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Historical Perspectives and Insights on Nuclear Reactor Accident Consequence Analyses

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History balances the frustration of "how far we have to go" with the satisfaction of "how far we have come"

Lewis F. Powel, Jr.

(Former Associate Justice of the US Supreme Court)

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Background

- The NRC staff is currently implementing its plan for developing state-of-the-art reactor consequence analyses (SOARCA)
- This presentation is based on a white paper prepared for use by the ACRS in its dialogue with the staff regarding the feasibility of using a simplified approach for updating results from earlier Level-3 PRAs such as NUREG-1150 for comparison with aspects of SOARCA results

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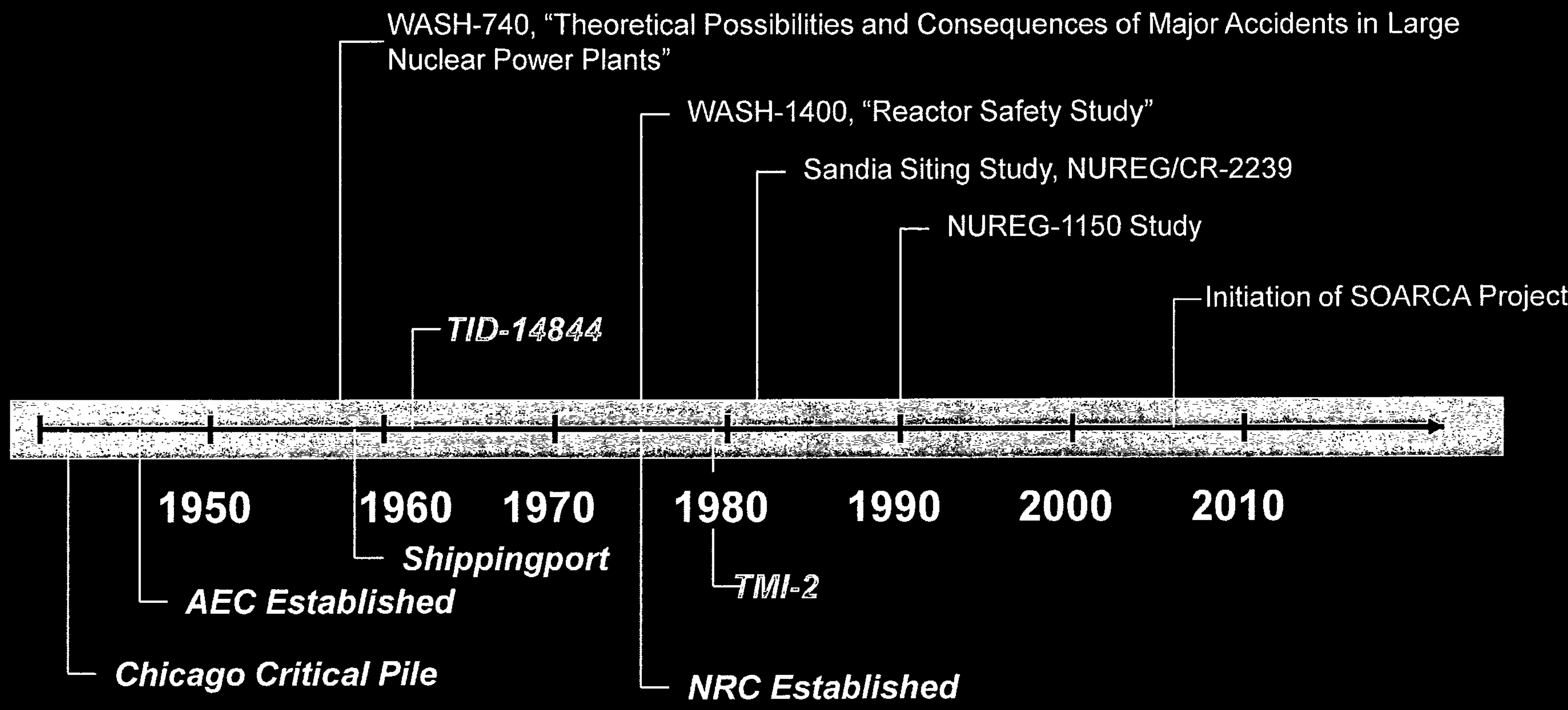
Objectives

- To provide historical perspectives and insights on previous state-of-the-art analyses of the consequences of severe reactor accidents
- To discuss the feasibility of using a simplified, yet systematic and defensible, approach to benchmark many aspects of SOARCA

Timeline of Major Studies of Reactor Accident Consequences



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WASH-740

- The first estimates of consequences of severe accidents were published in the 1957 AEC report (WASH-740), "Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants"
  - An attempt to provide upper bounds of the potential public hazards resulting from certain severe hypothetical accidents
  - No use of 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.

Reactor Safety Study (WASH-1400)

- The first systematic attempt to provide realistic estimates of public risk from potential accidents in commercial nuclear power plants
  - included analytical methods for determining both the probabilities and consequences of various accident scenarios
  - Two specific reactor designs were analyzed, Surry and Peach Bottom
  - Calculations were performed for a number of accident sequences and the results for these calculations were used to define a series of release categories (nine for PWR and five for BWR) into which all of the identified accident sequences could be distributed.

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Post TMI-2 Review of Source Term Technical Basis

- Following the publication of WASH-1400 and the accident at TMI-2, work initiated to review the predictive methods for calculating fission product release and transport
- Review resulted in several conclusions that represented significant departure from WASH-1400 assumptions including the suggestion that cesium iodide (CsI) will be the expected predominant iodine chemical form under most postulated LWR accident conditions
- These studies formed the basis for development of a generic set of radiological releases, characterized as Siting Source Terms (denoted SST1-5), used in Sandia Siting Study (NUREG/CR-2239)

Brief Descriptions of the Characteristics of the Accident Groups

(NUREG-0771, p. 8)

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)

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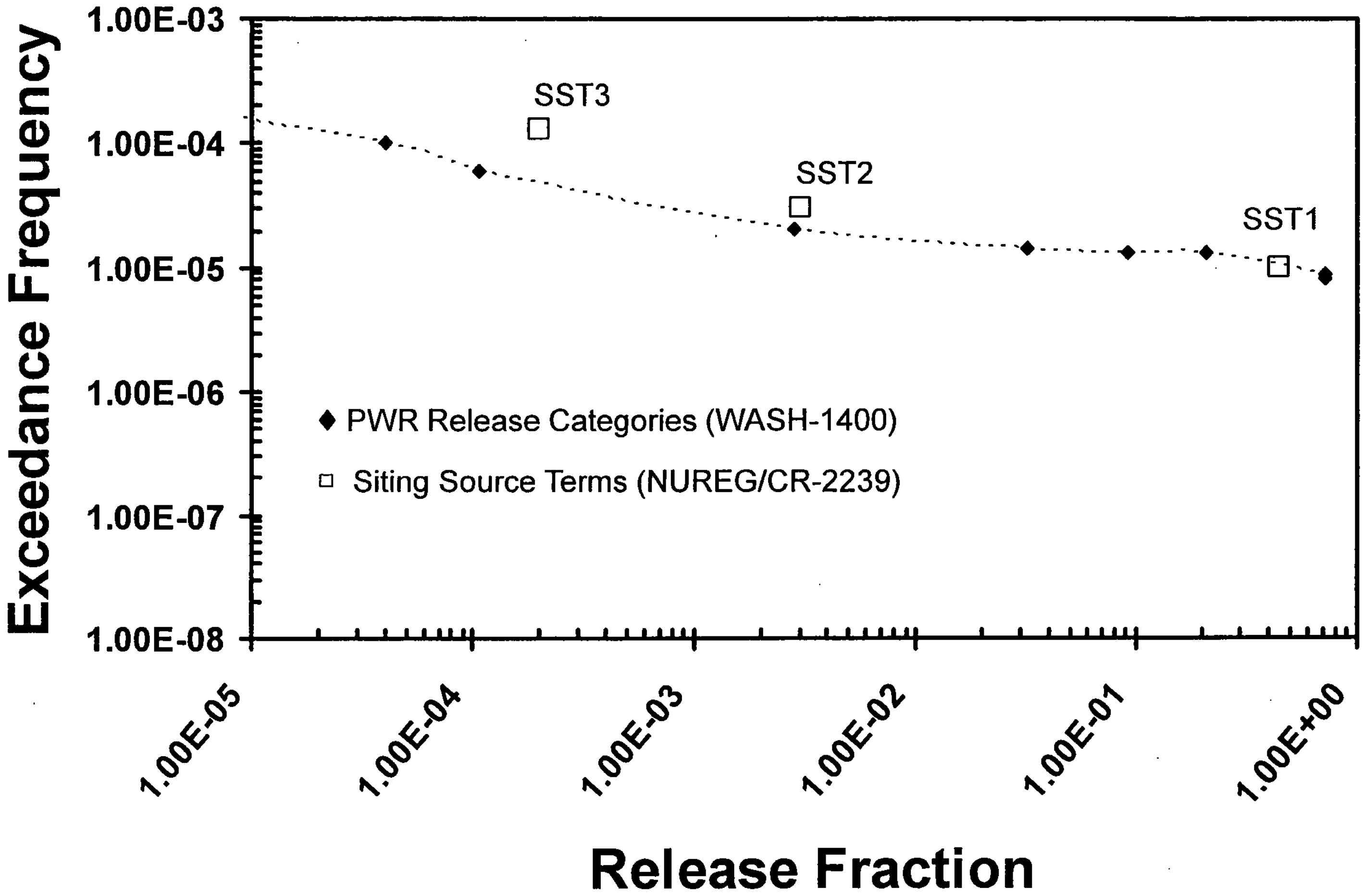
Sandia Siting Study (NURG/CR-2239)

- Used Siting Source Terms (SSTs) at 91 existing or proposed reactor sites to perform accident consequence analyses
- Detailed PRAs were not performed for all reactors. Based on available PRAs at the time, NRC suggested the following representative probabilities for the SSTs
  - SST1 1×10^{-5}
  - SST2 2×10^{-5}
  - SST3 1×10^{-4}

Frequecy of Release for Iodine

(Comparison of WASH-1400 PWR Release Categories and SSTs)



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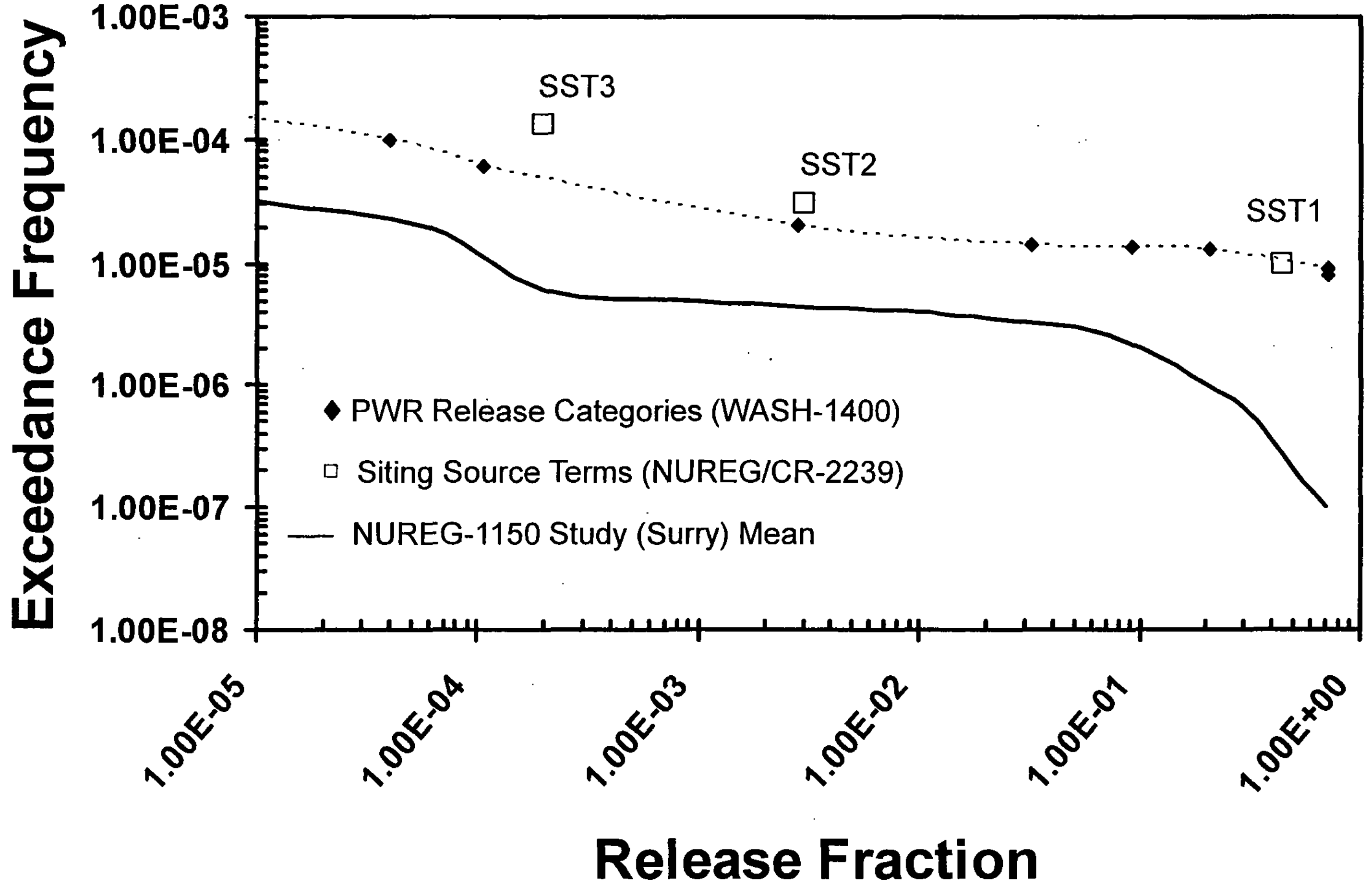
NUREG-1150 Study

- A major effort to put into a risk perspective the insights into system behavior and phenomenological aspects of severe accidents
  - inclusion of the uncertainties in the calculations of core damage frequency and risk that exist because of incomplete understanding of reactor systems and severe accident phenomena
- Five specific commercial nuclear power plants were analyzed :
  - Surry, a 3-loop Westinghouse PWR with a subatmospheric containment
  - Zion, a 4-Loop Westinghouse PWR with large dry containment
  - Sequoyah, a 4-loop Westinghouse PWR with ice-condenser containment
  - Peach Bottom, a BWR-4 reactor with a Mark I containment
  - Grand Gulf, a BWR-6 reactor with a Mark III containment

Frequecy of Release for Iodine Group

(Comparison of WASH-1400 PWR Release Categories, SSTs, and NUREG-1150)



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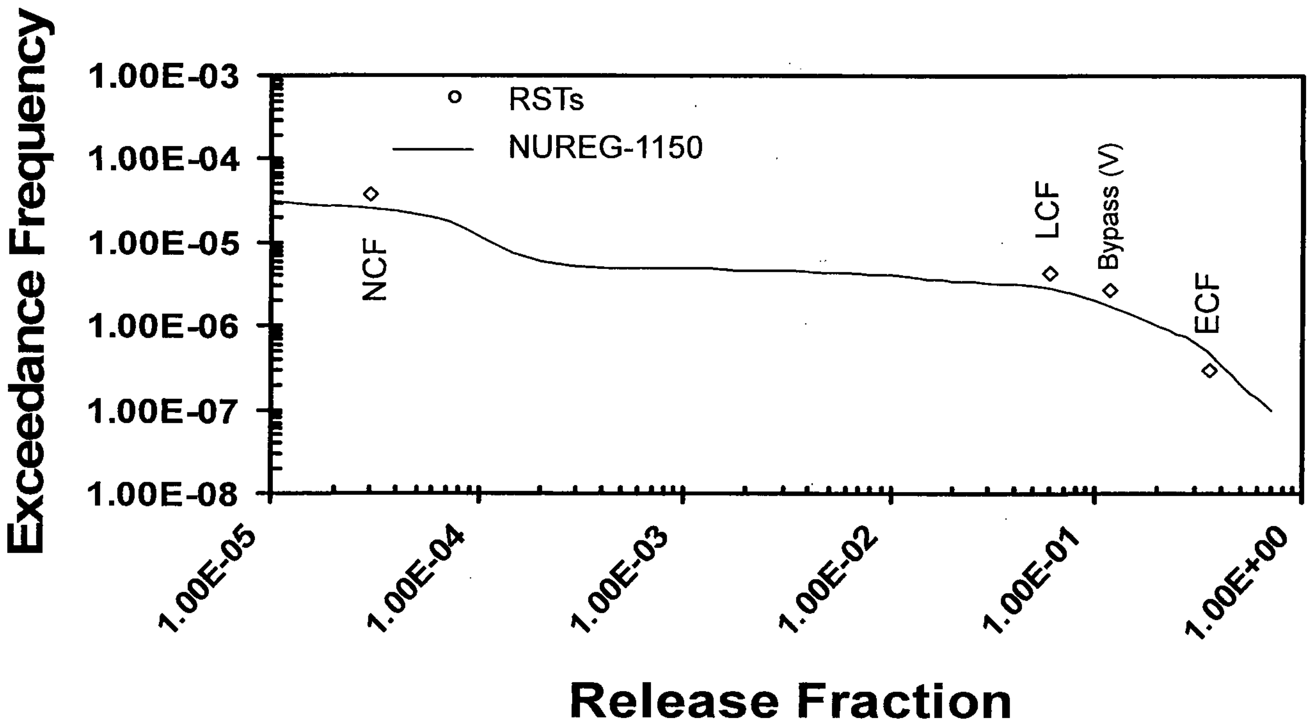
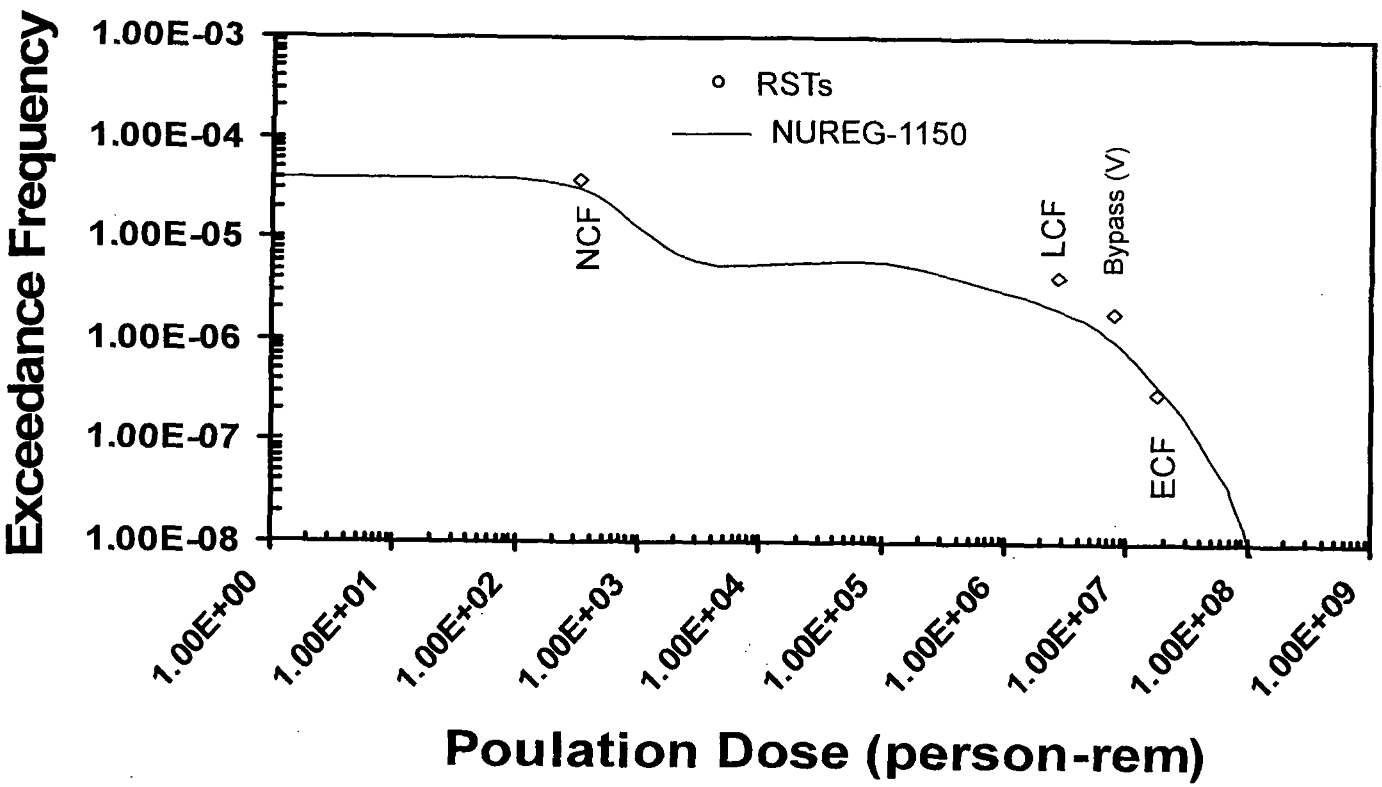
Reassessment of Selected Factors Affecting Siting of Nuclear Power Plants (NUREG/CR-6295)

- A series of probabilistic consequence assessment calculations were performed in support of an effort to re-assess reactor siting
- Insights from NUREG-1150 and the LaSalle independent risk assessment studies were used to develop representative source terms
  - A small set of source terms (4 to 7 for each plant) based on dominant plant damage states, accident progression groups and the associated release characteristics were developed for each reactor design to represent the full spectrum of severe accidents
- Examined consequences in a risk based format consistent with the quantitative health objectives (QHOs) of the NRC's Safety Goal Policy

Frequency of Release for Iodine Based on Representative Source Terms (RSTs) for Surry Internal Events



Frequency of Population Dose to Entire Region at Surry



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Recent Advances in Understanding of Severe Accident Phenomenology and Containment Failure Mechanisms

- Since the completion of NUREG-1150 Study, more analytical and experimental studies have been performed to address many severe accident issues including:
  - Direct Containment Heating (DCH) Issue
  - "Mark I Liner Attack" Issue
  - In-vessel steam explosion (alpha mode failure)

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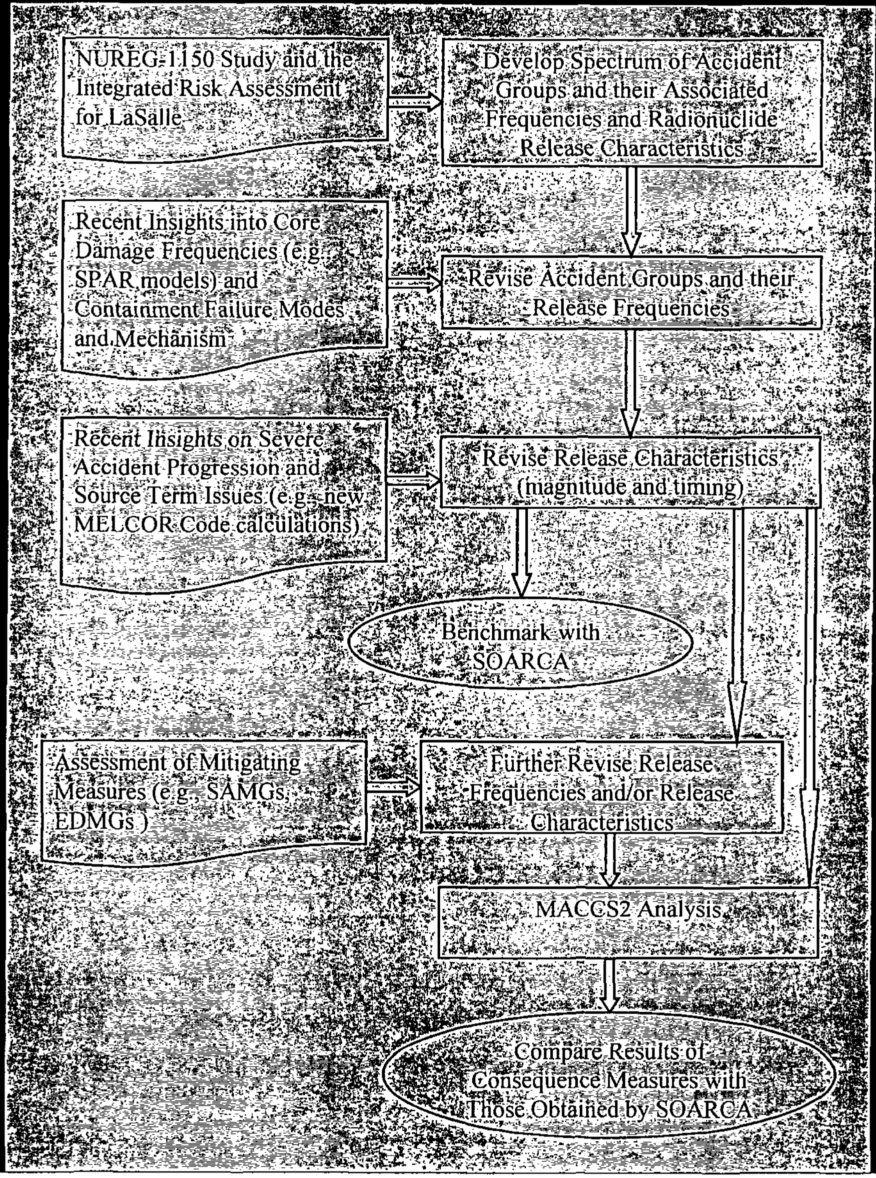
A Simplified Approach to Update the Results of Earlier Level-3 PRAs

- Although performing Level-3 PRAs 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.

Elements of the Proposed Approach



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Impact of current knowledge and understanding of early containment failure on NUREG-1150 results for the 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.

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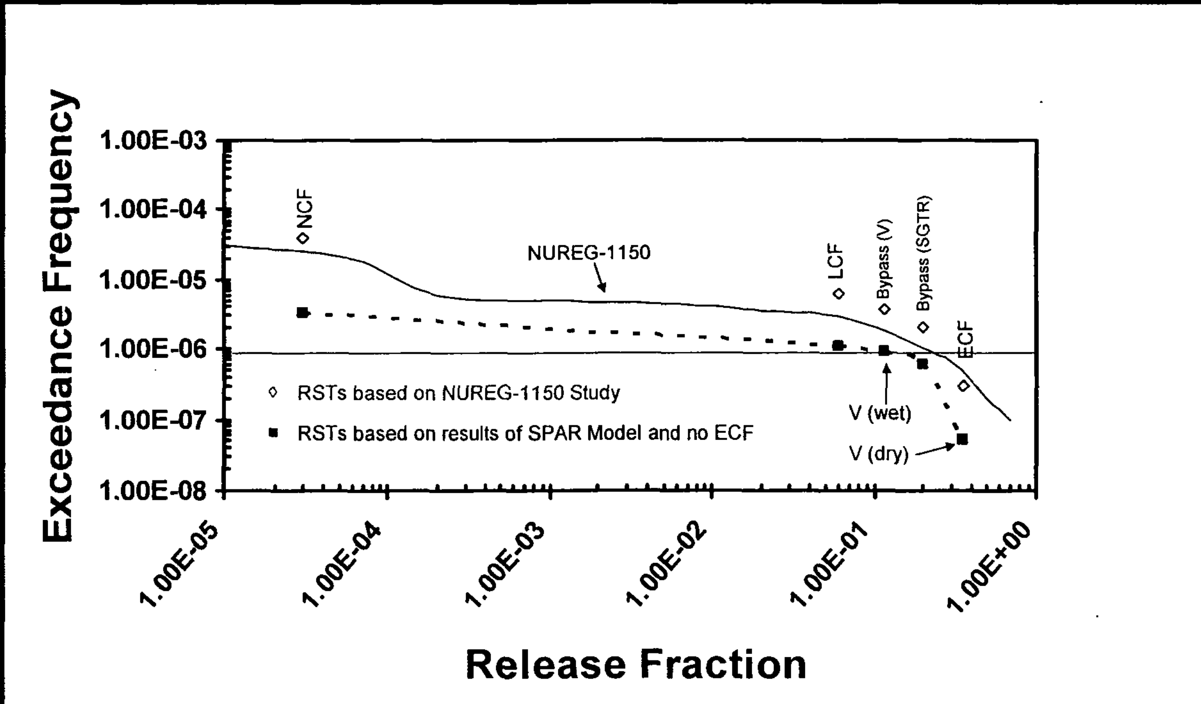
Frequencies and Magnitudes of Iodine Releases for Representative Source Terms for Surry (Internal Initiators)

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

Comparison of frequency distribution (CCDF) of iodine release predicted by NUREG-1150 Study for Surry with that obtained from the results of SPAR model and the recent insights on early containment failure mechanisms (Internal Events)



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Summary and Conclusion

- An overview of major contributions to consequence assessment was presented to provide historical perspectives and insights on previous state-of-the-art analyses of the consequences of severe reactor accidents
- 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 benchmark many aspects of SOARCA