Historical Perspectives and Insights on Reactor Consequence Analyses

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ABSTRACT

This paper has been prepared for use by the NRC Advisory Committee on Reactor Safeguards (ACRS) in its ongoing review of the State-Of-the-Art Reactor Consequence Analyses (SOARCA) Project. Major contributions to consequence assessment have been summarized to provide insights and historical perspectives on previous state-of-the-art analyses of the consequences of severe reactor accidents. The feasibility of using a simplified approach for updating results from earlier Level-3 probabilistic risk assessments (PRAs) such as the NUREG-1150 Study for comparison with aspects of SOARCA results has also been discussed.

The views expressed in this paper are solely those of the author and do not necessarily represent the views of the ACRS.

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ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
AEC	Atomic Energy Commission
ATWS	Anticipated Transient without Scram
BNL	Brookhaven National Laboratory
BWR	Boiling Water Reactor
CCDF	Complementary Cumulative Distribution Function
CCI	Core-Concrete Interaction
CF	Containment Failure
CV	Containment Venting
DBA	Design Basis Accident
DCH	Direct containment Heating
ECF	Early Containment Failure
EDO	Executive Director for Operations
EPRI	Electric Power Research Institute
ESF	Engineered Safety Feature
IPE	Individual Plant Examination
LCF	Late Containment Failure
LLNL	Lawrence Livermore National Laboratory
LOCA	Loss-of-Coolant Accident
LOSP	Loss of Offsite Power
LWR	Light Water Reactor
MACCS	MELCOR Accident Consequence Code System
NCF	No Containment Failure
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
PDS	Plant Damage State
PRA	Probabilistic Risk assessment
PWR	Pressurized Water Reactor
QHOs	Quantitative Health Objectives
RCS	Reactor Coolant System
RPV	Reactor pressure Vessel
RST	Representative Source Term
SAMGs	Severe Accident Management Guidelines
SBO	Station Blackout
SGTR	Steam Generator Tube Rupture
SNL	Sandia National Laboratories
SOARCA	State-Of-the-Art Reactor Consequence Analyses
SRM	Staff Requirements Memorandum
SRV	Safety Relief Valve
SST	Siting Source Term
STCP	Source Term Code Package
VB	Vessel Breach
WWF	Wet Well Failure

1 INTRODUCTION

The probability and offsite consequences of severe reactor accidents have been the subject of considerable interest and study since the earliest days of reactor development.

The first estimates of consequences of severe accidents were published in the 1957 U.S. Atomic Energy Commission report (WASH-740) [1], "Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants." This study was an attempt to provide upper bounds of the potential public hazards resulting from certain severe hypothetical accidents. Conservative values were used for many factors influencing the magnitude of the estimated accident consequences. At the time, the technology and the state-of-knowledge of severe accidents had not progressed to the point where it was possible to use quantitative techniques to estimate the probabilities of such accidents. However, there was a general agreement that the probability of occurrence of severe accidents in nuclear power reactors was exceedingly low. The following is quoted from the March 22, 1957 letter, from Harold S. Vance, Acting Chairman, Atomic Energy Commission, to Carl T. Durham, Chairman, Joint Committee on Atomic Energy, Congress of the United States, transmitting an advance copy of the WASH-740 report.

> *"As to the probabilities of major reactor accidents, some experts held that numerical estimates of a quantity so vague and uncertain as the likelihood of occurrence of major reactor accidents have no meaning. They declined to express their feeling about this probability in numbers. Others, though admitting similar uncertainty, nevertheless ventured to express their opinions in numerical terms. Estimations so expressed of the probability of reactor accidents having major effects on the public ranged from a chance of one in 100,000 to one in a billion per year for each large reactor. However, whether numerically expressed or not, there was no disagreement in the opinion that the probability of major reactor accidents is exceedingly low.*
>
> *…...To reduce the matter of assumed hazards to comparative numbers, let us take the most pessimistic assumptions used and apply them to a case of 100 power reactors in operation in the United States. Under these assumptions, the chances of a person being killed in any year by a reactor accident would be less than one in 50 million. By contrast, the present odds of being killed in any year by an automobile accident in the United States stands at about one in 5,000."*

Since the publication of WASH-740 report, several systematic studies have been made to search out a large spectrum of accidents and to use quantitative techniques to estimate the probabilities, radionuclide release characteristics (source terms), and potential offsite health consequences. Such studies include the NRC's WASH-1400 (1975) [2] and NUREG-1150 (1990) [3], as well as industry-sponsored probabilistic risk assessments (PRAs) such as those for Zion (1981) [4], Indian Point (1982) [5], Millstone 3 (1983) [6], and Seabrook (1983) [7].

In 1982, Sandia National Laboratories (SNL) performed a study of technical aspects of siting for nuclear power reactors. The results of this study, also known as Sandia Siting Study, were published in NUREG/CR-2239 [8], "Technical Guidance for Siting Criteria Development." This study used five generic source terms for analyzing the consequences and socio-economic impacts of possible plant accidents at 91 existing or proposed reactor sites. These source terms were derived from the Reactor Safety Study (WASH-1400) and subsequent evaluations.

Since the publication of the Sandia Siting Study, many events have brought a new focus to this study and its results. Despite accepted arguments that the results of this study are overly conservative and do not reflect current state-of-the-art in evaluating severe accident progression and offsite consequences, the results, in terms of predicted offsite early fatalities and latent cancers, have often been quoted by outside organizations to illustrate the potential consequences of a severe accident at a commercial nuclear power plant.

The NRC staff is currently implementing its plan for developing state-of-the-art reactor consequence analyses [9]. This work will: (1) evaluate and update, as appropriate, analytical methods and models for realistic evaluation of severe accident progression and offsite consequences; (2) develop state-of-the-art reactor consequence assessments of severe accidents; and (3) identify mitigative measures that have the potential to significantly reduce risk or offsite consequences. These analyses include external events; consideration of all mitigative measures, including the newly required extreme damage state mitigative guidelines (B.5.b); state-of-the-art accident progression modeling, based on 25 years of research, to provide a best estimate for accident progression, containment performance, time of release, and fission product behavior; more realistic offsite dispersion modeling; and site-specific evaluation of public evacuation based on updated emergency plans.

In a Staff Requirements Memorandum (SRM) dated April 14, 2006 [10], the Commission stated that the staff's proposal to examine significant radiological release scenarios having estimated likelihoods of one in a million or greater per year is an appropriate initial focus. The Commission also stated that "in applying a screening radiological release frequency of 10^{-6} per reactor year, the staff should be careful to define release groupings such that release characteristics are representative of scenarios binned into those groups. However, where possible, the groups should also be sufficiently broad to be able to include the potentially risk-significant but lower-frequency scenarios (for example, the interfacing systems LOCA scenarios that bypass the containment)."

In the April 14, 2006 SRM [10], the Commission specifically instructed the staff to "work with the ACRS on technical issues such as identification of accident scenarios to be evaluated, evaluation of source terms, credit for operator actions or plant mitigation systems, modeling of emergency preparedness, modeling of offsite consequences, and definition and characterization of analysis uncertainty."

In an April 2, 2007 SRM [11], the Commission directed the staff to "reduce the initial scope of this effort [SOARCA] to not more than eight plants representing a spectrum of plant vendors and technologies." The Commission also directed the staff to "conduct the first assessments on a subset of the eight plants, for example a selected BWR and PWR plant, in order to resolve issues associated with the integration of methods and resolve details associated with simulation of plant systems and procedures."

In its February 25, 2008 report to the Commission concerning the SOARCA Project [12], the ACRS recommended that "as a minimum, a limited set of updated Level-3 PRAs for the SOARCA pilot plants be performed to benchmark the consequence analyses and provide useful information to the Commission in deciding whether to proceed with a full set of consequence analyses." The ACRS further noted that "examination of the Level-3 PRA results for the SOARCA pilot plants may identify suitable Level-1 event scenario screening criteria and simplifying assumptions that could be used to develop a defensible, simplified approach."

In a letter dated April 7, 2008 [13], the Executive Director for Operations (EDO) responded to the ACRS report of February 25, 2008 on SOARCA, indicating that the staff did not agree with the ACRS recommendation that a limited set of Level-3 PRAs be performed to benchmark the SOARCA approach developed by the staff. In its April 21, 2008 response to the EDO [14], the ACRS noted that the Committee continues "to believe that the credibility of the SOARCA Project cannot rely on confidence in the judgment of the staff and on a novel analysis procedure that differs substantially from previous state-of-the-art analyses of the consequences of severe reactor accidents." The ACRS further noted that "without including benchmark analyses similar in scope, it will be difficult to demonstrate convincingly that reductions in consequences that might be indicated by the SOARCA results reflect the impact of enhancements in plant design and operation, and improvements in calculation methods for accident progression and consequence analysis, rather than changes in the scope of the calculation."

In a June 26, 2008 SRM [15], resulting from the June 5, 2008 meeting with the ACRS, the Commission directed the staff to "continue working to address Committee concerns, such as with SOARCA, … and, as necessary and appropriate, provide timely policy decision papers to the Commission to resolve any disagreements."

This report has been prepared for use by the ACRS in its continued dialogue with the staff regarding the feasibility of using a simplified, yet systematic and defensible, approach to update results from earlier Level-3 PRAs such as the NUREG-1150 Study for comparison with aspects of SOARCA results.

The report begins with an overview of major contributions to consequence assessment to provide historical perspectives and insights on previous state-of-the-art analyses of the consequences of severe reactor accidents. It then discusses how the results and insights from NUREG-1150 study and integrated risk assessment for La Salle together with recent advances in understanding of severe accident phenomenology and containment failure mechanisms could be used to update the results of such earlier Level-3 PRAs for comparison with aspects of SOARCA results.

2. MAJOR NRC-SPONSORED ASSESSMENTS OF REACTOR ACCIDENT CONSEQUENCES

2.1 Reactor Safety Study (WASH-1400)

The Reactor Safety Study (WASH-1400) [2], was the first systematic attempt to provide realistic estimates of risk to the public from potential accidents in commercial nuclear power plants. This 1975 study included 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 safety systems. Detailed investigations were performed to 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 were placed.

Two specific reactor designs were analyzed in WASH-1400: Peach Bottom Atomic Power Station, a boiling water reactor (BWR) with a Mark I containment and Surry, a 3-loop pressurized water reactor (PWR) with a subatmospheric containment. Nine PWR release categories and five BWR release categories were developed in the Reactor Safety Study. Each release category was represented by its release frequency and several other parameters that described the radionuclide release characteristics (source term). Table 1 presents a summary of release categories defined in the Reactor Safety Study. These release categories represent the full spectrum of possible accident progression and containment failure modes based on the state of knowledge and understanding of severe accidents at that time. For example, the dominant contributing sequence to PWR release category 1 is a transient event involving loss of offsite AC power with a failure to recover either onsite or offsite AC power within about 3 hours. This sequence considers that an in-vessel steam explosion ("alpha mode" failure) as the early containment failure mode (see Chapter 3 of this report for a more recent reassessment of this issue). It was also assumed that during steam explosion, the volatile fission products would be finely dispersed and air oxidation would enhance the magnitude of the radioactivity releases from the failed containment.

The Reactor Safety Study provided insights into expected individual risk from characteristic types of releases under average population and meteorological conditions at the 68 sites at which the first 100 reactors expected to be operating by about 1980. The risk was presented by the probability and magnitude of seven different consequence measures. These consequence measures were early fatalities (death within approximately one year after a potential accident), early illnesses, thyroid nodules, latent cancer fatalities, genetic effects, land contamination, and property damage costs. A major conclusion of the Reactor Safety Study was that the low probability-high consequence accidents involving core meltdown, containment failure, and failure of engineered safety features dominated the risk to public.

Figures 1 and 2 show the likelihood and number of fatalities from both nuclear and a variety of non-nuclear accidents, reported in the Executive Summary of

WASH-1400 report. These results indicated that non-nuclear events were about 10,000 times more likely to produce large numbers of fatalities than nuclear plants. It should be noted that the societal risks are stated as complementary cumulative distribution functions per year in Figures 1 and 2. The complementary cumulative distribution function (CCDF) shows the frequency that a consequence will exceed a given magnitude. Also note that the fatalities shown in Figures 1 and 2 are those that would be predicted to occur within a short period of time after the potential reactor accident. This was done to provide a consistent comparison to the non-nuclear events that also cause fatalities in the same time frame.

The WASH-1400 report stimulated a great deal of debate after its release. In June 1976, the Committee on Interior and Insular Affairs of the U.S. House of Representatives, chaired by Representative Morris Udall, held hearings on the findings of the Reactor Safety Study. These hearings found that the Reactor Safety Study seemed to be misleading in the certainty and comprehensiveness of its conclusions [16]. Rep. Udall suggested that an outside review panel be formed to take a closer look at how the study arrived at its conclusions [16]. The NRC then asked Dr Harold Lewis of the University of California-Santa Barbara to chair an independent review group, which produced what is now known as the “Lewis report” [17].

The Lewis report concluded that the WASH-1400 study was overall a “conscientious and honest effort", an “important advance” over earlier quantitative analysis of reactor safety, and with a “sound methodology” that should be used more widely by the NRC. Among the shortcomings that the Lewis Committee identified in the Reactor Safety Study was the lack of scrutability of the calculation/analysis process. The Lewis report was particularly critical of the Executive Summary of the WASH-1400 report for being “a poor description of the contents of the report” and for not adequately indicating the full extent of the consequences of, and the uncertainties in the probabilities of, reactor accidents. For this reason, the NRC withdrew its endorsement of the Executive Summary although it did not repudiate the study itself.

Table 1 Summary of Release Categories in Reactor Safety Study
(WASH-1400, NUREG-75/014, p. v-4) [2]

RELEASE CATEGORY	PROBABILITY per Reactor-Yr	TIME OF RELEASE (Hr)	DURATION OF RELEASE (Hr)	WARNING TIME FOR EVACUATION (Hr)	ELEVATION OF RELEASE (Meters)	CONTAINMENT ENERGY RELEASE (10^6 Btu/Hr)	FRACTION OF CORE INVENTORY RELEASED[a] Xe-Kr	Org. I	I	Cs-Rb	Te-Sb	Ba-Sr	Ru[b]	La[c]
PWR 1	$9x10^{-7}$	2.5	0.5	1.0	25	520[d]	0.9	$6x10^{-3}$	0.7	0.4	0.4	0.05	0.4	$3x10^{-3}$
PWR 2	$8x10^{-6}$	2.5	0.5	1.0	0	170	0.9	$7x10^{-3}$	0.7	0.5	0.3	0.06	0.02	$4x10^{-3}$
PWR 3	$4x10^{-6}$	5.0	1.5	2.0	0	6	0.8	$6x10^{-3}$	0.2	0.2	0.3	0.02	0.03	$3x10^{-3}$
PWR 4	$5x10^{-7}$	2.0	3.0	2.0	0	1	0.6	$2x10^{-3}$	0.09	0.04	0.03	$5x10^{-3}$	$3x10^{-3}$	$4x10^{-4}$
PWR 5	$7x10^{-7}$	2.0	4.0	1.0	0	0.3	0.3	$2x10^{-3}$	0.03	$9x10^{-3}$	$5x10^{-3}$	$1x10^{-3}$	$6x10^{-4}$	$7x10^{-5}$
PWR 6	$6x10^{-6}$	12.0	10.0	1.0	0	N/A	0.3	$2x10^{-3}$	$8x10^{-4}$	$8x10^{-4}$	$1x10^{-3}$	$9x10^{-5}$	$7x10^{-5}$	$1x10^{-5}$
PWR 7	$4x10^{-5}$	10.0	10.0	1.0	0	N/A	$6x10^{-3}$	$2x10^{-5}$	$2x10^{-5}$	$1x10^{-5}$	$2x10^{-5}$	$1x10^{-6}$	$1x10^{-6}$	$2x10^{-7}$
PWR 8	$4x10^{-5}$	0.5	0.5	N/A	0	N/A	$2x10^{-3}$	$5x10^{-6}$	$1x10^{-4}$	$5x10^{-4}$	$1x10^{-6}$	$1x10^{-8}$	0	0
PWR 9	$4x10^{-4}$	0.5	0.5	N/A	0	N/A	$3x10^{-6}$	$7x10^{-9}$	$1x10^{-7}$	$6x10^{-7}$	$1x10^{-9}$	$1x10^{-11}$	0	0
BWR 1	$1x10^{-6}$	2.0	2.0	1.5	25	130	1.0	$7x10^{-3}$	0.40	0.40	0.70	0.05	0.5	$5x10^{-3}$
BWR 2	$6x10^{-6}$	30.0	3.0	2.0	0	30	1.0	$7x10^{-3}$	0.90	0.50	0.30	0.10	0.03	$4x10^{-3}$
BWR 3	$2x10^{-5}$	30.0	3.0	2.0	25	20	1.0	$7x10^{-3}$	0.10	0.10	0.30	0.01	0.02	$3x10^{-3}$
BWR 4	$2x10^{-6}$	5.0	2.0	2.0	25	N/A	0.6	$7x10^{-4}$	$8x10^{-4}$	$5x10^{-3}$	$4x10^{-3}$	$6x10^{-4}$	$6x10^{-4}$	$1x10^{-4}$
BWR 5	$1x10^{-4}$	3.5	5.0	N/A	150	N/A	$5x10^{-4}$	$2x10^{-9}$	$6x10^{-11}$	$4x10^{-9}$	$8x10^{-12}$	$8x10^{-14}$	0	0

(a) A discussion of the isotopes used in the study is found in Appendix VI. Background on the isotope groups and release mechanisms is found in Appendix VII.

(b) Includes Mo, Rh, Tc, Co.

(c) Includes Nd, Y, Ce, Pr, La, Nb, Am, Cm, Pu, Np, Zr.

(d) A lower energy release rate than this value applies to part of the period over which the radioactivity is being released. The effect of lower energy release rates on consequences is found in Appendix VI.

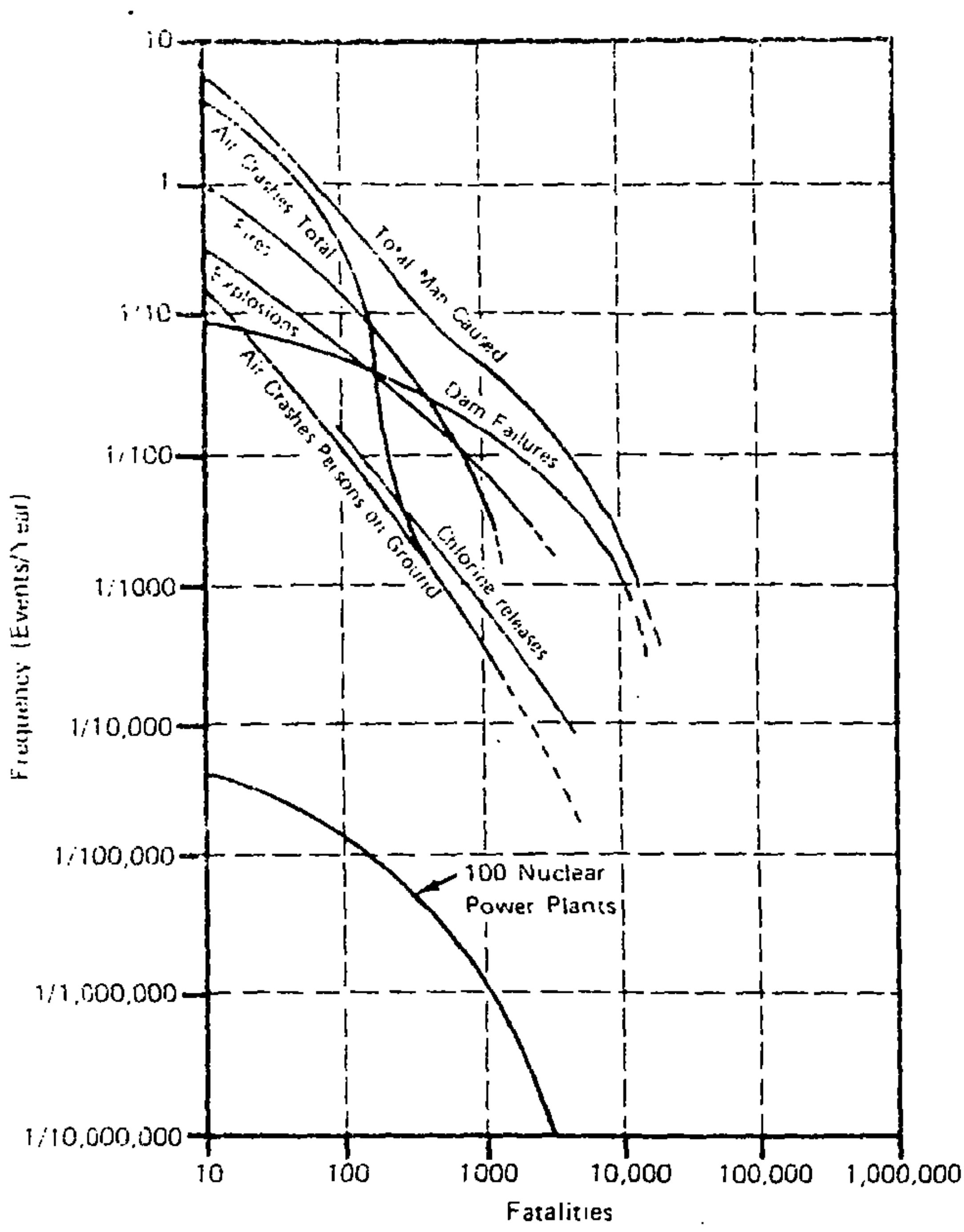

Figure 1: Frequency of Fatalities due to Man-Caused Events (WASH-1400) [2]

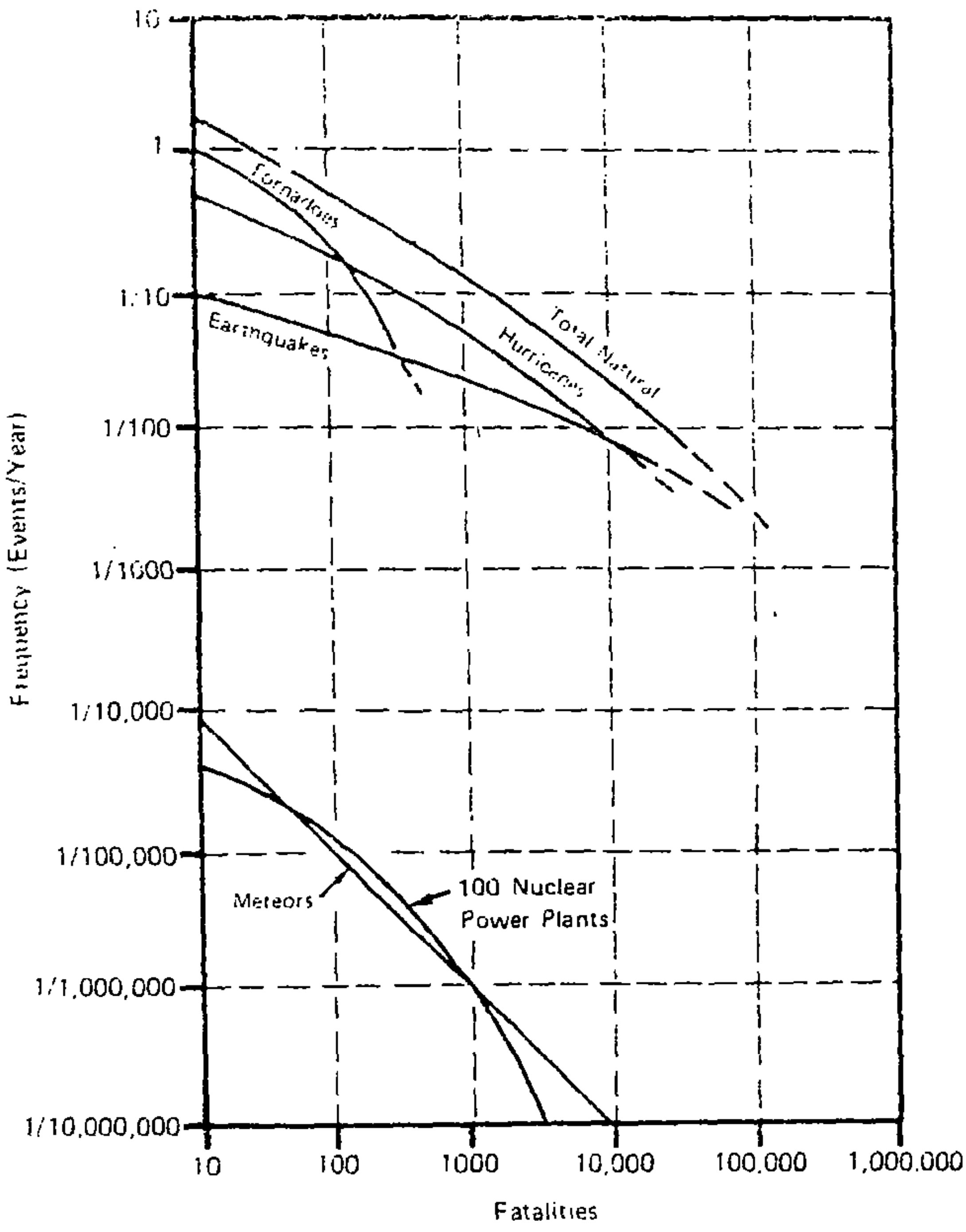

Figure 2: Frequency of Fatalities due to Natural Events (WASH-1400) [2]

2.2 Post TMI-2 Review of Source Term Technical Bases and Sandia Siting Study

Following the publication of WASH-1400 and the accident at Three Mile Island, Unit 2 (TMI-2), work was initiated to evaluate the predictive methods for calculating fission product release and transport. The results of this evaluation are contained in NUREG-0772 [18], "Technical Bases for Estimating Fission Product Behavior during LWR Accidents." The development of this report was prompted, in part, by the December 21, 1980 letter, from the Nuclear Safety Oversight Committee to President Carter [19], noting the questions raised at the time regarding iodine release and recommending that they should be answered by analyses and experimentation on an expedited basis. The NUREG-0772 evaluation resulted in several conclusions that represented significant departures from the Reactor Safety Study assumptions including the conclusion that cesium iodide (CsI) would be the expected predominant iodine chemical form under most postulated light water reactor (LWR) accident conditions. The potential impact of the NUREG-0772 findings on reactor regulation was also examined and the results were documented in NUREG-0771 [20]. These studies formed the basis for the designation of five accident groups as being representative of the spectrum of potential accident conditions. Brief descriptions of characteristics of the five accident groups are presented in Table 2.

Table 2 Brief Descriptions of the Characteristics of the Accident Groups (NUREG-0771, p. 8) [20]

Group 5 -	Limited core damage. No failures of engineered safety features beyond those postulated by the various design basis accidents are assumed. The most severe accident in this group includes substantial core melt, but containment functions as designed (siting DBA equivalent).
Group 4 -	Limited to modest core damage. Containment systems operate but in somewhat degraded mode (TMI-2 equivalent)
Group 3 -	Severe core damage. Containment fails by basemat melt-through. All other release mitigation systems have functioned as designed (analogous to Reactor Safety Study Pressurized Water Reactor, PWR, Categories 6 and 7)
Group 2 -	Severe core damage. Containment fails to isolate. Fission product release mitigating systems (e.g., sprays, suppression pool, fan coolers) operate to reduce release (analagous to Reactor Safety Study PWR Categories 4 and 5)
Group 1 -	Severe core damage. Essentially involves loss of all installed safety features. Severe direct breach of containment (analogous to Reactor Safety Study PWR Categories 1 and 3)

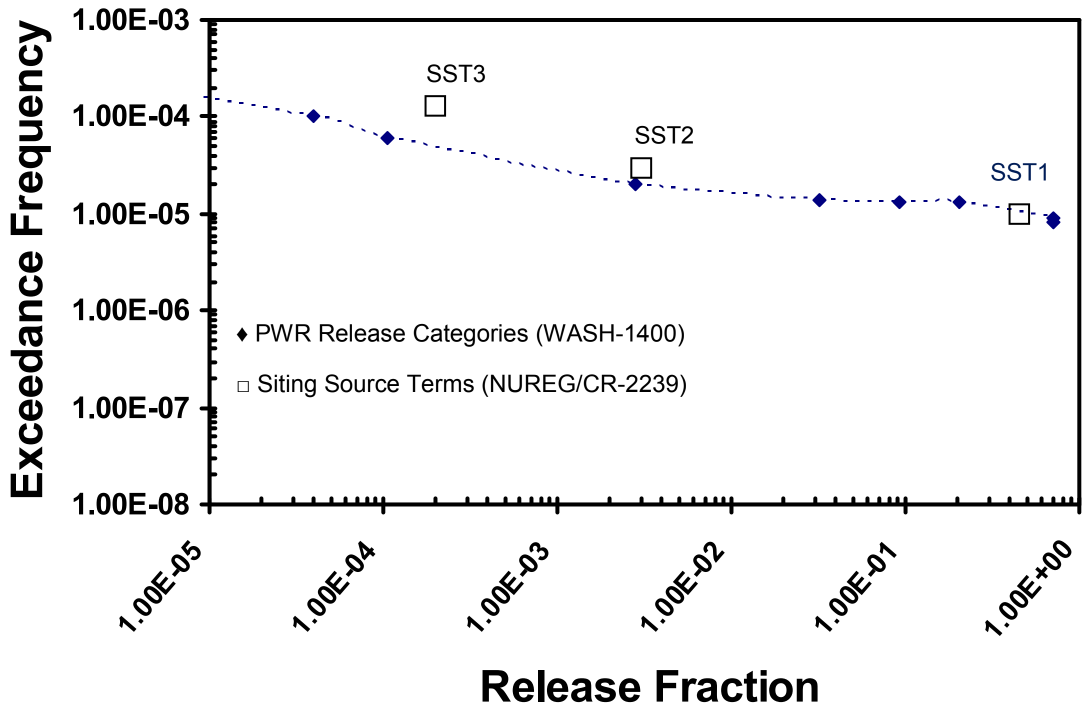

Figure 3: Comparison of frequency distribution (CCDF) of iodine release from Reactor Safety Study release categories for Surry with that assumed for Siting Source Terms

For the purpose of decisionmaking in such areas as siting and emergency response, NRC developed a generic set of radiological releases characterized as Siting Source Terms to represent the five accident groups. Table 3 presents the NRC-defined Siting Source Terms (denoted as SST1-5). The idea was that by adjusting the probabilities associated with each of the five source terms, the set could be made to approximately represent any current LWR design. Detailed probabilistic risk assessments (PRAs) were not performed for all reactors. However, based on the available PRAs at the time, NRC suggested that representative frequencies for the SST1, SST2, and SST3 were 1×10^{-5}, 2×10^{-5}, and 1×10^{-4} respectively. Figure 3 provides a comparison of the frequency distributions in the form of CCDF of iodine release from the Reactor Safety Study release categories for Surry with iodine release assumed in SST1, SST2, and SST3. Note that the frequencies shown in Figure 3 are exceedance frequencies (CCDFs) rather that frequency assigned to each source term. It should also be noted that iodine is not the sole radionuclide of importance to consequence analyses. Other fission product species notably cesium, tellurium, and ruthenium have also significant impact on the consequences of severe accidents. However, iodine is the dominant contributor to early fatalities and its frequency of release is presented throughout this report as an illustrative example.

Table 3 Siting Source Terms (SSTs)
(NUREG/CR-2239, p. 2-13) [8]

Release Characteristics[a]	Source Term SST1	SST2	SST3	SST4	SST5
Accident Type	Core Melt	Core Melt	Core Melt	Gap Release	Gap Release
Containment Failure Mode	Overpressure	H_2 Explosion or Loss of Isolation	-	-	-
Containment Leakage	Large	Large	1%/day	1%/day	0.1%/day
Time of Release (hr)	1.5	3	1	0.5	0.5
Release Duration (hr)	2	2	4	1	1
Warning Time (hr)	0.5	1	0.5	-	-
Release Height (meters)	10	10	10	10	10
Release Energy	0	0	0	0	0
Inventory Release Fractions					
Xe-Kr Group	1.0	0.9	6×10^{-3}	3×10^{-6}	3×10^{-7}
I Group	0.45	3×10^{-3}	2×10^{-4}	1×10^{-7}	1×10^{-8}
Cs-Rb Group	0.67	9×10^{-3}	1×10^{-5}	6×10^{-7}	6×10^{-8}
Te-Sb Group	0.64	3×10^{-2}	2×10^{-5}	1×10^{-9}	1×10^{-10}
Ba-Sr Group	0.07	1×10^{-3}	1×10^{-6}	1×10^{11}	1×10^{-12}
Ru Group	0.05	2×10^{-3}	2×10^{-6}	0	0
La Group	9×10^{-3}	3×10^{-4}	1×10^{-6}	0	0

a. As defined in the Reactor Safety Study [1].

Sandia National Laboratories (SNL) performed a study of the technical aspects of siting of nuclear power reactors. The results of this study were published in NUREG/CR-2239 [8], also known as the Sandia Siting Study. This study used Siting Source Terms at 91 existing or proposed reactor sites. Consequence analyses were performed for each of the five source terms. Table 4 compares the relative magnitude (normalized to 100 for source term SST1) of the mean values (using approximately 100 sampled weather sequences) of selected consequences, given the occurrence of each of the five SSTs. These calculations assumed an 1120 MWe PWR, population distribution (based on the 1970 census) and wind rose for Indian Point, New York City meteorology, and Summary Evacuation of persons within 10 miles. These results indicate that the mean consequences calculated for the SST1 release exceed those for the SST2 by 1 to 4 orders of magnitude and for SST3, SST4, and SST5 by 4 to 7 orders of magnitude.

As stated above, the Sandia Siting Study assumed an 1120 MWe PWR reactor at each of the 91 sites. Thus, its results were not directly representative of the actual potential consequences of an accident at the site. In a subsequent study, the results of NUREG/CR-2239 were scaled linearly by power level to derive an approximation of potential consequences for the actual reactor at the site. The results of this study were published in NUREG/CR-2723 [21]. This report also examined the financial consequences of potential accidents at nuclear power plants. Such information on the range of consequences was thought to be useful in a reevaluation of the liability limits of the Price-Anderson Act.

Table 4, Comparison of Conditional Mean Consequences Predicted for SSTs (NUREG/CR-2239, p. 2-14) [8]

Source Term	Mean Early Fatalities	Mean Early Injuries	Mean Latent Cancer Fatalities	Mean Thyroid Nodules	Mean Interdicted Land Area
SST1	100[b]	100	100	100	100
SST2	1×10^{-2}	0.5	7	3	1
SST3	0	0	2×10^{-2}	5×10^{-2}	0
SST4	0	0	4×10^{-4}	8×10^{-5}	0
SST5	0	0	4×10^{-5}	8×10^{-6}	0

a. Assumptions: 1120 MWe PWR, population distribution and wind rose for Indian Point, New York City meteorology, "Summary Evacuation" of persons within 10 miles.

b. All consequences are normalized to 100 for source term SST1.

2.3 NUREG-1150 Study

In the 1980s, a substantial research program on severe accident phenomenology was initiated. Updated computational models for severe accident analysis were developed and published in BMI-2104 [22]. A technical reassessment of severe accident source term technology for U.S. Light Water Reactors (LWRs) was published in NUREG-0956 [23]. This reassessment involved reviewing experimental and analytical results from severe accident research programs sponsored by the NRC and the nuclear industry. As a result of these activities, the Source Term Code Package (STCP) [24] was developed as an integrated tool for source term evaluation. Subsequently, the severe accident analysis code MELCOR [25] was developed based, in part, on the STCP.

The NUREG-1150 study [3] was a major effort to put the insights gained from the research on system behavior and phenomenological aspects of severe accidents into a risk perspective. An important characteristic of this study was the inclusion of the uncertainties in the calculations of core damage frequency and risk due to incomplete understanding of reactor systems and severe accident phenomena. The elicitation of expert judgment was used to develop probability distributions for many accident progression, containment loading, structural response, and source term issues. Five specific commercial nuclear power plants were analyzed in NUREG-1150: Surry, a 3-loop Westinghouse PWR with a subatmospheric containment; Zion, a 4-Loop Westinghouse PWR with a large dry containment; Sequoyah, a 4-loop Westinghouse PWR with an ice-condenser containment; Peach Bottom, a BWR-4 reactor with a Mark I containment; and Grand Gulf, a BWR-6 reactor with a Mark III containment. For Surry and Peach Bottom, the study included the analyses of both internal and external events.

Figure 4 shows the internal core damage frequency calculated in NUREG-1150 study for Surry as compared to that from the Reactor Safety Study and those reported in Individual Plant Examinations (IPEs) for PWRs. The IPE results presented are the range for the point estimate frequencies reported in NUREG-1560 [26] and do not include estimates of uncertainty. Also presented in Figure 4 are the recent results of the SPAR model for Surry obtained as part of the SOARCA Project. As shown in Figure 4, the recent results of the SPAR model for Surry indicate a core damage frequency that is one order of magnitude lower than the mean values obtained in the NUREG-1150 study. The staff has yet to report on how much of this difference is due to hardware modifications and procedural improvements that had been implemented by the plant since the NUREG-1150 study and how much of this difference is due to advances in the PRA state of the art.

Containment performance plays an important role in the assessment of the risk associated with severe accidents. The primary concerns for containment performance are how well the containment can withstand the pressure and temperature loads associated with severe core damage accidents and whether the containment is bypassed. For scenarios in which containment integrity is maintained, fission product release is small. For those scenarios leading to containment failure, fission product release depends on the timing as well as the size of the break in

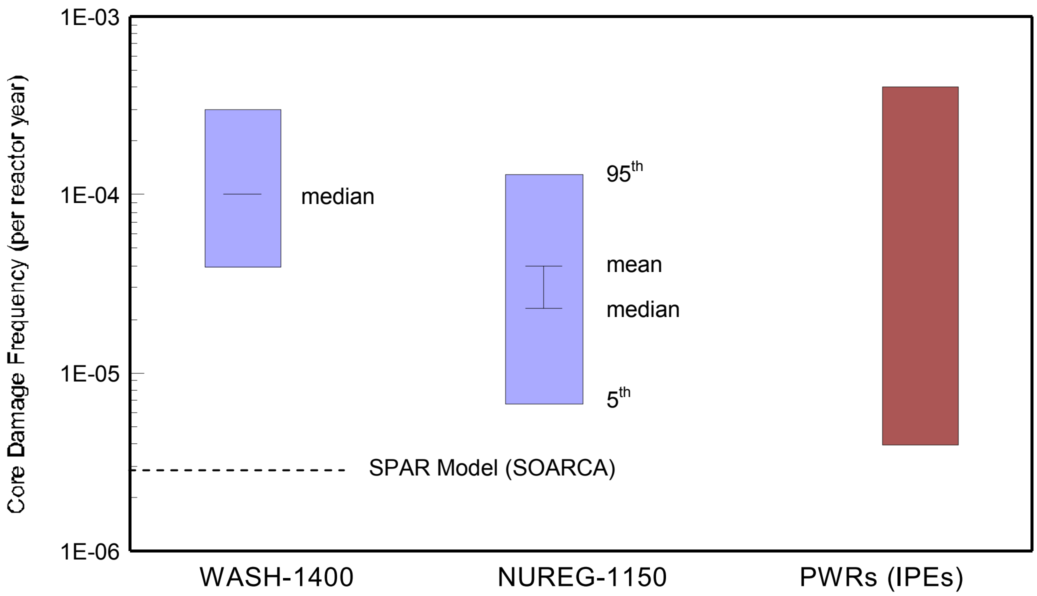

Figure 4: Surry frequency of core damage from internal events calculated in WASH-1400, NUREG-1150, the IPEs, and the recent result of the SPAR model for Surry (performed in support of SOARCA)

containment. Early containment failure can be important because it tends to result in shorter warning times for initiating public protective measures, and because radionuclide releases would generally be more severe than those from late containment failure. The mode of containment failure (i.e., gross failure versus leakages through failure of penetrations) influences the amount of radioactive materials inside the containment that would be released to the environment. For the NUREG-1150 study, accident progression and containment performance were analyzed using a single accident progression event tree developed for each plant, which was evaluated with the EVNTRE code [27]. The accident progression event trees made extensive use of the available severe accident computer code calculations and experimental results. As noted in NUREG-1150, "computer analyses cannot, in general, be used directly and alone to calculate branching probabilities in the accident progression event tree. Since the greatest source of uncertainty is typically associated with the modeling of severe accident phenomena, the results of a single computer run (which uses a specific model) do not characterize the branching uncertainty." It was therefore necessary to use sensitivity studies, uncertainty studies, and expert judgment to characterize the likelihood of alternative events that affect the course of an accident. The elicitation of expert judgment was used to develop probability distributions for many accident progression, containment loading, and structural response issues.

The NUREG-1150 results of the containment analyses for Surry and Peach Bottom are summarized in Figures 5 and 6, respectively. Figure 5 displays the Surry NUREG-1150 results for the conditional probabilities of seven containment-related accident progression bins (e.g., vessel breach (VB), alpha, early containment failure) for each of seven plant damage states (e.g., LOSP)

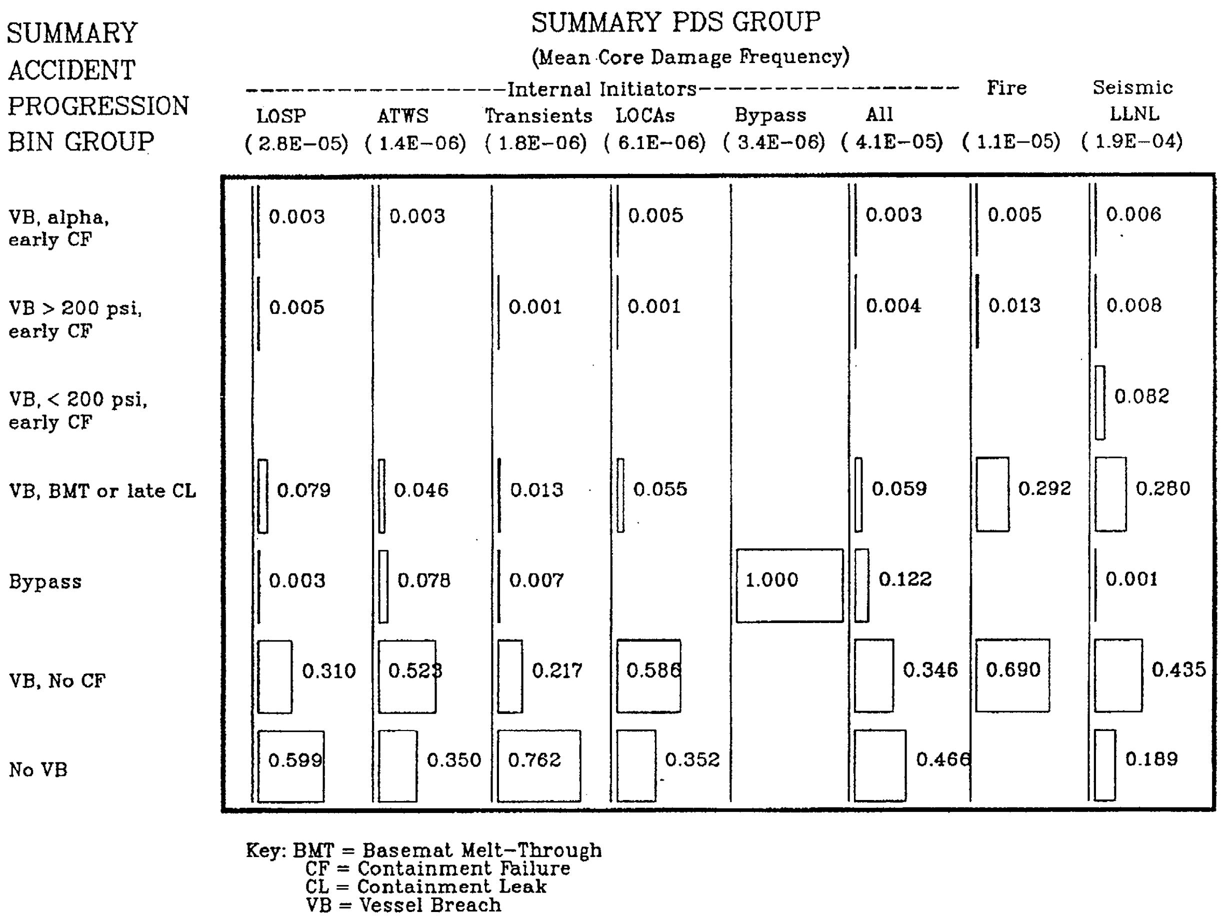

Figure 5: Conditional Probability of Accident Progression Bins at Surry (NUREG-1150, p. 3-12) [3]

PLANT DAMAGE STATE
(Mean Core Damage Frequency)

ACCIDENT PROGRESSION BIN	Internal Initiators: SBO (2.08E-06)	Internal Initiators: LOCAs (1.50E-07)	Internal Initiators: ATWS (1.93E-06)	Internal Initiators: Transients (1.81E-07)	Internal Initiators: All (4.34E-06)	Fire (1.98E-05)	Seismic: LLNL (7.52E-05)
VB > 200psi, early WWF	0.045		0.006		0.022	0.045	0.028
VB < 200 psi, early WWF	0.012	0.028	0.006	0.026	0.011	0.004	0.027
VB > 200 psi, early DWF	0.436		0.330		0.341	0.529	0.369
VB < 200 psi, early DWF	0.133	0.360	0.194	0.356	0.183	0.070	0.489
VB, late WWF	0.007			0.002	0.003	0.009	0.006
VB, late DWF	0.061	0.074	0.015	0.074	0.047	0.086	0.074
VB, CV	0.074	0.003	0.207	0.016	0.110	0.020	0.007
No CF	0.121	0.536	0.127	0.512	0.184	0.159	
No VB	0.112		0.091	0.014	0.089	0.078	
No Core Damage			0.024		0.010		

VB = Vessel Breach
WWF = Wetwell Failure
DWF = Drywell Failure
CV = Containment Venting
CF = Containment Failure

Figure 6: Conditional Probability of Accident Progression Bins at Peach Bottom (NUREG-1150, p. 4-13) [3]

As it is indicated in Figure 5, on a plant damage state frequency weighted average, the conditional mean probability of early containment failure from internally initiated accidents for Surry was found to be very small (about 1 percent) in the NUREG-1150 Study. The primary mechanism leading to early containment failure were loads from Direct Containment Heating (DCH) for accident sequences with high reactor coolant system (RCS) pressures at vessel breach and in-vessel steam explosion for sequences with low RCS pressure at vessel breach. The NUREG-1150 estimate of the likelihood of the early containment failure is substantially lower than the WASH-1400 result even though the phenomena of DCH had not been identified at the time of the Reactor Safety Study. In addition to the lower assumed containment capacity, the prediction of containment loading in the Reactor Safety Study was unrealistically high. It should be noted that the characterization of containment performance in the Reactor Safety Study was simplistic in comparison to the NUREG-1150 Study.

Figure 6 displays the Peach Bottom NUREG-1150 results for the conditional probabilities of 10 containment-related accident progression bins (e.g., vessel breach (VB), late wet well failure (WWF)) for each of six plant damage states such as station blackout (SBO). As it is indicated in Figure 6, on a plant damage state frequency weighted average, the conditional mean probability of early drywell failure from internally initiated accidents for Peach Bottom was found to be about 52 percent. For Peach Bottom, although the early containment failure was found to be quite likely in the NUREG-1150 Study, the mechanism resulting in failure as well as the modes and locations of failure were quite different from those considered in the Reactor Safety Study. In the Reactor Safety Study, the most likely failure location was assessed to be the upper portion of the suppression pool. In the NUREG-1150 Study, other mechanisms of containment failure, such as direct attack of drywell wall by molten core debris, were found to be also important. The dominant location of overpressure failure in the NUREG-1150 Study was assessed to be the lifting of drywell head by stretching the head bolts.

One of the major activities of the NUREG-1150 study was the development of fission product source terms for a spectrum of accident conditions. The source terms were calculated using a simplified parametric algorithm. The parametric equations describe the source terms as the product of release fractions and transmission factors at successive stages in the accident progression for a variety of release pathways, a variety of accident progressions, and nine classes of radionuclides. Probability distributions for important parameters were developed using the elicitation of expert judgment to augment the analytical results to reflect model uncertainties. The source term analysis resulted in characterizing thousands of source terms (20,000 for Surry) associated with tens of plant damage states, hundreds of accident progression bins, and the variation in source term phenomenological issues which were included in the propagation of uncertainties. In Figure 7, frequency distribution (CCDF) of iodine release predicted by the NUREG-1150 Study for Surry is compared with that obtained from the Reactor Safety Study release categories along with that assumed in Sandia Siting Study. As shown in Figure 7, the mean frequency of iodine release is generally lower than that obtained in the Reactor Safety Study. In particular, the mean frequency of high (more than 10 percent) release of iodine predicted in the NUREG-1150 Study is more than one order of magnitude less than that obtained in the Reactor Safety Study.

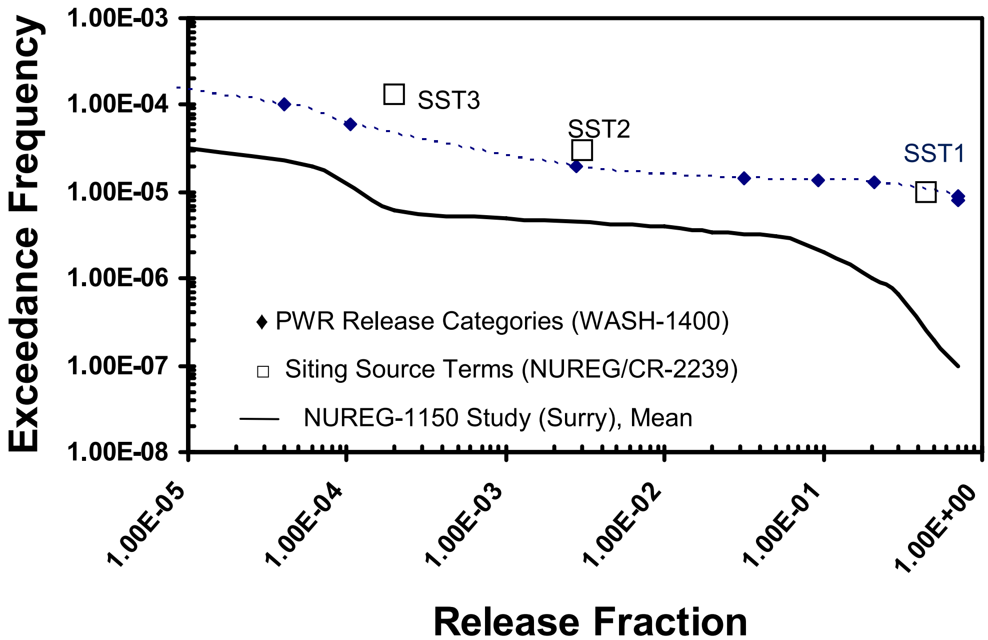

Figure 7: Comparison of frequency distribution (CCDF) of iodine release predicted by NUREG-1150 study for Surry with that obtained from the Reactor Safety Study release categories along with that assumed in the Sandia Siting Study

It should also be noted that the NUREG-1150 Study calculated the mean frequency of iodine releases similar to one assumed in SST1 to be less than 10^{-6} (the screening frequency used in SOARCA).

In NUREG-1150, the source terms were grouped according to the warning time and their potential to cause early or latent cancer fatalities. Through this "partitioning" process, the large numbers of calculated radioactive releases were collected into a small set of source term groups (30 to 60 in number for each plant). This set of groups was then used in the offsite consequence calculations.

The consequence measures, early fatalities, population dose (person-rem), and latent cancer fatalities, were calculated for each source term group by the MAACS code [28]. The output of MACCS for each source term group is a distribution of the consequences, conditional on occurrence of the source term, which incorporates the uncertainty (variability) due to weather as well as the uncertainty in the underlying health (dose-response) models.

Comparisons of frequency distributions (CCDFs) of early and latent cancer fatalities predicted by the NUREG-1150 Study with those reported in the Reactor

Safety Study are presented in Figures 8 and 9, respectively. These figures are reproduced from an August 1990 Report to the Congress from the Presidential Commission on Catastrophic Nuclear accidents [29]. As shown in Figures 8 and 9, the consequences calculated in the NUREG-1150 study are significantly lower than those obtained in the Reactor Safety Study.

The NUREG-1150 risk analyses were performed to provide the frequency of a large release, defined at the time as the one that could cause one or more early fatalities. The results of such evaluations for NUREG-1150 plants are shown in Figure 10, reproduced from NUREG-1150 (Vol. 1, pp. 13-10). As shown in Figure 10, the mean frequency of a release, due to internal events, to cause one or more early fatalities is less than 10^{-6}.

In spite of the plant specific nature of the NUREG -1150 quantitative results (e.g., core damage, frequency and offsite consequences), this study provides valuable insights into severe accident phenomenological issues and associated state-of-knowledge uncertainties which are very useful to the study of plants with similar NSSS and containment designs.

The insights from the NUREG-1150 Study have been used in several areas of reactor regulation including the development of alternative radiological source terms for evaluating design basis accidents at nuclear reactors. In 1995, the NRC published NUREG-1465 [30], which defined an alternative accident source term for regulatory applications. The release fractions for the alternative accident source terms were derived from the insights and simplifications of the NUREG-1150 source term analyses documented in NUREG/CR-5747 [31]. The NRC is also using source terms derived from the NUREG-1150 Study in an effort to revise criteria for protective action recommendations for severe accidents [32].

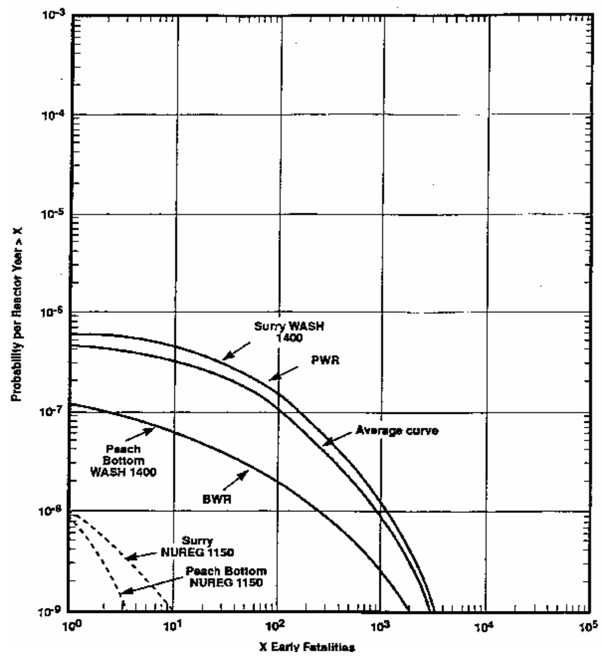

Figure 8: Comparison of frequency distributions (CCDFs) of early fatalities predicted by the NUREG-1150 study with those reported in the Reactor Safety Study
(Reproduced from the Report to the Congress from the Presidential Commission on Catastrophic Nuclear accidents, August 1990, Appendix B) [29]

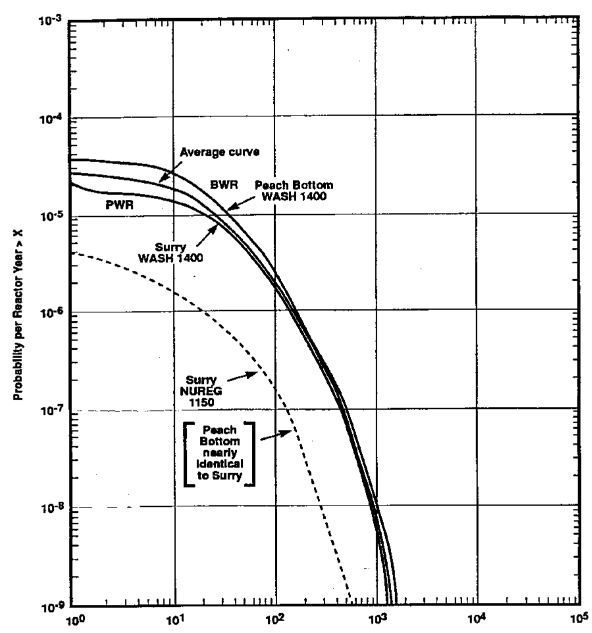

Figure 9: Comparison of frequency distributions (CCDFs) of latent cancer fatalities predicted by the NUREG-1150 study with those reported in the Reactor Safety Study
(Reproduced from the Report to the Congress from the Presidential Commission on Catastrophic Nuclear accidents, August 1990, Appendix B) [29]

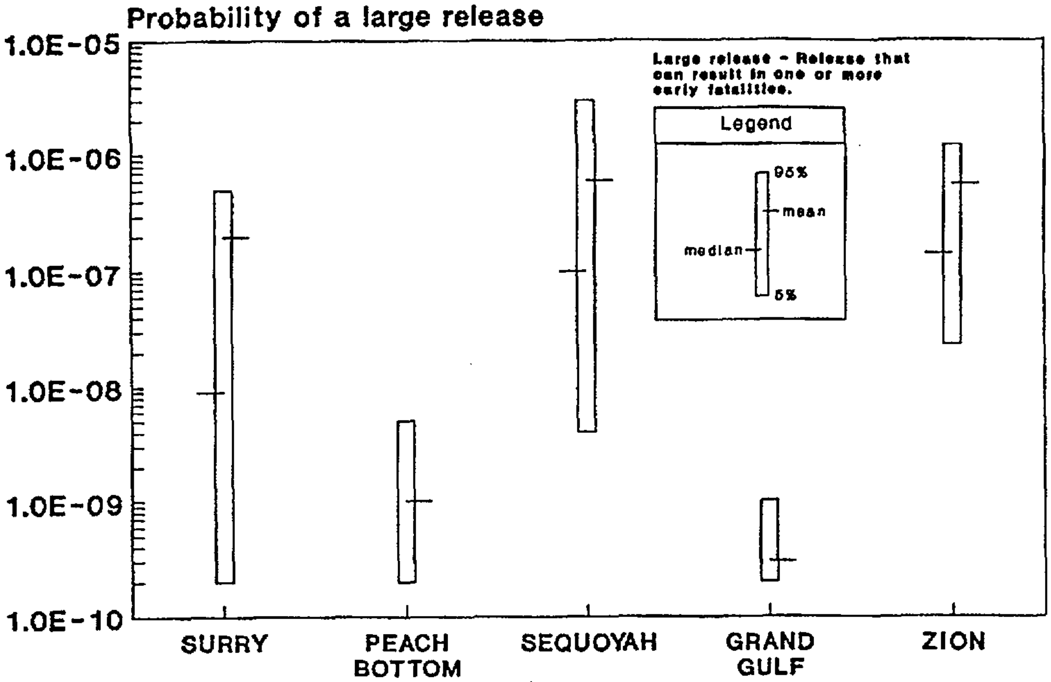

Figure 10: Frequency of a release that can result in one or more early fatalities due to internal events for NUREG-1150 Plants
(Reproduced from NUREG-1150, Vol. 1, pp. 13-10) [3]

2.4 Reassessment of Selected Factors Affecting Siting 0f Nuclear Power Plants

In 1997, Brookhaven National Laboratory performed a series of probabilistic consequence assessments in support of an effort to reassess siting criteria [33]. This study took into account the insights gained from the NUREG-1150 Study [3] and the integrated risk assessment for LaSalle[1] [34] and examined consequences in a risk-based format consistent with the quantitative objectives of the NRC's Safety Goal Policy.

The approach taken in this study was to use sets of representative source terms (for internal events) and their associated frequencies for LaSalle and each of the NUREG-1150 plants. These source terms were applied to a set of hypothetical sites with various population densities and locations of urban centers which would encompass most of the existing reactor sites. The site meteorology was selected such that, from a meteorological standpoint, the calculation of consequences would be likely to bound 80 percent of existing reactor sites. Sensitivity calculations were also performed to evaluate the effects of emergency protective action assumptions on the risk of prompt fatality and latent cancer fatality, as well as population relocation. The study concluded that the prompt and latent fatality risks at all generic sites met the Quantitative Health Objectives (QHOs) of the NRC's Safety Goal Policy by margins ranging from one to more than three orders of magnitude. Results from the study also indicated that both the quantity of radioactivity released during a severe accident, as well as the likelihood of release, were considerably lower than those predicted by earlier studies (e.g., the Sandia Siting Study of 1982). For this reason, the severe accident risks estimated in earlier studies were concluded to be unduly pessimistic, based on a revised understanding of severe accidents.

Developing representative source terms involved the determination of a set of reactor accident progression groups, the dominant plant damage states and the associated release characteristics for each reactor design which represented the full spectrum of severe accidents. A small set of source terms (4 to 7 for each plant) was developed [35] by considering release categories which accounted for a spectrum of possible timing and modes of containment failure. For each containment failure mode, the source terms were selected based on the dominant accident progression characteristics leading to the containment failure. The magnitudes of releases for each release category were obtained by using the mean values of the probability distributions of source term parameters used in NUREG-1150 and the LaSalle integrated risk assessment studies. For example, the representative source terms for Surry consist of four release categories as shown in Table 5. These release categories correspond to early containment failure (ECF), late containment failure (LCF), no containment failure, and containment bypass due to interfacing-system LOCA (Event V).

[1] It should be noted that the integrated risk assessment for LaSalle, a BWR with Mark II containment, was begun before the NUREG-1150 analysis and the LaSalle integrated risk assessment program supplied the NUREG-1150 Study with many analysis methods including methods for handling and propagating statistical uncertainties in an integrated way through the entire analysis, and BWR thermal-hydraulic models which were adapted for the Peach Bottom and Grand Gulf analyses.

Table 5: Characteristics of Surry Release Categories, Internal events
(NUREG/CR-6295, pp 3-19) [33]

Release Category	Plant Damage State	Accident Progression Characteristics						
		Containment Failure Time	Containment Failure Mode	CCI	Amt CCI	RCS Pres.	VB Mode	Sprays
RSUR1	LOSP	CF at VB	Rupture	Prm Dry	Medium	Low	Alpha	No
RSUR2	LOSP	Late CF	Leak	Prm Dry	Large	Low	Pour	No
RSUR3	LOSP	No CF	No CF	Prm Dry	Large	Low	Pour	L+VL
RSUR4	Bypass (V)	No CF	Bypass	Prm Dry	Large	Low	Pour	No

Table 6: Radionuclide Release Characteristics into Environment for Surry, Internal Events
(NUREG/CR-6295, pp3-19) [33]

Release Category	Frequency	Elevation (m)	Energy (W)	Time of Core Uncovery	Time of Release (hrs)	Release Duration	Fractional Releases								
							Ng	I	Cs	Te	Sr	Ba	Ru	La	Ce
RSUR1	2.9E-7	10	2.8E+7	5.0 hrs	6.0	200 sec	1.0E+0	2.5E-1	1.8E-1	8.0E-2	2.0E-2	2.0E-2	5.0E-3	1.0E-3	5.0E-3
		10	1.6E+6		6.06	2 hrs	0.0E+0	1.0E-1	1.3E-1	1.0E-1	4.0E-2	4.0E-2	1.0E-3	5.0E-3	5.0E-3
RSUR2	2.4E-6	10	5.2E+5	5.0 hrs	12.0	3 hrs	1.0E+0	6.0E-2	3.0E-2	9.0E-2	3.0E-3	3.0E-3	1.0E-3	4.0E-4	4.0E-4
RSUR3	3.3E-5	0	0.0E+0	5.0 hrs	6.0	10 hrs	2.5E-3	1.5E-5	1.2E-8	7.5E-9	2.5E-9	2.5E-10	2.0E-10	3.0E-10	4.0E-10
		0	0.0E+0		16.0	10 hrs	2.5E-3	1.5E-5	1.2E-8	7.5E-9	2.5E-9	2.5E-10	2.0E-10	3.0E-10	4.0E-10
RSUR4	1.6E-6	0	1.9E+6	20 min	1.0	30 min	1.0E+0	7.5E-2	6.0E-2	2.0E-2	5.0E-3	5.0E-3	1.0E-3	3.0E-4	1.0E-3
		0	1.7E+5		1.5	2 hrs	0.0E+0	4.0E-2	6.0E-2	6.0E-2	2.0E-2	2.0E-2	6.0E-4	3.0E-3	3.0E-3

The results of the NUREG-1150 Study indicate that the station blackout sequences are the largest contributors to internal event mean core damage frequency for Surry and are the dominant plant damage states leading to the first three release categories (see Figure 5). The radionuclide release characteristics into environment for different Surry release categories are presented in Table 6. The magnitudes of releases for each release category were obtained by utilizing the basic parametric equation used in NUREG-1150. The mean values of the probability distributions of source term parameters associated with the corresponding accident progression characteristics were used in these calculations. The energy, timing, and duration of releases are based on the results of STCP calculations performed in support of NUREG-1150. These representative source terms have been used in a number of other studies including evaluation of risk importance of containment and related ESF system performance requirements [36] and the Regulatory Analysis Technical Evaluation Handbook (NUREG/BR-0184) [37].

Figure 11 presents a comparison of frequency distribution (CCDF) of iodine release predicted by the NUREG-1150 Study for Surry with that obtained from the representative source terms used in NUREG/CR-6295. As shown in Figure 11, these source terms represent well the spectrum of mean iodine releases predicted by the NUREG-1150 Study. Consequence calculations using representative source term are reported in References 33 and 36. A comparison of the mean frequency distribution of population dose (person-rem) to the entire region at Surry predicted by the NUREG-1150 Study with those obtained by using representative source terms is presented in Figure 12.

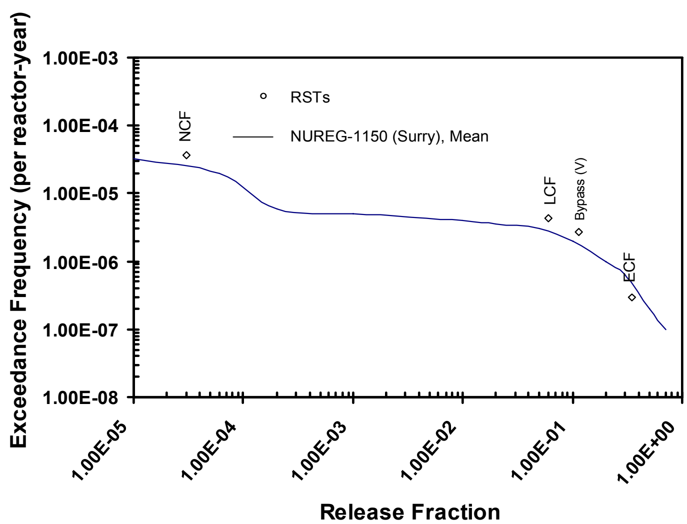

Figure 11: Comparison of frequency distribution (CCDF) of iodine release predicted by NUREG-1150 Study, internal events, for Surry with that obtained from the representative source terms (RSTs)

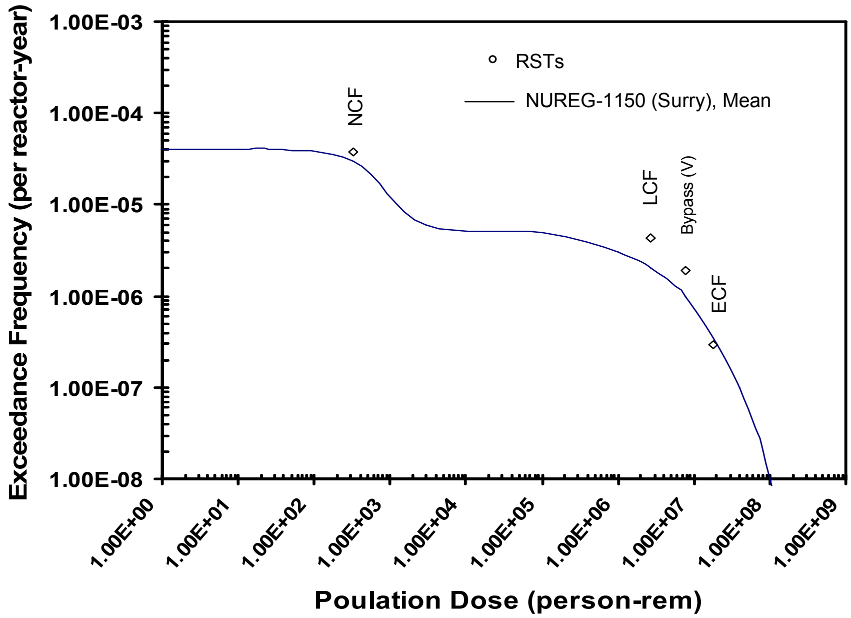

Figure 12: Comparison of frequency distribution (CCDF) of population dose (person-rem) to entire region at Surry predicted by the NUREG-1150 Study, internal events, with those obtained by using representative source terms (RSTs)

It should be noted that a generic set of accident sequences and associated frequencies and source term parameters has also been recently developed by Electrical Power Research Institute (EPRI) and Nuclear Energy Institute (NEI) as a part of risk-informed evaluation of protective action strategies for nuclear plant off-site emergency planning [38]. However a review of these source terms is beyond the scope of the present study.

3. RECENT ADVANCES IN UNDERSTANDING OF SEVERE ACCIDENT PHENOMENOLOGY AND CONTAINMENT FAILURE MECHANISMS

Since completion of the NUREG-1150 Study, more analytical and experimental studies have been performed to address many severe accident issues, including direct containment heating, Mark I liner attack, and in-vessel steam explosion. Reference 39 provides an overview of the state of knowledge and uncertainties associated with severe accident phenomena and source tem issues. Some recent advances in understanding of the severe accident issues and containment failure mechanisms important to consequence assessment are provided below for use in demonstrating the feasibility of developing a simplified approach for updating the results of NUREG-1150 Study.

3.1 Direct Containment Heating

The concern for DCH arises only if vessel breach occurs while the reactor is at elevated pressure. For such cases, the expulsion of molten core debris could lead to very rapid and efficient heat transfer to the containment atmosphere, possibly accompanied by oxidation reactions and hydrogen combustion that further enhance the energy transfer. The pressurization accompanying this process is referred to as direct containment heating.

The potential for DCH to cause containment failure depends on several factors, such as the primary system pressure, the size of opening in the vessel, the temperature and composition of the core debris exiting the vessel, the containment pressure and composition before the vessel breach, the amount of water in the cavity, and the dispersive characteristics of the reactor cavity.

Since the completion of NUREG-1150, advances have been made in understanding of the DCH phenomena. The NRC staff identified DCH as a major issue for resolution in its revised severe accident research plan [40] and sponsored analytical and experimental programs for understanding the key physical processes in DCH. A number of experiments were performed in support of DCH issue resolution for PWRs. These experiments included both separate effects tests and integral effects tests, simulating the DCH processes in scaled models of the Zion, Surry, and Calvert Cliffs containments. The results of an assessment of the probability of containment failure due to DCH for the Zion nuclear power plant were published in NUREG/CR-6075 and its supplement [41]. By using the Risk Oriented Accident Analysis Methodology (ROOAM) [42], it was concluded that the containment failure probability due to DCH at Zion is so low as to be considered physically unreasonable. The basic understanding upon which the approach to quantification of DCH loads is based is that intermediate compartments trap most of the debris dispersed from the reactor cavity and that the thermal-chemical interactions during this dispersal process are limited by the incoherence in the steam blowdown and melt entrainment processes. With this understanding, it was possible to reduce most of the complexity of DCH phenomena to a single parameter: the ratio of the melt-entrainment time constant to the blowdown time constant, which is referred to as the coherence ratio. Reference 43 provides further discussions on application of ROAAM to address the DCH issue for 34 Westinghouse plants with large dry or subatmospheric containment. As a part of a study to assess the risk importance of containment and related ESF system performance requirements, the accident progression event trees (APET) for Zion that had been used for NUREG-1150 was modified to reflect the latest understanding of DCH phenomena in Zion [36]. This also included incorporation of the containment fragility curve that was used in the NUREG-6075 study [41] and the Zion IPE [44]. The results of updated evaluation of the conditional probability of accident progression bins for internal events, as compared with the original results of NUREG-1150, are summarized in Table 7 (see also Figure 13). The conditional probability of early containment failure was found to be very low for the Zion plant. This is due to the fact that the Zion containment capacity was high and the expected containment loads from the core melt accidents were not high enough to threaten the integrity of the containment during the early stage of an accident. In addition, a large fraction of the plant damage states resulted in low RCS pressure at the time of vessel breach, which lowered the potential for loads associated with DCH. In view of more recent understanding of in-vessel steam explosion (alpha mode failure), the only major physical phenomena contributing to early containment failure in PWRs with large dry containments is containment isolation failure.

Table 7: Impact of latest understanding of DCH on probability of accident progression bins for internal events at Zion (NUREG/CR-6418, pp. 2-9) [36]

Accident progression bin	Plant damage state (Mean core damage frequency)				
	SBO (6.3×10^{-6})	LOCAs (3.14×10^{-4})	Transients (1.92×10^{-5})	VSGTR (1.58×10^{-6})	All (3.4×10^{-4})
CF before VB (Isolation failure not followed by CF)	0.004 (0.004)[a]	0.004 (0.004)	0.004 (0.004)	-	0.004 (0.004)
Early CF	0.006 (0.021)	0.008 (0.009)	0.004 (0.009)	-	0.008 (0.009)
Late CF	0.313 (0.315)	0.245 (0.246)	0.193 (0.192)	-	0.243 (0.244)
Bypass	0.001 (0.01)	-	0.004 (0.004)	1.0	0.005 (0.007)
No CF	0.674 (0.656)	0.739 (0.737)	0.794 (0.790)	-	0.740 (0.737)

VB = vessel breach
CF = containment failure
(a) Numbers in parentheses are based on the original NUREG-1150 assumptions.

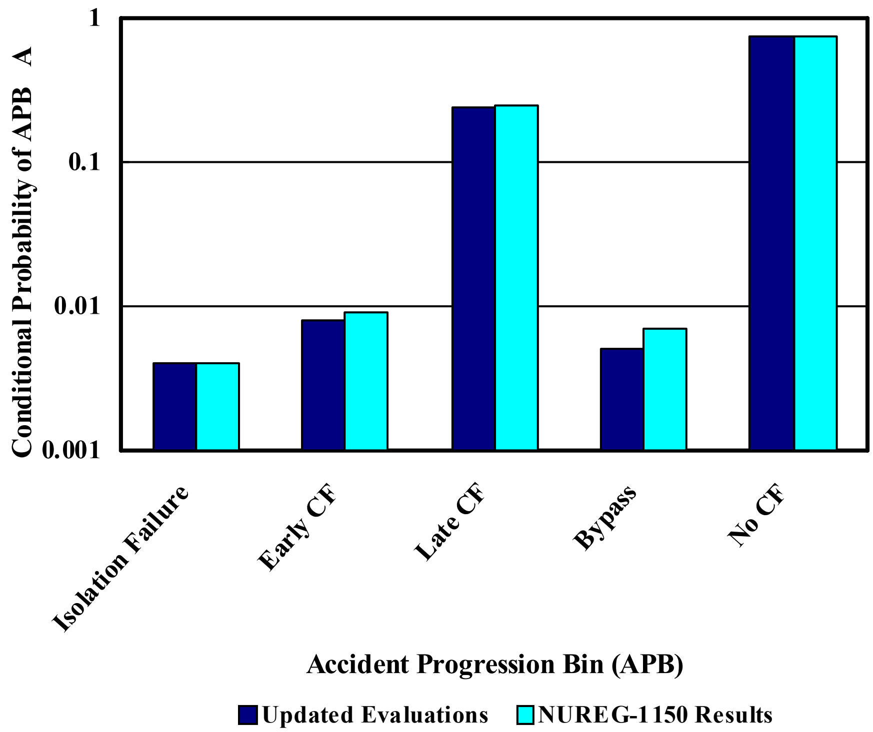

Figure 13: Impact of latest understanding of DCH on probability of accident progression bins for internal events at Zion [39]

3.2 Steam Explosion

During progression of severe accidents, molten debris from the damaged core would at some point begin to fall into the lower plenum of the reactor vessel. If an appropriate amount of water remained in the lower plenum, molten core material falling into the lower plenum could potentially cause a steam spike and, if severe enough, an explosion. Rapid steam pressure rise and missile, resulting from in-vessel steam explosion (alpha mode failure) has been identified as a potential challenge to the containment in past studies [2,3]. However, a more recent assessment of this issue in 1996 by an NRC sponsored steam explosion review group [45] concluded that alpha mode failure is of very low probability that is of little or no significance to the overall risk.

3.3 Drywell Melt-through in Mark I Containments

Drywell liner melt-through (caused by direct contact with core debris) has been found to be the most important contributor to early containment failure for Mark I containments. This failure mode is only possible for Mark I containments because the pedestal and drywell floor are at the same level, and core debris can easily reach the containment liner. The steel liner is the containment pressure boundary, and such a breach (i.e. drywell melt-through) would constitute an early containment failure.

The issue of Mark I drywell shell (liner) melt-through at Peach Bottom was assessed by the NUREG-1150 molten core-containment interaction panel. The

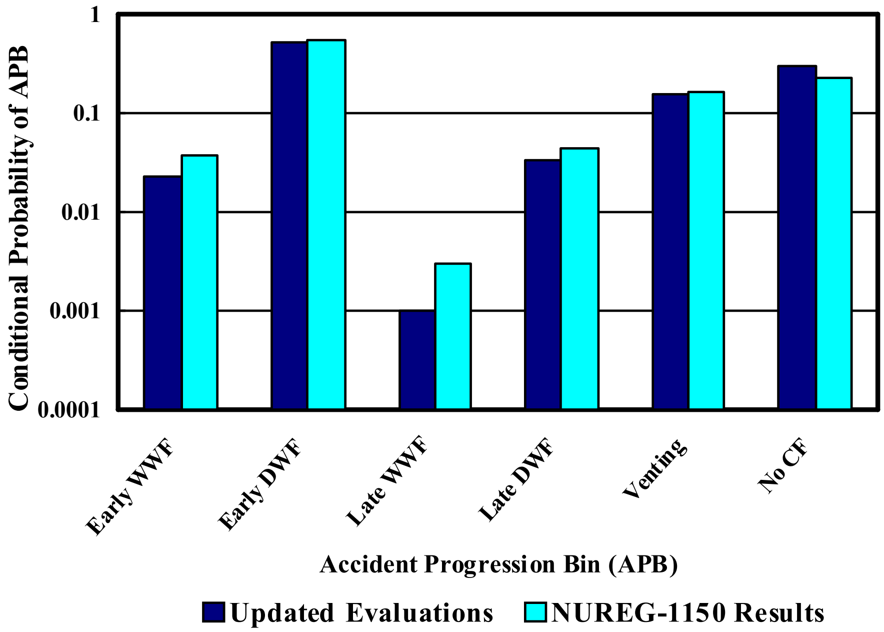

Figure 14: Impact of latest understanding of drywell melt-through on probability of accident progression bins for internal events at Peach Bottom [39]

results of expert panel elicitation are reported in Reference 46. There were two schools of thought on this issue. Some experts felt that melt spreading is a hydrodynamically limited phenomenon and that there is a high probability of drywell failure, even with the presence of water. Other experts felt the movement of the debris is thermodynamically limited and will be impeded by crust formation and the presence of water. The experts provided subjective probability of drywell failure for several different scenarios, characterized by five parameters. These parameters were the RPV pressure, debris flow rate from the vessel, debris superheat, un-oxidized metal content of the debris, and the presence of water on the drywell floor.

Since the completion of NUREG-1150, the NRC has sponsored analytical and experimental programs to address and resolve this so-called "Mark I Liner Attack" issue. The results of an assessment of the probability of Mark I containment failure by melt attack of the liner were published in NUREG/CR-5423 [47] and NUREG/CR-6025 [48]. It was concluded that, in the presence of water, the probability of early containment failure by melt-attack of the liner is so low as to be considered physically unreasonable.

As part of a study to assess the risk significance of containment and related ESF system performance requirements [36], the Peach Bottom APET that had been used for the NUREG-1150 study was modified to reflect the more recent understanding of the drywell melt-through mechanism. The results of the updated evaluation of the conditional probability of accident progression bins for internal events, as compared with the original results of NUREG-1150, are summarized in Figure 14 (see also Table 8). The results of updated evaluations indicate a decrease in the probability of containment failure but not as much as one might expect. This is because of the possibility of multiple failure modes. Also, in the ATWS and SBO accident groups (the risk dominated plant damage states), there is a significant probability that the vessel failure will occur when there is no water in the pedestal.

Table 8: Impact of latest understanding of drywell melt-through on probability of accident progression bins for internal events at Peach Bottom (NUREG/CR-6418, pp. 2-15) [36]

Accident progression bin	Plant damage state (Mean core damage frequency)				
	SBO (2.08×10^{-6})	LOCAs (1.5×10^{-7})	ATWS (1.93×10^{-6})	Transients (1.81×10^{-7})	All (4.34×10^{-6})
Early WWF	0.024 (0.06)[a]	0.04 (0.028)	0.018 (0.013)	0.04 (0.027)	0.023 (0.037)
Early DWF	0.606 (0.581)	0.052 (0.36)	0.515 (0.516)	0.053 (0.354)	0.51 (0.532)
Late WWF	0.001 (0.007)	—	—	0.001 (0.001)	0.001 (0.003)
Late DWF	0.026 (0.06)	0.112 (0.074)	0.023 (0.015)	0.112 (0.074)	0.034 (0.043)
Containment venting	0.126 (0.118)	0.003 (0.003)	0.212 (0.24)	0.01 (0.008)	0.15 (0.159)
No CF	0.21 (0.172)	0.793 (0.536)	0.265 (0.199)	0.78 (0.532)	0.3 (0.224)

WWF = Wetwell failure
DWF = Drywell failure
CF = Containment failure
(a) = Numbers in parentheses are based on the original NUREG-1150 assumptions.

4. A SIMPLIFIED APPROACH TO UPDATE THE RESULTS OF EARLIER LEVEL-3 PRAs FOR COMPARISON WITH SOARCA

Although performing Level-3 PRAs for the pilot plants is the best way to benchmark the SOARCA methodology, results and insights from the NUREG-1150 Study and Integrated Risk Assessment for LaSalle, together with more recent advances in understanding of the severe accident issues and containment failure mechanisms, could be used for developing a simplified, yet systematic and defensible, approach to update the results of such earlier Level-3 PRAs for comparison with aspects of SOARCA results. The major elements of such an approach are depicted in Figure 15.

Developing a small set of accident progression groups and their associated frequencies and release characteristics (source terms) to represent the full spectrum of potential severe accidents is an essential element of the proposed approach. The representative source terms developed previously [35, 33] and used in a number of other NRC studies [33, 36, 37] could be used as a starting point. As it was noted in Section 2.4, these representative source terms were developed based on the insights from the results of the NUREG-1150 Study and the Integrated Risk Assessment for LaSalle. Extending these representative source terms to include fire and seismic initiators (for Surry, Peach Bottom, and LaSalle Plants) as well as their further refinements to represent more accident progression groups and/or plant damage states may be necessary. Presenting these representative source terms and their associated frequencies in the form of complementary cumulative distribution functions (CCDFs) provides a logical framework for the subsequent adjustments and comparison with the SOARCA results.

The representative accident groups and their associated frequencies can be revised based on the more recent insights on core damage frequencies (e.g., results of SPAR models) and containment failure modes and mechanisms. For example, the results of NUREG-1150 Study for conditional probability of accident progression bins for each "summary plant damage state" at Surry (see Figure 5) can be revised to reflect the current knowledge of containment failure modes and mechanisms The results of SPAR model for core damage frequencies obtained as a part of SOARCA, together with the results of conditional probabilities of accident progression bins can then be used to revise the representative accident groups and their associated frequencies for Surry.

Table 9 presents an illustrative example of the impact of current knowledge and understanding of early containment failure on NUREG-1150 results for the conditional probability of accident progression bins at Surry. The results of NUREG-1150 Study indicate that, for the internal and fire initiators, the mean conditional probability of early containment failure is on the order of 0.01 and the early containment failure is only due to in vessel steam explosion and DCH. It should be noted that the Surry containment is maintained at a subatmospheric pressure (10psia) during operation with a continued monitoring of the containment leakage, and thus the NUREG-1150 Study concluded that the likelihood of pre-existing leaks of significant size is negligible. Therefore, with the more recent understanding of DCH and in vessel steam explosion phenomena (as discussed in Chapter 3),

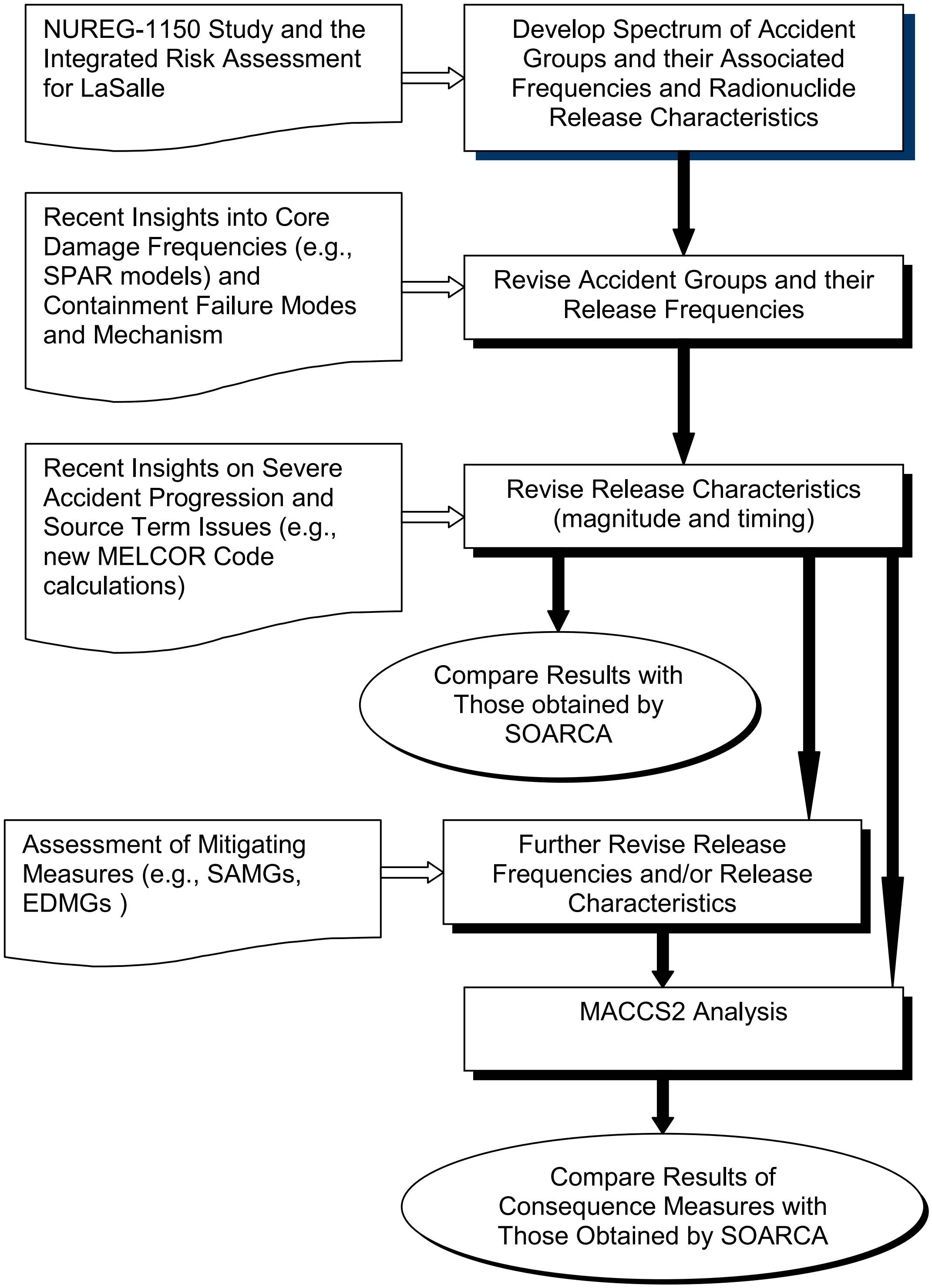

Figure 15: Elements of the Proposed Approach to Update the Results of Earlier Level-3 PRAs for Comparison with SOARCA

Table 9: Revised mean conditional probability of accident progression bins at Surry

Summary Accident progression Bin Group	Summary PDS Group (Mean Core Damage Frequency)							
	Internal initiators (4.1E-05)						Fire (1.1E-05)	Seismic LLNL (1.9E-04)
	LOSP (2.8E-05)	ATWS (1.4E-06)	Transients (1.8E-06)	LOCAs (6.1E-06)	ISLOCA (1.6E-06)	SGTR (1.8E-06)		
Early CF	-- (0.008)[(a)]	-- (0.003)	-- (0.001)	-- (0.006)			-- (0.018)	0.082 (0.096)
Late CF	0.084 (0.079)	0.046 (0.046)	0.014 (0.013)	0.056 (0.055)			0.305 (0.292)	0.288 (0.280)
Bypass	(0.003)	(0.078)	(0.007)		(1.0)	(1.0)		(0.001)
No CF	0.913 (0.909)	0.876 (0.873)	0.979 (0.979)	0.944 (0.939)			0.695 (0.690)	0.630 (0.624)

(a) Numbers in parentheses are the results of the NUREG-1150 Study.

it can be concluded that, for the internal and fire initiators, the early containment failure for Surry is of very low conditional probability that it is of little or no significance to the overall risk. However, this conclusion does not hold for the seismic initiators. The results of the NUREG-1150 Study indicate that, for the seismic initiators, the mean conditional probability of early CF is on the order 0.1 and most of these early failures of the containment are initial failures due to steam generator (SG) and reactor coolant pump (RCP) support failures. The staff has yet to report whether it has addressed this separate failure mechanisms in SOARCA. Revisions to frequencies of plant damage states together with the conditional probability of accident progression bins for each “summary plant damage state” provide a framework for systematic evaluation of selection of the accident groups and their associated containment failure modes and release frequencies in SOARCA.

As an illustrative example, for the purpose of comparing the results of radionuclide releases in SOARCA for internal initiators at Surry with those of the updated NUREG-1150 Study, the representative source terms for Surry, discussed in Section 2.4 (see Table 6), can be refined by defining an additional source term representing steam generator tube rupture (SGTR) plant damage state (PDS GROUP 7 of NUREG-1150 Study for Surry, Internal Initiators). For this additional bypass release category, based on the results of NUREG-1150 for the most probable bin, it is assumed the safety relief valves (SRVs) on the secondary system are stuck open, the containment spray system never operates during the accident, the core-concrete interaction (CCI) takes place promptly following vessel breach (VB), and there is no overlaying water pool to scrub the releases due to CCI. The magnitude of radionuclide releases into the environment for this release category can be obtained by the basic parametric equation used in the NUREG-1150 Study. The mean values of the probability distributions of the source term parameters associated with the corresponding accident progression characteristics can be used in these calculations. Table 10 provides a summary of the frequencies and magnitudes of Iodine releases into the environment for representative source terms for Surry. The mean fractional iodine release of 0.20 for the SGTR representative source term is based on the mean value reported for the most probable bin within that plant damage state group (NUREG/CR-4551, Vol. 1, Table 3.3-7) [49]. A comparison of frequency distribution (CCDF) of iodine release predicted by the NUREG-1150 Study for Surry with that obtained from these representative source terms is shown in Figure 16.

Based on discussions provided earlier, for internal initiators, the early containment failure for Surry is of very low probability that it is of little or no significance to the overall risk. Therefore, a representative source term associated with the early containment failure can be discarded. It should be noted that each representative source term represents many accident progression bins with different frequencies and release magnitudes within the accident progression group it represents. There is an overlap of release magnitudes between the accident progression groups. Therefore, by eliminating the representative source term associated with early containment failure, care must be taken to represent other low probability and high release bins. The results of the NUREG-1150 Study for Surry (NUREG/CR-4551, Vol. 1 Table 2.5.4) [49] indicate that for the bypass accident progression group due to interfacing-system LOCA (Event V), there is a conditional probability of ~0.15 that there is no water in the path of the radionuclide releases before entering into

Table 10: Frequencies and Magnitudes of Iodine Releases for Representative Source Terms for Surry (Internal Initiators)

Release Category	Summary PDS Group	Containment Failure Time	Containment Failure Mode	Frequency		Fractional Release for Iodine Group
				Based on NUREG-1150 Study	Revised Based on Results of SPAR Model and no Early Failure of Cont.	
RSUR1	LOSP	CF at VB (ECF)	Rupture	2.9E-07	----	0.35
RSUR2	LOSP	Late CF (LCF)	Leak	2.4E-06	1.5E-07	0.06
RSUR3	LOSP	No CF (NCF)	No CF	3.3E-05	1.95E-06	3.E-05
RSUR4	Bypass (V)	NCF	Bypass	1.6E-06 Wet (~85%) Dry (~15%)	3.5E-07 Wet (~3.0E-07) Dry (~5.0E-08)	0.115 0.115 (Wet) 0.37 (Dry)
RSUR5	Bypass (SGTRs)	NCF	Bypass	1.8E-06	5.5E-07	0.2

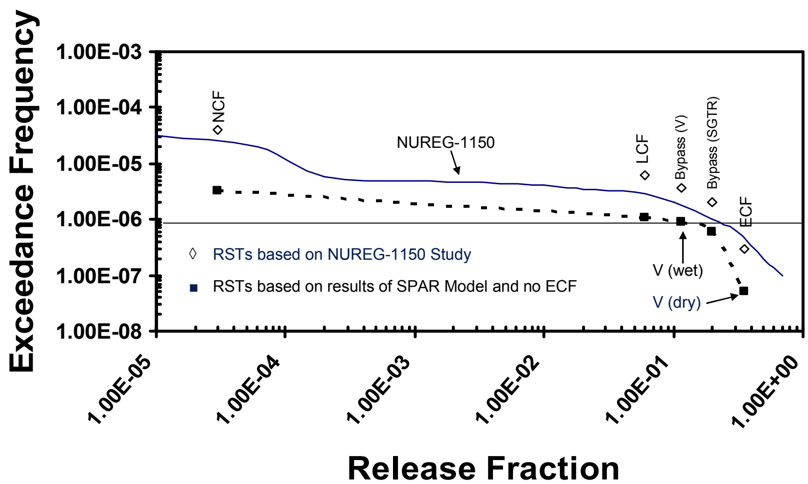

Figure 16: Comparison of frequency distribution (CCDF) of iodine release predicted by NUREG-1150 Study for Surry with that obtained from the representative source terms (RSTs) and their revised frequencies based on results of SPAR model and recent insights on early containment failure mechanisms

the environment (dry releases). In the absence of the decontamination effect of water in the release path, the release magnitude would be higher. An additional source term representing this bypass sub-group can be defined. The magnitude of radionuclide releases into the environment for this release sub-category can be obtained by the basic parametric equation used in the NUREG-1150 Study. The results of the revised frequencies of the representative source terms for Surry is also presented in Table 10. The mean fractional iodine release of 0.37 shown in Table 10 for the V bypass accident progression sub-group (Dry releases) is based on the mean value reported for the most probable bin within that sub-group (NUREG/CR-4551, Vol. 1, Table 3.3-4). The frequency distribution (CCDF) of iodine release for Surry obtained from revision to representative source terms, based on the results of SPAR model and recent insights on early containment failure mechanisms, is also shown in Figure 16. As shown in Figure 16 (see also Table 10), the revised frequency of the V bypass accident progression sub-group (dry releases) is about 5.0 x 10^{-8}. The staff has yet to report whether it has included the releases associated with such a bypass sequence for SOARCA. It is not clear how such an accident progression group can be excluded from SOARCA based on the screening frequency (10^{-7} for bypass sequences) argument, without the benefit of such information obtained from the Level-2 PRAs.

The release characteristics (magnitude and timing) of the representative source terms can also be revised based on the more recent insights on severe accident progression and source term issues (e.g., new MELCOR code calculations). The staff has not yet provided any document on accident progression and source term analysis using severe accident code MELCOR. It is not clear how the state-of-knowledge uncertainties associated with the accident progression phenomena and source term analyses are addressed.

The revised representative source terms and their associated frequencies in the form of CCDFs provide a possible framework for evaluating the selection of the accident sequences and the magnitude of the radionuclide releases into the environment obtained as a part of the SOARCA process. These radionuclide release characteristics are functions of the design and operating procedures of the plant and, unlike the results of consequence measures, they are independent of the population distribution, meteorology, and the emergency protective assumptions at the reactor site.

The frequencies and/or release characteristics of the representative source terms may be further revised based on the results of the assessment of the effectiveness of mitigating measures. Such mitigating measures are based on Severe Accident Management Guidelines (SAMGs) and other new procedures, such as mitigating measures resulting from B.5.b and other like programs that were not in place when the earlier consequence studies, including the NUREG-1150 Study, were performed.

Development of a simplified accident progression event tree which only considers significant high level issues, based on the insights from the NUREG-1150 Study and more recent advances in understanding of the severe accident issues, containment failure mechanisms, and effectiveness of potential mitigating measures would facilitate the revision to release frequencies and providing a systematic approach to justify the selection of sequences for SOARCA. Such simplified accident progression event tree would be particularly useful for assessment of the effectiveness of mitigating measures resulting from SAMGs and other new programs (e.g. B.5.b).

Using MACCS2 code, Consequence analyses could be performed to determine consequence measures (e.g., population dose, early fatalities, and latent cancer fatalities) for each representative source terms. The frequency distributions of consequence measures in the form of CCDFs provide a useful framework for comparison with the results of SOARCA, as well as for their comparison with historical results.

5. SUMMARY AND CONCLUSIONS

The probability and offsite consequences of severe reactor accidents have been the subject of considerable interest and study since the earliest days of reactor development. Several systematic studies have been made in the past to search out a large spectrum of accidents and to use quantitative techniques to estimate the probabilities, radionuclide release characteristics (source terms), and potential offsite health consequences. An overview of major contributions to consequence assessment was presented to provide a historical perspective and insights on previous state-of-the-art analyses of the consequences of severe reactor accidents.

The staff is currently implementing its plan for developing state-of-the-art reactor consequence analyses, using a "novel analysis procedure that differs substantially from previous state-of-the-art analyses of the consequences of severe reactor accidents" [13]. Although performing Level-3 PRAs for the pilot plants is the best way to benchmark SOARCA methodology, it is feasible to use the results and insights from the NUREG-1150 Study and Integrated Risk Assessment for LaSalle, together with more recent advances in understanding of the severe accident issues and containment failure mechanisms, and develop a simplified, yet systematic and defensible, approach to update the results of such earlier Level-3 PRAs for comparison with aspects of SOARCA results.

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