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U.S. NUCLEAR REGULATORY COMMISSION

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DOCKET NUMBER: 50-289

FILE NUMBER:

TO: Mr. Robert W. Reid

FROM: Metropolitan Edison Company
Reading, Pa.
R. C. Arnold

DATE OF DOCUMENT: 8/20/76

DATE RECEIVED: 8/23/76

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DESCRIPTION

Ltr. re our 6/9/76 ltr. and their 7/9/76 ltr.....concerning Reactor Vessel Supports Analysis.

(2-P)

ENCLOSURE

PLANT NAME:

Three Mile Island #1

DISTRIBUTION FOR REACTOR VESSEL SUPPORT INFO FOR OPERATING REACTORS PER MR. TRAMMELL 7-12-76

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August 20 , 1976
GQL 1189

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AUG 23 1976
U.S. NUCLEAR REGULATORY COMMISSION

Director of Nuclear Reactor Regulation
Attn: Mr. Robert W. Reid, Chief
Operating Reactors Branch #4
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Sir:

Three Mile Island Nuclear Station Unit 1 (TMI-1)
Docket No. 50-289
Operating License No. DPR-50
Reactor Vessel Supports Analysis

Your letter of June 9, 1976 requested additional information evaluating the adequacy of the reactor pressure vessel supports under loads not previously considered. As indicated in our July 9th response, we have met with other Babcock & Wilcox NSSS users in an effort to evaluate the concern and discuss possible resolutions.

Two paths are being pursued in parallel to determine our required action. The first approach is to undertake a review of work sponsored by a group of utilities evaluating the absolute and relative probability of a postulated pipe break between the reactor vessel nozzle and the reactor cavity wall. Science Applications, Inc. (SAI) is submitting their results in the form of a topical report in late August, 1976. The NRC staff was introduced to the study in a presentation involving SAI on July 13, 1976.

The second approach is a review of the additional information requested in your June 9th letter which requires a costly, multi-year analysis for each plant. Discussions with Babcock & Wilcox and other consulting firms, review of ACRS hearings transcripts, and your letter itself indicates uncertainties in the present state of the art analysis to evaluate this particular case.

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As a result of our preliminary review, it is our belief that the SAI study substantiates the extremely low probability of a pipe rupture in the reactor vessel cavity area at TMI-1; therefore, the postulated break represents no significant hazard to the health and safety of the public. In this light, no further analysis is considered required.

Following your review of the SAI study, we would like to meet with you to discuss this issue and address any further concerns you might have.

Sincerely,

RC Arnold

R. C. Arnold
Vice President

RCA:LCL:mft

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