UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

9 MAR 1987

Docket Nos.: 50-327
and 50-328

LICENSEE: Tennessee Valley Authority

FACILITY: Sequoyah Nuclear Plant

SUBJECT: MINUTES OF FEBRUARY 13, 1987 MEETING TO DISCUSS NUREG-1150

On February 13, 1987, members of the staff met with representatives from the Tennessee Valley Authority (TVA) and several other utilities. The purpose of the meeting was to provide TVA with a preliminary overview of NUREG-1150, "Reactor Risk Reference Document," and describe the Sequoyah results. Enclosure 1 is a list of attendees and Enclosure 2 is a copy of the staff's presentation.

Joseph J. Holonich

Joseph J. Holonich, Project Manager
PWR Project Directorate #4
Division of PWR Licensing-A

Enclosures: As stated

cc: See next page

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/S/

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Division of PWR Licensing-A

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Sequoyah Nuclear Plant

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MEETING SUMMARY DISTRIBUTION

Docket File
NRC PDR
L PDR
NSIC
PRC System
PWR#4 Reading File
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OGC-Bethesda
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ACRS (10)
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NRC Participants
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C. Tinkler
F. Eltawila
S. Newberry
S. Long
W. Lyon
C. Rossi
B. Morris

bcc: Licensee & Service List

Enclosure 1

ATTENDEES

NAME	ORGANIZATION
Joe Youngblood	NRC/Licensing
Joe Holonich	NRC/Licensing
John Austin	NRC/OCM
R. Gridley	TVA - Nuclear Safety & Licensing
Vince Noonan	NRC/Licensing
Milton Alexich	UP Nuclear OPS/Americal Elec. Power
Goutam Bagchi	NRC/NRR/Engineering Branch
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C. Tinkler	NRC/NRR
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Frank Carson	TVA/Public Information
William Hannum	TVA - Chairman, NSRB
Scott Newberry	NRC/NRR
Steve Long	NRC/NRR
Warren Lyon	NRC/NRR/PWR-A
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T.A. Ippolito	TVA Licensing

ENCLOSURE 2

METHODOLOGY

- o ABBREVIATED ANALYSIS OF FREQUENCY OF SEVERE ACCIDENTS, EMPLOYING INSIGHTS FROM PREVIOUS STUDIES TO ALTER DEPTY OF ANALYSIS.
- o EXTREMELY DETAILED CONTAINMENT EVENT TREES,
- o ANALYSIS OF SEVERE ACCIDENT PHENOMENOLOGY USING STATE-OF-THE-ART TOOLS,
- o CONSEQUENCE ANALYSES EMPLOYING IMPROVED MODELING AND LATEST HEALTH EFFECTS MODELS,
- o RISK ESTIMATION WITH COMPREHENSIVE UNCERTAINTY ANALYSES

Table D.13 Ranges of Risk Parameters

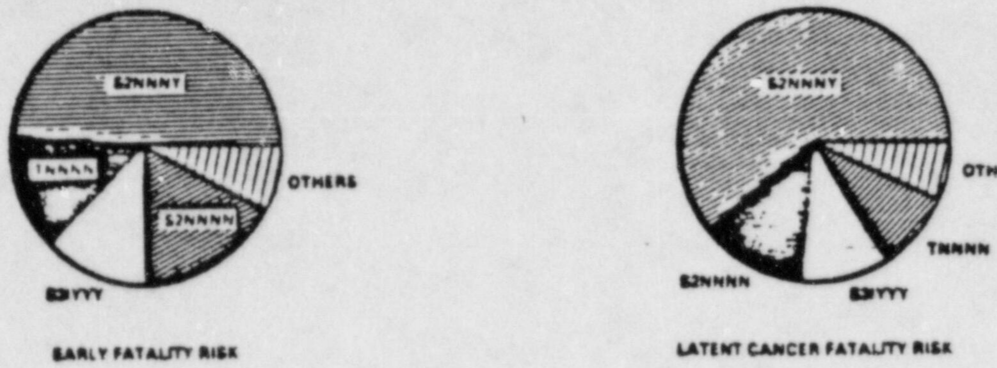
Risk Measure	Lower Bound	Upper Bound
Early Fatalities (/r-yr)	$3.5x10^{-6}$	$3.1x10^{-4}$
Individual Risk of Early Fatalities (/r-yr)	$1.4x10^{-8}$	$3.1x10^{-7}$
Latent Cancer Fatalities (/r-yr)	$4.3x10^{-2}$	$5.1x10^{-1}$
Early Injuries (/r-yr)	$3.6x10^{-5}$	$2.0x10^{-3}$
Population Dose Within 50 miles (person-rem/r-yr)	106.	906.
Offsite Costs ($/r-yr)	4721.	155500.

(r-yr) = reactor year of operation

Table D.3 (Continued)

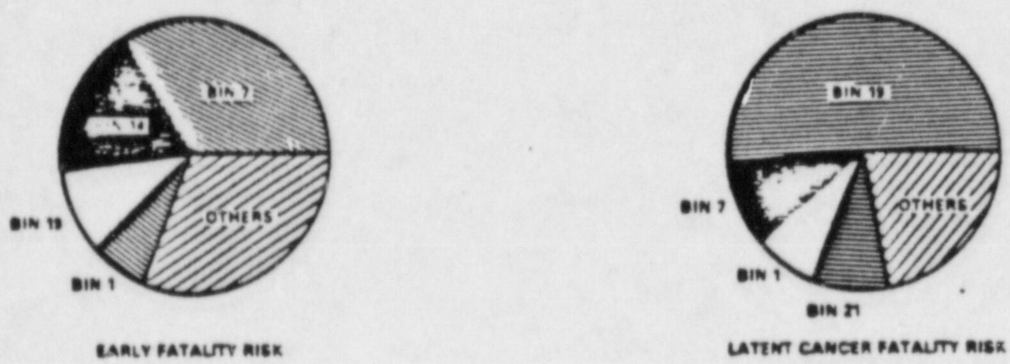
	Issue	Description of Sensitivity Study	Total Core Damage Frequency
10.	Recovery from Small LOCAs	Assume that blowdown of the steam generator secondary is a viable recovery procedure for small LOCAs with failure of high pressure injection where AFW is available.	8.4×10^{-5}
11.	ECCS Operability Following Containment Failure	Assume that ECC systems fail following containment failure.	1.2×10^{-4}
12.	Common-Cause Failure Rates	Use alternative (more pessimistic) interpretation of beta factors for common-cause failure rates.	1.8×10^{-4}
13.	Common-Cause Failure Rates	Eliminate the use of beta factors in modeling common-cause failure rates.	5.4×10^{-5}
14.	Check Valve Failure Rates	Use a more optimistic distribution for failure rates of check valves in interfacing systems LOCA sequences.	1.2×10^{-4}(a)
15.	Check Valve Failure Rates	Use a more pessimistic distribution for failure rates of check valves in interfacing systems LOCA sequences.	1.1×10^{-4}

(a) The change in mean total core damage frequency in this sensitivity case is considered to represent expected variation with a statistical sampling code and is not indicative of the impact of the modified parameters.



PLANT DAMAGE STATES	DESCRIPTION
S2NNNY	Loss of component cooling water system, causing ECCS, containment heat removal and spray failure
S2NNNN	Small LOCA with failure of ECCS and containment systems due to failure of AC power.
S3YYY	Reactor coolant pump seal LOCA with failure of ECCS in recirculation. Containment heat removal, sprays and AC power available
TNNNN	Transient with loss of all AC power. All ECCS and containment systems unavailable.

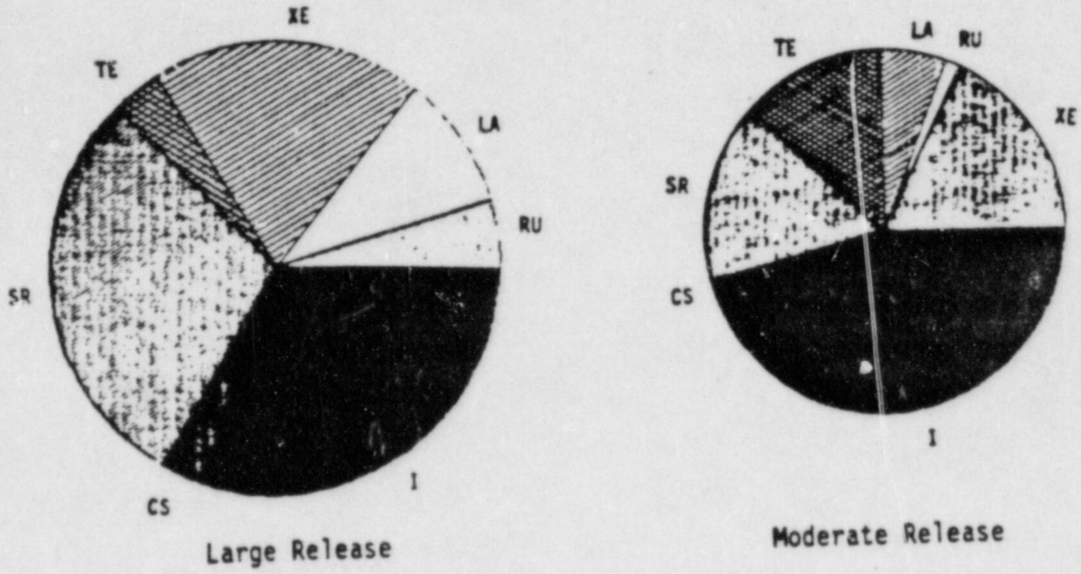
Figure 7.12 Contributions of plant damage states to Sequoyah risk



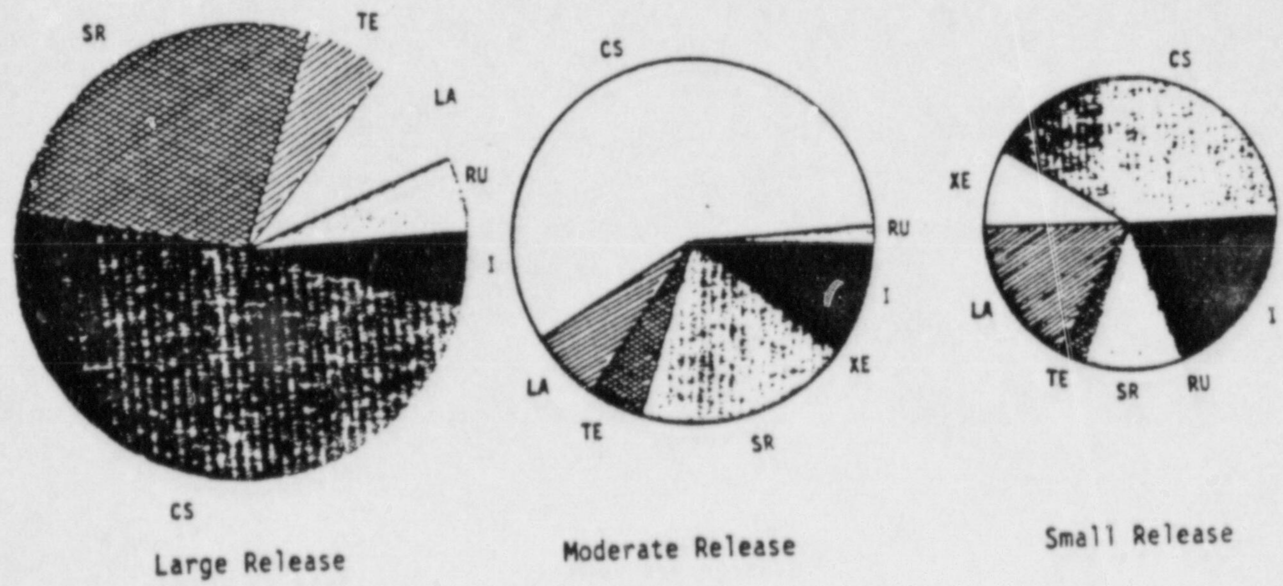
BIN 1	Reactor vessel pressure intermediate to high. Fans and sprays not operating. Cavity dry at vessel breach and ice present.
BIN 7	Reactor vessel pressure intermeidate to high. Fans operate, sprays do not. Cavity dry at vessel breach. Ice present at vessel breach.
BIN 14	Early containment failure. Reactor vessel pressure intermediate to high. Fans operate, sprays do not. Cavity flooded at vessel breach. Ice present at vessel breach. Core-concrete interractions delayed one day.
BIN 19	Containment failure several hours after vessel breach at high or intermediate pressure. Containment sprays failed, but fans working.
BIN 21	Late containment failure, containment sprays continue to operate. No coolable debris bed.

Figure 7.13 Contributions of containment failure bins to Sequoyah risk

Relative Contribution of Isotope Groups to
Early Fatality Consequences



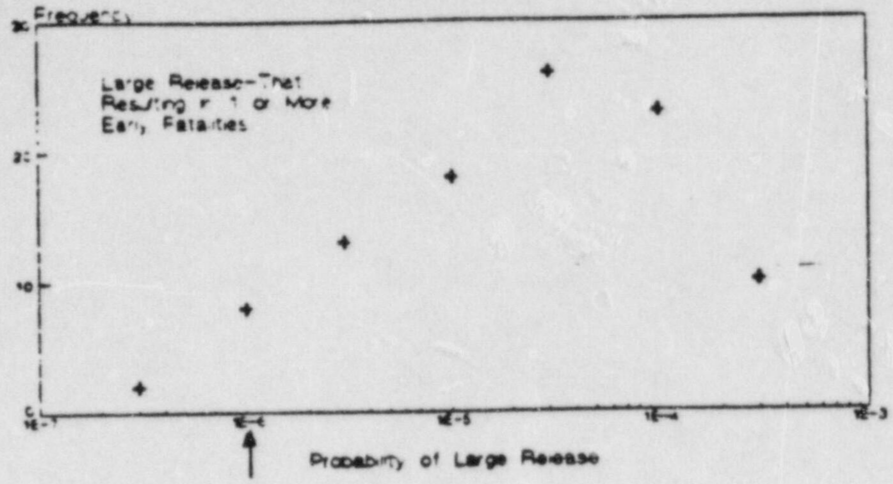
Relative Contribution of Isotope Groups to
Latent Fatality Consequences



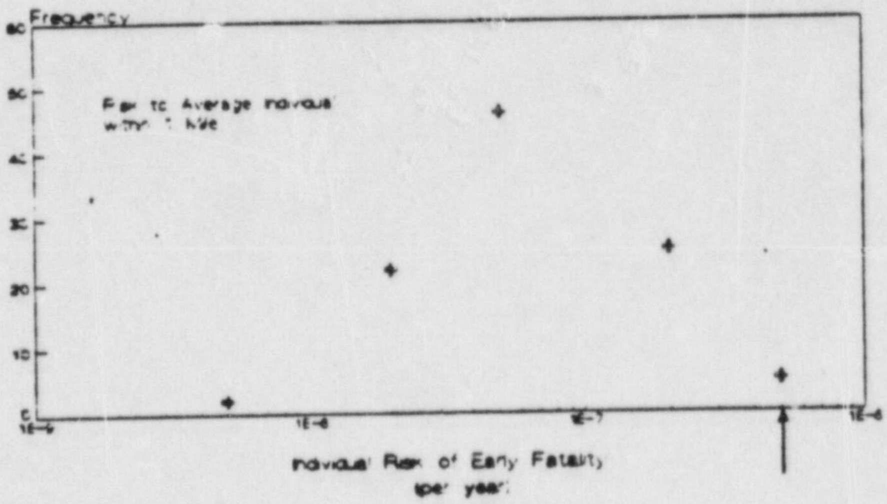
Sequoyah

Figure D.8 Relative Contribution of Isotope Groups to
Early and Latent Fatality Consequences

Large Release Probability



Individual Risk of Early Fatality



Early Containment Failure probability

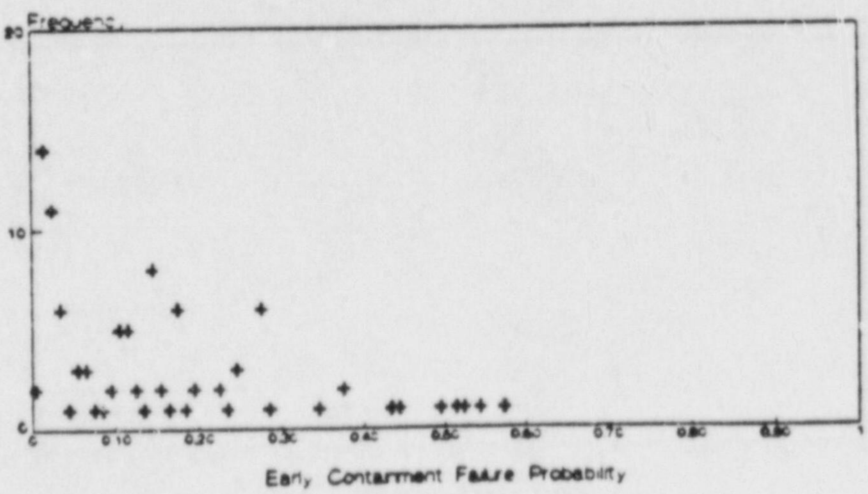


Figure D.13 (Continued)

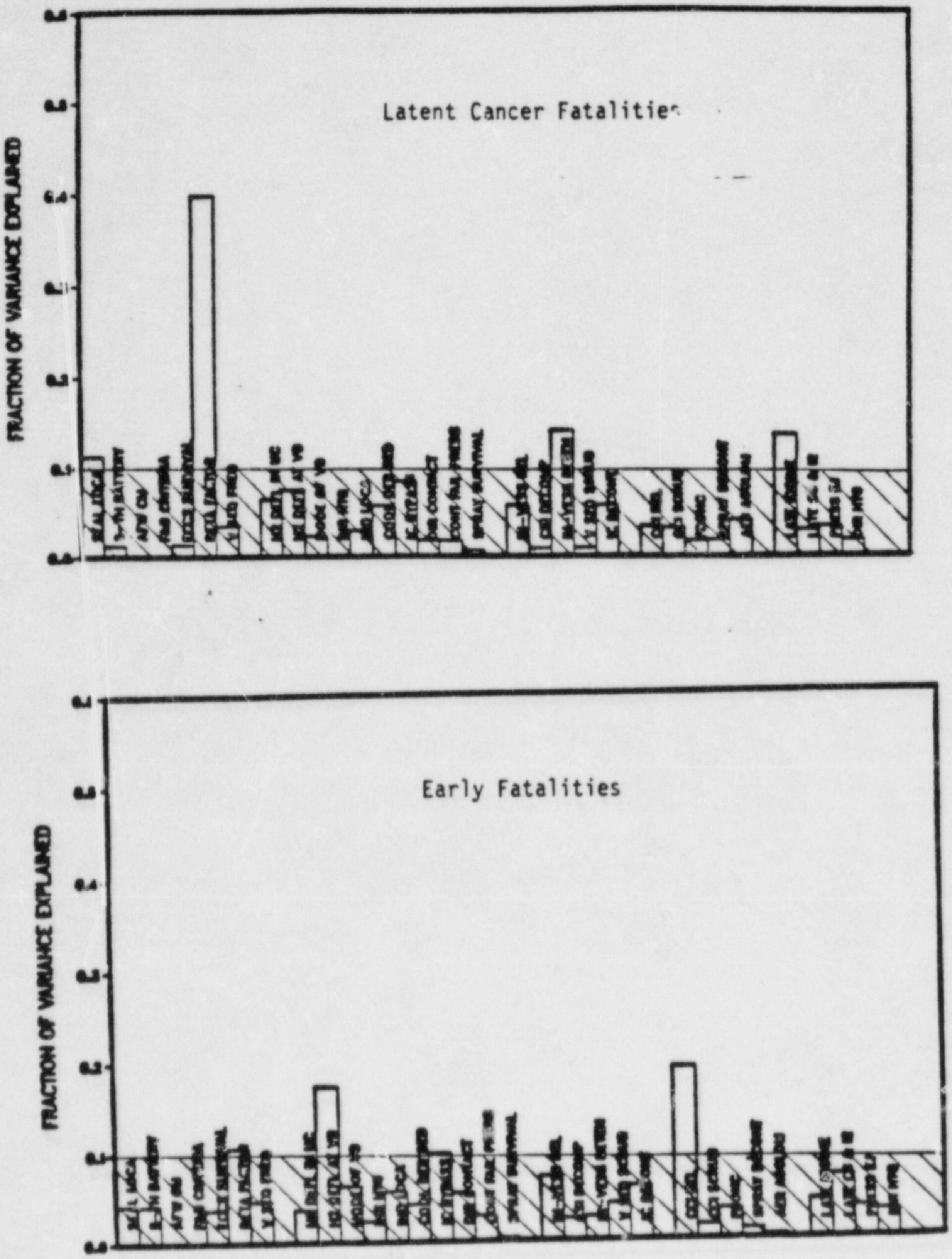


Figure D.10 Results of Rank Regression Analyses for Latent Cancer and Early Fatalities

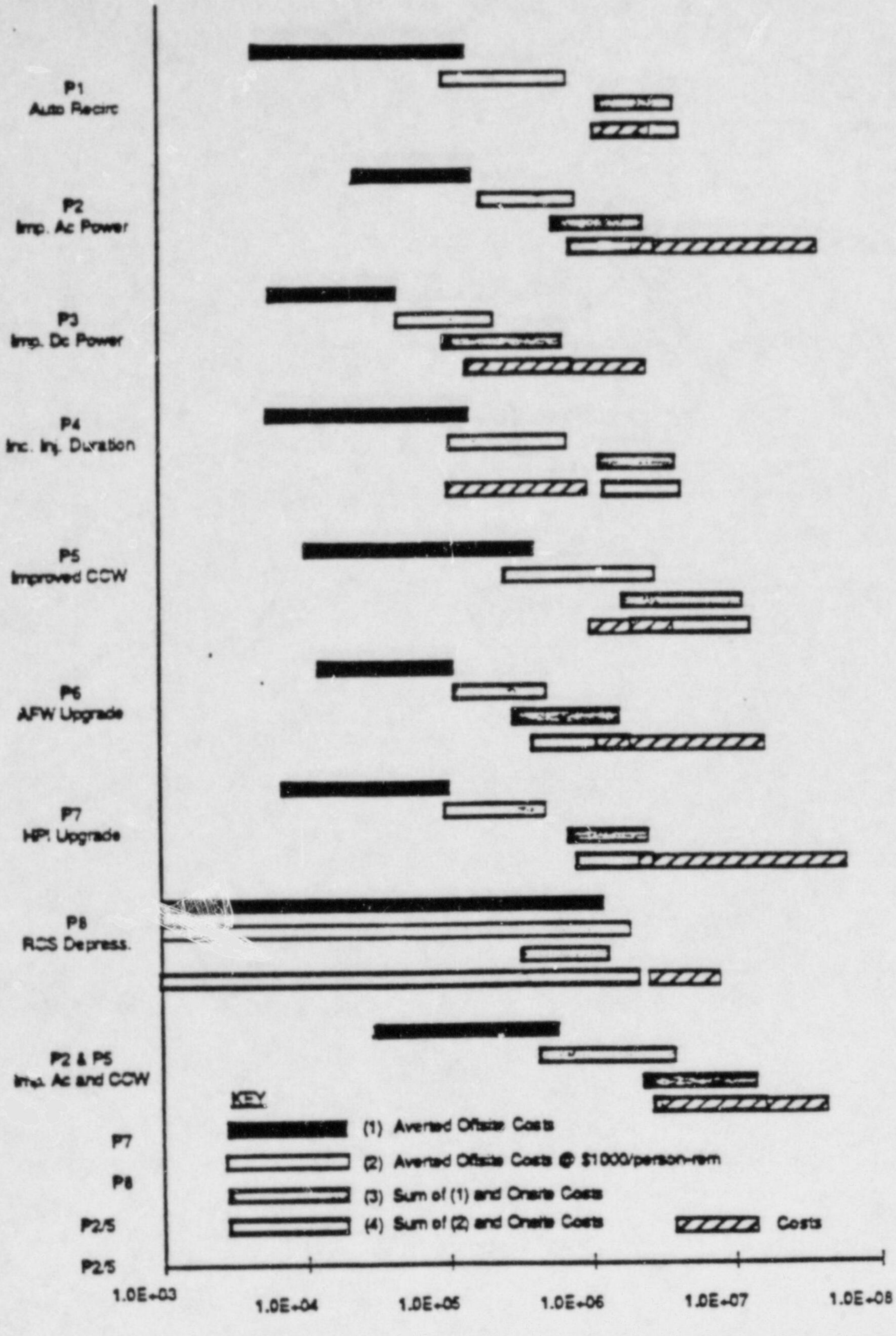


Figure D.11 Comparison of Costs and Averted Risk for Preventive Options

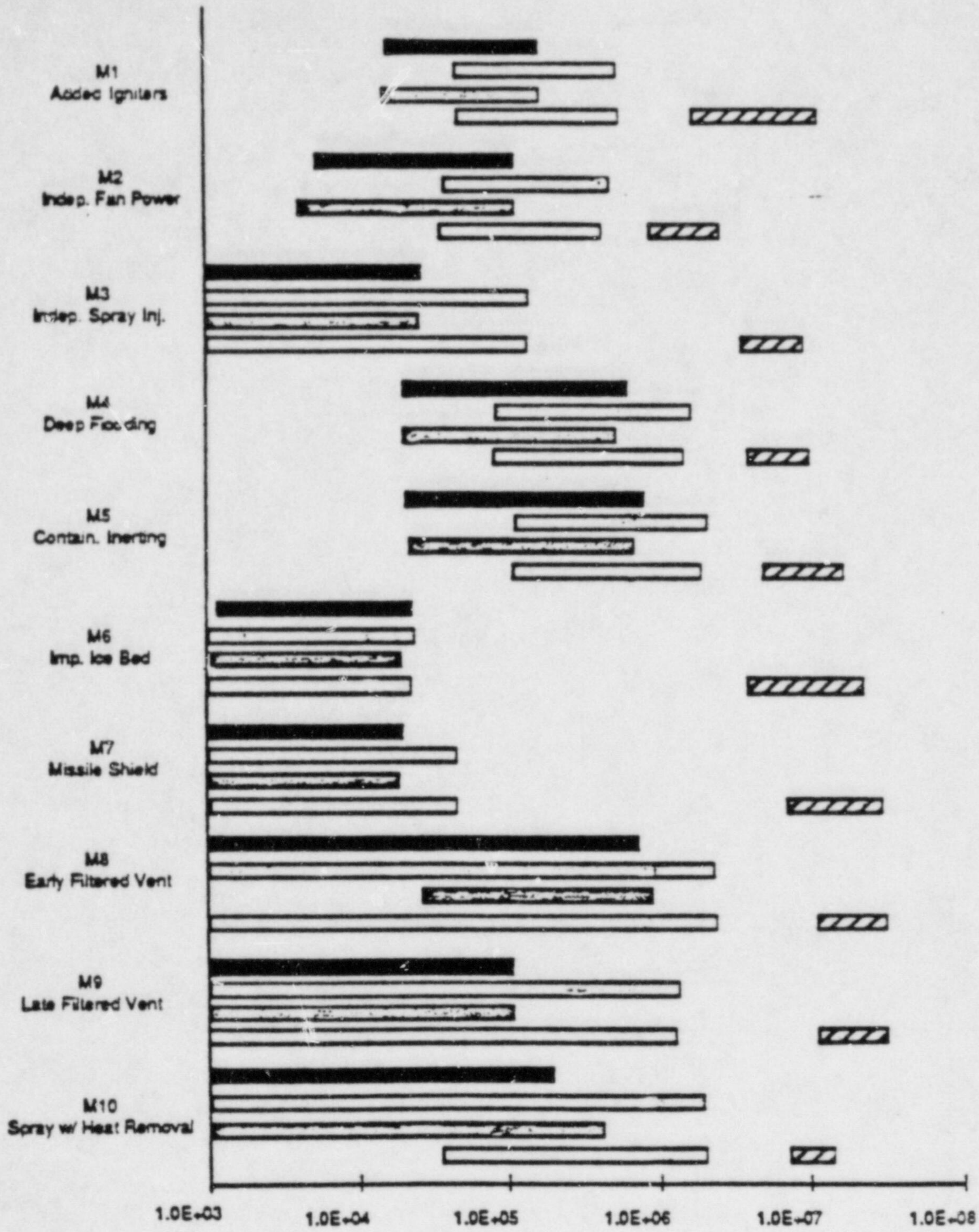


Figure D.12 Comparison of Costs and Averted Risk for Mitigative Options

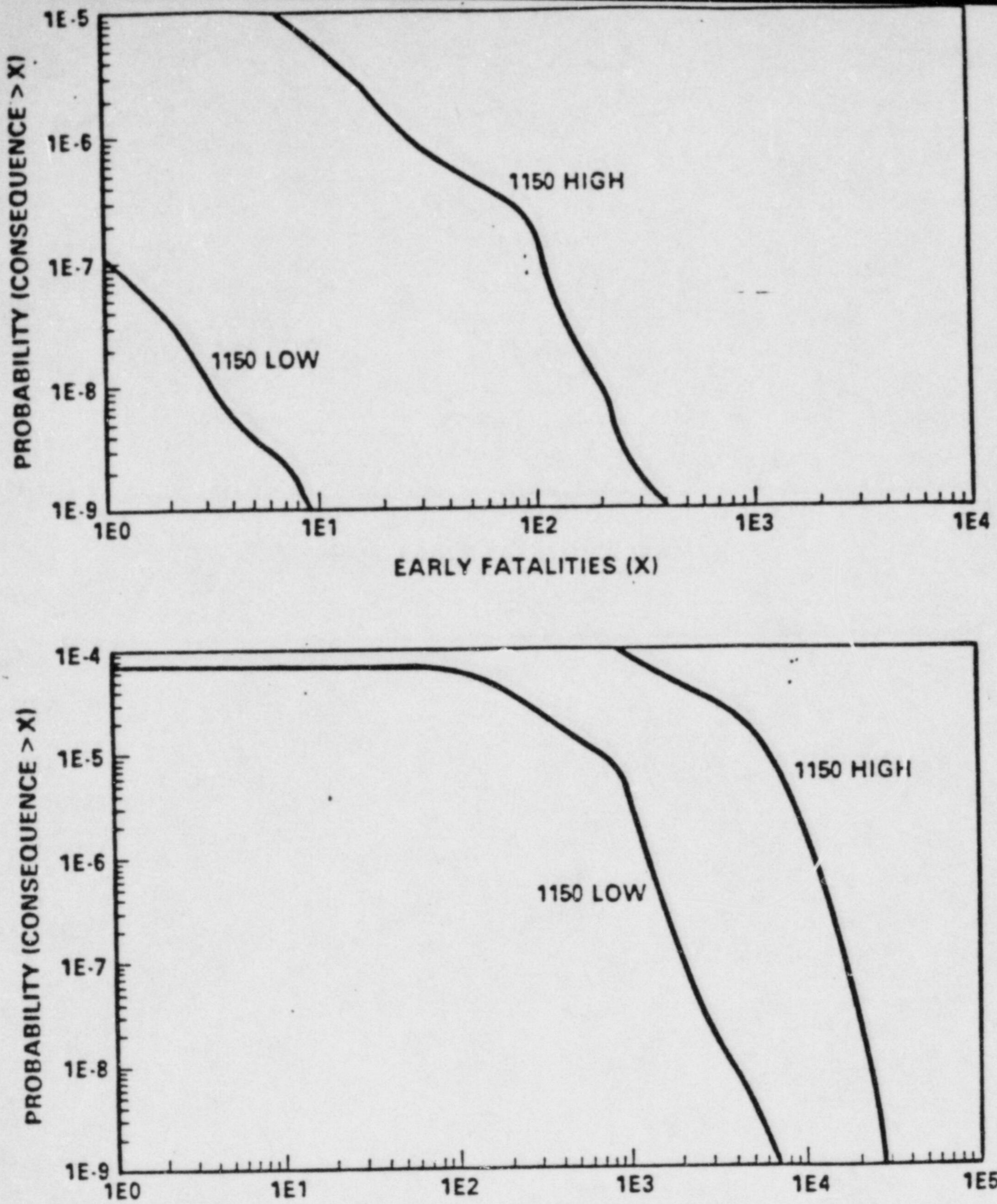


Figure 7.11 NUREG-1150 complementary cumulative distribution function for Sequoyah plant

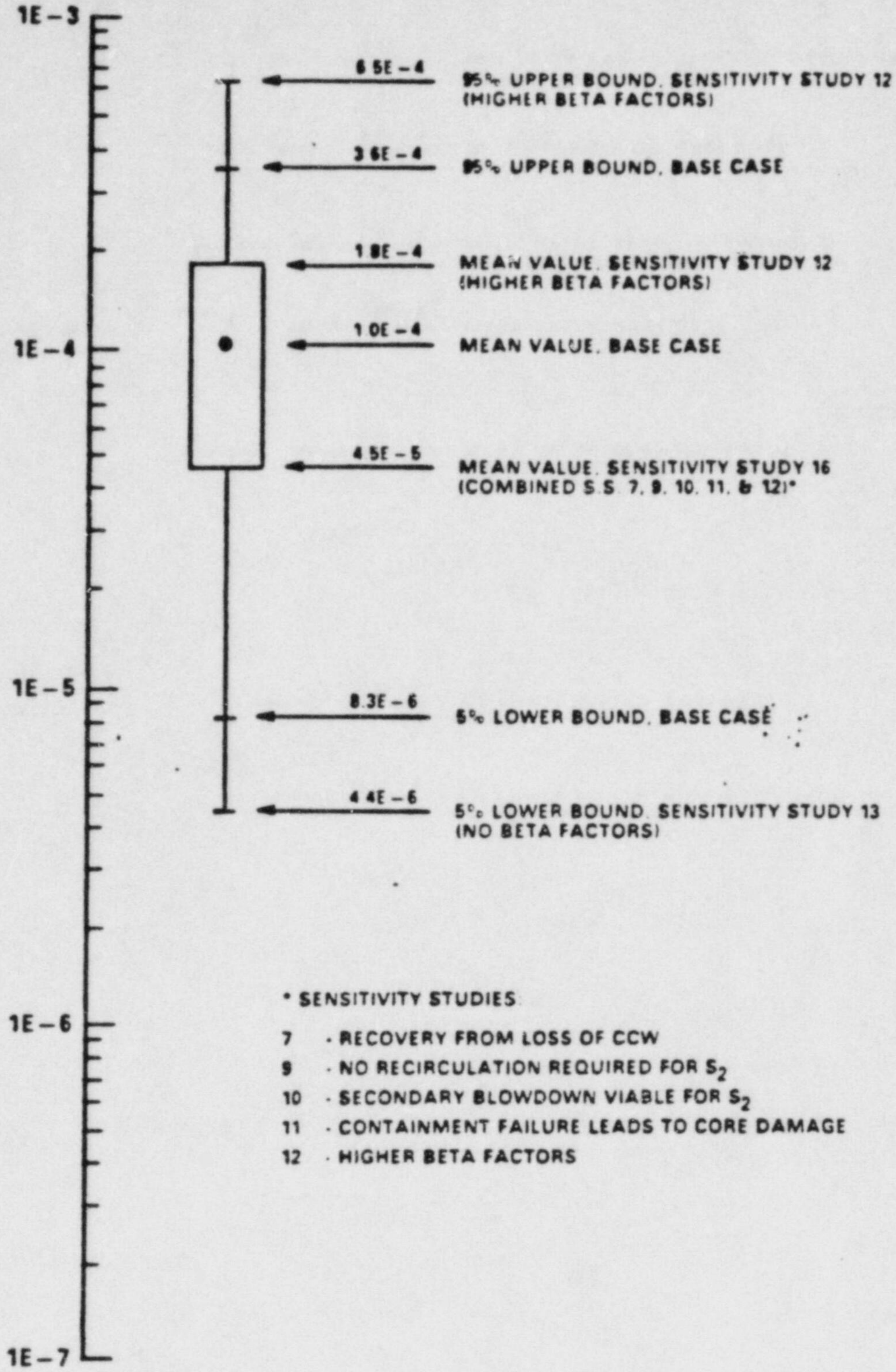
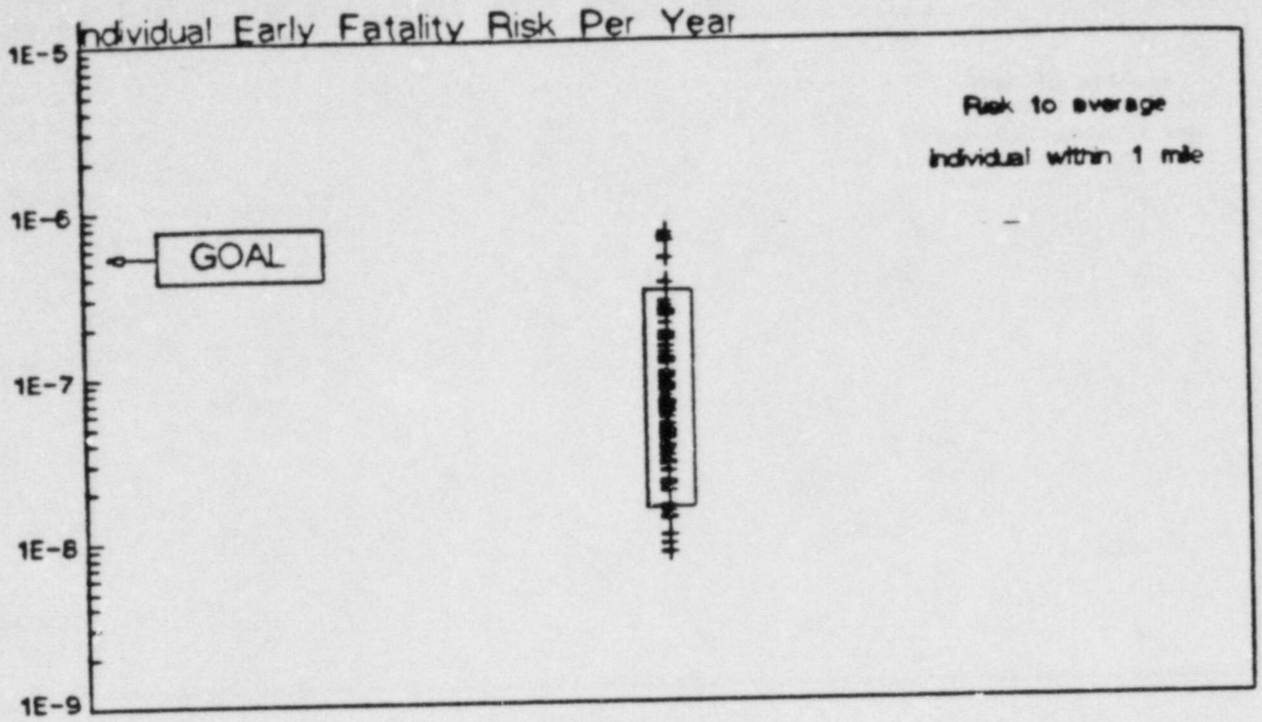


Figure 3.6 "Box-and-whisker" display of uncertainties for core damage frequency at Sequoyah



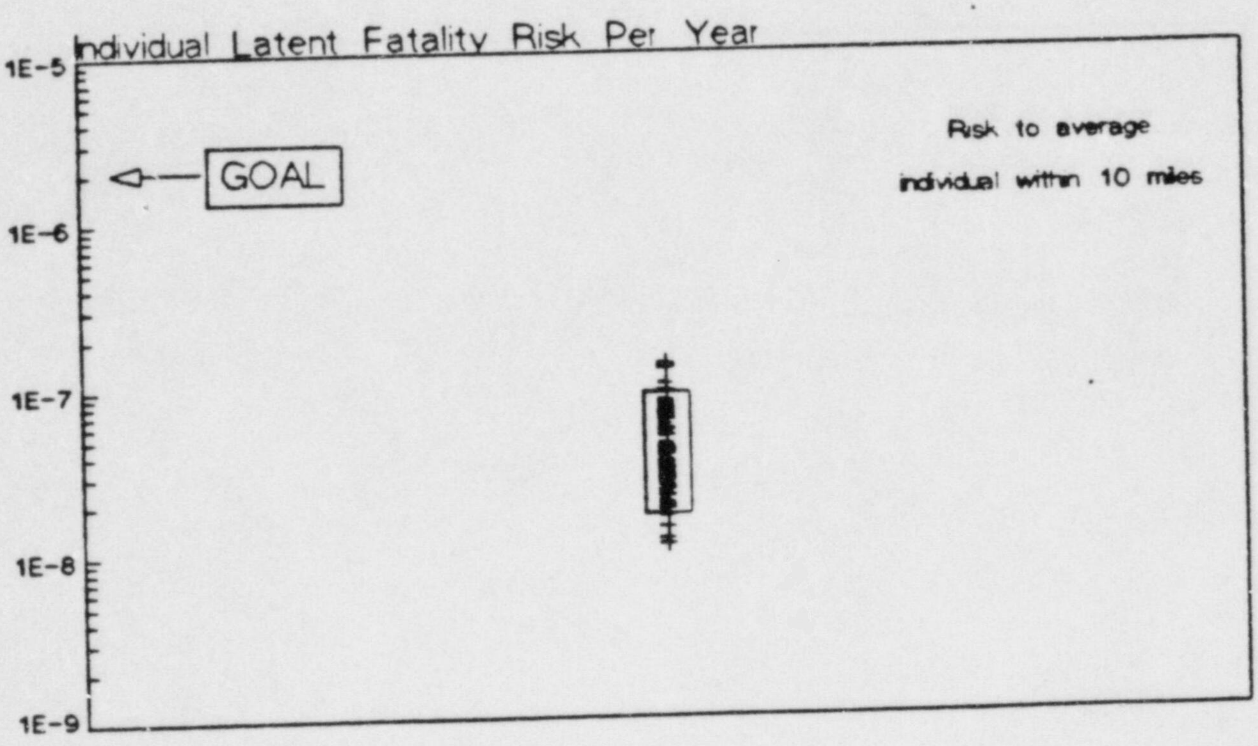


Figure 7.14 Comparison of Sequoyah internal event risk with safety goals

Sequoyah Probability of Large Release

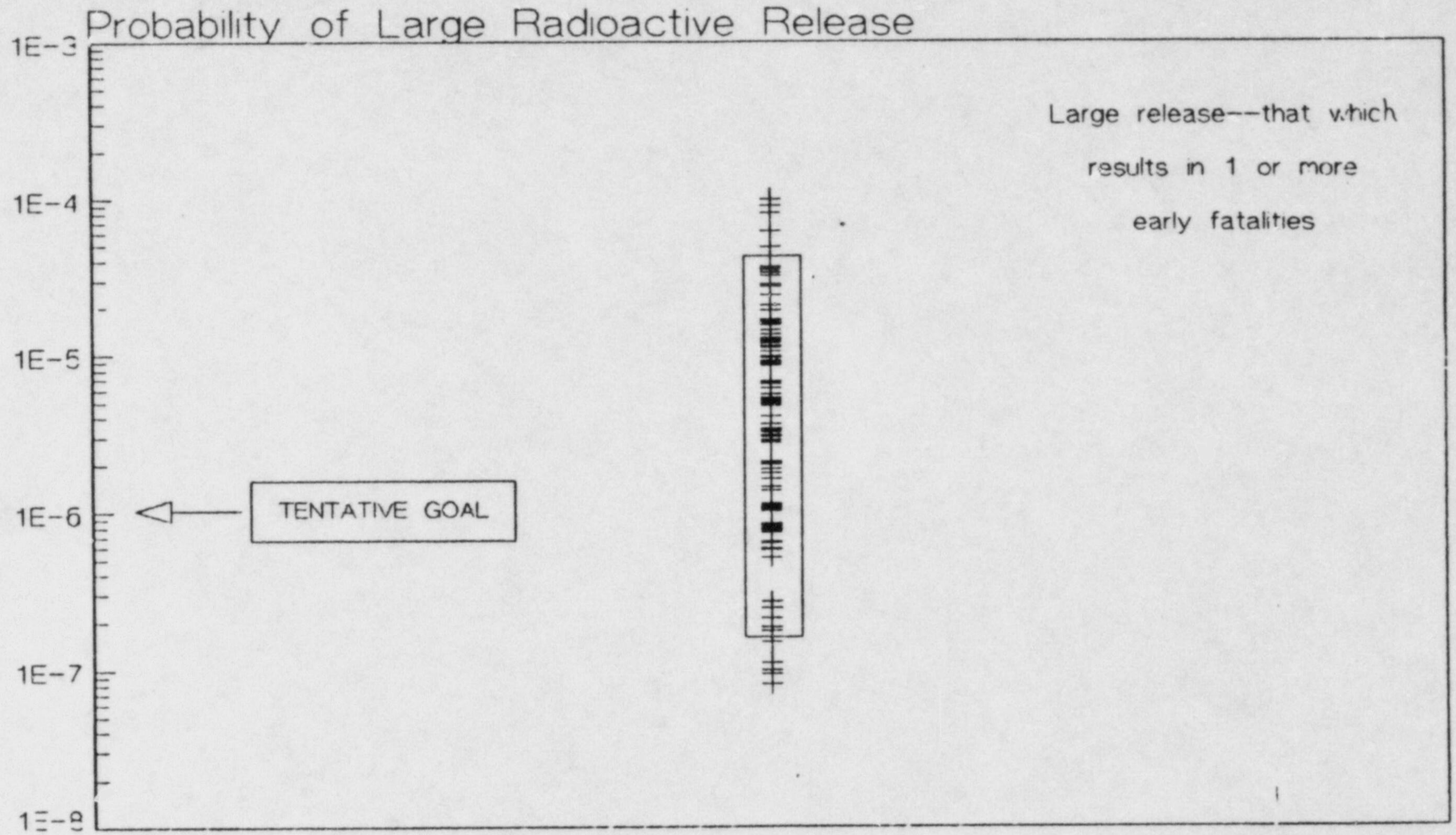


Table D.3 Sensitivity Studies for Core Damage Frequency Analysis

Issue	Description of Sensitivity Study	Total Core Damage Frequency
	BASE CASE	1.0×10^{-4}
1. RCP Seal LOCA	Decrease the probability of an RCP seal LOCA following station blackout sequences to 0.05. Also, increase the probability of recovery from a RCP seal LOCA during T_{CCW} to 0.10.	7.5×10^{-5}
2. RCP Seal LOCA	Assume a seal LOCA occurs 1/2 hr after loss of all seal cooling and that recovery of AC power within 1 hr of the seal LOCA prevents core damage.	1.1×10^{-4}
3. RCP Seal LOCA	Assume the size of a RCP seal LOCA is 1/4 the size assumed in the base case. The time for recovery of HPI flow was increased from 1 hr to 2 hr.	1.0×10^{-4}
4. Recovery of AC Power	Use alternative (more optimistic) set of data for experience in recovery of offsite electric power.	1.0×10^{-4}
5. Steam Binding of Aux. Feedwater Pumps (AFW)	Assume an increased probability of common mode failure of all AFW pumps due to steam binding.	1.5×10^{-4}(a)
6. Loss of Component Cooling Water (CCW)	Assume only one CCW pump is sufficient to service the cooling loads for both Sequoyah units, rather than one for each unit as in the base case.	5.8×10^{-5}
7. Recovery from Loss of CCW	Assume 70-95 percent of loss of CCW sequences are recoverable.	6.3×10^{-5}
8. Feed and Bleed Success Criteria	Assume one PORV is sufficient to provide cooling under loss of feedwater conditions.	1.0×10^{-4}
9. Requirement for Recirculation Cooling After an S_2 Initiator	Assume that for 75 percent of S_2 LOCAs, containment sprays are not actuated.	8.5×10^{-5}

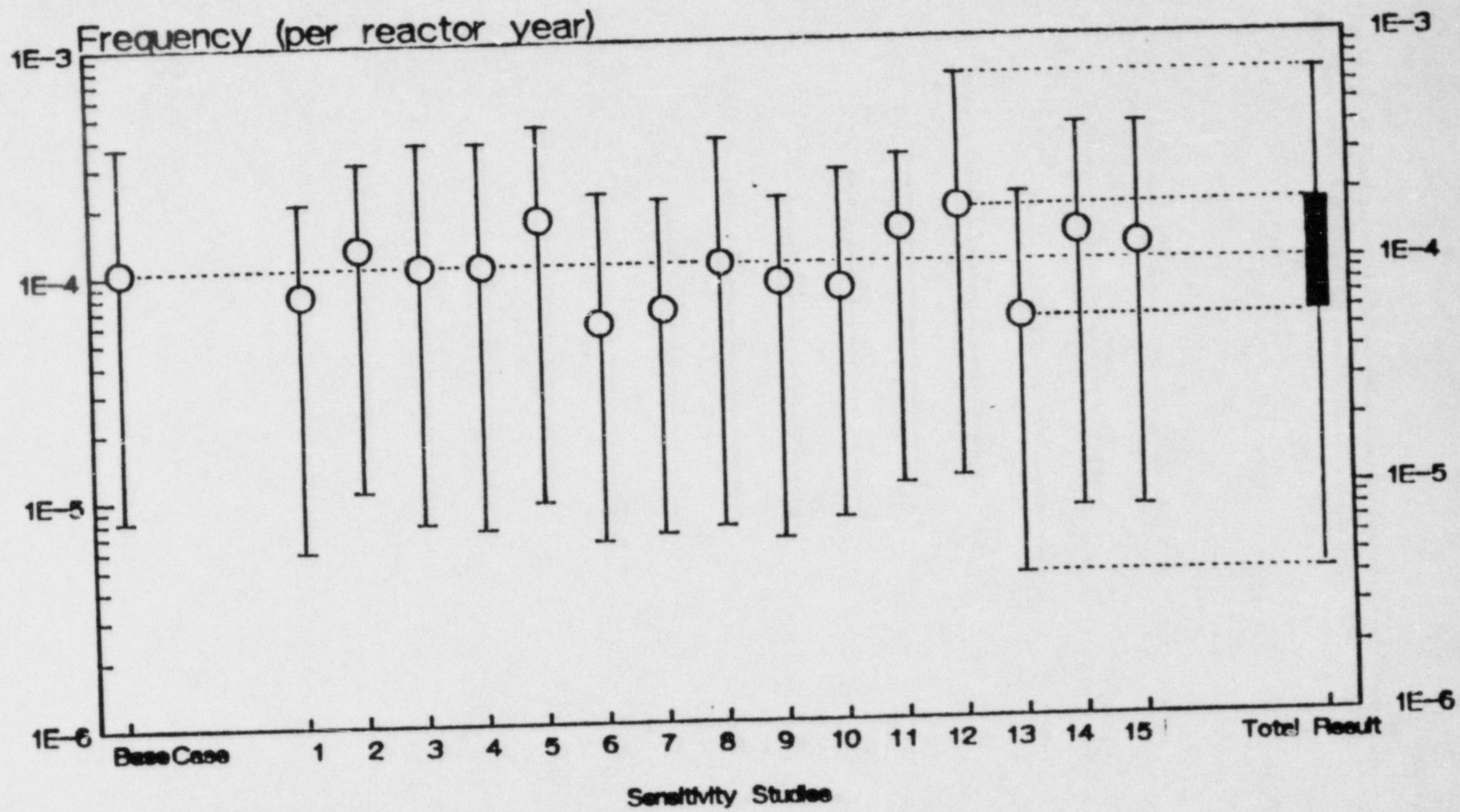


Figure D.1 Ranges of Core Damage Frequency - Sequoyah

TOTAL CORE DAMAGE FREQUENCY

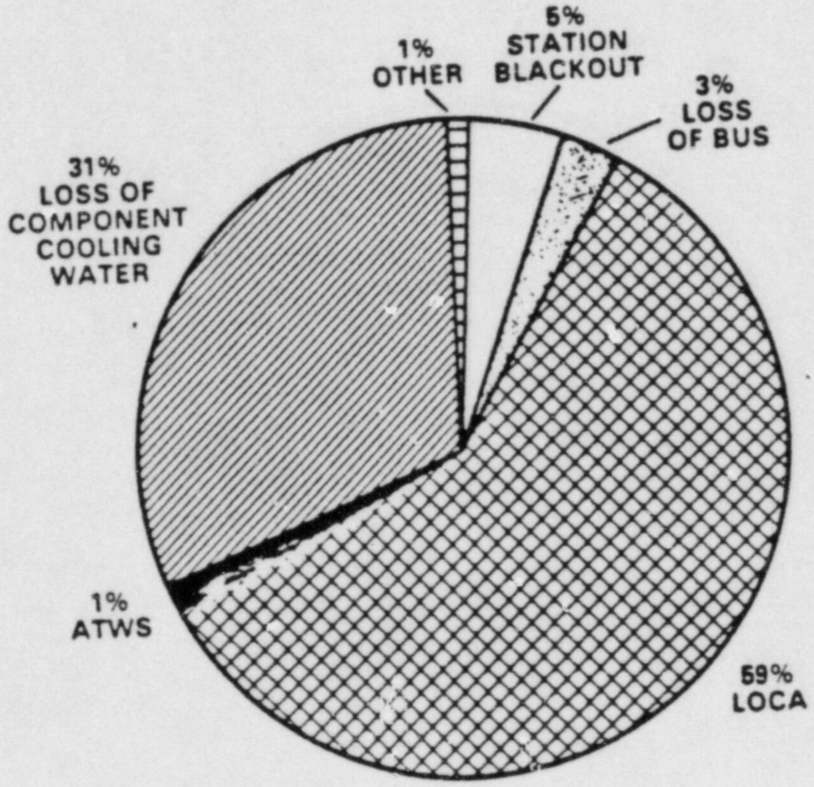


Figure 3.5 Principal contributors to core damage frequency at Sequoyah