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- TAbernathy
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Docket No. 50-289

MAR. 29 1976

Metropolitan Edison Company  
 ATTN: Mr. R. C. Arnold  
 Vice President - Generation  
 P. O. Box 542  
 Reading, Pennsylvania 19603

Gentlemen:

We are currently reviewing your reload application of February 11, 1976 for the Three Mile Island Nuclear Station Unit 1 Cycle 2 core. In order for us to complete our review it will be necessary for you to provide additional information in support of this application.

Specific questions and areas needing further attention are itemized in the enclosure to this letter.

In order to keep the reload application review timely, you are requested to provide this information within 10 days of the receipt of this letter. Your reply should include three (3) signed originals and 37 copies of the requested information.

Sincerely,

Original Signed by

Robert W. Reid, Chief  
 Operating Reactors Branch No. 4  
 Division of Operating Reactors

Enclosure:  
 Request for Additional  
 Information

cc:  
 See next page

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OFFICE	ORB4 <i>Car</i>	ORB4	<del>ORB4</del> C:ORB4		
SURNAME	DBridges:mmt	RIngram	RWReid		
DATE	3/29/76	3/29/76	3/29/76		



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

March 29, 1976

Docket No. 50-289

Metropolitan Edison Company  
ATTN: Mr. R. C. Arnold  
Vice President - Generation  
P. O. Box 542  
Reading, Pennsylvania 19603

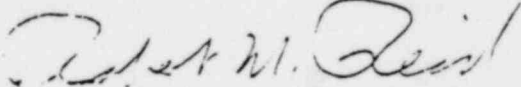
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Sincerely,

  
Robert W. Reid, Chief  
Operating Reactors Branch No. 4  
Division of Operating Reactors

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Request for Additional  
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See next page

1490 316

cc: w/enclosure

G. F. Trowbridge, Esquire  
Shaw, Pittman, Potts, Trowbridge & Madden  
Barr Building  
910 17th Street, N. W.  
Washington, D. C. 20006

GPU Service Corporation  
Richard W. Heward, Project Manager  
Thomas M. Crimmins, Jr., Safety  
and Licensing Manager  
260 Cherry Hill Road  
Parsippany, New Jersey 07054

Pennsylvania Electric Company  
Mr. R. W. Conrad  
Vice President, Generation  
1001 Broad Street  
Johnstown, Pennsylvania 15907

Mr. Weldon B. Arehart, Chairman  
Board of Supervisors of Londonberry  
Township  
2148 Foxiana Road  
Middletown, Pennsylvania 17057

Miss Mary V. Southard, Chairman  
Citizens for a Safe Environment  
P. O. Box 405  
Harrisburg, Pennsylvania 17108

Government Publications Section  
State Library of Pennsylvania  
Box 1601 (Education Building)  
Harrisburg, Pennsylvania 17126

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REQUEST FOR ADDITIONAL INFORMATION

CYCLE 2 RELOAD

THREE MILE ISLAND UNIT 1

1. Three Mile Island, Unit 1, submittal of February 11, 1976, is based on measured core flow rate (with 1.5% margin) rather than the design flow rate. The use of measured rather than design flow rate is acceptable provided:
  - (1) The flow measurement technique is acceptable.
  - (2) An error analysis is provided for the data measured.
  - (3) There is adequate conservatism between the flow rate actually measured and the flow which was used to develop plant technical specifications.
  - (4) There is periodic surveillance and/or additional testing to assure that the core flow rate does not decrease as a result of crud buildup, steam generator tube plugging, or other causes.

Based on these considerations, provide the following:

- (1) A description of the flow measurement technique used along with the data measured and an error analysis for the measurements.
- (2) A discussion of the bases for the power to flow into reactor trip setting and the overpower trip setting (if it is based on measured flow).
- (3) A proposed surveillance and/or test program to confirm that the value of the core flow rate has not decreased below the value used as the basis for reactor power/flow trip (and/or the overpower trip), including appropriate uncertainties.

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2. Rod bowing is not considered in the report. A discussion should be included on the effect of rod bowing and either justification given for not including the effects or revised technical specifications to account for rod bowing should be proposed.
3. Provide justification for the moderator and doppler coefficients used in the accident and transient analysis (Table 7.1-1 of Reload Report) and explain how these values relate to those listed in Table 5.1-1.
4. Indicate which of the values listed in Table 5.1-1 were observed in actual Cycle 1 operation and compare these observed values with calculations. Parameters of specific interest include critical boron concentrations, control rod worths, and core temperature coefficients.
5. It is our understanding that the check value proposed in your correspondence of April 19, 1975 to address the problem of long term cooling capability following a loss-of-coolant accident (LOCA) will not be available for installation prior to startup of Cycle 2. If this is the case it will be necessary for you to provide specific information on how power would be restored to the necessary values following a failure of the LC Engineered Safeguard Valve 480V control center prior to utilization of that aspect of the cooling system.

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