ACCELERATED DISTRIBUTION DEMONSTRATION SYSTEM

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9403310200 DOC.DATE: 94/03/21 NOTARIZED: NO DOCKET #
FACIL:50-387 Susquehanna Steam Electric Station, Unit 1, Pennsylva 05000387
50-388 Susquehanna Steam Electric Station, Unit 2, Pennsylva 05000388
AUTH.NAME AUTHOR AFFILIATION
BYRAM,R.G. Pennsylvania Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION
MILLER,C.L. Project Directorate I-2

SUBJECT: Forwards Rev 0 to EPIP EP-IP-055, "Post-Accident Reactor Bldg HVAC (Reactor Bldg Non-1E Electrical Load Shed)," per NRC 940317 request.

DISTRIBUTION CODE: A045D COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 4+23
TITLE: OR Submittal: Emergency Preparedness Plans, Implement'g Procedures, C

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD1-2 PD	1	1	CLARK,R	1	1
INTERNAL:	AEOD/DOA/IRB	1	1	NRR/DRSS/PEPB	1	1
	NUDOCS-ABSTRACT	1	1	REG FILE 01	1	1
EXTERNAL:	NRC PDR	1	1	NSIC	1	1

NOTE TO ALL "RIDS" RECIPIENTS:

PLEASE HELP US TO REDUCE WASTE! CONTACT THE DOCUMENT CONTROL DESK, ROOM P1-37 (EXT. 20079) TO ELIMINATE YOUR NAME FROM DISTRIBUTION LISTS FOR DOCUMENTS YOU DON'T NEED!

TOTAL NUMBER OF COPIES REQUIRED: LTTR 8 ENCL 8

MAY

RIDS/ADDS

PP&L Pennsylvania Power & Light Company
Two North Ninth Street • Allentown, PA 18101-1179 • 610/774-5151

Robert G. Byram
Senior Vice President—Nuclear
610/774-7502
Fax: 610/774-5019

MAR 21 1994

Director of Nuclear Reactor Regulation
Attention: Mr. C. L. Miller, Project Director
Project Directorate I-2
Division of Reactor Projects
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUSQUEHANNA STEAM ELECTRIC STATION
INFORMATION ON FUEL POOL COOLING -
ELECTRICAL LOAD SHED

Docket Nos. 50-387
& 50-388

PLA-4109 **FILE R41-2**

Dear Mr. Miller:

Attached are the documents that were requested by your staff on March 17, 1994. These documents include the approved procedure EP-IP-055 "Post Accident Reactor Building HVAC (Reactor Building Non-1E Electrical Load Shed)" and its associated safety evaluation. In addition, we have provided a brief explanation of how this procedure would be used in our Emergency Response Organization and on our calculations performed to define the Post Accident temperatures in our reactor building.

If you have any questions, please contact Mr. J.M. Kenny at (610) 774-7904.

Very truly yours,

R. G. Byram

Attachment

cc: NRC Document Control Desk (original)
NRC Region I
Mr. G. S. Barber, NRC Sr. Resident Inspector - SSES
Mr. R. J. Clark, NRC Sr. Project Manager - Rockville
Mr. J. W. Shea, NRC Project Manager - Rockville

A045 1/1

The attached documents are provided for your use. These documents include the originally approved procedure EP-IP-055 "Post Accident Reactor Building HVAC (Reactor Building Non-1E Electrical Load Shed)" and its associated 50.59 evaluation. In addition, we have provided a document which discussed the design features of the Auxiliary Load Shed which occurs at the initiation of the LOCA.

The EP-IP procedures (or EP-PS, as they are currently designate) are considered to be guidance documents used by our Emergency Response Organization. They are defined as follows:

> "A document describing how to perform the tasks assigned to an emergency position. Each document includes an overview of a position's tasks, detailed instructions, and relevant reference material. Used together, these documents are designed to implement the Emergency Plan during a declared emergency."

During a declared emergency, the engineering organization would be involved in the decision making process for recovery of the plant. Typically, the EP-IP procedures include guidance on response to engineering issues which involve long-term recovery actions. These actions do not include first priority vital actions, and generally fall outside of the first 8 hours of the emergency response. With respect to the attached subject matter, the existing emergency response procedure (EP-PS-102) provides specific guidance on both the loss of Fuel Pool Cooling Issue and the Reactor Building Non-1E Electrical Load Shed. These topics are discussed within separate tabs in the procedure, thus assuring the Technical Support Center (TSC) Coordinator would be cognizant of both issues.

The electrical load shed discussed in EP-IP-055 was intended to provide specific guidance to the Emergency Director (TSC Coordinator) to maintain equipment/component temperatures for the accident unit within EQ limits to assure long-term functionality of this equipment. The procedure directs the operators to restore the normal HVAC systems first, if possible. If this cannot be accomplished, then shedding non-1E electrical loads in the accident unit reactor building <u>only</u> will be necessary to assure temperatures remain within the EQ limits. During the response to such an emergency, the Engineering organization would be involved in evaluation of the event, monitoring the building temperatures and providing guidance to restoration of the HVAC systems. The operators would utilize existing procedures for restoration of the normal HVAC systems. Once the normal HVAC systems are restored, the EP-IP-055 or (EP-PS-102) procedure would be exited.

Upon review of Safety Evaluation #NL 88-026 Section II, the consequences of performing the electrical load shed are clearly discussed. The document identifies the major loads which are lost and notes that the consequences of their loss would not impact the ability to safety shutdown the plant during an accident condition. In addition, it is important to note that the original plant design included an auxiliary load shed of the 13.8kV loads from the auxiliary busses to assure adequate voltage is maintained during startup of the major safety related equipment loads, such as ECCS pumps. As a result of this load shed, the

Service Water pumps are lost and the cooling function of the Fuel Pool Cooling system will no longer be provided for the accident unit only. These loads can be restored ten minutes after initiation of the event. Complete restoration of the Service Water system would take approximately one 12 hour shift. The original design basis assumed the fuel pool cooling function would be restored, thus preventing the fuel pool from reaching a boiling condition. Restoration of the fuel pool cooling function is consistent with the SSES licensing basis, which does not consider a boiling fuel pool concurrent with a DBA LOCA/LOOP event. Operators would utilize existing Off-Normal procedures to restore the normal Fuel Pool Cooling or RHR Fuel Pool Cooling systems.

Calculation M-RAF-024 was developed to establish transient temperature responses for the reactor building for use in evaluating equipment qualification. The calculation was initially completed in October 1988. All previous analyses were either performed as steady state or simplified single node transient models of the reactor building. The Compartment Transient Temperature Program (COTTAP) computer code was developed for this specific application.

The initial calculation (M-RAF-024, Rev. 0) utilized the assumption that the fuel pool cooling function would be restored, which was consistent with the licensing basis The calculation notes that RHR Fuel Pool Cooling mode may be utilized to provide cooling to the fuel pool, if the normal Fuel Pool Cooling system cannot be restored. This prevented the need for consideration of a boiling pool environment. The fuel pool temperature was maintained at its design maximum value of 125°F for the 100 day duration of the analysis. The calculation includes heat loads from the normal Fuel Pool Cooling system equipment. To increase our design margins, subsequent revisions to the calculation incorporated sensible heatup of the fuel pool to 212°F in 24 hours and maintained these conditions for the duration of the event. This temperature response conservatively bounds any fuel pool heatup prior to restoration of cooling to the Fuel Pool.