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

August 16, 1989

Docket Nos.: 50-352/353

LICENSEE: Philadelphia Electric Company

FACILITY: Limerick Generating Station, Unit 1 and Unit 2

SUBJECT: SUMMARY OF JULY 27, 1989 MEETING ON SEVERE ACCIDENT ISSUES

References:
(1) Letter from G. Hunger, Philadelphia Electric Company, to NRC Document Control Desk, dated June 23, 1989.
(2) Letter from R. Clark, NRC, to G. Hunger, Philadelphia Electric Company, dated May 23, 1989.
(3) Letter from G. Suh, NRC, to G. Hunger, Philadelphia Electric Company, dated July 31, 1989.

On July 27, 1989 at the NRC offices in Rockville, Maryland, a meeting was held between the NRC staff and representatives of the Philadelphia Electric Company and its consultants to discuss severe accident issues related to the Limerick Generating Station. The items discussed at the meeting were identified by the staff in its review of the licensee's response (Reference 1) to the staff's request for additional information (Reference 2). Enclosure 1 provides a list of the meeting participants. For the purpose of keeping a record of the meeting, the meeting was recorded. Copies of the meeting transcript have been placed in the public document room. A summary of some of the meeting highlights are described in the following.

The licensee presented information related to a comparison of the core damage frequencies for each of the accident classes in the original and updated probabilistic risk assessments. The original PRA was completed in 1982. Since then, the PRA has been updated in 1986, 1988, and most recently in June 1989 as presented in Reference 1. The reasons for changes in core damage frequencies were discussed which included changes in the PRA model, hardware improvements in the plant, and increased credit for certain operator actions. The licensee confirmed that the improvements presented in Table 8 of NUREG-1068, titled "Review Insights on the Probabilistic Risk Assessment for the Limerick Generating Station," dated August 1984, had been implemented at the plant. These improvements consisted of improved automatic depressurization system initiation logic following the potential loss of high pressure coolant sources and improved design to achieve alternate methods of HPCI/RCIC room cooling during loss of offsite power events, and were stated in NUREG-1068 to result in a reduction of the core damage frequency estimate by a factor of 2.5. For other hardware and procedural improvements made to the plant, the change to the estimated values of plant risk were not available for any specific improvement made to the plant. Licensee representatives stated that the plant PRA model had gone through a number of revisions which made difficult the assignment of an effect from any single change, given the interactive effect between the various changes that had been made.

Models used to develop the release fractions for each of the accident classes and estimates of the offsite consequences were discussed. These models were essentially those used in the development of the 1982 PRA. The methodology

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used in assessing the risk reduction benefits of various severe accident mitigation design alternatives was summarized. Other topics of discussion included the containment event tree, conditional early fatality estimates of the dominant accident sequences, the modeling of the dominant fire sequences, and transient initiator frequencies.

The licensee discussed the process used for the derivation of cost estimates for severe accident mitigation design alternatives. The process included the development of the basic design bases for each alternative; formulation of conceptual designs by engineering teams; review of the design concepts for consistency with the risk reduction analyses effort; and estimation of material quantities, man-hour estimates, and schedules.

A discussion was also held of potential mitigation alternatives which have been considered in the Containment Performance Improvement Program being conducted by the Office of Nuclear Regulatory Research within the NRC. These alternatives are generic items that are being considered for Mark II containment plants, and were not alternatives which had been previously analyzed by the licensee in Reference 1. The specific alternatives which were discussed included provisions for an additional standby diesel generator, low pressure backup water supply system, methods for removing drywell spray water from the suppression pool to prevent containment failure, and actions to flood the outer surface of the drywell head. The licensee provided general comments as to the feasibility of these generic items as they related to the Limerick station, but had not performed specific analyses of the benefits and costs associated with these potential improvements.

During the meeting, several follow-up questions (see Enclosure 2) were identified. These were transmitted to the licensee in Reference 3. Enclosure 3 provides a copy of the licensee handouts which were distributed at the meeting.

Gene Y. Suh

Gene Y. Suh, Project Manager
Project Directorate I-2
Division of Reactor Projects I/II

Enclosures:
As stated

cc w/encls. 1 & 2:
Licensee/Applicant &
Service List

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/S/

Gene Y. Suh, Project Manager
Project Directorate I-2
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As stated

cc w/encls. 1 & 2:
Licensee/Applicant &
Service List

DISTRIBUTION: w/encls. 1 & 2

Docket File*	PDI-2 Reading*	GSuh/RClark	WButler	MO'Brien
Meeting participants		SVarga/BBoger	TSpeis	NRC PDR/LPDR*
FCongel	AThadani	LChandler	OGC	EJordan
BGrimes	ACRS (10)			

*w/encls. 1, 2 & 3

[GYSA]

PDI-2/PM	PDI-2/D
GSuh:tr	WButler
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ENCLOSURE 1

NRC/PECo MEETING ON
SEVERE ACCIDENT MITIGATION ISSUES

NAME	ORGANIZATION
R. M. Krich	PECo/Licensing
E. Robert Schmidt	NUS Corp.
Robert M. Rader	Conner and Wetterhahn
Dennis A. Klein	Bechtel
James T. Hearn	Bechtel
J. L. Phillabaum	PECo
A. J. Marie	PECo
P. J. Fulford	NUS Corp.
R. J. Barrett	NRC/NRR
R. L. Palla	NRC/NRR
E. S. Chelliah	NRC/RES/DSR
Trevor Pratt	BNL
Charles Hinson	NRC/NRR
John N. Ridgely	NRC/RES
Charles Ferrell	NRC/RES/SAIB
Brad Hardin	NRC/RES
Tim Margulies	NRC/NRR
Ann Hodgdon	NRC/OGC
Ajit K. Bhattacharyya	Comm. of PA/DER/BRP
Sidney Feld	NRC/RES
Gene Y. Suh	NRC/NRR/DRP

ENCLOSURE 2

QUESTIONS FROM JULY 27, 1989 NRC MEETING

WITH PHILADELPHIA ELECTRIC COMPANY

1. With regard to the contribution of fire initiated sequences to core damage frequency estimates, do the cables for both trains of safety-related systems have three hour fire ratings?

2. For the severe accident mitigation design alternatives considered in the June 23, 1989 submittal, what occupational exposure would be incurred in installation and in recurring operation and maintenance? Please provide the bases for the estimates, including work hour estimates, location of the work performed, and applicable radiation dose rates.

3. Please provide a reference to drawings for the drywell head assembly and the associated HVAC systems in the immediate area.

4. In the consideration of mitigation alternatives, one potential alternative would be leaving Unit 2 idle and providing replacement power from other sources. If Unit 2 was idle, what would be the source(s) of the replacement power, including a description of the type of power source and whether the source would be from inside or outside the utility's system? What would be the environmental impacts which would result from the use of these other sources?

5. What capacity factor was assumed for Unit 2 in terms of replacement energy costs and what variability was assumed in that factor as a function of plant life?