THE CLINCH RIVER BREEDER REACTOR PLANT SAFETY STUDY: SUMMARY AND RESULTS

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ABSTRACT

The Clinch River Breeder Reactor Plant (CRBRP) Safety Study is an assessment of risks to the public associated with the operation of a 380-MWe liquid-metal fast breeder reactor (LMFBR) of the type to be built on the Clinch River site in Oak Ridge, Tennessee. The objectives of the study were to provide a realistic assessment of accident risks associated with the CRBRP, to place them in perspective in relation to other societal risks, and to aid in determining whether accident risks from the CRBRP are comparable to those for previously licensed reactors. Although this study was necessarily not completed with the depth or precision possible for operating reactors, its timing has allowed a systematic and disciplined evaluation of the plant design to search for and identify potential accident scenarios and to provide additional early assurance that important safety considerations have been identified. Results of the study indicate that CRBRP operational risks are small when compared with other societal risks and are comparable to those associated with current-generation Light Water Reactors (LWRs).

DISCLAIMER

I. INTRODUCTION

A Safety Study has been conducted for the Clinch River Breeder Reactor Plant (CRBRP). This study(1) was performed to evaluate the risk to the public associated with the operation of the CRBRP, to provide perspective on this risk in relation to other societal risks, and to develop a basis for assessing the comparability between risk from the CRBRP and that from previously licensed nuclear power plants. This paper presents a summary of the methods used in the CRBRP Safety Study to evaluate radionuclide releases to the environment and the associated health effects of these releases. Results from these analyses, from which insights into important differences between LWRs and the CRBRP regarding radionuclide transport and release can be gained, are also presented.

This summary is divided into four sections:

- II. Core Accident Analysis and Evaluation
- III. Radioactive Material Transport and Release Analysis
- IV. Health Consequence Analysis
- V. Results

Some of the material supporting the conclusions in the results section relies on probabilistic results obtained in other portions of the CRBRP Safety Study (1). A brief introduction to the structure of the Safety Study will be presented here to allow the reader to understand the part that core accident and radionuclide release analysis played:

First a list of potential accident initiators was formulated. An accident initiator had the potential to lead to an accident sequence involving plant protective features. The states of these protective features, and thus the possible accident sequence paths, were described using event trees. The probability of occurrence of each branch of the event tree was estimated using experience data together with fault tree methodology. Radionuclide releases from the core to the reactor containment building (RCB) resulting from the accident were then evaluated. Characteristics of the behavior of the material released to the RCB and the releases to the environment were assessed. Finally, the public health consequences of the radionuclide releases to the environment were evaluated.

The following sections will focus on quantification of radionuclide releases and associated health effects. The starting assumption in this analysis is that conditions exist which are sufficient to cause a core disruptive accident (CDA), defined here as an accident in which core coolable geometry is lost. The development of the probabilities that such conditions exist is presented elsewhere. (1)

II. CORE ACCIDENT ANALYSIS AND EVALUATION

Plant protection and safety systems are provided to prevent the occurrence of core accidents. If an initiator should occur these features are expected to function as designed resulting in no loss of core coolable geometry. Should a sufficient number of these systems fail to function as designed fuel integrity may be lost followed by the possibility of release of radioactive material to the environment. The accident sequences developed in this work were evaluated and generic classes of CDAs were defined. Among these generic CDA classes are loss of decay heat removal capability following reactor shutdown, loss of heat removal capability at power, loss of piping integrity, and loss of flow with failure to shutdown.

CDAs can lead to a wide spectrum of effects. For most classes of CDAs, the predicted result is relatively slow melting of several fuel assemblies followed perhaps by collection of the molten fuel and steel in the bottom of the reactor vessel. If a sufficient fraction of the core is involved, penetration by the molten fuel material through the bottom of the reactor vessel and guard vessel may occur. This is termed thermal damage. It has been conservatively assumed that all events leading to loss of core coolable geometry also lead to melt-through of the reactor and guard vessels.

In addition to thermal damage, mechanical damage resulting from the CDA has been considered. The degree of mechanical damage to the vessel as a result of a CDA can be related to the amount of fuel vapor formed during the accident. In general, the larger the fraction of the core vaporized, the greater will be the energy transmitted to the reactor vessel and head and the greater will be the potential for mechanical damage. For each type of CDA probabilities have been assigned to the various degrees of mechanical damage which might result. Three classes of mechanical damage have been defined and the expected occurrence of each of the damage classes is given below.

- No seal damage (for approximately 90% of the CDAs),
- Moderate seal damage (for approximately 10% of the CDAs),
- Massive seal failure (for approximately 1% of the CDAs).

These categories of mechanical damage are related to three expected ranges of accident energetics:

- Non-energetic termination (energetics so low that seal design performance is not impaired),
- Energetics less severe than structural capabilities of head seals,
- Energetics exceeding the structural capabilities of head seals.

A fourth mechanical damage class has been defined:

- Highly energetic CDA.

The highly energetic CDA has been defined, non-mechanistically, to be an accident with sufficiently high energetics to cause significant damage both to the reactor vessel head and to the reactor containment building. Thus, a significant fraction of the radionuclides released from the core could escape into the environment through the damaged containment. Given the occurrence of a CDA, such severe levels of mechanical damage are judged to be extremely improbable. However, the current status of CDA analysis techniques is such that occurrence of such a severe event cannot be ruled out. Therefore, the highly energetic CDA has been considered in this safety study. The probability of the highly energetic CDA has been estimated to be 10% of the probability of the massive seal failure class (the third class noted above). Table 1 is an example of the format in which the probabilities of various degrees of mechanical damage to the reactor vessel head have been tabulated. Such tables were prepared to specify mechanical damage probabilities for each of the nine (9) generic CDAs.

Analytical results from several computer codes developed for the purpose have been employed in evaluating the accident progressions and in determining the resultant mechanical damage to the reactor, vessel and primary system.

The SAS-3A (2) code was used to analyze the initiating phase of each accident, and the VENUS-II (3) code was used to analyze the hydrodynamic disassemblies when these were predicted to occur. In the transition phase analysis for the CRBRP, key phenomena were examined as separate effects, judgment was employed to construct the scenario. The damage evaluations for events that produce structural loadings have been accomplished using the REXCO-HEP (4) and ANSYS (5) computer codes.

Each category of vessel mechanical damage has been conservatively analyzed to determine the release of radionuclides and sodium from the reactor vessel and into the reactor containment building (RCB) as a result of the initial accident energetics phase. The radionuclide and sodium releases into the RCB arise from two sources. First, radionuclides are released and some sodium may be released (depending upon the degree of seal damage) to the RCB through the head during and shortly after the energetic CDA, and second, material is released to the RCB during the approximately two hundred hours following the CDA as a result of sodium boiling in the reactor cavity (RC) which is vented to the RCB. This boiling is caused by decay heat generated in the molten fuel following melt-through of the reactor vessel and guard vessel.

Table 1 shows the fractional releases of nonvolatile core material and sodium entering the RCB in each of the four classes of mechanical damage associated with a core disruptive event.

III. RADIOACTIVE MATERIAL TRANSPORT AND RELEASE ANALYSIS

As shown in Figure 1, three principal paths exist for transport of radionuclides and sodium from the reactor vessel: one path through the head, one path involving melt-through of the reactor vessel/guard vessel combination and boil-off of sodium from the RC into the RCB, and one path involving possible penetration of the molten fuel through the concrete base mat into the ground water and eventually to the surface water.

The first two of these three paths lead to releases of radionuclides to the RCB where they can either fall out, decay, be removed by scrubbing and filtration, or be released to the environment. This section will discuss the analysis of these release paths. The third path has been evaluated and, as in the Reactor Safety Study (RSS) (6), was shown to contribute insignificantly to the overall risk from the CRBRP.

1. Description of Release Paths

A significant fraction of the gaseous fission products, a smaller fraction of volatile solids, and an even smaller fraction of nonvolatile solids could be released through the head shortly after the energetic CDA occurs. Estimates of these fractional releases (see Table 1) were derived for each of the categories of vessel/head mechanical damage by the following procedure:

o It was assumed that a partial failure of the head seals occurs instantaneously upon initiation of the CDA.

o Using a pressure-time history within the reactor vessel and time required for the fuel vapor bubble to rise from the core to the reactor head, the release of sodium through the reactor head and the fraction of fuel and fission products released was estimated.

Following the initial releases of sodium and radionuclides through the reactor head (occurring immediately after the CDA), some fraction of the core would be expected to melt and to collect in the bottom of the vessel. If a sufficient fraction collects in the bottom of the vessel melt-through would be expected (perhaps occurring as early as 15 minutes after the CDA). Breach of the reactor vessel and guard vessel would lead to draining of the available sodium from the primary system into the RC. This release of sodium would be followed shortly by melting of the remainder of the core and its accumulation in the bottom of the RC. Following vessel melt-through, the fission product decay energy would heatup and boil-off sodium from the RC into the RCB. During this boil-off of sodium, a significant fraction of volatile fission products would follow the sodium into the RCB. The material in the RCB would be gases, vapors, and aerosols from the boil-up material source as well as from the source initially released through the head.

Natural depletion mechanisms including plate-out, condensation, and aerosol agglomeration and settling exist to remove fission products, core material, and sodium from the RCB atmosphere. These depletion mechanisms do not rely on any active system to initiate them, but are benevolent phenomena which occur in the presence of sodium/sodium oxide aerosols, dust particles, and metal surfaces. Pressurization of the RCB, caused by heat evolved in the oxidation of sodium and fission product decay leads to leakage of radionuclides into the environment, and eventually, assuming availability of containment systems, venting of the RCB atmosphere through a scrubber/filter system. In the analysis performed for this assessment, the radionuclide releases which occurred over a period of hundreds of hours were integrated to determine the total release during the course of the accident. These total releases are assumed to issue from the RCB as a single puff. This is a conservative assumption in that if the release were uniformly distributed as a function of time, the effects would be diminished due to dilution and environmental processes.

2. Analytical Tools for Release Analysis

A number of computer codes were used to analyze radionuclide transport and release from containment. These codes have been developed over many years, during which considerable experimental verification of their predictive capabilities has been performed. The key assumptions employed in the radionuclide transport and release analyses are shown in Table 2. The SOFIRE (7) and SPRAY (8) computer codes have been used to evaluate the temperature and pressure history in the RCB following the initial (Path 1 on Fig. 1) release of sodium and radionuclides. This pressure history was used in the RCB leakage analysis during the time between initial material release through the head and the onset of sodium boiling in the RC.

The CACECO computer code (9) has been used to calculate the pressure and temperature in the RC and RCB following vessel melt-through. The results of this analysis were used to evaluate the radionuclide source term to the RCB and, together with the aerosol analysis code HAA-3 (10), to evaluate the release of radionuclides from the RCB to the environment.

CACECO is a computer code used to calculate the heat and material transport within the RC and RCB during the time following melt-through of the vessel until the sodium has been completely boiled away from the RC. All of the important heat sources (e.g., fission product decay heating and sodium vapor combustion) and heat sinks (e.g., the walls of the RC and the containment cooling system) were modeled in the CACECO analysis.

The COMRADEX code(11) has been used to calculate the total radionuclide release based on its calculated time histories of radionuclide releases from the RCB.

3. Cases Analyzed for Radionuclide Releases

Radionuclide releases to the environment have been characterized as dependent only on the initial release of radioactive material through the reactor vessel head to the RCB and on the state of availability of the containment systems. The releases of radioactive material through the reactor vessel head during the accident energetic phase have been discussed and the results presented in Table 1. The possible availability states of containment systems are shown in the containment event tree, Figure 2. Analysis of the various possible sequences representing system availability states to determine expected probabilities and radionuclide releases led to the selection of four containment event tree paths as bounding sequences for detailed analysis using the methods discussed earlier. Those paths are A, C, I, and L. For each of these four paths, the three categories of primary system mechanical damage shown in Table 1 have been considered. Analysis of the releases represented by the eleven combinations (L1 and L2 were combined and represented as L1) of containment system availability state and degree of primary system mechanical damage shown in Figure 2 has resulted in release conditions shown in Table 3 and in the fractional releases of core inventory shown in Table 4. The twelfth release evaluated, shown in Table 4, is that for the highly energetic CDA (HECDA).

These values represent the fraction of the radionuclide inventory in the CRBRP core at the end of an equilibrium cycle calculated to be released to the environment. These radionuclide releases were used in evaluating the public health consequences.

IV. HEALTH CONSEQUENCE ANALYSIS

Following a radioactivity release from the containment building, material would be transported downwind according to the prevailing meteorological conditions. Ultimately, the radioactivity is either uniformly mixed with the global atmosphere or is deposited on the surface by wet (rain) or dry (fallout) deposition.

The radiation exposure to people caused by release of radioactivity to the atmosphere was divided into three components:

- Direct λ-ray exposure from the passing cloud
- Irradiation from material deposited on the ground
- Desposition of radioactive material in the body by inhalation of the passing cloud

Possible medical consequences considered include:

- Early death from acute whole body exposure
- Respiratory impairment from acute lung exposure
- Growth of thyroid nodules from cumulative thyroid exposure
- Fatal latent cancer from cumulative whole body exposure

A computer model similar to that used in the RSS was employed to characterize potential consequences. The block diagram for this model is shown in Figure 3. The radionuclide releases associated with each of the twelve release categories were assumed to occur as a "puff" releases. Health effects arising from each release category were calculated using a spectrum of atmospheric dispersion conditions characteristics and the population distribution surrounding the site. The computer model used dose conversion factors derived from the International Committee on Radiation Protection (ICRP) model to compute health effects.

V. RESULTS

The above analyses led to results of two general types, radionuclide releases and health effects, for each of the important release categories. When combined with the probabilities of each of the twelve release categories (derived as discussed in Reference (1)), these results can be presented in the form of complementary cumulative distributions (here referred to as consequence curves) of health effects and radionuclide releases. Figures 4 and 5 are examples of consequence curves for early and latent fatalities. To gain some insight into the comparison between Pressurized Water Reactors (PWRs) and CRBRP, consequence curves were developed for fractional releases to the environment of each of the important radionuclide groups. Figures 6, 7, and 8 are examples of these curves. The form in which these curves are presented normalizes out reactor siting effects by presenting radionculide releases from containment to the environment, and normalizes out, to a degree, the effect of reactor size by presenting radionuclide releases as a percent of whole core inventory. Thus, Figures 6, 7, and 8 are the basis for a comparison between the radionculide releases from PWR accidents as identified in the RSS (6), and those from CRBRP accidents. Figure 6 indicates that the noble gas release distributions are quite similar for PWRs and the CRBRP. The only deviation is for relatively high probability events in PWRs in which noble gases are released without melting of the core. No similar events have been identified for the CRBRP. Figures 7 and 8 show radionuclide releases for halogens and heavy metals respectively. As shown on those figures, across a wide range of probabilities (from about 10^{-5} to about 5×10^{-9}) there is a significant difference between the fraction of fission products released to the environment with CRBRP releases being significantly lower. At probabilities lower than 5×10^{-9} the CRBRP releases are depicted as being equal to or, for heavy metals, greater than those from a PWR. The reasons for these observations are as follows:

1. For most types of accidents, containment integrity for the CRBRP is maintained for a considerably longer time following the occurrence of events beyond the design basis than for PWRs. During that longer time, naturally occurring radionuclide depletion mechanisms (e.g. aerosol agglomeration and settling) lead to significant reduction in the quantity of radionuclides available for release.

2. The highly energetic CDA is the only significant mechanism assumed to lead to early failure of the containment and large radionuclide releases to the environment. Because of the nature of the HECDA, a considerable fraction of heavy metals has been assumed to be released to the environment, thus accounting for the crossing of the CRBRP release curve over that for a PWR at the low probability end.

The curves on which the above observations are bases have been normalized to eliminate, as nearly as possible, the effects of reactor site and size. However, it should be pointed out that the curves should not be interpreted as generic comparisons since the CRBRP curves have been developed specifically for the 380 MWe CRBRP which at the time of the analysis was in its final design stage, while the PWR curves are based on operating reactors.

Thus, the uncertainties in the CRBRP curves are larger than those for the PWR curves, and the effect of variations in system design and accident phenomenology resulting from size increases have not been included in the study.

VI. REFERENCES

(1) "CRBRP Safety Study," CRBRP-1, March 1977.

(2) M. G. Stevenson, et. al., "Current Status and Experimental Basis of SAS LMFBR Accident Analysis Code System," CONF-740401-P3, 1303-1321.

(3) J. F. Jackson and R. B. Nicholson, "VENUS-II: An LMFBR Disassembly Program," ANL-7951, September 1952.

(4) Y. W. Chang and J. Guildys, "REXCO-HEP: A Two Dimensional Computer Code for Calculating the Primary System Response in Fast Reactors," ANL-75-19, June 1975.

(5) ANSYS, Swanson Analysis Systems, Inc., 870 Pine View Drive, Elizabeth Pa. 15037. (ANSYS is a proprietary computer code of Swanson Analysis Systems, Inc.).

(6) "Reactor Safety Study," WASH-1400, (NUREG 75/014), October 1975.

(7) P. Beiringer, et. al., "SOFIRE-II Users Report", AI-AEC-13055, March 1973.

(8) R. P. Shire, A Combustion Model for Hypothetical Sodium Spray Fire within Containment of a Liquid Metal Fast Breeder Reactor, MS Thesis, University of Washington, 1972.

(9) R. D. Peak, "User's Guide to CACECO Code," Hanford Engineering Developments Laboratory, Richland, Washington, HEDL-TC-859, 1977.

(10) R. S. Hubner, et. al., "HAA-3 User Report," AI-AEC-13038, March 1973.

(11) G. N. Spangler, et. al., "Description of the COMRADEX Code", AI-67-TDR-108, 1967.

Table 1. Typical Vessel Mechanical Damage and Release Potential Matrix for a Core Disruptive Event.

Core Damage \ Vessel Damage	Mechanical Damage			
	No Head Seal Damage (Case 1)	Moderate Head Seal Damage (Case 2)	Massive Head Seal Failure (Case 3)	Highly Energetic CDA
Whole Core Melting	0.9	0.1	0.01	0.001
Non-Volatile Material Release to Containment (% of Core)	0.01	1.0	10.0	**
Sodium Released to Containment (lb)	10.0	100.0	1000.0	

**Releases to Environment:

Noble Gases	100%
Halogens	70%
Volatile Solids	50%
Non-Volatile Core Material	10%

Table 2. Assumptions in Radioisotope Release Analysis

- Hydrogen recombination was assumed
- Sodium penetration into concrete
  - 1/2 inch per hour
  - Maximum penetration is 2 inches
- RCB Failure at 20 psig
- RCB Purge and Vent at 10 psig
- 99% Scrubber/filter efficiency
- Containment leakage rate is 0.1 volume percent per day at 10 psig

Table 3. Basic Features of Radioisotope Release Cases

Containment Event Tree Path	Annulus Cooling	Scrubber/ Filter	Vent Type	Vent Time (hr)	Purge Time (hr)
C	yes	no	10 psig	20.7	24.1
A	yes	yes	10 psig	20.7	24.1
I	yes	no	hole	none	22.0
L	no	no	overpressure failure	31.2	none

Table 4. Summary of CRBRP Releases to Environment

Containment Event Tree Path	Probability*	Release (Percent of Core Inventory)			
		Halogens	Nobels	Volatile Solids	Pu, Am, Cm
HECDA**	4.9-9	70	100	50	10
I3	1.5-10	3.3	61	3.3	3.2
L3	2.5-8	0.91	45	0.96	1.0-3
L1	1.2-6	0.78	34	0.82	1.2-6
I2	2.3-9	0.48	52	0.50	0.22
I1	8.5-8	0.48	51	0.30	0.0021
C1	5.5-9	0.38	76	0.40	1.7-6
C2	1.3-10	0.38	76	0.40	1.7-4
C3	7.7-12	0.35	76	0.37	1.0-3
A3	4.9-8	0.0075	76	0.0078	9.4-4
A2	6.1-7	0.0075	76	0.0078	1.4-4
A1	2.0-5	0.0074	76	0.0077	1.4-6

*These numbers should be interpreted as follows 4.9-9 is 4.9×10^{-9}.

**This HECDA case refers to the highly energetic CDA.

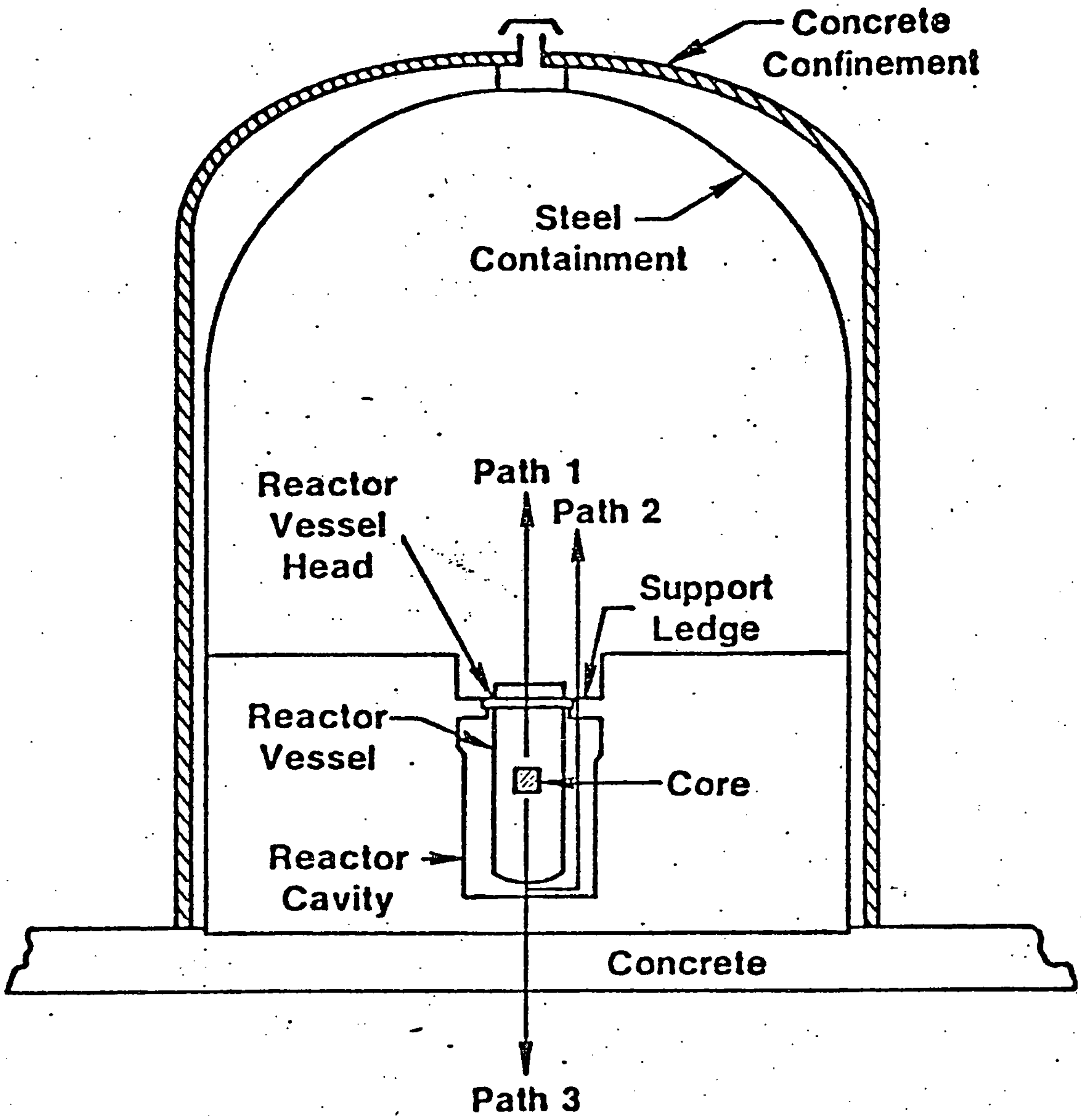
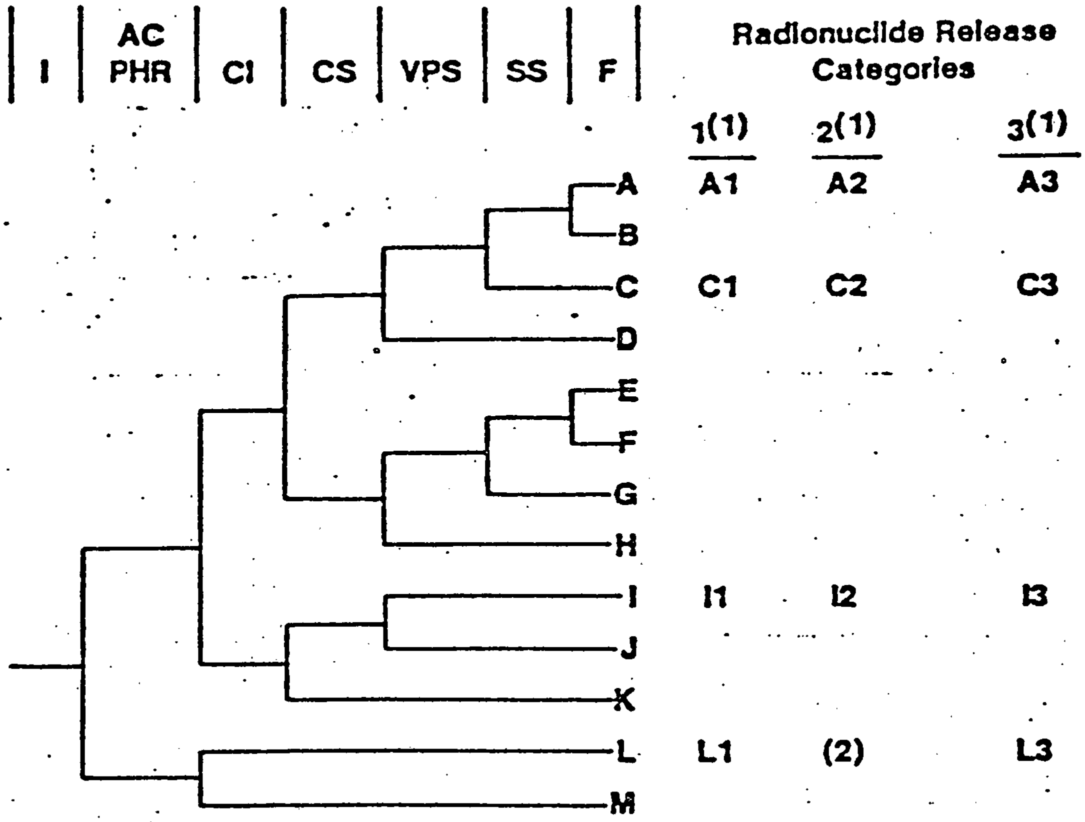


Figure 1. Schematic Diagram of Post-CDA Radionuclide Flow Paths



I—Initiating Event (High Radiation In Containment)

AC PWR—AC Power

CI—Containment Isolation

CS—Containment Cooling System

F—Filters

VPS—Vent & Purge System

SS—Scrubber System

Notes:

(1) Vessel Damage Designation
3: Massive; 2 Moderate: 1: Insignificant

(2) The Probability Of This Release Category Has Been Conservatively Added To That Of Category L1

Figure 2. Correlation of Release Calculations and Categories with the Containment Event Tree

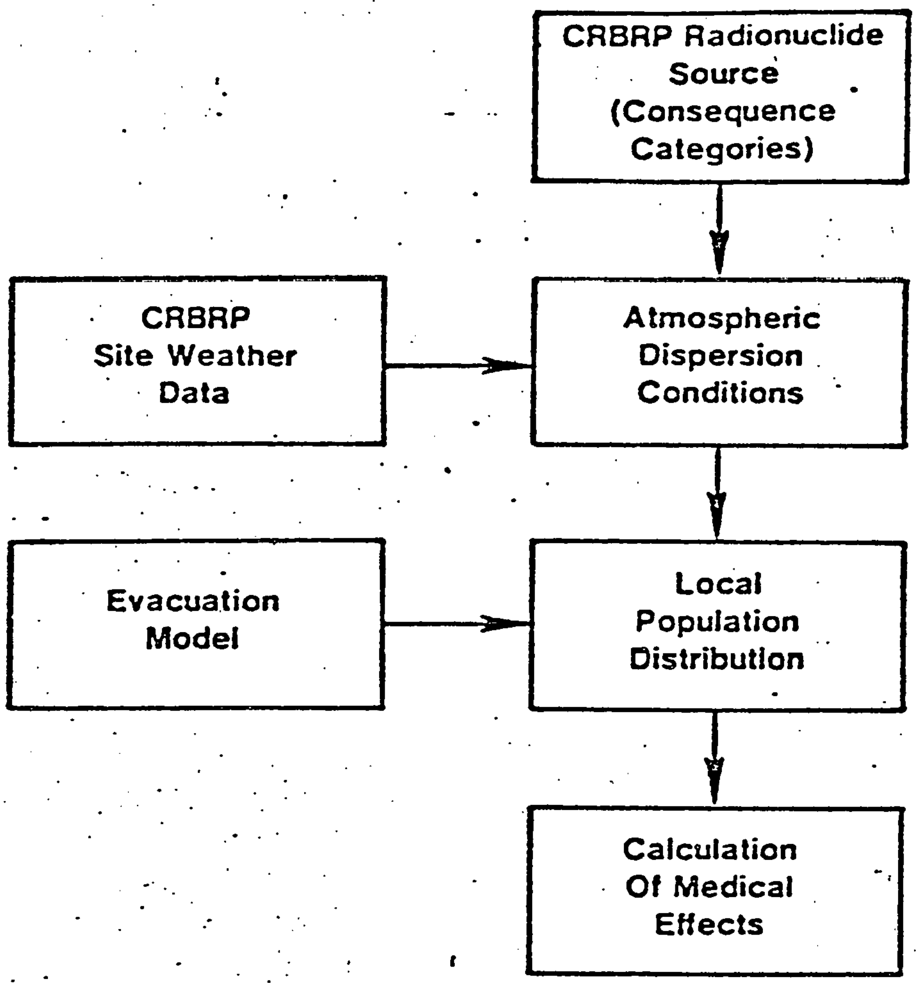


Figure 3. Schematic View of Consequence Calculational Model

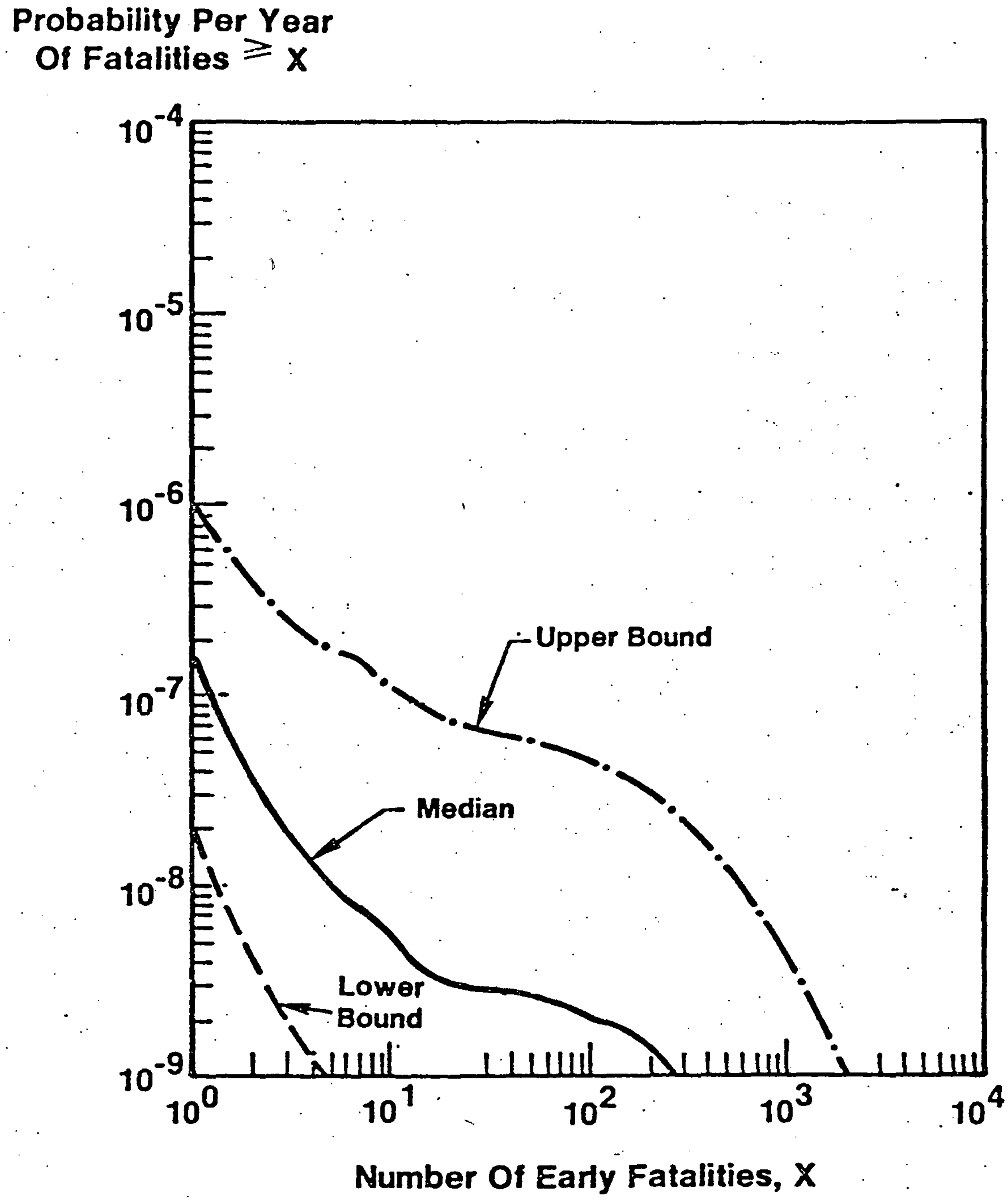


Figure 4. Cumulative Probability Distribution for Early Fatalities

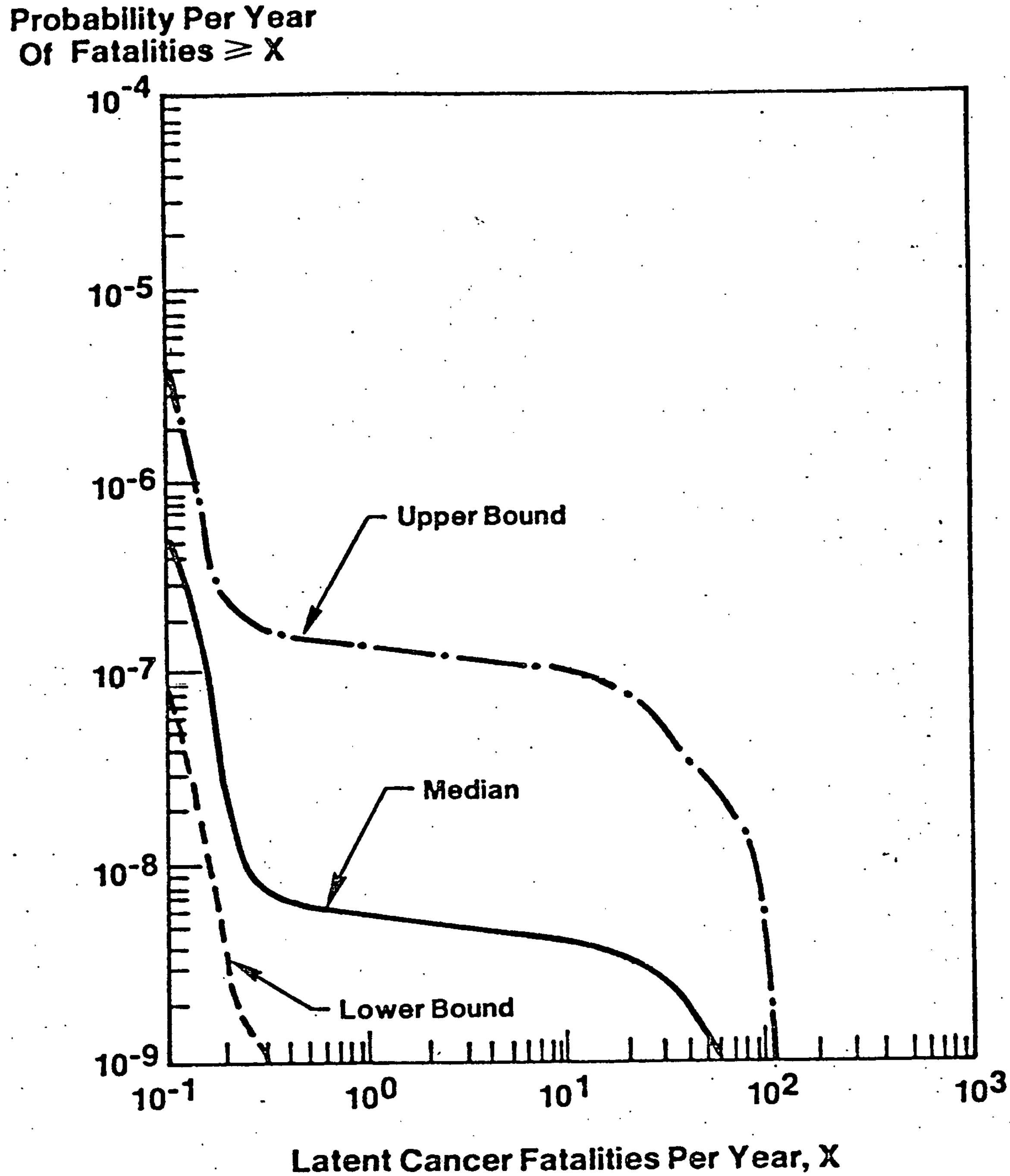


Figure 5. Cumulative Probability Distribution for Latent Cancer Fatalities per Year

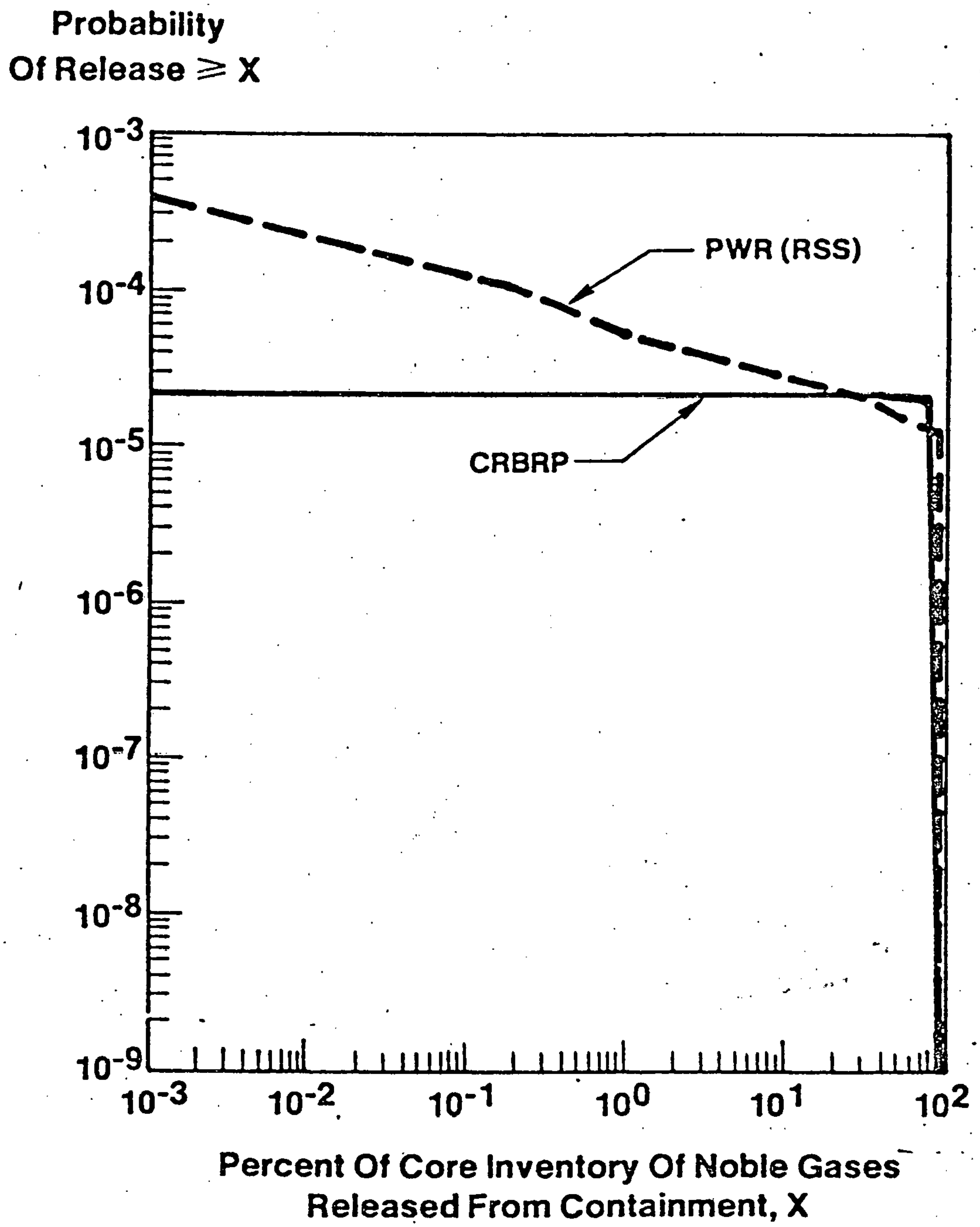


Figure 6. Comparison of Cumulative Probability Distributions for Release of Noble Gases from Containment

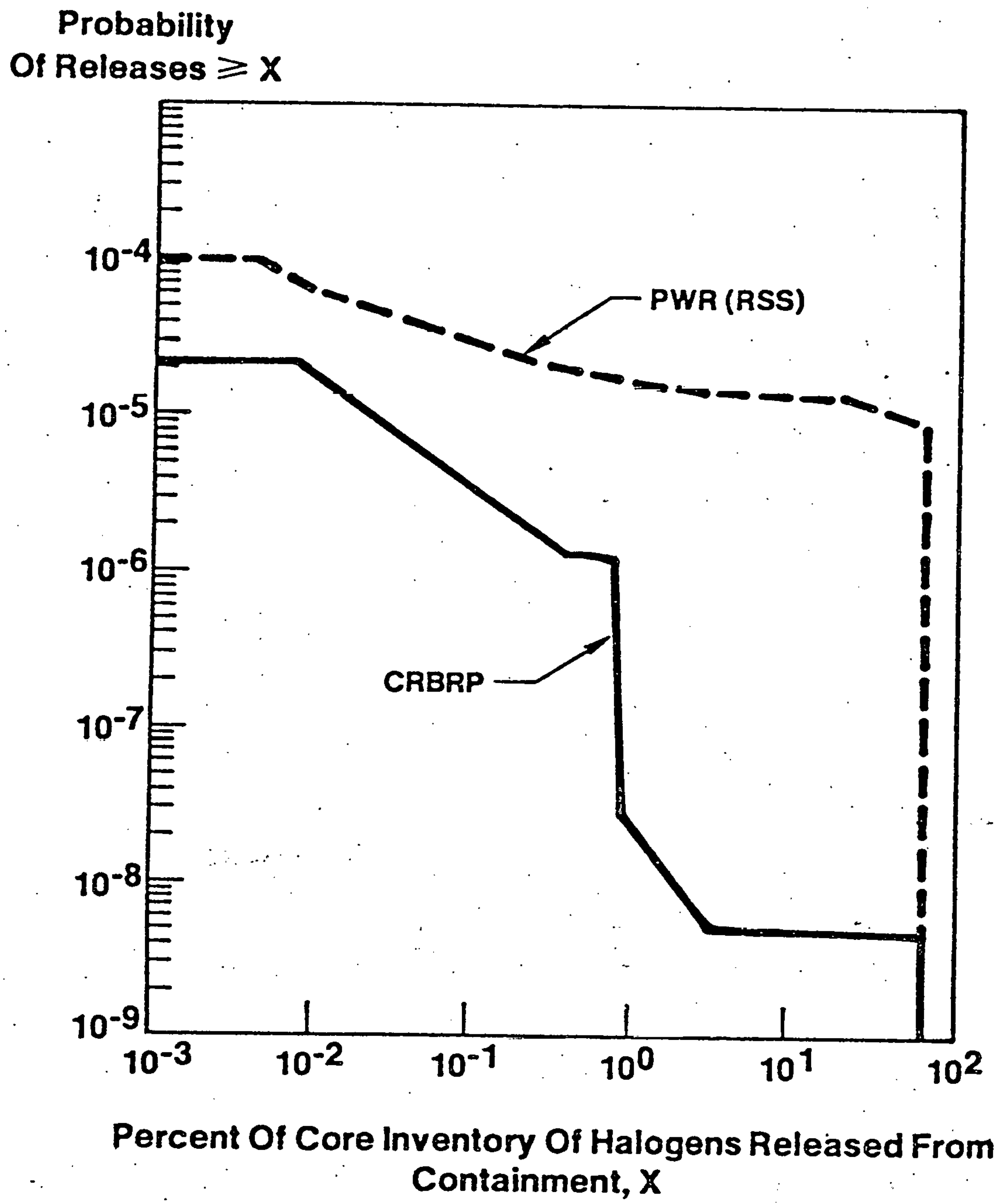


Figure 7. Comparison of Cumulative Probability Distribution or Release of Halogens from Containment

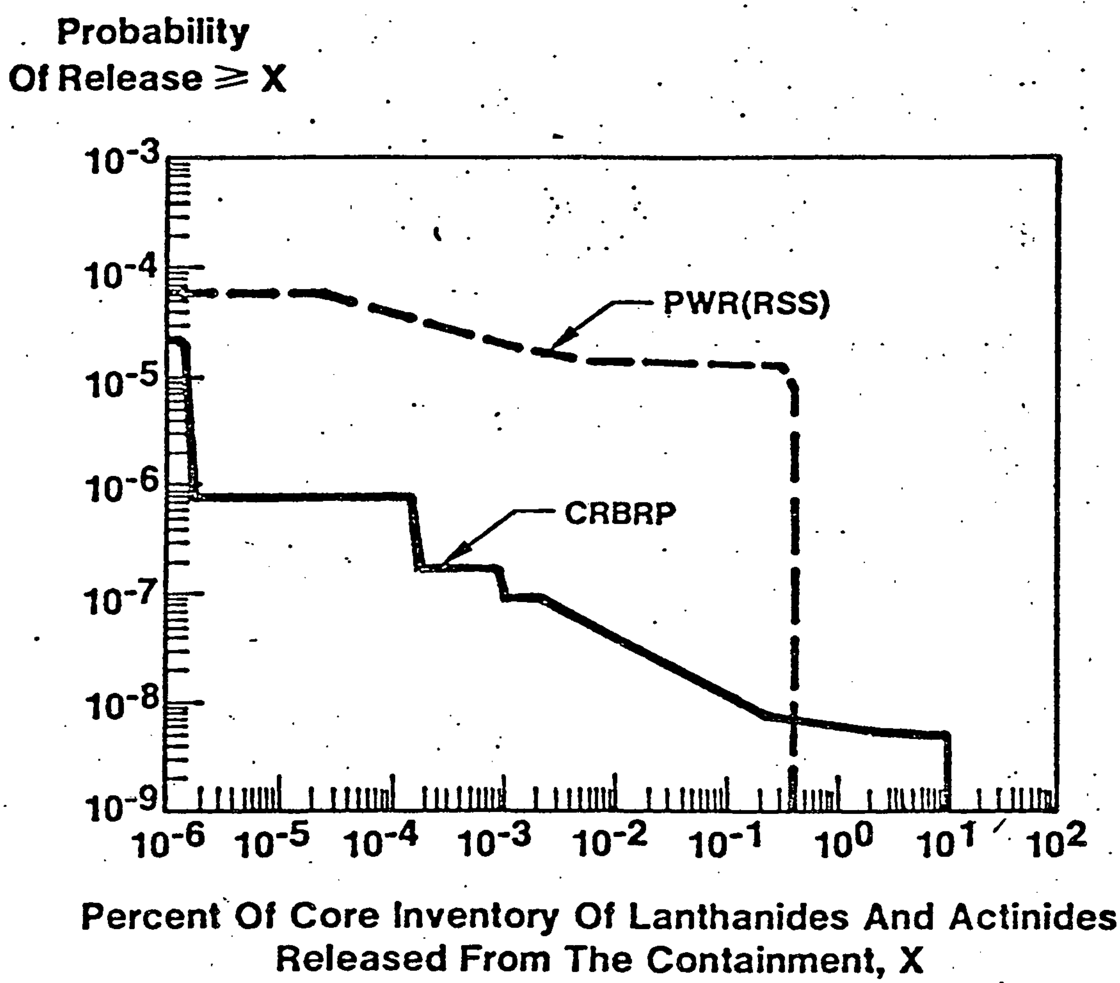


Figure 8. Comparison of Cumulative Probability Distributions for Release of Non-Volatiles from Containment