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CONTAINMENT PERFORMANCE ANALYSIS OF THE ADVANCED NEUTRON SOURCE REACTOR AT THE OAK RIDGE NATIONAL LABORATORY

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ABSTRACT

This paper discusses salient aspects of methodology, assumptions, and modeling of various features related to estimation of source terms from two conservatively scoped severe accident scenarios in the Advanced Neutron Source (ANS) reactor at the Oak Ridge National Laboratory. Various containment configurations are considered for steaming-pool-type accidents and an accident involving molten core-concrete interaction. Several design features (such as rupture disks) are examined to study containment response during postulated severe accidents. 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 for each scenario, which are to be used for studying off-site radiological consequences and health effects for these postulated severe accidents. Also highlighted will be a comparison of source terms estimated by two different versions of the MELCOR code.

I. INTRODUCTION

The 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 purpose of the reactor will be condensed matter physics, materials science, isotope production, and fundamental physics research.[1,2] ANS is planned to be a 330-MW research reactor that uses U_3Si_2–Al cermet fuel in a plate-type configuration. 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-on in the design phase. This was done to aid in designing 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 Levels-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 is being 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. An ancillary purpose is to highlight differences in predictions from using two different versions of the MELCOR code. 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, conservatively scoped scenarios were 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 for scenarios analyzed and reported in this paper. In addition, containment response for an intact containment configuration is analyzed. Modeling and specific analysis results for two of these scenarios are described in this paper.

II. DESCRIPTION OF ANS SYSTEM DESIGN

The ANS is currently in the conceptual design stage. As such, design features of the containment and reactor

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

systems are evolving, based on insights from ongoing 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, as opposed 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 so-called core pressure boundary tube and enveloped in a reflector vessel. This reactor system 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 primary containment of the ANS consists of 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); whereas, 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 scenarios postulated in this study, modeling of the ANS containment thermal-hydraulic analysis, and a radionuclide retention and transport study of containment.

A. Description of Severe Accident Scenarios

Because the ANS is in the preliminary stage of severe accident technology development, it has not been possible to develop mechanistic tools for capturing core-melt progression phenomena. Two severe accident scenarios are postulated for this study with a view toward evaluating conservatively estimated source terms. The first scenario (SC-1) evaluates maximum possible steaming loads and associated radionuclide transport. The second scenario (SC-2) is designed to evaluate maximal containment loads from the release of radionuclide vapors and aerosols and the associated generation of combustible gases.

1. SC-1: Severe Accident Steaming Event. The evaluation of loads from steaming events during severe accidents is modeled along the lines of the Nuclear Regulatory Commission's guidance for power reactors[5] and will be called Scenario 1 (SC-1). The core debris for this case is assumed to be confined within a 100-m^3 volume of water. At the beginning of the calculations, it is assumed that a partitioning of fission products has occurred. All noble gases and 50% of the halogen inventory are assumed to escape from the water and move directly into the atmosphere of the primary containment high-bay area. The balance of the radionuclides would remain behind and cause the water to boil. This prescription would be characterized

as conservative, because no time-span allowance is made for core material degradation, relocation, fission-product release, and possible retention. Also, the prescription does not take into account iodine removal caused by scrubbing as iodine passes through the large reactor pool in the ANS. However, the prescription does represent a conservative guide for evaluating source terms in the absence of mechanistic melt progression analysis and has a long history of similar usage[5] for the power reactor licensing process. For the maximum possible source-term estimate, failure of both primary and secondary containment is assumed to exist in the third-floor high-bay region as the initial condition (SC-1A) . Therefore, this failure allows a direct pathway of radionuclides from the high-bay region to the environment. Intact containment is another case of the current study to determine a containment response to maximum steaming load (SC-1B).

2. SC-2: Molten-Core-Concrete Interaction (MCCI) Event. 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 becoming relocated outside of the primary system and onto a concrete surface. MCCI events can release large amounts of combustible gases (CO and 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 onto the concrete floor of the subpile room. Thereafter, core debris would spread, and an MCCI event would begin. The scenario postulated for the current study conservatively assumes that core debris would relocate at 50 s after reactor scram onto a dry concrete floor in the 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. Additional conservatism is factored into the scenario through the assumption that none of the more than 100 m^3 of heavy water from the primary coolant system would relocate through the same breach (as the core debris) into the subpile room. As assumed for Scenario 1 both configurations of containment are analyzed for Scenario 2, viz., early containment failure (SC-2A) and the intact containment (SC-2B) case.

B. MELCOR Modeling of ANS Containment

The MELCOR severe accident analysis code (Version 1.8.1) was used to develop an overall representation of 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 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 also are modeled, as are several complex aerosol and vapor transport phenomena associated with various severe accident scenarios. Fission-product inventory and its associated decay heat have been calculated using the ORIGEN2 code for the ANS core-averaged end-of-cycle, assuming a 17-day core life at an operating power level of 330 MW.

For the steaming-pool case (SC-1), all noble gases and 50% of iodine inventory (in vapor form) initially are sourced into the high-bay region at the start of the calculations. As the reactor pool is heated to saturation because of decay heating of the rest of the fission products, cesium and tellurium are assumed to be released at a rate proportional to the steaming rate. Cesium is modeled as being in hydroxide form (i.e., CsOH). The remaining iodine release (i.e., the other 50% not released initially) is modeled

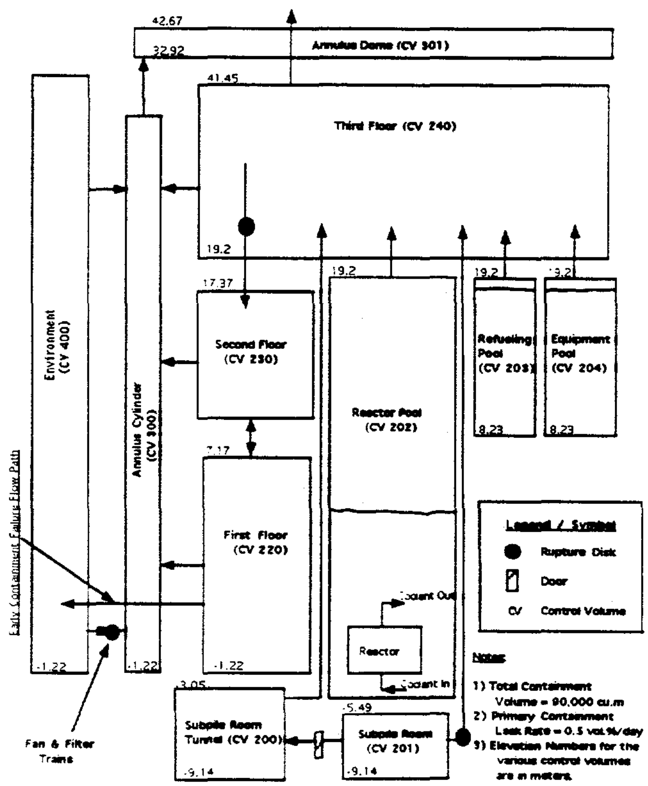

Figure 1. MELCOR representation of ANS containment.

mechanistically. Chemical interactions between radionuclides are neglected, while aerosol formation, deposition, and transport are allowed.

For Scenario 1 cases, it is assumed that, because of some events (e.g., beam-tube rupture), the reactor pool water becomes depleted to the level of the beam tubes. This gives rise to a pool volume of 100 m^3. It is further assumed that pool cooling equipment (for all pools in the high-bay area) does not function.

For the MCCI cases (SC-2), all volatile fission products were 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. The nonvolatile species contribute to the continuation of MCCI and stay in the debris; that is, they are not allowed to volatilize or form aerosols. Fifty percent of the total core decay power is assumed to be associated with nonvolatile fission products. For this study, mass and energy of gases generated from the MCCI are obtained through an independent study[7] and then specified through user input.

For modeling cases with containment failure, upon occurrence of a severe accident, a 0.5-m-diam 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 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.

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-diam 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 existed. Therefore, if oxygen is available there, any combustible gases will be allowed to deflagrate (but not to detonate). The basement of the subpile room is modeled as being made of limestone-common sand concrete.

Rupture disks are in place (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 psi) 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 psi) 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. Comparisons of the results obtained from new (Version 1.8.1) and old (Version 1.8.0) versions of MELCOR also are described.

A. Severe Accident Steaming Event (SC-1)

Key results of interest for the intact containment configuration (SC-1B) are shown in Figs. 2 and 3. Pressurization traces for various regions of containment are shown in Fig. 2. Iodine left in the pool is released into the atmosphere quickly as the pool heats. The reactor pool starts steaming at 4 h, and cesium and tellurium are released at a rate proportional to the steaming rate. As seen in Fig. 2, high-bay volume pressure rises quickly after about 4 h when pool steaming begins (about 50% of the pool steams during 70 h). Thereafter, rupture disks between the high-bay and experiment areas of the first and second floors provide pressure relief when a pressure difference of 115 kPa (2 psi) is reached. Eventually, the entire containment volume pressure levels off at about 120 kPa because of continuing condensation of steam on various structure surfaces in the containment. A mild atmospheric temperature increase of various containment regions is predicted. Specifically, the atmospheric temperature in the high-bay area rises to 335 K (140°F), primarily because of steam condensation and radionuclide deposition on various heat structures. During 70 h of transient duration, about 0.05 kg of radionuclides are predicted to be deposited on the structural surfaces. Deposition seems to keep increasing linearly at about 0.67 g/h. In the first few hours, revaporization of radionuclides deposited on the structures is predicted when surface temperatures of the structures increase and sufficient vapor pressure of a specific radionuclide element is built up. Fractional radionuclide mass released into the environment is shown in Fig. 3. The figure shows that only about 0.1% of the noble gases and $< 6 \times 10^{-4}\%$ of the halogen inventory is released over 70 h. About $10^{-5}\%$ of the cesium and tellurium inventory is released in this time frame.

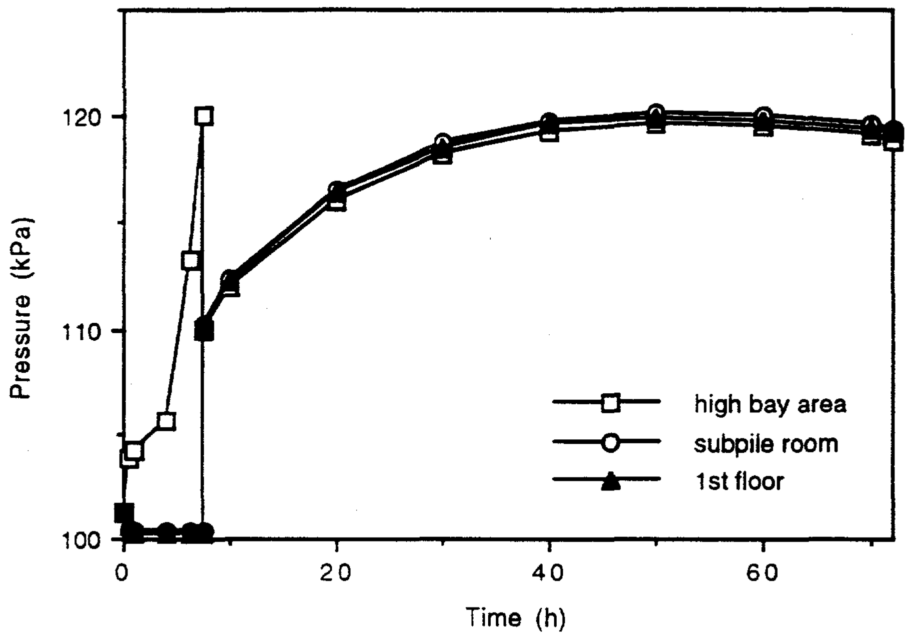

Figure 2. Variations in containment pressures for steaming-pool-type accident with intact containment.

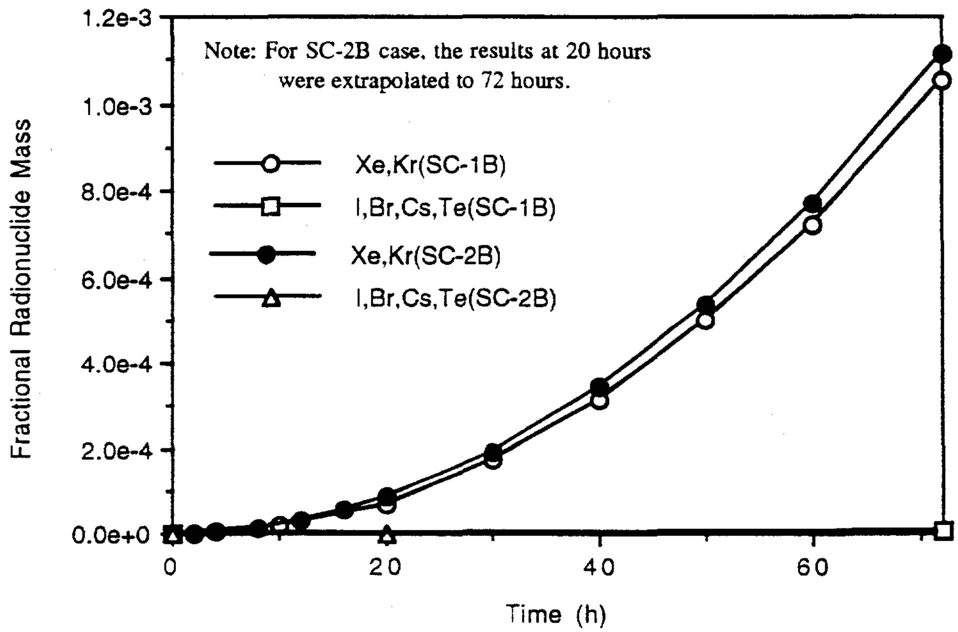

Figure 3 Fractional radionuclide mass released into environment for intact containment cases.

The results of the MELCOR calculations for SC-1A (i.e., steaming-pool case with early containment failure) are shown in Fig. 4. In this case, negligible pressurization results in the various control volumes. The rupture disk leading to the first- and second-floor volumes remains intact, because the high-bay region pressure does not exceed 115 kPa (2 psi). Characteristics of radionuclide deposition onto heat structures are like those for the intact containment configuration (SC-1B). However, because of early containment failure, the total amount deposited is about 20% lower than that seen for SC-1B. The principal difference in results concerns the magnitude of the source term. Figure 4 provides the transient variation of the radionuclides leaving containment (i.e., source term) and entering the environment. A sharp increase in aerosol and vapor mass release to the environment at the onset of steaming and the leveling-off characteristic behavior are seen. As seen in Fig. 4, approximately 28% of the noble gases, about 26% of the halogen inventory, and about 1.6% of the cesium and tellurium inventories are released into the environment.

B. Molten-Core-Concrete Interaction Event (SC-2)

Key results of interest for SC-2B are given in Figs. 3 and 5. As noted in Fig. 5, the subpile room pressure rises rapidly because of the intensity of the MCCI and 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 subpile room tunnel to rise concomitantly. The high-bay region pressure does not exceed 115 kPa (2 psi); hence, the first- and second-floor volumes are not subject to pressurization and radionuclide transport. The short spike in subpile room pressure lasting a few seconds is caused partly by hydrogen and carbon monoxide 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., on the order of a few thousands of degrees Celsius) can result in the subpile room 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. When compared to an equivalent steaming event (SC-1B), about five times more radionuclides are deposited on heat structures. Figure 3 provides the transient variation of the source term. As seen, about 0.009% of the noble gas inventory, about $4 \times 10^{-5}\%$ of the halogen inventory, about $6 \times 10^{-5}\%$ of the cesium-class inventory, and about $5 \times 10^{-4}\%$ of the tellurium-class inventory are released into the environment over 20 h. These 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. No radionuclides enter the first- and second-floor areas.

Results for the MCCI case with early containment failure (SC-2A) are shown in Fig. 4. Variations of important parameters in the subpile room are like those seen for SC-2B. One major difference, which can be expected, deals with the degree of high-bay region pressurization. A very mild pressurization results in the various control volumes, as seen from the containment failure case of steaming event. The high-bay region pressure is well below 115 kPa (2 psi). Consequently, the first- and second-floor areas are not available to receive radionuclide vapors and aerosols. As seen in Fig. 4, about 10.5% of the noble gases, 9.9% of the halogen inventory, and 10% of the cesium and tellurium inventories are released into the environment over 70 h. It should be noted that for the MCCI case (SC-2), most radionuclide releases occur well within the first hour of the start of MCCI. This contrasts sharply with the steaming pool cases described earlier, in which significant releases to the environment occur only after the reactor pool water starts steaming.

C. Comparison Against MELCOR 1.8.0 Results

This section describes comparisons of the results predicted by the new version of MELCOR (Version 1.8.1-HN) against those predicted by the old version (Version 1.8.0). In general, the new version's prediction is close to that of the old version, as far as the amount of steam involved in the radionuclide transport and retention process is limited. For the MCCI event (SC-2), results from both versions of MELCOR agree very well, because the magnitude of steam content in containment in this case is not significant. In the steaming-pool event (SC-1), however, substantial differences are seen, specifically in the transport and retention of radionuclides. At the end of the calculation, a noticeable difference is seen in the amount of iodine source term (26% from the new version vs 8% from the old version for the early containment failure configuration, SC-1A). This difference in results mainly is caused by an error in the old version associated with evaporation and condensation of fission products on various surfaces, viz., aerosol and heat structures. Therefore, caution is advised to users of the MELCOR Version 1.8.0 code for situations involving significant vapor condensation/ evaporation phenomena.

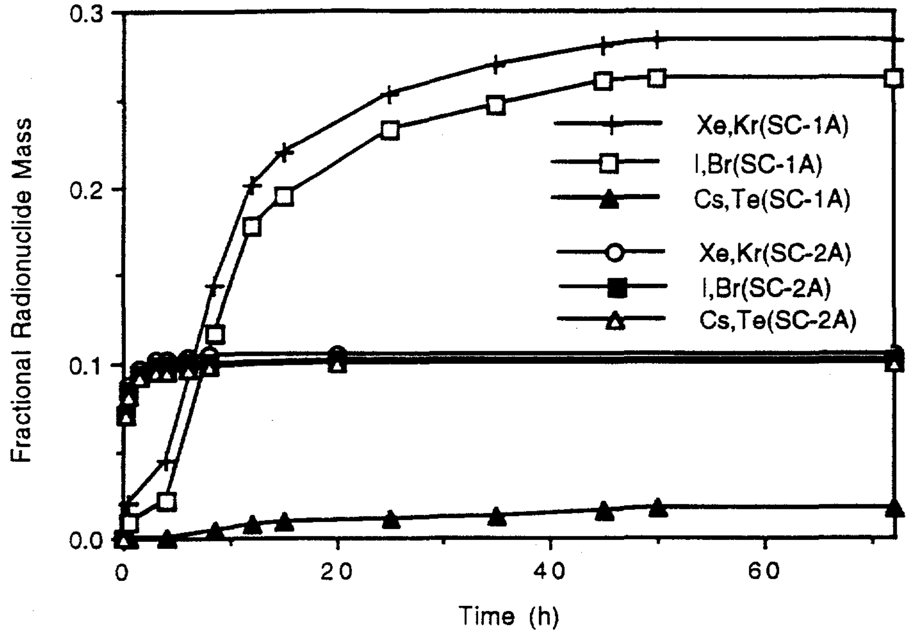

Figure 4. Fractional radionuclide mass released into environment for early containment failure cases.

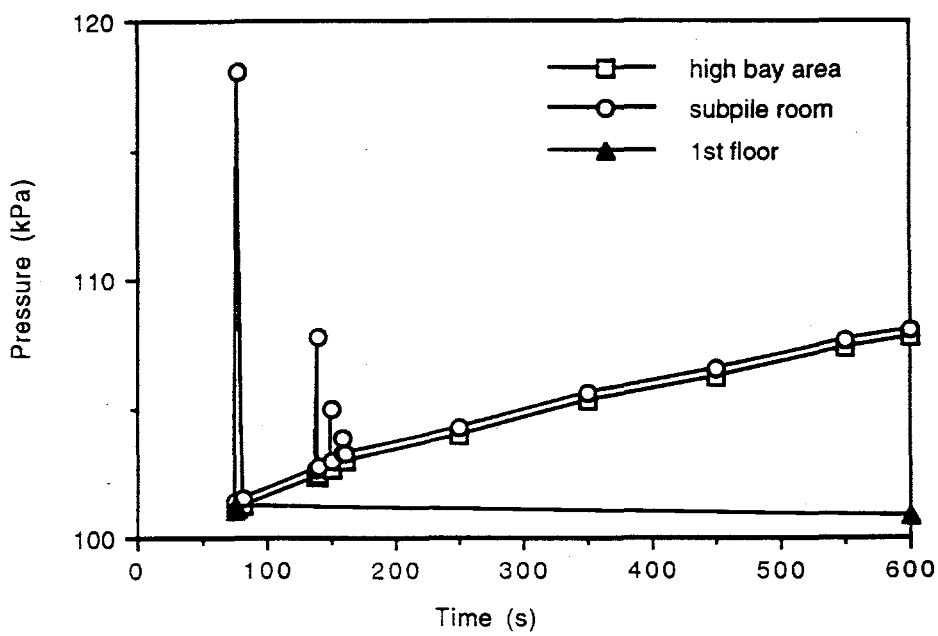

Figure 5. Variations in containment pressures for MCCI accident with intact containment.

V. SUMMARY AND CONCLUSION

To summarize, this paper has provided conservatively scoped estimates for source terms arising from two different severe accident scenarios for two different containment configurations. In addition, potentially erroneous predictions that can arise when using the MELCOR (Version 1.8.0) code have been highlighted. Caution is advised to users of this MELCOR code version for situations involving significant vapor condensation/evaporation phenomena. From the standpoint of severity, Scenario 2 (MCCI event) is expected to dominate in terms of health risks (for ANS), primarily because of the rapidity with which source terms are released to the environment.

As a cautionary note, it should be realized that severe accidents coupled with early containment failure in the ANS are very unlikely events. Preliminary PRA scoping studies indicate probability levels of 2.5×10^{-8}/year for Scenario 1 with early containment failure (SC-1A) and about 10^{-8}/year for Scenario 2 with early containment failure (SC-2A). Nevertheless, these calculations provide bounding estimates of health risk arising from hypothetical severe accidents in the ANS as part of the CSAR and provide insights into the development of mitigative features. Health risks from these postulated severe accidents are described in a companion paper.[8]

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