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IMPACT OF CORE-CONCRETE INTERACTIONS IN THE MARK I CONTAINMENT DRYWELL ON CONTAINMENT INTEGRITY AND FAILURE OF THE DRYWELL LINER

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Impact of Core-Concrete Interactions* in the Mark I Containment Drywell on Containment Integrity and Failure of the Drywell Liner

ABSTRACT

Previous containment analyses of the Mark I BWR have considered the γ-mode of containment failure as the dominant mode. The γ-mode is over-pressure 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 at 132 psia. However, results from the SASA program analyses of the 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 over-pressurization. It has become evident that a third mode of drywell failure must be considered under these specified accident conditions, in addition to the gross over-pressure failure and the leak-before-failure modes. This third mode of failure is local ablation of the steel drywell liner due to contact with the molten corium. In order to assess the drywell liner response to heat transfer from a pool of molten core debris during a core-concrete interaction, a calculational procedure consisting of both code calculations and hand calculations was developed. The general methodology was to calculate the melting attack on the steel liner by molten core debris that is simultaneously attacking the drywell concrete floor.

A comparison of the results of the calculations indicates that all three containment failure modes need to be considered simultaneously in order to accurately predict the pressure-temperature history in a Mark I BWR drywell. Leakage through drywell seals, as well as through local breaches in the liner due to melting, must be considered when estimating the structural response of the drywell. The transport of fission products and aerosols will also be affected by the location and timing of containment failure, as well as mode of failure, leakage area, and flow rate through the leakage area.

1. INTRODUCTION

The potential for containment failure from core melt accidents has been under review by the Nuclear Regulatory Commission (NRC) for some time. The possibility of early failure with the

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

potential for a large release of radioactivity (aerosol concentration is higher early in the accident) is the principle reason for this attention. Containment loads that might lead to such failure can result from severe accidents not normally considered in the design basis of nuclear power plants. In order to assess the inherent capability of various containment designs to mitigate the effects of a broad range of severe accidents, the NRC formed the Containment Loads Working Group (CLWG) with the objective of developing an updated evaluation of containment loads (temperature and pressure history) and associated challenges to containment integrity.

The overall approach was based on a standard problem methodology. The CLWG management team selected a specific reactor to represent each of the six containment designs deployed in the U.S. These were chosen to overlap with previous probabilistic risk assessments in order to provide a basis for evaluating progress in understanding severe accident phenomena.

This paper is an outgrowth of the BNL and ORNL participation in the CLWG and specifically deals with the likely failure mechanisms for Standard Problem 4 (SP-4).

The Containment Loads Working Group (CLWG) Standard Problem 4 is a TQUV-type accident sequence in a Mark I BWR containment in which all coolant injection fails at the time of reactor SCRAM from 100% power. Without coolant injection, the core uncovers within 30 minutes and since the ADS is assumed not activated, the primary system remains at high pressure. Shortly, the uncovered core of the reactor begins to melt, slumps into the RPV lower plenum, and eventually causes the reactor lower head to fail at approximately three hours after accident initiation. The molten corium is assumed to be displaced onto the reactor containment drywell floor immediately and to begin to attack the drywell concrete.

The Mark I containment consists of the drywell, pressure suppression pool, downcomer vents connecting the drywell and suppression pool, a containment cooling system, isolation valves, etc. The drywell is a steel pressure vessel, cylindrical at the top and spherical at the bottom. The vent system to the wetwell has eight circular downcomer pipes which penetrate the steel drywell liner, terminating in the pressure suppression pool. The pool is a toroidal steel pressure vessel which contains subcooled water for condensing primary system steam during normal transients.

The particular containment design chosen for Standard Problem 4 was that of the Browns Ferry Nuclear Power Station. In this containment, the molten core debris, consisting of approximately 80% of the core inventory, is assumed to fall downward into the reactor pedestal region forming a deep pool, filling the two containment sumps, and then flowing outward through the doorway over the annular drywell floor area. The sump volumes are approximately 3.8 m^3. Subtracting this from the initial corium inventory of 32.3 m^3 leaves 28.5 m^3 to be spread over a total of 132 m^2 of floor area. Assuming an even spread of all the debris over the entire floor results in a corium pool depth of 22 cm. Although this spreading is not mechanistically calculated, it is considered reasonable for the limiting high temperature debris case since pathways through the many obstructions are available, and there is empirical evidence that corium will flow at depths characteristic of this calculation [1]. For the high temperature limiting case, it is assumed that the debris will spread up to the steel containment liner itself.

Previous containment analyses of the Mark I BWR [2] have considered the γ-mode of containment failure as the dominant mode. The γ-mode is over-pressure 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 at 132 psia [3]. However, recent results from the SASA program analyses of the 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 over-pressurization [4,5,6]. Recent efforts by the Containment Performance Working Group (CPWG) have concentrated on determining the probability and timing of over-temperature failure of these penetrations, and the rate of leakage into the reactor building [7].

It has become evident that a third mode of drywell failure must be considered under these specified accident conditions in addition to the gross over-pressure failure and the leak-before-failure modes. This third 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 is assumed to flow outward from the pedestal region and contact the drywell liner. 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, a 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 volatile fission products. Although some of the gap between the drywell liner and the concrete is filled with fiberglass and polyester foam (see Figure 1), it is doubtful that they will present a significant obstacle to the flow of these high temperature gases from the drywell.

The objectives of this study are to:

(1) Develop a methodology to calculate the attack of molten core debris on the drywell liner,

(2) Parametrically study the impact of corium temperature, concrete composition, and fraction of core in corium on liner melt-through, and

(3) Compare the results to over-pressure and over-temperature failure times for a Mark I BWR.

2. PROBLEM SPECIFICATIONS FOR SENSITIVITY STUDIES

The CLWG Standard Problem 4 addresses the timing of the failure of the drywell due to over-temperature soaking of penetration seals (leak-before-fail) versus gross over-pressure failure of the steel liner (γ-mode failure). For SP-4, the core debris temperature and composition, the concrete composition, and the fraction of the core released were specified [8,9]. The specifications of the corium and concrete compositions as well as a summary of the sensitivity calculation specifications for SP-4 are listed in Tables 1 and 2, respectively.

The approach taken in the local liner failure calculations was somewhat different than for the SP-4 calculations reported in the CLWG report [10]. For SP-4, radiative heat transfer from the surface of the corium debris to the drywell containment structures and atmosphere was eliminated. All the sensible energy in the debris was thus forced into ablation of concrete, maximizing the concrete erosion rate and the generation of concrete decomposition gases. For the local liner failure calculations, however, radiative heat transfer from the corium debris surface was modeled. This enabled a more accurate calculation of the transient corium temperature, the most important variable in the calculation of the liner ablation rate. The concretes that were used in the calculations were a basalt- and a limestone-type, identical in composition to those specified for SP-4. The actual concrete composition at Browns Ferry is approximately an average of these two generic concretes (see Table 1). Three core debris temperatures were assumed: 2550 K, 1900 K, 1775 K. Mechanistically, the low temperature debris case is inappropriate since the debris probably would not be able to flow to the liner prior to solidifying. The radius of spreading of the debris on the drywell floor was assumed to be approximately 7 meters and the depth of the debris was held uniform. The debris required to fill the drywell sumps was subtracted from the debris inventory in order to calculate the corium depth. The radiative emissivity of the corium was given a constant value of 0.5. The fraction of the core that was allowed to participate in the core/concrete interaction was assumed to be 80% or 60%.

Although the TQUV accident sequence is a high pressure sequence with failure of the ADS, this was assumed to have no impact on the disposition of the corium in the drywell upon failure of the RPV. In other words, the debris was allowed to spread uniformly and homogeneously across the floor; high pressure jetting, impaction on the steel liner, and direct atmospheric heating were neglected. Although modeling of these phenomena may be desirable, they were neglected since they were beyond the scope of this study. A complete list of the parametric calculations chosen for the local liner melt-through evaluations is shown in Table 3.

3. CALCULATIONAL MODEL

In order to assess the drywell liner response to heat transfer from a pool of molten core debris during a core-concrete interaction, a calculational procedure consisting of both code calculations and hand calculations was developed. The general methodology was to calculate the melting attack on the steel liner by molten core debris that is simultaneously attacking the drywell concrete floor. The calculational tool that was used to analyze the attack of molten core debris on the drywell concrete floor was a modified version of the CORCON-MOD1 computer code [11].

CORCON-MOD1 is a general model describing the thermal and chemical interactions between molten core debris and structural concrete. The major components of the system are the concrete cavity, the molten debris pool, and the gas atmosphere and surroundings above the pool. The geometry of the system is formulated as a two-dimensional, axisymmetrical cavity, although specific geometries not available as code-supplied options may be user-input.

From the results of the CORCON code calculations, the maximum sidewards heat transfer coefficient across the gas film to the ablating concrete, h_i, was calculated at each time step as

$$h_i = \frac{q''_{conv} + q''_{rad}}{T_{interface} - T_{abl,concrete}}$$

where q''_{conv} and q''_{rad} are the convective and radiative components of heat transfer per unit area across the gas film, and $T_{interface}$ and $T_{abl,concrete}$ are the melt-gas film interfacial temperature and the concrete ablation temperature, respectively. This heat transfer coefficient was then used as input for the calculation of the transient heat-up and ablation of the steel liner. The heat transfer from the molten corium to the steel liner was modeled as one-dimensional transient convection 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

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

where ρ is the steel density, 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

$$\delta(t = t_0) = 3\ cm$$

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 time 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 when the downward erosion depth into the concrete exceeds the bubbled-up 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 when 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. Some of the physical properties and physical constants used in the calculations to be discussed are listed below:

ρ_{steel} = 8000 kg/m^3,
$h_{fs,steel}$ = 2.7 x 10^5 J/kg,
c_{psteel} = 500 J/kg K,
δ_{wall} = 3 cm .

4. RESULTS OF PARAMETRIC CALCULATIONS

The results of the calculations that were performed for the local liner failure problem are indicated in Table 4. Indicated on the table are the concrete type, corium temperature, percent of core participating in the interaction, total time to fail liner, total downward erosion at end of calculation, and thickness of liner ablated. 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 1775 K and 1900 K corium temperature cases on the basaltic concrete. Due to the low ablation temperature assumed for the basaltic concrete cases (≈1450K), the corium temperature dropped quickly upon contact since the basaltic concrete acts as a rapidly ablating, low temperature heat sink. As a result, the corium debris fell very rapidly below the steel ablation temperature, 1750 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 would 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 insure the eventual melt-through failure of the drywell liner, even for the case that the debris initial temperature was 1775 K.

It is apparent from these results that variation of the fraction of core in the core-concrete interaction had no impact on the ablation rate for both the high debris temperature limestone and basalt concrete cases. In none of the calculations did the corium debris penetrate deep enough into the concrete to terminate the calculations.

It is not clear if assigning the same ablation temperature to both the limestone concrete and the steel liner had any impact on the results of the low temperature limestone concrete-liner failure calculations. It would be desirable to lower the concrete ablation temperature by 25 K to determine if it would lower the debris temperature below the steel ablation temperature in time to prevent failure of the drywell by melt-through, in much the same way the basalt concrete calculations behaved. It is clear, however, that the only cases that liner failure by melt-through was avoided were those for which the corium debris temperature fell below 1750 K prior to liner melt-through.

5. DISCUSSION OF RESULTS

Until recently, the most likely modes of containment failure in a Mark I BWR were considered to be over-pressurization of the drywell and structural failure of the drywell liner or failure of sealing materials due to degradation at elevated temperatures and leakage through these degraded seals into the reactor building.

It is now apparent that if the Mark I containment is going to fail under the threat presented by an ex-vessel core-concrete interaction, it may occur early in the interaction due to melt-through of the steel drywell liner if the core debris is able to flow to and ablate the liner. In some cases, the drywell liner was calculated to fail within five minutes of contact with molten core debris, taking as long as 45 minutes in one case. In only two cases, with relatively low temperature debris interacting with a highly basaltic concrete, was the liner calculated to survive.

A comparison of calculated or estimated drywell failure times (time after RPV failure) for these three failure modes discussed is presented in Table 5. The calculations are for a TQUV accident sequence in a Browns Ferry-type Mark I containment with no CRD flow. In these calculations, the containment response calculations were performed with the MARCH 1.1B computer code [12] developed at ORNL, which contains some modeling changes specific to the Mark I not available in MARCH 1.1 [13]. The containment failure results which are presented employed CORCON-MOD1 calculations which were input to MARCH 1.1B in tabular form, bypassing the INTER model [14] in MARCH, which has been shown to overpredict concrete erosion rates and gas generation rates during core-concrete interactions.

The containment leakage times quoted in Table 5 are estimated from Reference [7] using the pressure-temperature histories from Reference [10]. Using the medium pre-existing leak area results for ethylene propylene seal material at 500 F, the seal soak time to initiate leakage is 18 minutes and the ramp time to totally degrade the seal material is 16 minutes. The over-temperature failure times listed indicate the sum of the times to achieve 500 F in the drywell atmosphere plus an additional 34 minutes. All times listed in Table 5 are "time after RPV failure."

Note that the over-pressurization failure times vary from over two hours for CLWG Case 1 to over eight hours for Cases 2 and 3. Case 4, with an extrapolated over-pressure failure time of 16 hours, is considered highly unlikely to actually fail the containment at all on pressure. The over-temperature failure times from the CPWG criteria are significantly shorter, varying from one hour for Case 1 to 3-1/2 hours for Case 2. Cases 3 and 4 are not calculated to fail at all on over-temperature. However, the local liner melt-through calculations indicate that failure may be expected as early as 3-1/2 to 5-1/2 minutes after the initiation of ex-vessel core-concrete interactions for Cases 1 and 3, to as much as 45 minutes for Case 2. These times are much less than the failure times for either of the other two failure modes. Case 4 was not calculated to melt through the liner.

What is evident from this comparison is that all three containment failure modes need to be considered simultaneously in order to accurately predict the pressure-temperature history in a Mark I BWR drywell. Leakage through drywell seals as well as through local breaches in the liner due to melting must be considered when estimating the structural response of the drywell. The transport of fission products and aerosols [15] will also be affected by the location and timing of containment failure, as well as mode of failure, leakage area, and flow rate through the leakage area.

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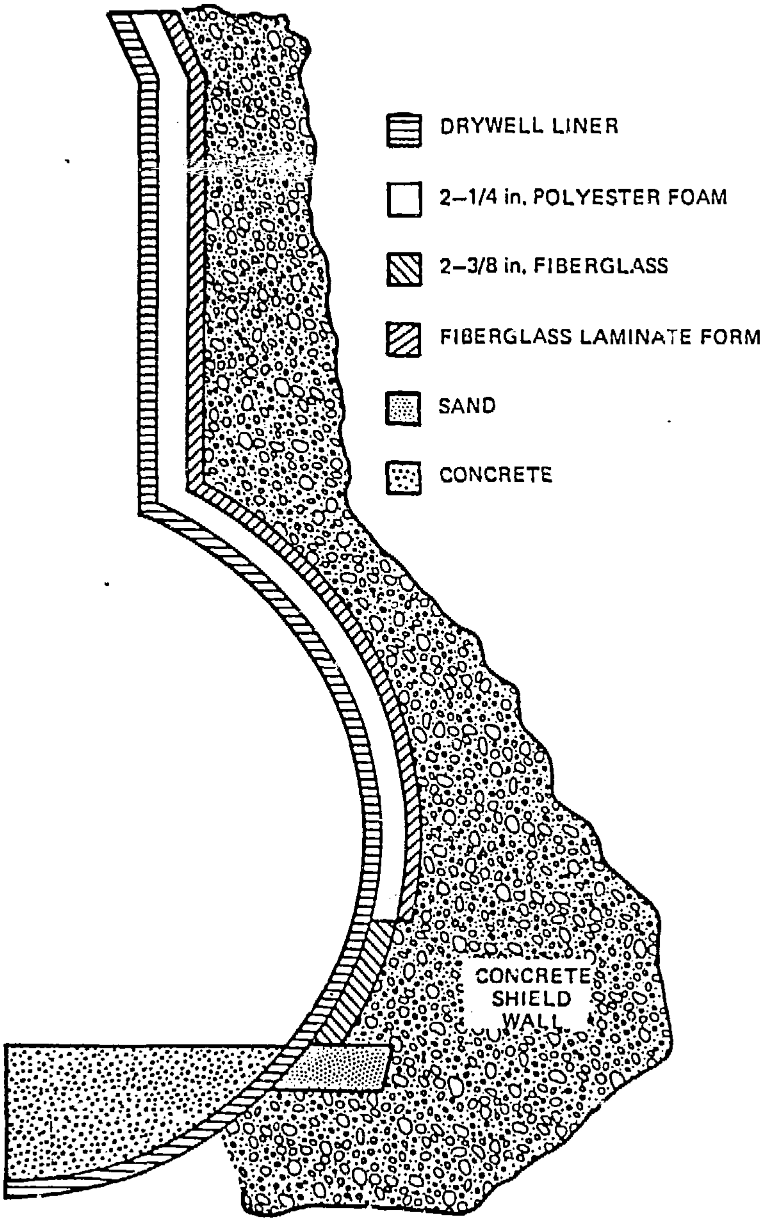

Figure 1 Drywell Liner - Concrete Shield Wall Gap Geometry (ORNL-DWG 83-4244 ETD)

TABLE I

SPECIFICATION OF CORIUM AND CONCRETE COMPOSITIONS FOR SP-4

CONCRETE	LIMESTONE	BASALT	BROWNS FERRY
WEIGHT FRACTIONS:			
$CaCO_3$	0.80	0.01	0.45
$Ca(OH)_2$	0.15	0.18	0.07
SiO_2	0.01	0.57	0.39
Free H_2O	0.03	0.04	0.05
Al_2O_3	0.01	0.20	0.04

CORIUM	
UO_2	127000 kg
ZrO_2	9160 kg
FeO	12250 kg
Fe	41920 kg
Zr	45380 kg
Ni	4450 kg
Cr	8000 kg

TABLE II

SUMMARY OF SENSITIVITY CALCULATION SPECIFICATIONS FOR SP-4

Case Number	1	1a	2	3	3a	4
Corium Spread (m)	5	5	3	5	5	3
Debris Temperature (K)	2550	2550	1755	2550	2550	1755
Concrete Type	L	L	L	B	B	B
Free H_2O (%)	3	6	3	4	8	4
Steel in Corium (lb)	140K	140K	140K	140K	140K	140K

TABLE III

MATRIX OF BWR MARK I LOCAL FAILURE CALCULATIONS

CASE NUMBER	1	2	3	4	5	6	7	8
Corium Spread (m)	6	6	6	6	6	6	6	6
Debris Temperature (K)	1775	1775	1900	1900	2550	2550	2550	2550
Concrete Type	B	L	B	L	B	L	B	L
Corium Fraction (%)	80	80	80	80	80	80	60	60
Corium Composition	---------------------- See Table 1 -----------------							

TABLE IV

SUMMARY OF BWR MARK I LOCAL FAILURE CALCULATION 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 V

COMPARISON OF APPROXIMATE DRYWELL FAILURE TIMES BY OVER-PRESSURE, OVER-TEMPERATURE, AND LINER MELT-THROUGH

CLWG CASE	DEBRIS TEMPERATURE CONCRETE COMPOSITION	MAXIMUM DRYWELL+ P AND T	CLWG OVER-PRESSURE FAILURE(MIN)	CPWG OVER-TEMPERATURE FAILURE (MIN)	LINER MELT-THROUGH FAILURE(MIN)
1	2550 K, Limestone	145 psia 622K(660F)	133	62	3.5
2	1755 K, Limestone	88 psia 533K(500F)	500*	329	45
3	2550 K, Basalt	108 psia 477K(400F)	460*	No Leakage Calculated	5.5
4	1755 K, Basalt	65 psia 411K(280F)	950* Failure Unlikely	No Leakage Calculated	No Melt-Through Calculated

* Extrapolated value.
\+ Maximum during five hours of core/concrete interaction.

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