, but at these low noncondensable gas production rates, the leakage model will keep the pressure near .70 MPa.

The previous calculations assume that the spreading debris will not reach the drywell wall. The results of the calculations that were performed for the local liner failure problem assuming that the debris does reach in the wall are indicated in

Table 1. It is clear from the table that in most cases studied, the steel liner was calculated to fail by ablation very rapidly, in one case as rapidly as 3-1/2 minutes after contact with the molten core debris. In two of the eight cases studied, it was calculated that the liner would not fail by local melt-through at all. This occurred for the low temperature corium cases (1775 K and 1900 K) on the basaltic-type concrete. Due to the low ablation temperature assumed for the basaltic concrete cases (~1450 K), the corium temperature dropped very rapidly upon contact with the concrete since the basaltic concrete acts as a rapidly ablating, low temperature heat sink. As a result, the corium debris temperature fell very rapidly below the steel ablation temperature, 1775 K, ending the ablation of the liner early. If at this time the liner had not been calculated to have been penetrated, it was assumed that no further threat by local melt-through will occur and the calculation was terminated. The only basalt concrete cases in which the drywell liner failed by melt-through were for the high corium temperature cases of 2550 K. For these two cases, it took only 5-1/2 minutes to ablate the liner and fail the drywell.

For all the limestone concrete cases studied, the steel drywell liner was calculated to melt through rapidly. The time to melt through varied from 3-1/2 minutes for the 2550 K corium cases to 45 minutes for the 1775 K corium case. Once again as for the 2550 K basalt cases, varying the percent of the core from 80% to 60% had little impact on the failure times. Since the ablation temperature of the limestone-type concrete was assumed to be 1750 K, the same as the melting temperature of the steel liner, the debris remained slightly above this temperature long enough to ensure the eventual melt-through failure of the drywell liner, even for the case that the debris initial temperature was 1775 K.

RESULTS FOR THE MARK II CONTAINMENT

Since there is considerable uncertainty as to the debris conditions at the time of reactor vessel failure, a sensitivity study was conducted from the basic TQUV sequence outlined in Section 3. Comparisons of peak pressures and temperatures are included in Table 2. The design pressure for the Mark II containment is .48 MPa, the design temperature is 444 K for the drywell chamber and 378 K for the wetwell chamber. For the eight cases involved in this study, the predicted atmospheric temperatures in both drywell and wetwell chambers exceed the design temperatures. But the concrete and steel liner temperatures in both chambers are lower than the corresponding design values. The predicted peak pressures are also higher than the design pressure. For three cases (Cases 5, 5a, and 7a in Table 1) the peak pressure is above 0.9 MPa which is close to (but does not exceed) the estimated containment failure pressure (1.0 MPa).

Note that two of the Mark II cases (5c and 5d) assume that some of the core debris flows through the downcomers and is quenched by the pool.

The high temperature limestone case (Case 5) is chosen to illustrate the effect of the predicted leakage for the Mark II containment. The pressure response for this case with and without leakage is shown in Figure 4. For the no leakage case, the predicted pressure rapidly approaches the ultimate capacity of the containment (1.0 MPa). But for the estimated leakage area (6), the compartment pressure is predicted to remain well below the ultimate capacity.

The sensitivity study results for the Mark II containment indicate that the containment pressure will not increase enough to cause catastrophic failure. However, the combination of the pressure and temperature loading is predicted to cause significant leakage from the containment. Thus, seal leakage rather than catastrophic failure may be the dominant failure mode for the TQUV sequence. Note that the leakage path is predicted (6) to be through the drywell head into the refueling area. For some Mark II containments, the releases in the refueling area could be mitigated by availability of the SGTS.

For the Mark II containment, the drywell liner is fastened directly to the concrete wall and no significant leakage is expected even if the debris spreads across the drywell floor and melts the liner.

CONCLUSIONS

For a TQUV accident sequence, the mode and timing of containment failure is closely related to the temperature and quantity of corium exiting the primary system. There is a large uncertainty as to the condition and location of the core debris after vessel failure. However, even if we assume that the corium is at a high temperature with maximum non-condensable gas generation from the decomposing concrete, the containment performance results (6) indicate that there are potential sources of leakage which may be sufficient to prevent catastrophic overpressure failure of the Mark I containment. However, the results also indicate that if the debris spreads all the way to the drywell wall, the debris could cause local liner penetration via melting. This melt-through has the potential to cause containment failure prior to reaching overpressure failure point.

For a TQUV accident in the Mark II containment, even without seal leakages, catastrophic failure is not predicted to occur during the five-hour simulation. However, containment temperatures and pressures are predicted to be well above the design value and substantial leakage through the drywell seals could occur using the leakage modeling developed by the Containment Performance Group (6).

Although fission product release calculations have not been performed at BNL, it is clear that a slow release over many hours would be less than for the previously assumed catastrophic failure. There is also the potential for additional reduction due to the action of the standby gas treatment system.

REFERENCES

1. Reactor Safety Study, "An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400, 1975.

2. Griemann, L. G., Famous, F., et al., "Reliability Analysis of Steel Containment Strength," NUREG/CR-2442, 1984.

3. Yue, D. D., and Cole, T. E., "BWR 4/MARK I Accident Sequence Assessment," NUREG/CR-2825, November 1982.

4. Cook, D. H., et al., "Station Blackout at Brown's Ferry Unit One--Accident Sequence Analysis," NUREG/CR-2182, November 1981.

5. Cook, D. H., et al., "Loss of DHR Sequence at Brown's Ferry Unit One--Accident Sequence Analysis," NUREG/CR-2973, May 1983.

6. Containment Performance Working Group, "Containment Leak Rate Estimates," NUREG-1037, to be published.

7. Perkins, K. R., Pratt, W. T., and Greene, G. A., "Containment Loading for Severe Accidents in BWRs with a Mark I Containment," BNL Informal Report, November 1984.

8. Mayo, S. E., et al., "IREP: Analysis of the Brown's Ferry Unit 1 Nuclear Power Plant," NUREG/CR-2802, July 1982.

9. Curry, J. J., et al., "IREP: Analysis of the Millstone Point Unit 1 Nuclear Power Plant," NUREG/CR-3085, January 1983.

10. Harrington, R. M., and Ott, L. J., "MARCH 1.1 Code Improvements for BWR Degraded Core Studies," Appendix B of NUREG/CR-3179, September 1983.

11. R. O. Wooton and H. I. Avci, "MARCH Code Description and User's Manual," Battelle Columbus Laboratories/USNRC Report, NUREG/CR-1711, October 1980.

12. J. F. Muir, et al., "CORCON-MOD 1: An Improved Model for Molten-Core/Concrete Interactions," NUREG/CR-2142, July 1981.

13. Murfin, W. B., "A Preliminary Model for Core/Concrete Interactions," SAND77/0370, August 1977.

14. USNRC, "Severe Accident Containment Loads," NUREG-1079, to be published.

Table 1

SUMMARY OF BWR MARK I CONTAINMENT LINER MELT-THROUGH RESULTS

RUN	CONCRETE*	CORIUM TEMPERATURE (K)	% OF CORE	TIME TO FAIL LINER(S)	AXIAL+ CONCRETE EROSION (cm)	THICKNESS+ OF LINER ABLATED (cm)
1	B	1775	80	NO MELT-THROUGH	3.3	0.1
2	L	1775	80	2842	1.2	3.0
3	B	1900	80	NO MELT-THROUGH	7.4	0.3
4	L	1900	80	895	1.5	3.0
5	B	2550	80	328	4.0	3.0
6	L	2550	80	208	1.6	3.0
7	B	2550	60	325	3.6	3.0
8	L	2550	60	226	1.6	3.0

* B = Basalt, L = Limestone

\+ At liner melt-through time.

Table 2

SUMMARY OF MARCH/CORCON RESULTS FOR THE TQUV SEQUENCE IN A MARK II CONTAINMENT

Case	5	5a	5c	5d	6	7	7a	8
Corium Spread (m)	5	5	5	5	3	5	5	3
Debris Temperature (°F)	4130	4130	4130	4130	2700	4130	4130	2700
Concrete Type	L	L	L	L	L	B	B	B
Free H_2O (%)	3	6	3	3	3	4	8	4
Pool Losses (%)	0	0	25	50	0	0	0	0
Results								
Peak Pressure (psia)	130	135	102	83	118	114	140	94
Peak Temperature (°F)								
Drywell Atmosphere	623	670	570	510	600	480	585	450
Drywell Concrete	320	330	305	280	340	310	325	280
Drywell Steel Liner	315	330	275	265	330	280	325	280
Wetwell Atmosphere	360	360	345	335	345	345	355	345
Wetwell Concrete	205	205	195	170	190	200	200	190
Wetwell Steel Liner	205	205	195	170	190	200	200	190

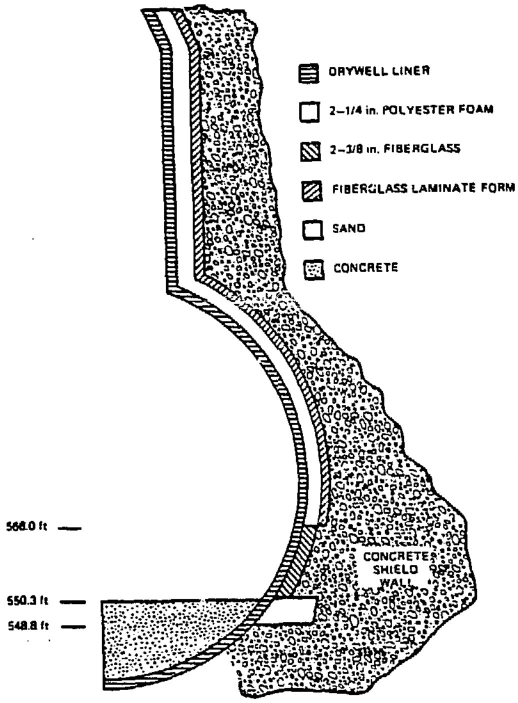

Figure 1 Drywell liner-concrete shield wall gap geometry (reproduced from Reference 5).

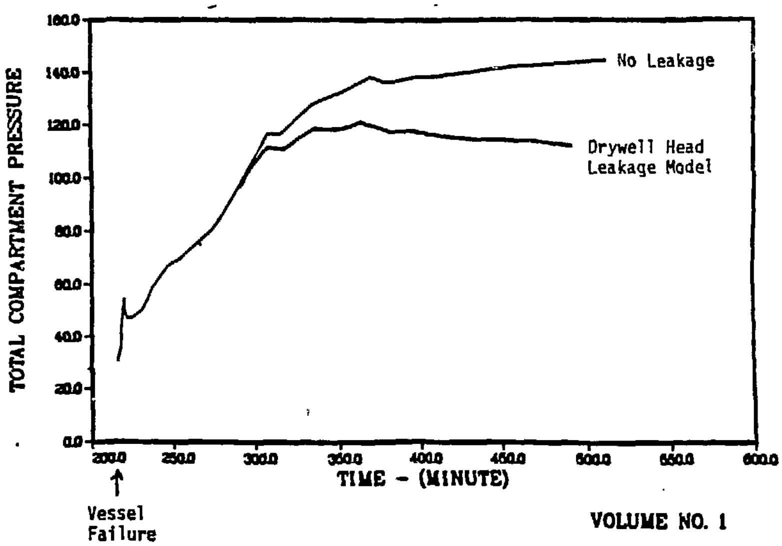

Figure 2 Calculated pressure response to a Class I loss-of-make-up accident in a BWR Mark I containment for two limiting core debris temperatures at vessel failure based on CORCON gas generation rates.