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A STUDY OF LIGHT WATER REACTOR CONTAINMENTS UNDER IMPORTANT SEVERE ACCIDENT CONDITIONS

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Abstract

The U.S. Nuclear Regulatory Commission has sponsored studies to develop a "LEAKAGE-BEFORE-FAILURE" model for use in severe accident risk assessments to provide a means of accounting for significant containment leakge prior to reaching the containment threshold pressure. Six containment types have been studied (large dry, subatmospheric, ice condenser, Mark I, II and III). Potential leak paths through major containment penetration assemblies were investigated and upper-bound estimates of leak areas established. These leak areas may result from increasing internal pressure and degradation of nonmetallic seal materials due to severe accident conditions. This paper describes the approach and summarizes the results and conclusions of this study.

1. Introduction

Most severe accident risk assessments have utlized a "THRESHOLD" model to characterize loss of containment integrity. If the containment pressure is below a certain threshold pressure it is assumed that the containment does not fail and offsite consequences are quite low. If containment pressure is above the threshold pressure it is assumed that the containment fails and significant fission product inventory is released.

The U.S. Nuclear Regulatory Commission (USNRC) has sponsored studies to develop a "LEAKAGE-BEFORE-FAILURE" model to provide a means of accounting for significant containment leakage prior to reaching the threshold pressure. Six containment types have been studied (large dry, subatmospheric, ice condenser, Mark I, II and III). The pressure and temperature response of each containment type under no leakage conditions for certain important severe accident sequences were reviewed. Based on these responses, potential leak paths through major containment penetration assemblies (such as the equipment hatch, airlock, purge and vent valves and drywell head for BWRs) were investigated. Upper-bound estimates of leak areas that may result from increasing internal pressure and degradation of nonmetallic seal materials due to severe accident conditions have been made. These leak area estimates are then incorporated into thermodynamic analyses of important accident sequences to calculate containment response and leak rates as a function of time.

This paper describes the approach and summarizes the results and conclusions of this study. A complete description of the details and results is provided in NUREG-1037 [1].

2. Approach

The approach taken in conducting this study involved the detailed review of containment penetration designs and an analytical treatment of penetration performance. The studies are based on investigations of the following six reference plants in the U.S.A:

(1) PWR, Large Dry - Zion
(2) PWR, Subatmospheric - Surry
(3) PWR, Ice-Condenser - Sequoyah
(4) BWR, Mark I - Peach Bottom
(5) BWR, Mark II - Limerick
(6) BWR, Mark III - Grand Gulf

The loads used to assess the performance of the containment penetrations were based on USNRC studies that developed a number of standard problems to test the integrity of the six containment types. The standard problems were carefully selected to represent the most severe pressure/temperature histories that could be expected to occur over an extended period of time in the containment buildings during postulated core meltdown accidents. These loads are discussed in NUREG-1079 [2].

This study concentrated on identifying potential leakage paths that may occur prior to reaching currently reported containment shell capability pressures. These capability pressures generally correspond to the point when the containment first reaches an initial general yield state. Consequently, the study did not consider potential leak paths that may result from large containment deformations. The capability pressures used in the study are based on references [3] through [7].

For each containment type the major penetrations having the greatest potential for leakage were identified and evaluated. The study included the following penetrations:

- Large Opening Penetrations
  - -- equipment hatch
  - -- personnel airlock
  - -- drywell head (BWR)
  - -- fuel transfer tube
  - -- CRD removal hatch
- Purge and Vent System Isolation Valves
- Piping Penetrations
- Electrical Penetration Assemblies

For all six reference plants, it was found that the above penetrations would maintain their structural integrity up to the capability pressures used in this study. However, the flanges of pressure unseating equipment hatches and BWR drywell heads were predicted to separate under the effects of the severe accident pressure. For some personnel airlocks, the flat bulkhead door frames were predicted to separate from the airlock doors. In these cases, the leak area was dependent upon the magnitude of the pressure and the integrity of the penetration seals. Since little data was available regarding penetration seal behavior under severe accident condtions, it was decided to report only upper bound leak area estimates. For the above penetrations, upper bound estimates are considered to correspond to the separation area predicted for the penetration. In essence, it is assumed that the seals have little resilience. (Some credit for seal resilience was given in that the seals are assumed to be effective in preventing leakage for separation areas predicted to occur below the design pressure). In addition, it was also assumed that the personnel airlocks would have only one door available to resist the containment pressure and the possible effect of thermal loads on the separation area associated with pressure unseating penetrations was neglected. Both assumptions are consistent with the goal of determining upper bound leak area estimates.

For two other types of penetrations, purge and vent valves and personnel airlocks with inflatable seals, the upper bound leak area estimates were not pressure dependent. For the purge and vent valves, which are generally large diameter butterfly valves, the maximum potential leak area corresponds to the metal-to-metal clearance between the valve disc and the body. For airlock doors with inflatable seals, the maximum potential leak area corresponds to the clearance between the door and the door frame. For upper bound leak area estimates, it was assumed that the seals of these penetrations become totally degraded when the containment temperatures remain high enough for a sufficient length of time to exceed the reported design life of the sealing material.

3. Summary of Results

A summary of the findings for each containment type is presented in this section. As noted above, the upper bound leak area estimates are limited to the containment shell capability pressures used in this study.

For the PWR large dry containment (Zion), the only significant leak source is from the personnel airlock. The door frame was predicted to yield at a pressure of 107 psig with a corresponding leak area of 1 in^2. At the capability presssure of 134 psig the leak area was estimated to be 5 in^2. Containment temperatures were predicted to be below 400°F and did not threaten purge and vent valve seal integrity. The above leakage estimates would delay the time required for the containment to reach its capability pressure. However, such leakage is expected to have little impact on offsite consequences.

For the PWR subatmospheric containment (Surry), the leakage was relatively small with a leak area of 0.4 in^2 at the capability pressure of 119 psig. This leak area was attributed to the pressure unseating airlock barrel flange seal which was predicted to unseat at a pressure of 81 psig and result in a leak area of 0.3 in^2 at 119 psig. The remainder of the leak area was attributed to the personnel airlock. These leak areas did not affect containment response and are also expected to have little impact on offsite consequences. The same conclusion was made for the PWR ice-condenser containment (Sequoyah) which was predicted to have a leak area of 0.3 in^2 (due to leakage through two personnel airlocks) at the capability pressure of 50 psig.

A. For the BWR Mark I containment (Peach Bottom), a leak area of 35 in^2 was estimated at the capability pressure of 117 psig (this pressure was determined from analyses of the Browns Ferry plant). Most of this leak area (approximately 95 percent) is attributed to the drywell head which is predicted to unseat at a pressure of 27 psig. The remainder of the leak area results from the equipment hatch which unseats at a pressure of 82 psig and the personnel airlock whose door frame yields at 94 psig. As discussed in Section 2, the above leak areas are attributed to the potential lack of seal resilience. However, the seals may also become degraded as a result of high containment temperatures which, for some studies, are predicted to exceed 700°F in the drywell. Such temperatures could create an environment for exceeding the life of the sealing materials used in the above penetrations, even though the seals are silicon rubber. The high containment temperatures also may result in a leakage path through the purge and vent lines which use double isolation valves with ethylene propylene for the seat material. This material is more susceptible to high temperature conditions than the silicon rubber seals used in the above penetrations. A leak area of approximately 14 in^2 would occur is these seals fail. However, this leakage path is considered to be less likely than the path through the drywell head since the second seal is well isolated from the containment atmosphere.

For the BWR Mark II containment (Limerick), a leak area of 42 in^2 was estimated at the capability pressure of 140 psig. Approximately 80 percent of this leak area is attributed to the drywell head which is predicted to unseat at a pressure of 85 psig. The remainder of the leak area results from the two equipment hatches which unseat at a pressure of 75 psig. As in the case of the Mark I containment, this leakage could result not only from the lack of seal resilience, but also from seal degradation due to high containment temperatures. For the Mark II containment, drywell temperatures are predicted between 600°F and 800°F. For the case of Limerick, purge valve leakage is not expected since the valves are equipped with a metal-to-metal seal.

The leakage estimated for the BWR Mark I and II containments does alter containment response and its effect on offsite consequences needs to be carefully assessed. Tests should be conducted to better characterize the size and likelihood of such leakage. However, it is also noted that changes in the BWR operating procedures involve the use of wetwell venting. If the wetwell is vented then there will be no driving force to produce significant drywell leakage (even with high drywell temperatures).

For the BWR Mark III containment it was determined that no significant leakage would result from pressure loadings up to the capability pressure of 60 psig. Furthermore, it was concluded that the drywell and containment personnel airlocks would maintain their integrity even in the presence of diffusion flames in the wetwell. Since suppression pool by-pass early in the accident is important, tests should be conducted to confirm this finding. The drywell personnel airlock, which utilizes a double inflatable seal design, is predicted to contribute to significant by-pass leakage (approximately 125 in^2 with the seals entirely blown out) due to high temperatures in the drywell during core/concrete interactions. During this period, some studies have predicted that the drywell temperature will reach 900°F, or above, and remain at this level until the containment reaches its capability pressure.

However, the resulting by-pass leakage occurs late in the accident sequence and is expected to have little impact on offsite consequences.

It should be noted that all of the above comments regarding offsite consequences are qualitative and intended to indicate potential trends. The leakage estimates must be incorporated into containment failure mode and fission product release analyses to determine the quantity and characteristics of the radionuclide release. It should also be noted that the point of release of the radionuclides is important (auxiliary building vs directly to the atmosphere) and will certainly influence the consequences.

4. <u>Conclusions</u>

It is not likely for the severe accident conditions considered in this study that the reported leak area estimates will be exceeded. On the other hand, smaller leak areas may also be justified. The results presented should be utilized to assess the impact of containment leakage on the radiological consequences of an accident. However, until more test data is available, these results should be coupled with the results obtained from threshold models.

In addition, this study has led to the following general conclusions:

- The potential for containment leakage through penetrations prior to reaching currently reported containment capability pressures should be considered in severe accident risk estimates.
- The potential for significant leakage before reaching currently reported containment capability pressures appears to be greater for BWRs than PWRs.
- Leakage before reaching containment capability pressures can also occur with PWRs, but such leakage is much more plant specific.
- Failure of nonmetallic seals for containment penetrations (primarily pressure unseating equipment hatches, personnel airlocks, drywell heads and purge valves) are the most significant sources of containment leakage.
- Although generic studies of containment types are useful in identifying sources of containment leakage, final conclusions may need to be plant specific.
- Current efforts rely on analysis and engineering judgement. Additional test data is needed to better quantify the leak tightness of containment penetrations when subjected to severe accident conditions.

Based on the results to date, both analytical and experimental studies should continue to better characterize containment leakage prior to reaching containment capability pressures as defined above. Furthermore, efforts should be made to better define the confidence levels associated with these capability pressures. Future studies should include the following:

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- Tests to fully assess the behavior of penetration seals under severe accident pressure and temperature conditions, including the effects of aging and radiation.
- Sensitivity studies to assess the potential variation of containment leakage within the family of each containment type.
- Sensitivity studies to determine the magnitude and timing of containment leakage which can have a significant effect on radiological consequences.
- An assessment of the potential for plugging of identified leak paths.
- An assessment of the survivability of equipment inside containment during important severe accident sequences.
- Identification of leakage paths after release from containment and an assessment of the effect of holdup of releases to the auxiliary or reactor buildings.

5. References

/1/ U.S. Nuclear Regulatory Commission, "Containment Performance Working Group Report", NUREG-1037, 1985.

/2/ U.S. Nuclear Regulatory Commission, "Containment Loads Working Group Report", NUREG-1079, 1985.

/3/ IDCOR Technical Report 10.1, "Containment Structural Capability of Light Water Nuclear Power Plants", Technology for Energy Corporation, July 1983.

/4/ Pananos, W.J. and Reeves, C.F., "Containment Integrity at Surry Nuclear Power Station", Stone and Webster Engineering Corporation Report TP-84-13, February 1984.

/5/ Greimann, L.G., et al., "Reliability Analysis of Steel Containment Strength", NUREG/CR-2442, June 1982.

/6/ Greimann, L.G., et al., "Reliability Analysis of Containment Strength - Sequoyah and McGuire Ice-Condenser Containments", NUREG/CR-1891, August 1982.

/7/ Sharma, S., et al., "Failure Evaluation of a Reinforced Concrete Mark III Containment Structure Under Uniform Pressure", NUREG/CR-1967, September 1982.

NOTICE

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