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Estimate of Radionuclide Release Characteristics Into Containment Under Severe Accident Conditions

Draft Report for Comment

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Prepared for
U.S. Nuclear Regulatory Commission

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Abstract

A detailed review of the available light water reactor source term information is presented as a technical basis for development of updated source terms into the containment under severe accident conditions. Simplified estimates of radionuclide release and transport characteristics are specified for each unique combination of the reactor coolant and containment system conditions. A quantitative uncertainty analysis in the release to the containment using NUREG-1150 methodology is also presented.

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Executive Summary

Estimation of accident source terms is important in nuclear safety regulation. Current regulations (10 CFR Part 100) require that the suitability of the reactor site be judged based in part on a postulated fission product release associated with a substantial core-melt accident. The release into containment for this accident is derived from the 1962 Atomic Energy Commission (AEC) report TID-14844 ("Calculation of Distance Factors for Power and Test Reactor Sites"). The instantaneous release into containment of 100 percent of full power noble gas fission products, 50 percent of iodine fission products, and 1 percent of the solid fission products in the core is postulated. Regulatory Guides 1.3 and 1.4 suggest that the bulk of the iodine (about 90 percent) should be assumed to be elemental (I_2). This "maximum credible" accident, postulated for site analysis is a nonmechanistic event and no specific accident sequence leading to the postulated release is specified. Regulatory Guides 1.3 and 1.4 specify a large loss-of-coolant accident in conjunction with this accident.

The use of TID-14844 release assumptions has not been confined to a determination of site suitability alone. The regulatory applications of releases of this type cover a wide range, including the basis for (1) performance of important fission-product clean-up systems such as sprays and filters, (2) post accident habitability requirements for the control room, (3) the radiation environment for safety related equipment qualification, (4) post-accident sampling systems and (5) containment leak rates.

There has been significant research activity regarding severe accidents following the accident at Three Mile Island Unit 2 (TMI-2). A detailed review of the available source term information for light water reactors from this extensive research has been performed for the present study. This information is provided to support the generation of an updated estimate of source terms appearing in containment under severe accident conditions.

Estimates of radionuclide release and transport characteristics are specified for each unique combination of reactor coolant and containment system condition. The characteristics of the radionuclide releases in this study are clearly different than the hypothetical source terms proposed in TID-14844.

The similarities between the NUREG-1150 method to estimate source terms and that proposed in this study allowed the use of NUREG-1150 expert elicitation on source term issues to assess the uncertainty inherent in the release estimates. An assessment of the range and distribution of the source terms appearing in containment is also presented in this report.

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Foreword

The information in this report will be considered by the NRC staff in the formulation of updated accident source terms for light water reactors (LWRs) to replace those given in report TID-14844. These source terms are used in the licensing of nuclear power plants to assure adequate protection of the public health and safety.

Any interested party may submit comments on this report for consideration by the staff. To be certain of consideration, comments on this report must be received by the due date published in the Federal Register Notices. Comments received after the due date will be considered to the extent practical. Comments should be sent to the Chief, Regulatory Publications Branch, Division of Freedom of Information and Publications Services, Mail Stop P-223, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Further technical information can be obtained from Mr. Leonard Soffer, Office of Nuclear Regulatory Research, Mail Stop NL/S-324, U.S. Nuclear Regulatory Commission, Washington, DC 20555. Telephone (301) 492-3916.

1. Introduction

1.1 Background

The release of fission products from the core could occur as a result of many accident sequences. The accidents in safety analysis reports are termed design basis accidents (DBAs) because they establish criteria for the design and evaluation of a variety of safety-related systems and equipment. Design basis accidents, which consider the release of fission products, include: (1) accidents involving the release of activity normally circulating in the primary coolant (e.g. steam line break, steam generator tube rupture, instrument line break), (2) accidents involving the release of radioisotopes contained in the void space between the fuel and cladding (e.g. rod ejection (PWR) or rod drop (BWR), fuel handling accidents), and (3) the design basis accident postulated for site analysis (siting DBA-LOCA), involving the release of fission products from the fuel, in addition to coolant and gap activity [1].

Current regulations (10 CFR Part 100) [2] require that the suitability of the reactor site be judged based in part on a postulated fission product release associated with a substantial core-melt accident. The release into containment for this accident is derived from the 1962 Atomic Energy Commission (AEC) report TID-14844 [3] ("Calculation of Distance Factors for Power and Test Reactor Sites"). Postulated is the instantaneous release into containment of 100 percent of full power noble gas fission products, 50 percent of iodine fission products, and 1 percent of the solid fission products in the core. Regulatory Guides 1.3 and 1.4 suggest that the bulk of the iodine (about 90 percent) should be assumed to be elemental (I_2). This "maximum credible" accident, postulated for site analysis is a non-mechanistic event and no specific accident sequence leading to the postulated release is specified. Regulatory Guides 1.3 and 1.4 specify a large loss-of-coolant accident in conjunction with this accident.

Use of TID-14844 release assumptions has not been confined to a determination of site suitability alone. The regulatory applications of releases of this type cover a wide range, including the basis for (1) performance of important fission-product cleanup systems such as sprays and filters, (2) post-accident habitability requirements for the control room, (3) the radiation environment for safety related equipment qualification, (4) containment leak rates including the transfer of highly radioactive containment sump water into another building, and (5) post-accident sampling systems and accessibility.

At some relatively low frequency level a combination of system failures can lead to accidents that would be more severe than the current set of DBAs. These beyond design basis accidents are not usually analyzed in safety analysis reports. However, they are included in risk assessment studies. Those beyond design basis accidents which results in significant core failure are called severe accidents.

There has been significant research activity of severe accidents following the accident at Three Mile Island Unit 2 (TMI-2). Updated fission product source term methods were developed and published in BMI-2104 [4]. A technical reassessment of severe accident source term technology for U.S. Light Water Reactors (LWRs) was published in NUREG-0956 [5]. This reassessment involved reviewing experimental and analytical results from severe accident research programs sponsored by the NRC and the nuclear industry. As a result of these activities the Source Term Code Package (STCP) [6] was developed as an integrated tool for source term evaluation. Subsequently, the MELCOR severe accident analysis program has been developed based in part, on the STCP. MELCOR is currently considered to be a state of the art analysis program for reactor accidents.

NUREG-1150 [7] was a major effort to put into a risk perspective the insights that have been generated as a result of recent research into systems behavior and phenomenological aspects of severe accidents. One of the major activities of this study was the development of fission product source terms for a spectrum of accident conditions. A limited number of source term calculations were performed using the STCP for selected plant's accident sequences found to be most important to risk in NUREG-1150. Radiological source terms for other accident scenarios in NUREG-1150 were extrapolated from the STCP results. The uncertainty analyses in NUREG-1150 involved using expert opinion to augment the analytical results to reflect uncertainties in the input data and modeling uncertainties. The process included an assessment of phenomena believed to be important to source term predictions, but which were not modeled in STCP. These phenomena include High Pressure Melt Ejection (HPME) and revaporization of radionuclides retained in the RCS and their subsequent release into the containment after vessel failure.

1.2 Objectives

The objective of this report is to review the available light water reactor source term information and to formulate an approach for estimating the magnitude, timing and composition of radionuclide releases into containment under severe accident conditions. The information in this report may provide a part of the basis to update and revise, as appropriate, the formulation given in TID-14844.

2. Historical Development and Applications of Severe Accident Models

The ability to predict radionuclide release characteristics into containment following a postulated severe accident requires the detailed modeling of a wide range of physical and chemical phenomena associated with core melt progression and fission product release and transport.

The first estimates of severe accidental releases of radioactive material, found in the U.S. Atomic Energy Commission report WASH-740 [8], published in 1957, was an attempt to provide realistic upper bounds of the potential public hazards resulting from certain severe hypothetical accidents. Conservative values were used for many factors influencing the magnitude of the estimated accident consequences.

Regulations for site selection were developed as 10 CFR Part 100, "Reactor Site Criteria" in 1962. In conjunction with Part 100, the concept of a maximum credible accident was developed to evaluate the acceptability of a potential site (siting limits) and containment design requirements. In 1962, the maximum credible accident was described in TID-14844. The TID source term, as it is known, postulated a loss of coolant accident (LOCA). The LOCA is a double ended guillotine rupture of a major coolant pipe. The TID assumed a core meltdown and release of all noble gasses, fifty percent of the iodine, and one percent of the other core particulate materials (solids) to the containment atmosphere.

Since issuance of the reactor site criteria, several systematic attempts have been made to search out a large spectrum of accidents and to use quantitative techniques to estimate the probabilities, source terms, and public consequences. Models of physical processes associated with different accident sequences have been developed to assess the magnitudes and timing associated with the release, transport, and deposition of the radioactive materials from the core, through the reactor coolant system and the containment, and into the environment. Major contributions to source term assessment will be summarized in this chapter to provide a historical perspective.

2.1 Reactor Safety Study (WASH-1400)

The Reactor Safety Study [9] (WASH-1400), was the first systematic attempt to provide realistic estimates of public risk from potential accidents in commercial nuclear power plants. The 1975 study includes analytical methods for determining both the probabilities and consequences of various accident scenarios. Event trees and fault trees were used to define important accident sequences and to quantify the reliability of engineered systems. Detailed investigations were performed to realistically predict fission product release from the reactor fuel and the subsequent transport and behavior within the reactor coolant and containment systems. Calculations were performed for a number of accident sequences and the results for these calculations were used to define a series of release categories into which all of the identified accident sequences could be distributed.

A list of fission product releases from the reactor core, as considered in the Reactor Safety Study (RSS) is shown in Table 2.1. These releases are divided into four major components:

Table 2.1 Fission Product Releases Developed in the RSS*

Fission Product	Gap Release Fraction	Meltdown Release Fraction	Vaporization Release Fraction [d]	Oxidation Release Fraction [e]
Xe, Kr	0.030	0.870	0.100	0.90
I, Br	0.017	0.883	0.100	0.90
Cs, Rb	0.050	0.760	0.190	--
Te[a]	0.0001	0.150	0.850	0.60
Sr, Ba	0.000001	0.100	0.010	--
Ru[b]	--	0.030	0.050	0.90
La[c]	--	0.003	0.010	--

(a) Includes Se, Sb

(b) Includes Mo, Pd, Rh, Tc

(c) Includes Nd, Eu, Y, Ce, Pr, Pm, Sm, Np, Pu, Zr, Nb

(d) Exponential loss over 2 hours with halftime of 30 minutes. If a steam explosion occurs prior to this, only the core fraction not involved in the steam explosion can experience vaporization.

(e) This release fraction is applied to the fraction of core involved in the steam explosion and the fraction of inventory remaining for release by oxidation.

* From the "Reactor Safety Study," Appendix VII, WASH-1400, October 1975, Table VII 1-6.

gap release, meltdown release, vaporization release, and oxidation release. The gap release occurs when the fuel cladding initially experiences a failure. This release consists mostly of activity that was released to gas spaces within the fuel rods during normal reactor operation. Meltdown release occurs from the core heatup and melting within the reactor vessel. The vaporization release occurs as a result of core-concrete interaction. The oxidation release occurs as a result of a steam explosion event.

In the RSS, the fission product species were grouped into seven categories in accordance with similarities in their chemical and physical behavior during severe accidents. The footnote in Table 2.1 gives the grouping of the various fission product species.

Generalized bounding calculations of fission product behavior were used in the RSS to develop simple retention factors for the reactor coolant system transport. These factors were described in terms of primary system escape fractions. A summary of reactor coolant system escape fraction is presented in Table 2.2. An escape fraction of one for all fission products

Table 2.2 Summary of RSS Reactor Coolant System Escape Fractions

Fission Product	Escape Fraction			
	PWR Systems	BWR Systems		
		Boil Off After ECC Interruption	ECC With Core Meltdown	No ECC
Xe, Kr (Group 1)	1	1	1	1
Groups 2-7	1	1	0.1	0.67

ECC = Emergency Core Cooling

was used in all calculations of PWR accidents regardless of pipe break location. In BWR accident sequences where the Emergency Core Cooling System (ECCS) is operational, an escape fraction of one was used for noble gases, but a value of 0.1 was used for all other fission products. In the absence of ECCS for BWR accidents, it was assumed that at the end of core meltdown, 2/3 of all fission products that had been released would have escaped the pressure vessel.

Two specific reactor designs were analyzed in WASH-1400: Peach Bottom Atomic Power Station, a Boiling Water Reactor (BWR) with a Mark I containment and Surry, a 3-loop Pressurized Water Reactor (PWR) with a large dry containment . Five BWR release categories and nine PWR release categories were developed in the RSS. Each category was represented by several parameters that describe the release characteristics.

The RSS has been subjected to critical reviews, and brief descriptions are given in References 10 and 11. The Reactor Safety Study analytical procedure has been used in several areas of reactor regulation such as emergency planning, establishing priorities for safety issues resolution and environmental impact statements.

2.2 Post TMI-2 Review of Source Term Technical Bases

Following the publication of the RSS and the accident at TMI-2, work was initiated to review the predictive methods for calculating fission product release and transport. The results of this review are contained in NUREG-0772 [12]. That review resulted in several conclusions that represented significant departures from the RSS assumptions including the suggestion that cesium iodide (CsI) will be the expected predominant iodine chemical form under most postulated LWR accident conditions.

The potential impact of the NUREG-0772 findings on reactor regulation was examined and the results were issued for public comment in NUREG-0771 [1].

These studies formed the basis for the development of a generic set of radiological releases (NUREG/CR-2239) [13] characterized as siting source terms (SST). These source terms were based on individual computer calculations that had been completed and documented in NUREG-0773 [14].

2.3 Current Source Term Studies

Much of the quantitative assessment in NUREG-0772 was based on scoping calculations that were applicable only to the specific conditions assumed for the calculations. In order to achieve an integrated application of the findings of NUREG-0772, the Battelle Columbus Laboratories performed a source term study. This study involved the development and modification of a number of severe accident computer codes based on emerging severe accident research results. These codes were then coupled to form a suite of codes that would provide feedback in accident sequences. The Battelle suite of codes and the sample analyses were reported in the multi-volume report, BMI-2104 [4].

As a result of the reassessment activities, the Source Term Code Package (STCP) emerged as an integrated tool for severe accident analysis. The STCP is an upgraded version of the BMI-2104 suite of codes and has been used in support of the NUREG-1150 study.

A second-generation source term code, MELCOR [15], has been developed at Sandia National Laboratories as the successor to the STCP. MELCOR has been especially designed to facilitate sensitivity and uncertainty analyses and is currently being used to estimate severe accident source terms and their associated sensitivities and uncertainties in a variety of applications including the NUREG-1150 study and the Independent Risk Assessment Plant study (in which the LaSalle plant is being considered).

2.3.1 Source Term Code Package (STCP)

The Source Term Code Package (STCP) is an integrated set of computer codes which more mechanistically simulates severe accident progression and which was believed to provide more realistic estimates of severe accident source terms than previous studies, such as the Reactor Safety Study. In particular, the characteristics of the source terms obtained with STCP (or other current methods) are clearly different than the hypothetical source term proposed in TID-14844.

The codes are basically those used in the analyses performed for the BMI-2104 report, but have been integrated into one self-consistent code package. A number of changes were made in the process of integrating these codes. Many of the changes merely simplified the use of the codes and reduced the potential for input errors during data transfer by automating the data transfer between some of the codes. The other changes, however, involved actual improvements in the models or in the coupling between models.

The STCP consists of four major computer codes (Figure 2.1). The MARCH3 code is a combination of the MARCH2, CORSOR and CORCON-MOD2 codes. The TRAPMELT3 code is a combination of the TRAPMELT2 and MERGE codes that takes input from the MARCH3

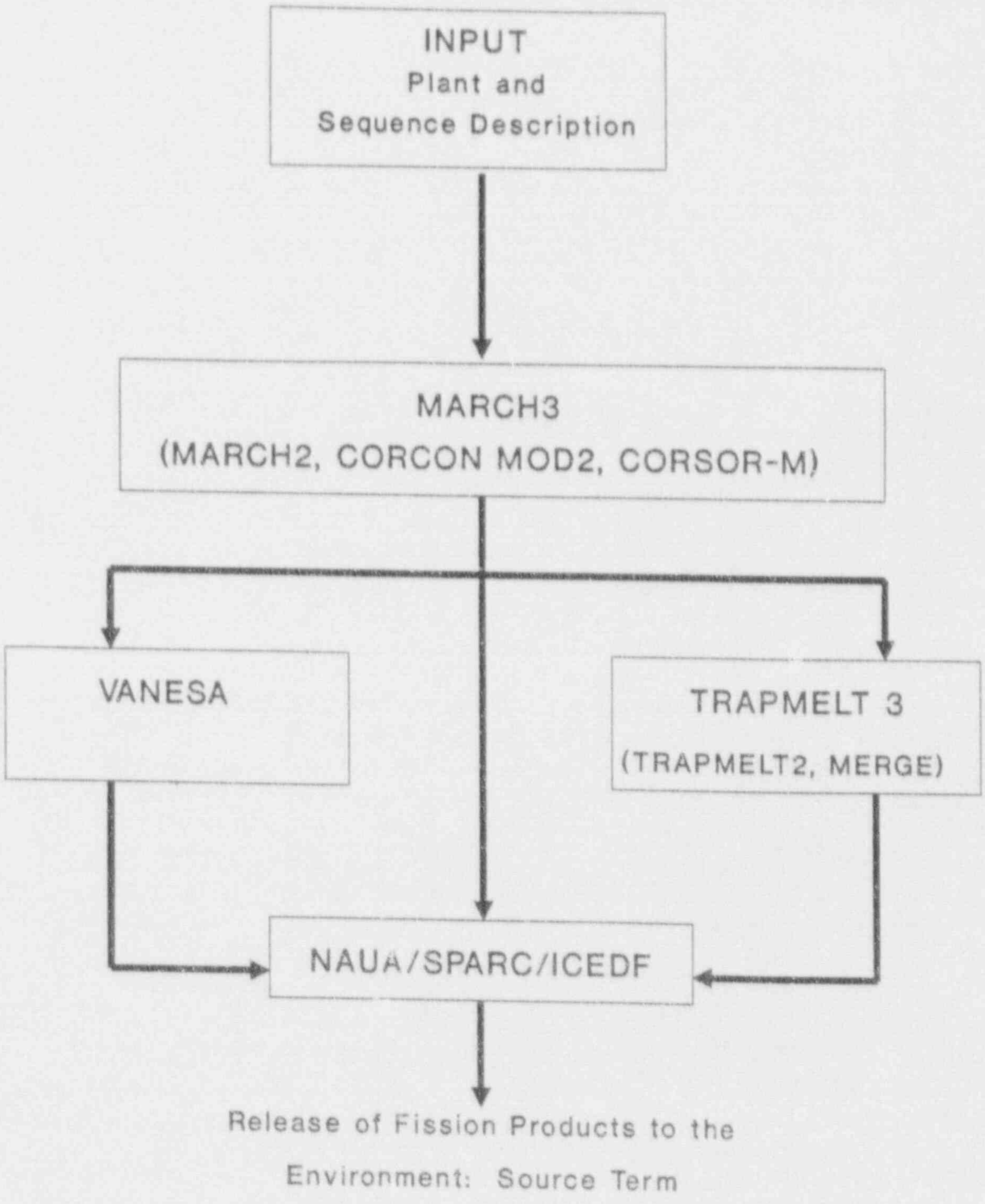


Figure 2.1 Source Term Code Package

code. The VANESA code takes input from MARCH3 and the NAUA/SPARC/ICEDF codes take input from MARCH3, VANESA and TRAPMELT3.

MARCH2: This code simulates the overall thermal hydraulics of severe accidents for light water reactors. MARCH2 consists of many individual subroutines. BOIL and associated subroutines perform reactor coolant system (RCS) calculations including the water boil-off from the reactor, core heatup and meltdown, the time of vessel breach, and the temperature and composition of molten materials released from the vessel. The MACE subroutine evaluates the containment thermodynamic behavior following an accident condition.

CORSOR: This code uses the temperatures calculated by MARCH2 to calculate the transient release of fission products from the fuel within the RCS. CORSOR-M with the Arrhenius form of the release coefficients is the default (preferred) version.

CORCON-MOD2: The mass, composition and temperature of the core debris released from the vessel as determined by MARCH are used by CORCON-MOD2 to evaluate the thermal-hydraulic behavior of corium during core-concrete interactions. The time-dependent temperature and composition of the corium and the accumulated gas generation calculated by CORCON-MOD2, together with the initial inventory of fission products in the corium pool provided by the CORSOR-M module of MARCH3, are passed to the VANESA code for calculation of aerosol generation from corium during core-concrete interactions.

The CORCON-MOD2 code assumes an immediate separation of corium into two immiscible layers: a metallic and an oxidic layer. As the core-concrete interaction proceeds, the decomposed concrete forms a second oxide layer. The orientation of layers depends on their relative densities. Heat transfer from the corium pool to concrete is governed by convective and radiative processes across a gas film that is assumed to exist at the core-concrete interface. In CORCON, the oxidation reaction between the metallic constituents and the concrete decomposition gases is assumed to proceed to equilibrium. It should be noted that the constraint of a metallic-oxidic stratified pool has been dropped and calculations with a mixed-phase debris pool are now possible.

TRAPMELT3: Coupling of the TRAPMELT2 and MERGE codes has led to TRAPMELT3. The TRAPMELT2 code treats radionuclide and inert aerosol transport within the reactor coolant system using the CORSOR-M calculated releases from the fuel as a boundary condition. The TRAPMELT2 code models the RCS by an arbitrary number of interconnected well-mixed control volumes. MERGE accepts the gas flow, temperature, and pressure conditions exiting the core as predicted by

MARCH to perform thermal-hydraulic analyses for connected subvolumes of the RCS and supplies the fluid dynamic variables and structure temperatures to TRAPMELT2.

In the STCP analyses, the iodine and cesium are assumed to be in the form of CsI and CsOH and tellurium is assumed to be in elemental form. These three species are treated as vapors as they are transported from the core. However, in calculating the transport and retention in the RCS, they can condense on walls as aerosol particles, evaporate from where they have condensed, or become chemically absorbed by the surfaces. The remaining less volatile fission products are treated as aerosols.

The TRAPMELT2 treatment of aerosol behavior within the RCS includes models for different processes of agglomeration (i.e., Brownian, gravitational, and turbulent agglomeration) as well as natural removal mechanisms (i.e., Brownian, gravitational, turbulent, and thermophoretic deposition).

VANESA: This code calculates the release of fission products and structural material during core-concrete interaction (CCI). The VANESA code models the vaporization of melt species into gases that are produced from the decomposition of concrete. The thermochemistry and kinetics of this process are modeled mechanistically. As the gases leave the melt, the code empirically models aerosol formation from bubbles breaking the melt surface and from the condensation/nucleation of vapors.

The corium is modeled as a layered two-phase system: an oxide layer above a dense metallic layer that is in contact with the concrete basemat. The reactions of H_2O and CO_2 from concrete decomposition with the major metallic constituents are evaluated to determine the equilibrium oxygen potential. This oxygen potential is assumed to hold for the oxide phase and is used to calculate the equilibrium vapor pressures of species in the M-O-H ternary, where M is the element of interest. A kinetic analysis, which considers condensed phase transport, transport across the gas/melt interface, and gas phase transport, estimates the amount of material transferred from the melt to the gas bubbles.

NAUA-4: This code uses the aerosol release rates from TRAPMELT3 and VANESA and the steam condensation rates from MARCH to calculate the aerosol behavior within the containment.

SPARC: This code calculates the scrubbing of fission products in the suppression pools of boiling water reactors (BWRs) during severe accidents.

ICEDF: This code calculates the fission product and aerosol attenuation in the ice chests of pressurized water reactors (PWRs) with ice condenser containments.

Table 2.3 identifies the radionuclide groups used in the STCP. These groups are an expansion of the original WASH-1400 and BMI-2104 groups.

Table 2.3 STCP Radionuclide Groups

Group	Elements
1	Xe, Kr
2	I, Br
3	Cs, Rb
4	Te, Sb, Se
5	Sr
6	Ru, Rh, Pd, Mo, Tc
7	La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y
8	Ce, Pu, Np
9	Ba

The Severe Accident Risk Reduction Program (SARRP), used the STCP to perform calculations for a number of accident sequences. In addition, STCP calculations have also been performed in support of the effort directed towards the development of a simplified source term methodology. The sequences for which STCP calculations have been performed are summarized in Tables 2.4 through 2.10. Refer to References [16] through [20] for further information.

2.3.2 MELCOR

MELCOR is a fully integrated computer code that models the progression of severe accidents in LWR nuclear power plants. MELCOR has been developed at Sandia National Laboratories as a second generation PRA tool and the successor to the STCP. The entire spectrum of severe accident phenomena, including RCS and containment thermal-hydraulic response, core heatup, degradation and relocation, and radionuclide release and transport, is treated in MELCOR in a unified code framework for both BWRs and PWRs. The development emphasis for MELCOR has been focused on building a reasonably fast-running code capable of being used in parametric studies with detailed mechanistic modeling where possible.

Table 2.4 Large Dry PWR (Zion) STCP Calculated Accident Sequences

Sequence	Description
S2DCR	A LOCA initiated by rupture of primary coolant system (2" break) accompanied by failure of the emergency core cooling injection as well as containment spray recirculation systems. Fan coolers are initially operable, but are assumed to fail at the time of vessel failure. Late over-pressure failure has been selected as the containment failure mode.
S2DCF1	A LOCA initiated by primary pump seal rupture (2" break) accompanied by failures of emergency core cooling, containment sprays as well as containment coolers. An early containment failure mode due to hydrogen combustion and/or direct heating.
S2DCF2	Same as S2DCF1 except a late containment failure mode due to delayed hydrogen burn or overpressurization.
TMLU	Initiated by a transient and is accompanied by the loss of power conversion, auxiliary feedwater, and emergency core cooling systems, both containment coolers and sprays are available. Early containment failure due to direct heating.

Table 2.5 Large Dry PWR (OCONEE 3) STCP Calculated Accident Sequences

TMLB'	A station blackout sequence with loss of all electric power. None of the active engineered safety systems are available.
S1DCF	A LOCA sequence (3 inches diameter break). The emergency core cooling system and the engineered containment safety features are assumed to fail.

Table 2.6 Subatmospheric PWR (Surry) STCP Calculated Accident Sequences

Sequence	Description
AG	A large hot leg break LOCA accompanied by failure of containment heat removal system; the emergency core cooling injection and containment spray systems are available.
TMLB'	Failure of power conversion and auxiliary feedwater systems given the initiating transient event of loss of offsite AC power.
V	Interfacing systems LOCA with containment bypass.
S3B	A station blackout with an induced reactor coolant pump (RCP) seal LOCA.
S2D-δ	A small pipe break with failure of ECC systems and early overpressurization failure due to hydrogen combustion.
S2D-β	A small break, both containment sprays and heat removal systems are operable with sprays continuing to operate after containment failure, a containment isolation failure is assumed (a containment leakage area associated with a 6-inch diameter hole was assumed at the start of accident).

Table 2.7 Ice Condenser PWR (Sequoyah) STCP Calculated Accident Sequences

Sequence	Description
S3HF1	A very small pump seal LOCA with emergency core cooling and containment spray recirculation failure. In this sequence the bottom of the reactor vessel is submerged in the reactor cavity water at the time of vessel failure.
S3HF2	A variation of S3HF1 in which a hot-leg LOCA is induced by high temperature during core degradation.
S3HF3	Another variation of S3HF1 when the reactor cavity was not permitted to fill with water.
S3B	A small break LOCA (1/2 inch diameter break) accompanied by station blackout, none of the active safety features, with the exception of the turbine-driven auxiliary feedwater pump, is available. Containment fails shortly after vessel failure due to a hydrogen burn in the upper compartment.
TBA	A station blackout accompanied by an accident induced large break in a hot leg. A hydrogen burn initiates in the lower compartment and propagates to the upper compartment, causing containment failure before vessel breach.
ACD	This sequence is initiated by a large, hot-leg break after which the emergency core cooling injection and containment spray system fail; the containment air return fans and hydrogen igniters were assumed to be available.
S3B1	A station blackout with the delayed failure of RCP seals in all four pumps three hours into the accident. Auxiliary feedwater (steam driven) operates throughout the accident and (according to plant procedural instructions) the secondary side of the steam generators is depressurized to maintain a reduced primary coolant system pressure.
S3HF	RCP seal LOCA with failure of ECC and containment spray recirculation. The initiating event is an RCP seal failure (in a single pump) of a size sufficient to yield a maximum leak rate (at full primary system pressure) of 480 gpm.
S3H	RCP seal LOCA with failure of ECC recirculation.

Table 2.8 BWR Mark I (Peach Bottom) STCP Calculated Accident Sequences

Sequence	Description
TC1	An anticipated transient without scram accompanied by the failure to achieve early power reduction but successful depressurization of the primary system.
TC2	A variation of TC1 sequence, the failure to scram is accompanied by failure to achieve early power reduction and the failure to achieve emergency depressurization.
TC3	A variation of TC2 with containment venting in the wetwell gas space.
TB1	Loss of all offsite and onsite AC power accompanied by loss of all active engineered safety features except the steam powered emergency core cooling systems. The latter, however, would fail when the station batteries are depleted (6 hours after start of accident).
TB2	A variation of TB1 with containment failure due tc rapid pressurization following failure of the reactor vessel.
S2E1	A small break (2" in diameter) LOCA accompanied by the complete failure of the emergency core cooling systems. For the purpose of this analysis the Automatic Depressurization System (ADS) was not actuated.
S2E2	A variation of S2E1 assuming a basaltic concrete composition.
V	A rupture in the low pressure emergency core cooling system piping in the reactor building outside the primary containment envelope.
TBUX	A station blackout initiated by a loss of all DC power. The operators are assumed to be unable to depressurize the reactor vessel because DC power is unavailable.

Table 2.9 BWR Mark II (LaSalle) STCP Calculated Accident Sequence

Sequence	Description
TB	A station blackout accident with late containment failure mode.

Table 2.10 BWR Mark III (Grand Gulf) STCP Calculated Accident Sequences

Sequence	Description
TC	An anticipated transient without scram. The containment was assumed to fail by over-pressurization prior to core melting due to elevated power input to the suppression pool; containment failure was assumed to lead to failure of emergency core cooling system pumps.
TB1	Loss of all AC power accompanied by loss of all active engineered safety features with the exception of the steam-turbine driven emergency core cooling systems. The latter, however, would fail when the station batteries are depleted (6 hours after start of accident).
TB2	A variation of TB1 with containment failure due to hydrogen burn following failure of the reactor vessel.
TBS	Loss of AC power accompanied by loss of all active engineered safety features. However, the operator was assumed to successfully depressurize the primary system.
TBR	A variation of TBS except that electric power is reestablished shortly after vessel melt-through and thus the sprays in containment operate.

Release of fission products from the fuel is determined by application of CORSOR or CORSOR-M models as specified by the user. A dynamic surface-to-volume ratio can also be applied to either model to account for changes in the core geometry. Released fission products may exist as vapors, aerosols, or both, depending on the material's vapor pressure. If the vapor mass is greater than the saturation value for the fission product vapor, the excess vapor mass is converted to aerosol mass.

MELCOR contains a number of physics packages or modules which model all essential phenomena and plant features. Key packages include those modelling control volume thermodynamics and hydrodynamics, heat structure thermal response, core heatup and degradation, reactor cavity interactions (i.e., core-concrete interaction), and radionuclide behavior.

Thermal-hydraulic behavior is modeled in MELCOR in terms of control volumes and flow paths in Control Volume Hydrodynamics (CVH) and Flow Path (FL) packages. No formal distinction is made between the RCS and containment; the same models and solution algorithms are used for both and the resulting equations are solved simultaneously.

The COR package calculates the thermal response of structures in the core and lower plenum. This package treats all important modes of heat transfer within the core, as well as oxidation, debris formation, and relocation of core and structural materials during melting, candling, and slumping. Lower head heatup, failure, and debris ejection are also modeled. The COR package represents a significant improvement in modeling capabilities over the models in the STCP, especially in the area of core relocation.

The aerosol dynamics portion of MELCOR is based on the MAEROS computer code. MELCOR does not calculate resuspension of deposited aerosol mass. Vapor condensation and evaporation on heat structures and aerosol surfaces are evaluated by the same models as in TRAPMELT3.

For ex-vessel release of radionuclides, the VANESA model has been implemented in MELCOR and has been coupled to CORCON-MOD2 for each time step.

2.3.3 NUREG-1150 Parametric Source Term Models

NUREG-1150 put into a risk perspective the insights that have been generated as a result of recent research into system behavior and phenomenological aspects of severe accidents. One of the major activities of this study was the estimation of the radionuclide release into the environment for a spectrum of accident conditions.

For the NUREG-1150 risk analyses, parametric models were developed that allowed the calculation of source terms for a wide spectrum of accidents. The parametric equations do not contain any chemistry or physics (except mass conservation), but describe the source terms as the product of release fractions and transmission factors at successive stages in the accident progression for a variety of release pathways and a variety of accident sequences. This approach led to development of separate computer codes for each plant; these were labeled following a pattern of XSOR, where "x" identifies the plant. For example, SURSOR was used for Surry, GGSOR was used for Grand Gulf, and so on. It should be emphasized that the parametric models used in the XSOR codes are not time dependent. These codes generate source terms only in terms of early or late releases.

None of the basic parameters used in the XSOR codes are internally calculated. The values for the parameters must be specified by the user or chosen from a distribution of values by a sampling algorithm. The input data on the more important parameters were constructed in the form of probability distributions. Such distributions were developed using expert judgment to interpret the available data or calculations. For a few parameters that were judged of lesser importance or not considered as uncertain, single-valued estimates were used in XSOR models. These estimates were derived from other calculations and adjusted as needed for the boundary conditions associated with the accident progression characteristics.

3. Phenomenological Aspects of Severe Core Damage Accidents

To provide a framework for developing updated source terms (magnitude, timing and composition of radionuclide releases) into containment under severe core damage accident conditions, it is essential to have an understanding of the phenomena which could occur.

Severe core damage accidents involve substantial melting of the reactor core. A characteristic accident sequence leading to severe core damage would be one in which a combination of system failures results both in loss of water from the reactor coolant system and in the failure of the emergency core cooling system to function properly. In such an event, the loss of coolant inventory would result in core uncovery with subsequent heatup and damage to the fuel rods. In the case of delayed operation or partial performance of the emergency core cooling system, core damage may be arrested as occurred in the TMI-2 accident. However, if coolant flow is not restored in time, complete meltdown of the reactor core and subsequent vessel breach could result.

During the early stages of a severe accident fission products are released from the damaged core into the RCS. This material is then transported through the RCS to the containment. During transport, the initial releases are subject to natural deposition processes which can result in substantial retention in the RCS. In some accident sequences, there could be a failure between the RCS and one of the support systems, which could result in a release path which bypasses the containment (Event V in WASH-1400). After the core debris penetrates the reactor pressure vessel, it may attack concrete structures below the vessel and, depending on the cavity configuration, there could be extensive core-concrete interactions.

This chapter gives a brief description of the physical and chemical processes which could take place during the progression of severe accidents. This chapter also describes how these phenomena affect radionuclide releases into the containment.

3.1 In-Vessel Release

The "in-vessel release phase" of a severe accident refers to that period of time during which the reactor core is damaged and begins to melt, but is still retained within the RCS. The characteristics of the fission products released to the containment during the in-vessel phase are controlled by several mechanisms. The relative importance of each of these mechanisms to determining the releases depends on the RCS conditions during the course of the accident and the radionuclides being considered.

As the reactor core uncovers and heats-up, steam may begin to react chemically with the zirconium cladding surrounding the uncovered fuel elements to produce hydrogen and heat. Provided sufficient steam is available for the reaction, the heat released during this zirconium oxidation is of a similar magnitude to the core decay heat.

Phenomenological Aspects

As the core continues to heat-up the zirconium cladding would begin to weaken, balloon and rupture. Upon rupture of the cladding, a small quantity of radionuclides that resides in the gap between fuel pellets and the cladding would be released. This release, which is termed the gap release, would consist mostly of volatile fission products. The gap release would be a relatively small fraction of the total fission product release if a severe accident were to occur.

Following the gap release, any radionuclides remaining in the gap space between the cladding and the fuel rods will diffuse out of the rupture opening. This diffusional release of the gap contents is a relatively slow process and the release is small unless the fuel rods are held at an elevated temperature for a substantial period of time.

As the core temperature increases further, the fuel cladding would begin to melt. During the melting, some of the more volatile components could evaporate from the various liquid surfaces. The details of the melting process are complex. Eutectic interactions can occur simultaneously with the melting process which alter the melting points of key materials. Detailed discussions of physical and chemical processes occurring during melting are given in Reference [21].

As the accident progresses further, the melting fuel elements could eventually destroy or bypass the reactor's core support structure. The molten and unsupported materials would fall ("slump") into the lower head of the Reactor Pressure Vessel (RPV). As hot core debris falls into water remaining in the lower head, significant steam generation would occur. The core debris (fuel, control blades, fuel cladding, and structural materials, called corium) could attack and eventually penetrate the RPV lower head.

The cladding failure and subsequent melting of the core could occur on a region by region basis. Thus, the total release of any given fission product species or other material could occur over an extended period of time. However, the more volatile radionuclides would tend to be released during the early stages of heatup and melting. The less volatile fission products would tend to be released when the core reaches higher temperatures. In a complete meltdown, these releases could continue after RPV melt-through.

The fission products and other materials which are released from the fuel prior to melt-through of the RPV are likely to be transported through the various portions of the RCS. The dominant pathways out of the RCS generally are determined by the location of the pipe break in the case of a loss of coolant accident (LOCA), or by the nearest relief or safety valve in the case of a transient-initiated event. As they move through the RCS, fission products may be retained as a result of various types of interactions. The extent of this retention depends on the fission product chemical and physical form and the thermal hydraulic conditions along the flow path.

The more volatile fission products would tend to enter the RCS as gases while the less volatile elements would tend to condense. The released fission product gases could absorb or condense onto particulates and RCS surfaces, react chemically with other species in the RCS atmosphere or with RCS surfaces, or dissolve in or otherwise react with any water present in the dominant pathway(s) through the system. The aerosols released from the core would tend to increase in size by the agglomeration process. As time passes, some aerosols would be removed by settling or be transported to surfaces by diffusiophoresis, thermophoresis, or other processes. Some of the removed material could subsequently be resuspended, revaporized, or otherwise entrained in the RCS fluids and subsequently transported out of the RCS. A detailed review of the major processes that could occur in the RCS and their effects are discussed in Reference [5].

The extent of retention of any fission product species in the RCS depends on several accident characteristics. Higher surface temperatures in the RCS, higher velocities of gases and particulates through the RCS, and lower aerosol generation rate in the RCS would tend to decrease the extent of retention in the RCS for most species.

Accidents in which the ECCS partially operates or where operation has been delayed could result in extensive core damage without progressing to full core meltdown. If an accident sequence does not progress to full core meltdown, then adequate cooling water must have become available to arrest core damage. The presence of water in the dominant pathway(s) through the RCS would tend to increase the retention of fission products in the RCS.

3.2 Ex-Vessel Release

Only a partial release of the fission product species occurs while the core is in the reactor vessel because of the limited time the core is at high temperatures before it melts its way through the bottom of the pressure vessel. This point represents a logical division in the progression of the accident. After vessel penetration the molten material (including the control blades and part of the support structure and pressure vessel) and most of the remaining radioactive materials would be transferred to the containment. Some of the radioactive material can reasonably be expected to remain within the reactor vessel. Whether this would occur slowly or rapidly as materials enter containment would depend on the accident progression and on the pressure in the RCS at the time of vessel breach.

If the RCS is pressurized at the time of vessel breach the corium will be ejected under pressure in a process which has been demonstrated experimentally to result in significant aerosol generation. In some containment designs where suitable pathways exist, elevated pressure in the reactor coolant system may also cause core debris ejected from vessel to be dispersed out of the pedestal region or reactor cavity as fine droplets. Since the fine debris particles have a large surface area for transferring heat to the atmosphere, the containment atmosphere could experience a rapid rise in temperature and pressure directly from the core debris. This heating phenomenon is called Direct Containment Heating (DCH). If DCH does occur, it implies additional exposure of highly heated and fragmented debris to a possibly oxidizing atmospheric environment, and this exposure is expected to lead to additional aerosol and radionuclide release from the expulsed core debris.

If depressurization occurred prior to reactor pressure vessel failure, then the molten core debris (corium) will relocate below the vessel without being dispersed into the containment atmosphere. Contact of molten core debris with the concrete in the reactor cavity, pedestal, drywell floor, or basemat could lead to core-concrete interaction. The extent of core-concrete interaction and the possibility of cooling the core debris are affected by many factors, including the amount of water available and the geometry. Core-concrete interactions liberate copious amounts of concrete decomposition gas products. As the gas passes through the core debris, volatile elements would be sparged from the molten mass into the containment atmosphere. The water released from the concrete will disassociate and the oxygen will react with the unoxidized zirconium until all of the zirconium has been oxidized. During this oxidation, the oxidation produced heat will be the major contributor to debris heating. After completion of the oxidation process and with the reduction in decay heat and incorporation of concrete residue into the debris, the molten mass will cool eventually. Although complete cooling may take a long time, it could only take several hours after the consumption of the zirconium for sufficient cooling to occur such that further fission product release would be negligible.

The composition and temperature of the corium and the composition of concrete influence the magnitude and timing of the ex-vessel release and the amount of aerosols carried into the containment atmosphere.

The continuous water pool (if any) overlying the corium during the core-concrete interaction would retain some of the releases from the core-concrete interaction. If the water pool is permitted to boil dry, the release to the containment atmosphere would merely be shifted in time, that is the cumulative aerosol release would be the same as if there had been no water pool.

4. Approach to Development of Updated Source Terms

The approach suggested for estimating source terms into the containment is based on two underlying assumptions. First, the fission product species are grouped according to their respective chemical forms and release characteristics. Secondly, the accident conditions will be categorized into appropriate categories somewhat similar to the approach utilized for the source term analysis used in NUREG-1150.

For simplicity the total radiological release fraction into the containment is represented by

$$ST_{CON}(i) = ST_{INV}(i) + ST_{VB}(i) + ST_{EXV}(i) + ST_{REV}(i) \qquad (4.1)$$

where (i) represents the radionuclide group, $ST_{CON}(i)$ represents the total source term for species (i), $ST_{INV}(i)$ represents release from the reactor coolant system into the containment prior to vessel failure, $ST_{VB}(i)$ represents the releases at vessel breach, $ST_{EXV}(i)$ represents the ex-vessel releases into the containment, primarily during core-concrete interactions, and $ST_{REV}(i)$ represents the releases due to late revolatalization from the reactor coolant system.

When using Equation 4.1, appropriate decontamination factors (DFs) must be applied to account for retention of fission products at various stages in the release path. For example, aerosol fission products would be retained in BWR suppression pools and in any water that might be overlying core debris interacting with concrete.

In the simplified formulation for the appearance rate into the containment, the fission product releases are treated as being proportional to time after the initial release (See Figure 4.1). The in-vessel release duration is assumed to be the time interval extending from core melt initiation to reactor pressure vessel bottom head failure (t_{VF}). Δt_{exV} and Δt_{rev} are the release durations for the ex-vessel releases (due to corium-concrete interactions) and late revolatilization release from the reactor coolant system.

The categorization of radionuclide releases into containment are determined by four key characteristics, namely;

1) Reactor type (BWR versus PWR),
2) RCS pressure prior to pressure vessel breach (high versus low),
3) Concrete aggregate/composition (limestone versus basalt), and
4) Cavity/pedestal condition (dry versus flooded).

The individual terms on the right hand side of equation 4.1 can be represented as products of release fractions and transmission factors:

$$ST_{INV}(i) = FCOR(i) * FVES(i) \qquad (4.2)$$

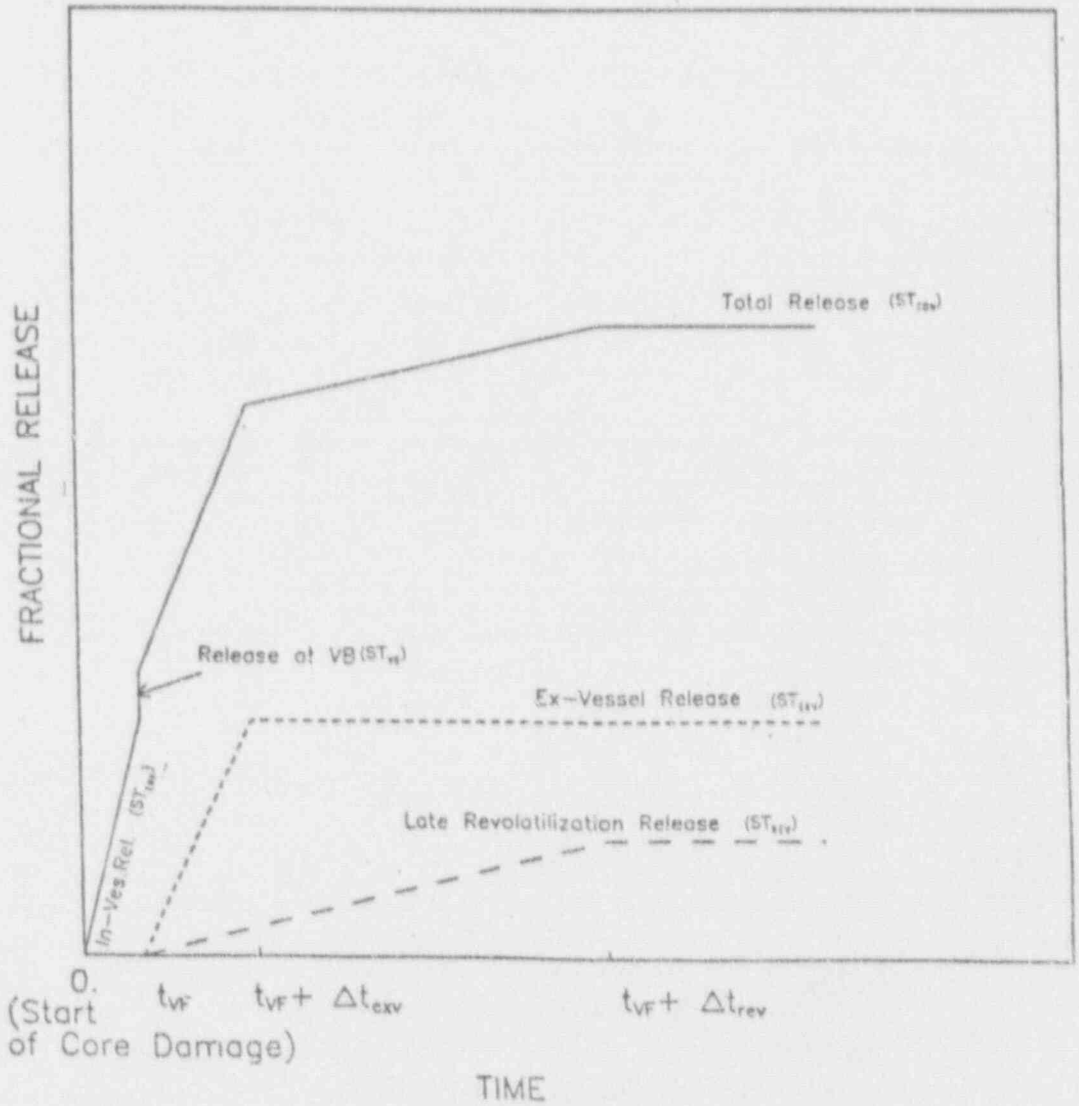


Figure 4.1 Time Variation of Simplified Releases Into Containment

$$ST_{VB}(i) = (1 - FCOR(i)) * FPME * FDCH(i) \tag{4.3}$$

$$ST_{EXV}(i) = FPART * FCCI(i) * (1 - FCOR(i)) \tag{4.4}$$

$$ST_{REV}(i) = FCOR(i) * (1 - FVES(i)) * FREV(i) \tag{4.5}$$

where:

FCOR(i) = fraction of initial core inventory of species (i) released from the core to the vessel before vessel failure,

FVES(i) = fraction of fission products of species (i) released from the core to the vessel which is released into containment at, or before vessel failure,

FPME = fraction of core debris participating in pressurized melt ejection,

FDCH(i) = fraction of fission products of species (i) present in the melt participating in pressure-driven melt expulsion that is released to containment in a direct containment heating event,

FCCI(i) = fraction of the fission products of species (i) released from the melt during core-concrete interaction,

FPART = fraction of the core that participates in core-concrete interaction, and

FREV(i) = fraction of the fission products of species (i) remaining in RCS that is re-volatilized and released to containment late in the accident (after vessel breach).

The parameters entering the updated source term formulations will be derived from the existing data base. These data consists mostly of STCP calculations performed in support of NUREG-1150.

The parametric representation of appearance rate into the containment (Equations 4.1 - 4.5) allows quantitative uncertainty analysis for releases into containment to be performed by using the probability distributions for the important parameters used in NUREG-1150 analysis. Such distributions have been developed using expert judgment to interpret the available data or calculations.

5. Quantification of the Updated Source Term Parameters

An updated formulation for estimating the radionuclide releases into containment was presented in the previous chapter. This chapter focuses on quantifying the parameters to be used in the updated source term model through a detailed examination of the available source term information.

The similarities between the NUREG-1150 method to estimate source terms and that proposed in this study allows the use of NUREG-1150 expert opinion on source term issues to assess the uncertainty inherent in the release estimates. An assessment of the range and distribution of the source terms into containment obtained by NUREG-1150 methodology will also be presented in this chapter.

5.1 Release of Fission Products from the Fuel Into the RCS Before Vessel Breach (FCOR)

The STCP results for fraction of initial core inventory released to the reactor vessel prior to pressure vessel failure (FCOR) for different sequences and plants are tabulated in Tables 5.1 through 5.4. Generally, most or all the volatile species are released from the fuel in the vessel while most of the non-volatile species remain with the fuel and are available for release during the ex-vessel phase of the accident. Higher in-vessel release of tellurium in some accident sequences is due to higher zirconium oxidation when the ECCS is operating (e.g., high pressure coolant injection (HPCI) with steam driven pump in Sequoyah S3HF and S3B sequences) which adds an additional source of steam for zirconium oxidation.

The STCP (CORSOR-M Code) calculation of in-vessel release of fission products from the fuel is based on a model developed from an experimental data base that included work at Oak Ridge National Laboratory and at Karlsruhe in Germany. The CORSOR-M model is first order (i.e., assumes rates are proportional to the amount of fission product remaining). With the exception of tellurium, the first order fission product release coefficients in CORSOR-M depend only on temperature. Tellurium, which reacts readily with the unoxidized zircaloy of the cladding, has been given an additional dependency to account for the ability of unoxidized zircaloy to retain tellurium.

There are large uncertainties associated with the data and fission product release models. As described in Reference [22], the data taken from different sources span several orders of magnitude of release rates for the same fuel temperature. The release of medium and low volatility fission products is even more uncertain. The variability in the data may be most important for mitigated accidents, since the fuel could be maintained at temperatures that do not imply total release for relatively long periods of time. There are other factors such as fuel geometry (surface-to-volume ratio), system pressure, chemical potentials, and flow velocities that can influence the release rates of fission products. The temperature dependent models for fission product release consistently overpredict the rate of release and the total release of fission products for integral heatup and meltdown experiments.

Table 5.1 STCP Results for Fraction of Initial Core Inventory Released to Vessel Prior to RPV Failure (FCOR) PWR, High RCS Pressure Sequences

	Surry		Zion		Sequoyah			OCONEE
	TMLB'	S3B	TMLU	S2DCR/S2DCF	S3HF/S3B	S3B1	TMLB'	TMLB'
NG	0.98	0.98	1.0	0.99	0.97	0.99	0.97	0.99
I	0.98	0.98	1.0	0.99	0.97	0.99	0.97	0.99
Cs	0.98	0.98	1.0	0.99	0.97	0.99	0.97	0.99
Te	0.46	0.3	0.54	0.43	0.84	0.85	0.36	0.3
Sr	$7x10^{-4}$	$5x10^{-4}$	$2x10^{-3}$	$4x10^{-4}$	$6x10^{-4}$	$8x10^{-4}$	$5X10^{-4}$	$7x10^{-4}$
Ba	0.013	0.01	0.02	$8x10^{-3}$	0.01	0.014	0.01	0.013
Ru	10^{-6}	10^{-6}	$2x10^{-6}$	$5x10^{-7}$	10^{-6}	$2x10^{-6}$	10^{-6}	10^{-6}
Ce	0	0	0	0	0	0	0	0
La	10^{-7}	10^{-7}	$2x10^{-7}$	$5x10^{-8}$	10^{-7}	10^{-7}	10^{-7}	10^{-7}

Table 5.2 STCP Results for Fraction of Initial Core Inventory Released to Vessel Prior to RPV Failure (FCOR) PWR, Low RCS Pressure Sequences

	Surry		Sequoyah		OCONEE
	V	AG	TBA	ACD	S1DCF
NG	1.0	1.0	1.0	1.0	1.0
I	1.0	1.0	0.98	1.0	1.0
Cs	1.0	1.0	0.98	1.0	1.0
Te	0.63	0.86	0.80	0.51	0.35
Sr	$1.5x10^{-3}$	10^{-3}	$2x10^{-3}$	10^{-3}	$7x10^{-4}$
Ba	0.03	0.02	0.04	0.01	0.014
Ru	$3x10^{-6}$	$2x10^{-6}$	$3x10^{-6}$	10^{-6}	$\approx 10^{-6}$
Ce	0	0	0	0	0
La	$2x10^{-7}$	$2x10^{-7}$	$3x10^{-7}$	10^{-7}	$\approx 10^{-7}$

Table 5.3 STCP Results for Fraction of Initial Core Inventory Released to Vessel Prior to RPV Failure (FCOR) BWR, High RCS Pressure Sequences

	Peach Bottom				La Salle	Grand Gulf
	TC2	TB	TBUX	S2E	TB	TB
NG	0.87	0.88	0.93	0.90	0.99	0.96
I	0.87	0.88	0.93	0.84	0.99	0.96
Cs	0.87	0.87	0.93	0.84	0.99	0.37
Te	0.62	0.38	0.32	0.19	0.43	0.38
Sr	$5x10^{-4}$	$2x10^{-3}$	$2x10^{-3}$	$4x10^{-4}$	10^{-3}	10^{-3}
Ba	0.01	0.02	0.02	$7x10^{-3}$	0.02	0.02
Ru	10^{-6}	$2x10^{-6}$	$2x10^{-6}$	10^{-6}	$2x10^{-6}$	$2x10^{-6}$
Ce	0	0	0	0	0	0
La	10^{-7}	$2x10^{-7}$	$2x10^{-7}$	5x10[illegible]	$1.5x10^{-7}$	$2x10^{-7}$

Table 5.4 STCP Results for Fraction of Initial Core Inventory Released to Vessel Prior to RPV Failure (FCOR) BWR, Low RCS Pressure Sequences

	Peach Bottom		Grand Gulf	
	TC1	V	TC1	TBS/TBR
NG	0.92	0.83	1.0	1.0
I	0.92	0.83	1.0	0.97
Cs	0.91	0.82	1.0	0.97
Te	0.3	0.12	0.41	0.39
Sr	$6x10^{-4}$	$3x10^{-4}$	$1.1x10^{-3}$	$5x10^{-4}$
Ba	0.01	$5x10^{-3}$	0.02	0.01
Ru	$8x10^{-7}$	$4x10^{-7}$	10^{-6}	$6x10^{-7}$
Ce	0	0	0	0
La	$9x10^{-7}$	$6x10^{-8}$	$2x10^{-7}$	$7x10^{-8}$

Quantification

In addition to uncertainties in the data base for fission product release, there is a significant uncertainty in the prediction of the fuel temperature and the extent of local oxidation during core degradation. Even if there were excellent fission product release models, the boundary conditions and the parameters fed into these release models would be highly uncertain.

The STCP overpredicts the maximum temperature in the core and does not account well for the formation of eutectics and gradual core relocation. Accounting for these processes would tend to reduce the total quantity of fission products released from the fuel.

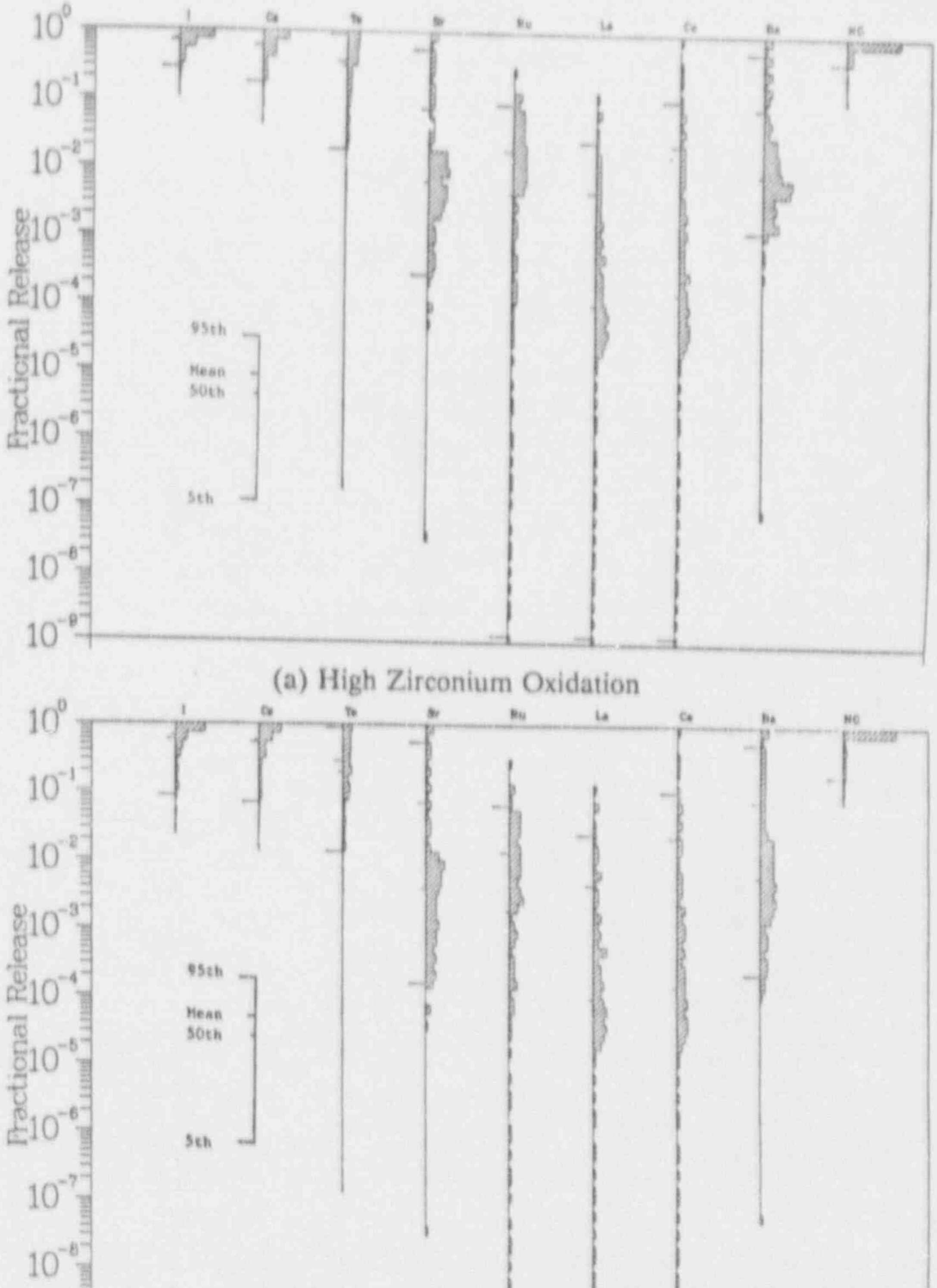
The issue of in-vessel release of fission products from the fuel was assessed by the members of the NUREG-1150 source term expert panel. The results of expert panel elicitation are presented in detail in Reference [23]. Two experts concluded that there were no significant differences between PWRs and BWRs as far as FCOR was concerned. The other two not only made distinctions between BWRs and PWRs, but they also considered a high Zirconium oxidation (greater than 21%) in-vessel subcase and a low Zirconium oxidation (less than 21%) in-vessel subcase for FCOR.

The distribution of release magnitude for the various isotopic groups are shown in Figures 5.1 - 5.2. These distributions have been obtained from aggregate cumulative probability distributions tabulated in Reference [23] using a specialized Monte Carlo sampling method, namely the Latin Hypercube Sampling (LHS). Logarithmic interpolation was used to determine source terms between fractiles. The mean and median values for these distributions are presented in Table 5.5. The estimated fractional releases depend strongly on the volatility of the fission products, as might be expected. The difference between semi-volatile fission products Sr and Ba are not great. Low volatile fission products Ce and La have similar releases. Note that the difference between the high-Zr-oxidation case and the low-Zr-oxidation case is small. Furthermore, except for Tellurium, there is no differences between PWR and BWR in-vessel releases. Due to the large amount of Zircaloy in the BWR core, the tellurium release tends to be less for BWRs.

The uncertainties in release of fission products (except noble gases) are high. Lower bound releases for volatiles fission products (iodine, cesium and tellurium) are small. This is very significant, indicating that substantial quantities of these species could remain in the melt and escape retention within the RCS and the pressure suppression pool (for BWRs) and thus remain substantially accessible for release into the containment atmosphere after corium discharge from the RPV.

5.2 Fission Product Transmission Within the RCS

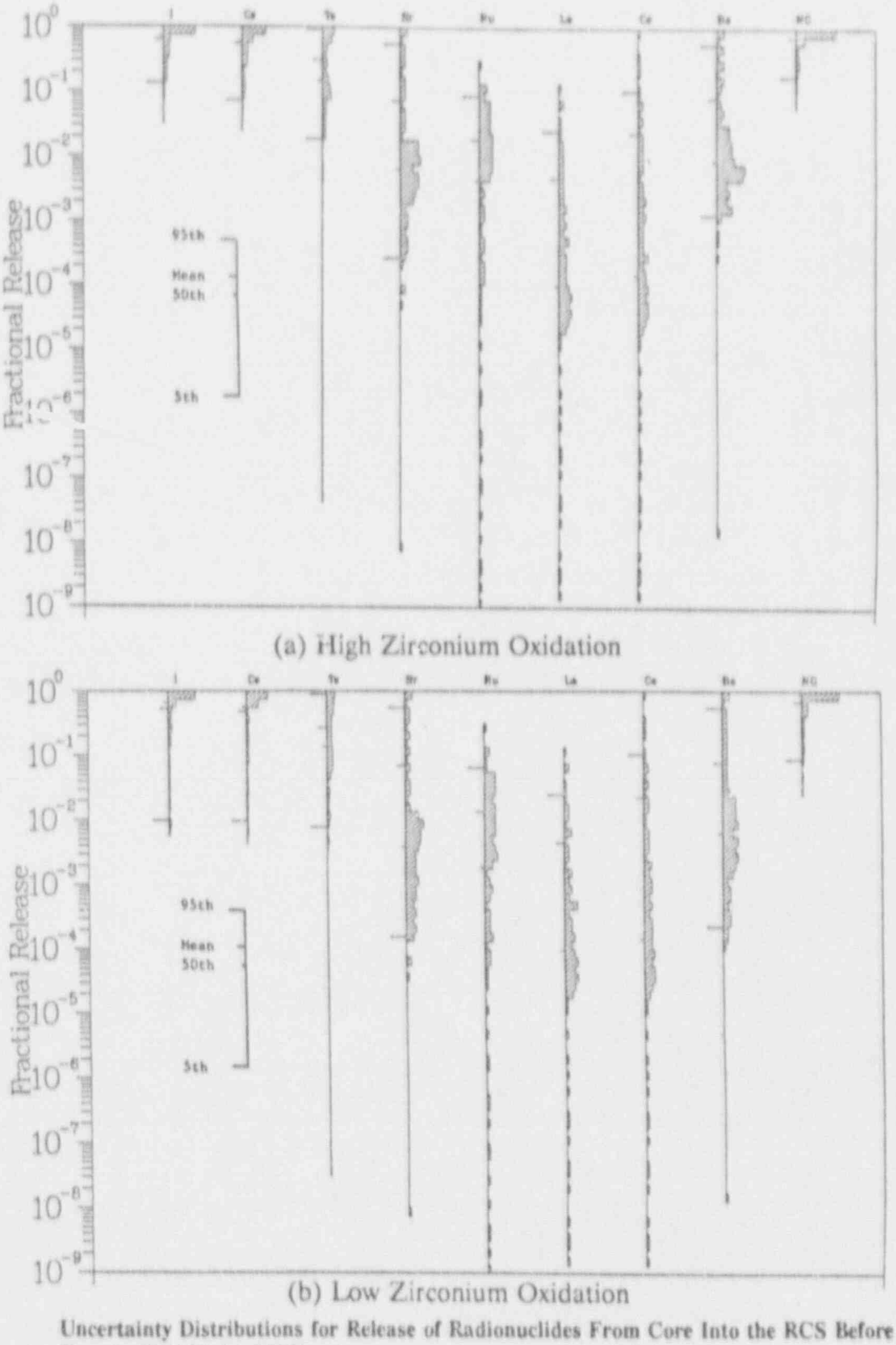
A convenient way to describe the overall effect of retention phenomena in the reactor coolant system is to state the fraction of fission products released from the core that is released into containment prior to vessel failure (FVES). The STCP results for FVES values are shown in Tables 5.6 through 5.9. A comparison of these values indicate that the retention fraction is a function of the fission product group and of the RCS pressure.



(a) High Zirconium Oxidation

(b) Low Zirconium Oxidation

Figure 5.1 **Uncertainty Distributions for Release of Radionuclides From Core Into the RCS Before Vessel Breach (FCOR) for PWRs.**



(a) High Zirconium Oxidation

(b) Low Zirconium Oxidation

Figure 5.2 **Uncertainty Distributions for Release of Radionuclides From Core Into the RCS Before Vessel Breach (FCOR) for BWRs.**

Table 5.5 Mean and Median Values for Fission Product Releases From the Core Into RCS (FCOR)

Conditions		FCOR[a]								
		NG	I	Cs	Te	Sr	Ba	Ru	La	Ce
PWRs	High Zr Oxidation	0.92	0.75	0.62	0.33	0.006	0.009	0.005	0.0001	0.00015
		(0.83)	(0.71)	(0.61)	(0.36)	(0.07)	(0.08)	(0.02)	(0.004)	(0.02)
	Low Zr Oxidation	0.9	0.69	0.58	0.19	0.004	0.006	0.002	0.0001	0.00015
		(0.8)	(0.6)	(0.55)	(0.3)	(0.07)	(0.08)	(0.01)	(0.004)	(0.02)
BWRs	High Zr Oxidation	0.9	0.73	0.58	0.15	0.006	0.009	0.005	0.0001	0.00015
		(0.74)	(0.63)	(0.55)	(0.31)	(0.07)	(0.08)	(0.02)	(0.004)	(0.02)
	Low Zr Oxidation	0.9	0.69	0.58	0.14	0.004	0.006	0.002	0.0001	0.00015
		(0.7)	(0.54)	(0.48)	(0.27)	(0.07)	(0.08)	(0.01)	(0.004)	(0.02)

[a] The mean values are shown in parenthesis.

Table 5.6 STCP Results for Fraction of FCOR Released Into Containment (FVES) PWR, High RCS Pressure Sequences

	Surry		Zion		Sequoyah			OCONEE
	TMLB'	S3B	TMLU	S2DCR/S2DCF	S3HF/S3B	S3B1	TMLB'	TMLB'
NG	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
I	0.22	0.27	0.22	0.28	0.32	0.53	0.36	0.41
Cs	0.21	0.24	0.19	0.28	0.25	0.46	0.32	0.29
Te	0.62	0.24	0.47	0.47	0.09	0.1	0.46	0.3
Sr	0.26	0.22	0.16	0.34	0.25	0.46	0.24	0.21
Ba	0.26	0.22	0.26	0.37	0.25	0.46	0.22	0.24
Ru	0.26	0.22	0.22	0.31	0.25	0.45	0.22	0.24
Ce	0	0	0	0	0	0	0	0
La	0.3	0.22	0.21	0.27	0.23	0.46	0.24	0.24

Table 5.7 STCP Results for Fraction of FCOR Released Into Containment (FVES) PWR, Low RCS Pressure Sequences

	Surry	Sequoyah		OCONEE
	AG	TBA	ACD	S1DCF
NG	1.0	1.0	1.0	1.0
I	0.87	0.94	0.92	0.91
Cs	0.87	0.92	0.92	0.91
Te	0.83	0.84	0.78	0.46
Sr	0.75	0.87	0.83	0.71
Ba	0.78	0.88	0.77	0.71
Ru	0.76	0.89	0.83	0.9
Ce	0	0	0	0
La	0.78	0.86	0.87	0.8

Table 5.8 STCP Results for Fraction of FCOR Released Into Containment (FVES) BWR, High RCS Pressure Sequences

	Peach Bottom				LaSalle	Grand Gulf
	TC2	TB	TBUX	S2E	TB	TB
NG	1.0	0.97	1.0	1.0	1.0	1.0
I	0.90	0.23	0.73	0.69	0.37	0.70
Cs	0.80	0.14	0.38	0.65	0.27	0.54
Te	0.15	0.07	0.04	0.18	0.13	0.13
Sr	0.62	0.09	0.10	0.70	0.23	0.31
Ba	0.60	0.13	0.15	0.70	0.24	0.30
Ru	0.60	0.14	0.20	0.74	0.23	0.31
Ce	0	0	0	0	0	0
La	0.78	0.15	0.20	0.86	0.25	0.31

Table 5.9 STCP Results for Fraction of FCOR Released Into Containment (FVES) BWR, Low RCS Pressure Sequences

	Peach Bottom	Grand Gulf	
	TC1	TC1	TBS/TBR
NG	0.98	1.0	1.0
I	0.81	0.54	0.86
Cs	0.81	0.56	0.83
Te	0.13	0.37	0.31
Sr	0.80	0.35	0.85
Ba	0.80	0.37	0.84
Ru	0.80	0.38	0.86
Ce	0	0	0
La	0.76	0.35	0.87

Quantification

Low pressure sequences are characterized by rapid blowdown of the RCS and with little gravitational settling (the dominant mechanism for aerosol deposition in the reactor coolant system). On the other hand, for high pressure sequences the fission products released from the fuel (with the exception of noble gases) are retained in the reactor coolant system with higher efficiency. The PWR results show a fairly regular trend toward increasing FVES with decreasing RCS pressurization. Trends among the BWR data are less clear and the results are more difficult to interpret. The reduction in the RCS retention (higher FVES values) for the volatile materials in BWR accident sequences illustrate the effect of revolatization because of fission product decay heating of the structures where fission products had originally deposited.

In the Source Term Code Package analyses, the iodine and cesium are assumed to be in the form of CsI and CsOH while tellurium is assumed to be in elemental form. These three species are treated as vapors as they are transported from the core. However, in calculating the transport and retention in the reactor coolant system they can condense on walls and aerosol particles, evaporate from where they have condensed, or become chemically absorbed by the surfaces. The remaining less volatile fission products are treated as aerosols. The STCP does not account for chemical reactions of Cs and I. Several processes have been postulated (and are currently being investigated) to alter the chemical form of iodine. These include but may not be limited to reactions of CsI with borates, metal surfaces, water pools, and the steam-hydrogen atmosphere during a hydrogen burn. Experimental evidence of the release of other forms of Te, e.g. CsTe, has been published. Generally, the treatment of fission product chemistry in the STCP is simplistic when it exists, and there remains a very high degree of uncertainty in the chemical forms of released fission products.

In the STCP modeling of fission product retention in the reactor coolant system (TRAPMELT3 code), the condensation or evaporation of vapor species is calculated by taking the product of a mass transfer coefficient and the difference between the gas phase concentration and the equilibrium vapor concentration of the species at the temperature of surfaces. The rate of adsorption of the volatile species is modeled through empirical deposition velocities which are based on the work of Elrick, Sallach, and others [24]. The CsI is assumed to condense only on metal surfaces whereas the metal-Te reaction is assumed to be much more reactive than the CsI (chemical absorption is modeled).

The STCP treatment of aerosol behavior within the reactor coolant system (TRAPMELT3 code) includes models for different processes of agglomeration (i.e., Brownian, gravitational, and turbulent agglomeration) as well as natural removal mechanisms (i.e., Brownian, gravitational, turbulent and thermophoretic deposition).

Retention within the RCS was considered by the NUREG-1150 source term expert panel. It was proposed that FVES depended upon the pressure in the vessel as well as upon the type of reactor. Thus, for FVES, four cases were proposed for the PWRs, and three cases were proposed for the BWRs:

PWR-1 -- System setpoint pressure (2500 psia); release through a cycling PORV.
PWR-2 -- High pressure (600 to 2000 psia); release through a very small break or pump seal LOCA.
PWR-3 -- Intermediate Pressure (200 to 600 psia); release through a break of approximately two inches diameter.
PWR-4 -- Low pressure (below 200 psia); release through a large break.

BWR-1 -- High pressure fast station blackout (TBUX).
BWR-2 -- Low pressure fast station blackout (TBU).
BWR-3 -- High pressure ATWS sequences (TCUX).

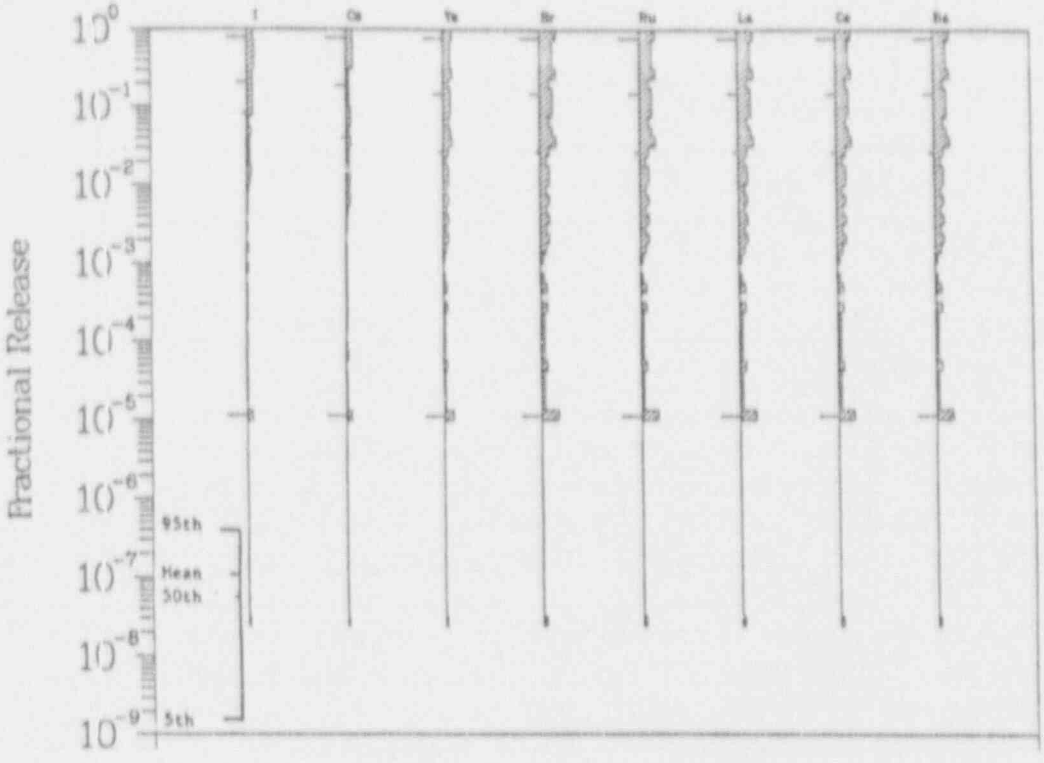
The panel agreed before elicitation that cases PWR-2 and PWR-3 could be considered together. The detailed results of expert panel elicitation may be found in Reference [23]. The uncertainty distributions of FVES values are shown in Figure 5.3 - 5.4. These distributions were obtained from aggregate cumulative probability distributions tabulated in Reference [23], using the LHS sampling method. Logarithmic interpolation was used to determine FVES values between fractiles. The mean and median values for these distributions are presented in Table 5.10. The estimated values for FVES depends strongly on the RCS pressure during the release, as might be expected. The uncertainties in retention of fission products (except noble gases) within the RCS are high. There is much uncertainty as to the kinetics and mechanics of the interactions of volatile fission products within RCS gases and on solid structures. These uncertainties are compounded by uncertainties about aerosol agglomeration and deposition rates and chemical interactions of fission products on the RCS structural surfaces.

5.3 Summary of In-Vessel Releases Into the Containment at, or Before Vessel Breach

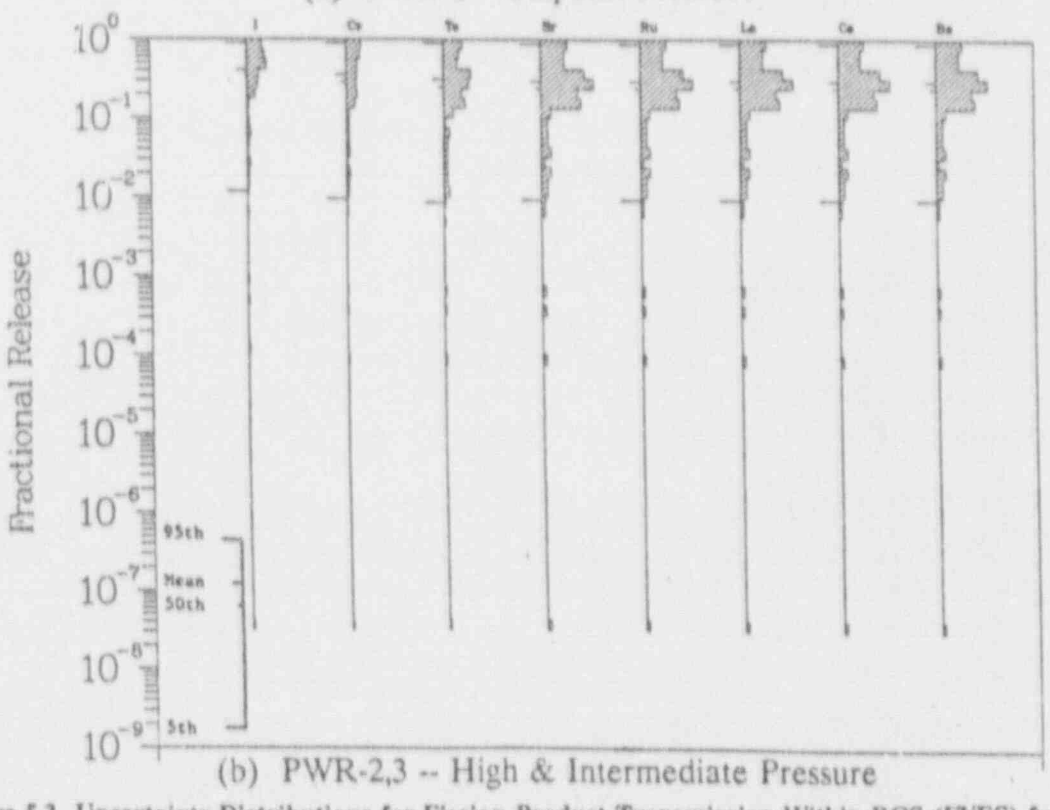
Since FCOR and FVES are correlated in a phenomenological sense, it is more reasonable to present the STCP results in terms of ST_{INV} (FCOR*FVES). The STCP results for fraction of initial core inventory released from the vessel into the containment at, or before, vessel failure (ST_{INV}) are tabulated in Table 5.11 through 5.14. In the STCP modeling, it is assumed that the fission products in the atmosphere of the RCS are released instantaneously to the containment as a puff release at the time of vessel failure.

The NUREG-1150 source term expert panel did not identify any correlation between the FCOR and FVES distributions that they provided. The mean and median values for the distributions of in-vessel releases into the containment (ST_{INV}) are presented in Tables 5.15 and 5.16. These values were obtained from propagation of the uncertainty distributions for FCOR and FVES (discussed in previous sections), using the LHS Sampling method.

The estimated fractional releases depends strongly on the volatility of the fission products, as might be expected. Volatile fission products, iodine and cesium, have similar releases. The difference between semi-volatile fission products Sr and Ba are not great. Low volatile fission products Ce and La have also similar releases.

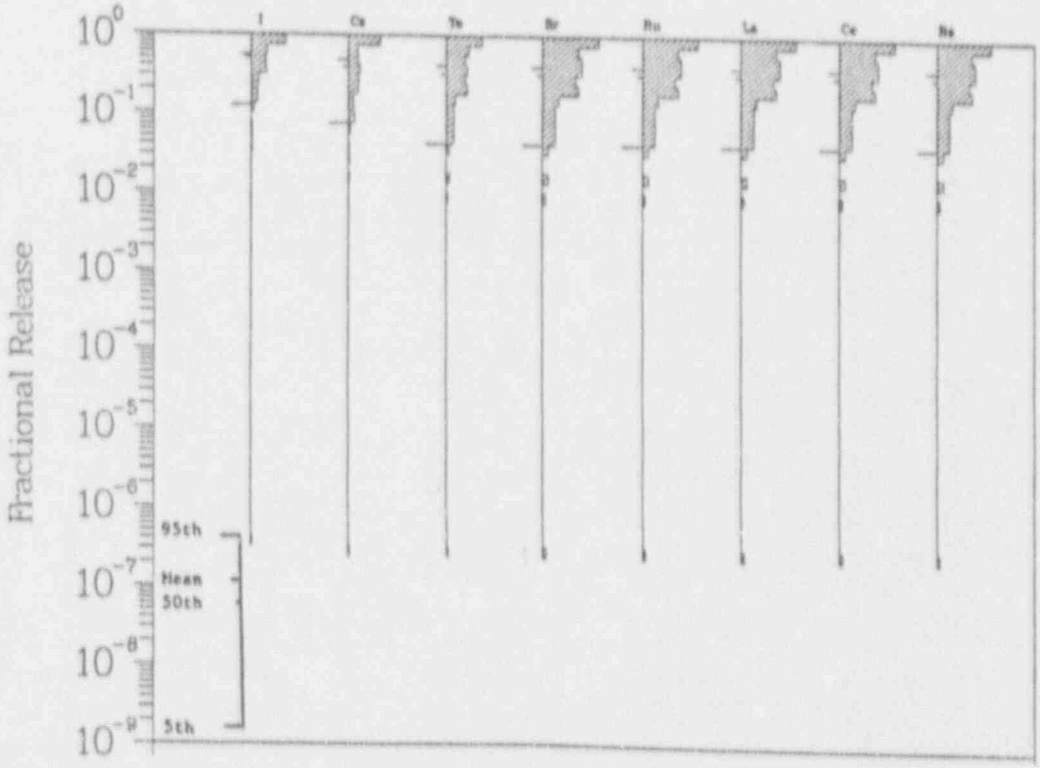


(a) PWR-1 -- Setpoint Pressure



(b) PWR-2,3 -- High & Intermediate Pressure

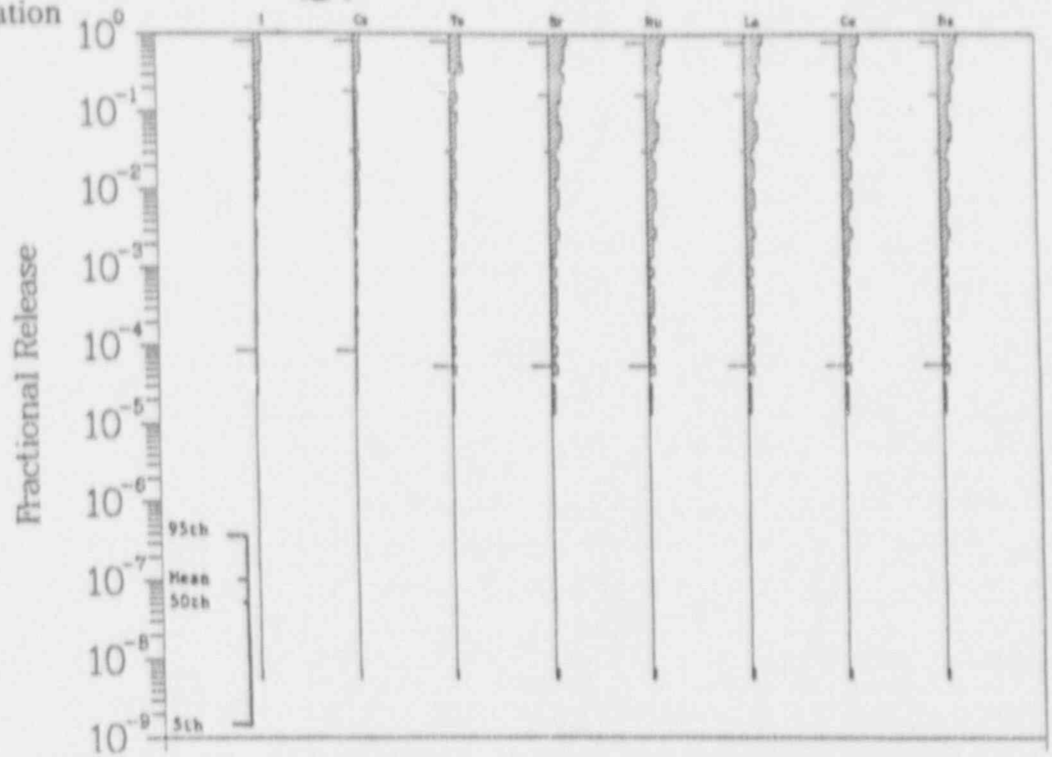
Figure 5.3 Uncertainty Distributions for Fission Product Transmission Within RCS (FVES) for PWRs.



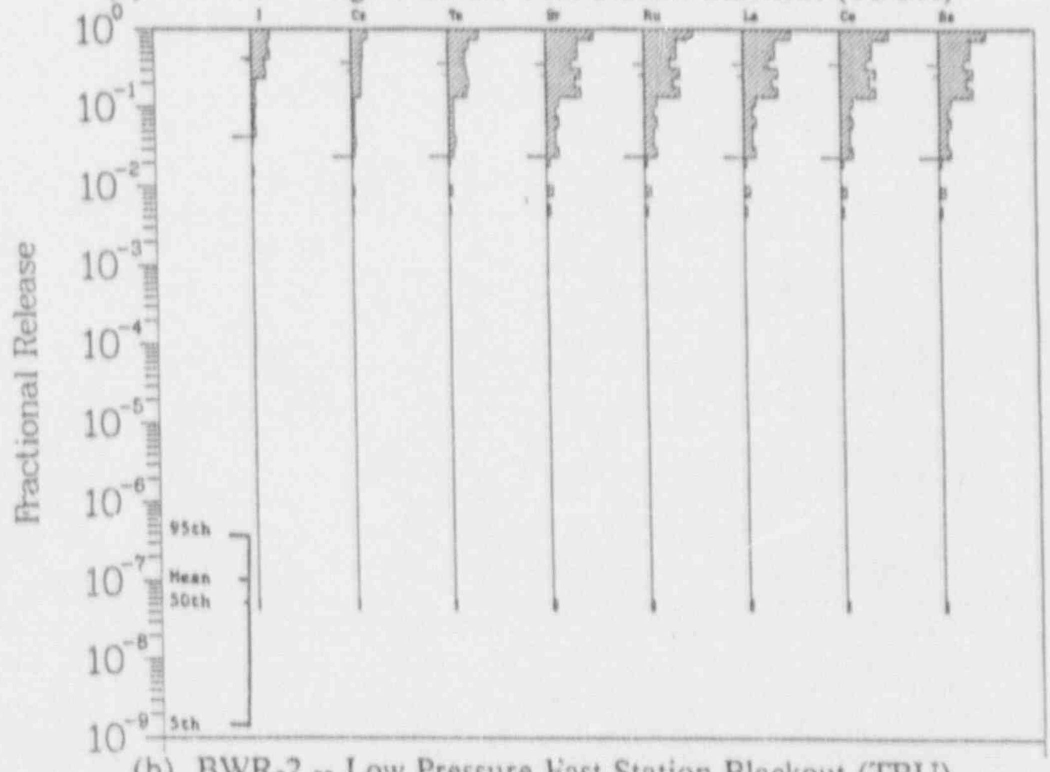
(c) PWR-4 Low Pressure

Figure 5.3 Uncertainty Distributions for Fission Product Transmission Within RCS (FVES) for PWRs (Continued)

Quantification

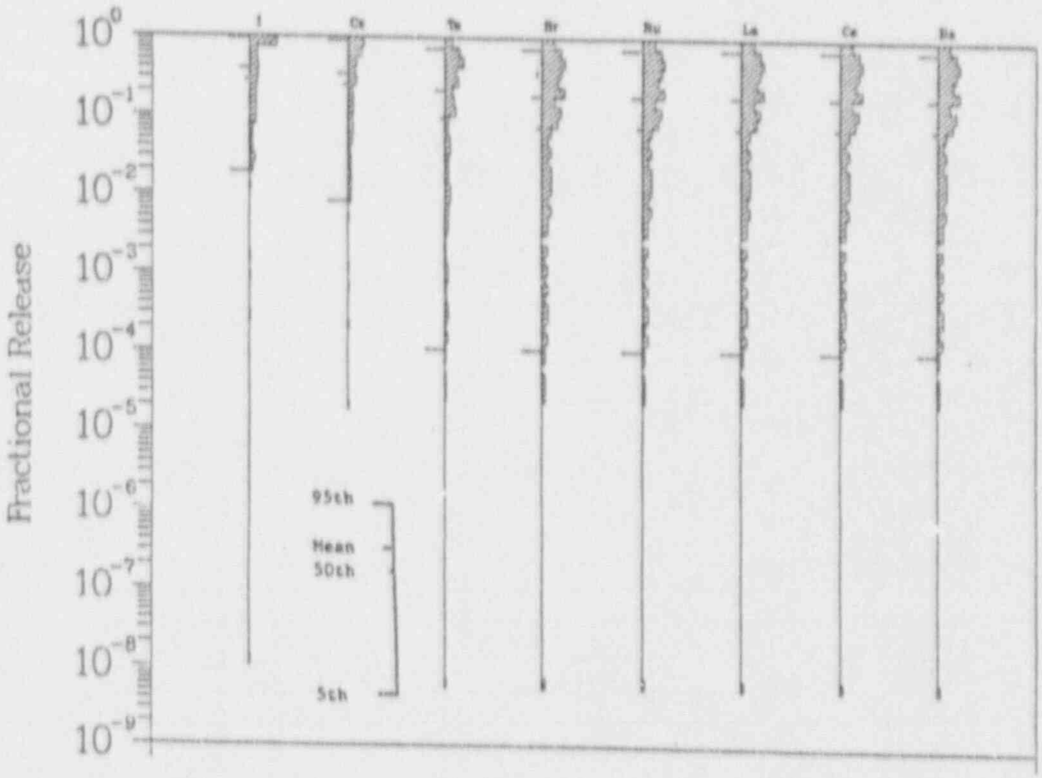


(a) BWR-1 -- High Pressure Fast Station Blackout (TBUX)



(b) BWR-2 -- Low Pressure Fast Station Blackout (TBU)

Figure 5.4 Uncertainty Distributions for Fission Product Transmission Within RCS (FVES) for BWRs.



(c) BWR-3 -- High Pressure ATWS Sequences (TCUX)

Figure 5.4 Uncertainty Distributions for Fission Product Transmission Within RCS (FVES) for BWRs. (Continued)

Table 5.10 Mean and Median Values for Fission Product Transmission Within RCS (FVES)

	Conditions	FVES[a]								
		NG	I	Cs	Te	Sr	Ba	Ru	La	Ce
PWRs										
	Setpoint Pressure	1.0	0.09 (0.2)	0.04 (0.19)	0.03 (0.15)	0.03 (0.15)	0.03 (0.15)	0.03 (0.15)	0.03 (0.15)	0.03 (0.15)
	High & Intermediate Pressure	1.0	0.41 (0.4)	0.29 (0.36)	0.25 (0.3)	0.24 (0.3)	0.24 (0.3)	0.24 (0.3)	0.24 (0.3)	0.24 (0.3)
	Low Pressure	1.0	0.52 (0.55)	0.40 (0.48)	0.33 (0.4)	0.33 (0.4)	0.33 (0.4)	0.33 (0.4)	0.33 (0.4)	0.33 (0.41)
BWRs										
	Fast, High Pressure	1.0	0.08 (0.2)	0.03 (0.18)	0.03 (0.17)	0.03 (0.17)	0.03 (0.17)	0.03 (0.17)	0.03 (0.17)	0.03 (0.18)
	Fast, Low Pressure	1.0	0.41 (0.44)	0.30 (0.38)	0.27 (0.36)	0.27 (0.36)	0.26 (0.36)	0.26 (0.36)	0.26 (0.36)	0.26 (0.37)
	Slow, High Pressure CRD	1.0	0.28 (0.4)	0.25 (0.34)	0.10 (0.18)	0.10 (0.18)	0.08 (0.18)	0.08 (0.18)	0.08 (0.18)	0.08 (0.2)

[a] Mean values are shown in parenthesis.

Table 5.11 STCP Results for Fraction of Initial Core Inventory Released Into Containment Including Puff Release (ST_{INV}) PWR, High RCS Pressure Sequences

	Surry		Zion		Sequoyah			OCONEE
	TMLB'	S3B	TMLU	S2DCR/S2DCF	S3HF/S3B	S3B1	TMLB'	TMLB'
NG	0.98	0.98	1.0	0.99	0.97	0.99	0.97	0.99
I	0.22	0.26	0.22	0.28	0.31	0.52	0.35	0.40
Cs	0.20	0.24	0.19	0.28	0.24	0.46	0.31	0.29
Te	0.28	0.072	0.25	0.20	0.076	0.085	0.17	0.09
Sr	$2x10^{-4}$	$1x10^{-4}$	$3x10^{-4}$	$1x10^{-4}$	$1x10^{-4}$	$4x10^{-4}$	$1x10^{-4}$	$1x10^{-4}$
Ba	$3x10^{-3}$	$2x10^{-3}$	$5x10^{-3}$	$3x10^{-3}$	$2x10^{-3}$	$6x10^{-3}$	$2x10^{-3}$	$3x10^{-3}$
Ru	$3x10^{-7}$	$2x10^{-7}$	$4x10^{-7}$	$2x10^{-7}$	$2x10^{-7}$	$9x10^{-7}$	$2x10^{-7}$	$2.6x10^{-7}$
Ce	0	0	0	0	0	0	0	0
La	$3x10^{-8}$	$2x10^{-8}$	$4x10^{-8}$	10^{-6}	$2x10^{-8}$	$5x10^{-7}$	$2x10^{-8}$	$2x10^{-8}$

Table 5.12 STCP Results for Fraction of Initial Core Inventory Released Into Containment Including Puff Release (ST_{INV}) PWR, Low RCS Pressure Sequences

	Surry	Sequoyah		OCONEE
	AG	TBA	ACD	S1DCF
NG	1.0	1.0	1.0	1.0
I	0.87	0.92	0.92	0.91
Cs	0.87	0.90	0.92	0.91
Te	0.71	0.67	0.40	0.16
Sr	$8x10^{-4}$	$2x10^{-3}$	$8x10^{-4}$	$5x10^{-4}$
Ba	0.02	0.04	$7x10^{-3}$	$9x10^{-3}$
Ru	10^{-6}	10^{-6}	10^{-6}	10^{-6}
Ce	0	0	0	0
La	$2x10^{-7}$	$3x10^{-7}$	$9x10^{-8}$	$8x10^{-8}$

Table 5.13 STCP Results for Fraction of Initial Core Inventory Released Into Containment Including Puff Release (ST_{INV}) BWR, High RCS Pressure Sequences

	Peach Bottom				LaSalle	Grand Gulf
	TC2	TB	TBUX	S2E	TB	TB
NG	.87	0.85	0.93	0.90	0.99	0.96
I	.78	0.20	0.68	0.58	0.37	0.67
Cs	.70	0.12	0.35	0.55	0.27	0.52
Te	.09	0.03	0.01	0.03	0.05	0.05
Sr	$3x10^{-4}$	$2x10^{-4}$	$2x10^{-4}$	$3x10^{-4}$	$2x10^{-4}$	$3x10^{-4}$
Ba	$6x10^{-3}$	$3x10^{-3}$	$3x10^{-3}$	$5x10^{-3}$	$4x10^{-3}$	$6x10^{-3}$
Ru	$6x10^{-7}$	$3x10^{-7}$	$4x10^{-7}$	$7x10^{-7}$	$5x10^{-7}$	$5x10^{-7}$
Ce	0	0	0	0	0	0
La	$8x10^{-8}$	$3x10^{-8}$	$4x10^{-8}$	$4x10^{-8}$	$4x10^{-8}$	$5x10^{-8}$

Table 5.14 STCP Results for Fraction of Initial Core Inventory Released Into Containment Including Puff Release (ST_{INV}) BWR, Low RCS Pressure Sequences

	Peach Bottom	Grand Gulf	
	TC1	TC1	TBS/TBR
NG	0.90	1.0	1.0
I	0.74	0.54	0.83
Cs	0.74	0.56	0.80
Te	0.04	0.15	0.12
Sr	5×10^{-4}	4×10^{-4}	4×10^{-4}
Ba	8×10^{-3}	7×10^{-3}	8×10^{-3}
Ru	6×10^{-7}	4×10^{-7}	5×10^{-7}
Ce	0	0	0
La	2×10^{-8}	7×10^{-8}	6×10^{-8}

Table 5.15 Mean and Median Values of In-vessel Releases Into Containment up to Vessel Breach (ST_{INV}) for PWRs

Conditions		ST_{INV}[c]								
RCS Pressure[a]	Zr Oxidation[b]	NG	I	Cs	Te	Sr	Ba	Ru	La	Ce
SP	H	0.92 (0.83)	0.05 (0.14)	0.02 (0.11)	0.007 (0.05)	1.5×10^{-4} (0.01)	3×10^{-4} (0.015)	2×10^{-5} (3×10^{-3})	3×10^{-6} (9×10^{-4})	3×10^{-6} (4×10^{-3})
SP	L	0.9 (0.79)	0.04 (0.12)	0.02 (0.1)	0.005 (0.04)	8×10^{-5} (0.01)	10^{-4} (0.01)	10^{-5} (2×10^{-3})	2×10^{-6} (9×10^{-4})	3×10^{-6} (4×10^{-3})
H & I	H	0.92 (0.83)	0.26 (0.29)	0.16 (0.22)	0.06 (0.11)	10^{-3} (0.025)	2×10^{-3} (0.03)	6×10^{-4} (6×10^{-3})	2×10^{-5} (1.5×10^{-3})	3×10^{-5} (8×10^{-3})
H & I	L	0.9 (0.79)	0.18 (0.24)	0.12 (0.2)	0.04 (0.09)	7×10^{-4} (0.02)	10^{-3} (0.03)	3×10^{-4} (4×10^{-3})	2×10^{-5} (1.5×10^{-3})	3×10^{-5} (8×10^{-3})
L	H	0.92 (0.83)	0.34 (0.39)	0.21 (0.3)	0.08 (0.15)	2×10^{-3} (0.03)	3×10^{-3} (0.04)	10^{-3} (8×10^{-3})	3×10^{-5} (2×10^{-3})	6×10^{-5} (0.01)
L	L	0.9 (0.79)	0.26 (0.33)	0.17 (0.26)	0.06 (0.12)	10^{-3} (0.03)	2×10^{-3} (0.04)	4.5×10^{-4} (6×10^{-3})	3×10^{-5} (2×10^{-3})	5×10^{-5} (0.01)

[a] SP, H & I and L refer to setpoint, High and Intermediate, and Low RCS Pressure, respectively.

[b] H and L refer to High and Low In-Vessel Zr Oxidation.

[c] The mean values are presented in parenthesis.

Table 5.16 Mean and Median Values of In-vessel Releases Into Containment up to Vessel Breach (ST_{INV}) for BWRs

Conditions	ST_{INV}[a]								
	NG	I	Cs	Te	Sr	Ba	Ru	La	Ce
Fast, High Pressure, High Zr Oxidation	0.9 (0.7)	0.04 (0.12)	0.02 (0.09)	0.004 (0.05)	$2x10^{-4}$ (0.02)	$4x10^{-4}$ (0.02)	$2x10^{-5}$ ($3x10^{-3}$)	$2x10^{-6}$ (10^{-3})	$3x10^{-6}$ ($4x10^{-3}$)
Fast, High Pressure, Low Zr Oxidation	0.9 (0.8)	0.02 (0.1)	0.009 (0.08)	0.004 (0.05)	10^{-4} (0.015)	$2x10^{-4}$ (0.02)	$2x10^{-5}$ ($2x10^{-3}$)	$2x10^{-6}$ ($9x10^{-4}$)	$3x10^{-6}$ ($4x10^{-3}$)
Fast, Low Pressure, High Zr Oxidation	0.9 (0.7)	0.21 (0.27)	0.11 (0.2)	0.04 (0.11)	$2x10^{-3}$ (0.03)	$3x10^{-3}$ (0.03)	$7x10^{-4}$ ($7x10^{-3}$)	$3x10^{-5}$ ($2x10^{-3}$)	$4x10^{-5}$ ($9x10^{-3}$)
Fast, Low Pressure, Low Zr Oxidation	0.9 (0.8)	0.16 (0.23)	0.09 (0.17)	0.04 (0.1)	$8x10^{-4}$ (0.03)	$2x10^{-3}$ (0.03)	$4X10^{-4}$ ($5X10^{-3}$)	$3X10^{-5}$ ($2X10^{-3}$)	$4X10^{-5}$ ($9X10^{-3}$)
Slow, High Pressure, High Zr Oxidation	0.9 (0.7)	0.13 (0.24)	0.08 (0.18)	0.01 (0.06)	$3x10^{-4}$ (0.02)	$7x10^{-4}$ (0.02)	$4.5x10^{-5}$ ($3x10^{-3}$)	$6x10^{-6}$ (10^{-3})	$8x10^{-6}$ ($5x10^{-3}$)
Slow, High Pressure, Low Zr Oxidation	0.9 (0.8)	0.08 (0.2)	0.045 (0.16)	0.007 (0.05)	10^{-4} (0.015)	$3x10^{-4}$ (0.02)	$3x10^{-5}$ ($3x10^{-3}$)	$6x10^{-6}$ (10^{-3})	$8x10^{-6}$ ($5x10^{-3}$)

[a] The mean values are presented in parenthesis.

5.4 Radionuclide Releases Associated With High Pressure Melt Expulsion From the Reactor Pressure Vessel

As was discussed in Chapter 3, if the reactor coolant system (RCS) is pressurized at the time of vessel breach, molten corium will be ejected under pressure in a process which can result in significant aerosol generation. The phenomena of radionuclide release associated with the pressure-driven expulsion of the melt from the RCS is not modeled in the STCP. However, these releases were included, through the parameters of FPME and FDCH, in the present formulation in a manner similar to the approach utilized in NUREG-1150. The quantity that is added to the in-containment source term (see Chapter 4) is:

$$ST_{VB}(i) = (1.-FCOR(i)) * FPME * FDCH(i) \tag{5.1}$$

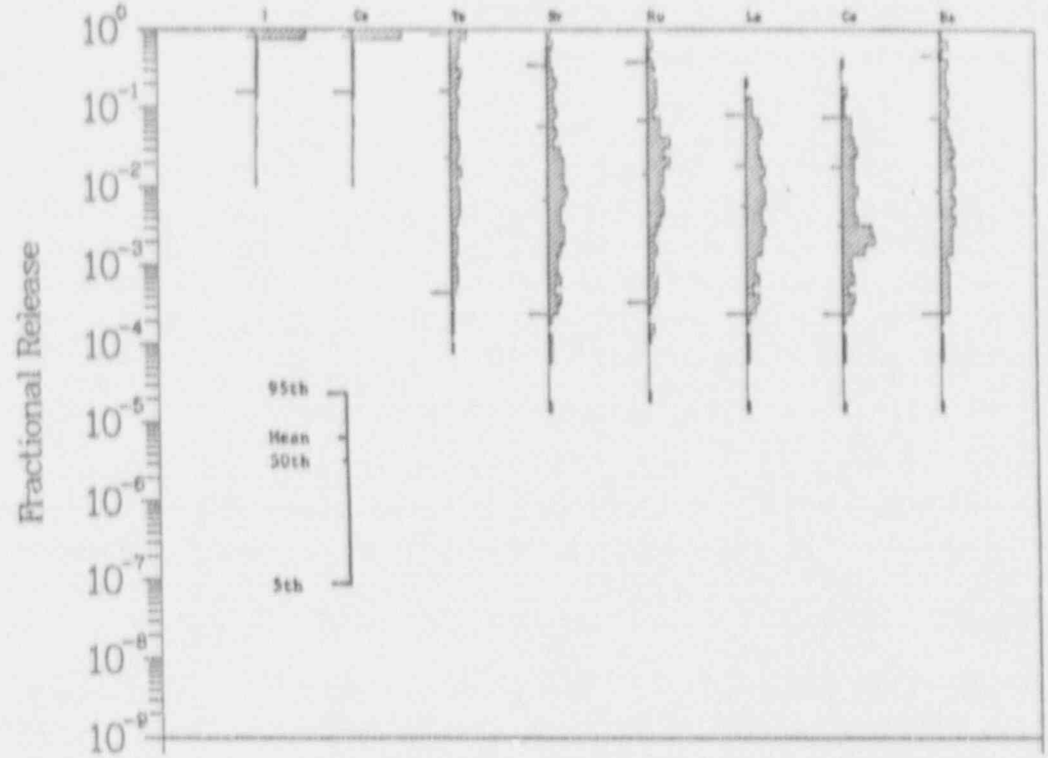
The debris mass and the mode of debris ejection into the reactor cavity/pedestal region cannot be deduced from the results of the STCP analysis. STCP does not include the complete physical mechanisms for these processes; it assumes that 100% of the core is deposited into the cavity/pedestal region at RPV failure in a coherent manner.

The issue of radionuclide release associated with pressurized melt expulsion from the RCS was assessed by the members of the NUREG-1150 source term panel. The detailed results of expert panel elicitation may be found in Reference [23]. The uncertainty distributions for the fraction of the inventory of radionuclide group i that is released to containment as a result of pressure-driven melt expulsion, FDCH(i) are shown in Figures 5.5 through 5.7. These distributions were obtained from aggregate cumulative probability distributions tabulated in Reference [23], using the LHS sampling method. Logarithmic interpolation was used to determine FDCH values between fractiles. The mean and median values for these distributions are also presented in Table 5.17. The estimated fractional releases depend strongly on the volatility of fission products. Most of the volatile fission products are released instantaneously at vessel breach. Note that the differences between high and intermediate pressure cases are small. Furthermore, there is not much difference between PWR and BWR releases.

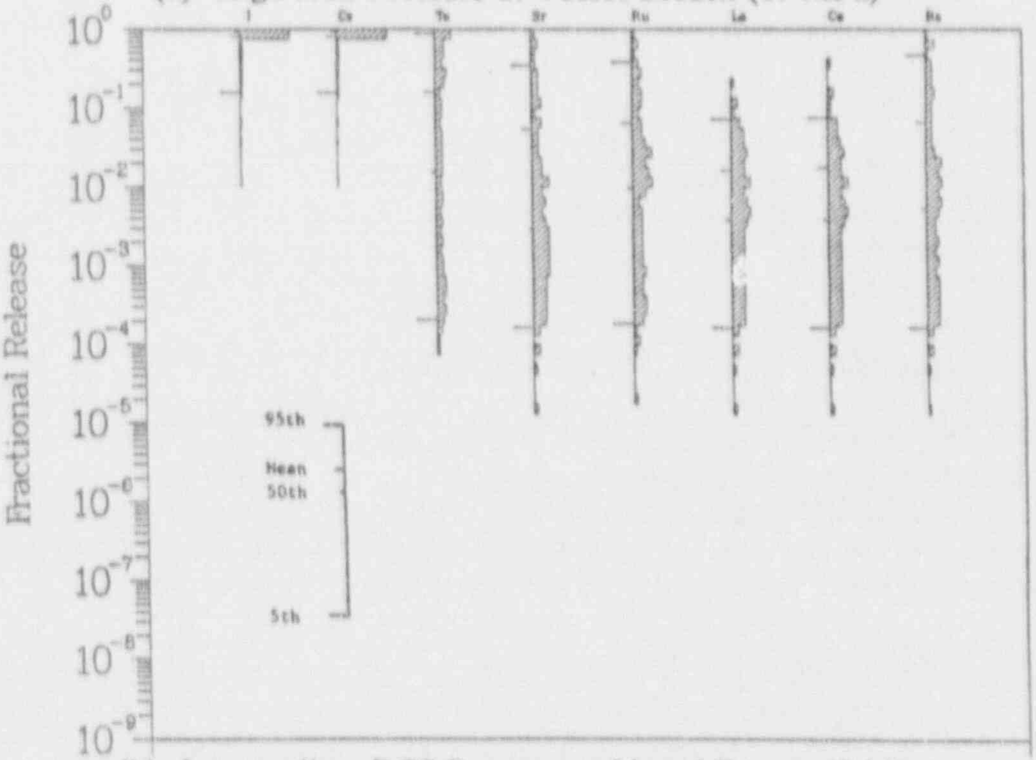
The mean and median values for the distributions of ST_{VB} (See Equation 5.1) are presented in Table 5.18. These values were obtained from propagation of the uncertainty distributions for FCOR and FDCH, assuming a value of 0.4 for FPME (the fraction of core debris participating in pressure melt expulsion). This value corresponds to the "large" fraction of core participating in high pressure melt expulsion assumed in NUREG-1150. Higher values of FDCH corresponding to Surry cases were used to calculate the distributions. It should be noted that the releases associated with HPME are not expected for low RCS pressure sequences.

5.5 Ex-vessel Releases Into the Containment Due to Core-concrete Interaction

Following reactor pressure vessel failure, the high temperature core debris (corium) will fall into the reactor cavity/pedestal where it will interact with structural concrete and any water that may be present. The corium may contain large amounts of unoxidized metals (Zircaloy clad

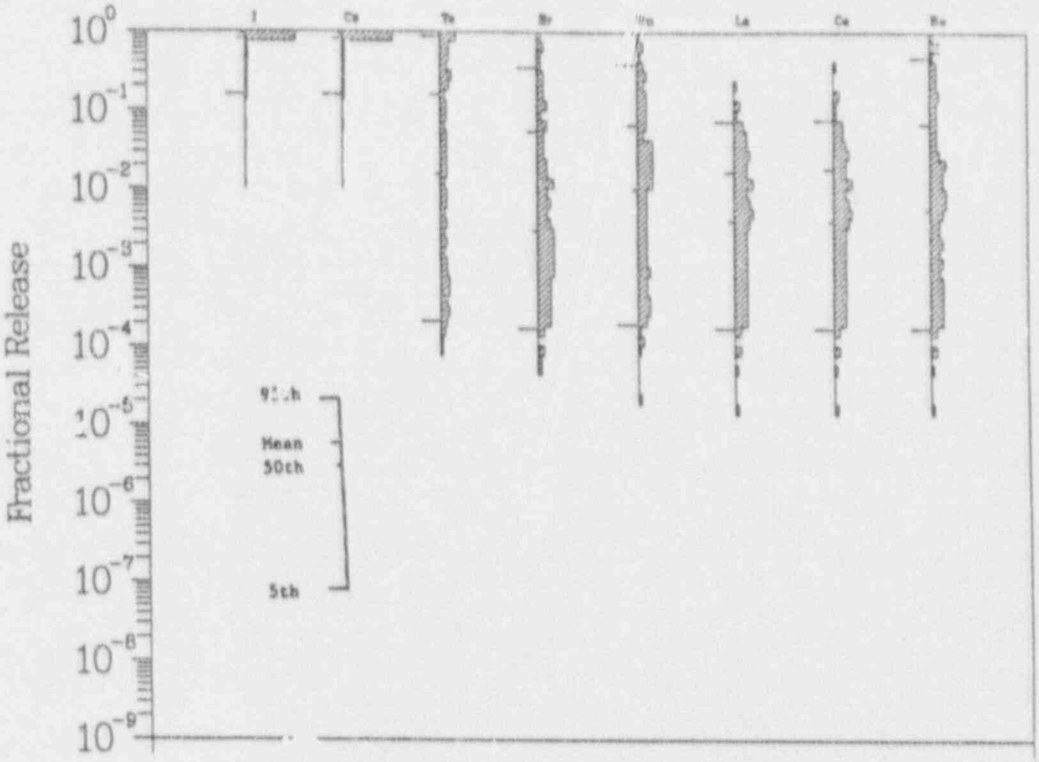


(a) High RCS Pressure at Vessel Breach (17 MPa)

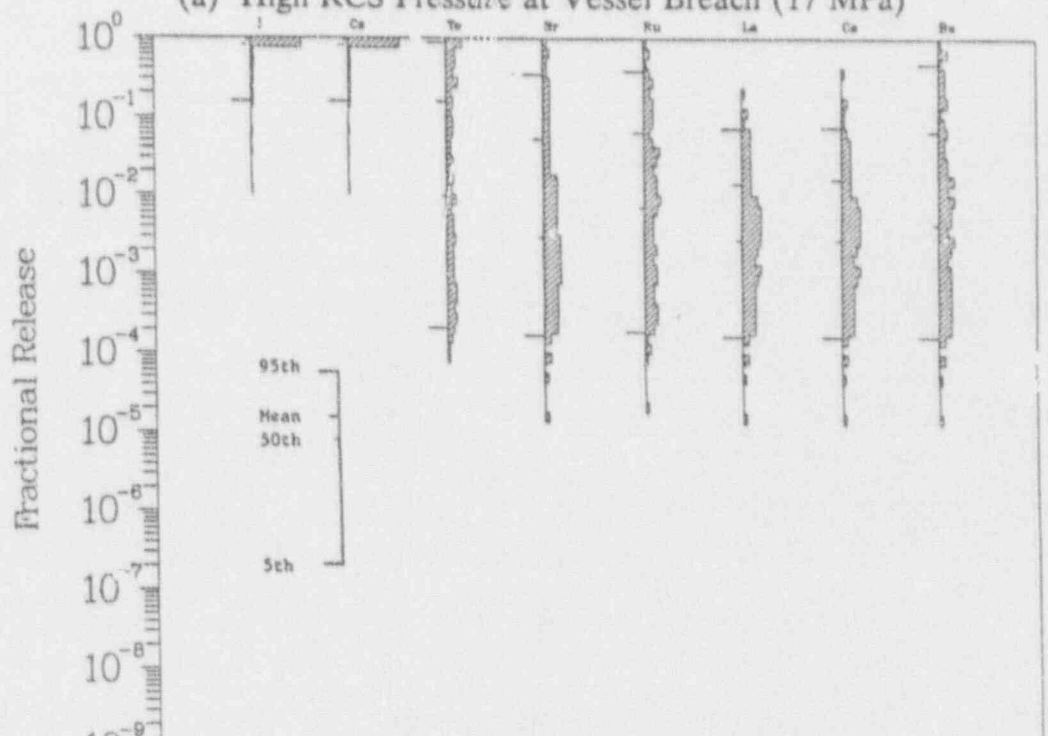


(b) Intermediate RCS Pressure at Vessel Breach (7 MPa)

Figure 5.5 Uncertainty Distributions of FDCH(I) for the Zion and Surry Plants



(a) High RCS Pressure at Vessel Breach (17 MPa)



(b) Intermediate RCS Pressure at Vessel Breach (7 MPa)

Figure 5.6 Uncertainty Distributions of FDCH(i) for the Sequoyah Plant

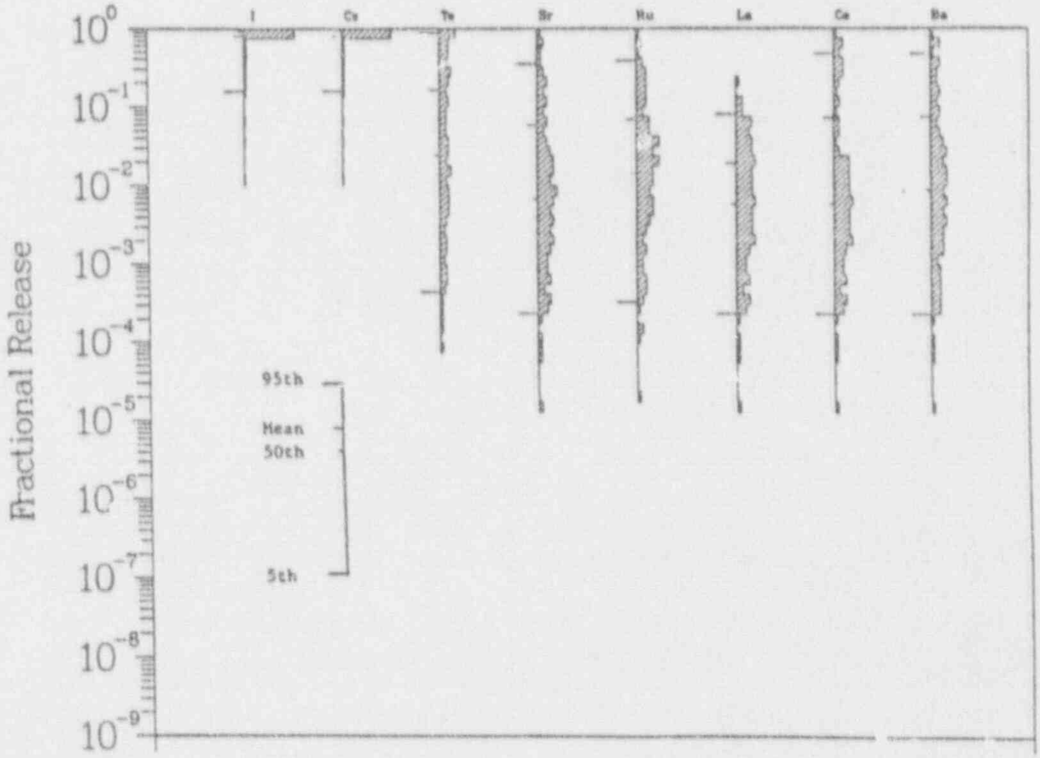


Figure 5.7 Uncertainty Distributions of FDCH(i) for BWRs

Table 5.17 Mean and Median Values of Fraction of Fission Product Species (i) Present in the Melt Participating in HPME That is Released to Containment in a Direct Containment Heating Event (FDCH)

	Plant	RCS Pressure[a] at Vessel Breach	FDCH[b] NG	I	Cs	Te	Sr	Ba	Ru	La	Ce
PWRs	Zion & Surry	H	1.0 (1.0)	0.94 (0.80)	0.94 (0.80)	0.025 (0.16)	0.007 (0.06)	0.009 (0.07)	0.015 (0.07)	0.006 (0.02)	0.003 (0.02)
		I	1.0 (1.0)	0.94 (0.80)	0.94 (0.80)	0.016 (0.16)	0.003 (0.05)	0.006 (0.07)	0.01 (0.06)	0.004 (0.02)	0.004 (0.02)
	Sequoyah	H	1.0 (1.0)	0.94 (0.80)	0.94 (0.80)	0.016 (0.16)	0.003 (0.05)	0.006 (0.07)	0.01 (0.07)	0.004 (0.02)	0.004 (0.02)
		I	1.0 (1.0)	0.94 (0.80)	0.94 (0.80)	0.01 (0.15)	0.003 (0.05)	0.005 (0.07)	0.007 (0.06)	0.003 (0.01)	0.003 (0.02)
BWRs		H	1.0 (1.0)	0.94 (0.80)	0.94 (0.80)	0.025 (0.16)	0.007 (0.06)	0.009 (0.07)	0.015 (0.07)	0.006 (0.02)	0.006 (0.07)

a) H & I refer to high (2465 Psig) and Intermediate (1015 Psig) RCS Pressure, respectively.

b) Mean values are presented in parenthesis.

Table 5.18 Mean and Median Values of Releases Into Containment at Vessel Breach Due to High Pressure Melt Ejection (ST_{VB})

Conditions	ST_{VB}							
	I	Cs	Te	Sr	Ba	Ru	La	Ce
PWRs (Surry)								
High Pressure, High Zr Oxidation	0.007 (0.09)	0.10 (0.12)	[illegible] (0.04)	0.0025 (0.02)	0.003 (0.03)	0.006 (0.03)	0.002 (0.007)	0.001 (0.007)
High Pressure, Low Zr Oxidation	0.09 (0.12)	0.11 (0.14)	0.005 (0.05)	0.0025 (0.01)	0.003 (0.03)	0.006 (0.03)	0.002 (0.007)	0.001 (0.007)
Int. Pressure, High Zr Oxidation	0.07 (0.09)	0.15 (0.12)	0.003 (0.04)	0.001 (0.02)	0.002 (0.02)	0.004 (0.03)	0.002 (0.006)	0.001 (0.007)
Int. Pressure, Low Zr Oxidation	0.09 (0.12)	0.11 (0.14)	0.004 (0.05)	0.001 (0.02)	0.002 (0.02)	0.004 (0.02)	[illegible]002 (0.06)	0.001 (0.07)
BWRs								
High Pressure, High Zr Oxidation	0.07 (0.12)	0.10 (0.14)	0.005 (0.07)	0.0075 (0.02)	0.003 (0.03)	0.006 (0.03)	0.002 (0.007)	0.002 (0.02)
High Pressure, Low Zr Oxidation	0.07 (0.14)	0.11 (0.17)	0.005 (0.05)	0.0025 (0.02)	0.003 (0.03)	0.006 (0.03)	0.009 (0.008)	0.009 (0.02)

ding, stainless steel from fuel assemblies (PWRs), and steel from the core support plate and vessel bottom head) as well as oxides such as UO_2 and ZrO_2. The consequences of these thermal and chemical core-concrete interactions may significantly impact the radionuclide releases into the containment atmosphere.

Aerosols are generated from the interactions of molten core debris with concrete. As the concrete is ablated, water vapor and carbon dioxide are released and sparge through the melt. Sparging releases the volatiles and refractory radionuclides as well as inert aerosols such as SiO_2 and Na_2O. The water vapor and carbon dioxide can also oxidize the unreacted metal (Zirconium and iron), producing more heat in the molten pool and thus enhancing the rate of aerosol production.

The STCP results for fraction of initial core inventory that is released from the melt during core-concrete interactions (ST_{EXV}) are tabulated in tables 5.19 through 5.21. ST_{EXV} represents the release from the top surface of the core debris. Retention in an overlaying water pool, if it exists, will be considered separately.

CORCON-MOD2 is used for modelling corium-concrete interaction. The CORCON-MOD2 code assumes an immediate separation of corium into two immiscible metallic and oxidic layers. As corium concrete interaction proceeds, the decomposed concrete forms a second oxidic layer. The order of layers depends on their relative densities. CORCON predicts the release of steam and CO_2 from concrete ablation. The gases that bubble up through the core debris are modeled to react chemically with the materials of the melt. The oxidation reaction between the metallic constituent and the concrete decomposition gases is assumed to proceed to equilibrium. The CORCON model accounts for the heat of reaction from the chemical processes and for the decay heat from fission products.

The STCP (VANESA code) calculation of the ex-vessel release of radionuclides is driven by bubbling of reaction gases into the melt. VANESA calculates the releases by vaporization of fission products and other melt constituents from the melt into the gas bubbles. As the gases leave the melt, the code empirically models aerosol formation from bubbles breaking the melt surface and from the condensation and nucleation of vapors.

Among the factors that influence the magnitude of the ex-vessel releases are the composition and temperature of the corium as it is released from the vessel. The composition of concrete can also have a major impact on the amount of aerosols carried into the containment atmosphere. Limestone concrete produces larger gas flows and is more oxidizing compared to basaltic. Of the six plants for which the STCP calculations have been performed, only Surry and Oconee have siliceous or basaltic concrete. Sequoyah, Zion, and Grand Gulf have limestone concrete which produces large gas flows compared to basaltic concrete. Peach Bottom has limestone-common sand concrete which produces less gas flow than limestone, but more than basaltic concrete. A variation of the STCP calculation of the Peach Bottom S_2E_1 sequence (S_2E_2) with basaltic concrete is also shown in Table 5.21.

Table 5.19 STCP Results for Fraction of Initial Core Inventory Released During Core-concrete Interactions (ST_{EXV}) PWR, Limestone Concrete

	Zion				Sequoyah					
	TMLU	S2DCR	S2DCF1	S2DCF2	S3HF	S3B	S3B1	TMLB'	TBA	ACD
NG	0	0	0	0	0	0	0	0	0	0
I	10^{-3}	$4x10^{-3}$	$4x10^{-3}$	$3.5x10^{-3}$	0.03	0.03	0.01	0.03	$0.9x10^{-3}$	$0.9x10^{-3}$
Cs	10^{-3}	$4x10^{-3}$	$4x10^{-3}$	$4x10^{-3}$	0.03	0.03	0.02	0.03	$0.8x10^{-3}$	$0.9x10^{-3}$
Te	0.17	0.22	0.40	0.28	0.06	0.08	0.07	0.40	0.09	0.22
Sr	$1.4x10^{-4}$	0.32	0.10	0.34	0.17	0.17	0.17	0.53	0.17	0.51
Ba	$1.3x10^{-3}$	0.23	0.07	0.23	0.10	0.10	0.10	0.29	0.10	0.27
Ru	$4x10^{-6}$	10^{-5}	$8x10^{-4}$	$3x10^{-6}$	$2x10^{-6}$	$4x10^{-6}$	$5x10^{-6}$	$3.5x10^{-3}$	$4x10^{-6}$	$0.8x10^{-6}$
Ce	$5x10^{-7}$	$7x10^{-3}$	$2x10^{-3}$	$8x10^{-3}$	$6x10^{-3}$	$7x10^{-3}$	$7x10^{-3}$	$3x10^{-2}$	$6x10^{-3}$	$0.8x10^{-3}$
La	$9x10^{-6}$	$7x10^{-3}$	$5x10^{-3}$	$8x10^{-3}$	$9x10^{-3}$	$9x10^{-3}$	$9x10^{-3}$	$2x10^{-2}$	$1.2x10^{-2}$	$6x10^{-2}$

Table 5.20 STCP Results for Fraction of Initial Core Inventory Released During Core-concrete Interactions (ST_{EXV}) PWR, Basaltic Concrete

	Surry				OCONEE	
	TMLB'	S3B	AG	V	S1DCF	TMLB'
NG	0	0	0	0	0	0
I	0.02	0.04	1.5×10^{-4}	0	0.001	0.005
Cs	0.02	0.04	1.6×10^{-4}	0	0.001	0.005
Te	0.12	0.11	0.02	0.06	0.31	0.29
Sr	0.17	0.09	0.09	0.33	0.60	0.18
Ba	0.10	0.06	0.06	0.16	0.30	0.11
Ru	4×10^{-6}	6×10^{-7}	4×10^{-9}	2×10^{-6}	5×10^{-6}	5×10^{-6}
Ce	7×10^{-3}	10^{-3}	10^{-3}	0.025	0.16	0.004
La	8×10^{-3}	4×10^{-3}	4×10^{-3}	0.02	0.12	0.002

Table 5.21 STCP Results for Fraction of Initial Core Inventory Released During Core-concrete Interactions (ST_{EXV}) BWR, Limestone Concrete

	Peach Bottom							LaSalle	Grand Gulf		
	TC1	TC2	TB	TBUX	S2E1	V	S2E2[a]	TB	TC1	TBS/TBR	TB
NG	0	0	0	0	0	0	0	0	0	0	0
I	0.08	0.04	0.11	0.07	0.16	0.17	0.16	0.008	0.002	0.03	0.02
Cs	0.08	0.04	0.12	0.06	0.16	0.18	0.16	0.008	0.002	0.03	0.03
Te	0.40	0.19	0.39	0.40	0.50	0.6	0.11	0.3	0.11	0.14	0.21
Sr	0.75	0.66	0.84	0.75	0.77	0.84	0.28	0.68	0.42	0.42	0.55
Ba	0.56	0.45	0.60	0.53	0.54	0.6	0.17	0.48	0.25	0.26	0.35
Ru	1×10^{-6}	3×10^{-6}	1×10^{-6}	1×10^{-6}	2×10^{-6}	3×10^{-6}	3×10^{-8}	2×10^{-7}	4×10^{-7}	3×10^{-7}	8×10^{-7}
Ce	0.04	0.03	0.09	0.06	0.07	0.08	8.4×10^{-3}	0.04	0.05	0.04	0.06
La	0.02	0.02	0.06	0.03	0.03	0.04	8×10^{-3}	0.02	0.03	0.02	0.04

[a] Basaltic concrete was assumed in calculations.

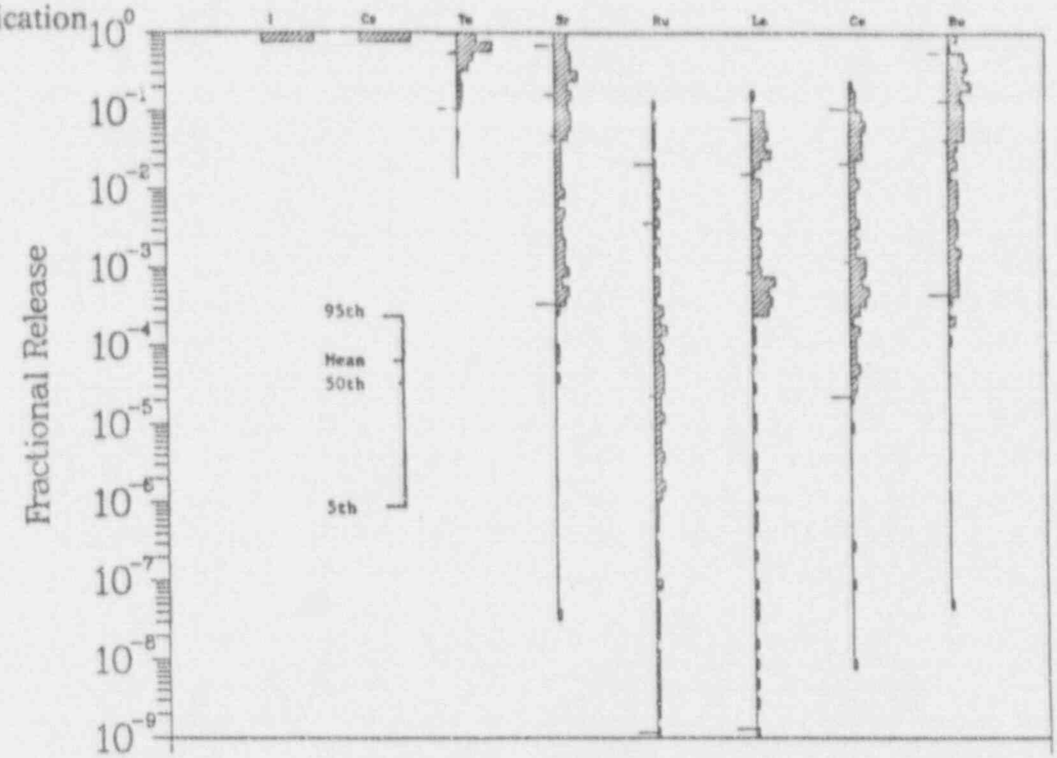
The uncertainties in ex-vessel release of fission products are high. There is much uncertainty in the concrete erosion processes and in the interaction between the molten pool and any water that may be present in the reactor cavity or pedestal region during core concrete interaction. These uncertainties are compounded by uncertainties associated with the thermochemistry and kinetics of the vaporization of melt species into gases.

The issue of the ex-vessel release of fission products during core-concrete interaction was assessed by the members of the NUREG-1150 source term expert panel. The experts were asked for the distributions that they believed would characterize the uncertainty in the fraction of radionuclide group i that is released from the molten fuel during corium-concrete interaction (FCCI). For each type of reactor (PWRs vs. BWRs), different cases were considered during the elicitation based on the type of concrete, the amount of Zr in the molten core, and the presence or absence of water in the reactor cavity or pedestal region during core concrete interaction.

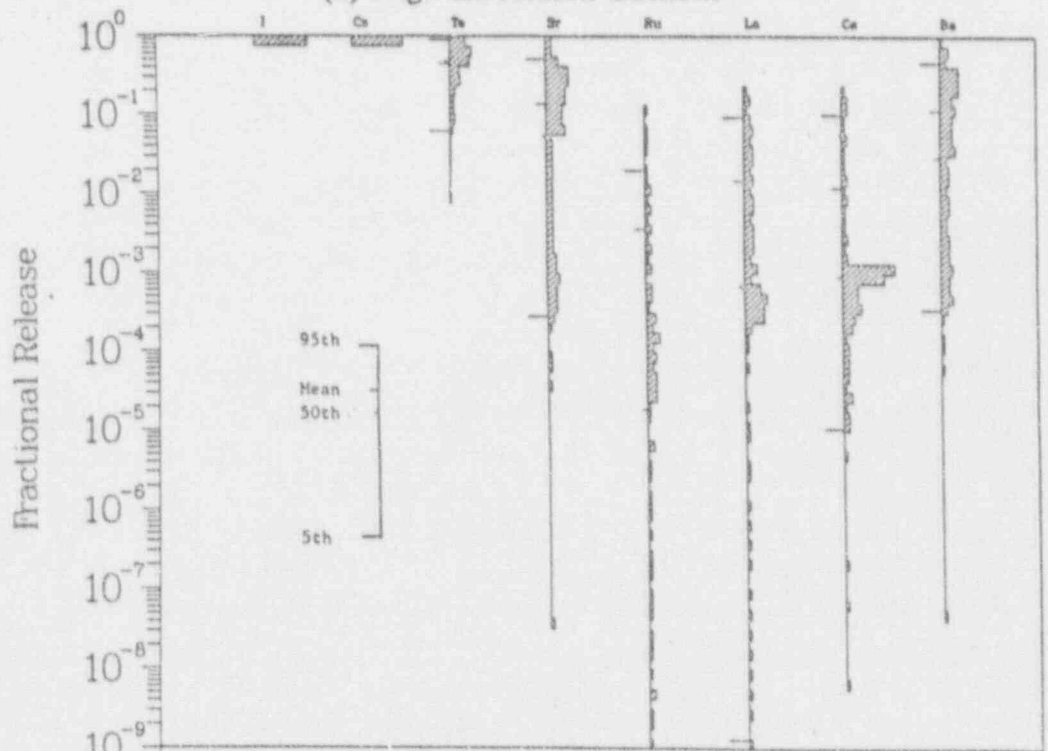
The detailed results of expert panel elicitation is in Reference [23]. The distributions of release magnitude (FCCI) for various isotopic groups are shown in Figures 5.8 through 5.13. These distributions were obtained from aggregate cumulative probability distributions presented in Reference [23], using the LHS sampling method. Logarithmic interpolation was used to determine CCI releases between fractiles. The mean and median values for these distributions are shown in Table 5.22 and 5.23. The estimated releases during CCI depends strongly on the volatility of fission products, as might be expected. The noble gas and volatile species (I and Cs groups) would certainly be completely released from the molten core-concrete interactions. The difference between releases of semi-volatile fission products Sr and Ba is not great. Low volatile fission products Ce and La groups have also similar releases. Note that whether or not there is water in the reactor cavity or pedestal region is the only important variable in characterizing the uncertainty distribution of CCI. Due to large amounts of Zircaloy in the BWR core, the CCI releases tend to be higher for BWRs.

5.6 Radionuclide Releases Into the Containment Associated With Late Revolatilization From the RCS

A phenomena which has a potential impact on the severe accident source term is the revaporization of radionuclides retained in the RCS and their subsequent release into the containment after vessel failure. The importance of this issue is the timing of radionuclide revaporization. The additional activity associated with the revaporized fission products is not expected to dramatically increase the overall releases except for some BWR accident sequences since the revolatilization release may not pass through the suppression pool. The impact of radionuclide revaporization is more noticeable and significant if the timing of revaporization is delayed. If fission products revaporize early, (shortly after vessel failure) the revaporized material will enter the containment and could interact with a concentrated source of aerosols generated by core-concrete interaction (and enter the suppression pool for BWR cases). The result is some mitigation of the revaporization process. If fission products revaporize slowly and the

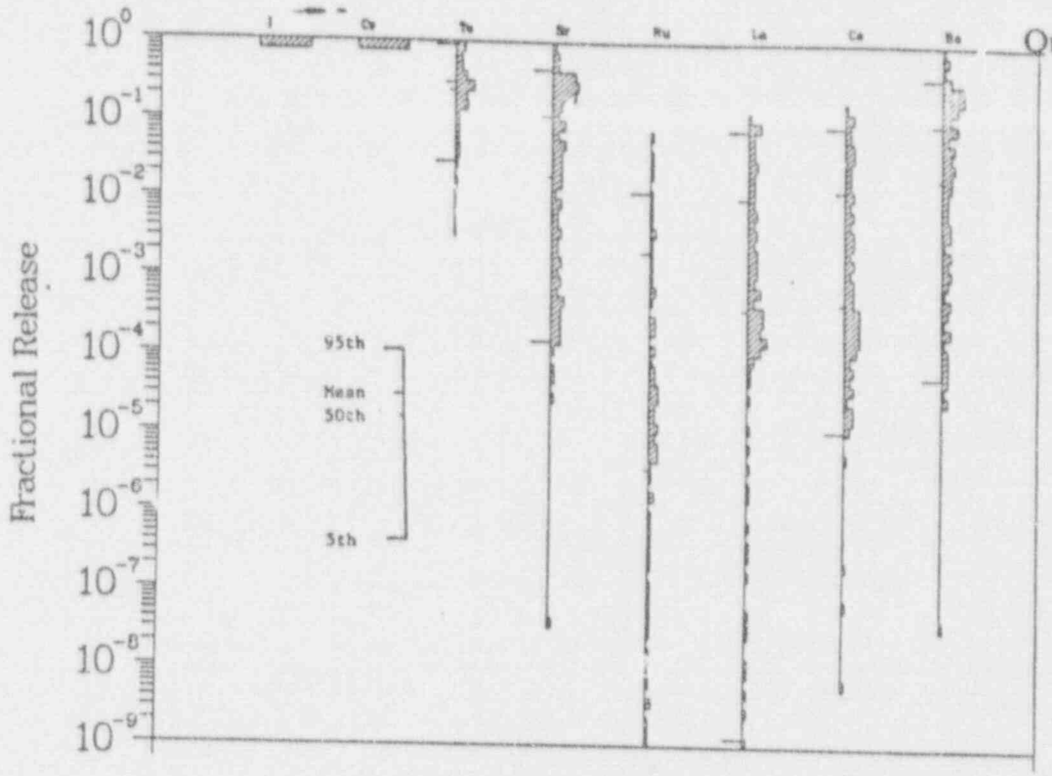


(a) High Zirconium Content

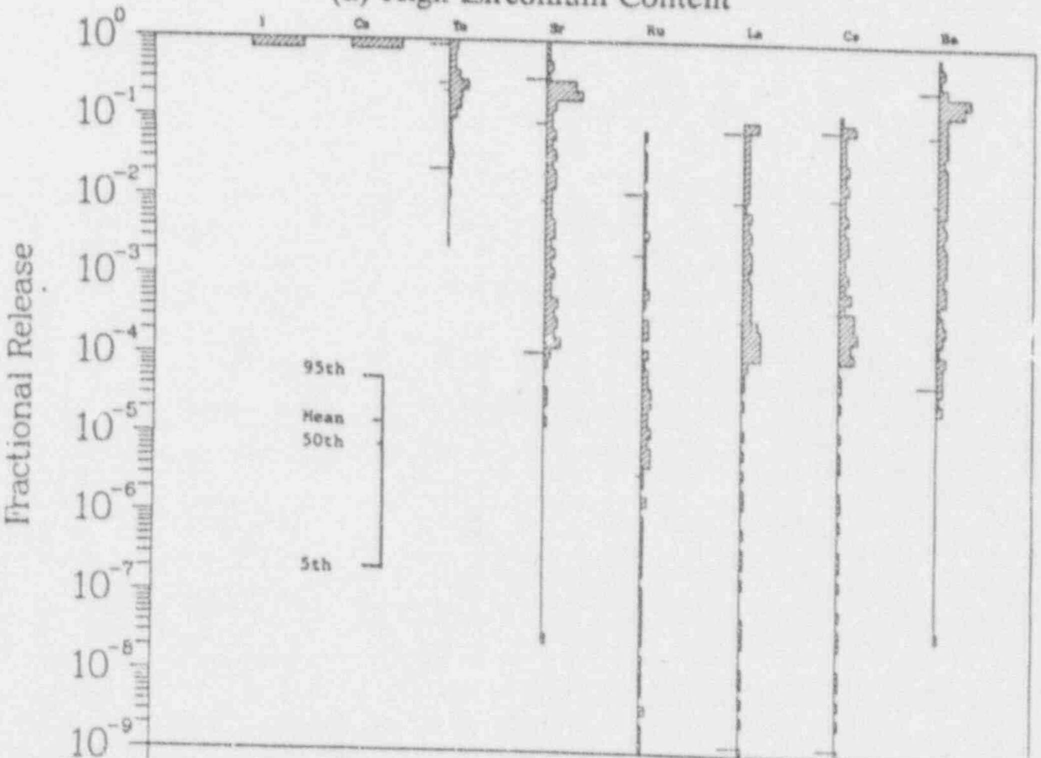


(b) Low Zirconium Content

Figure 5.8 Uncertainty Distributions for Fission Product Release During Core-concrete Interaction (FCCI), PWR, Limestone Concrete, Dry Cavity.



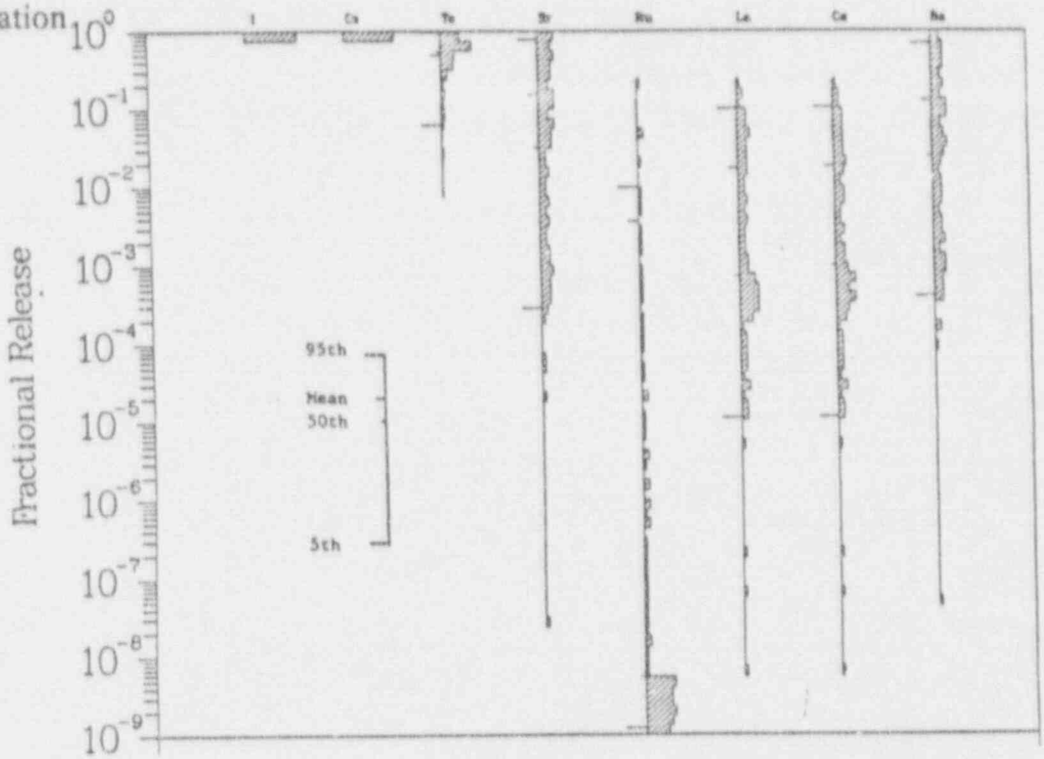
(a) High Zirconium Content



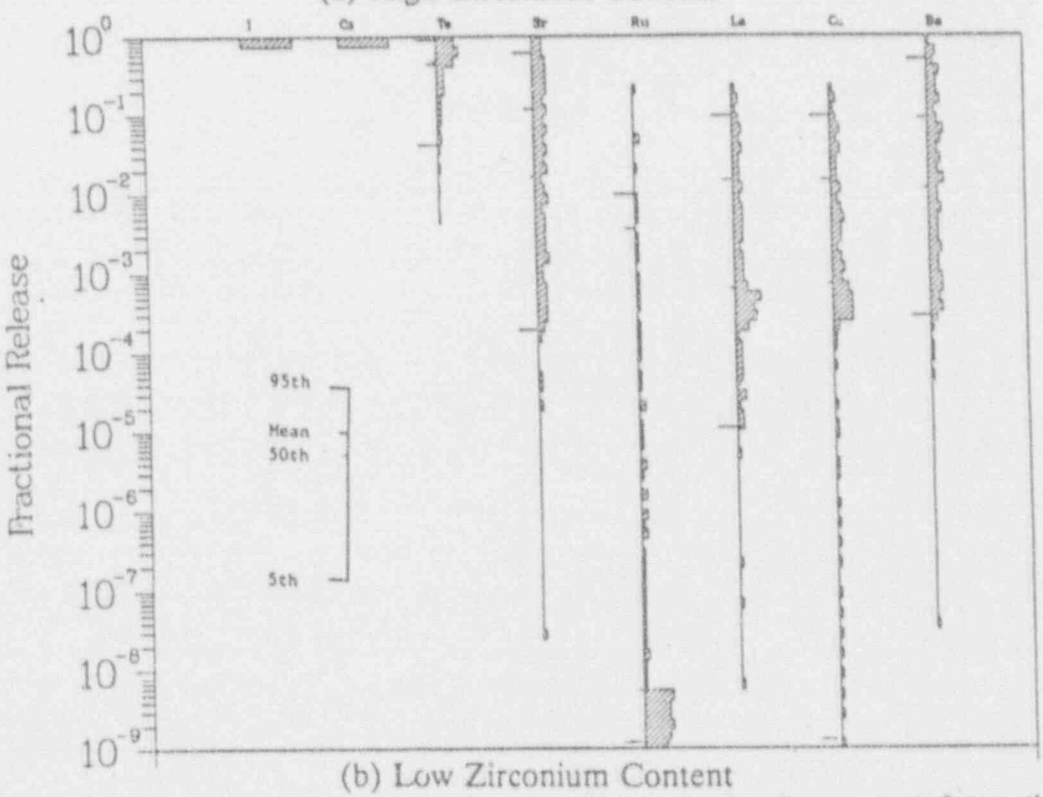
(b) Low Zirconium Content

Figure 5.9 Uncertainty Distributions for Fission Product Release During Core-concrete Interaction (FCCI), PWR, Limestone Concrete, Wet Cavity.

Quantification

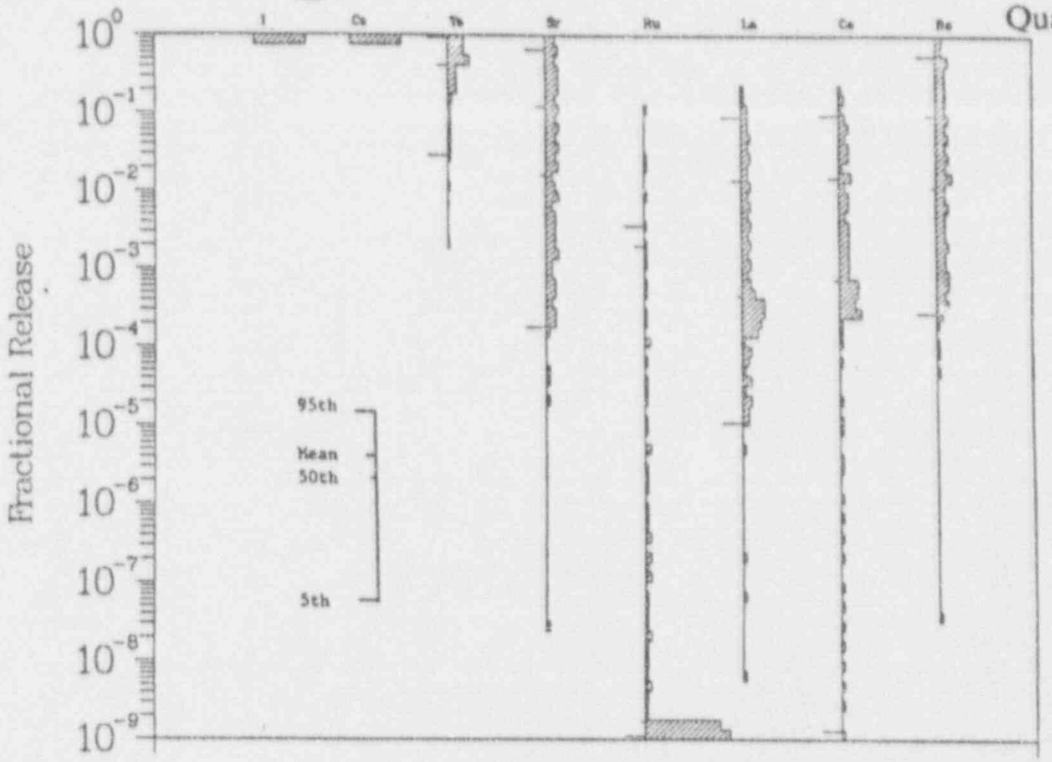


(a) High Zirconium Content

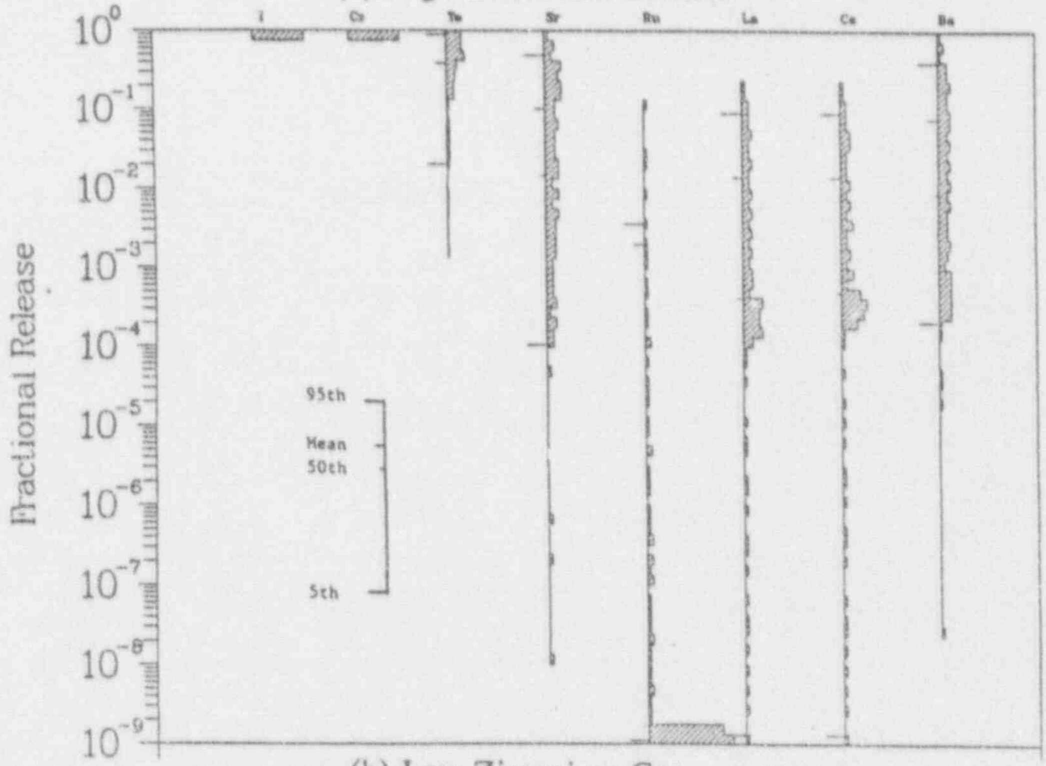


(b) Low Zirconium Content

Figure 5.10 Uncertainty Distributions for Fission Product Release During Core-concrete Interaction (FCCI), PWR, Basaltic Concrete, Dry Cavity.

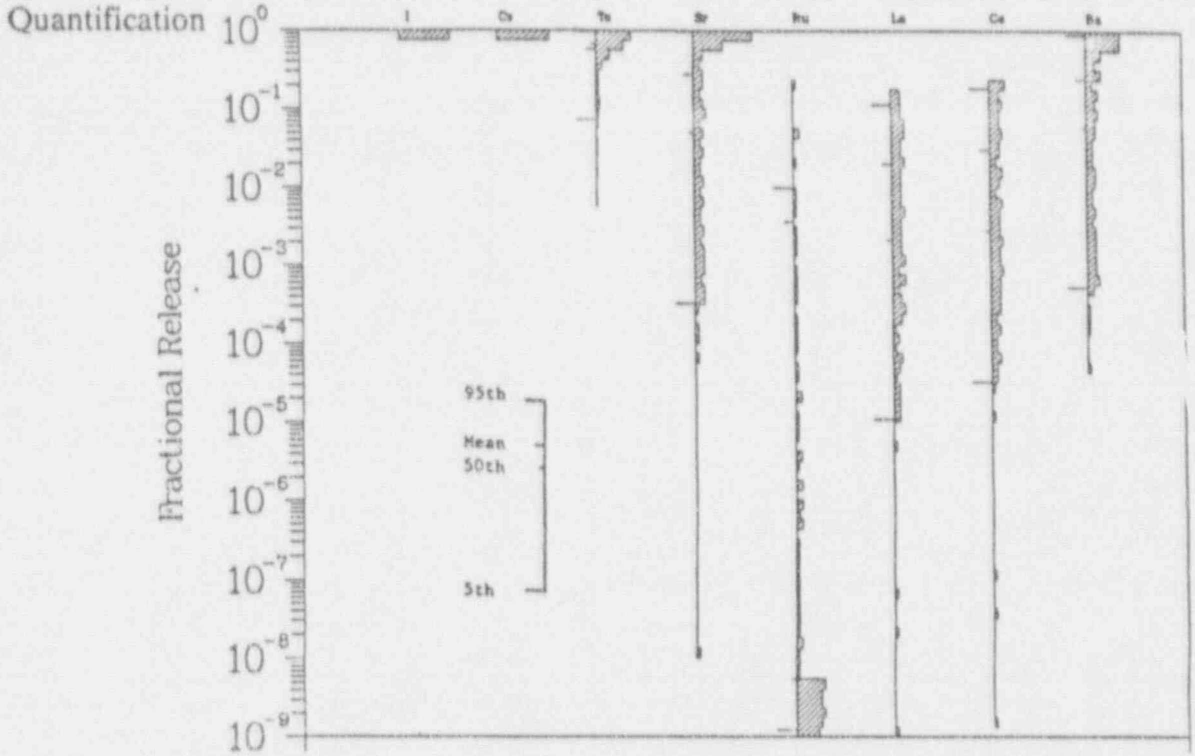


(a) High Zirconium Content

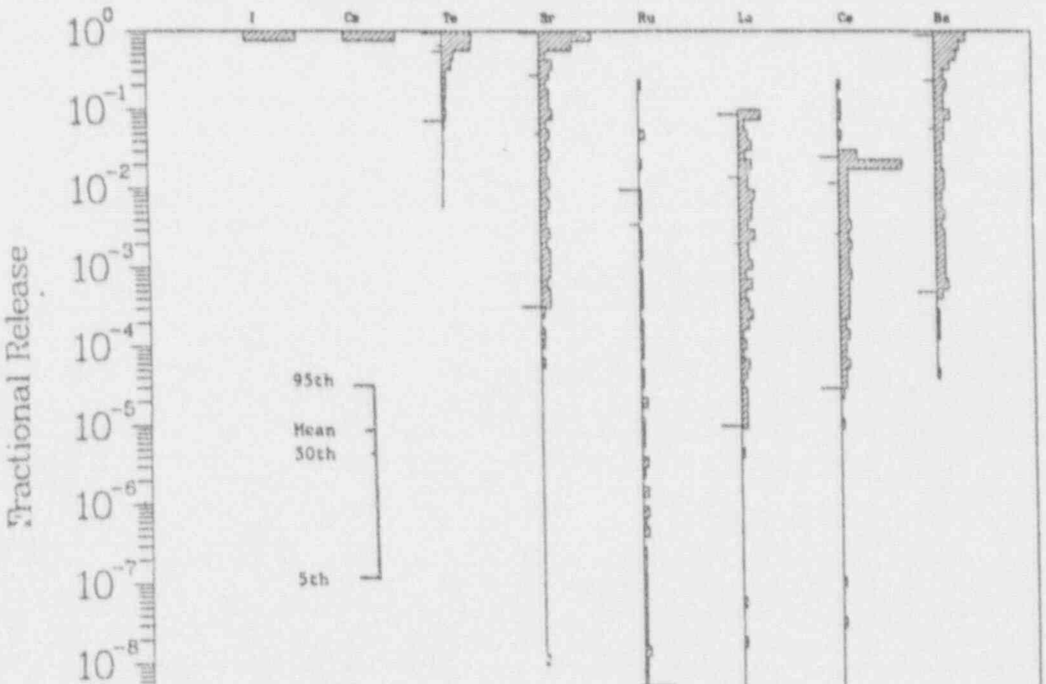


(b) Low Zirconium Content

Figure 5.11 **Uncertainty Distributions for Fission Product Release During Core-concrete Interaction (FCCI), PWR, Basaltic Concrete, Wet Cavity.**

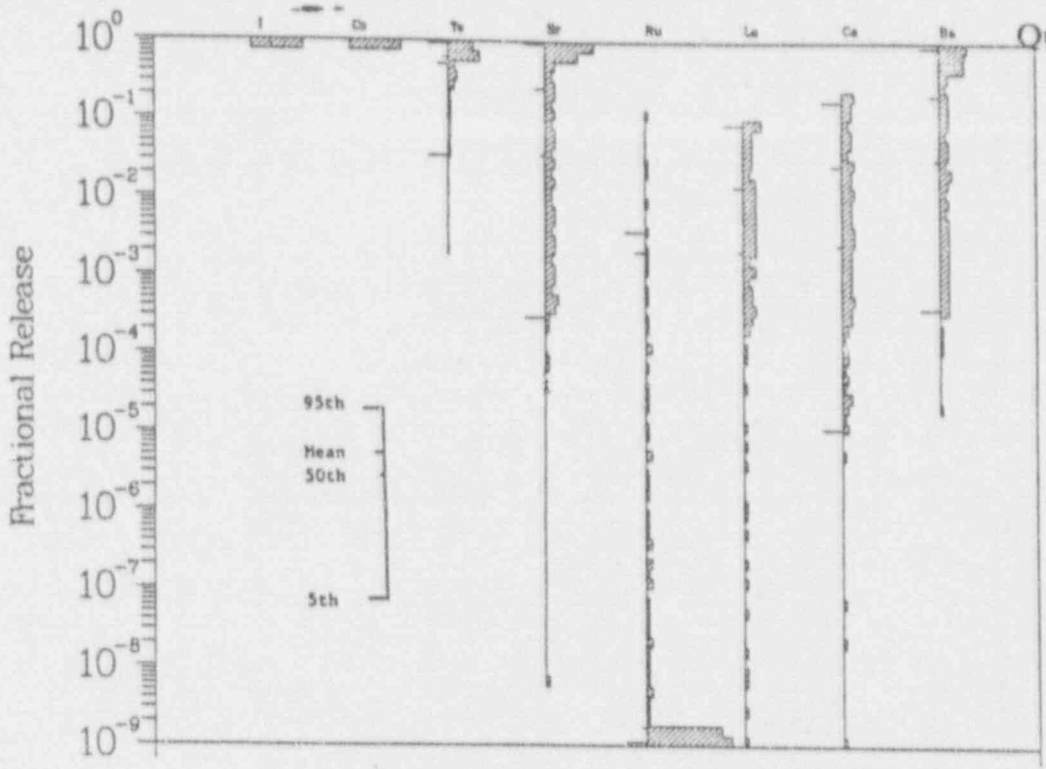


(a) High Zirconium Content

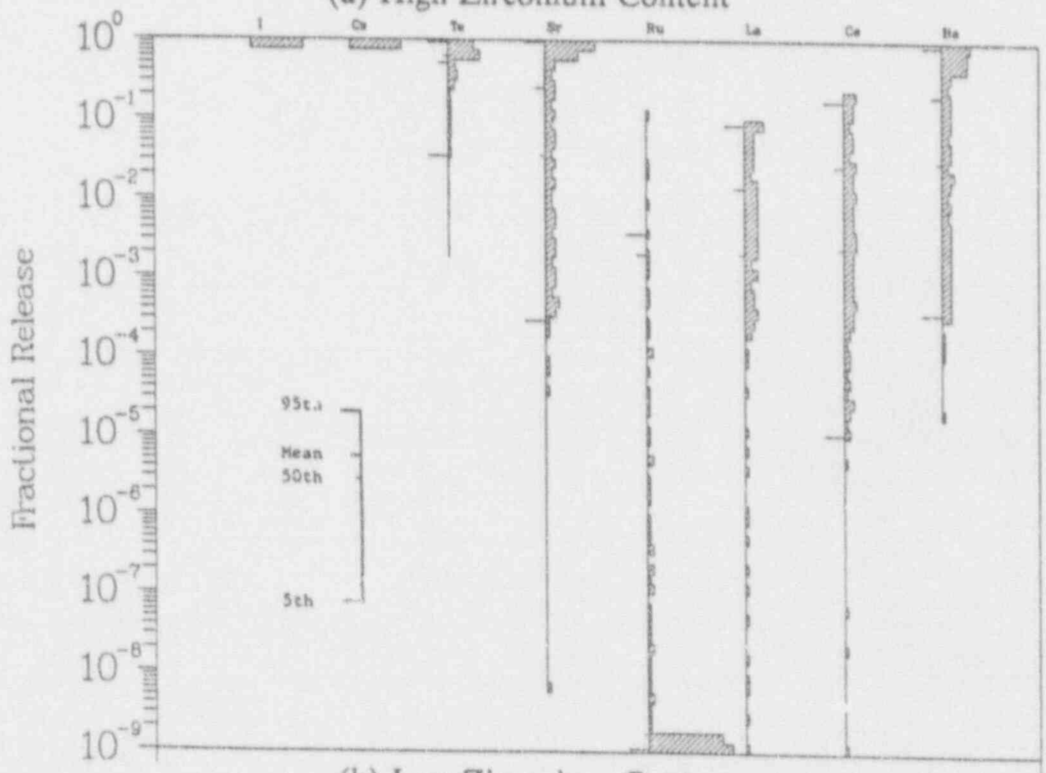


(b) Low Zirconium Content

Figure 5.12 **Uncertainty Distributions for Fission Product Release During Core-concrete Interactions (FCCI), BWR, Limestone Concrete, Dry Pedestal.**



(a) High Zirconium Content



(b) Low Zirconium Content

Figure 5.13 Uncertainty Distributions for Fission Product Release During Core-concrete Interactions (FCCI), BWR, Limestone Concrete, Wet Pedestal.

Table 5.22 Mean and Median Values for the Fractions of Radionuclide Group I Released During Core-concrete Interaction (FCCI) for PWRs.

Conditions			FCCI[d]							
Concrete Type[a]	Cavity Condition[b]	Zirconium Content in the Melt[c]	I,Cs	Te	Sr	Ba	Ru	La	Ce
L	D	H	1.0	0.56 (0.52)	0.05 (0.15)	0.04 (0.13)	$2X10^{-5}$ (0.004)	$8X10^{-4}$ (0.015)	10^{-3} (0.02)
L	D	L	1.0	0.5 (0.45)	0.05 (0.13)	0.03 (0.11)	$2X10^{-5}$ (0.004)	$7X10^{-4}$ (0.015)	$9X10^{-4}$ (0.01)
L	W	H	1.0	0.24 (0.30)	0.0[illegible] (0.1[illegible])	0.02 (0.10)	$3x10^{-6}$ (0.002)	$4x10^{-4}$ (0.01)	$4.5x10^{-4}$ (0.01)
L	W	L	1.0	0.23 (0.28)	0.009 (0.09)	0.01 (0.07)	$3x10^{-6}$ (0.002)	$3x10^{-4}$ (0.01)	$4x10^{-4}$ (0.01)
B	D	H	1.0	0.54 (0.49)	0.03 (0.15)	0.02 (0.13)	$5x10^{-9}$ (0.004)	$7x10^{-4}$ (0.02)	$9x10^{-4}$ (0.02)
B	D	L	1.0	0.48 (0.44)	0.02 (0.12)	0.01 (0.09)	$5x10^{-9}$ (0.004)	$6x10^{-4}$ (0.01)	$7x10^{-4}$ (0.01)
B	W	H	1.0	0.41 (0.41)	0.02 (0.12)	0.01 (0.10)	$2x10^{-9}$ (0.002)	$4x10^{-4}$ (0.01)	$7X10^{-4}$ (0.01)
B	W	L	1.0	0.38 (0.39)	0.01 (0.10)	0.009 (0.07)	$2x10^{-9}$ (0.002)	$4x10^{-4}$ (0.01)	$5X10^{-4}$ (0.01)

(a) L & B refer to limestone and basaltic concrete respectively.
(b) D & W refer to dry and wet cavity respectively.
(c) H & L refer to high and low Zirconium content in the melt.
(d) Mean values are presented in parenthesis.

Table 5.23 Mean and Median Values for the Fraction of Radionuclide Group i Released During Core-concrete Interaction (FCCI) for BWRs.

Conditions			FCCI						
Concrete Type[a]	Pedestal Condition[b]	Zirconium Content in the Melt[c]	I, Cs	Te	Sr	Ba	Ru	La	Ce
L	D	H	1.0	0.66 (0.57)	0.05 (0.27)	0.06 (0.24)	5×10^{-9} (0.004)	0.002 (0.02)	0.003 (0.03)
L	D	L	1.0	0.65 (0.55)	0.05 (0.26)	0.06 (0.23)	5×10^{-9} (0.004)	0.002 (0.01)	0.003 (0.01)
L	W	H	1.0	0.63 (0.51)	0.03 (0.25)	0.03 (0.20)	2×10^{-9} (0.002)	0.002 (0.01)	0.002 (0.01)
L	W	L	1.0	0.63 (0.51)	0.03 (0.25)	0.03 (0.20)	2×10^{-9} (0.002)	0.002 (0.01)	0.002 (0.02)

[a] L refers to limestone concrete.
[b] D and W refer to dry and wet pedestal conditions.
[c] H and L refer to high and low Zirconium content in the melt.

process continues during the latter stages of the accident when the containment aerosol concentration is low, retention of the revaporized fission products may be lower.

Fission product revaporization is affected by post-vessel-failure thermal hydraulics, reactor coolant system heat transfer, and the chemistry of retained radionuclides.

Extensive RCS retention during the in-vessel phase of the accident (high pressure sequences), high temperature of RCS structures, and high flow inside the RCS after vessel failure (to carry vaporized fission products to the containment) are prerequisites to fission product revaporization.

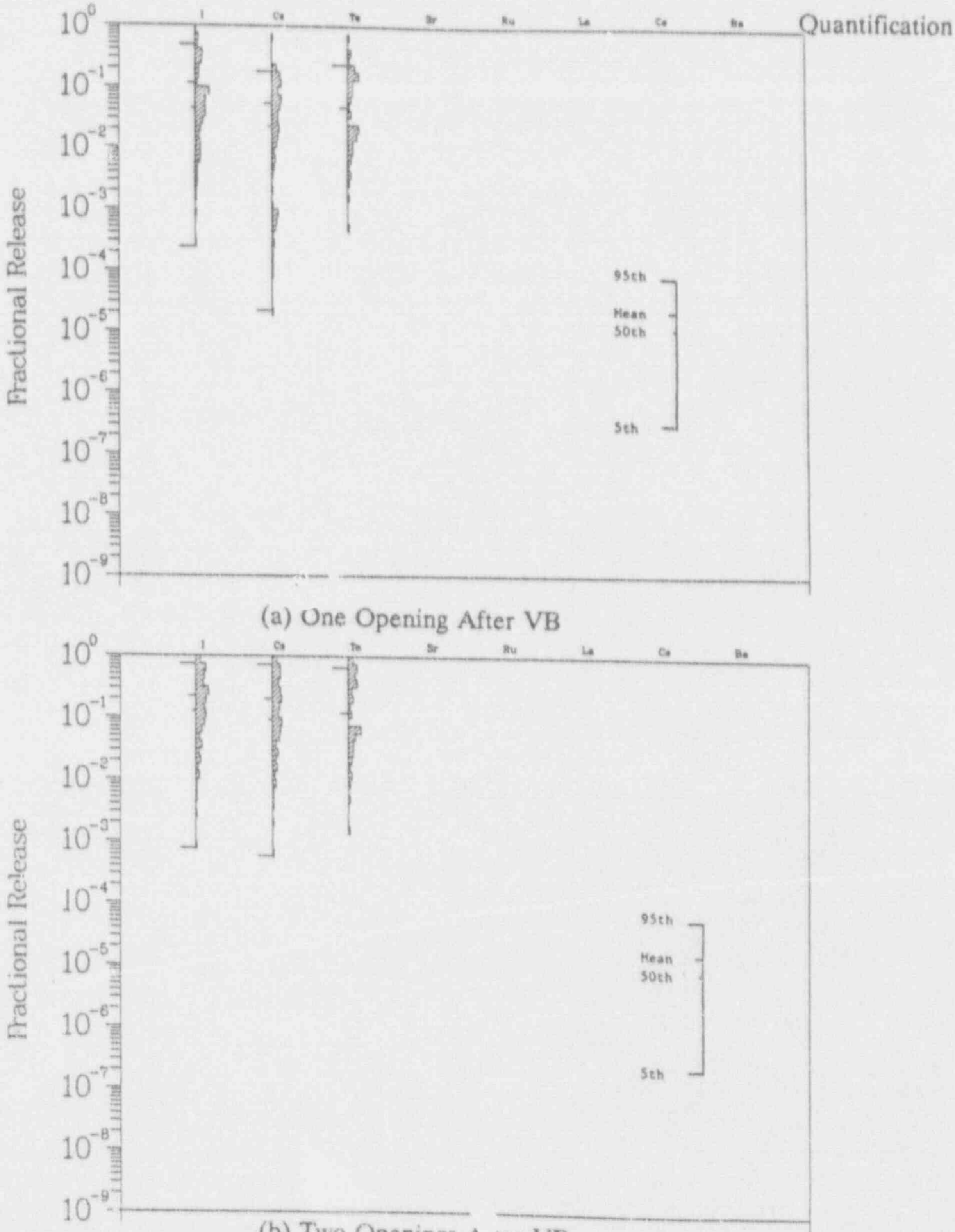
The STCP does not model revolatilization phenomena after the core debris has penetrated the vessel. However, late revolatilization is included in the present formulation through the parameter of FREV. The extent of radionuclide releases into the containment associated with late revolatilization from the RCS (see chapter 4) is:

$$ST_{REV}(i) = FCOR(i) * (1 - FVES(i)) * FREV(i) \tag{5.2}$$

Fission product releases into the containment by late revolatilization from the RCS were considered by the NUREG-1150 source term expert panel. The fraction of the fission product of group i retained in the RCS (at the time of vessel breach) which is released to containment at later times, FREV(i), depends upon the type of reactor and on the number of large openings in the RCS after vessel breach (for PWRs) and the drywell temperature (for BWRs). Two cases were considered for PWRs and three cases were proposed for the BWRs:

- PWR-1 -- One opening after vessel breach,
- PWR-2 -- Two openings after vessel breach,
- BWR-1 -- No water injection after vessel breach (TBUX or TBU) and high drywell temperature,
- BWR-2 -- No water injection after vessel breach (TBUX or TBU) and low drywell temperature, and
- BWR-3 -- Water injection available after vessel breach (e.g. TCUX)

The detailed results of expert panel elicitation is in Reference [23]. Only three groups - iodine, cesium, and tellurium were considered for the late revolatilization release. The uncertainty distributions of FREV values are shown in Figures 5.14 and 5.15. These distributions were obtained from aggregate cumulative probability distributions presented in Reference [23], using the LHS sampling method. Linear interpolation was used to determine FREV values between fractiles. The mean and median values of these distributions are presented in Table 5.24. Note that for PWRs, the revolatilization is higher if there are two RCS breaches (e.g. in the vessel and in the hot leg) due to establishment of natural circulation between the RCS and the containment. In BWRs, low drywell temperature would keep the reactor vessel walls cool and decrease the vaporization. Also, for the high pressure anticipated transients without scram (ATWS) sequence (TCUX), the low pressure injection system would become available for

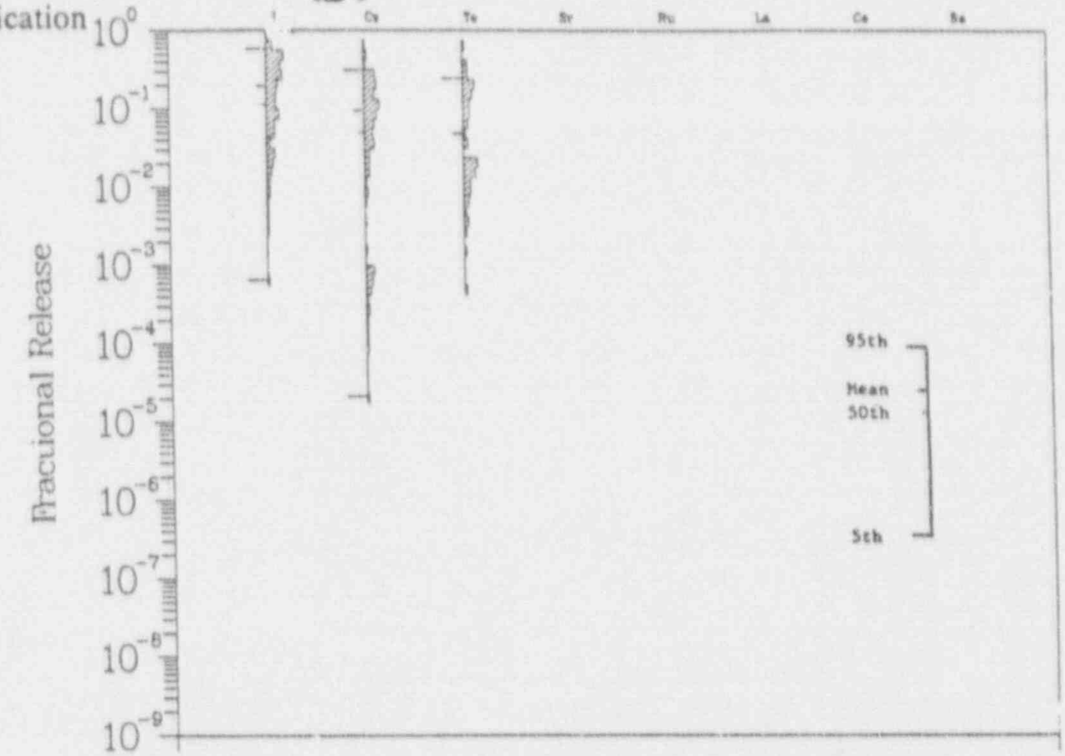


(a) One Opening After VB

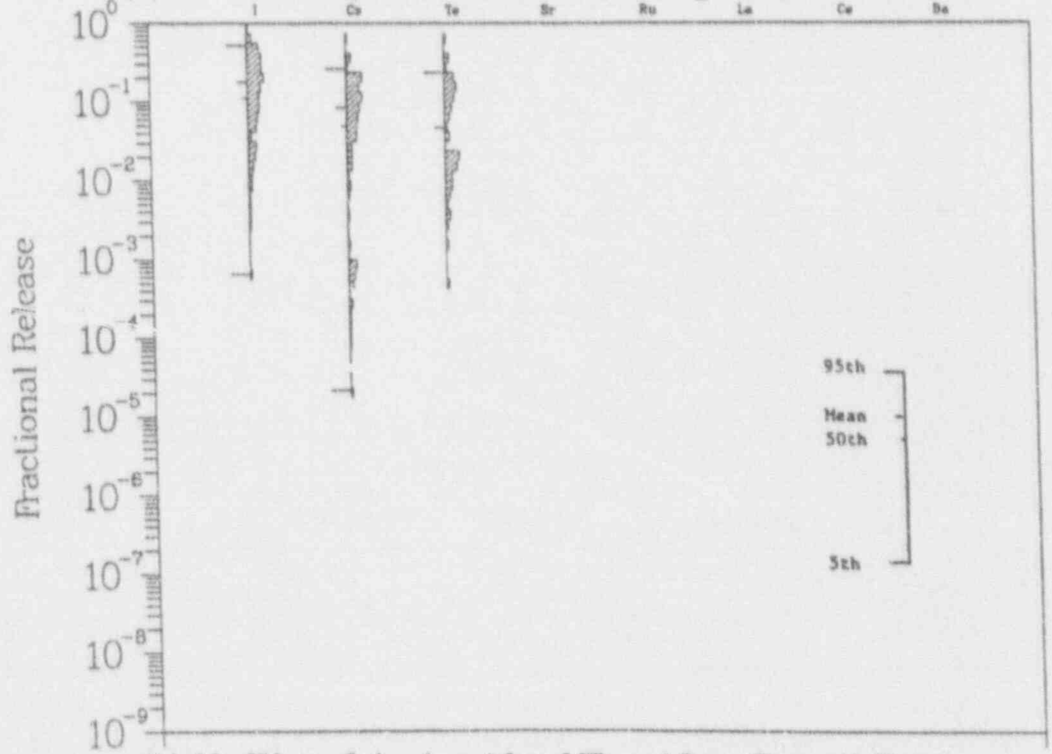
(b) Two Openings After VB

Figure 5.14 **Uncertainty Distributions for the Fraction of Radionuclide Group i Retained in RCS Which is Released Into the Containment at Later Times ($FREV_i$) for PWRs.**

Quantification

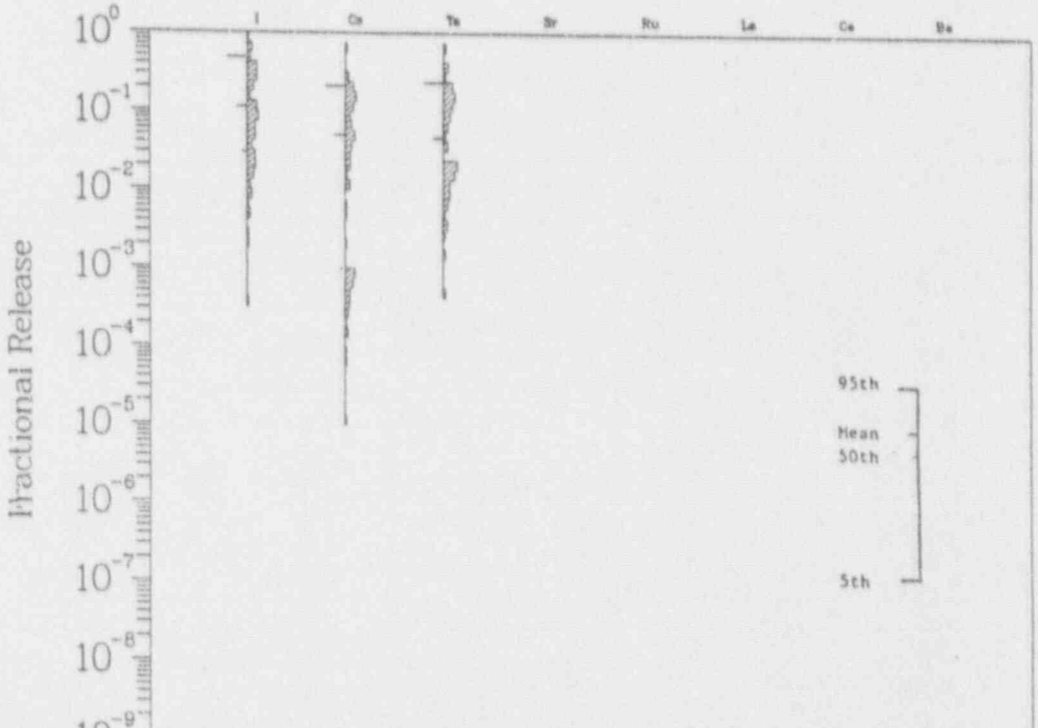


(a) No Water Injection After VB and High Drywell Temp.



(b) No Water Injection After VB and Low Drywell Temp.

Figure 5.15 **Uncertainty Distributions for the Fraction of Radionuclide Group i Retained in RCS Which is Released Into Containment at Later Times ($FREV_i$) for BWRs.**



(c) Water Injection After VB

Figure 5.15 **Uncertainty Distributions for the Fraction of Radionuclide Group i Retained in RCS Which is Released Into Containment at Later Times ($FREV_i$) for BWRs. (Continued)**

Table 5.24 Mean & Median Values for the Fraction of Radionuclide Group I Retained in RCS Released Into Containment After Vessel Failure (FREV)

	Conditions	FREV[a] I	FREV[a] Cs	FREV[a] Te
PWRs	One opening after vessel breach	0.04 (0.11)	0.02 (0.05)	0. (0.04)
	Two openings after vessel breach	0.13 (0.22)	0.095 (0.20)	0. (0.12)
BWRs	No water injection after vessel breach and high drywell temperature	0.11 (0.19)	0.05 (0.09)	0. (0.05)
	No water injection after vessel breach and low drywell temperature	0.11 (0.17)	0.05 (0.08)	0. (0.05)
	Water injection after vessel breach	0.03 (0.11)	0.001 (0.05)	0. (0.05)

[a] The mean values are shown in parenthesis.

injecting water into the reactor vessel. This would cool the reactor vessel and would result in a lower revolatilization release.

5.7 Effective Decontamination Factor (DF) of the Water Pool Overlying the Corium During Core-concrete Interaction

In some accident sequences, water will be in the reactor cavity or pedestal region during core-concrete interaction. If there is sufficient water maintained the water pool overlying the core debris can scrub aerosol particles released during CCI. The depth of the overlying water pool (if any) during core-concrete interaction is an important factor which influences the aerosol scrubbing effect and, therefore, the magnitude of ex-vessel release to the containment. The STCP (VANESA code) models this aerosol scrubbing by gravitational settling, random diffusion, and inertial impaction. Table 5.25 shows the STCP results for effective decontamination factor of the water pool overlying the corium melt during core-concrete interaction (DF_{pool}).

5.8 Timing of Releases

The release time and duration of fission products is an important parameter in specifying the source term into the containment. Current regulatory requirements assume the radionuclide release occurs immediately upon loss of core cooling, and occurs at such a rapid rate as to be virtually instantaneous. However, the actual time of appearance of radionuclides in the containment can vary depending on plant, accident sequence and radionuclide group. Tables 5.26 and 5.27 summarize the results of the STCP calculations of the time elapsed between the accident initiation and the actual appearance of radioactive material in the containment. The accident which has the shortest time (28 minutes) until a significant in-vessel radionuclide release into the containment is a large break LOCA.

The results of the STCP for in-vessel release duration are also summarized in Tables 5.26 and 5.27. These durations are the time intervals extending from release initiation to reactor pressure vessel bottom head failure.

The in-vessel release duration is generally longer for BWR accident sequences. This is due to lower power to moderator ratio and lower core power density in BWRs which would delay the time for complete core meltdown.

The STCP model for bottom head failure includes stresses due to reactor vessel internal pressure. Thus the high RCS pressure sequences show a trend toward shorter in-vessel release durations.

Although the release from core-concrete interaction is predicted to extend many hours beyond initiation of corium-concrete interactions (time of vessel breach), generally 90% of radionuclide releases (except Te and Ru) are predicted to occur within a two hour period for PWRs and three hours for BWRs. For tellurium and ruthenium, ex-vessel releases are predicted to occur within five hours for PWR and six hours for BWRs.

Table 5.25 STCP Results for Effective Decontamination Factor of the Water Pool Overlying the Corium During Core-concrete Interaction (DF_{POOL})

Species	Surry	Zion				Sequoyah			
	TMLB'	TMLU	S2DCR	S2DCF1	S2DCF2	S3HF1/S3HF2	S3HF3	S3B	TMLB'
I	2	16	13	2	20	32	6	6	15
Cs	2	9	14	2	17	23	4	4	2
Te	2	10	13	1	5	12	2	2	2
Sr	2	11	14	2	17	28	5	1	6
Ba	3	9	14	2	17	27	5	5	6
Ru	4	9	18	1	1	6	1	1	–
Ce	4	15	14	2	16	29	5	5	3
La	3	14	13	2	11	31	4	6	3

Table 5.26 STCP Results for Timing of In-vessel Releases Into Containment for PWR Accident Sequences

Plant	Accident Sequence	Time of In-Vessel Release Into Containment (min)	In-vessel Release Duration (min)
Surry	TMLB′	135	41
	S3B	110	36
	AG	3140	215
Zion	TMLU	150	41
	S2DCR/S2DCF	94	39
Sequoyah	S3HF	364	46
	S3B	327	46
	S3B1	434	75
	TMLB′	116	37
	TBA	52	195
	ACD	28	73
Oconee	TMLB′	83	35
	S1DCF	80	84

Table 5.27 STCP Results for Timing of In-vessel Releases Into Containment for BWR Accident Sequences

Plant	Accident Sequence	Time of In-Vessel Release Into Containment (min)	In-Vessel Release Duration (min)
Peach Bottom	TC2	62	66
	TC3	58	63
	TC1	134	97
	TB1/TB2	642	91
	S2E	110	81
	TBUX	134	67
LaSalle	TB	--	81
Grand Gulf	TB	579	122
	TC1	117	130
	TBS/TBR	85	96

Quantification

Due to larger amounts of Zircaloy in the BWRs, the duration of ex-vessel releases are generally longer for some accident sequences. The exothermic oxidation of Zirconium present in the melt during core-concrete interaction enhances the heat source to the melt and increases the ex-vessel release duration.

A review of STCP calculated results for the appearance rate into containment [25] indicated that the fission product releases can be treated as being proportional to time after the initial release.

6. Updated Source Term Formulation Parameters

Bounding values for radionuclide releases into the containment under severe accident conditions are tabulated in Table 6.1 and 6.2. The release fraction for each radionuclide group which is assigned to an accident category generally was taken as the highest STCP calculated fraction from all of those accident sequences assigned into the release category. Radionuclide releases due to direct containment heating (ST_{VB}) and late revolatilization from the RCS (ST_{REV}) are based on the assessment of NUREG-1150 expert elicitation discussed in the previous chapter.

The duration of these releases to containment have also been selected through an assessment of the existing STCP calculations. The duration of in-vessel releases are generally within 40 minutes for PWR and 1.5 hours for BWRs. Although the releases from core-concrete interaction are predicted to extend many hours beyond corium-concrete interaction initiation, generally 90% of the radionuclide releases (except Te and Ru) occur within two hours for PWRs and three hours for BWRs. For tellurium and ruthenium, ex-vessel release durations of five hours for PWRs and six hours for BWRs are assumed.

In order to make a general assessment of the source terms proposed in this study, the ranges and distributions of source terms into the containment obtained by the NUREG-1150 methodology are presented for comparison in Appendices A and B. Some statistical parameters for total release into containment are also presented in Tables 6.3 and 6.4. The mean values of radionuclide releases into containment for low RCS pressure conditions are listed in Tables 6.5 for PWRs and 6.6 for BWRs. It should be noted that there was no NUREG-1150 elicitation for ex-vessel releases for BWRs with basaltic concrete. The ST_{EXV} values in Table 6.6 are the ratios of PWRs (basaltic/limestone) times BWRs (limestone).

Table 6.1 Updated Bounding Value of Radionuclide Releases Into the Containment Under Severe Accident Conditions for PWRs

	ST_{INV} [a]		ST_{VB}	ST_{EXV} [b]		ST_{REV}	
	High RCS Pressure	Low RCS Pressure	High RCS Pressure	Limestone Concrete	Basaltic Concrete	High RCS Pressure	Low RCS Pressure
NG	1.0	1.0	0.	0.	0.	0.	0.
I	0.30	0.75	0.10	0.15	0.15	0.05	0.02
Cs	0.30	0.75	0.10	0.15	0.15	0.02	0.02
Te	0.20	0.50	0.05	0.40	0.30	0.02	0.01
Sr-Ba	0.003	0.01	0.01	0.40	0.15	–	–
Ru	0.003	0.01	0.05	0.005	0.005	–	–
La-Ce	5×10^{-5}	1.5×10^{-4}	0.01	0.05	0.05	–	–
Release Duration	40 minutes			2 hours [c]		10 hours	

[a] All entries are fractions of initial core inventory.

[b] Assuming 100% of the core participate in CCI.

[c] Except for Te and Ru where the duration is extended to five hours.

Table 6.2 Updated Bounding Value of Radionuclide Releases Into the Containment Under Severe Accident Conditions for BWRs

	ST_{INV}[a]		ST_{VB}	ST_{EXV}[c]		ST_{REV}	
	High RCS Pressure	Low RCS Pressure[b]	High RCS Pressure	Limestone Concrete	Basaltic Concrete	High RCS Pressure	Low RCS Pressure[b]
NG	1.	1.	0.	0.	0.	0.	0.
I	0.50	0.75	0.10	0.15	0.15	0.10	0.02
Cs	0.50	0.75	0.10	0.15	0.15	0.05	0.01
Te	0.10	0.15	0.05	0.50	0.30	0.02	0.02
Sr-Ba	0.003	0.01	0.01	0.70	0.30	–	–
Ru	0.003	0.01	0.05	0.005	0.005	–	–
La-Ce	5×10^{-5}	1.5×10^{-4}	0.01	0.10	0.10	–	–
Release Duration	1.5 hours			3 hours[d]		10 hours	

[a] All entries are fractions of initial core inventory.

[b] High pressure ATWS are also considered in this category.

[c] Assuming 100% of the core participate in CCI.

[d] Except for Te and Ru where the duration is extended to six hours.

Table 6.3 Some Statistical Parameters for Total Release Into a PWR Containment Using NUREG-1150 Methodology

Conditions		ST_{CON} I	Cs	Te	Sr	Ba	Ru	La	Ce
Setpoint Pressure, Low Zr Oxidation, Limestone Concrete, Dry Cavity	Median	0.54	0.55	0.31	0.05	0.05	0.006	0.007	0.009
	Mean	0.53	0.54	0.32	0.12	0.11	0.03	0.015	0.022
	95th Percentile	0.95	0.96	0.66	0.41	0.38	0.16	0.057	0.078
Low RCS Pressure, Low Zr Oxidation, Limestone Concrete, Dry Cavity, Two Openings After VB	Median	0.87	0.86	0.56	0.07	0.07	0.001	0.003	0.005
	Mean	0.78	0.77	0.51	0.18	0.16	0.009	0.015	0.031
	95th Percentile	0.997	0.997	0.89	0.70	0.61	0.056	0.075	0.15
Setpoint Pressure, Low Zr Oxidation, Basaltic Concrete, Dry Cavity	Median	0.54	0.55	0.3	0.05	0.05	0.009	0.007	0.007
	Mean	0.53	0.54	0.31	0.12	0.11	0.03	0.018	0.021
	95th Percentile	0.95	0.96	0.65	0.47	0.42	0.16	0.06	0.077
Low RCS Pressure, Low Zr Oxidation, Basaltic Concrete, Dry Cavity, Two Openings After VB	Median	0.87	0.86	0.53	0.06	0.05	0.0008	0.002	0.004
	Mean	0.78	0.77	0.49	0.17	0.15	0.009	0.019	0.027
	95th Percentile	0.997	0.997	0.90	0.79	0.7	0.056	0.103	0.142

Table 6.4 Some Statistical Parameters for Total Release Into a BWR Containment Using NUREG-1150 Methodology

		ST_{CON}							
Conditions		I	Cs	Te	Sr	Ba	Ru	La	Ce
High Pressure Fast Station Blackout, Low Zr Oxidation, Limestone Concrete, Dry Pedestal, High Drywell Temperature	Median	0.56	0.58	0.35	0.07	0.07	0.009	0.0085	0.011
	Mean	0.56	0.55	0.34	0.19	0.17	0.031	0.0175	0.04
	95th Percentile	0.94	0.96	0.71	0.58	0.54	0.166	0.06	0.24
Low Pressure Fast Station Blackout, Low Zr Oxidation, Limestone Concrete, Dry Pedestal, Low Drywell Temperature	Median	0.81	0.82	0.58	0.09	0.09	0.0006	0.004	0.0006
	Mean	0.74	0.72	0.53	0.29	0.26	0.008	0.02	0.04
	95th Percentile	0.998	0.999	0.90	0.97	0.89	0.05	0.116	0.19
High Pressure ATWS Sequences, Low Zr Oxidation, Limestone Concrete, Water Injection After VB	Median	0.75	0.7	0.35	0.06	0.07	0.009	0.008	0.011
	Mean	0.66	0.65	0.34	0.18	0.16	0.03	0.016	0.05
	95th Percentile	0.989	0.988	0.70	0.58	0.53	0.16	0.059	0.22

Table 6.5 Mean Values of Radionuclide Releases Into Containment Under Severe Accident Conditions (PWRs, Low RCS Pressure, High Zr Oxidation, Dry Cavity, Two Openings After VB)

	ST_{INV} [a]	ST_{EXV} [b]		ST_{REV}
		Limestone Concrete	Basaltic Concrete	
NG	1.0	0.	0.	0.
I	0.40	0.29	0.29	0.07
Cs	0.30	0.39	0.39	0.06
Te	0.15	0.29	0.28	0.025
Sr	0.03	0.12	0.11	–
Ba	0.04	0.1	0.08	–
Ru	0.008	0.004	0.004	–
La	0.002	0.015	0.01	–
Ce	0.01	0.02	0.01	–

[a] All entries are fractions of initial core inventory.

[b] Assuming 100% of the core participate in CCI.

Table 6.6 Mean Values of Radionuclide Releases Into Containment Under Severe Accident Conditions (BWRs, Low RCS Pressure, High Zr Oxidation, Dry Pedestal, High Drywell Temperature)

	ST_{INV}[a]	ST_{EXV}[b]		ST_{REV}
		Limestone Concrete	Basaltic Concrete	
NG	1.0	0.	0.	0.
I	0.27	0.37	0.37	0.07
Cs	0.2	0.45	0.45	0.03
Te	0.11	0.38	0.37	0.01
Sr	0.03	0.24	0.22	–
Ba	0.03	0.21	0.17	–
Ru	0.007	0.004	0.004	–
La	0.002	0.01	0.007	–
Ce	0.009	0.01	0.005	–

[a] All entries are fractions of initial core inventory.

[b] Assuming 100% of the core participate in CCI.

7. Summary

A detailed review of the available source term information for light water reactors has been performed. This information is provided to support the generation of updated source terms into containment under severe accident conditions.

Estimates of radionuclide release and transport characteristics were specified for each unique combination of reactor coolant and containment system conditions. The characteristics of the radionuclide releases in this study are clearly different than the hypothetical source term proposed in TID-14844.

The similarities between the NUREG-1150 method to estimate source terms and that proposed in this study allowed the use of NUREG-1150 expert elicitation on source term issues to assess the uncertainty inherent in the release estimates. An assessment of the range and distribution of the source terms into containment obtained by the NUREG-1150 methodology was also presented in this report.

8. References

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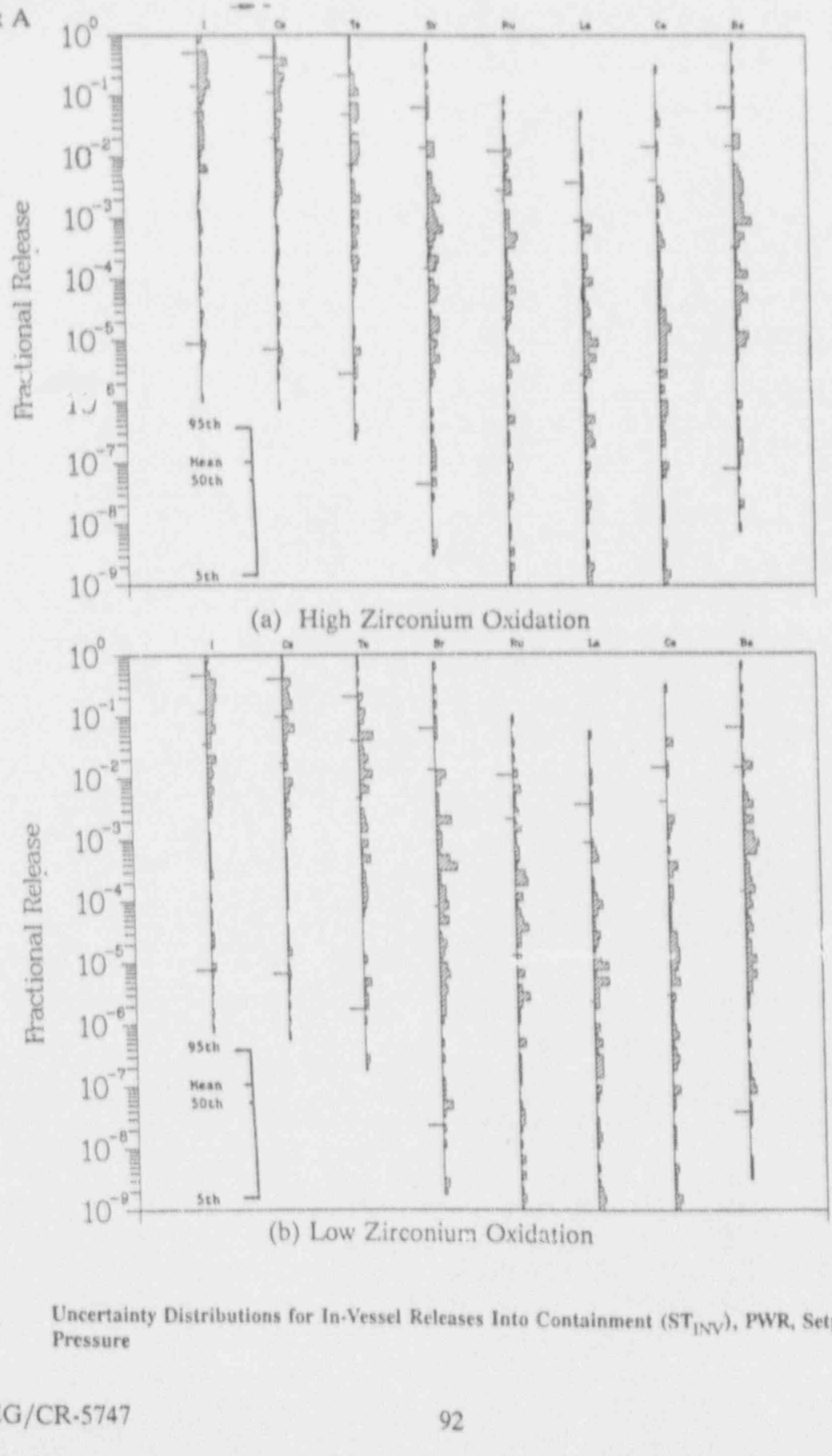
*Available in the NRC Public Document Room, 2120 L Street NW., Washington, D.C.

APPENDIX A

UNCERTAINTY DISTRIBUTIONS FOR IN-VESSEL RELEASES INTO CONTAINMENT



(a) High Zirconium Oxidation

(b) Low Zirconium Oxidation

Figure A.1 Uncertainty Distributions for In-Vessel Releases Into Containment (ST_{INV}), PWR, Setpoint Pressure

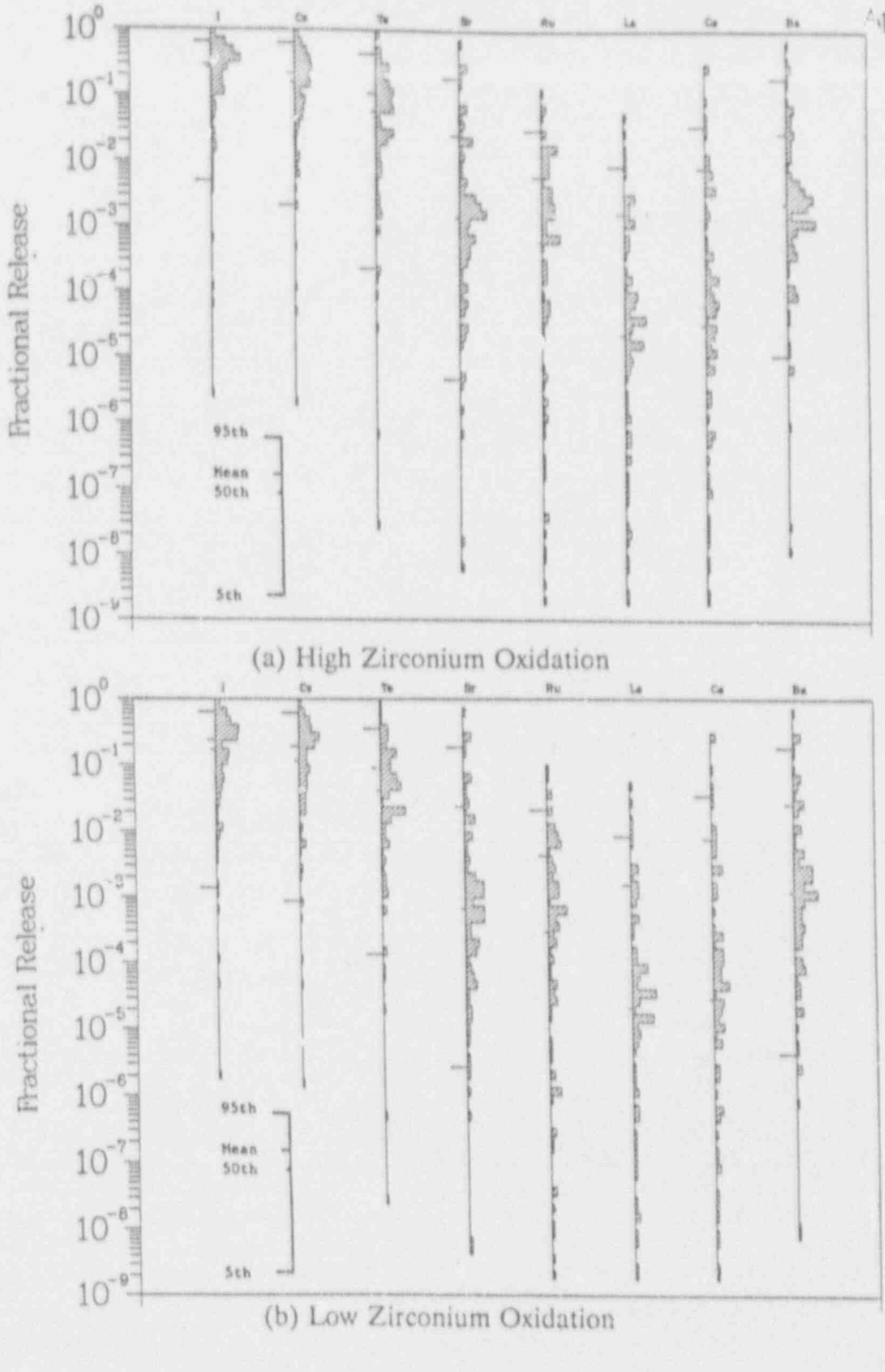


Figure A.2 **Uncertainty Distributions for In-Vessel Releases Into Containment (ST_{INV}) PWR, High and Intermediate RCS Pressure**

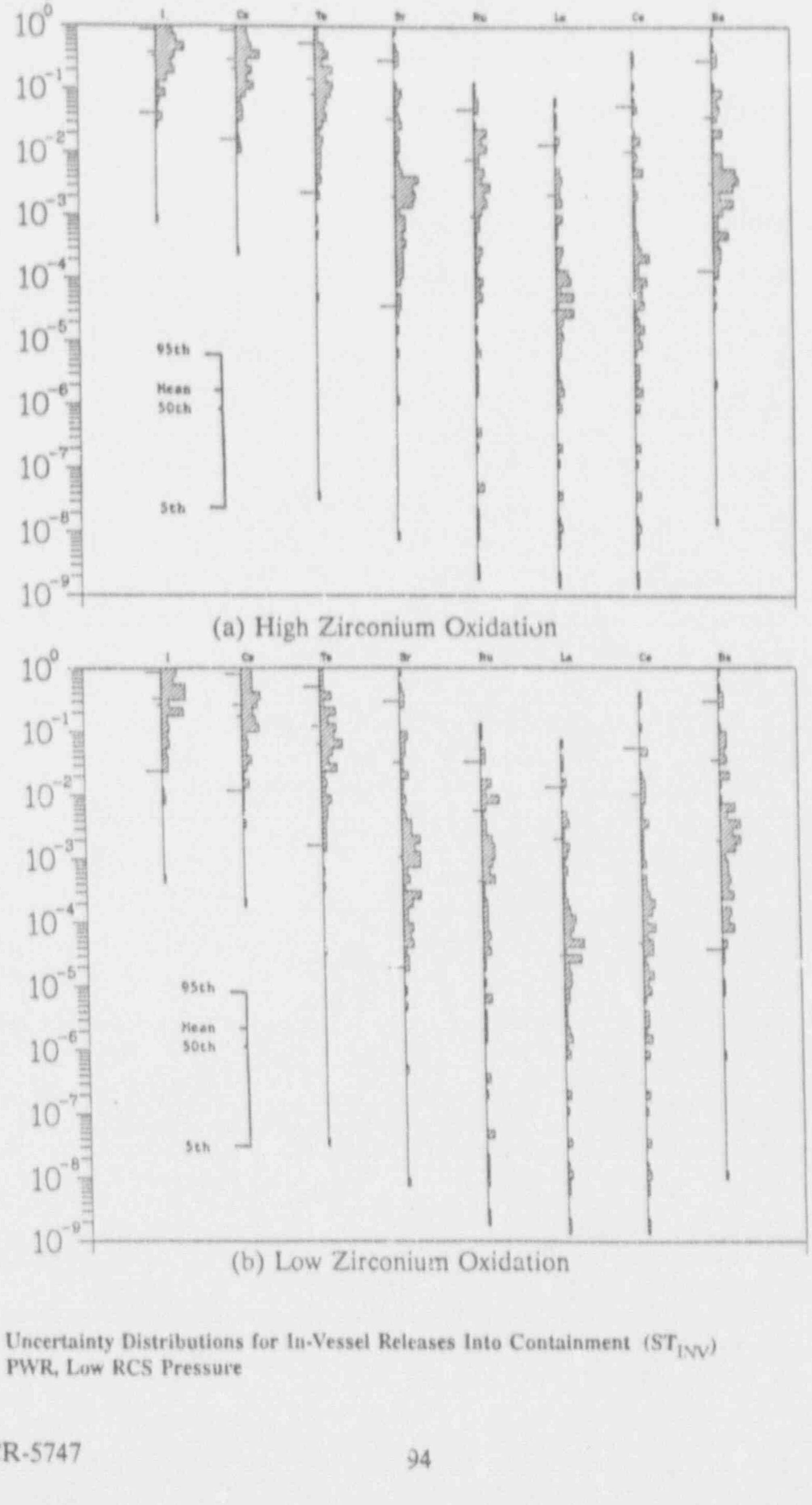
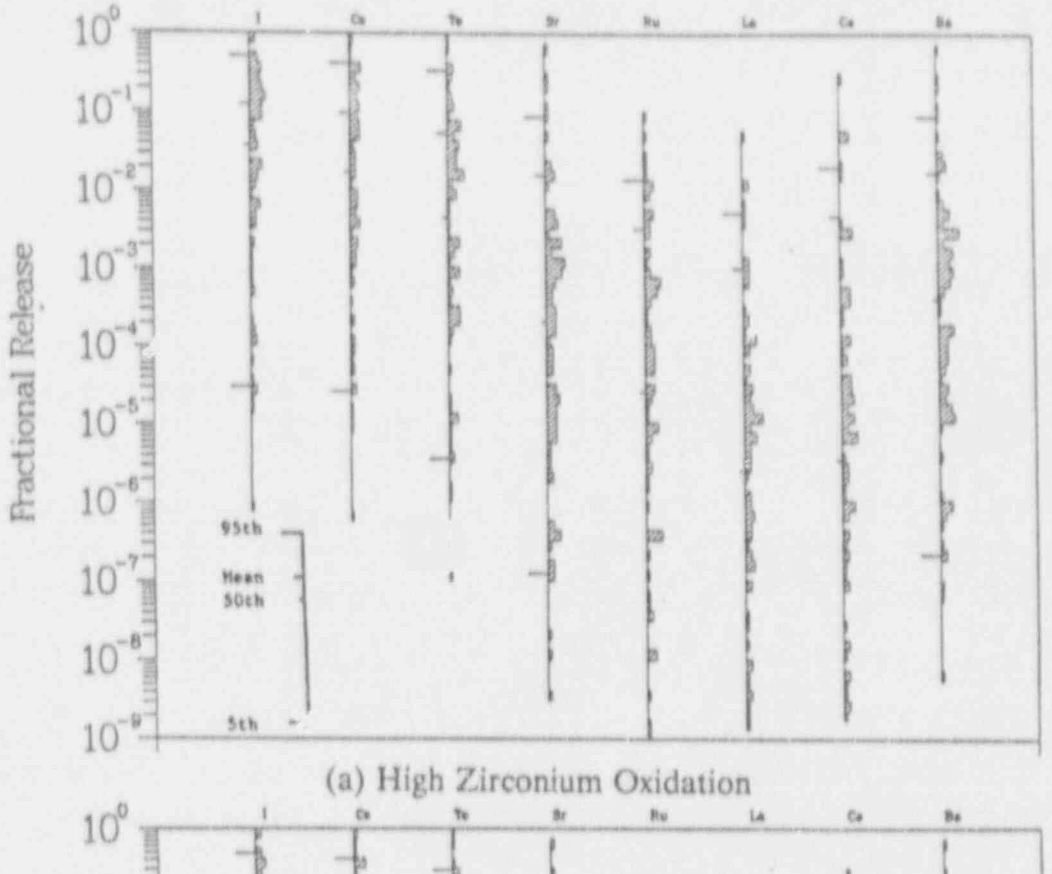


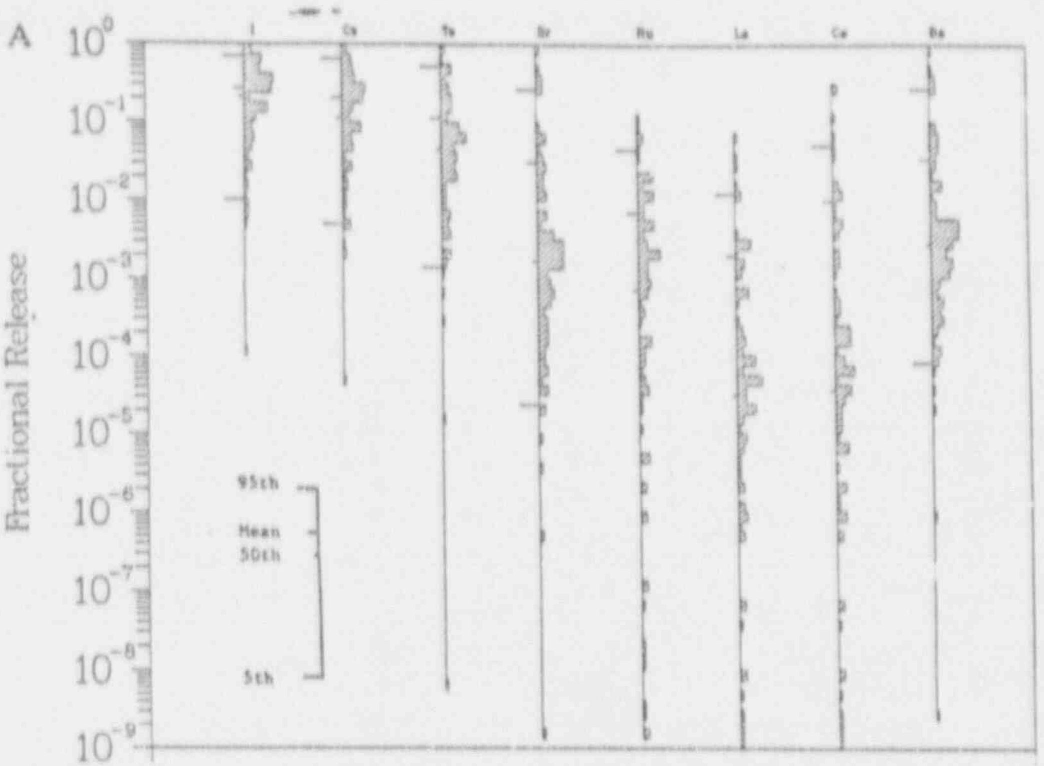
Figure A.3 Uncertainty Distributions for In-Vessel Releases Into Containment (ST_{INV}) PWR, Low RCS Pressure



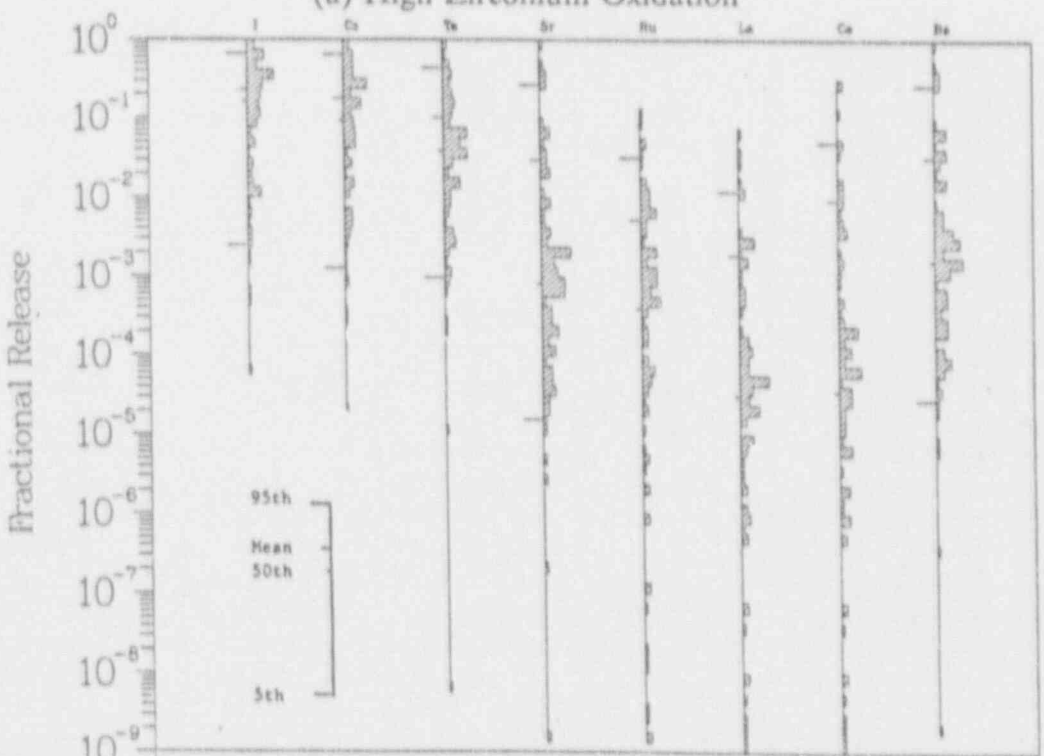
(a) High Zirconium Oxidation

(b) Low Zirconium Oxidation

Figure A.4 Uncertainty Distributions for In-Vessel Releases Into Containment (ST_{INV}) BWR, High Pressure Fast Station Blackout



(a) High Zirconium Oxidation



(b) Low Zirconium Oxidation

Figure A.5 Uncertainty Distributions for In-Vessel Releases Into Containment (ST_{INV}) BWR, Low Pressure Fast Station Blackout

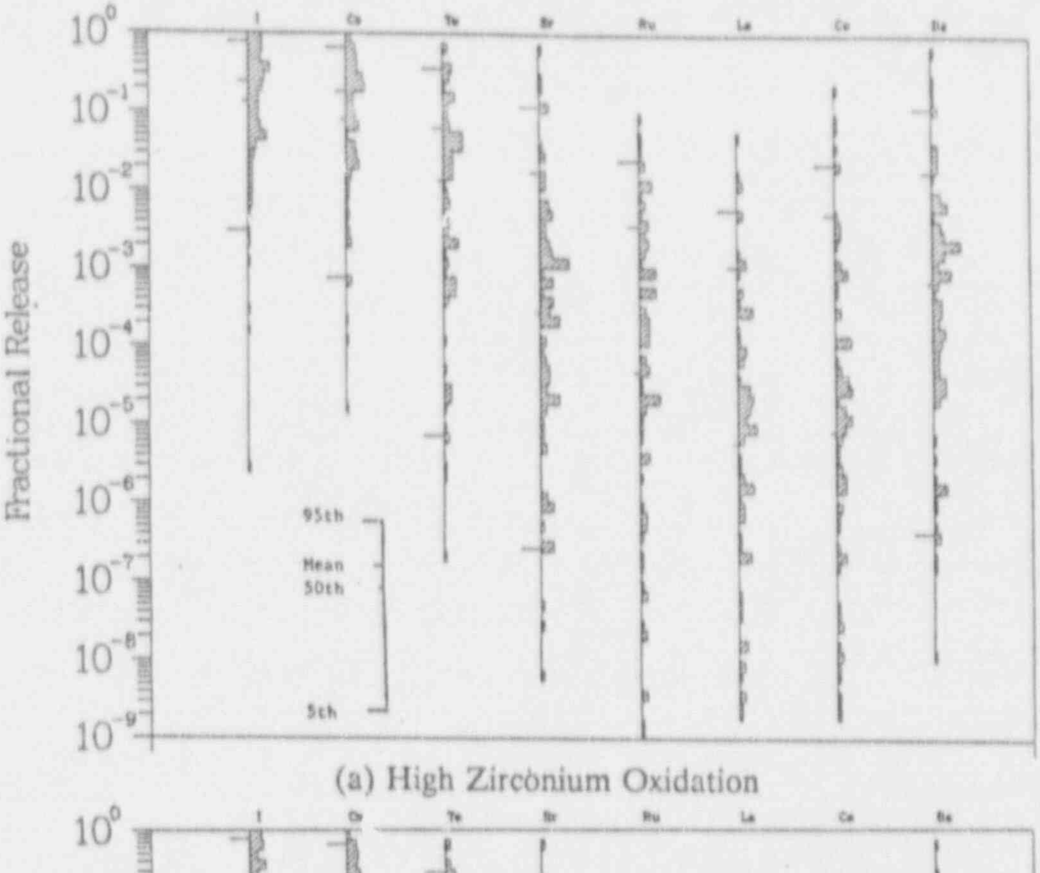
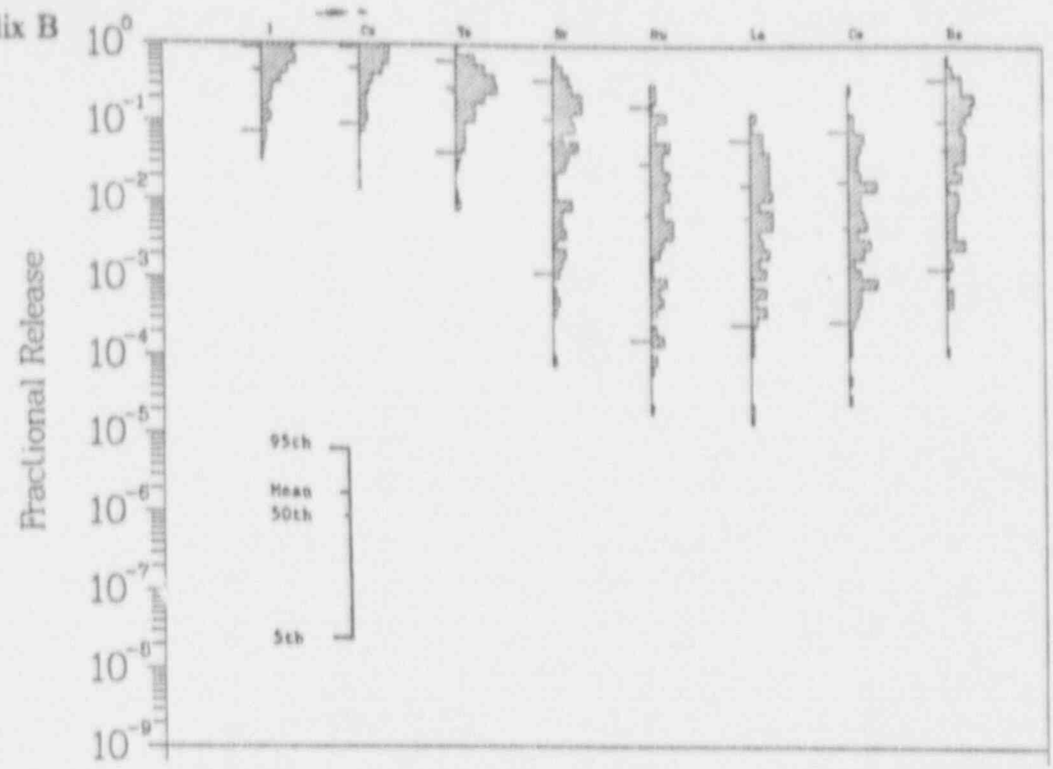


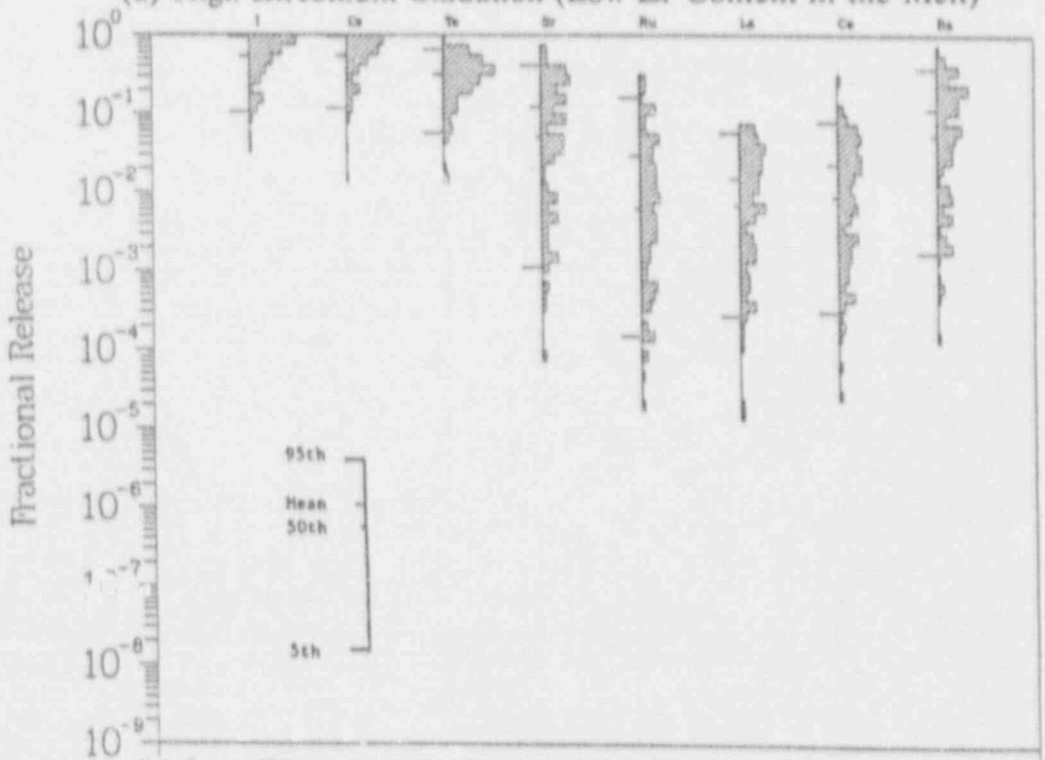
Figure A.6 Uncertainty Distributions for In-Vessel Releases Into Containment (ST_{INV}) BWR, High Pressure ATWS Sequences

APPENDIX B

UNCERTAINTY DISTRIBUTIONS FOR TOTAL RADIONUCLIDE RELEASES INTO CONTAINMENT

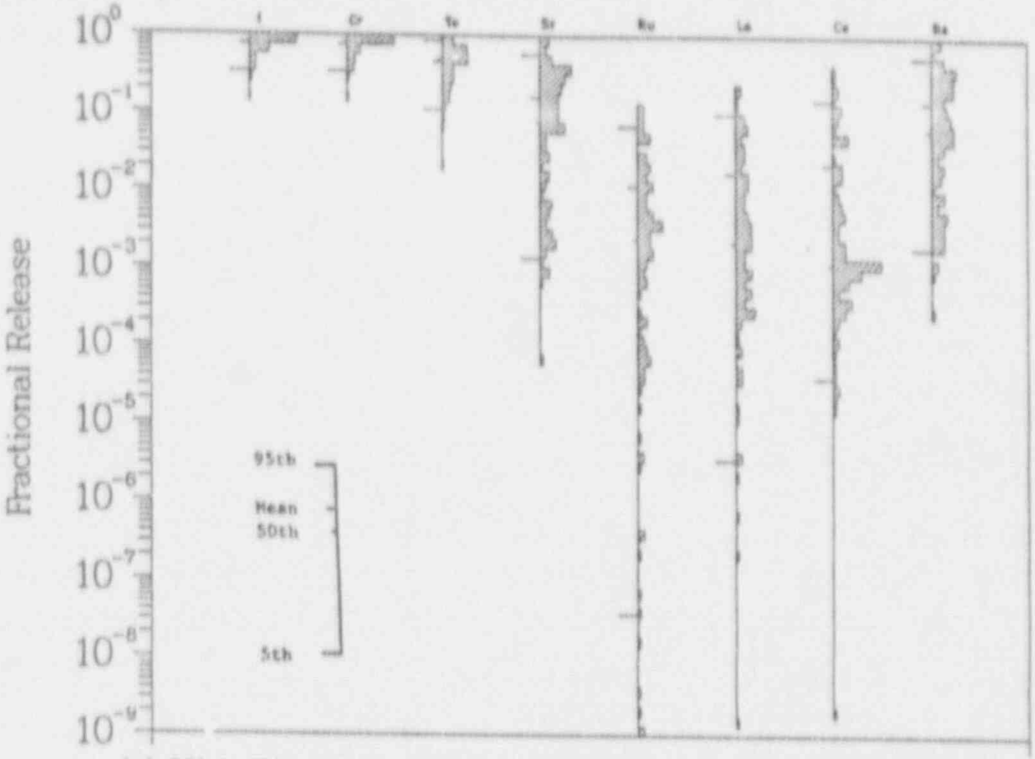


(a) High Zirconium Oxidation (Low Zr Content in the Melt)

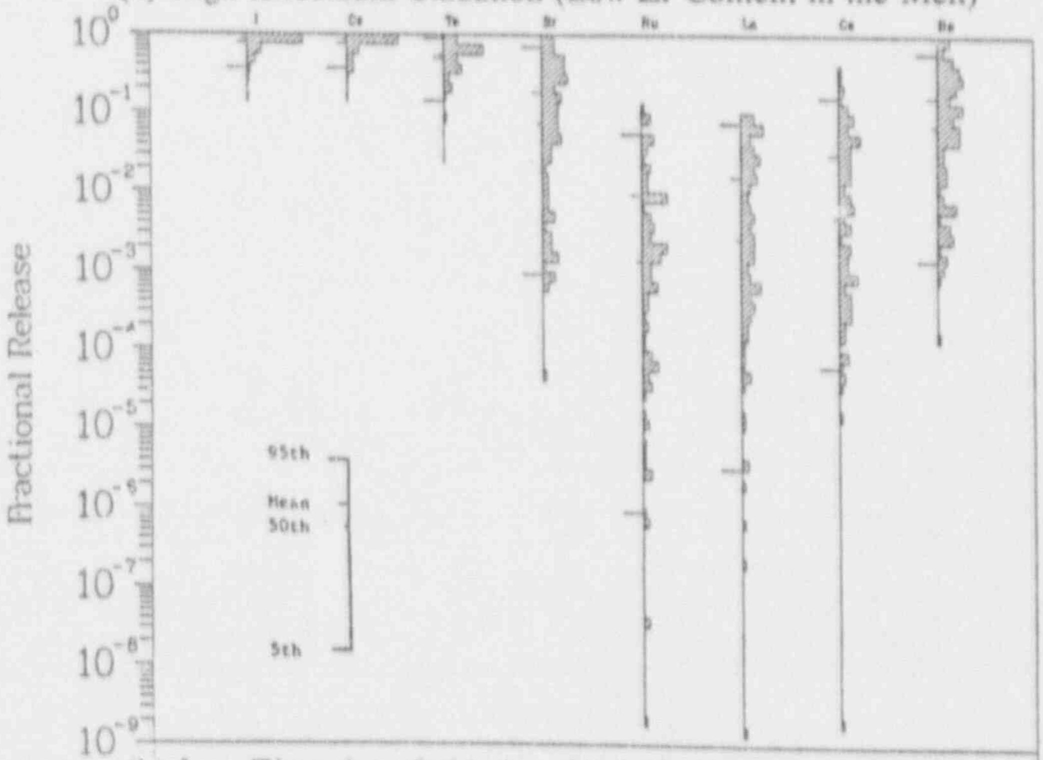


(b) Low Zirconium Oxidation (High Zr Content in the Melt)

Figure B.1 **Uncertainty Distributions for Total Releases Into Containment PWR, Setpoint Pressure, Limestone Concrete, Dry Cavity, FPART = 0.6, FPME = 0.4.**

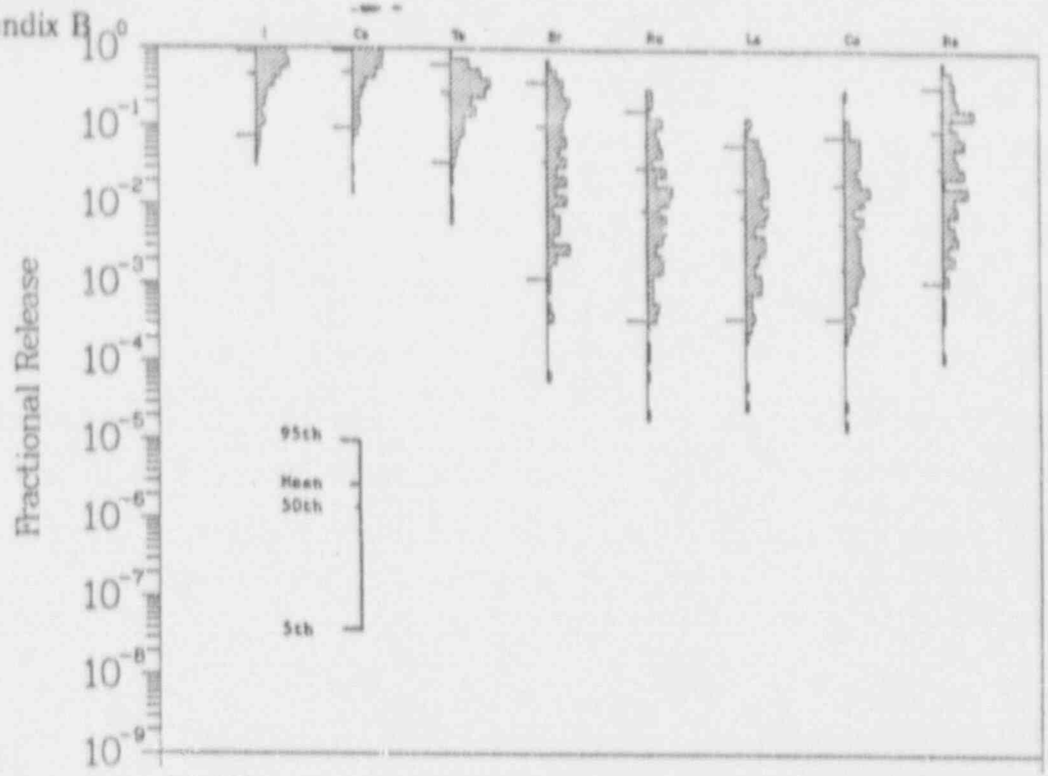


(a) High Zirconium Oxidation (Low Zr Content in the Melt)

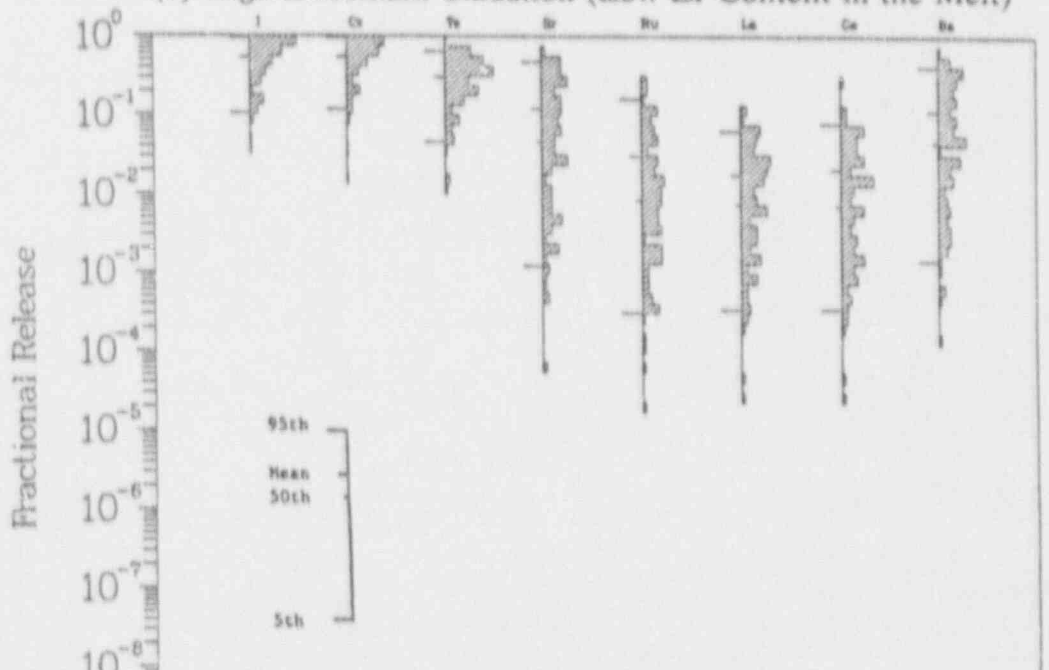


(b) Low Zirconium Oxidation (High Zr Content in the Melt)

Figure B.2 Uncertainty Distributions for Total Releases Into Containment PWR, Low RCS Pressure, Limestone Concrete, Dry Cavity, Two Openings After VB, FPART = 1.

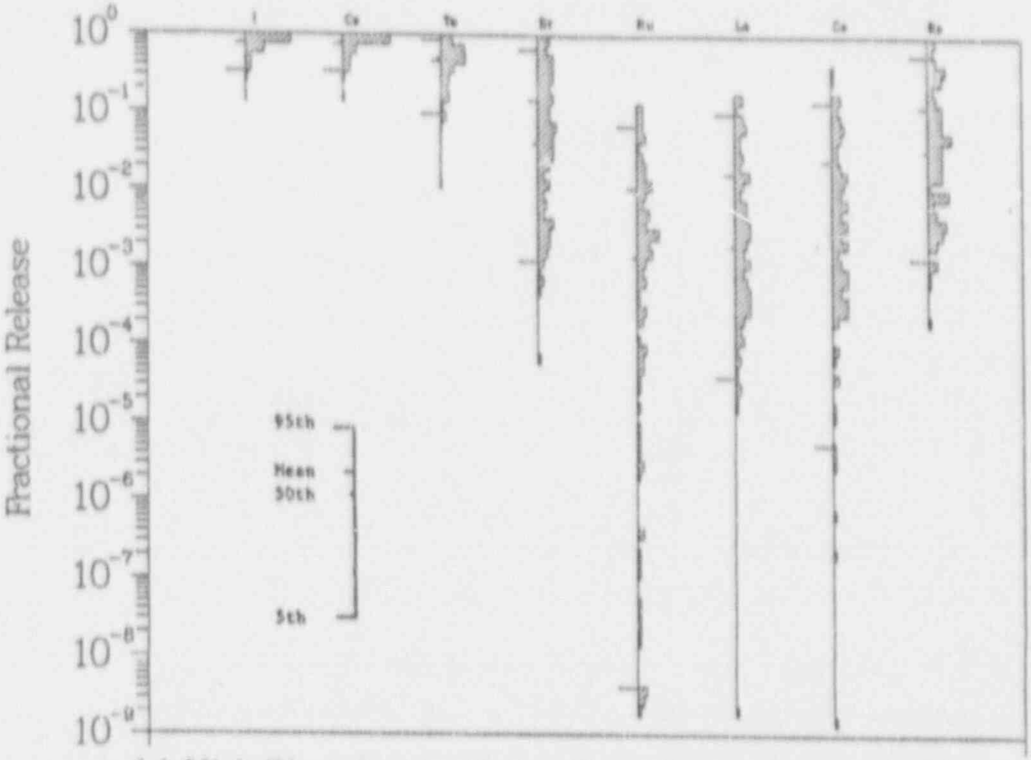


(a) High Zirconium Oxidation (Low Zr Content in the Melt)

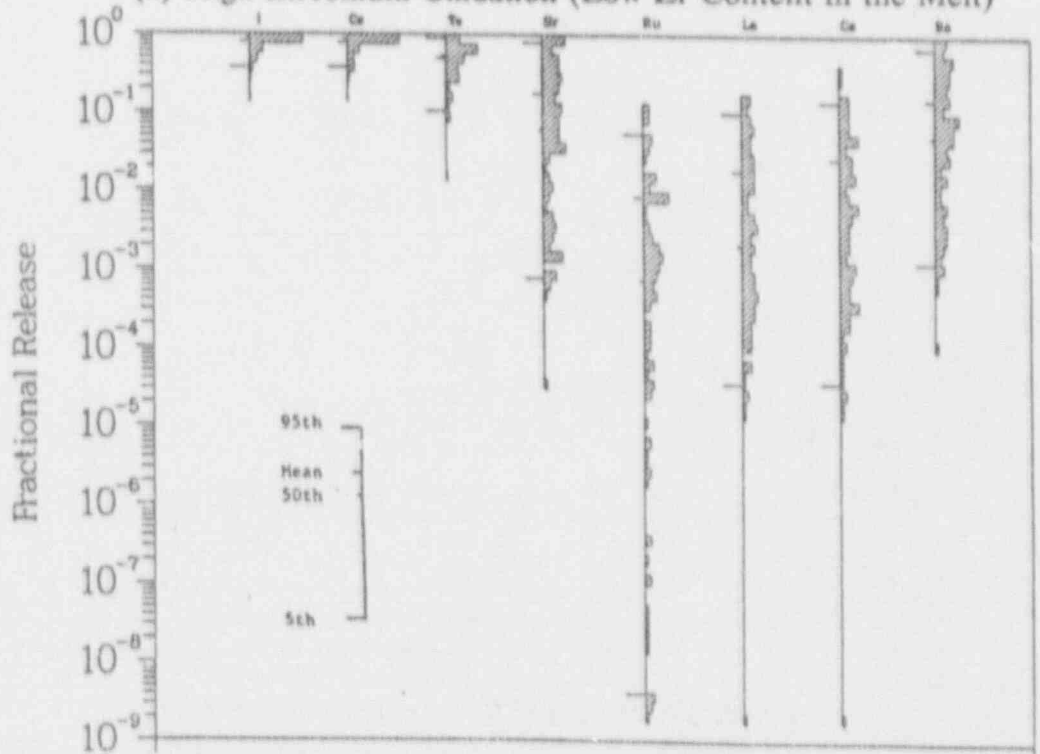


(b) Low Zirconium Oxidation (High Zr Content in the Melt)

Figure B.3 Uncertainty Distributions for Total Releases Into Containment PWR, Setpoint Pressure, Basaltic Concrete, Dry Cavity, FPART = 0.6, FPME = 0.4.

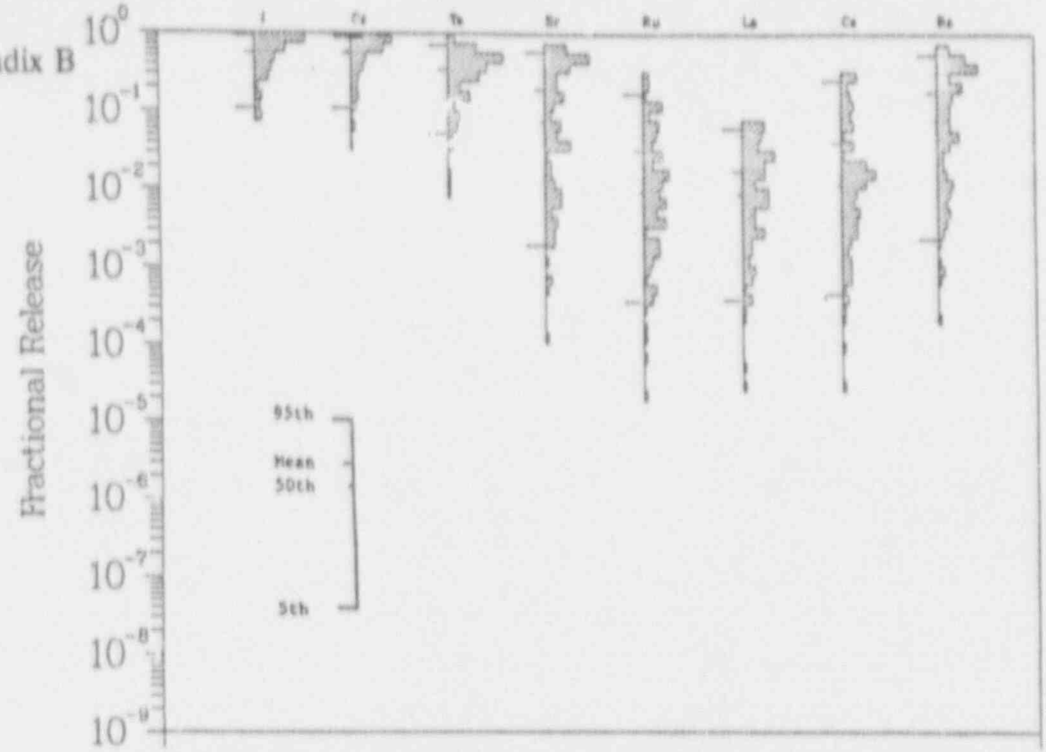


(a) High Zirconium Oxidation (Low Zr Content in the Melt)

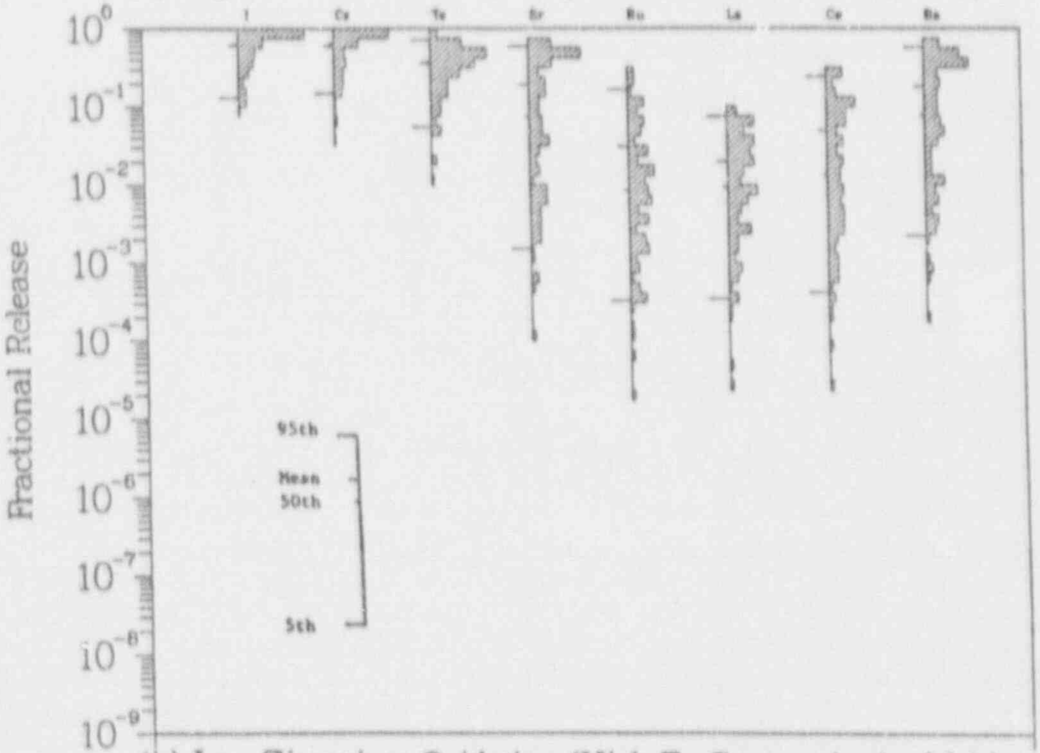


(b) Low Zirconium Oxidation (High Zr Content in the Melt)

Figure B.4 Uncertainty Distributions for Total Releases Into Containment PWR, Low RCS Pressure, Basaltic Concrete, Dry Cavity, Two Openings After VB, FPART = 1.



(a) High Zirconium Oxidation (Low Zr Content in the Melt)



(b) Low Zirconium Oxidation (High Zr Content in the Melt)

Figure B.5 **Uncertainty Distributions for Total Releases Into Containment BWR, High Pressure Fast Station Blackout, Limestone Concrete, Dry Pedestal, High Drywell Temperature, FPART = 0.6, FPME = 0.4.**

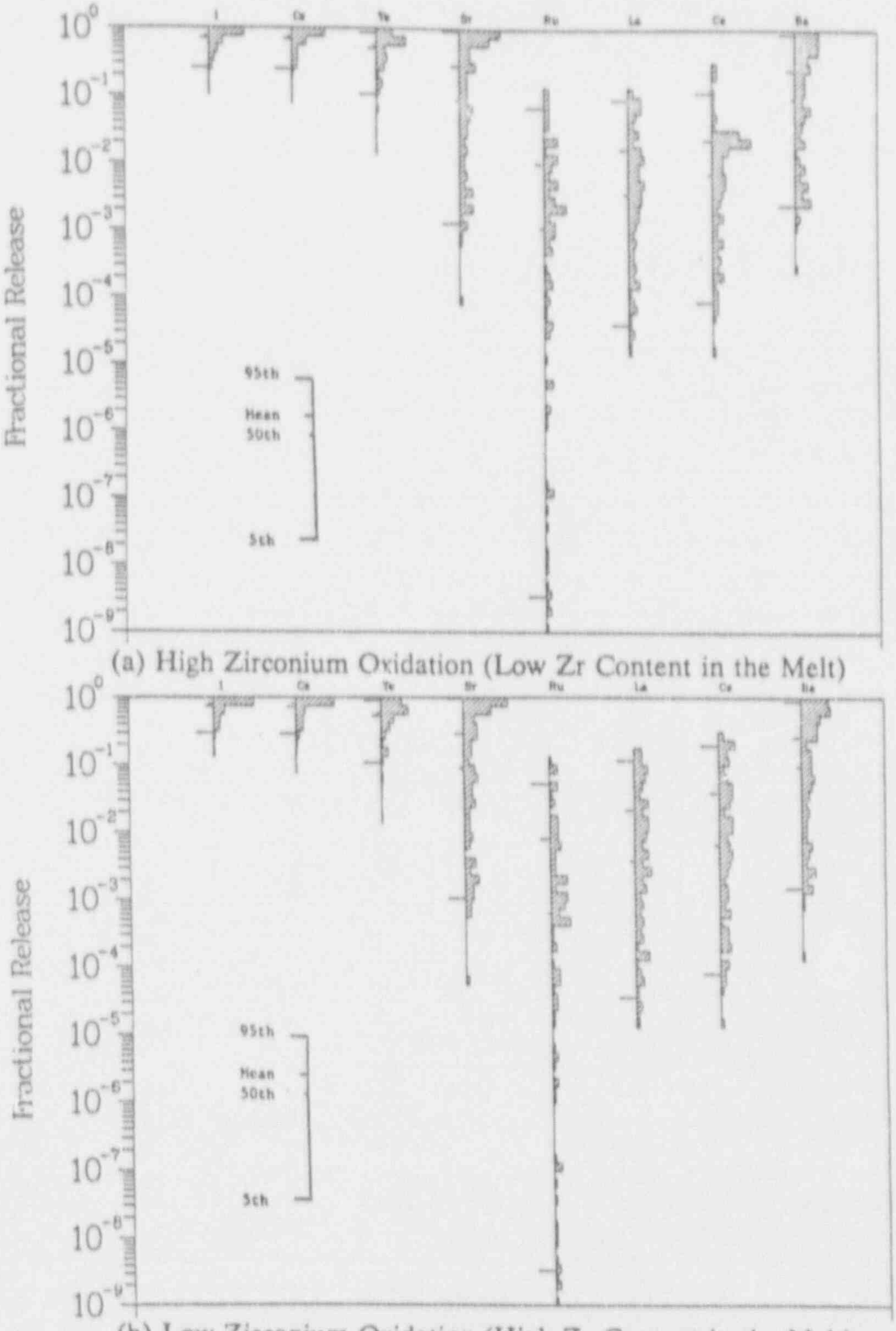
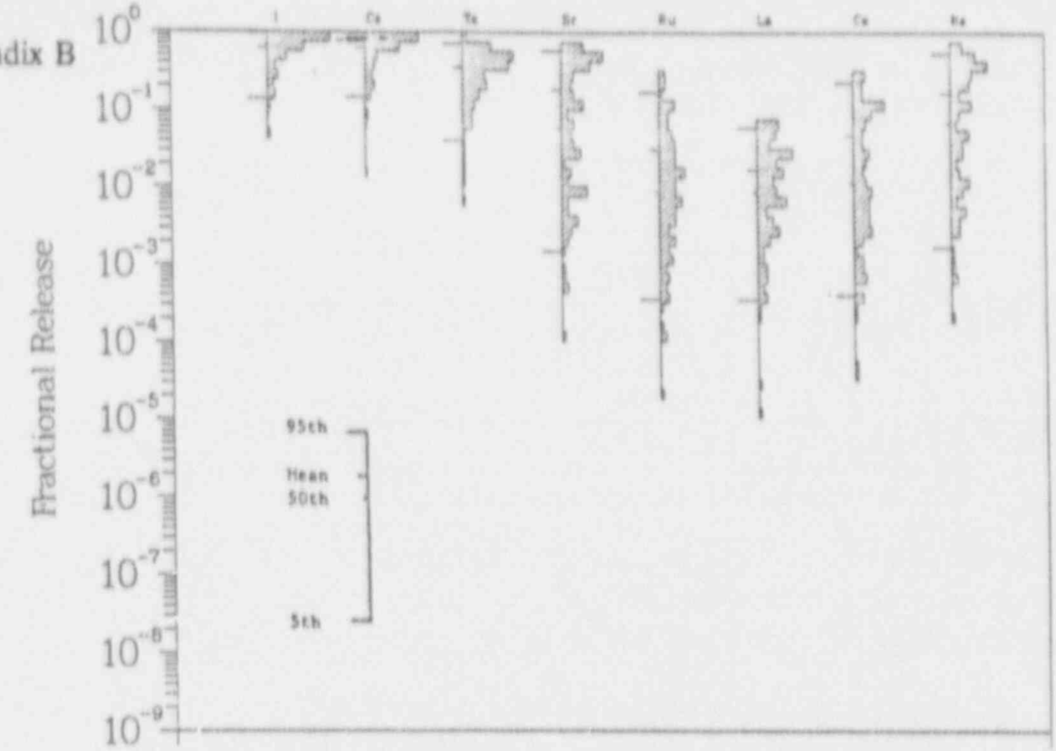
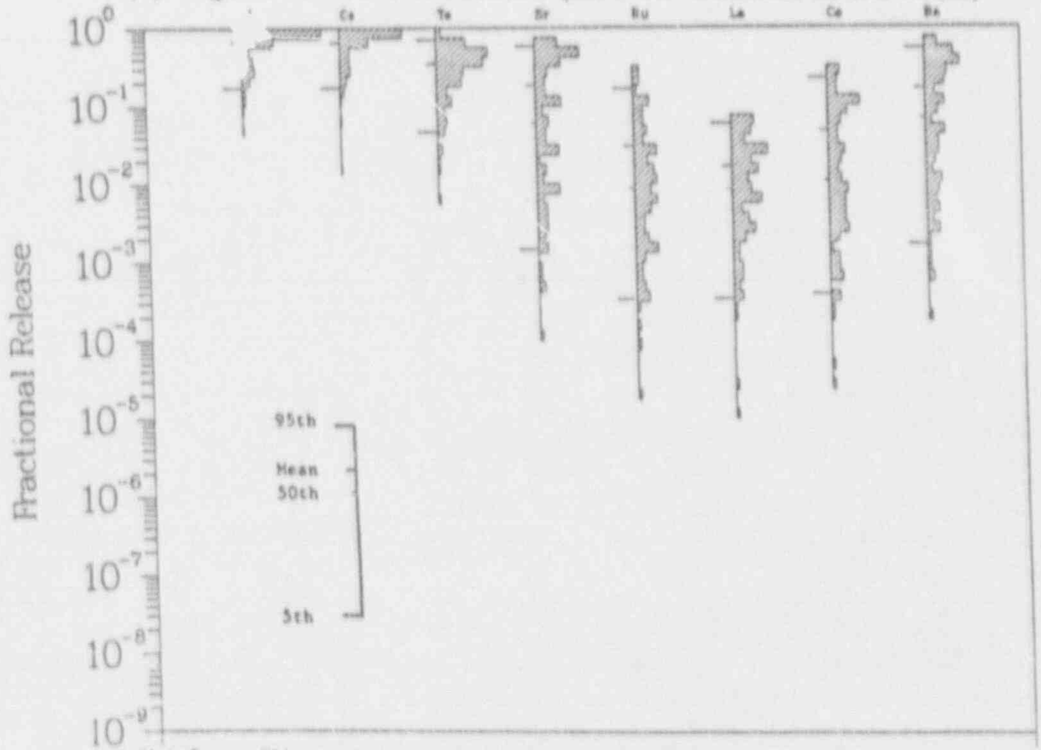


Figure B.6 Uncertainty Distributions for Total Releases Into Containment BWR, Low Pressure Fast Station Blackout, Limestone Concrete, Dry Pedestal, Low Drywell Temperature, FPART = 1.



(a) High Zirconium Oxidation (Low Zr Content in the Melt)



(b) Low Zirconium Oxidation (High Zr Content in the Melt)

Figure B.7 Uncertainty Distributions for Total Releases Into Containment BWR, High Pressure ATWS Sequence, Limestone Concrete, Water Injection After VB, FPART = 0.6, FPME = 0.4.

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A detailed review of the available light water reactor source term information is presented as a technical basis for development of updated source terms into the containment under severe accident conditions. Simplified estimates of radionuclide release and transport characteristics are specified for each unique combination of the reactor coolant and containment system combinations. A quantitative uncertainty analysis in the release to the containment using NUREG-1150 methodology is also presented.

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ESTIMATE OF RADIONUCLIDE RELEASE CHARACTERISTICS INTO CONTAINMENT UNDER SEVERE ACCIDENT CONDITIONS

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