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FISSION PRODUCT RELEASE CHARACTERISTICS INTO CONTAINMENT UNDER DESIGN BASIS AND SEVERE ACCIDENT CONDITIONS

H.P. Nourbakhsh, M. Khatib-Rahbar, and R.E. Davis

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ACCIDENT ANALYSIS GROUP
DEPARTMENT OF NUCLEAR ENERGY
BROOKHAVEN NATIONAL LABORATORY
UPTON, NEW YORK 11973

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ABSTRACT

A detailed review of the radiological release estimates for light water reactor accident sequences is presented as a basis for development of a simplified approach for prediction of characteristics of radiological releases into containments under design basis and severe accident conditions for both Pressurized Water Reactors (PWRs) and Boiling Water Reactors (BWRs). Resulting source term estimates are also compared with parallel results using the Source Term Code Package (STCP) methodology.

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FOREWORD

Accident source terms are important in nuclear safety regulation. Current NRC regulations (10 CFR 100) require that the suitability of the reactor site be judged in conjunction with 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"). Postulated are the instantaneous in containment releases of 100 percent of the full power noble gas fission products, 50 percent of the iodine fission products (half of which are assumed to deposit on interior surfaces), and 1 percent of the remainder of 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).

Use of the TID-14844 release 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) the radiation accident environment for which safety-related equipment has been qualified, (2) post-accident habitability requirements for the control room, (3) performance of important fission-product cleanup systems such as sprays and filters, (4) post-accident sampling systems and accessibility, (5) containment leak rates, and (6) containment isolation valve closure times.

The research results given in this report provide significant new information on the magnitude, timing and composition of fission product releases into containment based upon a spectrum of severe accidents. Therefore, the information can provide a useful basis to update and revise, as appropriate, the formulation given in TID-14844, as well as to allow for improved related regulatory practices.

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EXECUTIVE SUMMARY

The source term to containment is defined as the quantity, timing, and chemical form of the fission product species released to the reactor containment building atmosphere during core damage accidents.

The current source term assumption used in the licensing process in the United States is based on data that was obtained from burning irradiated uranium metal in air. This data formed the basis for the TID-14844 document published in 1962.

The present U.S. Nuclear Regulatory Commission (USNRC) regulatory framework, involving the use of the TID-14844 radiological release assumptions in connection with the evaluation of design basis accidents, treats design basis events in an inconsistent manner with respect to radiological source terms and thermal-hydraulic conditions. In this framework the design basis accident thermal-hydraulic conditions (namely, no core damage) are combined with fission product release assumptions associated with significant core damage. Significant research activity in the area of severe accidents has been undertaken following the accident at Three Mile Island Unit 2. Updated fission product source term methods have been developed and published in NUREG-0956.

The objectives of this report are to review the available light water reactor source term information, and to formulate a consistent and simplified approach for the estimation of radiological releases to containment for accidents involving significant fuel damage. The phenomenological aspects of degraded core accidents are discussed and key factors affecting the characteristics of fission product release into containment are identified.

A simplified formulation for source term releases to containment is proposed. Two basic assumptions govern the validity of the proposed approach; firstly, the fission product species are grouped according to their respective chemical form and release characteristics and, secondly, the accident conditions must be categorized into appropriate severe accident attributes which govern the release. These attributes include: (1) reactor type (BWR vs. PWR), (2) RCS pressure prior to vessel breach (high vs. low), (3) concrete aggregate (limestone vs. basaltic), and cavity/pedestal condition (dry vs. flooded). Appropriate decontamination factors, depending on the path of release, are also applied.

The relevant parameters, including the timing of release, are based upon the results of the recent Source Term Code Package calculations performed in support of draft NUREG-0956 and draft NUREG-1150 studies.

The approach presented in this report is expected to provide a simplified framework for estimating fission product release rates into the containments of Light Water Reactors (LWRs).

1. INTRODUCTION

1.1 Background

During accidents in Light Water Reactors (LWRs) the reactor core could be damaged and fission products may be released to the primary system. If the reactor coolant system is breached, fission products could in turn be released to the containment building. In containment there are systems available to help prevent the fission products from being released to the environment. If these systems fail or are compromised, a fraction of the radionuclides may be released to the atmosphere with corresponding adverse effects on the surrounding environment. There is potentially a large number of different accident sequences that could lead to core damage and ultimately to core meltdown. Each individual accident sequence could result in several possible paths for fission products to reach the environment.

In order to define a "source term," information is needed on the amount and chemical form of the fission product species released and also on the characteristics of the release.

The spectrum of accidents considered in the nuclear regulatory and licensing process includes those postulated as the design basis accidents, and beyond the design basis accidents.

The traditional and still current surrogate for beyond the design basis accident end of the accident spectrum used in the licensing process is based on the quantity of radioactivity released into the containment. This release is postulated for testing the acceptability of the combination of the plant design features for reducing the amount of radioactivity released from the containment and the proposed site.[1] The containment source term is derived from TID-14844,[2] published in 1962.

In TID-14844,[2] the postulation of a substantial core melt accident takes the form of the release of all of the noble gases, 50 percent of the iodine, and 1 percent of the radioactive solids to the containment. In addition, this document provides assumptions regarding containment leakage and the atmospheric transport of the fission products. The procedures and results embedded in the TID report are stated by the authors to be approximations, sometimes relatively poor ones, to actual accident conditions. Therefore, in setting the magnitude of releases into but not out of the containment, highly conservative estimates in the upward directions were assigned.

The evaluation of severe reactor accident source terms over the period from 1957 to approximately 1981 has been summarized in Reference 3 and fully described in Reference 4.

Significant research activity in the area of severe accidents has been undertaken following the accident at Three Mile Island Unit 2. Updated fission product source term methods were developed under the direction of the Accident Source Term Program Office (ASTPO) and initially published in BMI-2104.[5] The technical reassessment of severe accident source term technology for U.S. light water reactors has been published in NUREG-0956.[3] This reassessment involved reviewing experimental and analytical results from severe accident research programs that have received significant emphasis after the accident at TMI-2. As part of this major activity, the U.S. Nuclear Regulatory Commission (NRC) has developed the Source Term Code Package (STCP)[6] as a method to predict radiological releases from severe nuclear reactor accidents.

1.2 Objectives

The current NRC regulatory framework, involving the use of the TID-14844[2] in containment release assumptions in connection with the evaluation of design basis accidents, treats design basis events in an inconsistent manner with respect to radiological source terms and thermal hydraulic conditions, and thereby, the approach may be inappropriate with regard to the effects of severe accidents. An important NRC goal is the potential development of a revised regulatory framework or principle that employs the use of consistent and realistic source terms.[7]

The objective of the present report is to review the available[8-18] light water reactor source term information, and to formulate a consistent, realistic and simplified source term approach for estimation of radiological releases during all phases of (a) design basis accidents (DBAs), and (b) beyond the design basis conditions.

2. HISTORICAL DEVELOPMENTS

Over the past 30 years a continually developing understanding of reactor accident phenomena and the mechanisms affecting the release of radionuclides during reactor accidents has allowed more detailed and realistic estimates to be made of the effects of postulated accidents at light water reactors.

The first estimates of severe accidental releases of radioactive material, found in the U.S. Atomic Energy Commission report WASH-740[19], published in 1957, was an attempt to provide realistic upper bounds of the potential public hazards resulting from certain severe hypothetical accidents. Since a definitive understanding of all the physical processes associated with accident phenomena was lacking, pessimistic values were used for many factors influencing the magnitude of the estimated accident consequences. One particular limitation was the inability at that time to quantitatively describe the physical mechanism governing the release and transport of radionuclides from the core to the environment during severe accidents.

Using WASH-740 as a basis, regulations for site selection were developed as 10 CFR, Part 100, "Reactor Site Criteria." In conjunction with part 100, the concept of a maximum credible accident was developed as the mechanism to evaluate the acceptability of the potential site. The maximum credible accident concept was devised to evaluate siting limits and containment design requirements. In 1962, the maximum credible accident was described in TID-14844[2] "Calculation of Distance Factors for Power and Test Reactor Site." The TID source term, as it is known, postulated a loss of coolant accident (LOCA) upon complete rupture of a major coolant pipe, followed by substantial core meltdown and release of all noble gases, fifty percent of the iodine, and one percent of the other particulate materials (solids) to the atmosphere of the reactor building (containment).

Since the inception 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 in an integral way. In addition, 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 primary system and containment and into the environment. Each major contribution 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[8] (WASH-1400), was the first systematic attempt to provide realistic estimates of the public risk from potential accidents in commercial nuclear power plants. The study, published in 1975, 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 of 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: gap release, meltdown release, vaporization release, and oxidation release. The gap release occurs when the claddings initially experience a failure. This release consists mostly of activity that was released to void spaces within the fuel rods during normal reactor operation. Meltdown release occurs from fuel prior to reactor pressure vessel failure. The vaporization release occurs from the corium melt during 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 to develop simple retention factors for the primary system transport in the RSS. 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 was used in all calculations of PWR accidents regardless of pipe break location. In BWR accident sequences where ECCS is operational, an escape fraction of one was used for noble gas, 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; Surry, a 3-loop Pressurized Water Reactor (PWR) with a large dry containment and Peach Bottom, a Boiling Water Reactor (BWR) with Mark I containment. Nine PWR release categories and five BWR release categories were developed in 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 20 and 21. 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 RSS and the accident at TMI-2, work was initiated to review the state-of-the-art related to predictive methods for calculating fission product release and transport. The results of this review are contained in NUREG-0772.[9] That review resulted in several conclusions that represented significant departures from the RSS assumptions. These conclusions have been summarized in NUREG-0956[3] and are reproduced here:

Table 2.1 Fission Product Releases Developed in the RSS*

Fission Product	Gap Release Fraction	Meltdown Release Fraction	Vaporization Release Fraction[d]	Steam Explosion 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[8]," Appendix VII, WASH-1400, October 1975, Table VII 1-6.

Table 2.2 Summary of RSS Primary 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 2-6	1	1	0.1	0.7

1. The data base suggests that cesium iodide (CsI) will be the expected predominant iodine chemical form under most postulated LWR accident conditions. The evidence regarding the chemical form of iodine released from fuel at high temperature (>1400°C) is inconclusive. Thermodynamic calculations predict that formation of CsI should occur in the gaseous reducing atmosphere in the reactor coolant system following release from fuel even if iodine is not released from the fuel as CsI. However, the formation of some more volatile iodine species (e.g., elemental iodine and organic iodines) cannot be precluded under certain accident conditions.

2. The assumed form of iodine was not predicted to have a major influence on iodine attenuation in the containment for early containment failure. However, the assumed chemical form can influence calculated attenuation within the reactor coolant system, where, in general, the attenuation will be greater for CsI than was assumed in the Reactor Safety Study for elemental iodine.

3. Predictions of the retention of radioactive material within the reactor coolant system (which was not accounted for in the Reactor Safety Study for most accident sequences) range from very little to substantial, depending on the accident sequence. In addition, attenuation of fission products within the reactor coolant system could be substantial as a result of agglomeration and fallout of aerosols. Consequently, for certain accident sequences considered in the Reactor Safety Study, the release of radionuclides to the environment may have been significantly overpredicted. However, for other accident sequences, the estimated releases would be in approximate agreement with the Reactor Safety Study estimates.

4. Certain engineered safety features (e.g., containment sprays, suppression pools, and ice condensers) will perform effectively in removing fission products (except noble gases) regardless of their chemical form (i.e., vapor or particulate) and under most conditions. Other engineered safety features (e.g., filters) are less effective.[3]

In order to examine the potential impact of the NUREG-0772 findings on reactor regulation, a parallel effort was undertaken by the NRC, and its results were issued for public comment in NUREG-0771.[23]

These studies formed the basis for the development of a generic set of radiological releases (NUREG/CR-2239)[22] characterized as siting source terms (SST). These source terms were based on individual computer calculations that had been completed and documented in NUREG-0773.[4]

2.3 Current NRC Source Term Reassessment Study

Much of the quantitative assessment in NUREG-0772[9] 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[9], the NRC funded a source term study at Battelle Columbus Laboratories. 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 the appropriate feedback involved in realistic accident sequences. About 25 specific accident sequences for a selection of five operating plants were analyzed with this suite of codes. The purpose of the sequence analyses was to exercise the suite of codes over a wide range of conditions. Consequently, several sequences were chosen that had low probabilities of occurrence, and many sequences that might contribute significantly to the overall risk were not analyzed. The Battelle suite of codes and the sample analyses were reported in the multi-volume report, BMI-2104.[5]

As a result of these reassessment activities, the NRC's Source Term Code Package[6] (STCP) has emerged as an integrated tool for severe accident analysis. The STCP (an upgraded version of the BMI-2104 suite of codes) has been used extensively[11-15] in support of the on-going NUREG-1150 study. The current NRC analytical approach to source term assessment and its experimental basis, its associated review process, and related research programs are discussed in Reference [3].

2.4 Draft Reactor Risk Reference Document (NUREG-1150)[10]

The draft Reactor Risk Reference Document (NUREG-1150) is a major effort by the NRC to put into a risk perspective the insights that have been generated as a result of recent research into systems behavior and physical phenomena under severe accident conditions. Many key elements of the reassessment of the severe accident risk and risk reduction for five reference plants are now completed and the results have been published in a draft form.

One of the major activities of this study was the development of fission product source terms for a spectrum of accident conditions. The base case source term calculation for the accident sequences most important to risk were performed by Battelle Columbus Laboratories (BCL), using the STCP.[11-15] Radiological source terms for accident scenarios of secondary significance were derived from these limited STCP results. The uncertainty analyses involved the solicitation of input from experts that were used to augment the STCP results to reflect the significant phenomenological uncertainties related to severe accident source term predictions. The process included adjustment of results to account for phenomena believed to be important but not included in current STCP models.

2.5 IDCOR Source Term Studies

The Industry Degraded Core Rulemaking (IDCOR) Program[24], which operated under the auspices of the Atomic Industrial Forum, is another major undertaking. Within this program, an integrated computer code called MAAP, along with several other computer programs, were developed to predict the release of radionuclides to the environment and to evaluate the effects of potential plant modifications.

IDCOR studies predict in-vessel releases of the more volatile fission products similar to STCP analyses with the exception of Te (IDCOR predicts less release for Te). However, in the IDCOR analyses, Rb, Zr, Pu, La, Ba, Y, Tc, Rh and Pd are omitted from the in-vessel phase of release. In the STCP modeling, it is assumed that the fission products retained in the primary system at the point of vessel failure are retained permanently, while in the

IDCOR analyses a revaporization of these fission products after vessel failure has been included.

Due to modeling differences for fission product release as a result of core/concrete interactions, IDCOR analysis predicts substantially less ex-vessel fission product release.

2.6 Other Source Term Studies

A Special American Nuclear Society (ANS) Committee on Source Term[48] was given a charter to comment on the magnitude of the source term itself. Using not only the NRC's codes but all available methods, the ANS committee concluded that source terms from severe accidents to nuclear plants would be lower than WASH-1400 values by large factors, at least one order of magnitude and possibly several. While many subsidiary conclusions were drawn, the principal reason for the lower source term estimates was that if containment failed, failure would not occur until many hours after core melt began.

There are two foreign studies related to severe accidents and source terms which have been published. These include a German risk study[25] published in 1979 and a British source term study[26] published in 1982 in connection with the planned construction of Sizewell-B pressurized water reactor.

The German risk study used probability oriented methods for a safety evaluation. Objectives of this study were to determine collective risk of the plants in Germany and to compare results with the United States RSS. The study was based on a single representative operating PWR in Germany. Throughout the study assumptions that were believed to lead to higher estimated consequence were made and the estimated source term release for a release category was taken as the most unfavorable for those events included in the category. The German risk study concluded that despite differences in plant design and site characteristics, the results "more or less" confirmed those of the Reactor Safety Study.

Source term estimates were made for Sizewell-B reactor, a PWR with a large dry containment under construction for a site located in England. These estimates were done under the auspices of the Westinghouse Electric Corporation and the United Kingdom Atomic Energy Authority. The study was a scoping study showing the procedure and its application. The procedure consisted of assigning probabilities to arbitrary adjusted release fractions to account for changes in fission product transport effects believed to be missing or simplistically accounted for in a base case, which was based on the Reactor Safety Study analytical procedure.

An intercomparison between various source term studies has been performed by the Nuclear Energy Agency (NEA) Committee on the Safety of Nuclear Installations (CSNI) and the results were published in 1986.[27]

3. DEVELOPMENT AND APPLICATION OF THE SOURCE TERM CODE PACKAGE (STCP)

The Source Term Code Package (STCP)[6] is an integrated set of computer codes developed within the past decade which more mechanistically simulate severe accident progression and which is believed to provide more realistic estimates of severe accident source term than previous studies, such as the Reactor Safety Study.[8] 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.[2]

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 have been made in the process of integrating these codes. Many of the changes merely simplify the use of the codes and reduce the potential for input errors during data transfer by automating the data transfer between some of the codes. The other changes, however, involve actual improvements in the models or in the coupling between models.

The original BMI-2104 code suite consisted of:

MARCH2: Was the major code in the system and drove all of the other codes. MARCH2 consisted of many individual subroutines. The BOIL subroutine was one of the largest and most important and it simulated the overall thermal-hydraulics of the reactor plant, including the rate of water removal from the reactor coolant system (RCS), core heatup and meltdown, the time of vessel breach, and the temperature and composition of the molten materials released from the vessel. The INTER subroutine modeled core/concrete interactions after vessel breach. Note that CORCON Mod1 was used to model core/concrete interactions for the purpose of calculating fission product release in VANESA. This inconsistency has been eliminated in the STCP. The MACE subroutine calculated the containment thermodynamic behavior following an accident condition.

CORSOR: Used the temperatures calculated by BOIL to calculate the fission product release rates from the fuel within the RCS.

MERGE: Used the gas flow rates and temperatures calculated by BOIL to calculate the gas and surface temperatures along the flow path out of the RCS.

TRAPMELT: Used the temperatures calculated by MERGE, along with the fission product release rates obtained from CORSOR to determine the amount of fission product retention in the RCS and the fission product release to containment.

CORCON: The mass, composition and temperature of the core debris released from the vessel as determined by MARCH were used by CORCON Mod1 to calculate temperature and gas flow rates through the melt during core/concrete interactions. It also calculated the corium temperature and rate of attack on the concrete surfaces. CORCON Mod1 provided input to VANESA, however, INTER provided input to the MACE subroutine to calculate the containment thermal-hydraulics. As noted above, this has been corrected in the STCP.

VANESA: The time-dependent melt temperatures and gas flow rates determined by CORCON were used to calculate the aerosol release rate during core/concrete interactions in this code.

NAUA-4: Uses the aerosol release rates from TRAPMELT and VANESA, and steam condensation rates from MARCH (MACE subroutine) to calculate the agglomeration and settling of the aerosol within the containment building.

SPARC: Calculated the scrubbing of fission products in the suppression pools of Boiling Water Reactors (BWR) during severe accidents.

ICEDF: Calculated fission product and aerosol attenuation in the ice chests of Pressurized Water Reactor (PWR) with Ice Condenser containments.

Figure 3.1 illustrates the manner in which the codes have been grouped in the STCP. The MARCH2, CORSOR-M, and CORCON-Mod2 codes (which replaced the INTER subroutine) were coupled into MARCH3. The CORSOR-M version of the CORSOR code, which uses an Arrhenius form for the empirical fission product release correlation, has replaced CORSOR. CORCON-Mod2 replaced INTER and CORCON Mod1, predicts the thermal-hydraulic loads on containment due to core/concrete interactions, and provides input to the VANESA code to calculate the fission product release. Potentially significant changes also resulted from the intimate coupling of the MERGE and TRAPMELT (TRAPMELT3) codes in the code package.

Table 3.1 identifies the radionuclide groups used in the STCP. These groups are an expansion of the original WASH-1400 and BMI-2014 group structures.

As part of the NRC's Severe Accident Risk Reduction Program (SARRP), Source Term Code Package calculations have been performed for a number of accident sequences which were selected to be the potential contributors to risk at the Surry, Zion, Sequoyah, Peach Bottom, and Grand Gulf plants. In addition, STCP calculations have also been performed in support of the present effort in the development of a generic source term methodology. The conditions encountered in the analyzed sequences span wide ranges of the governing phenomena such as high versus low pressures in the reactor coolant system, rapid heating of the core versus quenching of the core during heatup, slow overpressurization of the containment (and hence delayed failure) versus early failure or bypass. The sequences for which STCP calculations have been performed are summarized in Tables 3.2 through 3.7. Further information may be found in References 11 through 18.

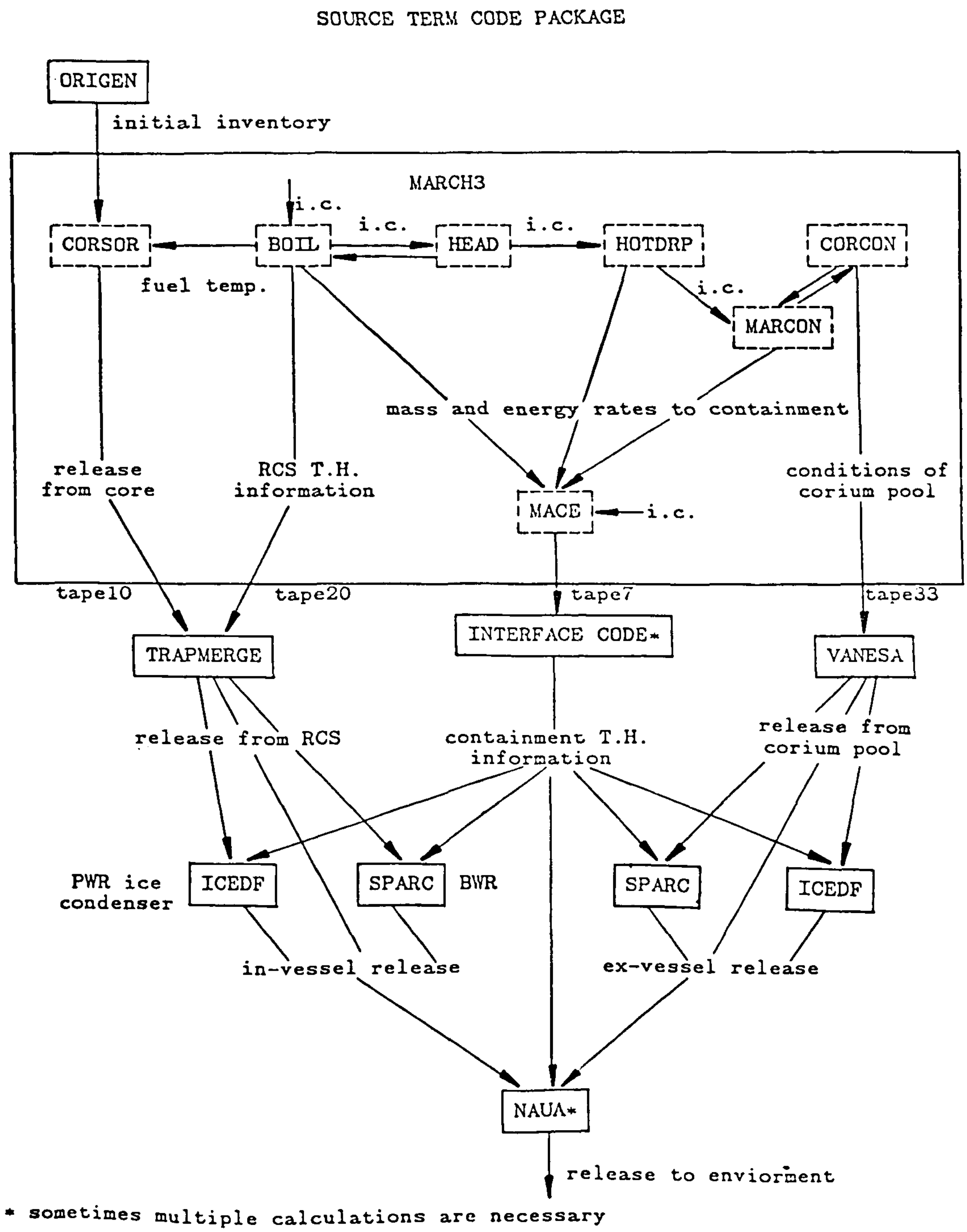

Figure 3.1 Source Term Code Package (STCP)

Table 3.1 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

Table 3.2 Large Dry PWR (Zion) STCP Calculated Accident Sequences[15]

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 overpressure 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 3.3 Subatmospheric PWR (Surry) STCP Calculated Accident Sequences[13,16]

SEQUENCE	DESCRIPTION
AG	A large hot leg break LOCA accompanied by failure of containment heat removal system; the emergency core cooling 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.

Table 3.4 Ice Condenser PWR (Sequoyah) STCP Calculated Accident Sequences[12,16,17]

SEQUENCE	DESCRIPTION
S3HF1	A very small pump seal LOCA with emergency core cooling and containment spray recirculation failure. In this sequence the bottom of reactor vessel is submerged in the reactor cavity water at the time of vessel failure.
S3HF2	A variation of S3HF1 in which a large hot-leg LOCA is induced by the high temperature during core degradation.
S3HF3	Another variation of S3HF1 when the reactor cavity was not permitted to fill with water.
S3B(TB)	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 break due to a burn in 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.

Table 3.5 BWR Mark I (Peach Bottom) STCP Calculated Accident Sequences[11,16-18]

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 wet well 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 later, 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 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.

Table 3.6 BWR Mark II (Lasalle) STCP Calculated Accident Sequence[30]

SEQUENCE	DESCRIPTION
TB	A station blackout accident with late containment failure mode.

Table 3.7 BWR Mark III (Grand Gulf) STCP Calculated Accident Sequences[14]

SEQUENCE	DESCRIPTION
TC	An anticipated transient without scram. The containment was assumed to fail by overpressurization 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 later, 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.

4. APPROACH TO DEVELOPMENT OF SIMPLIFIED SOURCE TERMS

The basis for development of the simplified source terms is the estimation for each unique combination of Reactor Coolant System (RCS) and containment conditions following core damage of the amount of each radionuclide species that is released to the containment.

The release of fission products to containment arises from contributions from the in-vessel and ex-vessel phases of an accident. During the in-vessel phase, fission products are released from the fuel rods to the reactor coolant system. These materials are then transported through the primary system to the containment. During transport, the initial releases are subject to natural deposition processes which can result in a substantial retention in the primary system. In the ex-vessel phase, fission products are released from the molten core/concrete interaction. The ex-vessel release from the melt can be attenuated by the water pool (if any) overlying the corium during core/concrete interaction.

In this chapter, the phenomenological aspects of core damage accidents will be discussed and the key factors affecting fission product release into containment will be identified. Finally, a simplified source term formalism will be proposed.

4.1 Phenomenological Aspects of Core Damage Accidents

Release of fission products from the core materials could occur as a result of many types of accidents. The most limiting accidents appearing in safety analysis reports are those postulated as design basis accidents because they establish criteria for the design and the performance of the engineered safety features. At some level of probability, combinations of system faults can lead to accidents that are more severe than accidents which fall within the current design basis envelope. These beyond design basis accidents are not usually analyzed in safety analysis reports. However, they are included in risk assessment studies.

In order to provide a framework for developing a simplified formalism for fission product release characteristics into containment, the phenomenological aspects of fission product release from the design basis accidents and the severe core damage accidents are discussed in the following sections.

4.1.1 Design Basis Accidents

By definition, Design Basis Accidents (DBAs) are to be terminated successfully through the action of the Engineered Safety Features (ESFs). In particular, 10CFR50.46[28] contains the acceptance criteria for Emergency Core Cooling (ECC) systems for LWRs, which in paragraph (b) (1) specifies that the maximum fuel element cladding temperature shall not exceed 2200°F (1200°C). Although the primary concern that established this particular criterion was the limitation of the extent of fuel cladding oxidation and the extent of hydrogen production, this temperature limit provides a substantial margin with respect to the melting point of the zircaloy cladding (~1800°C). Thus, a DBA should result only in minor damage to the fuel rods. However, other phenomena including complex fuel-cladding interactions and clad bursting may lead to a lower effective failure temperature than the clad melting point. Examination of highly irradiated fuel pins has also provided evidence that failure of the fuel cladding due to embrittlement could occur at a much lower temperature, 900°C.[9] Hence, current state-of-knowledge of severe accident progression suggests cladding failure may occur at or below the 1200°C peak cladding temperature limit set forth in 10CFR50.46 and far below the melting point of the cladding. On the other hand, the accident at TMI demonstrated that substantial core melting can occur without breaching 100% of the fuel pins.

Concurrent with the failure of cladding of some fraction of the fuel pins, a prompt release to the RCS of a small amount of radioactive material that resides in the gap between the fuel pellets and the cladding will occur. Then a mainly time, temperature-dependent release, known as the transient release, can also contribute to the release of fission products.

4.1.2 Severe Core Damage Accidents

These types of accidents are defined as those in which substantial or complete core meltdown occurs. 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 RCS feedwater system and in the failure of the emergency core cooling system to function properly. In such an event, the water loss would gradually 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 Three Mile Island accident. If the cooling water is not supplied in time, complete core debris and meltdown and subsequent vessel breach would result.

During a severe core damage accident, fission products are released from the core materials into the reactor pressure vessel and then are transported through the primary system to the containment. In addition, in accidents in which the core melts through the reactor pressure vessel, fission products are released from molten core debris and concrete interactions into the containment.

4.1.2.1 In-Vessel Release

The fission product release characteristics to the containment during the in-vessel phase of a severe accident are controlled by several mechanisms. The relative importance of each of these mechanisms in determining the overall releases would depend on the Reactor Coolant System (RCS) conditions during the course of the accident and the radionuclides being considered.

During a severe core damage accident, loss of coolant from the reactor system would result in uncovery and subsequent overheating of the core. As the core uncovers and heats up, the steam would begin to react chemically with the zirconium cladding of the uncovered fuel elements, to produce both hydrogen and heat. The heat released by this zirconium oxidation reaction would be on the order of decay heat of the reactor, if sufficient steam were available for the reaction.

As the core heats to a higher temperature, the zirconium cladding would begin to weaken, balloon and rupture. Upon rupture of the cladding, a prompt release of a small amount of the mostly volatile fission products (burst release) will occur. This release forms a relatively small component of the total fission product release for any severe accident.

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

As the temperature increases further, the fuel would begin to distort and melt. During the melting, some of the more volatile components could be evaporated from the various liquid surfaces. This release includes both structural and core materials. The details of the melting process are complex. Chemical transformations occur simultaneously with the melting process which alters the melting points of key materials. For example, while Zircaloy melts at ~1800°C, oxidation to ZrO_2 occurs at the same time during many accidents producing a material with a melting point of ~2700°C. Therefore, whether clad melting or clad oxidation occurs more quickly in an accident sequence can affect the subsequent course of events. Also the presence of molten cladding has a prominent effect on the pellet material as it approaches its melting point. At elevated temperatures, Zirconium diffuses into the pellet, reducing a part of the UO_2 to form a metallic phase with a much lower melting point. Therefore, fission products whose chemical affinities would cause them to move into the metallic phase would evolve more rapidly during melting than those that are retained in the higher melting UO_2 phase. More detailed discussions of physical and chemical processes occurring during melting are described in Reference 9.

As the accident progresses further, the melting fuel elements could eventually destroy and/or bypass the reactor's core support structure and some or all of the melted and unsupported solid materials would fall ("slump") or otherwise move into the lower head of the Reactor Pressure Vessel (RPV). As the hot material falls into any remaining water in the lower head, the steam generation process would be accelerated. The continued evolution of decay heat from molten and solid core materials would evaporate the remaining water in the RPV and the combined molten fuel material and structural materials (corium) would then attack the lower head of the RPV.

Due to the large temperature gradient within the core, the cladding rupture and subsequent melting of the core would occur on a region by region basis. Thus, the total release of any given fission product or other material would occur over a period of time. However, in general, the more volatile radionuclides would tend to be released in the early heatup and melting. On the other hand, the less volatile fission products would tend to be released toward the end of melting. In a complete meltdown, these releases would 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 transported through the various portions of the Reactor Coolant System (RCS) to the containment. The dominant pathways which the fission products follow through 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 rapidly condense into aerosol particles. The released fission product gases could absorb or condense onto particulates and reactor coolant system (RCS) surfaces, react chemically with other species in the RCS atmosphere or with RCS surfaces, or dissolve in and/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. Any of the removed material could subsequently be resuspended, revaporized, and/or otherwise entrained in the RCS fluids and subsequently transported out of the RCS. A detailed review of the major natural processes in the RCS and their effects are discussed in Reference [29].

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

Accidents in which the ECC system partially operates or has delayed operation could result in extensive core damage without progressing to full core meltdown. If an accident sequence does not progress to full core meltdown, it is an implicit requirement that cooling water must have been available in the latter stages of the accident to arrest core damage. The presence of water in the dominant pathway(s) through the RCS would tend to increase the extent of retention in the RCS.

4.1.2.2 Ex-Vessel Release

Only a partial release of the fission products occurs within 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. The molten material (including part of the support structure and pressure vessel) and all of the remaining radioactive materials would be transferred to the containment. Whether this would occur slowly or rapidly as materials are injected into the containment would depend on the previous history of the core melt scenario and, in particular, on the pressure in the RCS at the time of vessel breach.

If the primary system is pressurized at the time of vessel breach, corium will be ejected under pressure in a process which has been demonstrated experimentally to result in significant aerosol generation. This purely physical process is expected to occur more or less independently of whether or not the dispersed core debris causes 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 core debris.

The molten core debris (corium) eventually falls into a concrete cavity below the RPV. The molten corium would attack the concrete and liberate copious amounts of concrete decomposition gas products. As this occurred, aerosols would be sparged from the molten mass into the containment atmosphere.

Due to both loss of decay heat and incorporation of concrete residue, the molten mass eventually would cool. Although complete cooling may take a long time, it would only take several hours at most for sufficient cooling that further fission product release would be negligible.

Among the factors that influence the magnitude and timing of the ex-vessel release are composition and temperature of the corium as it is released from the vessel. The composition of concrete also has a major impact on the amount of aerosols carried into the containment atmosphere.

The water pool (if any) overlying the corium during core/concrete interaction would retain some of the releases from core/concrete interaction.

4.2 Key Factors Affecting Fission Product Release Into Containment

As discussed in previous sections, many factors influence the magnitude and timing of the fission product releases into the containment. In this section the dominant factors which determine the fission product release characteristics into containment are discussed.

4.2.1 Reactor Type

Boiling Water Reactors (BWRs) generally have larger amounts of Zircaloy in the core, lower core power density and lower operating pressure compared to Pressurized Water Reactors (PWRs). These factors influence the fission product release characteristics into the containment during postulated severe accidents.

Due to larger amounts of Zircaloy in the BWR core, the amount of zirconium would be larger in the corium as it is released from the vessel. The exothermic oxidation of zirconium ex-vessel provides a continuing heat source that adds to the decay heat and also produces more carbon, from the reduction of CO_2, a concrete decomposition gas. While zirconium oxidation is taking place, CO_2 which enters the melt, is converted to carbon with a resultant decrease in the amount of gas which sparges through the melt. After zirconium oxidation is completed, the carbon can be oxidized by either H_2O or CO_2. For each mole of H_2O or CO_2 which enters the melt, two moles of gas eventually sparge the melt. The increased amount of zirconium in BWRs enhances these zirconium related increases in the gas sparging rate.

Lower core power density delays the time for complete core melting for BWR accident sequences. Therefore, the time of in-vessel release is generally longer for BWR accident sequences.

4.2.2 Reactor Coolant System (RCS) Pressure Before Reactor Pressure Vessel (RPV) Failure

Accident sequences with low RCS pressure before RPV failure are characterized by having relatively large steam velocities and, therefore, short radionuclide residence times in the RCS before releasing to containment. The retention of aerosols in the RCS would, therefore, be small for such accident sequences. In contrast, the accident sequences with high RCS pressure typically would have low steam velocities and consequently long radionuclide residence times in RCS. For such accidents due to higher gravitational settling, the dominant mechanism for aerosol deposition in the RCS, the fission products released from the fuel (with the exception of noble gases) are retained in the RCS with higher efficiency.

4.2.3 Containment Concrete Aggregate/Composition

Limestone concrete produces larger gas flows and are more oxidizing compared to basaltic concrete. This has a major impact on the amount of aerosols carried into the containment atmosphere by the sparging of the concrete decomposition gases through the molten corium and in the generation of heat by zironium oxidation.

4.2.4 Cavity/Pedestal Condition

In some accident sequences, water will be in the reactor cavity while the corium melt attacks concrete. The water pool overlying the core debris can trap aerosol particles evolved by the core debris. The depth of water pool (if any) overlying the corium during core/concrete interaction is an important factor which influences the aerosol scrubbing effect and, therefore, the magnitude of ex-vessel release to containment. (An issue which could have a major effect on the source term to containment is the possibility that steam explosion(s) could fragment the core into a coolable configuration and mitigate the core/concrete interaction.[42] This issue has not been explicitly treated in the proposed formulation, but will be addressed as an uncertainty issue in Chapter 7.)

4.3 Proposed Simplified Formulation for Source Term to Containment

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

$$F_i = FRCS_i + FCCI_i \qquad (4.1)$$

where:

F_i = fraction of the initial core inventory of species i that is released into the containment,

$FRCS_i$ = fraction of the initial core inventory of species i released from the reactor coolant system into the containment, that is:

$$FRCS_i = FCOR_i * FVES_i \tag{4.2}$$

$FCOR_i$ = fraction of initial core inventory of species i released from the fuel prior to reactor pressure vessel failure,

$FVES_i$ = fraction of material of species i that is released from the fuel which is released from the vessel, and

$FCCI_i$ = fraction of initial core inventory that is released from the corium melt during core/concrete interactions (CCI).

In using Equation 4.1, appropriate decontamination factors (DFs) are applied to account for fission product retention inside BWR suppression pools and/or the overlying water pool for a flooded cavity/pedestal region during the core/concrete interactions.

Table 4.1 shows the grouping of the radionuclide releases which are adapted for the present treatment. The five groups provide an adequate simplification to the original nine group representation of the Source Term Code Package.[6]

The calculation of simplified source terms using Equation 4.1 contains two underlying assumptions. First, the fission product species are grouped according to their respective chemical forms and release characteristics as defined in Table 4.1. Secondly, the accident conditions will be categorized into appropriate categories somewhat similar to the approach utilized for the risk rebaselining studies[10].

The categorization of radionuclide releases will follow four key characteristics, namely;

1) Reactor type (BWR versus PWR),

2) RCS pressure prior to reactor pressure vessel breach (high, intermediate or low pressure),

3) Concrete aggregate/composition (limestone versus basaltic), and

4) Cavity/pedestal condition (dry versus flooded).

The parameters entering the model will be derived from the existing data base, consisting mostly of the STCP calculated source terms.[11-18] These calculations will be assessed in the next chapter to arrive at the source term parameters.

Table 4.1 Simplified Source Terms Radionuclide Groups

Group	Key Radionuclides	Definition
1	Xe, Kr	Noble gases
2	I, Cs	Halogens and alkali metals, also includes Br and Rb
3	Te	Tellurium, also includes Se and Sb
4	Sr, Ba	Strontium and barium
5	Ru, La, Ce	Noble metals, lanthanum and cerium groups, also includes Mo, Pd, Rh, Tc, Zr, Nd, Eu, Y, Pr, Pm, Nb, Sm, Np and Pu

5. ANALYSIS OF STCP SEQUENCES

Extensive STCP calculations have been performed in support of draft NUREG-1150, the effort by Battelle Columbus Division and Brookhaven National Laboratory.

In order to provide a framework for determining the parameters governing the proposed simplified formulation for source term to containment, the results of these calculations are presented in this section.

5.1 Nuclide Groups and Quantities

The STCP (CORSOR-M Code)[3] calculation of in-vessel release of fission products from 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 depend only on temperature. Tellurium, which reacts readily with unoxidized zircaloy of cladding, has been given an additional dependency to account for the ability of unoxidized zircaloy to retain tellurium.

The STCP results for fraction of initial core inventory released to 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 operation of the core heat removal system (e.g., HPC injection with steam driven pump in Sequoyah S3HF and S3B sequences) adds an additional source of steam for zirconium oxidation. The STCP results for the fraction of Zircaloy cladding reacted prior to RPV failure for different sequences and plants are tabulated in Tables 5.5 and 5.6.

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 chemisorbted by the surfaces[3]. 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,[49] metal surfaces[50], water pools[51], and the steam-hydrogen atmosphere during a hydrogen burn[52]. Experimental evidence of the release of other forms of Te, e.g. CsTe, has been published[53]. Generally, the treatment of fission product chemistry in the STCP is primative when it exists, and there remains a very high degree of uncertainty in the chemical forms of released fission products.

TABLE 5.1 STCP Results for Fraction of Initial Core Inventory Released to Vessel Prior to RPV Failure (FCOR) for PWR High and Intermediate Pressure Sequences.

SPECIES	SURRY[16]	ZION[15]			SEQUOYAH[12,16]		
	TMLB'*	TMLU	S2DCR	S2DCF	S3HF	S3B	TMLB'
NG	0.98	1.0	0.99	0.99	0.97	0.97	0.97
I	0.98	1.0	0.99	0.99	0.97	0.97	0.97
Cs	0.98	1.0	0.99	0.99	0.97	0.97	0.97
Te	0.46	0.54	0.43	0.43	0.84	0.84	0.36
Sr	$7x10^{-4}$	$2x10^{-3}$	$4x10^{-4}$	$4x10^{-4}$	$6x10^{-4}$	$6x10^{-4}$	$5x10^{-4}$
Ba	0.013	0.02	$8x10^{-3}$	$8x10^{-3}$	0.01	0.01	0.01
Ru	10^{-6}	$2x10^{-6}$	$5x10^{-7}$	$5x10^{-7}$	10^{-6}	10^{-6}	10^{-6}
Ce	0	0	0	0	0	0	0
La	10^{-7}	$2x10^{-7}$	$0.5x10^{-7}$	$0.5x10^{-7}$	10^{-7}	10^{-7}	10^{-7}

*See Chapter 3 for definition of accident sequences

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

SPECIES	SURRY[13,16]		SEQUOYAH[12,17]	
	V	AG	TBA	ACD
NG	1.0	1.0	1.0	1.0
I	1.0	1.0	0.98	1.0
Cs	1.0	1.0	0.98	1.0
Te	0.63	0.86	0.8	0.51
Sr	1.5×10^{-3}	1×10^{-3}	2×10^{-3}	0.6×10^{-3}
Ba	3×10^{-2}	2×10^{-2}	4×10^{-2}	1×10^{-2}
Ru	2.6×10^{-6}	1.6×10^{-6}	3×10^{-6}	1×10^{-6}
Ce	0	0	0	0
La	2×10^{-7}	1.6×10^{-7}	3×10^{-7}	0.8×10^{-7}

TABLE 5.3 STCP Results for Fraction of Initial Core Inventory Released to Vessel Prior to RPV Failure (FCOR) for BWR High and Intermediate Pressure Sequences.

SPECIES	PEACH BOTTOM[11,16,17,18]					LaSalle[30]	GRAND GULF[14]
	TC2		TC3/TC2	TB1/TB2	S2E	TB	TB
	MOD0	MOD1[a]					
NG	0.96	0.87	0.88	0.88	0.9	0.99	0.96
I	0.96	0.87	0.96	0.88	0.84	0.99	0.98
Cs	0.96	0.87	0.96	0.87	0.84	0.99	0.97
Te	0.67	0.62	0.69	0.38	0.19	0.43	0.38
Sr	$6.1x10^{-4}$	$5.3x10^{-4}$	$6.2x10^{-4}$	$1.6x10^{-3}$	$4.x10^{-4}$	$9.8x10^{-4}$	$1.1x10^{-3}$
Ba	$1.1x10^{-2}$	$9.4x10^{-3}$	$1.1x10^{-2}$	$2.2x10^{-2}$	$7.4x10^{-3}$	$1.8x10^{-2}$	$2.1x10^{-2}$
Ru	10^{-6}	10^{-6}	10^{-6}	$1.6x10^{-6}$	$0.7x10^{-6}$	$1.8x10^{-6}$	$1.5x10^{-6}$
Ce	0	0	0	0	0	0	0
La	10^{-7}	10^{-7}	10^{-7}	$1.6x10^{-7}$	$0.5x10^{-7}$	$1.5x10^{-7}$	$1.5x10^{-7}$

(a) MOD0 and MOD1 refer to an interim version of the STCP and the latest version of the STCP, respectively.

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

SPECIES	PEACH BOTTOM[11]		GRAND GULF[14]	
	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	5.9×10^{-4}	2.7×10^{-4}	1.1×10^{-3}	4.6×10^{-4}
Ba	1.1×10^{-2}	5.1×10^{-3}	2.1×10^{-2}	8.8×10^{-3}
Ru	8×10^{-7}	4×10^{-7}	10^{-6}	6×10^{-7}
Ce	0	0	0	0
La	9×10^{-7}	6×10^{-8}	2×10^{-7}	7×10^{-8}

Table 5.5 STCP Results for Fraction of Zircaloy Cladding Reacted Prior to RPV Failure for PWR Accident Sequences

Plant	Accident Sequence	Fraction of Clad Reacted	Remarks
Surry	TMLB'	0.52	
	V	0.4	
	AG	0.72	
Zion	TMLU	0.52	
	S2DCR	0.48	
	S2DCF	0.48	
Sequoyah	TMLB	0.5	
	TBA	0.7	
	ACD	0.48	
	S3HF	0.75	HPC Injection
	S3B	0.75	HPC Injection

Table 5.6 STCP Results for Fraction of Zircaloy Cladding Reacted Prior to RPV Failure for BWR Accident Sequences

Plant	Accident Sequence	Fraction of Clad Reacted	Remarks
Peach Bottom	TC2/TC3	0.6	CRD Flow
	TC1	0.25	
	TB1/TB2	0.25	
	V	0.32	
	S2E	0.38	
LaSalle	TB	0.25	
Grand Gulf	TB	0.32	
	TC1	0.3	
	TBS/TBR	0.4	

In the STCP modeling of fission product retention in the reactor coolant system (TRAPMELT code)[32], 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 Elrik, Sallach, and others[33,34]. The CsI is assumed to condense only on metal surfaces whereas the metal-Te reaction is assumed to be much more reactive (chemical absorption is modeled) than the CsI.

The STCP treatment of aerosol behavior within the reactor coolant system (TRAPMELT 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).

A convenient way to describe the overall effect of retention phenomena in the reactor coolant system is to state the fraction of materials that is released from the fuel which is released from the vessel (FVES). The STCP results for FVES values are shown in Tables 5.7 through 5.10. A comparison of these values indicate different retention for different fission product groups as a function of the RCS pressure.

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.

Since the FCOR and FVES are correlated in a phenomenological sense, it is more reasonable to present the results in terms of FRCS (FRCS = FCOR * FVES). The STCP results for fraction of initial core inventory released from the vessel to the containment (FRCS) for different sequences and plant types are tabulated in Tables 5.11 through 5.14.

STCP calculations have been performed only for accident sequences in which the complete core meltdown and subsequent vessel breach would result. However, it is recognized that operator/system actions to stem core degradation by injecting coolant into/onto an uncovered core may occur at any time during the sequence. If the cooling water is supplied in time, termination of the accident and long term cooling of the core is possible, as occurred during the Three Mile Island accident. STCP (MARCH3) calculations were performed for two seqeunces at Zion and Peach Bottom where Emergency Core Cooling (ECC) was assumed to be failed throughout the accident or be recovered just before core slump. The core damage was arrested only for the case of ECC recovery one

TABLE 5.7 STCP Results for Fraction of Material That is Released From the Fuel Which is Released From the Vessel Including the Puff Release (FVES) for PWR High and Intermediate Pressure Sequences.

SPECIES	SURRY[16]	ZION[15]			SEQUOYAH[12,16]		
	TMLB'	TMLU	S2DCR	S2DCF	S3HF	S3B	TMLB'
NG	1.0	1.0	1.0	1.0	1.0	1.0	1.0
I	0.22	0.22	0.28	0.28	0.31	0.32	0.36
Cs	0.21	0.19	0.28	0.28	0.24	0.25	0.32
Te	0.62	0.47	0.47	0.47	0.09	0.09	0.46
Sr	0.26	0.16	0.34	0.34	0.23	0.25	0.24
Ba	0.26	0.26	0.37	0.37	0.23	0.25	0.22
Ru	0.26	0.22	0.31	0.31	0.24	0.25	0.22
Ce	0	0	0	0	0	0	0
La	0.3	0.21	0.27	0.27	0.23	0.31	0.24

TABLE 5.8 STCP Results for Fraction of Material That is Released From the Fuel Which is Released From the Vessel Including the Puff Release (FVES) for PWR Low Pressure Sequences.

SPECIES	SURRY[13,16]		SEQUOYAH[12,17]	
	V	AG	TBA	ACD
NG	1.0	1.0	1.0	1.0
I	0.62	0.87	0.94	0.92
Cs	0.60	0.87	0.92	0.92
Te	0.25	0.83	0.84	0.78
Sr	0.35	0.75	0.87	0.83
Ba	0.35	0.78	0.88	0.77
Ru	0.35	0.76	0.89	0.83
Ce	0	0	0	0
La	0.35	0.78	0.86	0.87

TABLE 5.9 STCP Results for Fraction of Material That is Released From the Fuel Which is Released From the Vessel (FVES) for BWR High and Intermediate Pressure Sequences.

SPECIES	PEACH BOTTOM[11,16,17,18]					LaSalle[30]	GRAND GULF[14]
	TC2		TC3/TC2	TB1/TB2	S2E	TB	TB
	MOD0	MOD1[a]					
NG	1	1	1	0.97	1	1	1
I	0.96	0.9	0.96	0.23	0.69	0.37	0.7
Cs	0.83	0.8	0.82	0.14	0.65	0.27	0.54
Te	0.25	0.15	0.29	0.07	0.18	0.13	0.13
Sr	0.75	0.62	0.61	0.09	0.7	0.23	0.31
Ba	0.76	0.6	0.64	0.13	0.7	0.24	0.3
Ru	0.78	0.66	0.65	0.14	0.74	0.23	0.31
Ce	0	0	0	0	0	0	0
La	0.72	0.78	0.63	0.15	0.86	0.25	0.31

a MOD0 and MOD1 refer to an interim version of the STCP and the latest version of the STCP, respectively.

TABLE 5.10 STCP Results for Fraction of Material That is Released From the Fuel Which is Released From the Vessel Including Puff Release (FVES) for BWR Low Pressure Sequences.

SPECIES	PEACH BOTTOM[11]		GRAND GULF[14]	
	TC1	V	TC1	TBS/TBR
NG	0.98	0.99	1.0	1.0
I	0.81	0.89	0.54	0.86
Cs	0.81	0.87	0.56	0.83
Te	0.13	0.5	0.37	0.31
Sr	0.8	0.78	0.35	0.85
Ba	0.8	0.76	0.37	0.84
Ru	0.79	0.76	0.38	0.86
Ce	0	0	0	0
La	0.76	0.53	0.35	0.87

TABLE 5.11 STCP Results for Fraction of Initial Core Inventory Released From the Vessel to the Containment Including Puff Release (FRCS) for PWR High and Intermediate Pressure Sequences.

SPECIES	SURRY[16]	ZION[15]			SEQUOYAH[12,16]		
	TMLB'	TMLU	S2DCR	S2DCF	S3HF	S3B	TMLB'
NG	0.98	1.0	0.99	0.99	0.97	0.97	0.97
I	0.22	0.22	0.27	0.27	0.3	0.31	0.35
Cs	0.21	0.19	0.27	0.27	0.23	0.24	0.31
Te	0.28	0.25	0.2	0.2	0.075	0.075	0.17
Sr	$2x10^{-4}$	$3x10^{-4}$	$1.4x10^{-4}$	$1.4x10^{-4}$	$1.5x10^{-4}$	$2x10^{-4}$	$1.3x10^{-4}$
Ba	$3.3x10^{-3}$	$5x10^{-3}$	$3x10^{-3}$	$3x10^{-3}$	$3x10^{-3}$	$3x10^{-3}$	$2.5x10^{-3}$
Ru	$3x10^{-7}$	$4x10^{-7}$	$2x10^{-7}$	$2x10^{-7}$	$3x10^{-7}$	$3x10^{-7}$	$2x10^{-7}$
Ce	0	0	0	0	0	0	0
La	$4x10^{-8}$	$4x10^{-8}$	10^{-8}	10^{-8}	10^{-8}	10^{-8}	10^{-8}

TABLE 5.12 STCP Results for Fraction of Initial Core Inventory Released From the Vessel to the Containment Including the Puff Release (FRCS) for PWR Low Pressure Sequences.

SPECIES	SURRY[13]		SEQUOYAH[12,17]	
	V	AG	TBA	ACD
NG	1.0	1.0		1.0
I	0.62	0.87	0.92	0.92
Cs	0.6	0.87	0.90	0.92
Te	0.16	0.71	0.67	0.4
Sr	$5x10^{-4}$	$9x10^{-4}$	$2x10^{-3}$	$5x10^{-4}$
Ba	0.01	0.02	0.03	0.01
Ru	10^{-6}	10^{-6}	10^{-6}	10^{-6}
Ce	0	0	0	0
La	10^{-7}	10^{-7}	10^{-7}	10^{-7}

TABLE 5.13 STCP Results for Fraction of Initial Core Inventory Released From the Vessel (FRCS) for BWR High and Intermediate Pressure Sequences.

SPECIES	PEACH BOTTOM[11,16,17,18]					LaSalle[30]	GRAND GULF[14]
	TC2 MOD0	TC2 MOD1	TC3/TC2	TB1/TB2	S2E	TB	TB
NG	0.96	0.87	0.96	0.86	0.82	0.99	0.96
I	0.92	0.78	0.92	0.2	0.58	0.36	0.68
Cs	0.79	0.69	0.79	0.12	0.55	0.26	0.52
Te	0.17	0.095	0.2	0.03	0.035	0.056	0.05
Sr	4.6×10^{-4}	3.3×10^{-4}	3.8×10^{-4}	1.5×10^{-4}	2.8×10^{-4}	2.2×10^{-4}	3.4×10^{-4}
Ba	8.5×10^{-3}	5.6×10^{-3}	7×10^{-3}	2.8×10^{-3}	5.2×10^{-3}	4.3×10^{-3}	6.4×10^{-3}
Ru	7.8×10^{-7}	8×10^{-7}	6.5×10^{-7}	2.3×10^{-7}	5.2×10^{-7}	4.1×10^{-7}	4.7×10^{-7}
Ce	0	0	0	0	0	0	0
La	6.8×10^{-8}	1.1×10^{-7}	6×10^{-8}	2.4×10^{-8}	4.3×10^{-8}	3.7×10^{-8}	4.7×10^{-8}

a MOD0 and MOD1 refer to an interim version of the STCP and the latest version of the STCP, respectively.

TABLE 5.14 STCP Results for Fraction of Initial Core Inventory Released From the Vessel (FRCS) for BWR Low Pressure Sequences.

SPECIES	PEACH BOTTOM[11]		GRAND GULF[14]	
	TC1	V	TC1	TBS/TBR
NG	0.9	0.82	1.0	0.94
I	0.75	0.74	0.54	0.83
Cs	0.74	0.71	0.56	0.81
Te	0.04	0.06	0.15	0.12
Sr	$5x10^{-4}$	$2x10^{-4}$	$4x10^{-4}$	$4x10^{-4}$
Ba	$9x10^{-3}$	$4x10^{-3}$	$8x10^{-3}$	$7x10^{-3}$
Ru	$7x10^{-7}$	$3x10^{-7}$	$5x10^{-7}$	$5x10^{-7}$
Ce	0	0	0	0
La	$7x10^{-8}$	$3x10^{-8}$	$5x10^{-8}$	$6x10^{-8}$

Table 5.15 Effect of ECC Recovery on the STCP Results for Fractions of Initial Core Inventory Released to Vessel (FCOR) for High RCS Pressure Accident Sequence in a PWR (Zion)

Species	No ECC	ECC Recovery 1 Min. Before Core Slump (Terminated Accident)
NG	1	0.93
I	1	0.93
Cs	1	0.93
Te	0.56	0.4
Sr	$8x10^{-4}$	$8x10^{-4}$
Ba	10^{-2}	10^{-2}
Ru	10^{-6}	10^{-6}
Ce	0	0
La	10^{-7}	10^{-7}

Table 5.16 Effect of ECC Recovery on the STCP Results for Fractions of Initial Core Inventory Released to Vessel (FCOR) for a High RCS Pressure Accident Sequence in a BWR

Species	No ECC	ECC Recovery 1 Min. Before Core Slump (Terminated Accident)
NG	0.98	0.86
I	0.98	0.85
Cs	0.98	0.86
Te	0.33	0.3
Sr	$9x10^{-4}$	$8x10^{-4}$
Ba	$2x10^{-2}$	10^{-2}
Ru	10^{-6}	10^{-6}
Ce	0	0
La	10^{-7}	10^{-7}

minute before core slump. Tables 5.15 and 5.16 summarize the resulting fission product releases for the two cases.

Although the STCP (MARCH code) suffers from limitations in modeling some aspects of the progression of terminated accidents (e.g., shattering of the partially molten core due to cooling effects from ECC injection), most of the in-vessel releases would have occurred during early heat-up and melting and, thus, the STCP calculated results for the in-vessel releases for complete meltdown sequences can be considered as the upper bound for releases resulting from terminated accidents. However, as noted earlier, other chemical processes may take place and alter the chemical form of iodine. These processes were not considered in the thermochemical analysis[9] cited above.

ECC injection during terminated accidents also has an impact on total hydrogen generation resulting from the exothermic oxidation of Zr. An assessment of hydrogen generation during a terminated accident requires determination of the rate of cooling afforded by the injected coolant (which lowers the oxidation reaction rate) relative to the availability of steam for zirconium oxidation. The STCP results for effect of ECC recovery on the total in-vessel hydrogen release during station blackout sequences in Peach Bottom and Zion plants are shown in Table 5.17. These results indicate that in a terminating accident sequence the quantity of in-vessel hydrogen generation would be significantly less than the amount of hydrogen generation in a complete meltdown sequence.

The chemical form of iodine released during a severe accident is most likely cesium iodide. The available thermodynamic calculations[9] clearly supports this conclusion. These calculations indicate CsI stability is enhanced by hydrogen (reducing environment) and lower temperature. Though less hydrogen formation is expected in a terminated severe accident, the lower RCS temperatures favor CsI formation. Additionally, the extent of hydrogen production prior to core collapse in both the terminated and complete meltdown variations of the sequences examined is very similar. Since the majority of the Cs and I releases occur during this period, it is believed that the analysis, which predicts CsI as the dominant form of iodine, to be valid also for terminated accident sequences.

Following the reactor pressure vessel failure, the molten core debris (corium) which may contain large amounts of unoxidized metals (Zircaloy cladding, stainless steel from fuel assemblies, and steel from vessel bottom head) as well as oxides such as UO_2 and ZrO_2 falls into a concrete cavity below. In the Source Term Code Package calculation, CORCON-MOD2[35] is used for modeling corium/concrete interaction. CORCON predicts the release of steam and CO_2 from concrete decomposition (ablation). The gases that bubble up through the core debris are modeled to react chemically with the materials of the melt. The CORCON model accounts for the heat of reaction from the chemical processes and for the decay heat from the fission products[3].

In the STCP calculation the fission product release and aerosol production from the core concrete melt is predicted by the VANESA[36] code. The

Table 5.17 Effect of ECC Recovery on Hydrogen Generation During Station Blackout Accident Sequence in Peach Bottom and Zion

Plant	Total Hydrogen Generation (kg)	
	No ECC	ECC Recovery Before Core Slump (Terminated Accident)
Zion	515	386
Peach Bottom	673	363

TABLE 5.18 STCP Results for Fraction of Initial Core Inventory That is Released From the Melt During Core/Concrete Interactions (FCCI) for PWR High and Intermediate Pressure Sequences.

SPECIES	SURRY[16]	ZION[15]				SEQUOYAH[12,16]		
	TMLB'	TMLU	S2DCR	S2DCF1	S2DCF2	S3HF	S3B	TMLB'
NG	0	0	0	0	0	0	0	0
I	0.02	10^{-3}	$4\text{x}10^{-3}$	$4.\text{x}10^{-3}$	$3.5\text{x}10^{-3}$	0.03	0.03	0.03
Cs	0.02	10^{-3}	$4\text{x}10^{-3}$	$4.\text{x}10^{-3}$	$4.\text{x}10^{-3}$	0.03	0.03	0.03
Te	0.12	0.17	0.22	0.4	0.28	0.065	0.078	0.4
Sr	0.17	$1.4\text{x}10^{-4}$	0.32	0.1	0.34	0.17	0.17	0.53
Ba	$9.7\text{x}10^{-2}$	$1.3\text{x}10^{-3}$	0.23	0.07	0.23	0.1	0.10	0.29
Ru	$4\text{x}10^{-6}$	$4\text{x}10^{-6}$	10^{-6}	$8\text{x}10^{-4}$	$3\text{x}10^{-6}$	$2\text{x}10^{-6}$	$4\text{x}10^{-6}$	$3.5\text{x}10^{-3}$
Ce	$7\text{x}10^{-3}$	$5\text{x}10^{-7}$	$7\text{x}10^{-3}$	$2\text{x}10^{-3}$	$8\text{x}10^{-3}$	$6\text{x}10^{-3}$	$7\text{x}10^{-3}$	$2.8\text{x}10^{-2}$
La	$8\text{x}10^{-3}$	$9\text{x}10^{-6}$	$7\text{x}10^{-3}$	$5\text{x}10^{-3}$	$8\text{x}10^{-3}$	$9\text{x}10^{-3}$	$9\text{x}10^{-3}$	$2\text{x}10^{-2}$

TABLE 5.19 STCP Results for Fraction of Initial Core Inventory That is Released From the Melt During Core/Concrete Interaction (FCCI) PWR Low Pressure Sequences.

SPECIES	SURRY[13,16]		SEQUOYAH[12,17]	
	V	AG	TBA	ACD1
NG	0	0	0	0
I	0	1.5×10^{-4}	9.1×10^{-4}	9.2×10^{-4}
Cs	0	1.6×10^{-4}	8.1×10^{-4}	9.2×10^{-4}
Te	0.06	0.017	0.09	0.22
Sr	0.33	0.09	0.17	0.51
Ba	0.16	0.06	0.10	0.27
Ru	2×10^{-6}	4×10^{-9}	4×10^{-6}	8×10^{-7}
Ce	2.5×10^{-2}	1.06×10^{-3}	5.7×10^{-3}	8×10^{-2}
La	2×10^{-2}	4×10^{-3}	1.17×10^{-2}	6×10^{-2}

TABLE 5.20 STCP Results for Fraction of Initial Core Inventory That is Released From the Melt During Core/Concrete Interaction (FCCI) for BWR High and Intermediate Pressure Sequences.

SPECIES	PEACH BOTTOM[11,16,17,18]						LaSalle[30]	GRAND GULF[14]
	TC2 MOD0	TC2 MOD1	TC3/TC2	TB1/TB2	S2E1	S2E2	TB	TB
NG	0	0	0	0	0	0	0	0
I	0.04	0.13	0.04	0.11	0.16	0.16	0.008	0.02
Cs	0.04	0.13	0.04	0.12	0.16	0.16	0.008	0.03
Te	0.19	0.2	0.19	0.39	0.5	0.114	0.3	0.21
Sr	0.61	0.6	0.66	0.84	0.77	0.28	0.68	0.55
Ba	0.43	0.4	0.45	0.6	0.54	0.173	0.48	0.35
Ru	1.4×10^{-6}	1.5×10^{-6}	2.6×10^{-6}	1.4×10^{-6}	1.6×10^{-6}	2.6×10^{-8}	1.6×10^{-7}	8×10^{-7}
Ce	0.03	0.03	0.03	0.09	0.07	8.4×10^{-3}	0.04	0.06
La	0.02	0.02	0.02	0.06	0.03	8×10^{-3}	0.02	0.04

TABLE 5.21 STCP Results for Fraction of Initial Core Inventory That is Released From the Melt During Core/Concrete Interaction (FCCI) for BWR Low Pressure Sequences.

SPECIES	PEACH BOTTOM[11]		GRAND GULF[14]	
	TC1	V	TC1	TBS/TBR
NG	0	0	0	0
I	0.08	0.17	$2.\times10^{-3}$	0.03
Cs	0.08	0.18	$2.\times10^{-3}$	0.03
Te	0.4	0.6	0.11	0.14
Sr	0.75	0.84	0.42	0.42
Ba	0.56	0.6	0.25	0.26
Ru	1.2×10^{-6}	2.4×10^{-6}	3.7×10^{-7}	2.8×10^{-7}
Ce	0.04	0.08	0.05	0.04
La	0.02	0.04	0.03	0.02

TABLE 5.22 STCP Results for Effective Decontamination Factor of the Water Pool Overlying the Corium During Core/Concrete Interaction (DF_{POOL})

SPECIES	SURRY[16]	ZION[15]				SEQUOYAH[12,16]			
	TMLB'	TMLU	S2DCR	S2DCF1	S2DCF2	S3HF1/ S3HF2	S3HF3	S3B	TMLB'
I	2.3	16	13	2.25	19.7	32	5.5	5.7	15
Cs	2	9	14	1.5	17.3	23	4	4	2
Te	1.5	10	13	1.2	4.8	12	1.66	1.9	2.1
Sr	2.4	11	14	2	17	28	5.3	1.23	5.9
Ba	2.9	9.4	14	1.9	17	27	4.8	5	5.8
Ru	3.5	9.1	18	1.4	1.3	5.7	1.3	1.2	--
Ce	3.88	15	14	1.9	16	29	5.2	5.3	2.8
La	3.2	14	13.4	2	11	31	4	5.5	3.2

ex-vessel release is driven by bubbling of reaction gases into the melt. VANESA calculates the release by vaporization of fission products and other melt constituents from the melt into the gas bubbles.

Among the factors that influence the magnitude and timing of the ex-vessel release are the composition and temperature of the corium as it is released from the vessel. The composition of the concrete can also have a major impact on the amount of aerosols carried into the containment atmosphere by the sparging of the concrete decomposition gases through the molten corium. The STCP results for fraction of initial core inventory that is released from the melt during core/concrete interactions (FCCI) are tabulated in Tables 5.18 through 5.21. Peach Bottom, Sequoyah, Zion and Grand Gulf have limestone concrete which produces large gas flows compared to basaltic concrete (Surry). A variation of Peach Bottom S_2E_1 sequence (S_2E_2) with basaltic concrete is also shown in Table 5.20.

The depth of water pool (if any) overlying the corium during core/concrete interaction is another important factor which influences the magnitude of ex-vessel release to containment. The VANESA code accounts for this aerosol scrubbing by gravitational settling, random diffusion, and inertial impaction. Table 5.22 shows STCP/VANESA results for effective decontamination factor of the water pool overlying the corium melt during core/concrete interaction (DF_{pool}).

5.2 Timing of Releases

The fission products release time and duration is an important parameter in specifying the source term to containment.

The time of release of fission products into the containment can vary depending on plant, accident sequence and the specific fission product group. However, a review of the STCP calculated results for six reference plants revealed that the significant in-vessel fission product release into the containment starts at times earlier than about one-half hour after the core melt initiation. As discussed in the next chapter, core melt initiation occurs 6 or more minutes after the start of gap release in all of the plant types and accident sequences studied. Current regulatory requirements are based upon the assumption that the fission product release to occur immediately upon loss of core cooling, and to occur at such a rapid rate as to be virtually instantaneous.

The results of the STCP for in-vessel release duration are summarized in Tables 5.23 and 5.24. These durations are the time intervals extending from core melt 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 the reactor vessel internal pressure. Thus the high RCS pressure sequences show a trend toward shorter in-vessel release duration.

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 occur within a finite time interval (ex-vessel release duration).

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

Table 5.23 STCP Results for In-Vessel Release Duration for PWR Accident Sequences

Plant	Sequence	Release Duration (min)
Surry	TMLB'	41
	AG	215
	V	104
Zion	TMLU	41
	S2DCR	39
	S2DCF	39
Sequoyah	S3HF	46
	S3B	46
	TMLB'	37
	TBA	195
	ACD	73

Table 5.24 STCP Results for In-Vessel Release Duration for BWR Accident Sequence

Plant	Sequence	In-Vessel Release Duration (min)
Peach Bottom	TC2/TC3	66/68
	TC1	97
	TB1/TB2	91
	V	69
	S2E	81
LaSalle	TB	81
Grand Gulf	TB	122
	TC1	130
	TBS/TBR	96

6. RECOMMENDED PARAMETERS FOR SIMPLIFIED SOURCE TERM FORMULATION

6.1 Design Basis Accident

During an accident, fission product release from the fuel to the reactor coolant system is generally observed to occur in several steps (see Section 4.1.1). However, in the case of a DBA, operation of the ECCS should maintain the peak cladding temperature below 2200°F (1200°C). If one assumes complete failure of the cladding, i.e., 100% of the fuel pins are breached and a temperature of 1200°C for the entire core, it would take over 10 hours to release (via the transient release mechanism as modelled by the CORSOR-M code) an amount of fission products equivalent to that initially released from the gap. Hence, with the ECCS operational, the primary mode of release is the gap release.

The gap inventory of fission products thus provides a basis for the DBA source term. The fraction of the total core inventory which resides in the gap is specified in Reference 28. These values were obtained from WASH-1400. Table 6.1 summarizes the fractional inventory that resides in the gap. The nine fission product groups normally tracked by the STCP have been contracted here into five groups for the reasons given heretofore. The experimental data base that exists to establish the gap inventory was reviewed in References 9, 29 and 30. No more recent reviews addressing the gap inventory were found; this seems reasonable in that the major fission product releases in core melt accidents result from temperature dependent transient mechanism(s) and not from the gap release. The gap inventory data base is limited and summarized below.

The primary source of information was the study by Lorenz[38] in which the gap releases of cesium and iodine from simulated, trace irradiated and actual reactor fuel were examined. Additionally, a gap inventory of fission gas (0.25%) and a gap release of fuel dust (.003%) from actual fuel was reported. The study proposed an empirical model for the gap releases. The model generally predicted the observed cesium and iodine releases to within a factor of three. The model was then applied to a hypothetical PWR LOCA, in which 100% of the fuel pins were breached and the entire core experienced a transient to 1200°C for 10 minutes. Based upon these observations and analyses, the following estimates of gap release have been reported.[9,37,38]

The release of krypton was 2.8% of initial inventory and the release of xenon was estimated to be a factor of four lower than krypton. Cesium releases of 0.02 to 0.025% and iodine releases from 0.02 to 0.053% were reported. A comparison of these estimations and observations with the CORSOR values shown in Table 6.1 indicates that the releases of the dominant fission product groups are generally below the CORSOR code gap release estimates, assuming that fission products not explicitly measured are released in direct proportion to released fuel dust. However, the refactory species (Ba, Sr, Ru, Ce and La) appear to be under-estimated. Based again on the assumption of proportionality to fuel dust, these release fractions could be raised to at least $3x10^{-5}$.

Based upon the relative importance of individual species to early dose consequence as defined in Reference 38 and an assumed release fraction of

Table 6.1 DBA Source Term--Gap Release

Group	Fraction of Initial Core Inventory[a]
NG	0.03
Cs-I	0.05
Te	10^{-4}
Sr-Ba	10^{-6} $(3x10^{-5})$[b]
Ru-Ce-La	0.0 $(3x10^{-5})$

a Duration 0.5 hr.
b Recommended DBA release fraction given in parenthesis.

$3x10^{-5}$ for the Sr, Ba, Ru, La and Ce groups, it is expected that the volatiles (NG, Cs and I groups) will strongly dominate the radiation environment. The small releases of these groups could probably be neglected from the standpoint of dose. However, ancillary concerns, for example whether or not the radioactive wastes generated from the clean up of a DBA would be likely to exceed the low level radwaste limits on transuranic nuclides, could warrant their inclusion.

Reference 9 also mentions substantially higher estimates of noble gas release (~25%) under some circumstances. These estimates were based on calculations presented in Reference 39. A typical calculated result is shown in Figure 6.1. In the figure, the predicted inventory of noble gas that accumulates in the fuel rod gap is plotted against burnup. The results of two independent models for four steady-state constant-peak-power levels are shown.A power level of 7.3 kw/ft is typical for normal operation of this B&W fuel assembly. Hence, the elevated noble gas gap inventory of ~25% or greater is associated with a power level or burnup that is decidedly atypical of current reactor operation and should, therefore, not be considered in arriving at a best estimate value for current reactor operation. Recently a desire on the part of utilities to extend the level of fuel burnup has come about due to limitation on spent fuel storage capacity. Burnups as high as 60,000 Mw day/MTHM have been mentioned. If these high burnups were to become a reality, then the gap inventory should be increased. NUREG-0559 would support a fission gas gap inventory of 10 to 20% of the rod inventory. Commensurate increases in the cesium and iodine gap inventory appeared to be supported by Reference 30.

In addition to the gap release, the transient release initiates upon clad failure. Since the STCP was assembled to treat severe accidents, those accidents which result in complete core melting, the relative contributions from gap release and the transient release during core heatup is not resolved in the STCP output.

However, the calculation which was described at the beginning of this section clearly indicates that the gap release should be the dominant release under DBA conditions. Hence, adoption of the STCP gap release fractions, including the increases for the non-volatiles based wholly upon the experimentally observed fuel dust release, as the estimate of DBA source term to containment would appear to be reasonably conservative for current reactor practice if the fractions were applied to the entire core inventory (all rods fail) since the STCP volatile gap fractions are generally greater than existing best estimates.[38] Since the STCP, which is the major source of information available to formulate accident source terms, was designed to model complete core meltdown accidents, very little attention was given to the details of clad failure, especially early in the core heat up. Therefore, even if the STCP readily provided an estimate of the fraction of cladding failed at 1200°C, a large uncertainty would exist. It is felt that assuming 100% clad failure is prudent though conservative. These assumptions also provide a margin to compensate for any minor contribution of transient release from the fraction of fuel whose cladding has failed. If higher burnup fuel must be considered, then fractional release for the volatiles of ~0.2 are recommended. These should be applied to the core inventory.

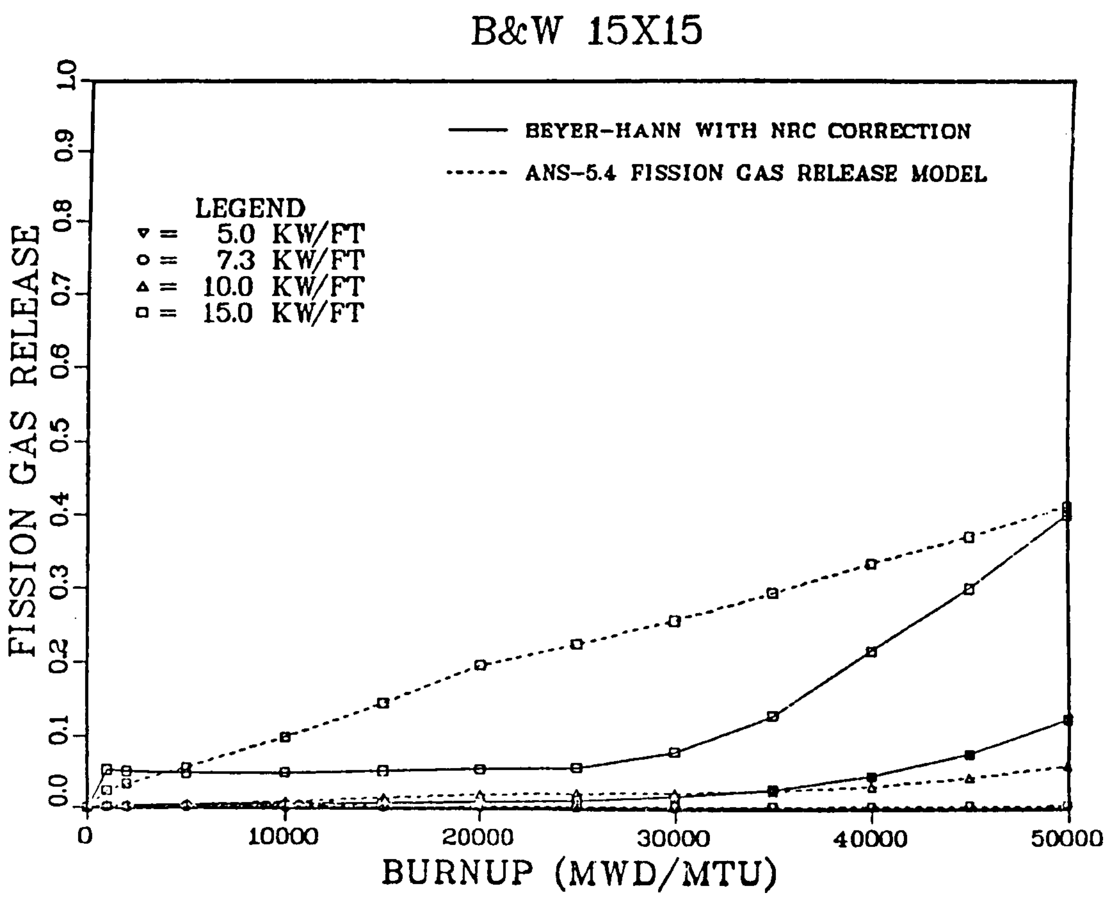

Figure 6.1 Fission Gas Release for Two Burnup Dependent Fission Gas Release Models[39]

Retention of fission products in the RCS during a DBA could vary widely. Large break LOCAs could result in high velocity steam flows with correspondingly short residence times in the RCS. Additionally, low material release would result in low aerosol concentrations which decrease the rate of aerosol depletion mechanisms. This, among other factors, would lead to low primary system retention. On the other hand, generally lower RCS temperatures might lead to enhanced vapor deposition and successful operation of the ECCS could provide liquid blockage to any release from the RCS in other DBAs.

The large break LOCA would appear to be the limiting case. In addition to low fission product retentions for vapor transported material, ultimately the RPV would be continually supplied with cooling water drawn from the reactor sump, which could, of course, also transport radioactive particulates and dissolved fission products to the containment. Hence, the assumption of no retention seems entirely appropriate in the regulatory context.

The chemical form of iodine released during a DBA is most likely cesium iodide. The available thermo-dynamic calculation[9] clearly supports this conclusion. These calculations indicate CsI stability is enhanced by hydrogen (reducing environment) and lower temperature. Though minor hydrogen formation is expected in a DBA, the low anticipated RCS temperatures favor CsI formation. The chemical processes that were discussed in section 4.3 and that could affect the form of iodine, cannot be ruled out for a DBA accident at this time. Hence, there is still some uncertainty regarding the amounts that might be generated of other chemical forms of iodine.

Another property important in specifying the source term is the duration of release. Table 6.2 summarizes estimates of the duration of the gap release for several typical STCP calculations, the estimated duration is the time needed to heat the core from the peak temperature of 900°C to a core wide average temperature of 900°C. The Surry AG sequence, in which temporary operation of the ECC system occurs, may be the most representative (from the standpoint that ECCS was actuated) STCP result for assessing the duration of the PWR DBA release. For BWRs, the release duration ranges from approximately 10 to 30 minutes. Given that ESFs will operate and DBA will be terminated, it is believed that the longer durations are more applicable. Therefore, a 30 minute duration for the DBA release duration, independent of reactor type, seems appropriate.

6.2 Severe Accidents

Generic fission product releases from reactor coolant system (FRCS) and from the melt during core/concrete interaction (FCCI) are tabulated in Tables 6.3 and 6.4. The release fraction for each radionuclide group which is assigned to an accident category generally is taken as the highest STCP calculated fraction from all of those accident sequences assigned into the release category. However, for in-vessel release fractions of the Ru-La-Ce group, the higher values (FRCS = 3×10^{-5}) based on similar arguments discussed in Section 6.1 are recommended.

Table 6.2 Estimated Duration of Gap Release

Plant		Sequence	Time (min)[a]
PWRs	Zion	S_2DC	6
		TMLU	6
	Surry	TMLB	7
		V	6
		AG	27[b]
	Sequoyah	TMLB	6
BWRs	Peach Bottom	TC1	10
		TC2/TC3	13
		TB1/TB2	24
		V	8
	Grand Gulf	TC	19
		TB	30
		TBS	16

a Time predicted between peak and core-wide average temperature reaching clad failure temperature.

b Assumed most representative of DBA for PWRs since ECC operates.

Table 6.3 Simplified PWR Fission Product Releases to Containment for Severe Accident Conditions

Groups	FRCS[a]		FCCI	
	H,I	L	Basaltic Concrete	Limestone Concrete
NG	1.0[b]	1.0	0	0
Cs, I	0.35	0.9	0	0
Te	0.3	0.65	0.15	0.35
Sr, Ba	$2x10^{-3}$	0.01	0.15	0.4
Ru, Ce, La	$3x10^{-5}$	$3x10^{-5}$	$6.x10^{-3}$	0.05
Release Duration	40 mins.		2 hrs.[c]	

a H, I and L refer to high, intermediate or low RCS pressure, respectively.

b All entries are fractions of the initial core inventory.

c Except for Te where the duration of ex-vessel release is extended to 5 hours for PWRs and 6 hours for BWRs.

Table 6.4 Simplified BWR Fission Product Releases to Containment for Severe Accident Conditions

GROUPS	FRCS[a]		FCCI	
	H,I	L	Basaltic Concrete	Limestone Concrete
NG	1.0[b]	1.0	0	0
Cs,I	0.7	0.8	0.15	0.15
Te	0.1	0.15	0.12	0.5
Sr,Ba	$6x10^{-3}$	$6x10^{-3}$	0.2	0.7
Ru,Ce,La	$3x10^{-5}$	$3x10^{-5}$	$6x10^{-3}$	0.06
Release Duration	1.5 hrs.		3 hrs.[c]	

a H, I and L refer to high, intermediate or low RCS pressure, respectively.

b All entries are fractions of the initial core inventory.

c Except for Te where the duration of ex-vessel release is extended to 6 hours.

The duration of these releases to containment have also been selected through an assessment of the existing STCP calculations. In-vessel releases generally occur within 40 minutes for PWR and 1.5 hours for BWRs. These releases are assumed to begin with DBA gap releases following initial core melt. The major in-vessel releases are then assumed to start at 10 minutes into the accident. Although the release from core/concrete interaction is predicted to extend many hours beyond corium/concrete interaction initiation, generally 90% of the radionuclide releases (except Te) occur within 2 hours for PWRs and 3 hours for BWRs. For tellurium, ex-vessel release durations of five hours for PWRs and 6 hours for BWRs is assumed.

DF values of 10, 3 and 1 have been assigned to conditions of very deep, deep and shallow water (or dry) pools overlying the corium during MCCI.

In the simplified formulation for appearance rate into the containment, the fission product releases are treated as being proportional to time. Figures 6.2 through 6.14 show comparisons of the proposed simplified appearance rates for the various radionuclides to the STCP calculations. As expected, the simplified releases generally envelop the STCP calculated results.

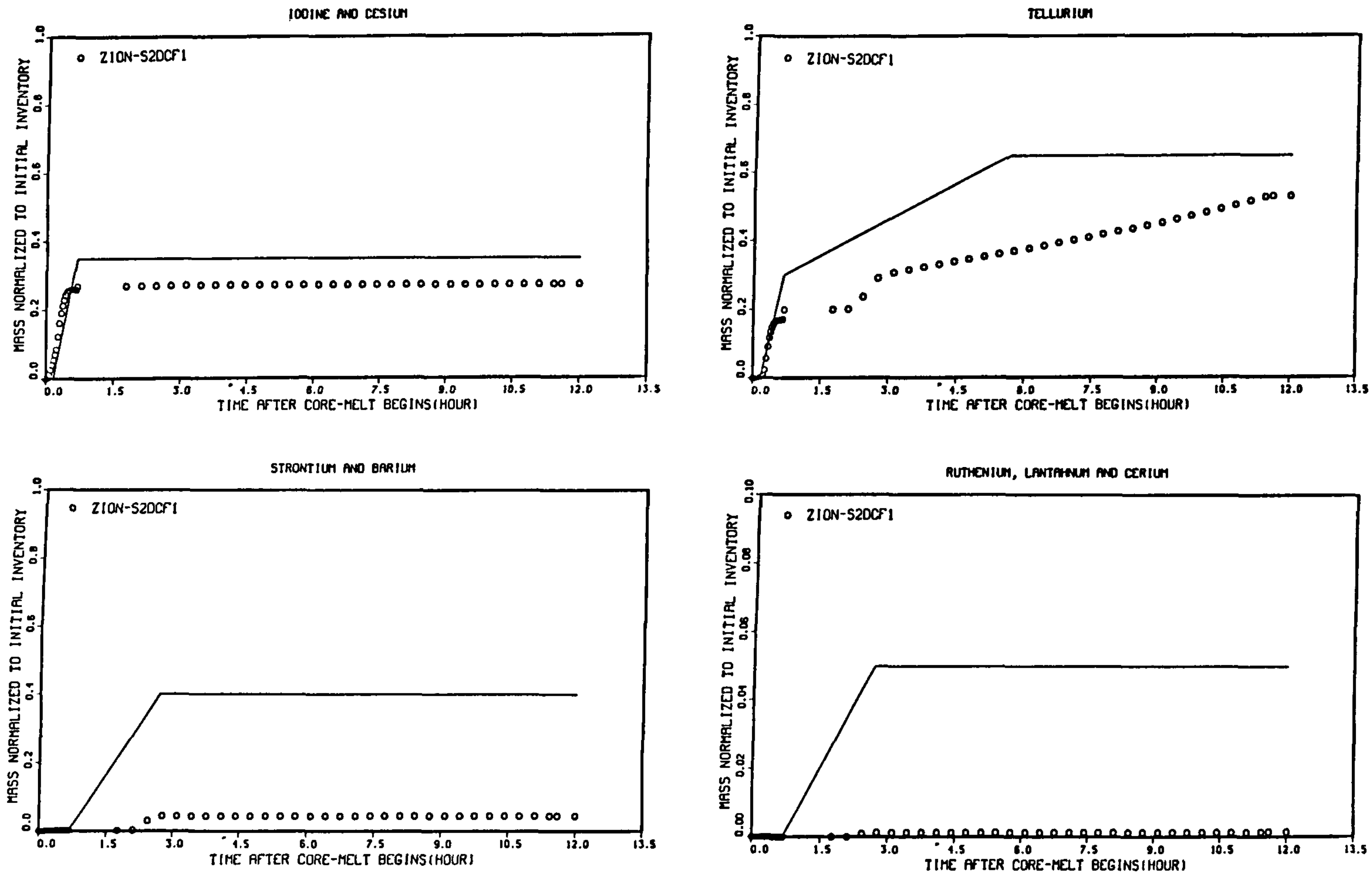

Figure 6.2 Cumulative Release Fractions to Containment
(PWR, High RCS Pressure, Limestone Concrete, Dry Cavity)

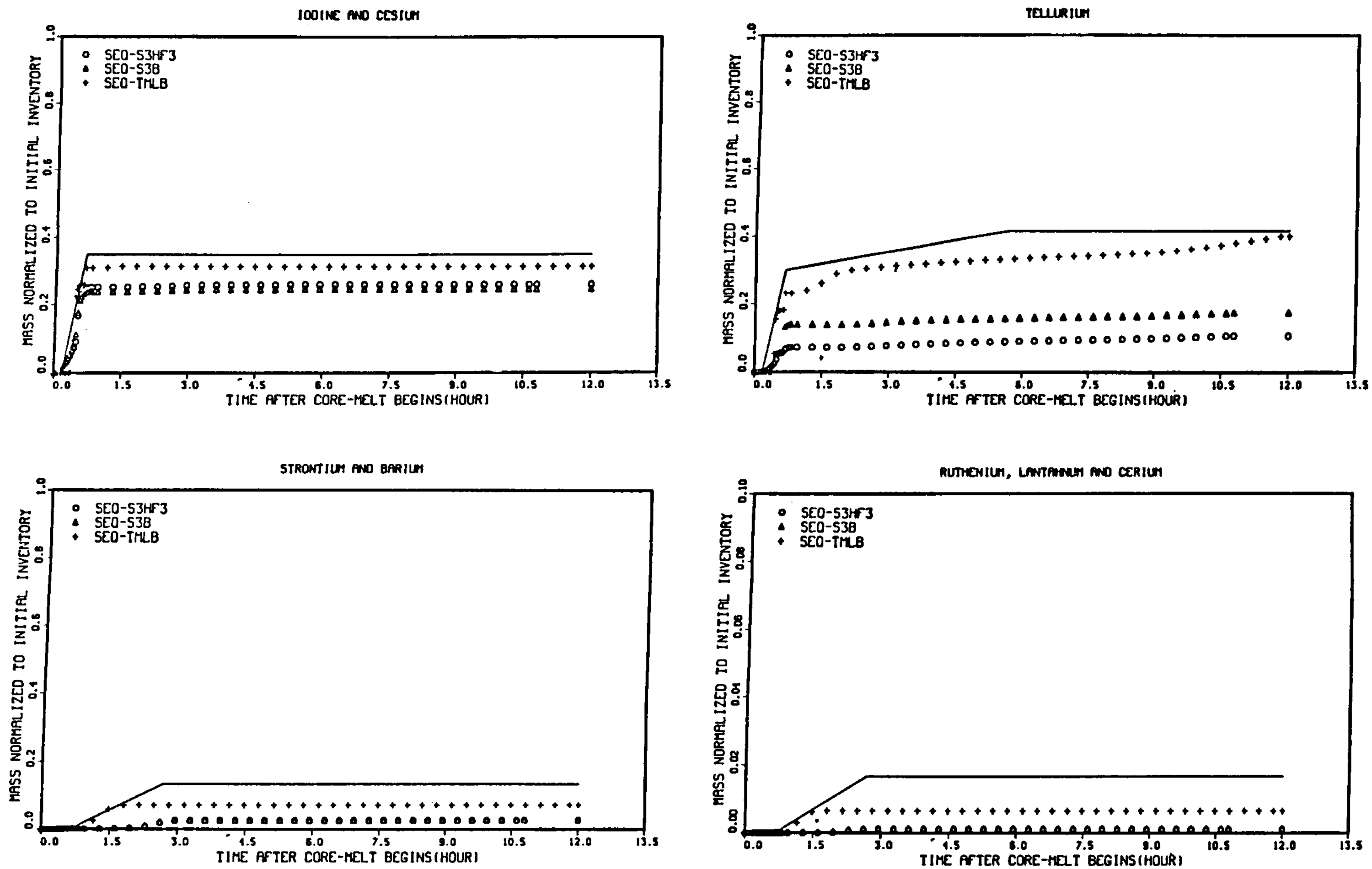

Figure 6.3 Cumulative Release Fractions to Containment
(PWR, High RCS Pressure, Limestone Concrete, Wet Cavity: DF=3)

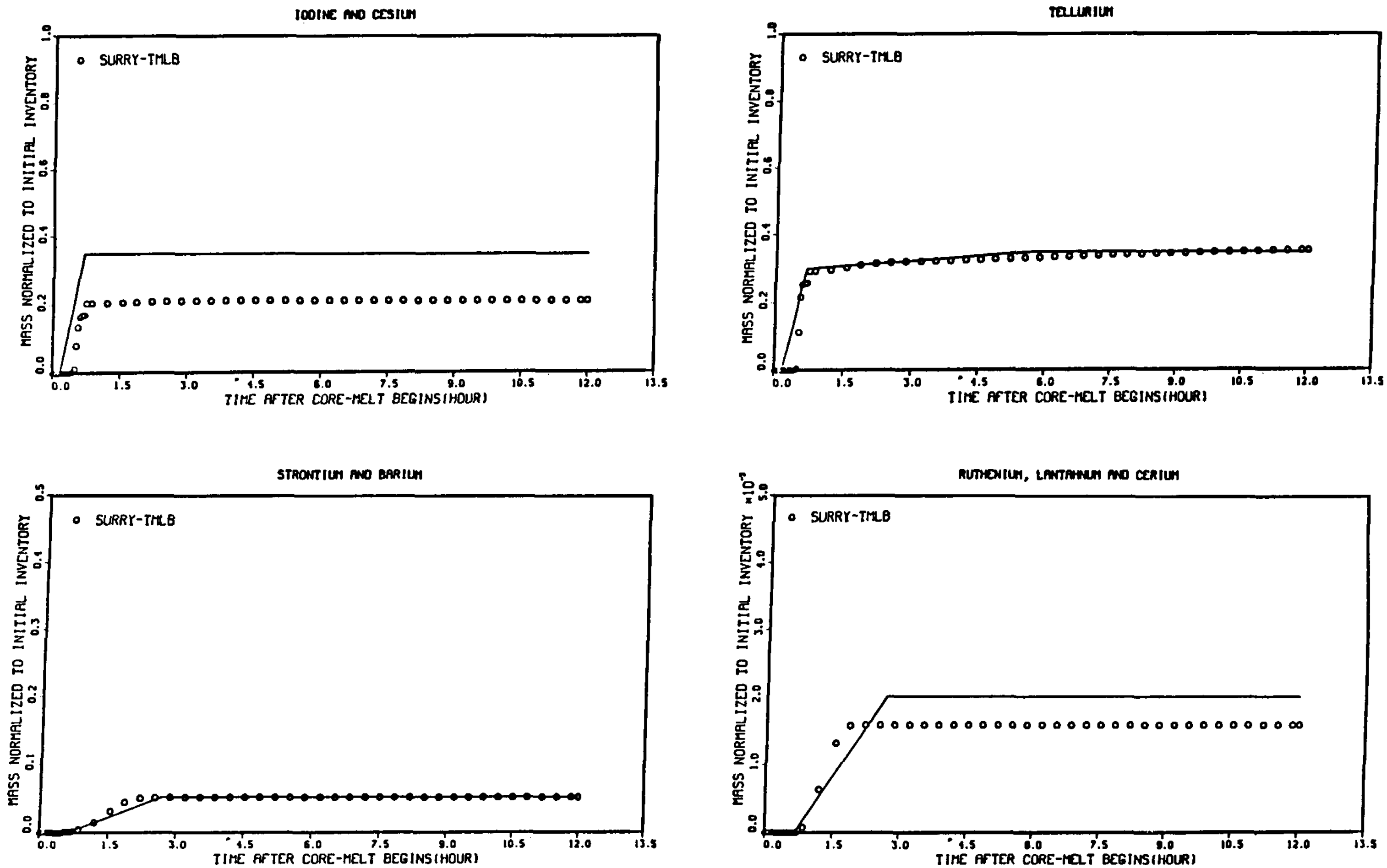

Figure 6.4 Cumulative Release Fractions to Containment
(PWR, High RCS Pressure, Basaltic Concrete, Wet Cavity: DF=3)

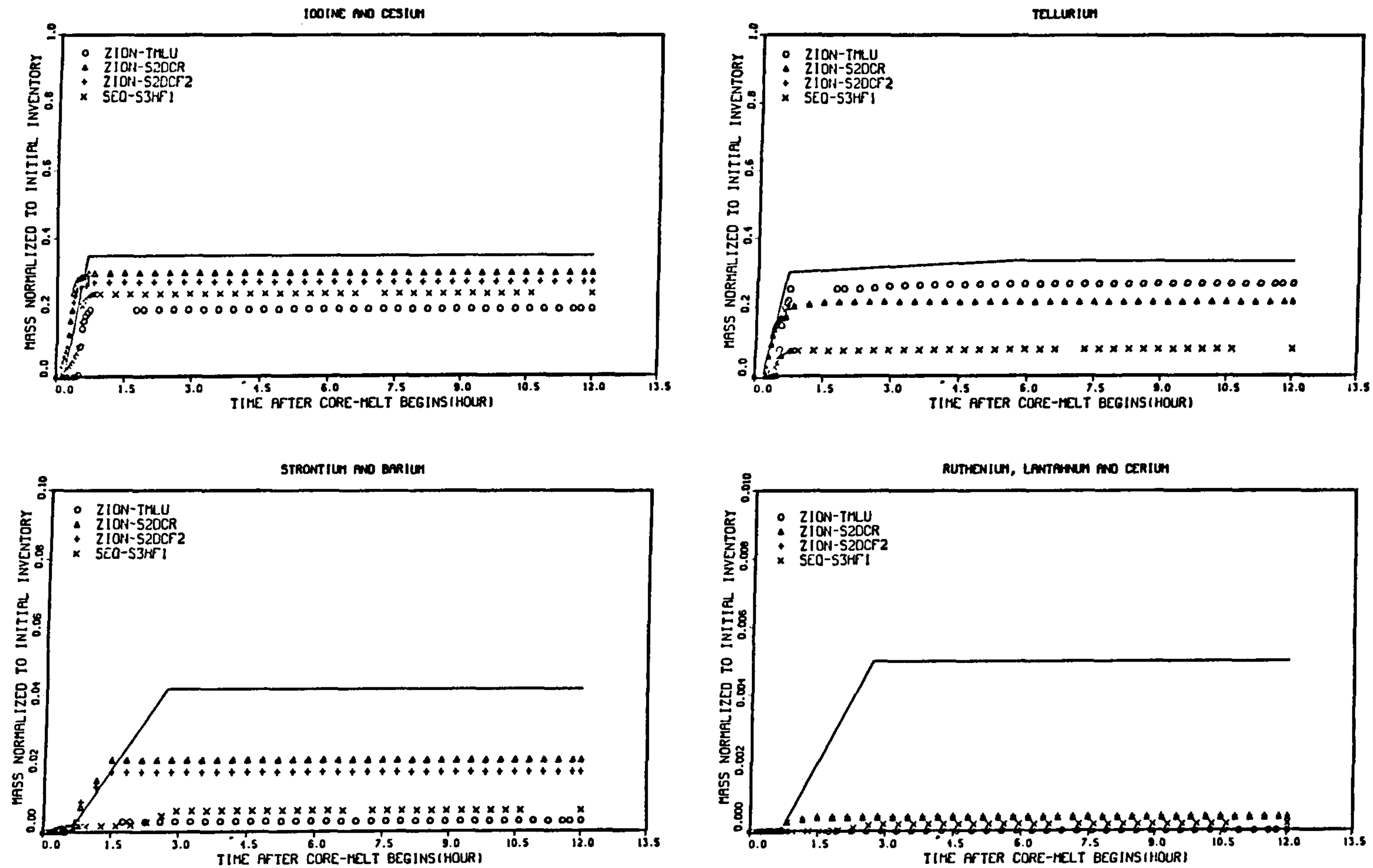

Figure 6.5 Cumulative Release Fractions to Containment
(PWR, High RCS Pressure, Limestone Concrete, Wet Cavity: DF=10)

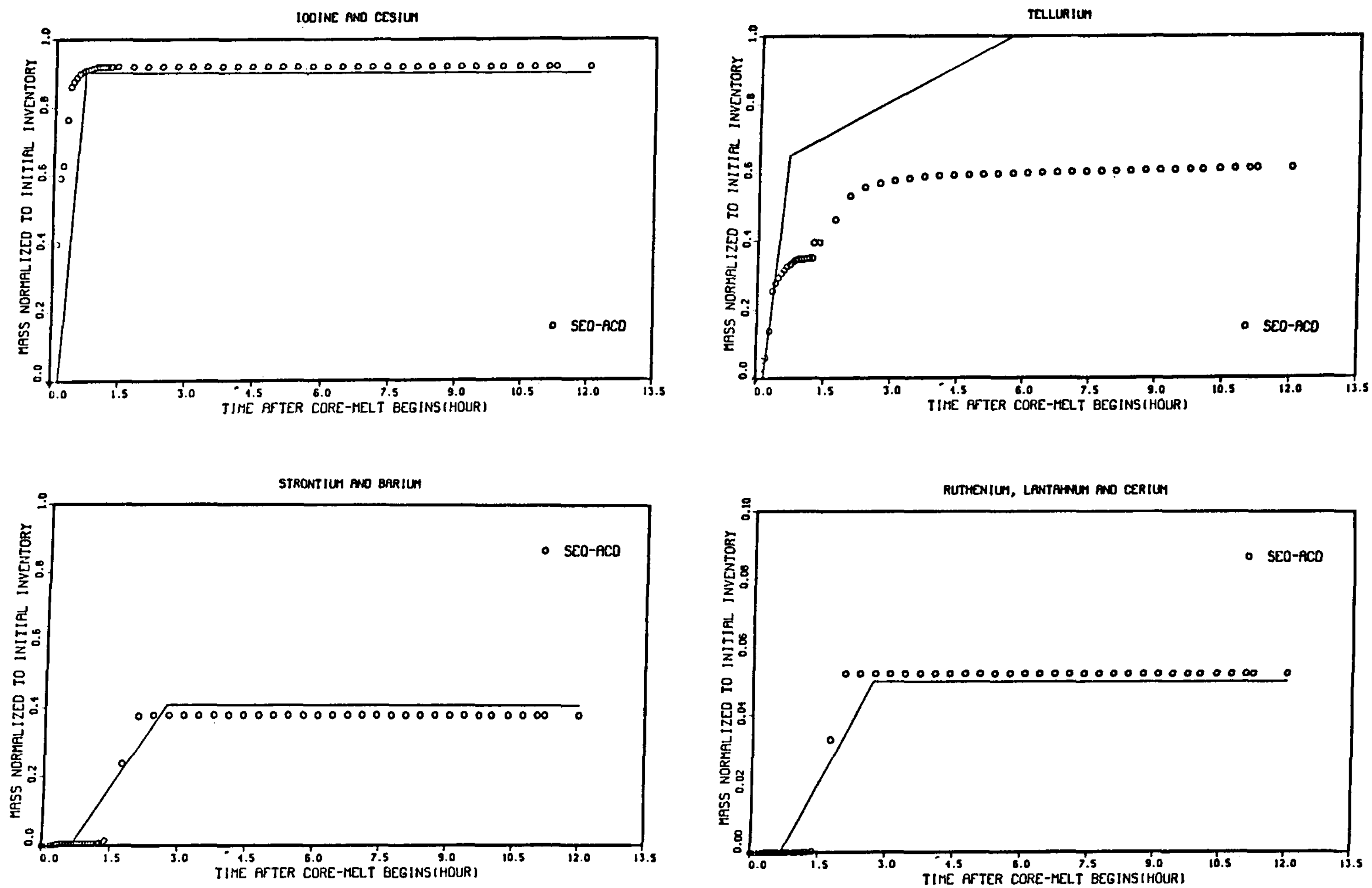

Figure 6.6 Cumulative Release Fractions to Containment
(PWR, Low RCS Pressure, Limestone Concrete, Dry Cavity)

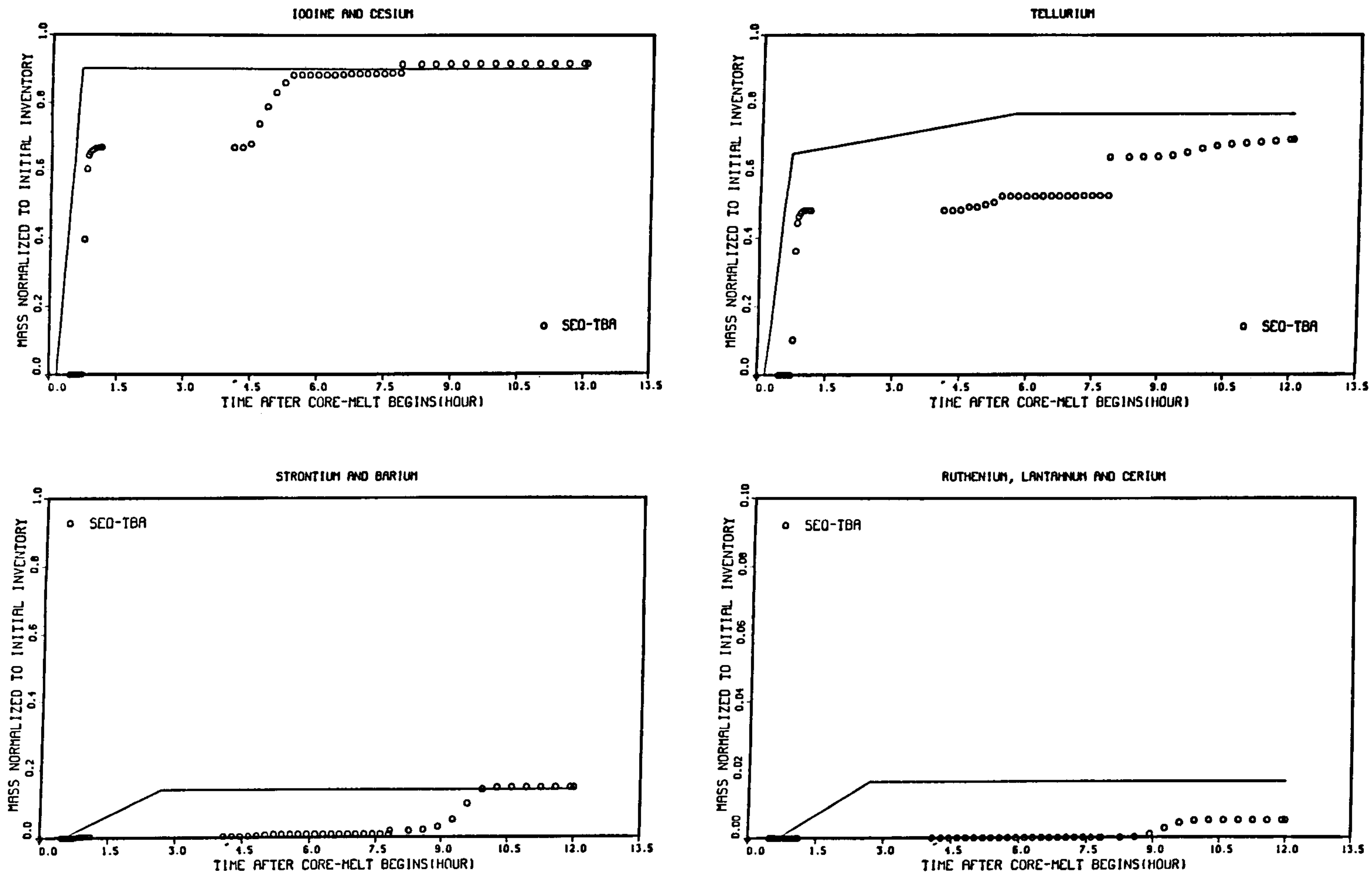

Figure 6.7 Cumulative Release Fractions to Containment
(PWR, Low RCS Pressure, Limestone Concrete, Wet Cavity: DF=3)

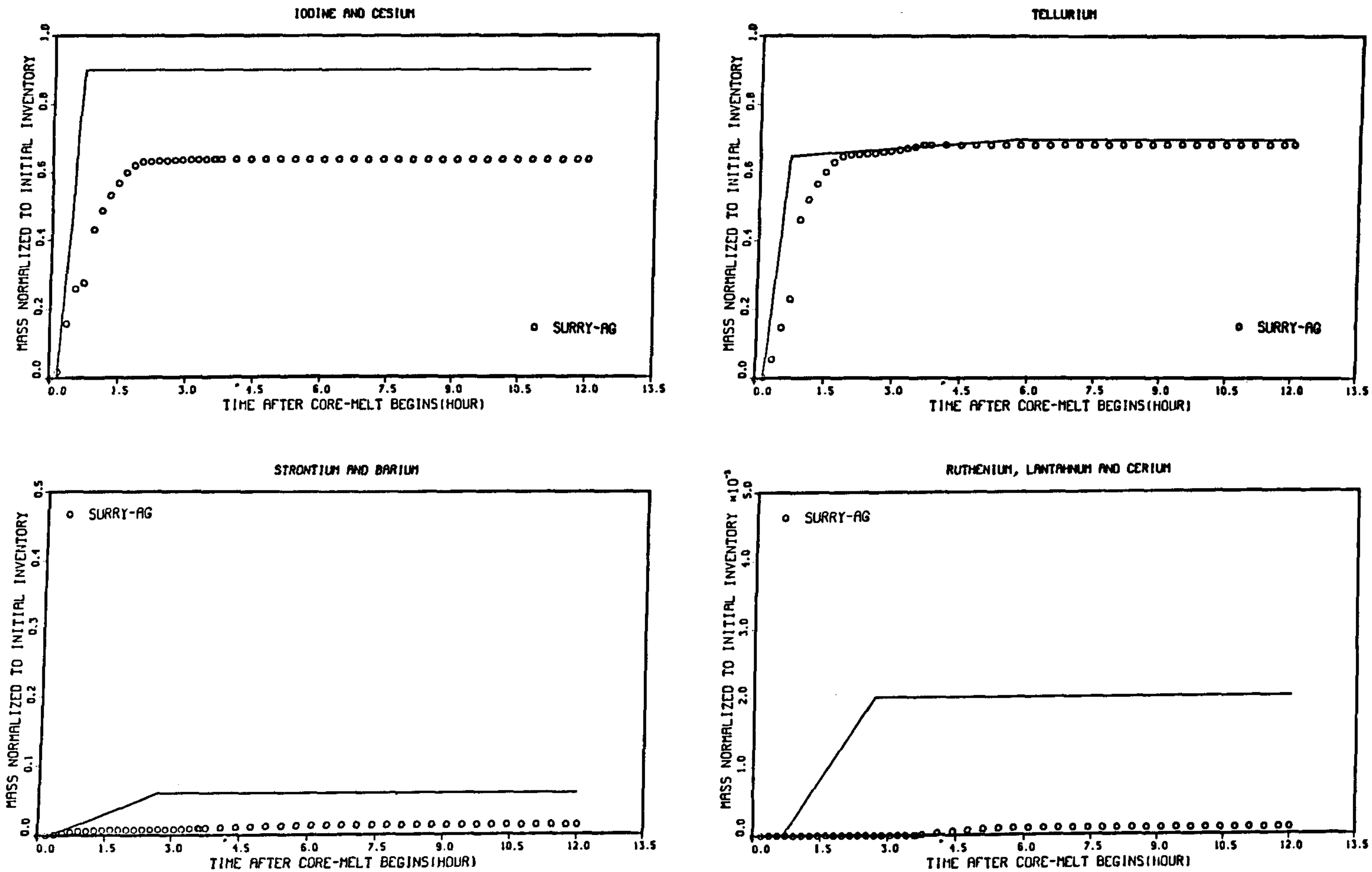

Figure 6.8 Cumulative Release Fractions to Containment
(PWR, Low RCS Pressure, Basaltic Concrete, Wet Cavity: DF=3)

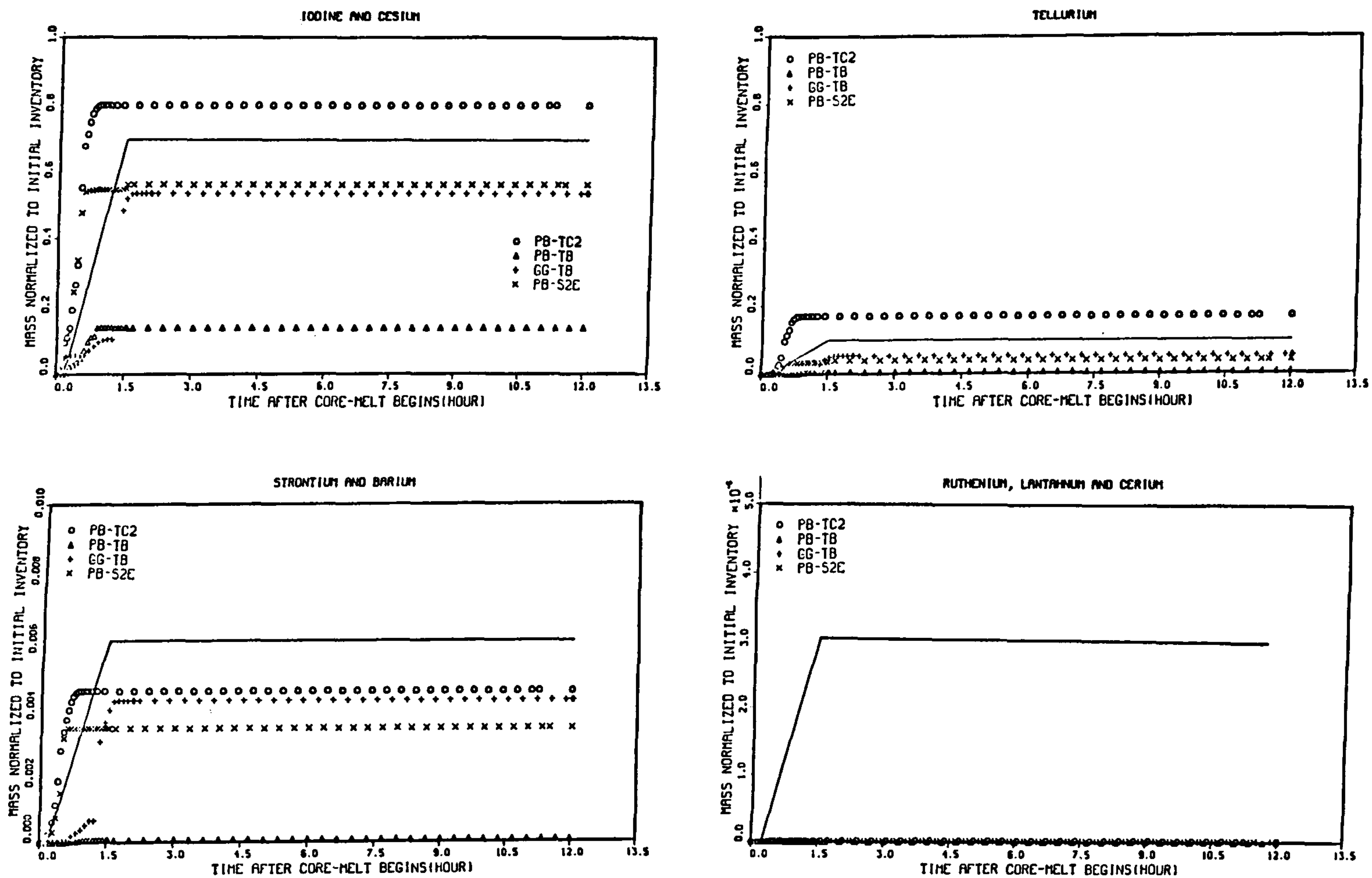

Figure 6.9 Cumulative In-Vessel Release Fractions
(BWR, High RCS Pressure)

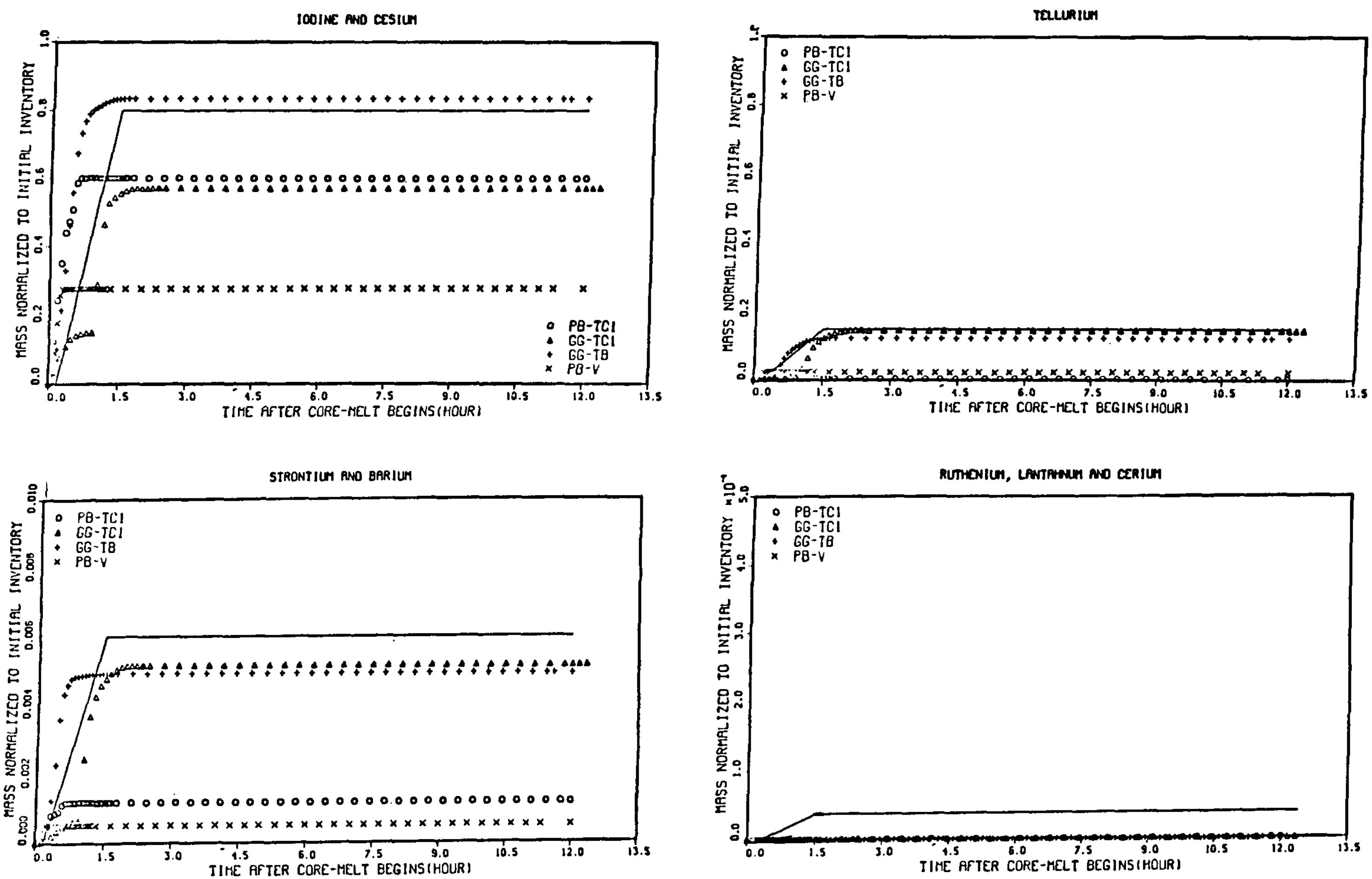

Figure 6.10 Cumulative In-Vessel Release Fractions (BWR, Low RCS Pressure)

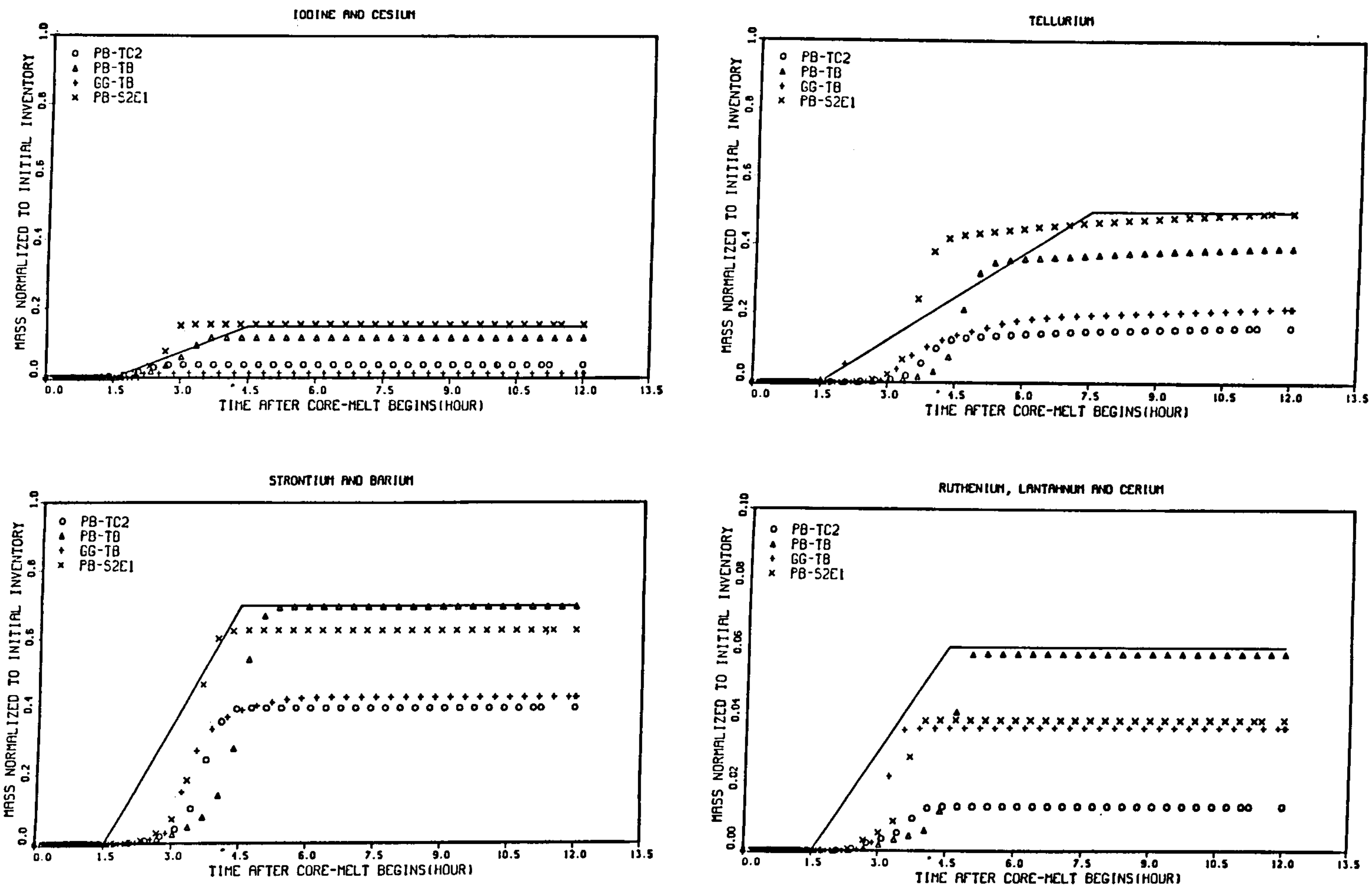

Figure 6.11 Cumulative Ex-Vessel Release Fractions to Containment (BWR, High RCS Pressure, Limestone Concrete, Dry Cavity)

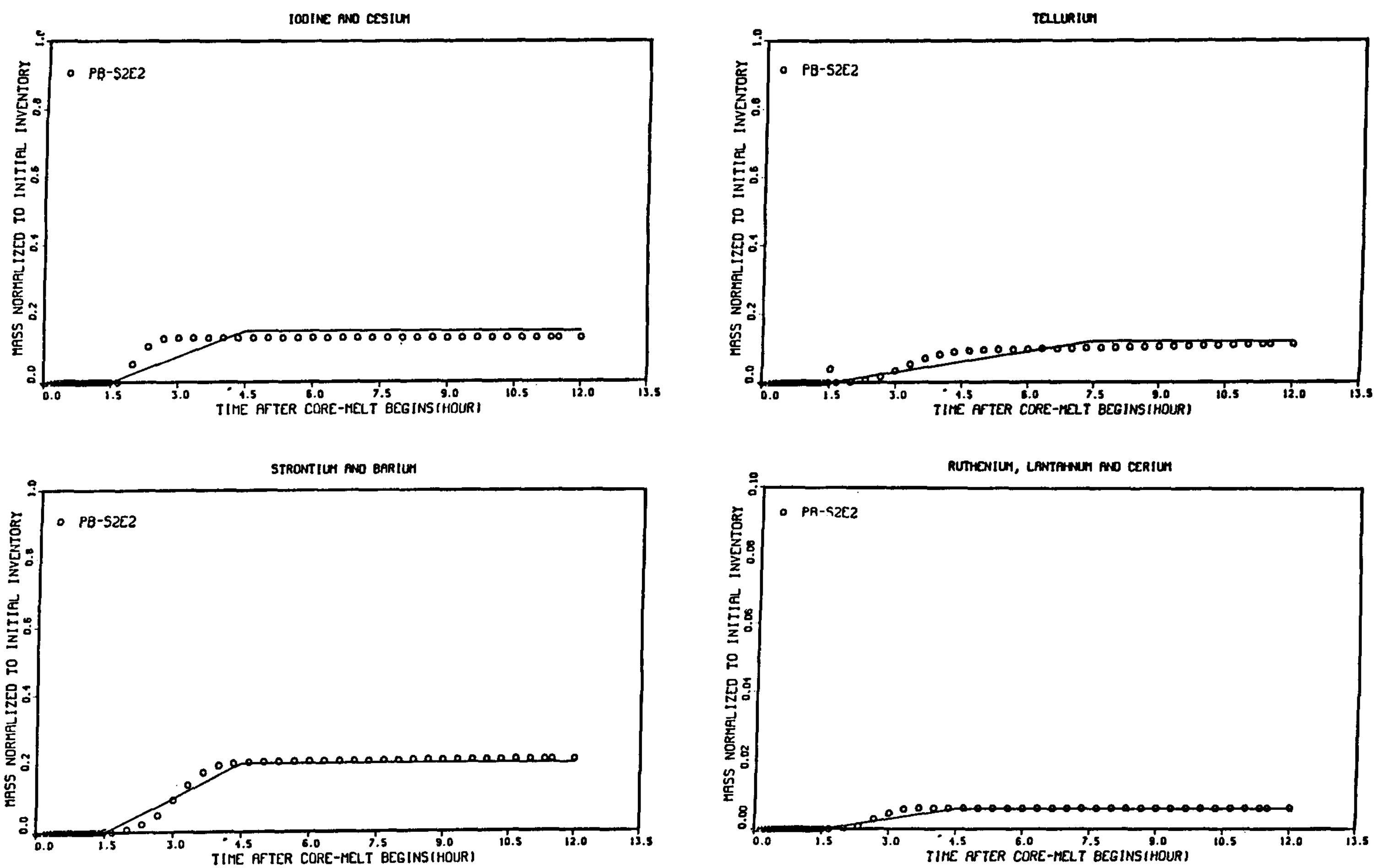

Figure 6.12 Cumulative Ex-Vessel Release Fractions to Containment (BWR, High RCS Pressure, Basaltic Concrete, Dry Cavity)

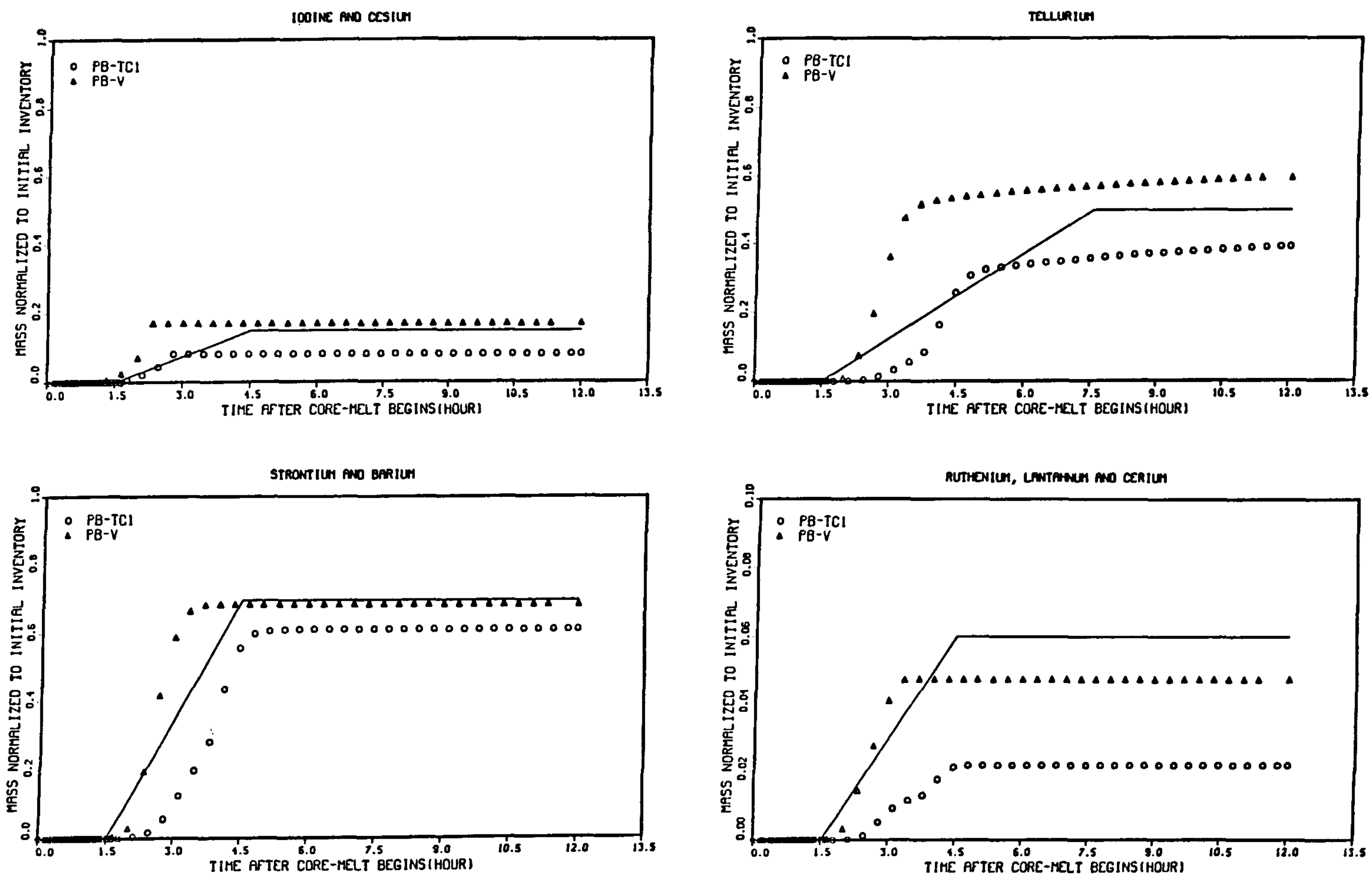

Figure 6.13 Cumulative Ex-Vessel Release Fractions to Containment
(BWR, Low RCS Pressure, Limestone Concrete, Dry Cavity)

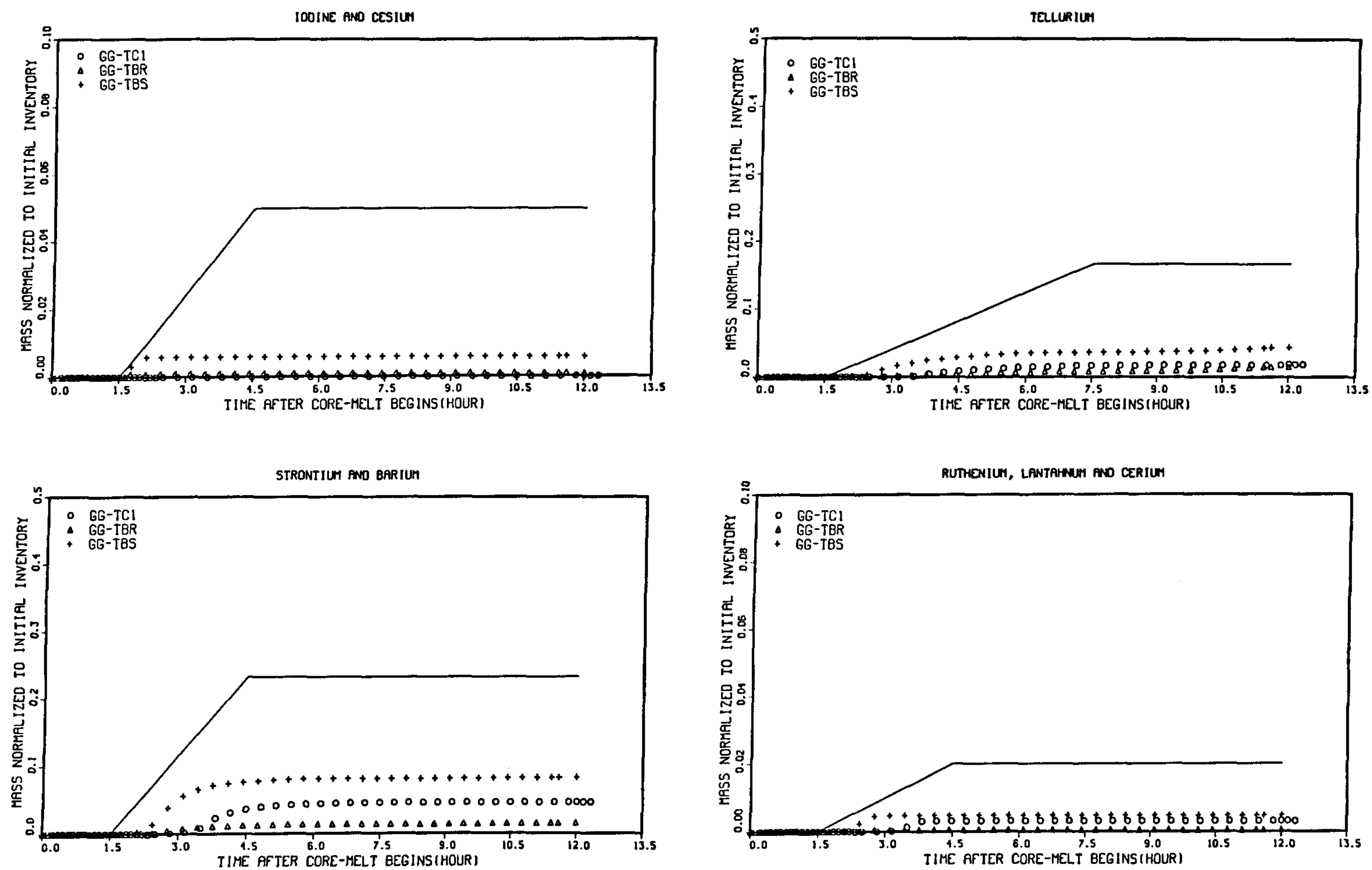

Figure 6.14 Cumulative Ex-Vessel Release Fractions to Containment
(BWR, Low RCS Pressure, Limestone Concrete, Wet Cavity: DF=3)

7. DISCUSSION OF UNCERTAINTIES AND THEIR TREATMENT

7.1 Background

The objective of this study was the development of a simplified formulation to estimate the fission product release characteristics into containments for both design basis and severe accidents in nuclear power plants. A complete estimation of a source term to containment includes the amount of material released including chemical forms, timing of the release, and the uncertainty inherent in the release estimates. The uncertainty provides a basis for developing explicit margins of safety that are applied to the realistic best estimate source term results when these results are used in regulation.

The method proposed in this study is an extension of the method that is currently being used to estimate source terms in the Reactor Risk Reference Document.[10] The data base, used to calculate the empirical inputs to both methods, is the series of detailed STCP calculations performed primarily by BCL in support of the SARRP effort.[11-18] The degree to which the simplified source term estimates developed here reproduce the results of detailed STCP calculations was examined in the previous section. Further refinements of the methods aimed at improving the fit, can only be made through the introduction of additional models to account for additional plant features, e.g., operation of the core heat removal system, not currently considered in this method.

Ultimately, the uncertainty associated with the simplified method of source term prediction described here must be based upon examination of the uncertainties associated with the detailed results of the STCP including phenomenological issues not addressed by the STCP methodology. Historically, the determination of the uncertainties in source term prediction have received less attention than the development of the predictive methodologies themselves. This area is being actively investigated at this time. The results of these efforts are integral, and indispensable, parts of a realistic uncertainty assessment.

Previous efforts in this area were reviewed and summarized in NUREG-0956.[3] These included reviews of the BMI-2104 suite of codes, which were enhanced and integrated into the STCP, by members of the technical community actively involved in source term research[37] and analytical efforts by an independent study group of scientists.[40] The major areas of phenomenological uncertainties identified with the source term predictive methods are summarized in Table 7.1. Also included are the issues which have been identified through a joint NRC industry review,[3,41] independent panels of experts[42] (referred to as the Kouts Panels), and several issues which are being addressed in the NUREG-1150 effort. Within major areas, e.g., Accident and Core Melt Progression, more specific items are listed. These are identified by a label, e.g., APS-7, which indicates the source of the items. These sub-listings are often redundant; APS-17, ANS-3 and IDCOR/NRC-2 basically identify the same issue. This was done intentionally to allow the reader to assess the level of technical agreement on the importance of a particular issue.

The first major attempt to quantify uncertainty in the source term pre-

Table 7.1 Major Areas of Source Term Uncertainty

Accident and Core Melt Progression

APS-7.	Damage progression in core
APS-8.	Fragmentation of corium at reactor pressure vessel melt-through
APS-14.	Change of sequence by fission product heating
APS-17.	Natural circulation research
ANS-1.	Mechanisms of core degradation: Space and time dependence. Silver aerosol generation.
ANS-3.	Thermal hydraulics (in RCS): Thermal convection loops. Pump seal failure.
IDCOR/NRC-2.	Recirculation of coolant in the reactor vessel
IDCOR/NRC-4.	In-vessel melt progression
IDCOR/NRC-6.	Direct heating of containment by ejected core material
IDCOR/NRC-13.	Hydrogen ignition and burning
IDCOR/NRC-14.	Accident sequence definition
IDCOR/NRC-5.	Containment failure by in-vessel steam explosion
SARRP-4.	High pressure melt ejection
KOUTS PANELS-1.	Natural Circulation in Reactor Coolant System
KOUTS PANELS-2.	Core Melt Progression and Hydrogen Generation
KOUTS PANELS-4.	High Pressure Melt Ejection

In-vessel Release from Fuel

APS-1.	Vaporization of low volatility fission products
APS-11.	Generation mechanisms for aerosols
IDCOR/NRC-1.	Fission product release prior to vessel failure

Table 7.1 Continued

Fission Product Transport and Retention in RCS

APS-4.	Transport of radionuclides through reactor
APS-16.	Aerosol deposition on pipes
ANS-2.	Aerosol transport in reactor coolant system: Absorption-desorption. Chemical reactions. Fission product self-heating.
IDCOR/NRC-3.	Deposition of fission products and aerosols in reactor coolant system
IDCOR/NRC-9.	Revaporization of fission products
SARRP-3.	Revolatilization of fission products from RCS
SARRP-5.	Direct heating fission product release
SARRP-6.	Retention in steam generator
KOUTS PANELS-8.	Fission Product Revaporization

Fission Product Chemistry

APS-5.	Tellurium behavior
APS-6.	Release of volatile forms of iodine
SARRP-1.	CsI decomposition
SARRP-2.	Late iodine release
KOUTS PANELS-7.	Iodine Chemical Form: CsI disociation, reaction with metal surfaces and B_4C, behavior in high temperature oxidizing atmosphere.

Table 7.1 Continued

Core/Concrete Interaction

APS-2.	Thermal hydraulics of core-concrete interaction
APS-3.	Release of refractory materials in core-concrete interaction
APS-11.	Generation mechanism for aerosols
ANS-4.	Corium-concrete reactions: Includes fission product and aerosol release
IDCOR/NRC-7.	Ex-vessel fission product and aerosol release
IDCOR/NRC-8.	Ex-vessel heat transfer from molten core to concrete and containment
KOUTS PANELS-5.	Core-Concrete Interactions

dictions methods was the QUEST study.[43] For two reactor types, this study has produced estimates of the sensitivity (rather than the uncertainty) of the source term results to variations of the computer code inputs and to changes in the phenomena modelled in the codes.

The major conclusions drawn in this study were:

Surry-TMLB'

- The uncertainty in the source term is broad and appears to increase during the course of the accident. The amount of suspended aerosol varied by at least two orders of magnitude. The modeling of phenomena (examined by either modification of existing (STCP) models, or use of alternate calculational tools) contributed more to the uncertainty of the source term estimate than reasonable variations to the inputs to the codes.

- Near the time of vessel failure, the source term uncertainty is dominated by the core-temperature history, release of fission products from the fuel, natural circulation in the RCS, and retention of fission products in the RCS.

- Late in the accident, the source term uncertainty is dominated by the corium temperature during core/concrete interaction, aerosol shape factors, turbulent agglomeration rates, and the potential for resuspension of settled fission product aerosols.

Surry S_2D

- The uncertainty in source term is related to the effect of containment sprays on the core debris temperatures and the debris configuration after the corium contacts water in the reactor cavity.

Grand Gulf TC

- The uncertainty in source term is wide.

- If the pressure suppression pool is not disrupted by a high pressure ejection of the melt from the reactor vessel, the uncertainty is biased to lower releases and is dominated by the release rate coefficients of tellurium during core melting, deposition of radionuclides in the reactor coolant system, and the effectiveness of fission product scrubbing by the pressure suppression pool.

- If suppression pool disruption occurs, large amounts of fission products may pass through the pool. This ineffective scrubbing by the suppression pool can lead to much larger release than predicted by the current source term codes.

The authors of this study noted that additional calculations for other reactors and sequences were needed to judge the generality of these conclusions.

The QUEST study used plausible ranges of inputs without statistical weighting to determine the source term output variations. The results are therefore sensitivity rather than uncertainty estimates. The QUASAR study[44] at Brookhaven will attempt to estimate the uncertainty in source term predictions by identifying reasonable input ranges and statistical distributions, where possible, and propagating these inputs through the STCP using a Latin Hypercube Sampling technique.[45] Phenomena uncertainties will also be addressed in the QUASAR program, however, routine methods for this evaluation do not exist and a major effort within the QUASAR program is the development of methods for assessing phenomenological uncertainties and combining them with input uncertainties.

Another ongoing effort that attempts to define source term uncertainties is the NUREG-1150 effort.[10] Less rigorous methods are being employed in this program than have been applied in the QUEST and QUASAR studies. However, the similarities between the NUREG-1150 method to estimate source terms and that proposed in this study suggest that the results of the NUREG-1150 effort with appropriate modifications to account for the time dependence, may provide a suitable means of directly evaluating uncertainty associated with the simplified method proposed in this report.

In summary, past efforts to evaluate the uncertainties of source term predictive methods have been limited to the technical reviews of the phenomena modelled within the BMI-2104 suite of codes, which were the basis of the STCP and an extensive sensitivity study performed with the BMI-2104 suite of codes. Ongoing efforts, NUREG-1150 and QUASAR, are attempting to more quantitatively define the inherent uncertainty in source term estimates. Due to the complexity of source term estimation, gaps in the experimental data bases, and the difficulties in quantitatively defining the associated uncertainties, even these studies are of limited scope. Hence, while the results of previous and current uncertainty efforts will form a basis to establish defensible uncertainty limits, one can expect refinements of these limits to parallel ongoing or future research efforts aimed at resolving the issues summarized in Table 7.1.

Not withstanding the intense research activity in the area of severe accidents, aimed specifically at reduction of the uncertainty, it is likely that large source term uncertainties will exist into the future. This seems quite inevitable, since industry and the NRC have expended so much effort to minimize the possibility of nuclear power plant accidents, and since experimental efforts by necessity must investigate analog or small scale prototypes.

7.2 Present Approach

In order to make a general assessment of the simplified source terms proposed in this study, a methodology similar to the one used for the risk rebaselining studies is used. The NUREG-1150 methodology is particularly applicable to uncertainty assessments of the estimate of simplified radiological release into the containment. Here a comparison between the simplified source term into the containment for PWRs as proposed in this report and ranges of source term into the containment obtained by NUREG-1150 methodology for Zion[46] will be presented.

7.2.1 Ranges of Radiological Release Fractions Into the Containment for PWRs (Dry Cavity, Limestone Concrete)

The simplified radiological release fraction into the containment as discussed in Chapter 4 is represented by:

$$Ft(i) = FRCS(i) + FCCI(i) \tag{7.1}$$

where:

$$FRCS(i) = FCOR(i) * FVES(i) \tag{7.2}$$

The Source Term Code Package (STCP) assumes that iodine will be present as CsI. Recent experimental data[50] so indicate that in the presence of steam and radiation (as in the primary system), CsI tends to decompose to elemental iodine (or some other volatile form) which is released from the primary system with zero retention, although the cesium derived from CsI may still be retained. In order to allow for this possibility Eq. (6.1) is modified for iodine as follows:

$$Ft(I) = FCOR(I) * [FI_2 + (1-FI_2) * FVES(I)] + FCCI(I) \tag{7.3}$$

Here FI_2 is the fraction of the iodine released from the fuel which is converted to the more volatile form (elemental I_2, hydrogen iodide, etc.); no in-vessel retention is credited for this more volatile form.

Several mechanisms have been identified by which there could be significant amounts of Cs and iodine in the containment at late times. These include:

(a) revolatilization from RCS,
(b) slow deposition of initial releases from RCS,
(c) radioactive decay chains,
(d) resuspension at containment failure, and
(e) retention in melt until after vessel failure.

All of these mechanisms are lumped together as a "delayed release" for I and Cs. Therefore, the actual contribution to the source term is treated as a delayed release through a parameter FLATE, which represents the amount of Cs or I that is released to the containment at late times; that is:

$$\text{Delayed Release} = FLATE * [1-Ft(i)] \tag{7.4}$$

where Ft(i) is the release fraction calculated from Eq. (7.1), without allowing for the possibility of late release.

Another important phenomena not modeled in the STCP methodology deals with radiological releases associated with high pressure melt ejection (HPE) and direct containment heating (DCH).

If the primary system is pressurized at the time of vessel breach, fuel will be ejected under pressure in a process which can result in significant aerosol generation, as has been demonstrated experimentally. This purely physical process is expected to occur more or less independently of whether or not the dispersed core debris results in significant containment pressurization due to 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 fuel.

Based on the recommendation of Powers,[47] the actual contribution to the source term from these processes are given by:

$$FHPE = (1-FCOR) * FREJ * RADEJL \tag{7.5}$$

and

$$FDCH = (1-FCOR) * FREJ * FRDH * (1-RADEJL) * RADDHL \tag{7.6}$$

where

FREJ = fraction of the core ejected
FREJ * FRDH = measure of the amount of direct heating
RADEJL = radionuclide release fraction for HPE
RADDHL = radionuclide release fraction for DCH

Table 7.2 through 7.8 show the values assigned to the issues for each level as defined for Zion, with the weights agreed upon by the expert review group. The definition of the source term issues and levels is identical to that found in Zion, with the exception of FVES and FCCI. For FVES only two categories of high and low RCS pressure is considered. For the corium concrete interaction release the outcomes are expressed in terms of the fractional releases of a given radiological species from the melt FCCI/(1-FCOR).

Values of FREJ and FRDH were assumed to be 0.73 and 0.75, respectively. These values correspond to the upper bound values for high pressure sequences with dry cavity used for Zion and Surry. However, it should be noted at this high value of the fraction of the core ejected, FREJ; the containment may not sustain the resulting pressure increase (130 psi pressure rise for Surry).

The HPE and DCH release fractions calculated apply only to that fraction of the inventory not released prior to vessel breach. It was also assumed that a fraction of the core equal to (FREJ * FRDH) will not be available for core concrete interaction, on the grounds that it will be too widely dispersed to reheat and attack the concrete.

The calculations were performed to obtain the source terms as a fraction of the initial core inventory released to the containment atmosphere using Eqs. (7.1) through (7.6). The results of LHS (Latin Hypercube Sampling) for the seven issues weighted as indicated before was used for these calculations. Thus a set of one thousand vectors, with each vector defining the level chosen by LHS for each issue per source term calculation, resulted in 1000 separate sets of release to containment.

Table 7.2 Issue 1: In-Vessel Release From the Fuel (FCOR)

SPECIES/LEVEL	1	2	3	4
I, Cs	0.5	1.0	1.0	1.0
Te	0.026	0.54	0.90	1.0
Sr, Ba	0.6×10^{-2}	1.2×10^{-2}	4.9×10^{-2}	9.8×10^{-2}
Ru-La, Ce	0.55×10^{-7}	0.55×10^{-6}	0.35×10^{-5}	0.28×10^{-4}
Weight	0.24	0.46	0.23	0.07

Table 7.3 Issue 2: Amount of CsI Decomposition (FI_2)

LEVEL	1	2	3	4
VALUE	0.	0.33	0.67	1.0
WEIGHTS	0.2	0.38	0.3	0.12

Table 7.4 Issue 3: In-Vessel Retention (FVES)

(a) High RCS Pressure

SPECIES/LEVEL	1	2	3	4
I, Cs	0.02	0.28	0.5	0.9
Te	0.04	0.48	0.7	0.9
Sr, Ba	0.05	0.21	0.41	0.77
Ru-La, Ce	0.05	0.21	0.4	0.8
Weight	0.11	0.39	0.39	0.11

(b) Low RCS Pressure

SPECIES/LEVEL	1	2	3	4
I, Cs	0.2	0.87	0.95	0.95
Te	0.1	0.83	0.90	0.95
Sr, Ba	0.1	0.77	0.85	0.95
Ru-La, Ce	0.1	0.77	0.85	0.95
Weight	0.11	0.39	0.39	0.11

Table 7.5 Issue 4: Core-Concrete Interaction Releases $\left(\frac{FCCI}{1-FCOR}\right)$

SPECIES/LEVEL	1	2	3
I, Cs	1.0	1.0	1.0
Te	0.015	0.75	1.0
Sr, Ba	0.2×10^{-2}	0.083	0.35
Ru-La, Ce	0.4×10^{-4}	0.16×10^{-2}	0.6×10^{-1}
Weight	0.215	0.57	0.215

Table 7.6 Issue 5: Late Releases of Cesium and Iodine

LEVEL	1	2	3	4
VALUE	0.	0.07	0.23	0.7
WEIGHTS	0.12	0.26	0.40	0.22

Table 7.7 Issue 6: High Pressure Ejection (RADEJL)

SPECIES/LEVEL	1	2	3	4
I, Cs	0.1	0.5	1.0	1.0
Te	0.01	0.1	0.1	1.0
Sr, Ba	10^{-4}	10^{-2}	10^{-2}	10^{-1}
Ru-La, Ce	6.3×10^{-5}	8.8×10^{-4}	6.3×10^{-3}	6.3×10^{-2}
Weight	0.07	0.32	0.58	0.03

Table 7.8 Issue 7: Direct Heating Releases (RADDHL)

SPECIES/LEVEL	1	2	3	4
I, Cs	0.5	1.0	1.0	1.0
Te	0.5	1.0	1.0	1.0
Sr, Ba	10^{-4}	10^{-2}	0.2	0.2
Ru-La, Ce	6.3×10^{-7}	1.6×10^{-2}	6.7×10^{-2}	0.25
Weight	0.14	0.27	0.47	0.12

Figures 7.1 through 7.10 show in detail the results for both high and low RCS pressure conditions. A comparison with the appropriate simplified source term proposed in this report is also shown: arrows on the figures denote the simplified source term developed here.

The results of calculations in terms of arithmatic average, 5th, 50th and 95th percentile over the one thousand performed calculations, with a comparison to the simplified source term are also presented in Tables 7.9 through 7.18.

An additional issue, which has not received much attention, is what effect could steam explosions have on the source term to containment. Several possibilities exist. A massive in-vessel steam explosion, leading to a α-mode containment failure, could conceivably result in an increased fission product release similar to that postulated for HPE. On the other hand, steam explosion(s) might lead to a corium debris configuration which is coolable[48]. The Kouts committee[42] reviewed research in this area. In general, they concluded that the major effort was focussed on the gross thermal hydraulic feature of a catastrophic steam explosion, that might lead to containment failure. Discussion of the impact of steam explosion(s) on the source term were qualitative and did not provide a basis to estimate source term uncertainty.

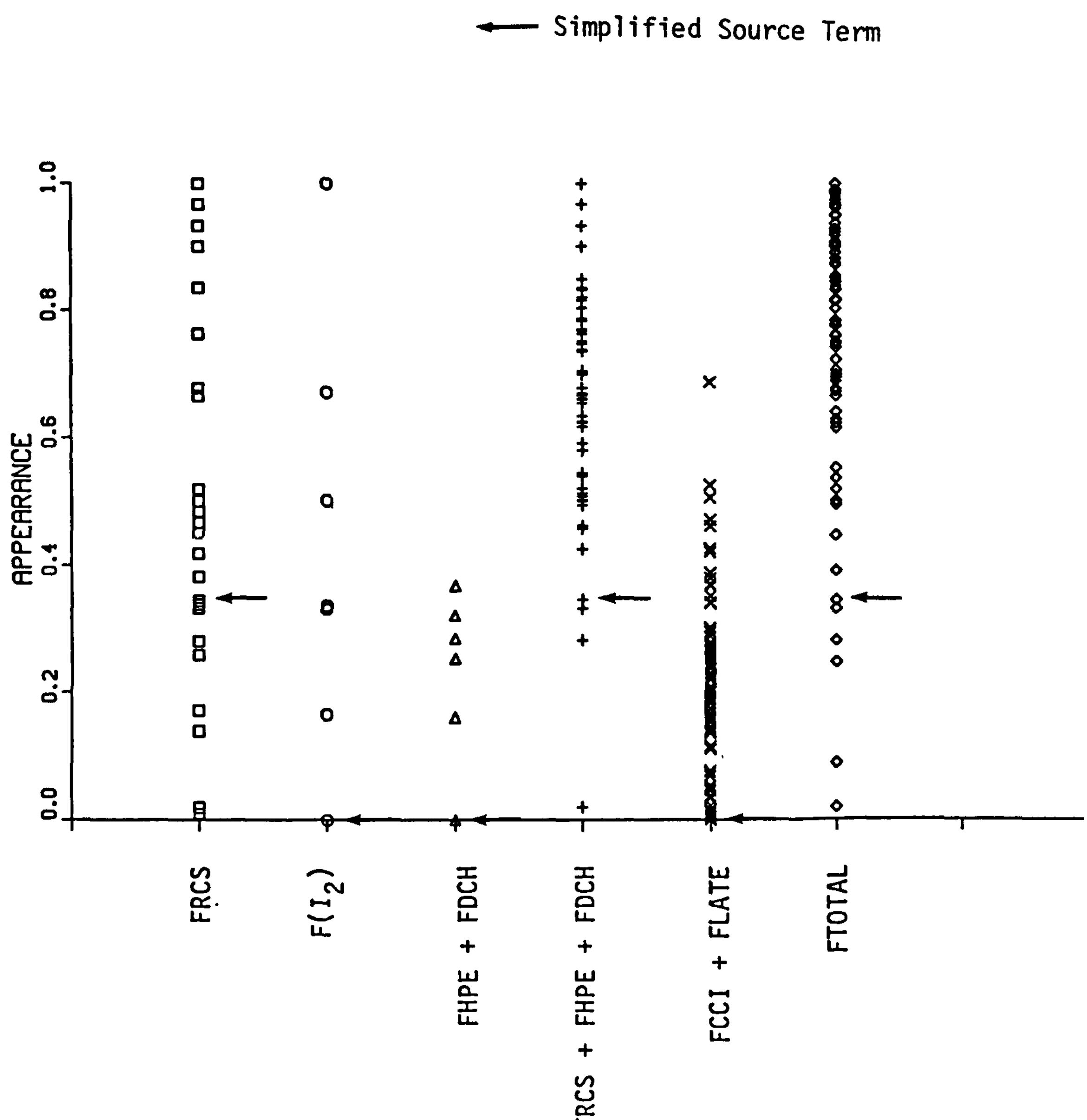

Figure 7.1 Uncertainties in Iodine release fraction to a PWR (Zion) containment compared with simplified source term results (PWR, High RCS Press., Limestone Concrete, Dry Cavity).

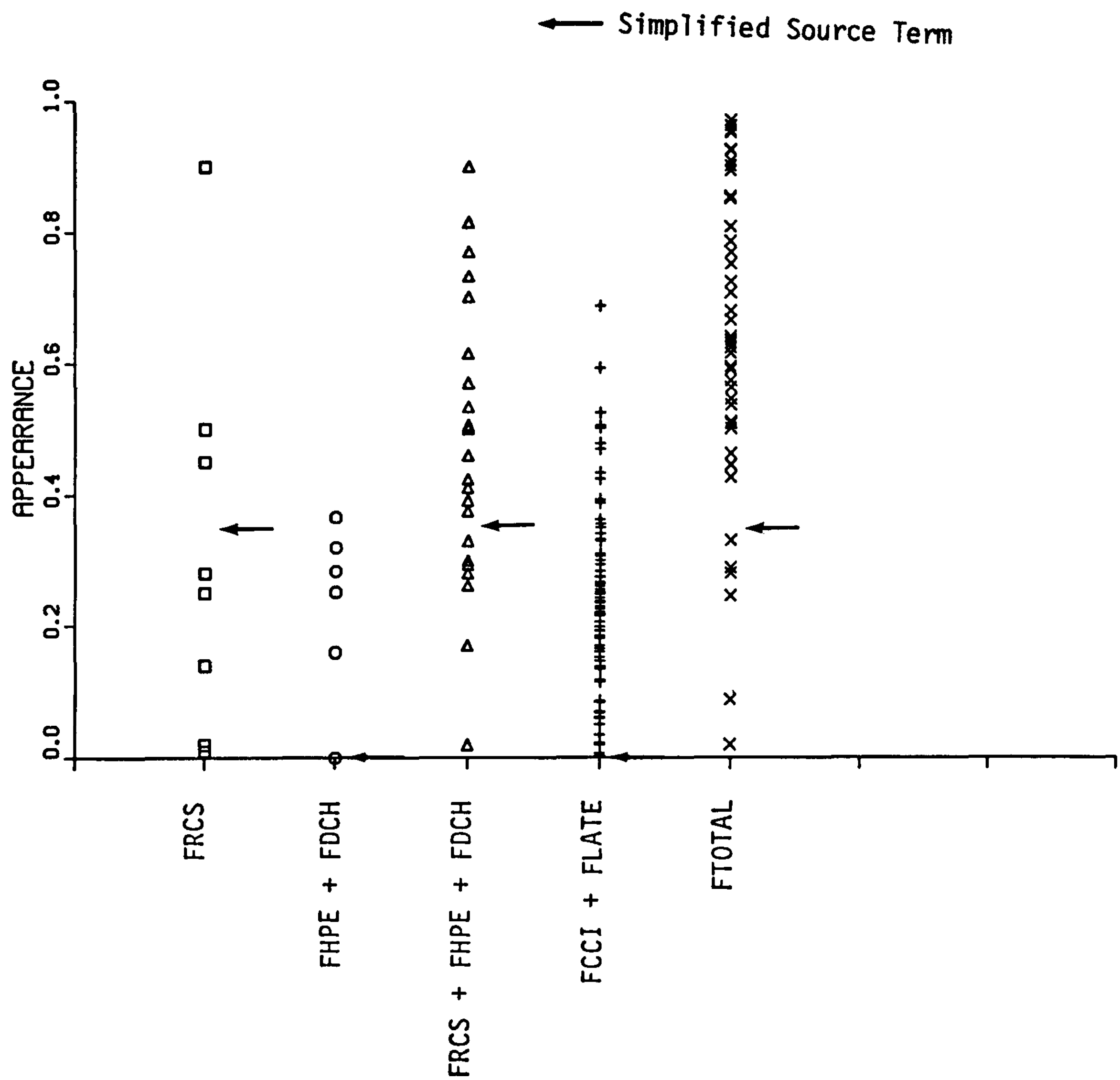

Figure 7.2 Uncertainties in Cesium release fraction to a PWR (Zion) containment compared with simplified source term results (PWR, High RCS Press., Limestone Concrete, Dry Cavity).

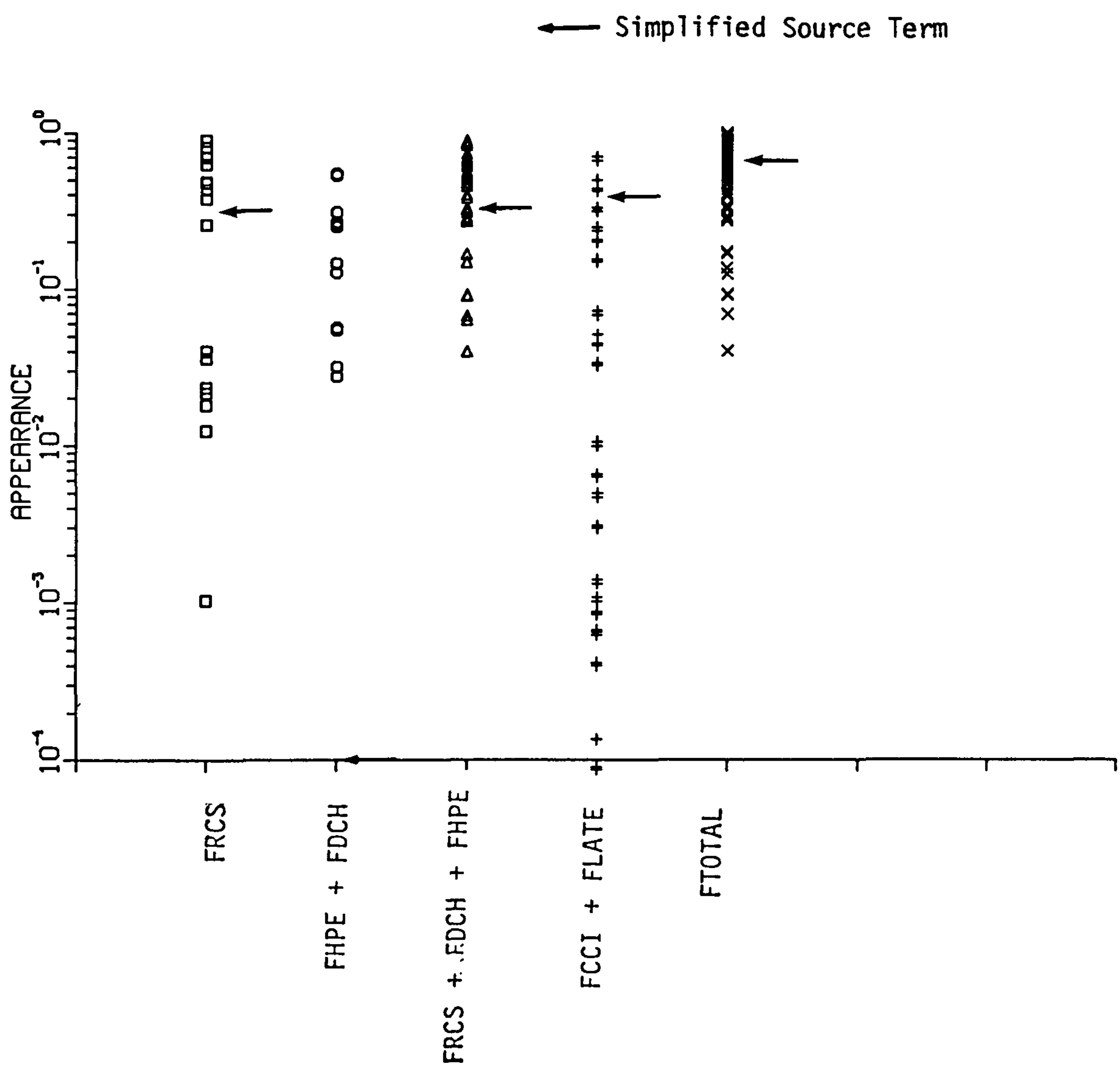

Figure 7.3 Uncertainties in Tellurium release fraction to a PWR (Zion) containment compared with simplified source term results (PWR, High RCS Press., Limestone Concrete, Dry Cavity).

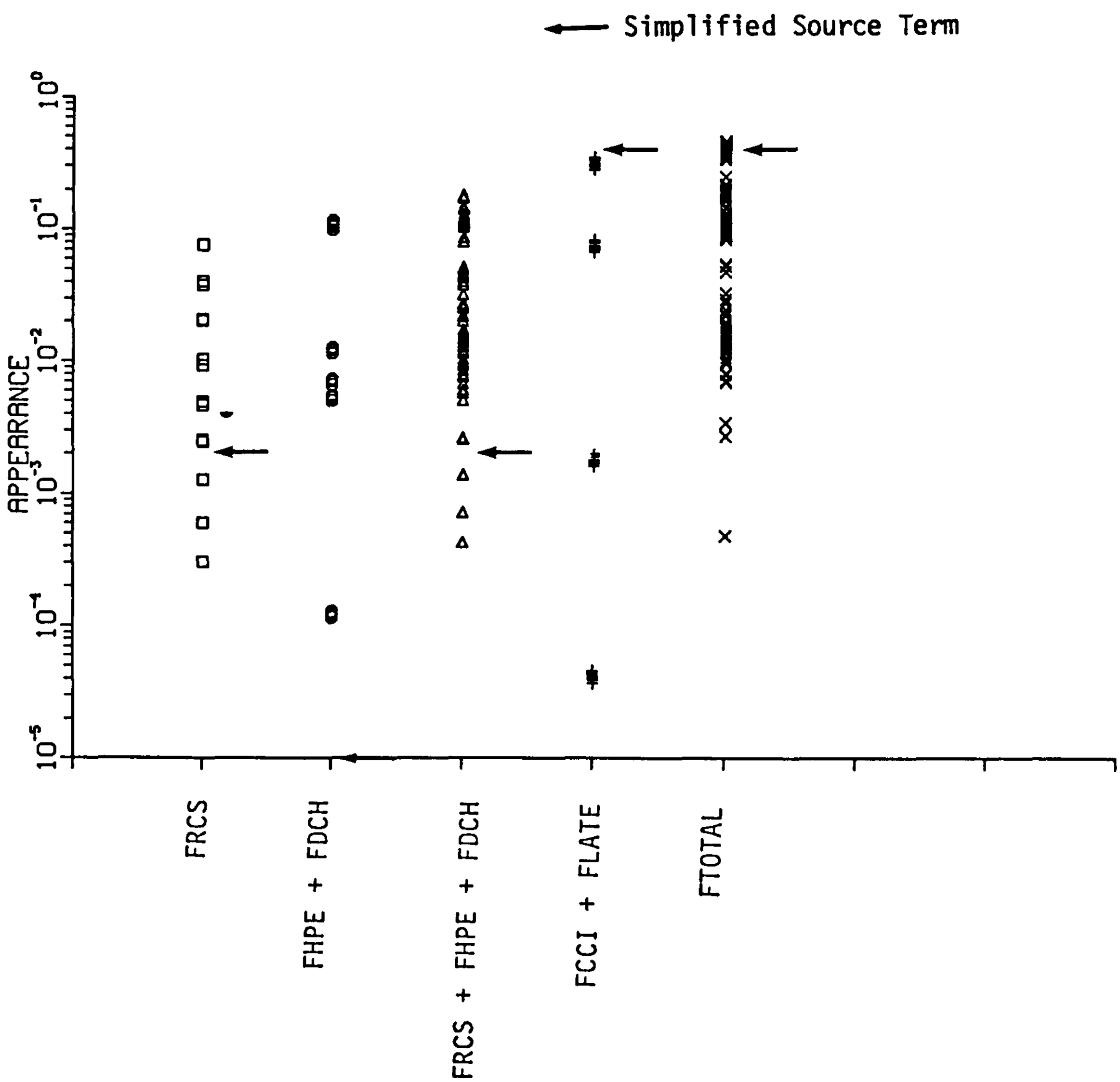

Figure 7.4 Uncertainties in Sr-Ba release fraction to a PWR (Zion) containment compared with simplified source term results (PWR, High RCS Press., Limestone Concrete, Dry Cavity).

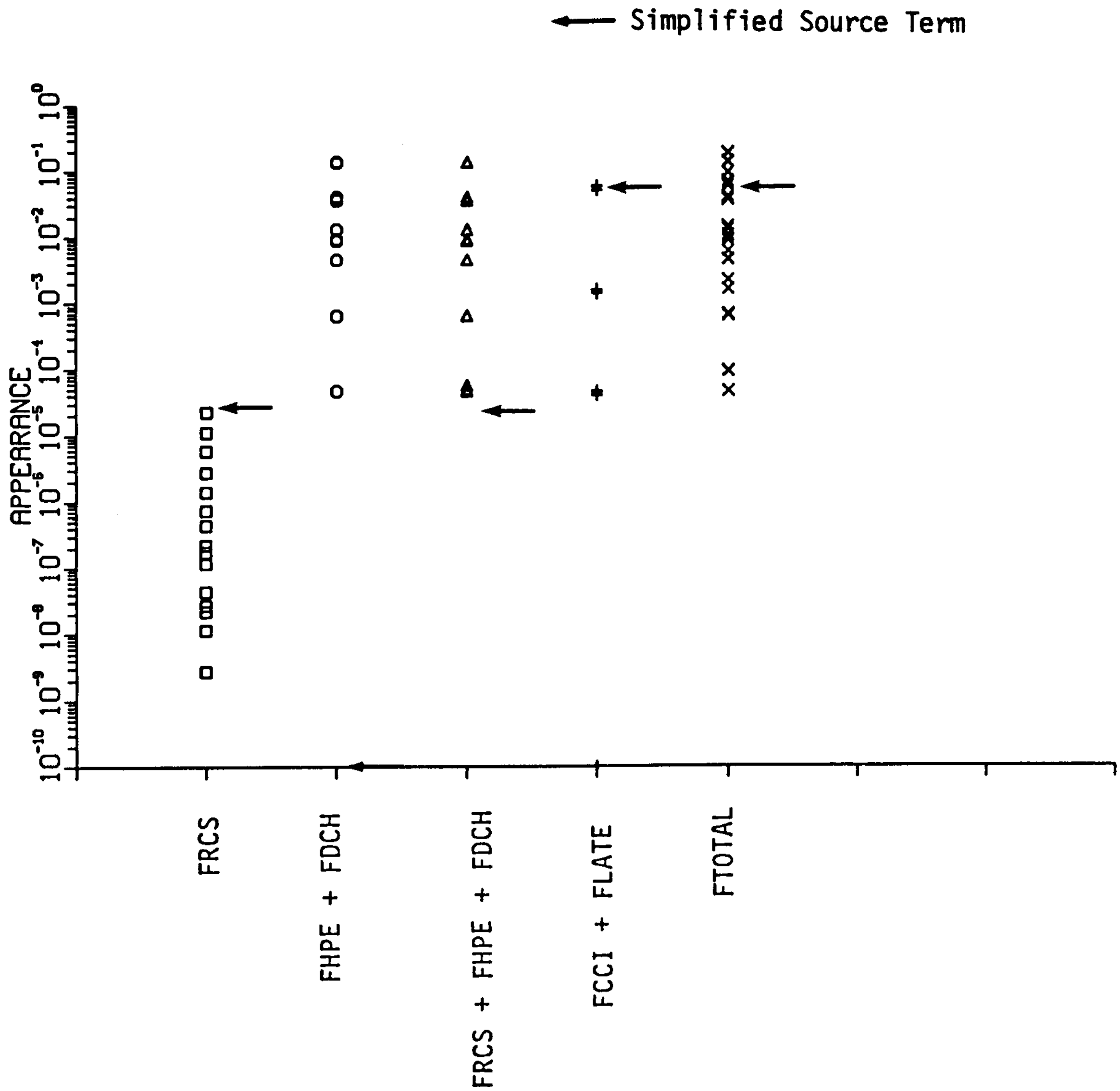

Figure 7.5 Uncertainties in Ru-La-Ce release fraction to a PWR (Zion) containment compared with simplified source term results (PWR, High RCS Press., Limestone Concrete, Dry Cavity).

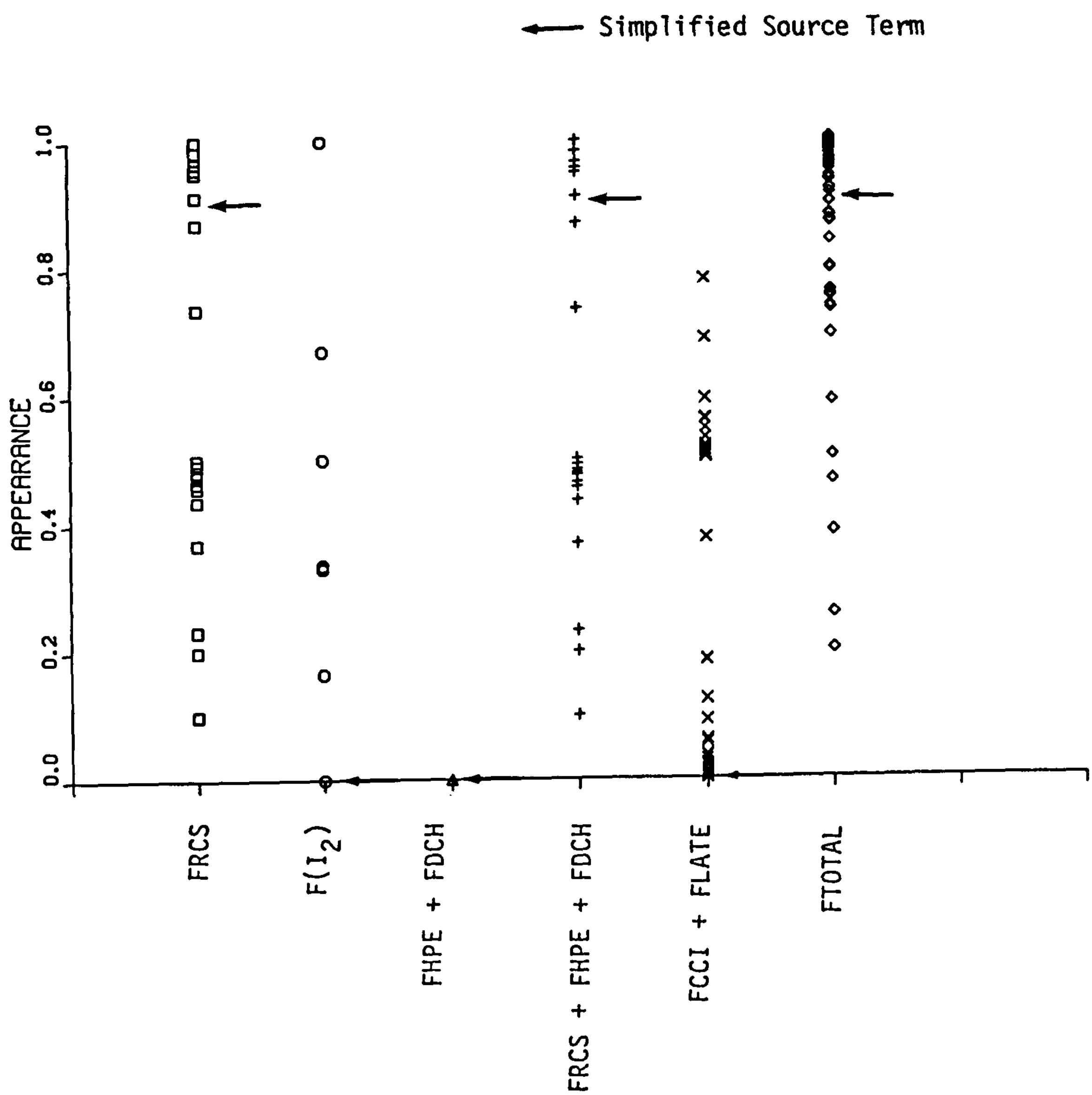

Figure 7.6 Uncertainties in Iodine release fraction to a PWR (Zion) containment compared with simplified source term results (PWR, Low RCS Press., Limestone Concrete, Dry Cavity).

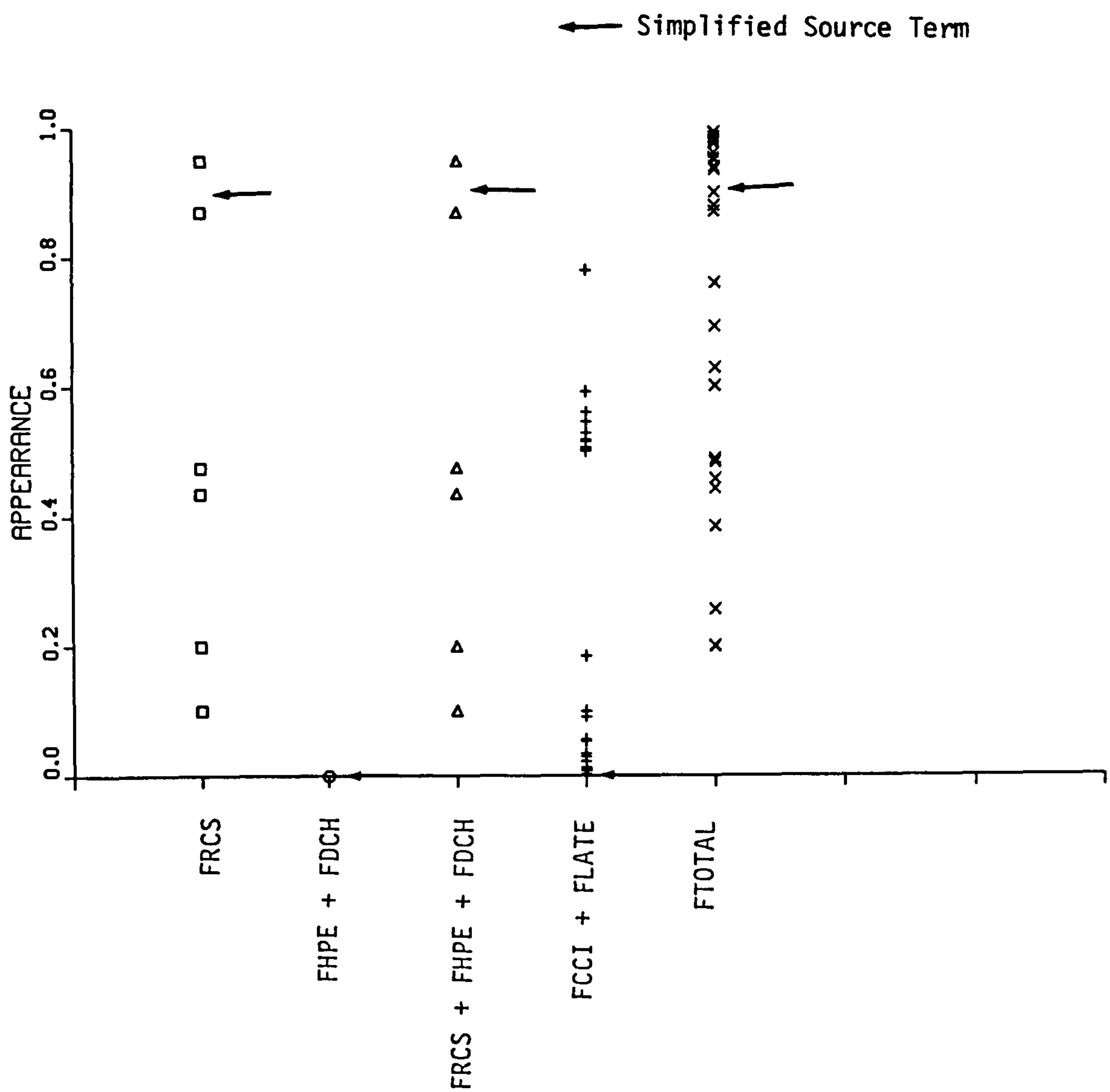

Figure 7.7 Uncertainties in Cesium release fraction to a PWR (Zion) containment compared with simplified source term results (PWR, Low RCS Press., Limestone Concrete, Dry Cavity).

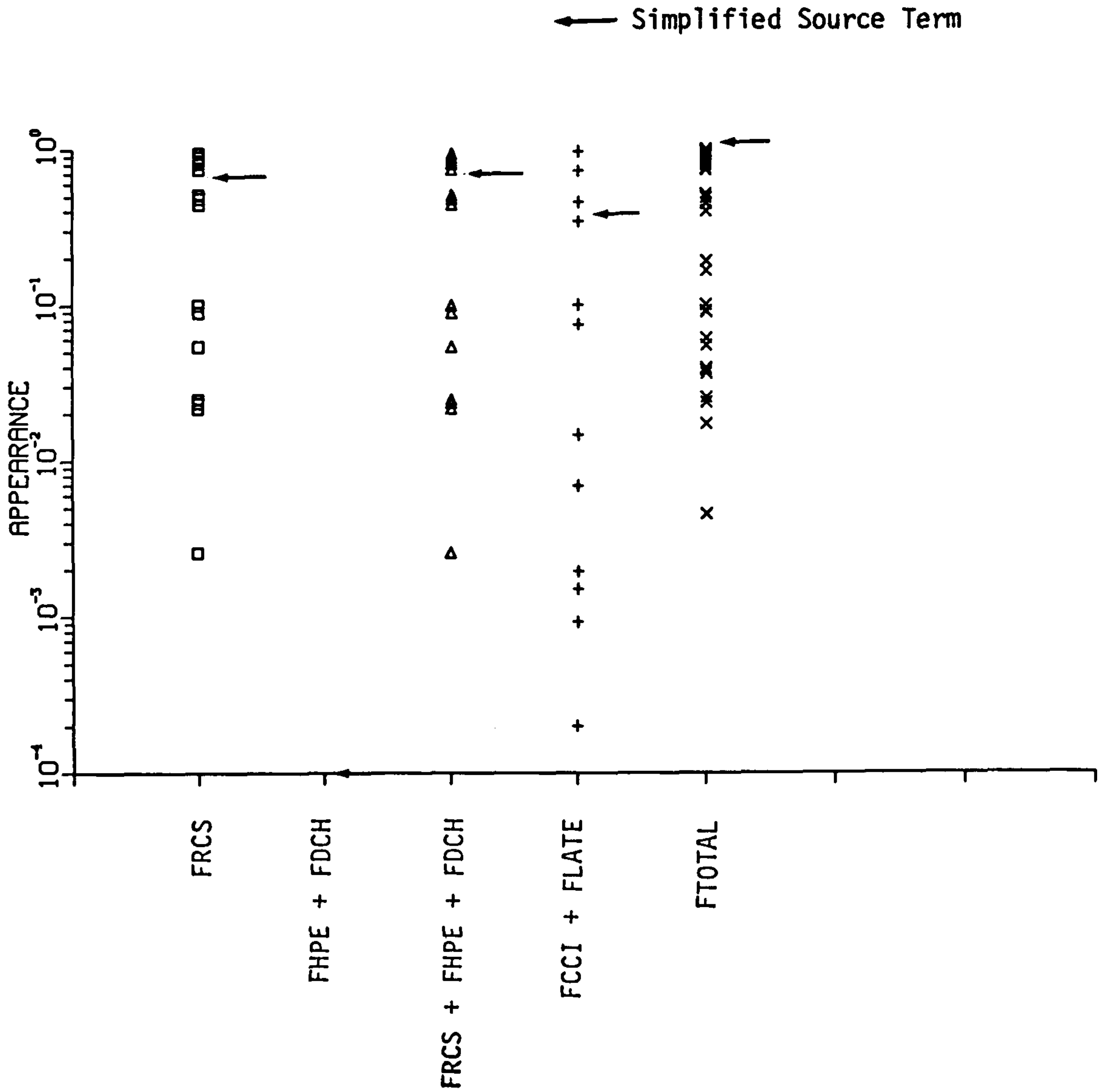

Figure 7.8 Uncertainties in Te release fraction to a PWR (Zion) containment compared with simplified source term results (PWR, Low LRCS Press., Limestone Concrete, Dry Cavity).

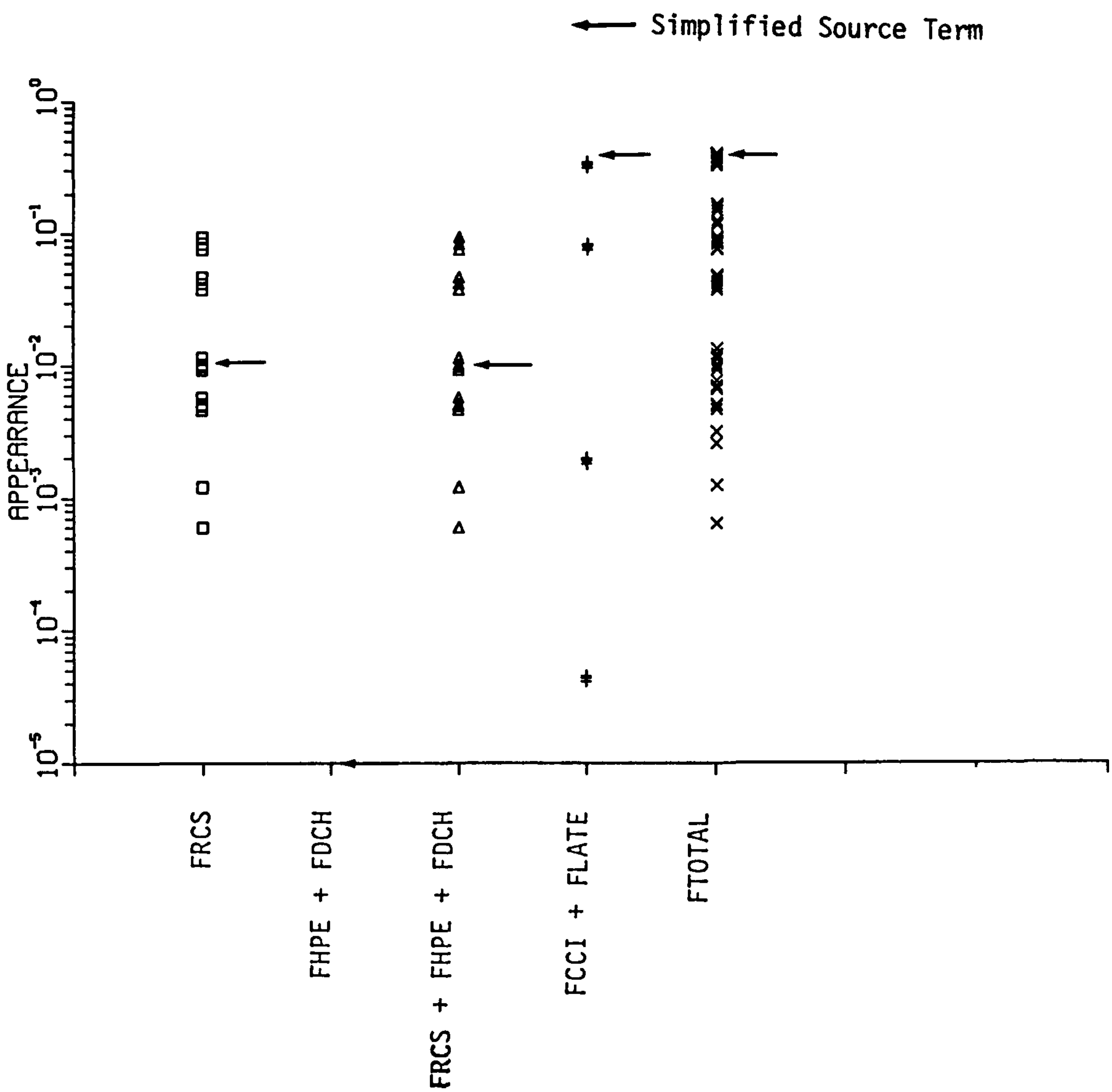

Figure 7.9 Uncertainties in Sr-Ba release fraction to a PWR (Zion) containment compared with simplified source term results (PWR, Low RCS Press., Limestone Concrete, Dry Cavity).

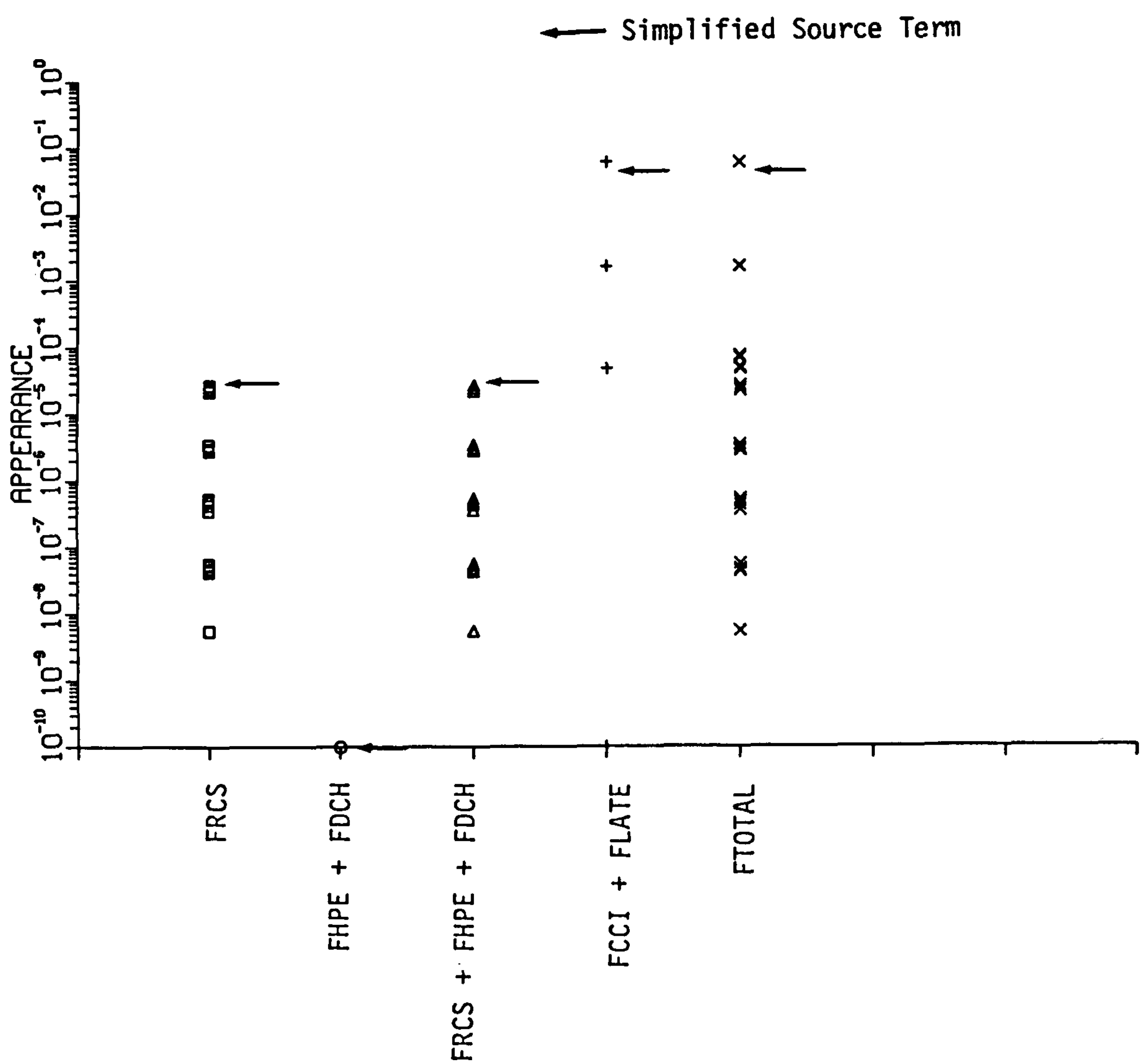

Figure 7.10 Uncertainties in Ru-La-Ce release fraction to a PWR (Zion) containment compared with simplified source term results (PWR, Low RCS Press., Limestone Concrete, Dry Cavity).

Table 7.9 Comparison of Some Statistical Parameters for Iodine Release Into the Zion Containment Using the NUREG-1150 Methodology With the Proposed Simplified Source Term (High RCS Pressure, Dry Cavity, Limestone Concrete)

Parameter	5th	50th	95th	Average	Simplified Source Term
FRCS	0.26	0.66	0.97	0.62	0.35
FI_2	0	0.33	0.67	0.46	0
FHPE + FDCH	0	0	0.32	0.07	0
FRCS + FDCH + FHPE	0.34	0.7	0.97	0.69	0.35
FCCI + FLATE	0	0.05	0.27	0.1	0
FTOTAL	0.5	0.82	0.98	0.8	0.35

Table 7.10 Comparison of Some Statistical Parameters for Cesium Release Into the Zion Containment Using the NUREG-1150 Methodology With the Proposed Simplified Source Term (High RCS Pressure, Dry Cavity, Limestone Concrete)

Parameter	5th	50th	95th	Average	Simplified Source Term
FRCS	0.02	0.28	0.9	0.32	0.35
FHPE + FDCH	0	0	0.4	0.075	0
FRCS + FDCH + FHPE	0.02	0.42	0.9	0.4	0.35
FCCI + FLATE	0	0.14	0.5	0.16	0
FTOTAL	0.09	0.54	0.92	0.56	0.35

Table 7.11 Comparison of Some Statistical Parameters for Tellurium Release Into the Zion Containment Using the NUREG-1150 Methodology With the Proposed Simplified Source Term (High RCS Pressure, Dry Cavity, Limestone Concrete)

Parameter	5th	50th	95th	Average	Simplified Source Term
FRCS	0.01	0.26	0.7	0.27	0.3
FHPE + FDCH	0	0.26	0.55	0.25	0
FRCS + FDCH + FHPE	0.09	0.54	0.75	0.52	0.3
FCCI	0	0.15	0.42	0.15	0.35
FTOTAL	0.14	0.72	0.98	0.67	0.65

Table 7.12 Comparison of Some Statistical Parameters for Sr-Ba Release Into the Zion Containment Using the NUREG-1150 Methodology With the Proposed Simplified Source Term (High RCS Pressure, Dry Cavity, Limestone Concrete)

Parameter	5th	50th	95th	Average	Simplified Source Term
FRCS	$6x10^{-4}$	$2.5x10^{-3}$	$3.7x10^{-2}$	$7.7x10^{-3}$	$2x10^{-3}$
FHPE + FDCH	$5.5x10^{-3}$	$1.08x10^{-1}$	$1.15x10^{-1}$	$7.27x10^{-2}$	0
FRCS + FDCH + FHPE	$7.8x10^{-3}$	$1.14x10^{-1}$	$1.3x10^{-1}$	$8.04x10^{-2}$	$2x10^{-3}$
FCCI	$4.02x10^{-5}$	$7.8x10^{-2}$	$3.4x10^{-1}$	$1.53x10^{-1}$	0.4
FTOTAL	$1.75x10^{-2}$	$1.9x10^{-1}$	$4.25x10^{-1}$	$2.33x10^{-1}$	0.4

Table 7.13 Comparison of Some Statistical Parameters for Ru-La-Ce Release Into the Zion Containment Using the NUREG-1150 Methodology With the Proposed Simplified Source Term (High RCS Pressure, Dry Cavity, Limestone Concrete)

Parameter	5th	50th	95th	Average	Simplified Source Term
FRCS	1.15×10^{-8}	1.15×10^{-7}	5.88×10^{-6}	9.4×10^{-7}	3×10^{-5}
FHPE + FDCH	6.42×10^{-4}	3.73×10^{-2}	1.37×10^{-1}	4.8×10^{-2}	0
FRCS + FDCH + FHPE	6.42×10^{-4}	3.73×10^{-2}	1.37×10^{-1}	4.8×10^{-2}	3×10^{-5}
FCCI	0	1.58×10^{-3}	5.97×10^{-2}	2.28×10^{-2}	0.05
FTOTAL	2.24×10^{-3}	6.06×10^{-2}	1.89×6^{-1}	7.12×10^{-2}	0.05

Table 7.14 Comparison of Some Statistical Parameters for Iodine Release Into the Zion Containment Using the NUREG-1150 Methodology With the Proposed Simplified Source Term (Low RCS Pressure, Dry Cavity, Limestone Concrete)

Parameter	5th	50th	95th	Average	Simplified Source Term
FRCS	0.37	0.91	0.98	0.78	0.9
$F(I_2)$	0	0.335	0.67	0.46	0
FCCI + FLATE	0	0.0115	0.515	0.14	0
FTOTAL	0.58	0.97	0.99	0.92	0.9

Table 7.15 Comparison of Some Statistical Parameters for Cesium Release Into the Zion Containment Using the NUREG-1150 Methodology With the Proposed Simplified Source Term (Low RCS Pressure, Dry Cavity, Limestone Concrete)

Parameter	5th	50th	95th	Average	Simplified Source Term
FRCS	0.2	0.87	0.95	0.68	0.9
FCCI + FLATE	0	0.03	0.54	0.15	0
FTOTAL	0.26	0.94	0.98	0.83	0.9

Table 7.16 Comparison of Some Statistical Parameters for Tellurium Release Into the Zion Containment Using the NUREG-1150 Methodology With the Proposed Simplified Source Term (Low RCS Pressure, Dry Cavity, Limestone Concrete)

Parameter	5th	50th	95th	Average	Simplified Source Term
FRCS	0.022	0.45	0.85	0.38	0.65
FCCI + FLATE	0	0.34	0.97	0.32	0.35
FTOTAL	0.039	0.82	0.99	0.7	1.0

Table 7.17 Comparison of Some Statistical Parameters for Sr-Ba Release Into the Zion Containment Using the NUREG-1150 Methodology With the Proposed Simplified Source Term (Low RCS Pressure, Dry Cavity, Limestone Concrete)

Parameter	5th	50th	95th	Average	Simplified Source Term
FRCS	1.2×10^{-3}	9.2×10^{-3}	0.075	0.017	0.01
FCCI	4.5×10^{-5}	0.082	0.35	0.16	0.4
FTOTAL	6.6×10^{-3}	0.093	0.37	0.18	0.41

Table 7.18 Comparison of Some Statistical Parameters for Ru-La-Ce Release Into the Zion Containment Using the NUREG-1150 Methodology With the Proposed Simplified Source Term (Low RCS Pressure, Dry Cavity, Limestone Concrete)

Parameter	5th	50th	95th	Average	Simplified Source Term
FRCS	4.2×10^{-8}	4.2×10^{-7}	2.16×10^{-5}	2.07×10^{-6}	3×10^{-5}
FCCI	0	1.6×10^{-3}	0.06	0.024	0.05
FTOTAL	4.2×10^{-7}	1.6×10^{-3}	0.06	0.024	0.05

8. SUMMARY AND CONCLUSIONS

A detailed review of the radiological release estimates for light water reactors was provided as a technical basis for development of a generic simplified source term approach possible for use in LWR licensing.

The simplified in-vessel and ex-vessel fission product release and transport characteristics were specified for each unique combination of reactor coolant and containment system conditions.

The methodology was designed to apply to a spectrum of accidents including Design Basis Accidents (DBAs) and severe accidents.

The methodology developed in the present study is limited to prediction of radiological source term behavior. However, a comparable simple approach for determination of reactor coolant and containment system thermal-hydraulic behavior could also be developed.

In chapter 7, a summary of outstanding issues which contribute to the level of source term uncertainty was presented and the proposed simplified source term estimates developed in this report were compared to the estimated range of Draft NUREG-1150 results developed for similar accident sequences. Clearly large uncertainties exist, and these will not be resolved in the near term.

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13. ABSTRACT (200 words or less)

A detailed review of the radiological release estimates for light water reactor accident sequences is presented as a basis for development of a simplified approach for prediction of characteristics of radiological releases into containments under design basis and severe accident conditions for both Pressurized Water Reactors (PWRs) and Boiling Water Reactors (BWRs). Resulting source term estimates are also compared with parallel results using the Source Term Code Package (STCP) methodology.

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