Analysis of Core-Concrete Interaction Event with Flooding for the Advanced Neutron Source Reactor

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ABSTRACT

This paper discusses salient aspects of the methodology, assumptions, and modeling of various features related to estimation of source terms from an accident involving a molten core-concrete interaction event (with and without flooding) in the Advanced Neutron Source (ANS) reactor at the Oak Ridge National Laboratory. Various containment configurations are considered for this postulated severe accident. Several design features (such as rupture disks) are examined to study containment response during this severe accident. Also, thermal-hydraulic response of the containment and radionuclide transport and retention in the containment are studied. The results are described as transient variations of source terms, which are then used for studying off-site radiological consequences and health effects for the support of the Conceptual Safety Analysis Report for ANS. The results are also to be used to examine the effectiveness of subpile room flooding during this type of severe accident.

I. INTRODUCTION

The Advanced Neutron Source (ANS) is to be a multipurpose neutron research center and is currently in the design stage at the Oak Ridge National Laboratory (ORNL). The major areas of research will concentrate on condensed matter physics, materials science, isotope production, and fundamental physics.[1,2] ANS is planned to be a 330-MW heavy-water-cooled, and moderated research reactor housed in a large, double-walled containment dome and surrounded by thermal neutron beam experimental facilities. The reactor uses U_3Si_2–Al cermet fuel in a plate-type configuration. Cooling systems are designed with many safety features, including large heat sinks sufficient for decay heat removal; passive inventory control by accumulators, pools, and flooded cells; a layout that maximizes natural circulation capabilities; and fast, redundant shutdown systems. A defense-in-depth philosophy has been adopted. In response to this commitment, ANS project management initiated severe accident analyses and related technology development early in the design phase. This was done to aid in designing a sufficiently robust containment for retention and controlled release of radionuclides in the event of an accident. It also provides a means for satisfying on- and off-site regulatory requirements, accident-related dose exposures, containment response, and source-term best-estimate analysis for Level-2 and -3 Probabilistic Risk Analyses (PRAs) that will be produced. Moreover, it will provide the best possible understanding of the ANS under severe accident conditions and, consequently, provide insights for development of strategies and design philosophies for accident mitigation, management, and emergency preparedness efforts.[3]

A focused severe accident study was conducted to evaluate conservatively scoped source terms to support the ANS Conceptual Safety Analysis Report (CSAR) and to aid in the introduction of built-in design features for mitigation and management controls. This paper describes thermal-hydraulic and radionuclide transport modeling aspects along with analyses conducted for deriving source terms in support of the ANS CSAR. Because severe accident technology for the ANS is in an early stage of development, relevant mechanistic tools have not been developed for evaluating core-melt-progression phenomena. Consequently, a conservatively scoped scenario was postulated and analyzed. To provide initial source-term estimates for the high-consequence, low-probability end of the severe-accident-risk spectrum, early containment failure cases also are evaluated and reported in this paper. In addition, containment response for an intact containment configuration is analyzed.

II. DESCRIPTION OF ANS SYSTEM DESIGN

The ANS is currently in the conceptual design stage. As such, design features of the containment and reactor systems are evolving, based on insights from on-going studies. Table 1 summarizes the current principal design features of the ANS from a severe accident perspective compared with ORNL's High Flux Isotope Reactor (HFIR)[4] and a commercial light-water reactor (LWR). Specifically, the ANS reactor will use about 15 kg of highly enriched (i.e., 93% ^{235}U enrichment) uranium silicide fuel in an aluminum matrix with plate-type geometry and a total core mass of 100 kg. The power density of the ANS will be about 2 to 3 times higher than that of the HFIR and about 50 to 100 times higher than that of a large LWR. Because of such radical differences, high-power-density research reactors may give rise to significantly different severe accident issues. Such features have led to increased attention being given to phenomenological considerations dealing with steam explosions, recriticality, core-concrete interactions, core-melt progression, and fission-product release. However, compared to power reactor scenarios, overall containment loads from hydrogen generation and deflagration are relatively small for the ANS.

The reactor core is enclosed within a core pressure boundary tube and enveloped in a reflector vessel, which is immersed in a large pool of water. Experiment and beam rooms for researchers are located on the first and second floors, which are connected to the third-floor high-bay region through a rupture disk. The subpile room housing the control rod drive mechanisms also is connected to the third floor through lines with a rupture disk in between. The approximately 95,000-m^3 (61-m diameter) primary containment of the ANS consists of a 25-mm steel shell housed in a 0.8-m-thick, reinforced concrete secondary containment wall with a 1.5-m gap in between. The targeted design leak rate for the primary containment is 0.5 vol %/day (to the annulus), and for the secondary containment, the design leak rate is 10 vol %/day. Annulus flow is exhausted through vapor and aerosol filters. The containment isolation system is designed to initiate closure of isolation valves automatically on lines that penetrate the primary containment wall.

III. MODELING OF ANS CONTAINMENT THERMAL-HYDRAULICS AND RADIONUCLIDE TRANSPORT

This section describes the accident scenario postulated in this study, modeling for the ANS containment, and thermal-hydraulic-cum-radionuclide transport analysis.

A. Description of Severe Accident Scenario

Because the ANS is in the preliminary stage of severe accident technology development, it has not been possible to develop mechanistic models for capturing core-melt progression phenomena. A severe accident scenario is postulated for this study with a view toward evaluating conservatively estimated source terms. The assumed scenario is designed to evaluate maximum containment loads from the release of radionuclide vapors and aerosols and the associated generation of combustible gases. Two accident scenarios can be postulated in the absence of analysis of the front-end accident progression. One represents containment loads due to reactor pool steaming following the reactor core being dispersed into the reactor pool. The other case postulates that the reactor core penetrates the primary system boundary and thereafter relocates into the subpile room where it undergoes an extensive interaction with the concrete floor. The first case was extensively analyzed and reported elsewhere along with the results of the second case for the dry subpile room configuration.[5] This paper will describe containment response and radionuclide transport following the molten core-concrete interaction (MCCI) in a flooded subpile room. An option for subpile room flooding was studied as a measure of accident management to minimize containment loads during the MCCI event.[3] Flooding may also result from core-melt progression whereby a significant quantity of water may accompany the core debris as it relocates from the reactor coolant system (RCS) to the subpile room floor. Results will be compared with those for the dry subpile room case to determine the effectiveness of the subpile room flooding during this severe accident.

After more than a decade of research into severe accidents for power reactors, it is now well known that the study of MCCIs represents an important phase of any hypothetical severe accident that results in core debris relocating outside of the primary system onto a concrete surface. MCCI events can release large amounts of combustible gases [Carbon monoxide (CO) and Hydrogen (H_2)] as well as considerable quantities of radionuclides in the form of vapors and aerosols. Because of the relatively high power density of the ANS fuel debris, it is postulated that, during a core-meltdown accident, core debris could ablate penetration seals or other reactor vessel boundary structures and fall on the concrete floor of the subpile room. Flooding may also be caused by reactor coolant water accompanying molten core debris relocating through the breached reactor vessel bottom structure. Thereafter, core debris would spread, and an MCCI event would begin. For this scenario, a water injection system is assumed to be actuated to flood the concrete floor in a way not to trigger an extensive interaction between the molten core debris and water (i.e., steam explosion). Based upon bounding heat transfer calculations, the scenario postulated for the current study conservatively assumes that core debris would relocate at 75 s after reactor scram on a concrete floor in the flooded subpile room. Thereafter, containment capacity will be challenged from the resulting loads arising from combustible gas deflagration and released radionuclides, in addition to other gases produced from MCCI. Three containment configurations are assumed regarding source term release. For the intact containment configuration, it is assumed that the containment isolation system functions upon demand and both containment walls (primary steel shell and secondary concrete wall) stay leak-tight. Another configuration assumes that the primary containment shell fails while the secondary wall remains intact. To evaluate a maximum possible source term, the configuration involving failure of both containment boundaries was also modeled and studied.

B. MELCOR Modeling of ANS Containment

The MELCOR severe accident analysis code (Version 1.8.2) was used to develop an overall representation of the ANS containment. MELCOR is a fully integrated computer code that has been developed primarily for power reactor severe accident analysis.[6] However, MELCOR cannot model specific ANS core-melt progression phenomena associated with radically different fuel types, power densities, materials, and geometry. Therefore, MELCOR was used at this stage primarily for capturing containment transport phenomena. The MELCOR model of the ANS containment is represented by 11 control volumes, 15 flow paths, and 21 heat structures (representing walls, ceilings, shells, and miscellaneous structures) of various shapes (Fig.1). Aerosol and vapor filtration processes are also modeled, as are several complex aerosol and vapor transport phenomena associated with various severe accident scenarios. Fission product inventory and its associated decay heat were calculated using the ORIGEN2 code for the ANS core-averaged end-of-cycle, assuming a 17-d core life at an operating power level of 330 MW.

For the study of the MCCI event, all volatile fission products (i.e., noble gases, halogens, cesium, and tellurium) were directly sourced into the subpile room atmosphere at the start of evaluations of radionuclide transport. Initially iodine is specified in vapor form, whereas cesium and tellurium species are specified to be in aerosol form. No fission product chemistry effects were accounted for in this study, and thus no chemical interaction was assumed between the fission products and water in the subpile room. The nonvolatile species contribute to the continuation of MCCI by staying in the debris, which continues to ablate the concrete and heat up the water pool above the concrete floor; that is, they are not allowed to volatilize or form aerosols. About 50% of the total core decay power is associated with nonvolatile fission products. For this study, mass and energy of gases generated from the MCCI were obtained through an independent study[7] and then specified through user input. Figure 2 shows MELCOR input values for noncondensable gases generated as a result of MCCI.

For modeling containment failure, upon occurrence of a severe accident, a 0.5-m diameter opening is made available in the high-bay region primary containment shell for release of radionuclides. Such a release can occur either directly to the environment without filtration or to the annulus region housed in the secondary containment. Release to the environment is conservatively modeled to occur at ground level. Such pathways simulate early containment failure from the possible effect of explosive and/or external events as well as the possibility of failure of isolation valves in ventilation ducts.

The ANS containment (normal and emergency) ventilation flow paths were not modeled or accounted for as being potential radionuclide release pathways. However, note that the 0.5-m diameter containment failure path postulated for some cases is based on the assumed failure-to-isolate of one normal containment ventilation line; it also could represent an opening created by missiles or shock waves generated during energetic events such as steam explosions.

The subpile room is modeled as though functioning igniters exist. Therefore, if oxygen is available there, any combustible gases (e.g., H_2 and CO) will be allowed to deflagrate (but not to detonate because models for shock wave generation, transport, and structural interaction aspects have not yet been incorporated in MELCOR). The basemat of the subpile room is modeled as being made of limestone–common sand concrete.

Rupture disks are planned for incorporation (and modeled) to allow passage of materials between the subpile room and the high-bay region and between the high-bay region and the first- and second-floor volumes (where experimental scientists are located), respectively. These rupture disks open if a pressure differential of 115 kPa (2 psig) or greater is imposed. The doorway in the subpile room leading to the access tunnel will fail to open if a pressure differential of 136 kPa (5 psig) or greater is imposed.

The filter trains are modeled to perform with decontamination factors of 100 for iodine and 200 for aerosols, respectively, without consideration of filter degradation.

IV. RESULTS OF SOURCE TERM EVALUATION

MELCOR predictions of containment thermal-hydraulic behavior, radionuclide transport, and source terms are presented in this section. The results will be compared with the dry subpile room case presented elsewhere.[5]

A. Molten-Core-Concrete Interaction Event with Flooding in the Subpile Room

Intact Containment Configuration

Key results of interest for the MCCI event are given in Fig. 3. As noted in Fig. 3, the subpile room pressure for the intact containment configuration rises rapidly because of the intensity of the MCCI. It causes the rupture disk to open and allow passage of radionuclides to the high-bay area. The pressure in the subpile room does not rise high enough to cause the door leading to the subpile room tunnel to fail. However, a direct pathway exists from the high-bay region to the subpile room tunnel, which causes the pressure in the subpile room tunnel to rise concomitantly. About 6-h into the transient, the 51-m^3 water pool in the subpile room starts to boil. Subsequently, a rapid pressure increase beyond 115 kPa (2 psig) in the subpile room and the high-bay area results in the failure of a rupture disk, allowing pressure relief through a passage to the experiment and beam rooms on the first and second floors, respectively. Thereafter, pressures in all the containment regions increase rather slowly to approximately 115 kPa at around 33-h . The containment pressure then slowly decreases due to steam condensation and continuing heat loss to the containment wall structures. A few short spikes in subpile room pressure, lasting a few seconds immediately following the debris relocation into the subpile room (not shown in the figure), are caused partly by H_2 and CO deflagration. Afterward the oxygen content is completely depleted. Because no ventilation flow path is available in the model to bring in a fresh supply of oxygen, hydrogen combustion stops. A very high temperature (i.e., around 1000 K) can result in the subpile room atmosphere because of heating from fission products and combustion of H_2 and CO. After the initial high temperature rise, subpile room air begins to cool as combustion ceases, and heat-producing radionuclides are transported to the high-bay region, coupled with energy absorption in structure materials. Many radionuclides are deposited on cold structural surfaces in this case. About 0.1% of the noble gas inventory, about 5 x 10^{-4}% of the halogen inventory, about 9 x 10^{-5}% of the cesium- and tellurium-class inventories are released into the environment over 72-h. These negligibly low source term values essentially are caused by the leak-tight nature of the intact ANS dual-containment design and by the containment size being large enough to accommodate significant pressure and thermal sources.

Failed Containment Configurations

Figure 4 shows the results for the MCCI case with early containment failure. Variations of important parameters in the subpile room are similar to those seen for the intact containment case. One major difference, which may be expected, deals with the degree of high-bay region pressurization. A very mild pressurization is observed in various control volumes, after the rupture disk between the subpile room and the high-bay region opens following debris relocation into the subpile room. This behavior is expected for early containment failure cases. The assumed opening (or break) size of a 0.5-m-diam hole is large enough to prevent a significant pressure buildup in the containment, of which both the primary- and the secondary-containment failure is assumed. Even the single (primary) containment failure case shows a very mild pressurization. This is mainly due to an increased containment volume (e.g., annulus region), additional heat sink, and structural surfaces for increasing heat transfer and steam condensation. Consequently, the high-bay region pressure is well below 115 kPa (2 psig), and thus the rupture disks leading to the first- and second-floor areas stay closed. For the dual-containment failure case, as seen in Fig. 4, about 20.1% of the noble gases, 19.8% of the halogen inventory, and 15% of the cesium and tellurium inventories are released into the environment over 72 h. During the same time period for the single- (primary) containment failure case, about 9.6% of the noble gas inventory, 1.3% of the halogen inventory, and 3.4% of the cesium and tellurium inventories are released. It is noteworthy to see that the source term of the single- (primary) containment failure case comprises only 1.3% of the halogen inventory compared with 19.8% for the dual-containment failure case. This substantial reduction in the halogen release is due to additional structural surface area for iodine deposition. With the annulus region available, deposition surface areas increase over three times. Also note that for the MCCI case with early containment failure, most radionuclide releases occur well within the first few hours since the debris relocation into the subpile room and steaming due to the heat-up of the water pool in the subpile room.

B. Comparison Against Dry Subpile Room Case

The motivation for flooding of the subpile room was to reduce thermal and hydraulic containment loading from noncondensable gas generation during the MCCI event. Furthermore, the reduced combustible gas generation (i.e., H_2 and CO) would contribute to decreasing the probability and severity of the combustion event including deflagration and detonation. As can be seen in Table 2, however, the MCCI with flooding yields a higher source term than the dry MCCI case. In the flooding case, more driving force exists to increase pressure difference between the containment and the environment as the subpile room water vaporizes. With intact containment and single- (primary) containment failure, however, the flooded MCCI case shows much less halogen release. This is because steam generation in the containment provides additional aerosol surface for iodine deposition. Another difference lies in the containment pressurization. Without the steaming source, the dry MCCI case shows that the high-bay area pressure remains below 115 kPa, and thus the rupture disk between the high-bay area and the experimental/beam room stays intact. However, as seen in Fig. 3, the subpile room flooding case shows that steaming in the subpile room results in rupture disk failure, allowing radionuclide transport into the experimental/beam room where scientists work. Flooding contributes to reduction of the subpile room temperature significantly (i.e., from a few thousands for the dry case to about 1000 K for the flooding case). Nevertheless, the idea of flooding the subpile room does not seem to be viable unless gas generation and associated combustion-related loading from the MCCI with flooding is also reduced substantially to an amount small enough to compensate for pressure buildup due to steaming. As seen in Fig. 2, flooding over the limestone–common sand concrete does not seem to be very effective in reducing combustible gas generation (i.e., H_2 and CO). An independent study, however, shows that the flooding the subpile room effectively reduces gas generation from the MCCI if alumina-concrete is used as a basemat material.[7]

V. SUMMARY AND CONCLUSION

To summarize, this paper has provided conservatively scoped estimates for source terms arising from the MCCI event in the flooded subpile room for three different containment configurations. For the intact containment configuration, source term magnitude is very small primarily because of tightness of the dual-containment configuration. Even with primary-containment failure, the source term is predicted to be very small as long as the secondary concrete containment remains intact because of the increased surface area for radionuclide deposition and heat transfer to the outside of the containment. The idea of intentionally flooding the subpile room during the MCCI event is being analyzed further in an integral manner (i.e., including impact on structures, explosions, etc.) because it leads to a larger source term. This increase is attributed to increased hydrodynamic driving force caused by steaming. As mentioned previously, flooding of the subpile room as a consequence of core melt progression may also be credible possibility with the existing design.

As a cautionary note, it should be recognized that severe accidents coupled with early containment failure in the ANS are very unlikely events. Preliminary PRA scoping studies indicate probability levels of about 10^{-8}/year for the MCCI event with early primary and secondary containment failure.

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Table 1. Severe accident characteristics of the ANS and other reactor systems

Parameter	Commercial LWR	HFIR	ANS
Power, MW(t)	2600	85	330
Fuel	UO_2	U_3O_8–Al	U_3Si_2–Al
Enrichment (m/o)	2–5	93	93
Fuel cladding	Zircaloy	Al	Al
Coolant/moderator	H_2O	H_2O	D_2O
Coolant outlet temperature, °C	318	69	92
Average power density, MW/ L	<0.1	1.7	4.5
Clad melting temperature, °C	1850	580	580
Hydrogen generation potential, kg	850	10	12

Table 2. Fission product fractonal mass released from ANS containment

Containment			Fission Product Released into Environment (%)			
Primary	**Secondary**	**Subpile Room**	**Noble Gases**	**Halogen**	**Cesium**	**Tellurium**
Intact	Intact	Flooded	0.1	4.71E-04	8.85E-05	8.91E-05
		Dry	4.93E-02	3.92E-05	5.58E-05	4.56E-04
Failed	Intact	Flooded	9.63	1.26	3.35	3.37
		Dry	4.81	1.36	1.62	1.62
Failed	Failed	Flooded	20.14	19.75	15.33	15.43
		Dry	10.5	9.95	10.1	10.1

Note : Numbers for the dry subpile room cases are extrapolated from 20 to 72 h.

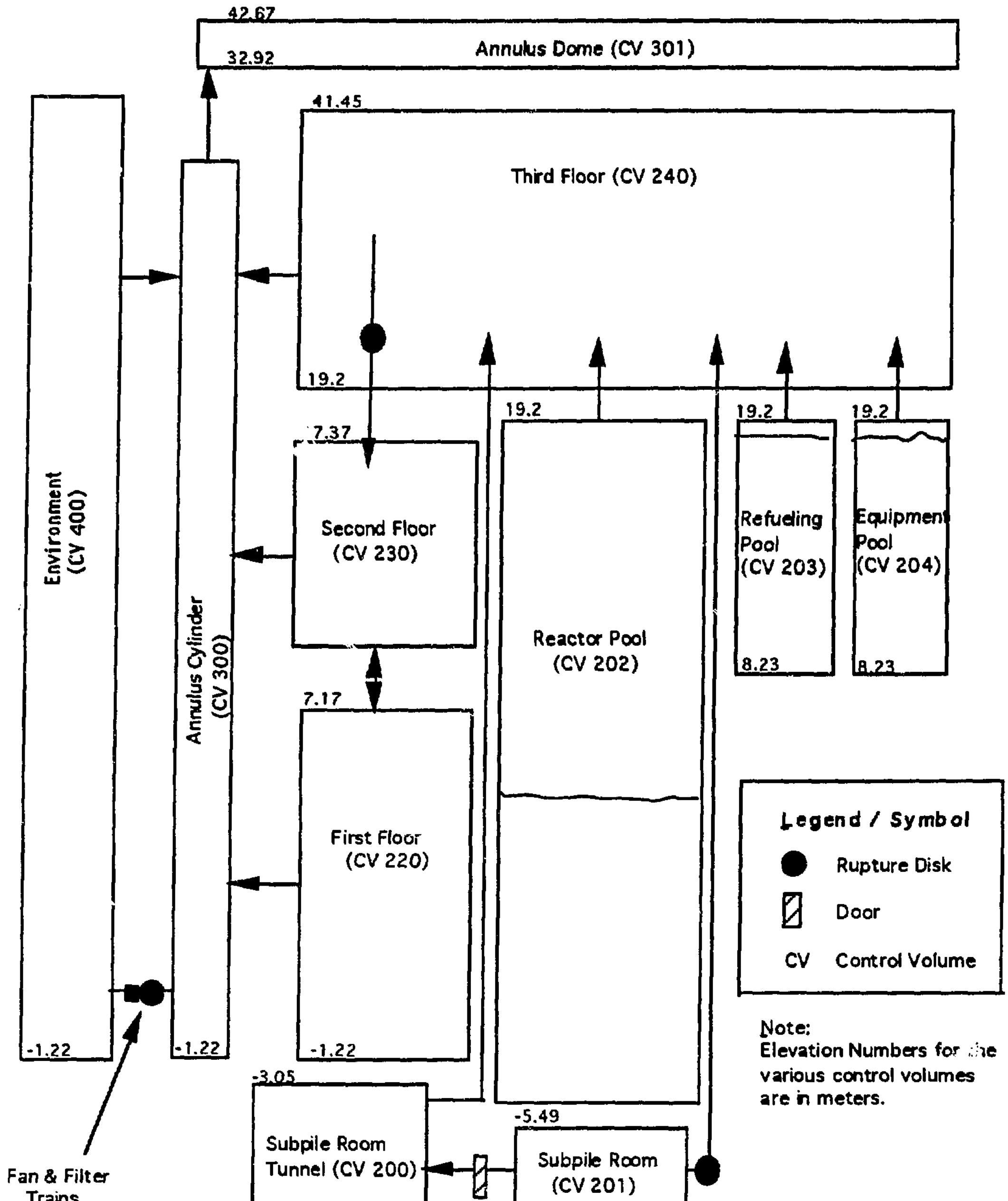

Figure 1 ANS Containment (MELCOR) Representation

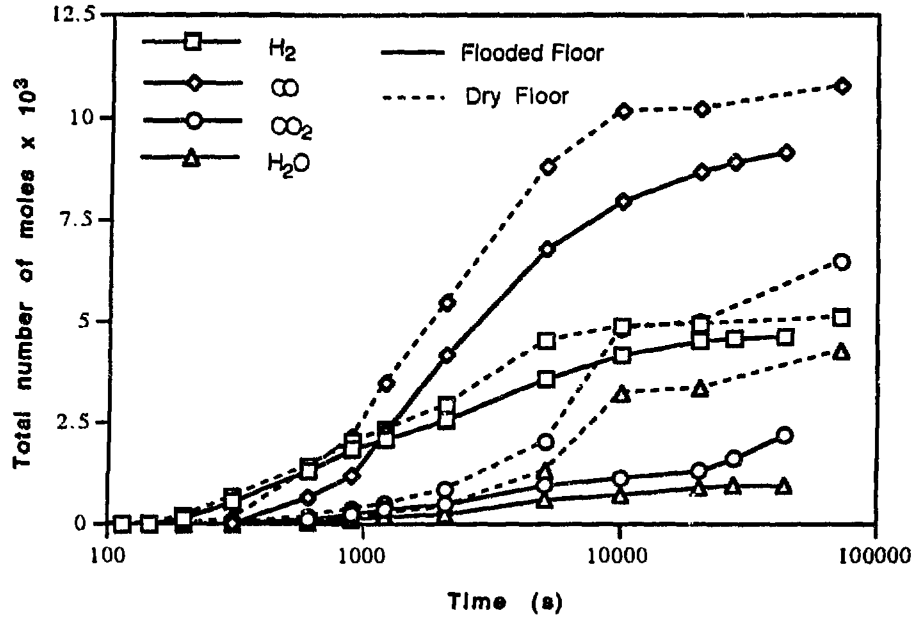

Figure 2 Gas Generation from MCCI with Flooding

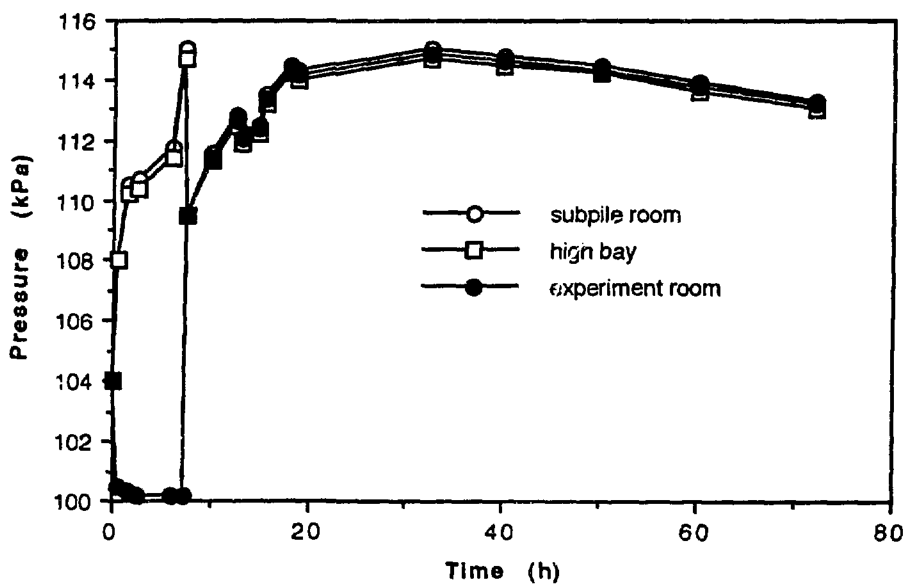

Figure 3 Pressure Variations for Flooded MCCI with Intact Containment

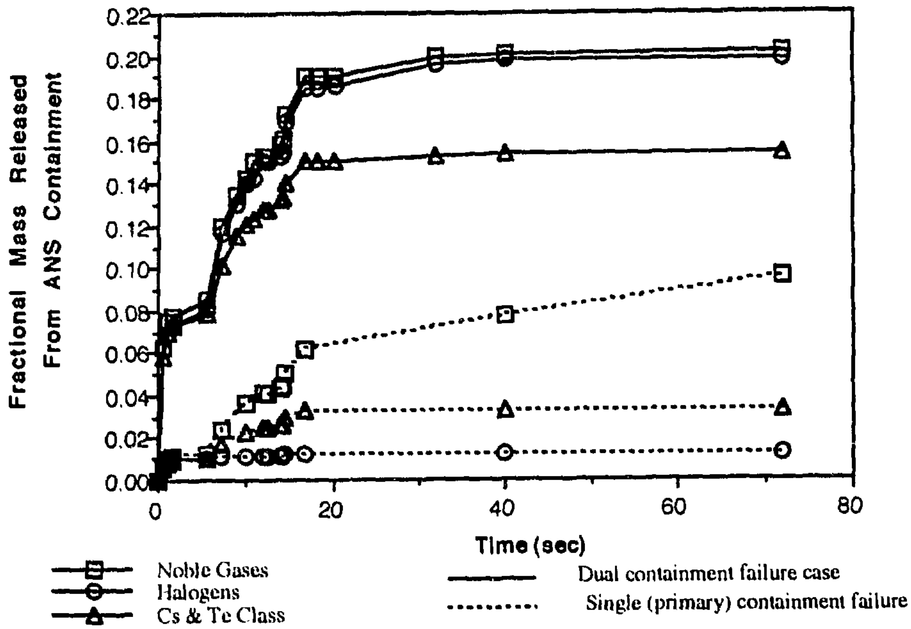

Figure 4 Fractional Radionuclide Mass Released from ANS Containment for Flooded MCCI Event with Early Containment Failure