ACRS-2250

PDR 062185

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10/26/84

MEETING SUMMARY FOR THE COMBINED SUBCOMMITTEES

ON LIMERICK UNITS 1 AND 2 AND

RELIABILITY AND PROBABILISTIC ASSESSMENT

OCTOBER 20, 1984

The combined ACRS Subcommittee on Limerick Units 1 and 2 and Reliability and Probabilistic Assessment held in Los Angeles, Ca. on October 20, 1984 to continue the Subcommittee discussions on the Limerick probabilistic risk assessment and the Philadephia Electric Company application for a license to operate the Limerick Station. The meeting was held in open session, begin at 8:30 a.m. and was adjourned at 6:00 p.m. The principle attendees were:

D. Okrent, Chairman	J. Ebersole, Member
R. Savio, ACRS Staff	S. Seth, Staff Fellow
P. Davis, Consultant	A. Garcia, Consultant
P. Pomeroy, Consultant	D. Powers, Consultant
M. Trifunac, Consultant	

S. Acharya, NRC	F. Coffman, NRC
J. Rosenthal, NRC	R. Martin, NRC
W. Kastenberg, RDA	W. Pratt, BNL
H. Ludewig, BNL	K. Shiu, BNL

V. Boyer, PECO G. Daebler, PECO

R. Henry, FAI G. Gabor, FAI

The highlights from this meeting were as follows:

HIGHLIGHTS

1. The PECO representatives summarized the results of the Limerick PRA/SARA evaluation. The frequency of core damage (points estimates) was estimated a 2.4×10^{-5}/reactor year with the frequency from interval events being estimated at 1.5×10^{-5}/reactor year, the frequency from earthquakes estimated at 5.7×10^{-6}. These estimates are displayed with their uncertainties on page 1 of Attachment A and the dominant sequences are listed on page 2. A comparison with other plants for which a full scope PRA has been done is given on page 3 of Attachment A. A estimated acute fatalties comparison with the Reactor Safety Study is given on page 4 of Attachment A. Comparisons with the proposal Commission SAfety Goal is given on page 5. PECO has concluded from the Limerick PRA/SARA analysis that core damage frequency is dominanted by interval events, and that no single sequence or plant system dominantes the frequency of core damage.

2. Accidents caused by a seismic event are a major contribution to the risk of early fatalities. The estimates of seismic risk have large recognized uncertainties, which appear to be controlled by the estimates of seismic hazard. There were additional questions as to

the adequacy of fragility analysis, and the effect of failures in systems which were not designed to withstand failure may effect safety equipment. BNL has raised the number of questions as to the methods which were used in the PRA/SARA.

3. THE PECO representatives discussed their experience with the use of the PRA/SARA analysis. The stated that the process involved will come increased from understaning in the performance of this evaluation increased their understanding of plant design and operation and lead to useful improvements in this area. They stated that proposed plant changes need to be evaluated in light of uncertainties and that they believed the results were best used to compare alternatives. Tney also stated that it was PECO intention to maintain the capability for updating the PRA/SARA, to update the PRA/SARA as required, and to train PECO employees in the use of PRA/SARA insights. Specifically, PRA/SARA results will be used in the evaluation of modifications and in procedure development, maintenance, and training activities. PRA/SARA results have already been used to evaluate plant modifications and to develop emergency procedures and a capability for updating the PRA/SARA is in place. The others uses described are planned for the future.

4. These was considerable discussion as to the comparison of the core/melt containment performance model used in the PRA/SARA and their comparison with the current state of the art and with WASH-1400. The PRA/SARA uses more advanced thermal-hydraulics and containment response models. WASH-1400 used significantly smaller

suppression pool decontamination factors. The CORRAL code was used to predict fission product transport in containment in WASH-1400 where as land calculations were used in the PRA/SARA. The source term calculations used in the PRA/SARA were based on methods different from what was used in WASH-1400. The PRA/SARA results were however, modified, to account for this (i.e., baselined to WASH1400 source terms) for use in the Environmental Statement.

(1)

ANNUAL CORE DAMAGE FREQUENCY

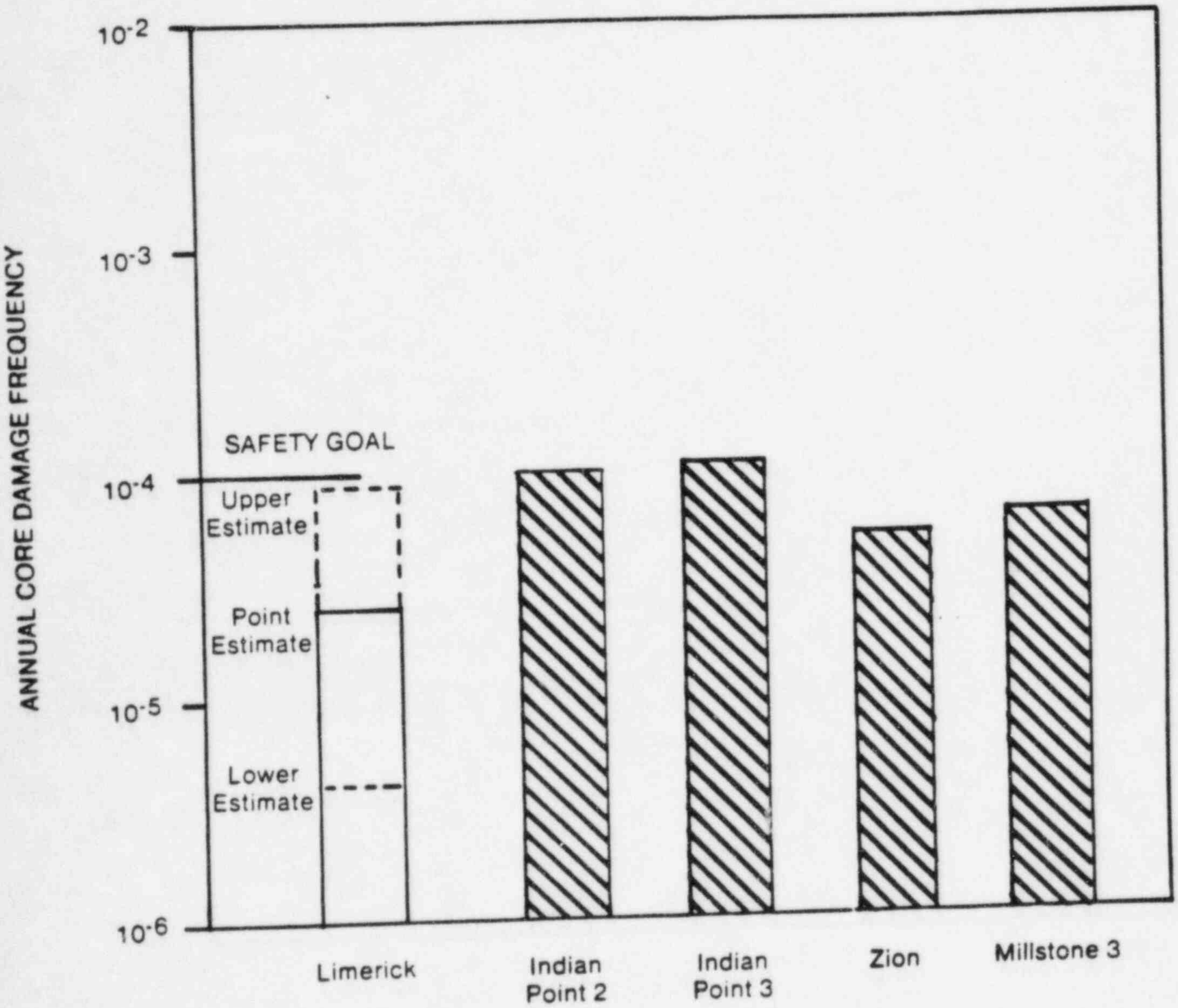
	LOWER ESTIMATE	MEDIAN	UPPER ESTIMATE	POINT ESTIMATE
INTERNAL	2.4×10^{-6}	9.2×10^{-6}	6.0×10^{-5}	1.5×10^{-5}
EXTERNAL				
SEISMIC	2.2×10^{-9}	3.3×10^{-7}	2.7×10^{-5}	5.7×10^{-6}
FIRES	1.7×10^{-7}	1.4×10^{-6}	1.2×10^{-5}	3.4×10^{-6}
OTHER		— NEGLIGIBLE —		
TOTAL	4.0×10^{-6}	1.8×10^{-5}	7.8×10^{-5}	2.4×10^{-5}

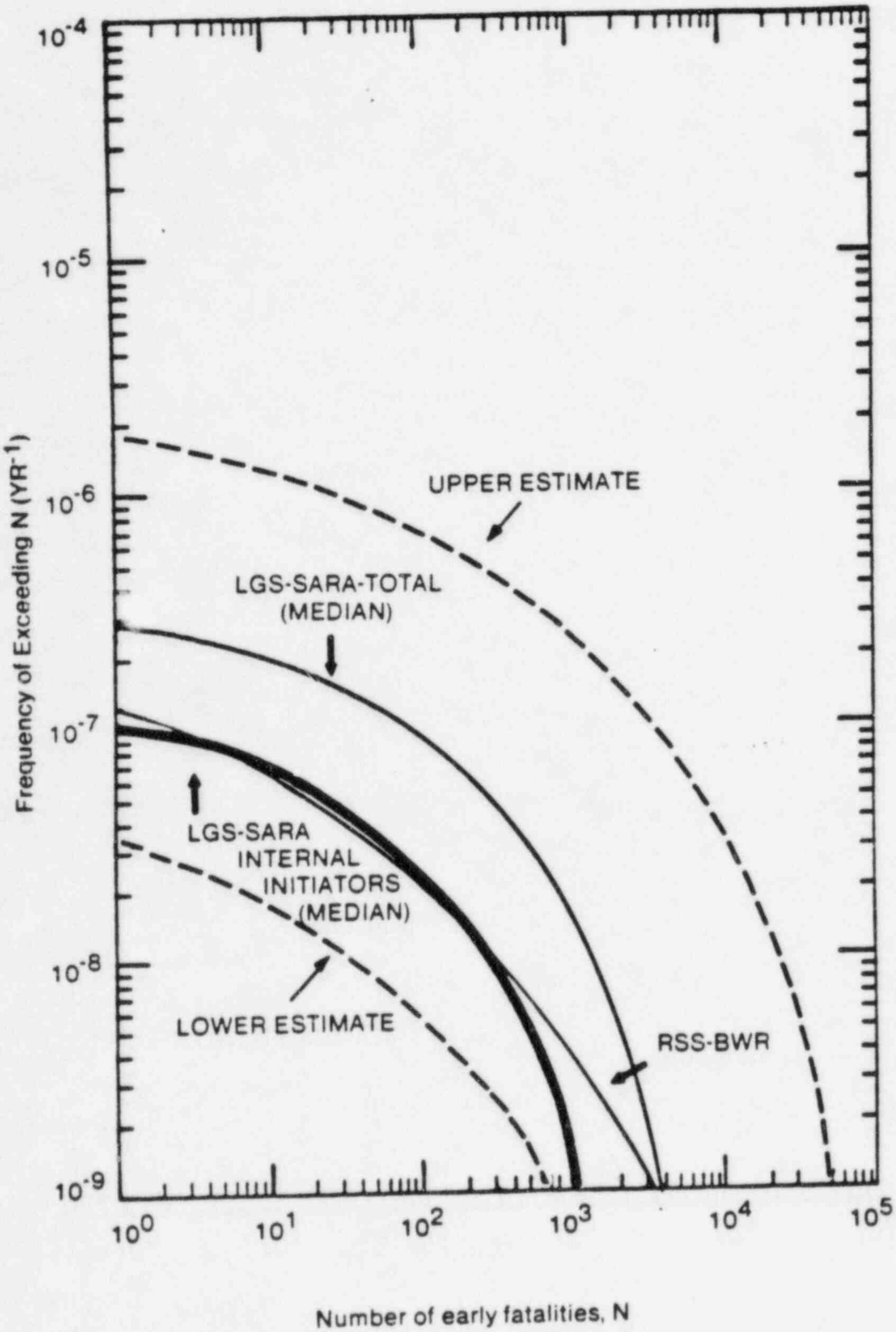
2

DOMINANT CORE DAMAGE SEQUENCES

DESCRIPTION	DESIGNATION	POINT ESTIMATE	PERCENT OF TOTAL
LOSS OF OFFSITE POWER COMMON CAUSE FAILURE OF ALL DIESELS FAILURE OF HPCI AND RCIC	T_EUV	5.9×10^{-6}	25
LOSS OF FEEDWATER FAILURE TO RESTORE FEEDWATER FAILURE OF HPCI AND RCIC FAILURE OF TIMELY DEPRESSURIZATION	T_FQUX	3.6×10^{-6}	15
SEISMIC LOSS OF OFFSITE POWER SEISMIC FAILURE OF AC/DC BUSES AND SWITCHGEAR	T_SE_SUX	3.2×10^{-6}	13

COMPARISON OF LIMERICK CDF WITH POINT ESTIMATES OF OTHER PLANTS





CCDFs of acute fatalities-comparison with the Reactor Safety Study.

ANNUAL INDIVIDUAL RISK

	EARLY FATALITY	LATENT CANCER FATALITY
U.S. Avg.	5×10^{-4} (1)	2×10^{-3}
Safety Goal	5×10^{-7} (2)	2×10^{-6} (3)
LGS Upper	7×10^{-8} (2)	1×10^{-8} (3)
LGS Lower	1×10^{-10} (2)	2×10^{-10} (3)

(1) Accidental Causes

(2) Avg. Within 1 Mile

(3) Avg. Within 50 Miles