Distribution
Docket
NRC PDR
Local PDR
ORB4 Rdg
KRGoller
TJCarter
OELD
OI&E (3)
RWReid
DBridges
RIngram
DEisenhut
TAbernathy
JBuchanan
ACRS (16)
VStello
file

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	ORB4	C:ORB4		
SURNAME	DBridges:mmt	RIngram	RWReid		
DATE	3/29/76	3/ /76	3/29/76		

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

March 29, 1976

Docket No. 50-289

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Robert W. Reid

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

Enclosure:
Request for Additional
Information

cc:
See next page

1490 316

cc: w/enclosure

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

1490 317

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.

1490 318

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 1C Engineered Safeguard Valve 480V control center prior to utilitization of that aspect of the cooling system.

1490 319