NRC FORM 195 (2-76)

U.S. NUCLEAR REGULATORY COMMISSION

DOCKET NUMBER 50-320

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TO: Mr. Steven A. Varga

FROM: Metropolitan Edison Company
Reading, Pa.
J. G. Herbein

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DESCRIPTION

Consists of info re Reactor Vessel Supports Adequacy.....

ENCLOSURE

PLANT NAME: Three Mile Island Unit No. 2

RJL 10/13/77 (2-P)

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

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METROPOLITAN EDISON COMPANY

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October 6, 1977
GQL 1367

Director of Nuclear Reactor Regulation
Attn: Steven A. Varga, Chief
Light Water Reactors Branch No. 4
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Sir:

Three Mile Island Nuclear Station Unit 2 (TMI-2)
License No. CPPR-66
Docket No. 50-320
Reactor Vessel Supports Adequacy

References:

1. Letter from Karl Kniel to R. C. Arnold, dated December 9, 1975
2. Letter from R. C. Arnold to Karl Kniel, GQL 0017, dated January 9, 1976
3. Letter from D. B. Vassallo to R. C. Arnold, dated July 12, 1976
4. Letter from R. C. Arnold to D. B. Vassallo, GQL 1250, dated August 31, 1976
5. An Analysis of the Probability of Pipe Rupture at Various Locations in the Primary Coolant Loop of a Babcock and Wilcox 177 Fuel Assembly Pressurized Water Reactor - Including the Effects of a Periodic Inspection, SAI-250-77-PA, September 27, 1977
6. An Analysis of the Relative Probability of Pipe Rupture at Various Locations in the Primary Cooling Loop of a Pressurized Water Reactor Including the Effects of a Periodic Inspection, SAI 001-PA, June 1976

References 1 through 5 present the chronology of regulatory requests and response regarding the adequacy of reactor vessel supports assuming an instantaneous double-ended guillotine break inside the reactor cavity. Our last letter to you (Ref. 4) indicated participation of Metropolitan Edison Company in an Owner's Group that is investigating this problem. In August 1976, this Group discussed the analytical options with a broad spectrum of consultants. Our conclusion at that time was, that, given the disparity of opinion between the NRC staff, ACRS, and the analytical consultants regarding the appropriateness of the analytical tools, we would defer further analysis of the event until the appropriateness of the available analytical tools are demonstrated.

Instead, we propose for existing plants, such as TMI-2, to determine if such an event is likely enough to represent a risk to the health and safety of the public. We believe that it is possible to perform a quantitative study to determine if indeed the event was probable enough to require further analysis.

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If the probability of the event is determined to be acceptably low, then further analysis is not warranted. The results of our quantitative analysis have been submitted as a Topical Report by Science Applications, Inc. (Ref. 5).

The report uses similar methodology as a report submitted by the Combustion Engineers User's Group (Ref. 6), but has taken advantage of experience gained in preparation of that report and expands the depth of analysis. The principal features that have been added to the 177 FA Report are:

1. Considering the critical length of through wall defects to vary with the maximum stress level, rather than using a constant conservatively estimated lower bound value.

2. Considering the material properties, stress levels, and parameters in the detection probabilities and initial defect size distribution to be statistically fixed. Monte Carlo calculations of the distribution of the failure rates then allowed the degree of conservatism of the results to be quantitatively estimated.

3. Refinements in the analysis of the effects of in-service inspection on the failure probabilities allowed more accurate treatment of the effects of ISI.

4. Addition of further data to the base of the report.

5. Factoring in of the effect of ultrasonic inspection on rupture probabilities.

By this letter, we are providing our endorsement of the SAI Report and its conclusions. We believe that it fully satisfies your request for information concerning the adequacy of Reactor Vessel Supports.

We believe our design and operation of TMI-2, consistent with the SAI Report, provides sufficient assurance that there is no undue risk to public health and safety and consider this matter to be adequately resolved. We are interested in discussing the salient features of this report with you at your earliest convenience.

Very truly yours,

J. G. Herbein
Vice President

JGH:JRS:tas

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