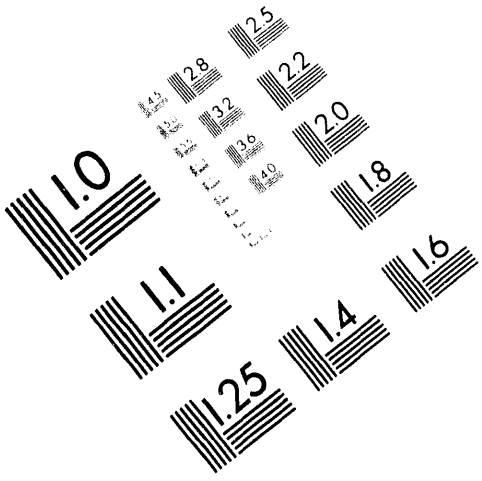


AIIM

Association for Information and Image Management

1100 Wayne Avenue, Suite 1100
Silver Spring, Maryland 20910

301/587-8202

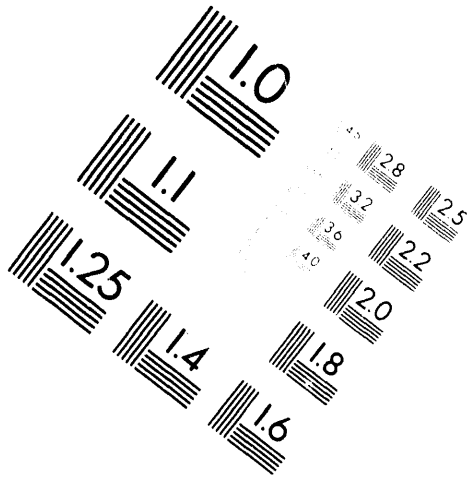
Centimeter

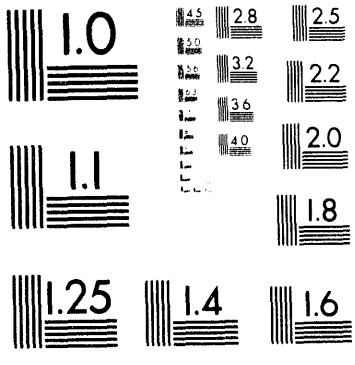
1 2 3 4 5 6 7 8 9 10 11 12 13 14 15 mm

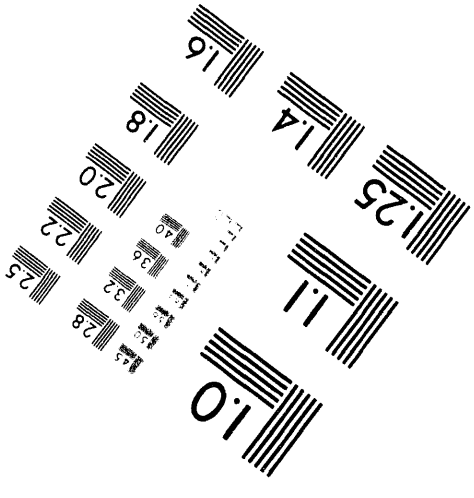
1 2 3 4 5

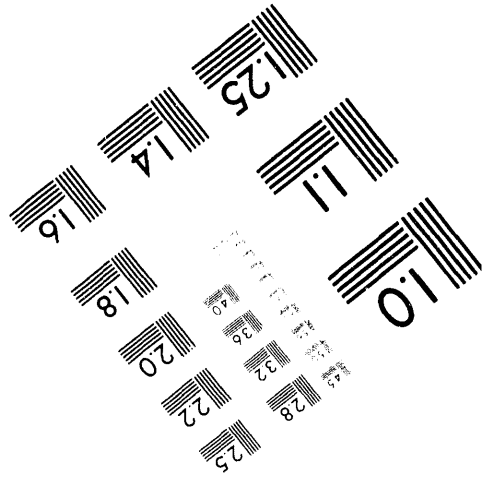
Inches

MANUFACTURED TO AIIM STANDARDS
BY APPLIED IMAGE, INC.









1 of 1

Conf-9303103--3

Modeling & Analysis of Criticality-Induced Severe Accidents During Refueling for the Advanced Neutron Source Reactor*

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

by

V. Georgevich
S. H. Kim
S. Jackson**
R. P. Taleyarkhan

Engineering Technology Division
Oak Ridge National Laboratory***
Oak Ridge, Tennessee, USA

(October, 1992)

*The submitted manuscript has been authored by a contractor of the U. S. Government under contract No. DE-AC05-84OR21400. Accordingly, the U. S. Government retains a nonexclusive, royalty-free license to publish or reproduce the published form of this contribution, or allow others to do so, for U. S. Government purposes."

** Summer intern; current address: University of Missouri, Kansas City, Missouri

*** Research sponsored by the U.S. Department of Energy under Contract No. DE-AC05-84OR21400 with Martin Marietta Energy Systems Inc.

Prepared for Presentation and Publication in Proceedings of Second International Conference on Nuclear Engineering, ICONE-2, March, 1993.

MASTER

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

MODELING AND ANALYSIS OF CRITICALITY-INDUCED SEVERE ACCIDENTS DURING REFUELING FOR THE ADVANCED NEUTRON SOURCE REACTOR

V. GEORGEVICH, S. H. KIM, S. JACKSON,[a] and R. P. TALEYARKHAN
Engineering Technology Division
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37831

ABSTRACT

This paper describes work done at the Oak Ridge National Laboratory (ORNL) for evaluating the potential and resulting consequences of a hypothetical criticality accident during refueling of the 330-MW Advanced Neutron Source (ANS) research reactor. The development of an analytical capability is described. Modeling and problem formulation were conducted using concepts of reactor neutronic theory for determining power level escalation, coupled with ORIGEN and MELCOR code simulations for radionuclide buildup and containment transport. Gaussian plume transport modeling was done for determining off-site radiological consequences. Nuances associated with modeling this blast-type scenario are described. Analysis results for ANS containment response under a variety of postulated scenarios and containment failure modes are presented. It is demonstrated that individuals at the reactor site boundary will not receive doses beyond regulatory limits for any of the containment configurations studied.

I. INTRODUCTION

ORNL's ANS will be a new facility[1] for all kinds of neutron research centered around a research reactor of unprecedented ($\sim 10^{20}$ m^{-2}-s^{-1}) neutron beam flux. A defense-in-depth philosophy has been adopted. In response to this commitment, ANS Project management has initiated severe accident analysis 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 such an accident. It also provides a means for satisfying on- and off-site regulatory requirements and accident-related dose exposures and for containment response and source-term best-estimate analyses for the Level-2 and -3 Probabilistic Risk Analyses 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.

This paper describes salient aspects of the work done to date on addressing a potentially important severe accident issue dealing with recriticality-induced severe accidents during refueling.

A. ANS System Design

The ANS is currently in the conceptual design stage. As such, design features of the containment and reactor system 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 to ORNL's High Flux Isotope Reactor[2] and a commercial light water reactor (LWR). As seen in Table 1, high power density research reactors can give rise to significantly different severe accident issues. Specifically, the ANS reactor will use about 15 kg of highly enriched (~93 m/o ^{235}U) uranium silicide fuel in an aluminum matrix with a plate-type geometry and a total core mass of 100 kg. Heavy water (D_2O) is used as moderator and coolant. Power density of the ANS will be about 50 to 100 times higher than that of a large LWR. A schematic representation of the reactor containment configuration is given in Fig. 1. The reactor core is enclosed within a core pressure boundary tube (CPBT) enveloped in a reflector tank and located at the bottom of a large reactor pool. The high-bay area is separated from first- and second-floor

[a]Summer intern; current address: University of Missouri, Kansas City, Missouri.

experiments and beam rooms via a rupture disk that opens to under a pressure differential of 114 kPa (2 psi).

B. Importance of Criticality Issue for ANS

Criticality-induced severe accidents are highly undesirable because they may lead to containment-damaging steam explosion loads and radiation exposure to on- and off-site individuals. The scoping study of a hypothetical, recriticality-induced severe accident in the ANS was motivated by the need to gage its potential for damage and by the need to consider designed mitigative features early on in the design process.

II. MODELING AND PROBLEM FORMULATION

This section describes salient aspects of modeling and problem formulation for the postulated refueling accident. Descriptions for various aspects are given subsequently.

A. Scenario Description

The refueling accident scenario under consideration assumes that a series of unlikely events occur during a refueling event. The entire reactor core is composed of two different diameter annular fuel elements (each consisting of several hundred fuel plates). The k_{eff} values[3] for unirradiated outer and inner fuel elements in an infinite D_2O medium are about 1.2 and 1.1, respectively. Fresh fuel was chosen for this analysis because spent fuel has considerably less potential for going critical. Ordinarily, refueling involves inserting each fuel element separately through a D_2O-filled transfer shaft accompanied by sufficient neutron-absorbing materials to keep the elements significantly subcritical. The accident under consideration postulates that, by some hypothetical means, the outer fuel element (consisting of about 63% of entire core fuel plates) becomes separated from the neutron poison and therefore undergoes a prompt critical power excursion. The purpose of this paper is to present a preliminary analysis for the resulting power excursion, determine the energetics of the fuel-coolant interaction (FCI), evaluate the containment thermal-hydraulic response, evaluate the resulting source terms to the environment, and conduct a scoping study of off-site radiological consequences.

B. Preliminary Modeling of Power Excursion, and FCI Energetics

Power excursions during a recriticality event are governed by several competing effects. Positive reactivity feedback from the moderator causes an exponential power rise. Simultaneously, fuel temperature increases and moderator voiding introduces negative reactivity insertion. Core reactivity also decreases because of Doppler broadening but should not be a significant effect for ANS fuel, which is 93% enriched ^{235}U. Fuel geometry changes also can cause a change in criticality. Apriori, this effect is very difficult to judge, because it is unclear what level of dispersion or geometrical change will occur under such conditions.

The rate of power generation will, in essence, depend on the previously mentioned competing nuclear-cum-thermal hydraulic effects, coupled with the existence of an inertial constraint. Inertial constraint provides the essential retention mechanism for power escalation by preventing expulsion of the fuel-cum-moderator mixture from a super-critical configuration. Clearly, a critical configuration cannot be sustained in the absence of a moderator or in the absence of a critical mass of fuel. A mechanistic core-melt-progression capability has not been developed yet for the ANS. In the absence of such a capability, it was conservatively assumed that the power excursion would stop when sufficient energy is generated to cause vaporization of the aluminum in the affected ANS fuel element. Sufficient pressure buildup at this stage should, as experienced in the SL-1 incident[4], cause expulsion of the dispersed core debris and moderator out of the open (i.e., at the top) fuel transfer channel into the high-bay atmosphere. At temperatures considerably lower than the aluminum vaporization temperature of 2400°K (i.e., around 1500°K), aluminum can react exothermically[5] with water, generating about 17 MJ/kg of aluminum consumed. Under such conditions, explosive FCIs also can occur and cause considerable debris dispersion and high-pressure buildup. Therefore, these mechanisms would permit shutdown of the nuclear excursion.

The time it takes for total power to cause aluminum vaporization is obtained using an exponential solution[6] to the point kinetics equation (neglecting delayed neutrons). This is represented by,

$$p(t) = p_o e^{t/T} \quad , \tag{1}$$

where p(t) is the power level at a given time t, p_o is the initial power, and T is the reactor period. Integrating Eq. (1) over a given time τ, we obtain the expression for energy generated, E. Therefore, the time it takes to produce a given energy-level buildup can be expressed as

$$\tau = \ln [1 + E/(p_o T)] T \quad , \tag{2}$$

where the period T is conventionally expressed in terms of the neutron diffusion time l_p, and k_{eff} is expressed as

$$T = l_p / (k_{eff} - 1) \quad . \tag{3}$$

For an infinitely dispersed thermal reactor employing D_2O, l_p equals[6] about 0.14 s. This magnitude can be lowered significantly by the state and composition of fuel. The l_p value for ANS-type reactors is usually in the 10^{-3} s range[7,8]. As can be seen from Eq. (3), the reactor period and, therefore, the value for τ will depend on excess reactivity, which may be different depending on the presence of neutron-absorbing materials. The value for τ also will depend on the transient buildup of power to arrive at the required energy level E. However, for a given value for l_p and k_{eff}, one can obtain an approximate value for τ using Eqs. (2) and (3) for various E/p_o ratios.

C. Modeling Fission-Product Buildup and Decay During Blast-Type Accident

The ORIGEN 2.1 fuel depletion code[9] was used for evaluating fission-product buildup during power burst events, as for tracking decay characteristics of the resulting fission products and daughter elements. Fission-product inventory buildup during the blast-type event is calculated using ORIGEN 2.1, wherein a given amount of energy required for aluminum vaporization is specified to be generated over a given time frame. Because of the uncertainty of the exact time scale involved [because of not knowing exact values for the parameters in Eqs. (2) and (3)], a sensitivity study was conducted to evaluate the variation in fission-product inventories from a specified amount of nuclear energy generated between 0.1 to 2 s. Initial variations were noted in the quantities of short-lived isotopes. However, a few minutes thereafter, the concentrations effectively equilibrated to the same value, regardless of initial irradiation time. This indicated that large uncertainties in the value of τ would not lead to significant uncertainties in the resulting containment response and off-site radiological consequences, where the time scale of events ranges from several minutes to hours.

D. Modeling ANS Containment Thermal-Hydraulics and Radionuclide Transport

The MELCOR severe accident analysis code[10] was used to develop an overall representation of ANS containment. The model, consisting of 11 control volumes, 15 flow paths, and 21 heat structures (representing walls, ceilings, shells, and miscellaneous materials) of various shapes, is shown in Fig. 2. A fan model also has been included to account for flow through the large annulus gap between the steel shell and outer containment. Aerosol and vapor filtration processes also are modeled, as are various complex aerosol and vapor transport phenomena associated with the severe accident scenario being evaluated. Iodine and aerosol filter trains have been incorporated to provide retention (of halogens and particulates) with decontamination factors of 100 and 200, respectively. Under containment isolation conditions, the ANS-specific leakage rates of 0.5 vol %/day from primary containment to annulus (under design pressure difference) and 10 vol %/day from annulus to the environment were modeled. Modeling of the annulus leak rate of 10 vol %/day was performed by conducting an inverse calculation. That is, the exhaust rate of 10 vol %/day was specified as a boundary condition, and the resulting pressure distributions in the annulus were back calculated. At the start of the calculation, 100% of the inventories of noble gases, iodine, and tellurium class inventories are sourced into the high-bay-area volume as vapors. The remaining core debris (including aluminum and uranium) is sourced into the high-bay area as aerosols. Such a prescription provides for maximum possible thermal-hydraulic and radionuclide loading into the containment atmosphere.

The MELCOR model shown in Fig. 2 was made flexible enough to allow modeling of intact and failed containment states. Failure is modeled via a 0.5-m-diam opening in containment along with a flow path between the environment and high-bay volumes. The model also allows for specification of various sensible heat loads accompanying the core debris into the containment (in addition to decay heating from fission products). This was done to allow study of various levels of energy deposition from events such as Al–H_2O chemical reactions.

E. Modeling Off-Site Radiological Consequences

Modeling of off-site radionuclide consequences was conducted using the well-known Gaussian plume formula[11] for plume transport. For purposes of the present analysis, it is assumed that the release of radionuclides will not occur at ground level but from a given height. This assumption is considered realistic because a containment failure, if it occurs, would occur in this way because of primary and secondary containment wall breach from interactions with an energetic missile or shock waves. The energetic missile or shock waves are postulated to be caused from the initial, explosive FCI. Transport of radionuclides is modeled under various conservative assumptions of wind speed, stability class, breathing rates, and dose conversion factors for key isotopes.

III. ANALYSIS

The modeling framework presented previously was used to analyze for radionuclide buildup during power excursion, containment response, source-term determination, and off-site radionuclide consequence evaluation. Various aspects are described subsequently.

A. Power Excursion and Energy Buildup

Calculations were conducted to evaluate the amount of energy required for causing the aluminum in the outer fuel element to reach aluminum vaporization conditions. It was found that this value amounts to about 660 MJ. As mentioned earlier, beyond about 1500°K, self-sustained aluminum-water chemical reactions set in, releasing about 17 MJ of energy for every kilogram of aluminum consumed. For the outer element, it was determined that about 931 MJ of energy may be released if all of the aluminum reacts with water. As discussed previously, under these conditions, the resulting pressure buildup from an explosive FCI very likely would disperse the core and shut down the nuclear reaction. Therefore, the maximum amount of energy that can be liberated during a refueling accident involving the outer ANS fuel element is about 1.6 GJ. One may hypothesize conditions whereby the nuclear reaction may continue because of a strong enough inertial constraint, providing for a larger energy source. However, an energy release level larger than 1.6 GJ is deemed unrealistic.

B. Buildup of Fission Products During Power Burst Accident

The time scale for excursion τ is impossible to evaluate precisely without mechanistic modeling of power escalation from a given initial power level p_o. Therefore, a series of scoping calculations were conducted with different values of p_o in Eq (2). For a k_{eff} value of 1.2 and a prompt neutron generation time of 10^{-3} s (viz., value for thermalization in ANS core), we obtain a reactor period of 5×10^{-3} s. Considering the long diffusion length for neutrons in D_2O and the fact that the transfer chute for the fuel element is surrounded on the outside with light water, this value for the reactor period may be somewhat lower in reality. However, assuming a 5×10^{-3} s reactor period, it was found that excursion time τ varies between 0.12 s to about 0.2 s for p_o varying between 1 W to 10^{-6} W. Separately, sensitivity studies were conducted with ORIGEN 2.1 code to deplete the upper element over a range of irradiation times varying from 0.1 s to several seconds. These sensitivity studies indicated that for irradiation times ranging in the few seconds, the radioisotope buildup characteristics and level had essentially converged to the same value in ~30 s. So long as the total number of fissions remained the same, irradiation over a few seconds produced the same isotope buildup characteristics as irradiation for less than a second. Thereafter, equilibrium concentration levels for important isotopes (e.g., ^{131}I, ^{90}Sr) are reached in ~30 min. It is to be realized that MELCOR is not capable of simulating shock wave physics-related phenomena. Very rapid energy deposition into the ANS containment atmosphere caused numerical difficulties. Again, due to void formation during the power burst accident, the actual time associated with the power excursion coupled with the time over which thermal energy is deposited into the ANS containment atmosphere can be expected to be longer than 0.1 to 0.2 seconds. Due to these reasons, and also because the precise irradiation time during the power burst does not cause significantly different isotope buildup over longer durations, the ORIGEN 2.1 code was used to deplete the upper element at a constant power level of 330 MW for 2 s.

C. Containment Response Analysis and Source-Term Evaluation

The MELCOR model of Fig. 2, and results of ORIGEN 2.1 calculations for fission-product inventory were used in conjunction with energy-level evaluations to perform containment response calculations. The test matrix of MELCOR calculations is shown in Table 2. As seen from Table 2, four different cases are analyzed. These cases are essentially combinations of two different energy levels (viz., with and without aluminum ignition) and two different containment configurations (viz., intact and failed).

Because of the relatively sudden (i.e., over 2 s) insertion of the large amounts of energy into the containment high-bay area, calculations for all of the cases showed that the high-bay volume pressures rise rapidly. Interestingly, for both intact and failed containment configurations, the pressure buildups amounted to ~106 kPa for the case without ignition and to ~113 kPa for the case with aluminum ignition. Figure 3 provides a sample pressure trace for containment pressure buildup for the case with aluminum ignition and an intact containment. These results indicated that, neglecting shock wave interaction phenomena, the pressure buildup in the high bay would not be beyond 115 kPa, which is the level required for rupture disk failure. These calculations also indicate that due to the rapidity of the event, the existence of a 0.5-m-diam venting of the ANS containment will not play a major role in reducing the pressure buildup magnitude. This is essentially due to the large overall volume of the ANS containment, whereby a 0.5-m-diam opening is similar to a pencil hole on a standard sheet of paper. Thereafter, equilibrium pressures are reached within 3 min for the case without aluminum ignition and within 4 s for the case with ignition.

During the 30 min of transient simulation, containment pressures were noted to decrease from 106 kPa to 102 kPa for the case without ignition and from 113 kPa to 103 kPa for the case with ignition. During this same time, it was found that the containment air temperature builds up to 311 K for the case without ignition and to 330 K with ignition. Thereafter, containment air cools down slowly as structural components absorb atmospheric energy.

An important feature of the MELCOR simulation deals with the treatment of isotopes and their related decay heats. From a study of the ORIGEN 2.1 results, it was found that due to the blast effect (i.e., short irradiation time) many of the isotopes immediately generated are neutron-rich and highly unstable. The majority of these have half-lives ranging from a few seconds to tens of seconds. Radionuclide class transformations take place. As structured, class transformation of individual elements is not explicitly permitted in MELCOR simulations of severe accidents. While this is considered a reasonable assumption for normal severe accident simulations involving time scales of several hours, this limitation presented a technical problem for simulating blast effects. A detailed study was conducted to note the class transformation behavior in relation to the ultimate source term predicted by MELCOR (without allowing for class transformations). The study indicated that a majority of radionuclide elements transformed from one volatile class to another and from one nonvolatile class to another. It was further found that the source-term transport behavior of all volatile classes was similar. The same pattern was observed for nonvolatile classes. This is seen clearly in Fig. 4 where the source term for a case involving containment failure coupled with aluminum ignition is presented. Because thermal-hydraulic transport behavior is governed by the decay heat of elements depending on their volatility, it was surmised that class transformations of the type noted from the ORIGEN 2.1 calculations should not significantly affect the ultimate source term predicted by MELCOR.

Radionuclide release to the environment follows containment-pressure buildup and decrease behavior. For failed containment cases, most radionuclide release occurs within the first 3 min, because containment pressure essentially equilibrates with environment pressure thereafter. Overall, for failed containment configuration cases, about 5.8% and 2.8% of the initial radioactive inventories (including aluminum and uranium) are released for the cases with and without ignition, respectively. The rate of radionuclide release is markedly different for intact containment configurations because of gradual containment-pressure decrease. Herein, the source-term release is continuous over the simulation duration. Radionuclide release fractions are negligibly small. For the case without aluminum ignition, fractional amounts of 2.7×10^{-8} noble gases and about 1.3×10^{-10} of other class element inventories are released. Again, for the case with aluminum ignition, fractional amounts of 4×10^{-8} for noble gases and 2×10^{-10} of other class element inventories are released. Such low values essentially are caused by the leak-tight nature of ANS containment (under isolation conditions).

D. Radionuclide Health Consequences at ANS Site Boundary

The source-term information generated was used in conjunction with the Gaussian plume model to evaluate reactor site-boundary doses from various radionuclides. As mentioned previously, most of the isotopes immediately produced during the accident are neutron-rich. These fission products are not in equilibrium. As an example, iodine concentration buildup as a function of time is shown in Fig. 5. As seen, peak concentration of the important ^{131}I isotope builds up only after about 30 min. Similar behavior was noted for cesium and strontium isotopes. Because of the significant class transformation explained earlier and unavailability of dose conversion factors for these neutron-rich isotopes, codes such as MACCS (developed for conventional severe accident analysis) were impossible to use. For example, MACCS only provides for one parent-daughter transformation and has built-in dose conversion information for only 60 isotopes, none of which are the neutron-rich isotopes in question.

Note that it also is impossible to capture transient variations in radionuclide transformations with the standard Gaussian plume model. To obtain conservative estimates for site-boundary doses, peak concentration values for the important iodine, cesium, and strontium isotopes were assumed to exist in equilibrium during the entire course of consequence evaluation (viz., 24 h). Calculations have been conducted making best-estimate assumptions, as well as worst-case assumptions, for determining wind speed and dispersion parameters σ_y and σ_z, respectively.

Table 3 provides estimates for computed dose rates for individuals at the site boundary under the assumption of release at ground level or at 15 m above ground level (which corresponds to the top of the containment dome). As can be seen, iodine elements dominate the dose levels received by an individual located at the site boundary. Doses from strontium or cesium isotopes are an order of magnitude lower than from iodines under comparable conditions. As expected, ground-level releases also are an order of magnitude higher than if the release occurred at a 15-m height. Even in the worst of circumstances, maximum thyroid dose over the duration of release at the site

boundary does not exceed 0.23 Sv (23 rem). This value is an order of magnitude lower than the allowable 10 CFR 100 limit[12] of 3 Sv (300 rem). A wide margin of safety exists with respect to radiological consequences at the site boundary. This can be attributed principally to the combined aspects of low initial inventory of fission-product buildup, large containment volume, and the resulting small amount (6%) of total inventory that is released, even under the assumptions of containment failure and aluminum ignition.

IV. SUMMARY AND CONCLUSION

This paper has provided an overview of modeling and analysis of a hypothetical severe accident caused by criticality of a fuel element in the ANS core. Modeling and analyses for the resulting power excursion, energetics of an FCI event, containment thermal-hydraulic response, radionuclide transport behavior, and health consequences at the ANS-site boundary have been described.

It was shown that, depending on the occurrence of aluminum ignition, total energy generated during the refueling accident would amount to either 660 MJ (without ignition) or about 1.6 GJ (with ignition). Preliminary calculations for the dynamics of power excursion indicate that nuclear reactivity excursion would occur very rapidly (in less than a second) before shutdown caused by dispersion of moderator and/or fuel debris. ORIGEN 2.1 calculations for determining fission-product inventory buildup indicated that significant quantities of neutron-rich isotopes would be formed initially. These neutron-rich isotopes would decay into the important iodine-, strontium-, and cesium-class isotopes in about 30 min. MELCOR containment response calculations indicated that sudden release of energy from the recriticality event would cause a step jump in containment high-bay-area pressure. Pressure levels up to 113 kPa would be reached for the case where aluminum ignition occurs. This level is lower than the 115 kPa necessary for rupture disks to open between the high bay and experimental rooms.

The cases with an intact containment led to a negligibly small source term. For the case where a missile-generated containment failure may have occurred, source terms for the cases with and without aluminum ignition amounted to about 3% and 6% of the fission product inventory, respectively. Radiological consequence modeling was done using the Gaussian plume model under worst-case and best-estimate assumptions. It was found that, even under the worst of circumstances, the dose given to an individual at the reactor site boundary would be several times lower than that allowed by the site-suitability limit criteria outlined in 10 CFR 100 guidelines.

REFERENCES

1. C. D. WEST, "The Advanced Neutron Source: A New Reactor-Based Facility for Neutron Research," *Transactions of the American Nuclear Society*, **61**, 375 (June 1990).

2. F. T. BINFORD, and E. N. CRAMER, *The High Flux Isotope Reactor, A Functional Description*, ORNL-3572, Oak Ridge National Laboratory (Rev. 2) (June 1968).

3. T. E. VALENTINE and J. T. MIHALZCO, "Subcriticality of Advanced Neutron Source Reactor Fuel Elements During Refueling," *Transactions of the American Nuclear Society, 1992 Annual Winter Meeting, Chicago, Illinois, November 1992.*

4. *Final Report of SL-1 Recovery Operation, May 1961–July 1962*, IDO-19311 (July 1962).

5. L. BAKER Jr., "Metal-Water Reactions," *Nuclear Safety*, 7 (Fall 1965).

6. S. GLASSTONE and A. SESONSKE, *Nuclear Reactor Engineering*, 3rd Ed., Van Nostrand Reinhold Company (1981).

7. F. C. DIFILIPPO et al., "Reactor Physics Calculations forthe Control of the Advanced Neutron Source Reactor," *Proceedings of the International Reactor Physics Conference, American Nuclear Society, September 18-21, 1986, Jackson Hole, Wyoming.*

8. J. M. HENDRIE, *Final Safety Analysis Report on the Brookhaven High Flux Isotope Research Reactor*, Brookhaven National Laboratory Report, April, 1964.

9. A. G. CROFF, *A User's Manual for the ORIGEN2 Computer Code*, ORNL/TM-7175, Oak Ridge National Laboratory (July 1980).

10. R. M. SUMMERS et al., *MELCOR 1.8.0: A Computer Code for Nuclear Reactor Severe Accident Source Terms and Risk Assessment Analyses*, NUREG/CR-5531 (January 1991).

11. D. H. SLADE (Ed.), *Meteorology and Atomic Energy*, USAEC Report TID-21490, Environmental Science and Services Administration (July 1968).

12. J. J. DINUNNO et al., *Calculation of Distance Factors for Power and Test Reactor Sites*, Atomic Energy Commission Technical Information Document TID-14844 (March 1963).

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

Parameter	Commercial LWR	HFIR	ANS
Power, MW(t)	2600	85	300
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	85
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. Summary of source-term calculations and calculational test matrix

Containment configuration	Aluminum ignition	Peak pressure (kPa)	Peak temperature (K)	Release fraction of fission products	
				Noble gases (%)	Others (%)
Failed	No	106	311	2.8	2.8
Failed	Yes	113	330	5.8	5.8
Intact	No	106	311	2.7×10^{-8}	1.3×10^{-10}
Intact	Yes	113	330	4.0×10^{-8}	2.0×10^{-10}

Table 3. Dose at site boundary (180-m radius) for containment failure with aluminum ignition case

	Release height = 0 m		Release height = 15 m	
Release height	Base case	Worst case	Base case	Worst case
Pasquill stability class	D	F	D	F
σ_y, m	12	8	12	12
σ_z, m	9	3.2	9	9
Wind speed, m/s	6	1	6	1
Iodine dose, rem	0.882	22.326	0.22	1.32
Strontium dose, rem	0.027	0.689	0.007	0.042
Cesium dose, rem	0.0053	0.134	0.0013	0.0078
Total dose, rem	0.9143	23.149	0.2283	1.3698

Note: D = neutral stability class
F = moderate stability class
σ_y = lateral dispersion parameter
σ_z = vertical dispersion parameter

For stack release, worst case corresponds to neutral stability class caused by combination with exponential term in the Gaussian plume model equation.

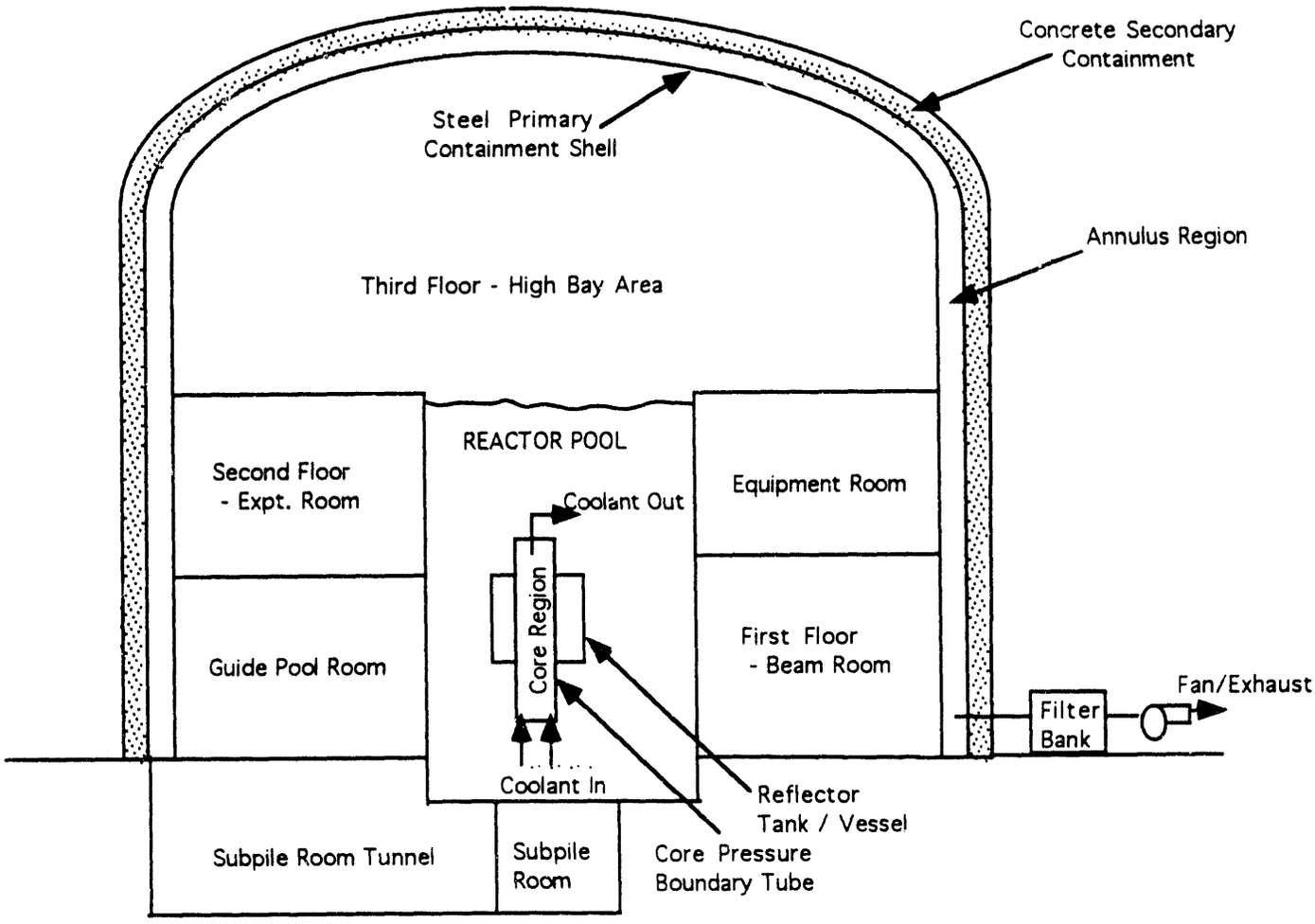


Figure 1. Schematic Representation of ANS Reactor & Containment

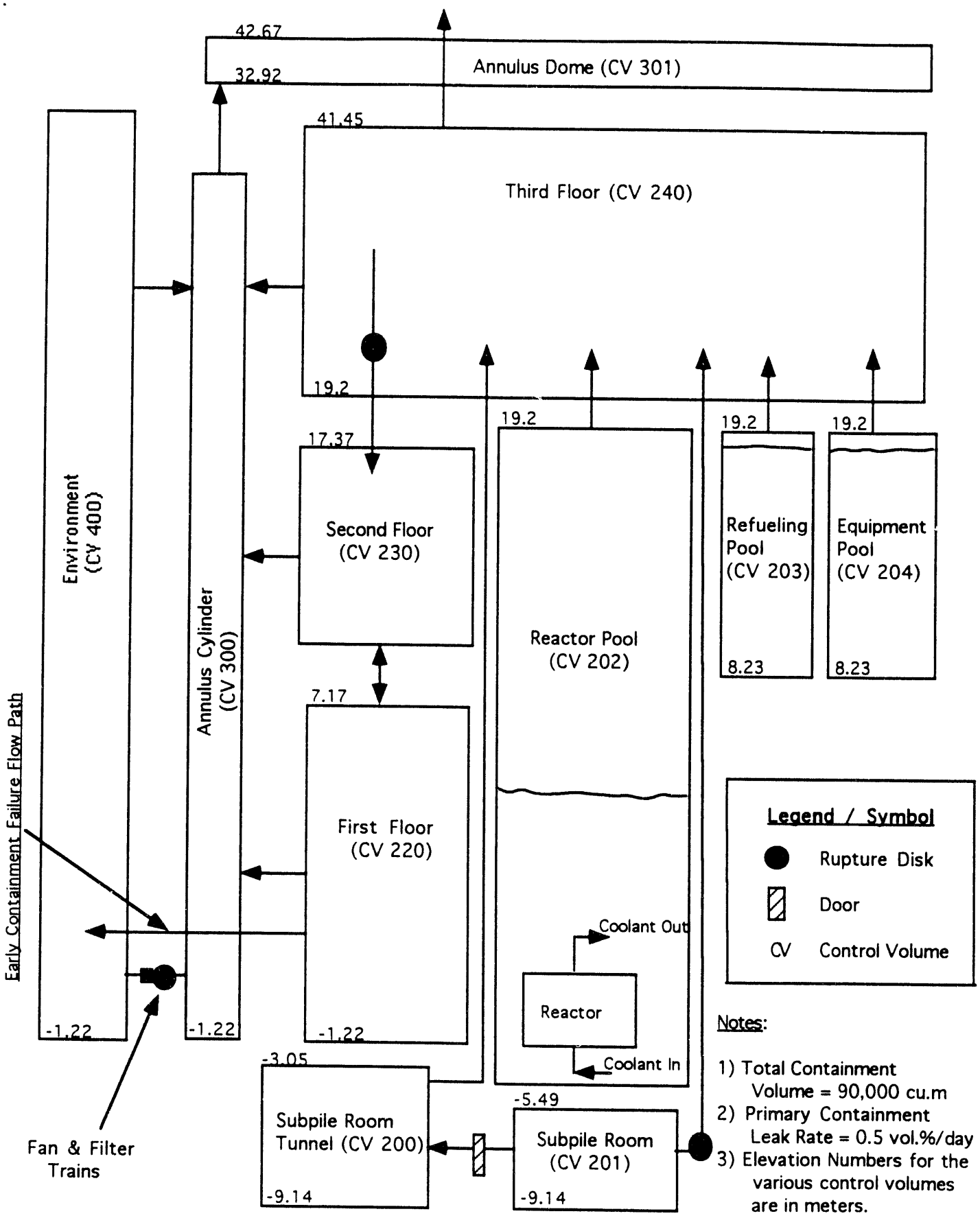


Figure 2. ANS Containment (MELCOR) Representation

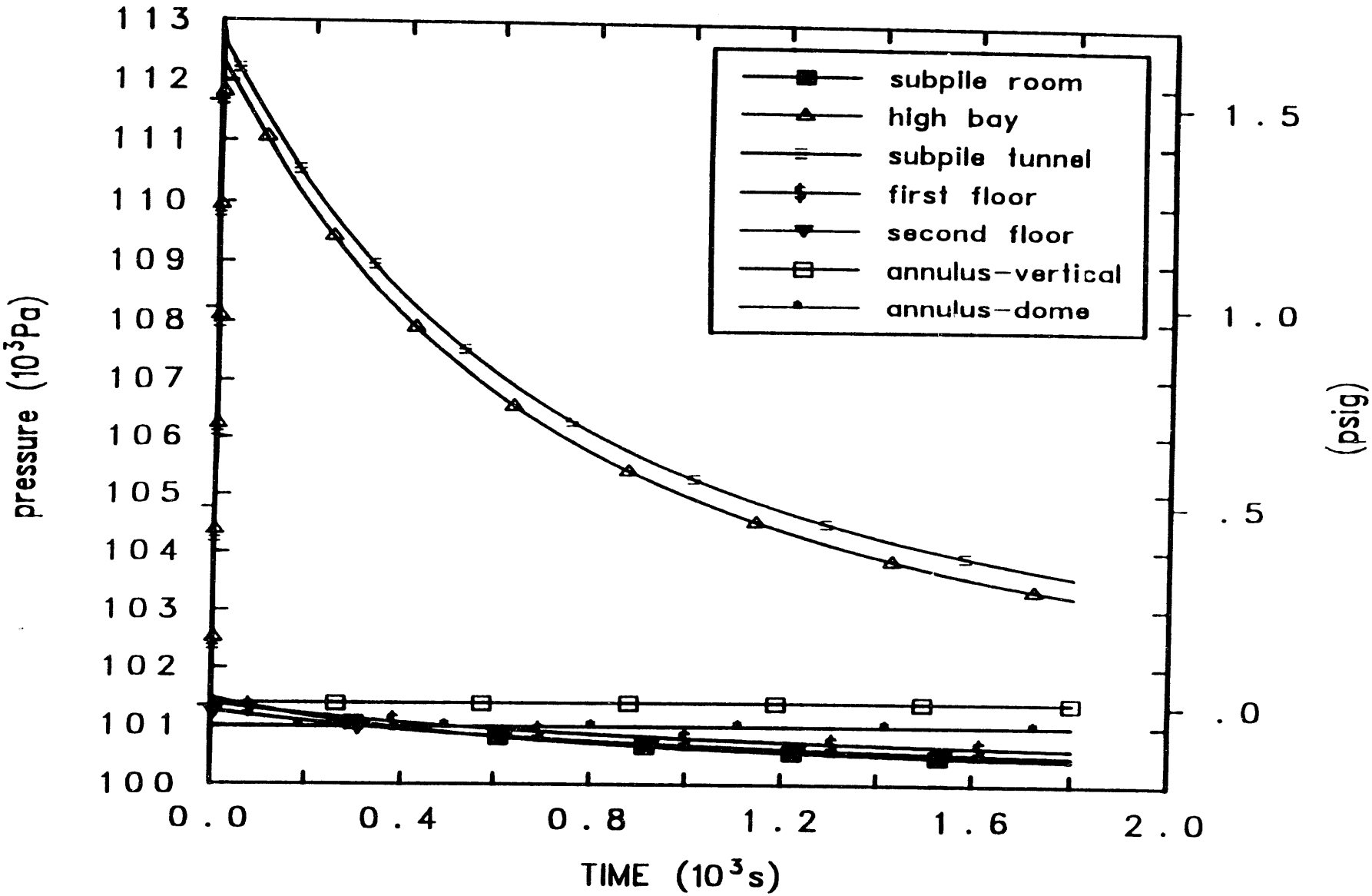


Figure 3. Variation of Containment Pressure with Time During Power Burst Refueling Accident (Containment - Intact, Aluminum Ignition - Included)

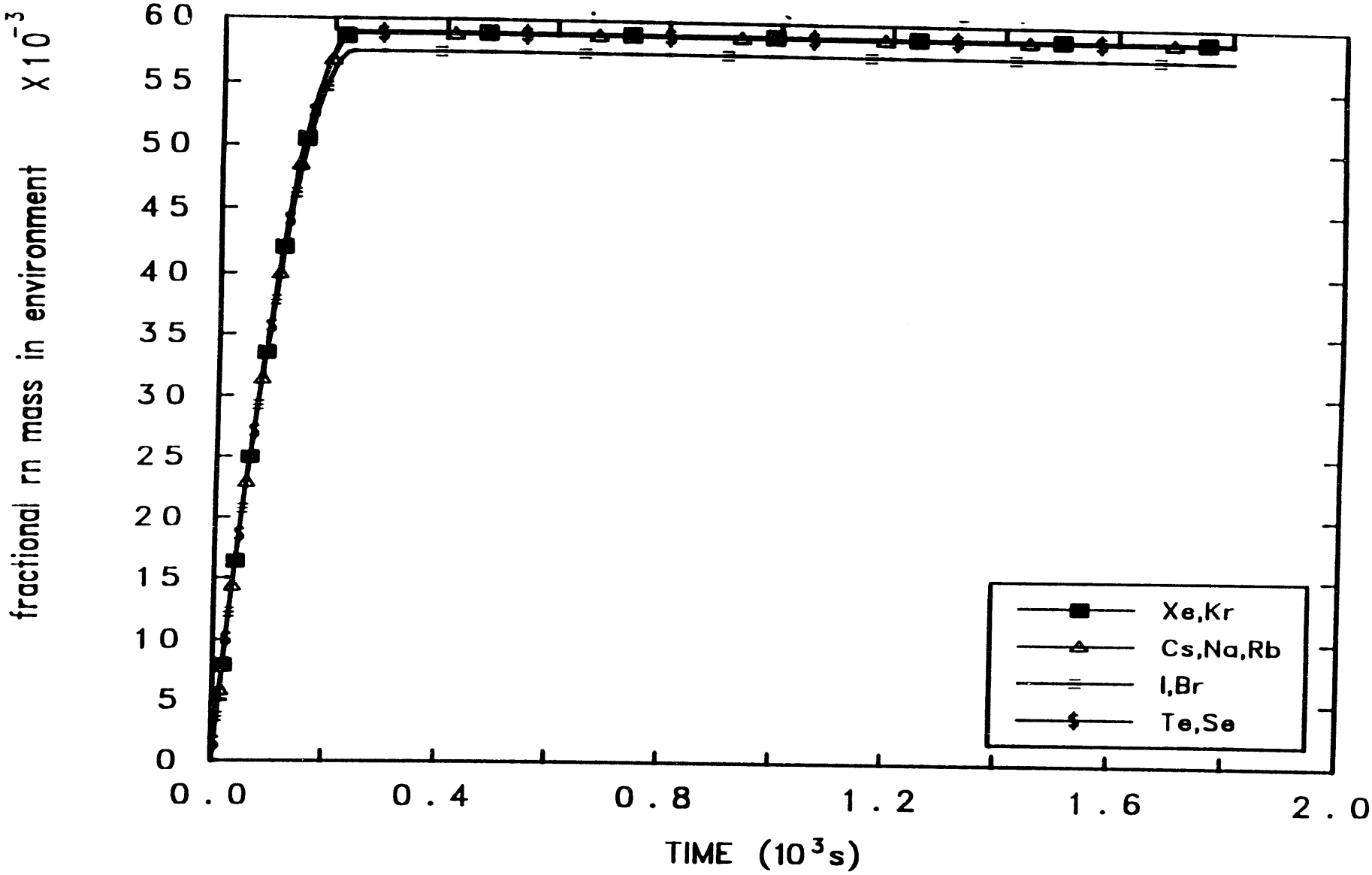


Figure 4. Variation of Source Term with Time During Power Burst Refueling Accident (Containment - Failed, Aluminum Ignition - Included)

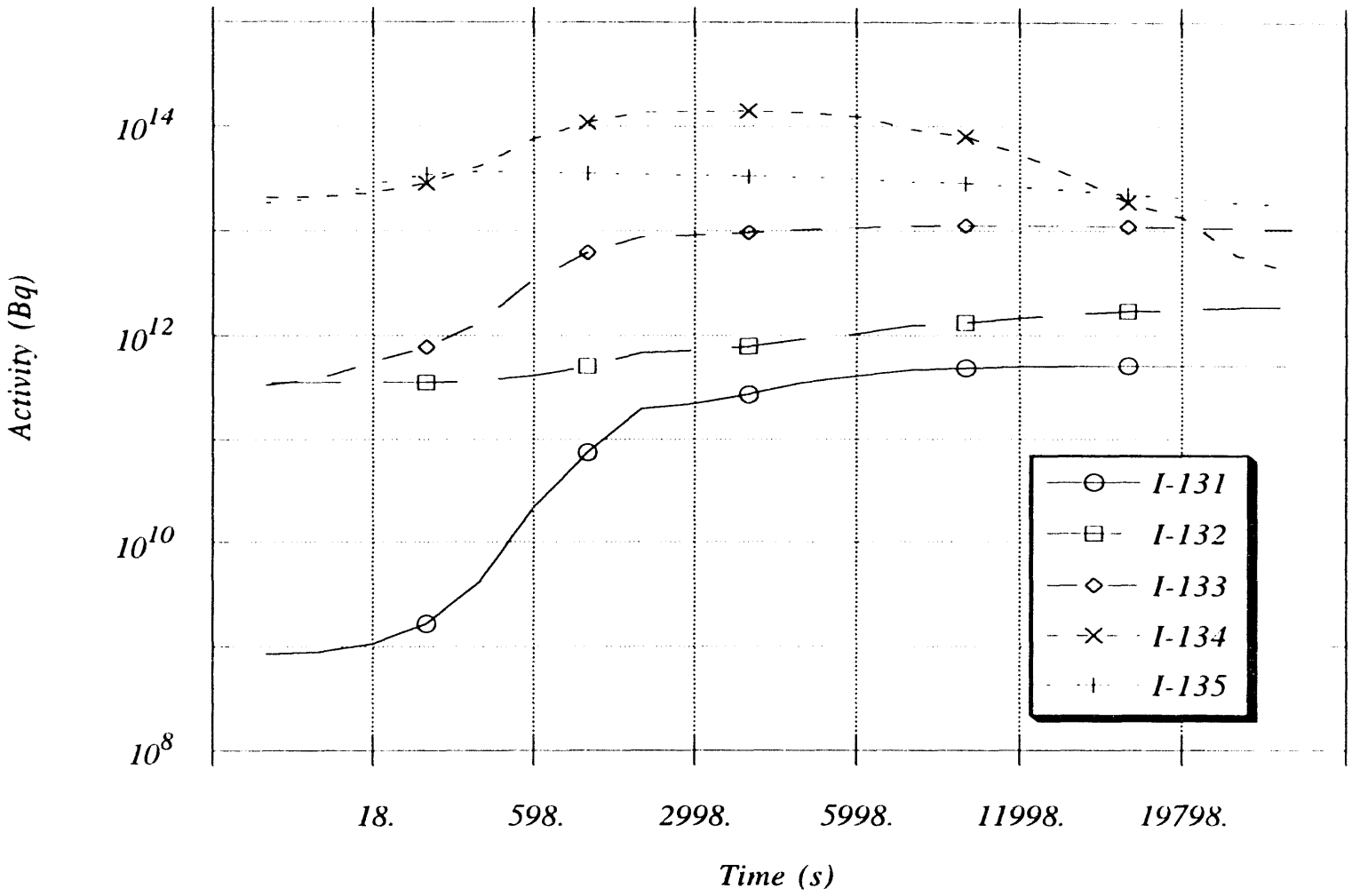


Figure 5. Variation of Important Radionuclides with Time (Irradiation time = 2 s at 330 MW)

DATE
FILMED
10/28/94

END