

May 13, 1988

Docket No. 50-341

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MEMORANDUM FOR: Edward G. Greenman, Director  
Division of Reactor Projects  
Region III

FROM: Dennis M. Crutchfield, Director  
Division of Reactor Projects - III,  
IV, V & Special Projects  
Office of Nuclear Reactor Regulation

SUBJECT: REQUEST FOR ASSISTANCE - QUESTIONS FROM MONROE CITY-COUNTY  
REGARDING FERMI-2 (AITS NO. F0301688) (TAC NO. 67894)

Enclosed are the responses to the five questions you identified in your April 12, 1988 request for NRR assistance. If you have any questions regarding the proposed responses, please contact Ted Quay of my staff (FTS 492-1325).

Original signed by  
Dennis M. Crutchfield

Dennis M. Crutchfield, Director  
Division of Reactor Projects - III,  
IV, V & Special Projects  
Office of Nuclear Reactor Regulation

cc w/enclosure:  
W. G. Rogers, SRI, Fermi  
R. W. Cooper, RIII  
A. Thadani  
F. Gillespie  
M. Caruso

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UNITED STATES  
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D. C. 20555

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ENCLOSURE

Monroe City-County Question:

1. What are the chances of a meltdown at Fermi 2?

Response:

The most recent thorough study of severe accidents has been published in Draft NUREG-1150 (February, 1987) "Reactor Risk Reference Document." This study included analyses of different reactor designs; one of those studied was a BWR Mark I reactor, namely the Peach Bottom nuclear plant. The Peach Bottom plant as a Mark I design is similar to Fermi-2 in reactor and containment design. The results of this study indicated an estimated mean frequency of core damage (i.e., core melt) to be approximately one chance in 100,000 per year of operations. Other studies indicate an estimated core damage likelihood of approximately one in 10,000 per year of reactor operation. These results are consistent with NRC's belief that core melt accidents are very unlikely.

Monroe City-County Question:

2. What are the chances of a severe accident?

Draft NUREG-1150 also investigated the probability of early containment failure following a core melt. It is this issue which has attracted considerable attention to Mark I reactors since the study concluded that there is large uncertainty regarding the probability of early containment failure for these extremely unlikely accidents. As a result of the study documented in Draft NUREG-1150, it was concluded that the containment failure probability for Peach Bottom, a Mark I reactor, could range from 10 to 90 percent, albeit for highly improbable accidents.

Even allowing the large uncertainties which result in a higher upper value for containment failure, the Draft NUREG-1150 study estimated that the probability of a large reactor accident that results in one or more early fatalities ranged from one in one million to one in one billion. Given a severe accident, the probabilities of very high radiation exposure and the distances over which they would occur were also estimated to be reasonably small. The risk levels for Fermi-2 or other Mark I reactors would of course, depend on its actual core melt probability, containment behavior, the local demography, and could vary somewhat from the results presented in Draft NUREG-1150. The results of this and related studies do, however, support the overall conclusion of low severe accident risk of nuclear reactors.

Monroe City-County Question:

3. The NRC recently released NUREG 1150, in which the steel lining [of] Mark I reactors are reported to melt in a few minutes in an accident. Why is Fermi 2 allowed to continue to operate with this new information?

Response:

See the response to question number 2.

Monroe City-County Question:

4. In addition to the melting of the steel lining the NRC has stated Fermi 2's containment has a 90% chance of failure in an accident. Why has this plant been given an operating license to operate with this type of peril?

Response:

See the response to question 2.

Monroe City-County Question:

5. What are the exemptions that Detroit Edison has received in regard to complying with license conditions and safety regulations required by the NRC's standards and laws?

Response:

The following exemptions have been granted for Fermi:

<u>DATE</u>	<u>DESCRIPTION</u>
April 15, 1988	Exemption from Paragraph III.D.3 of Appendix J to 10 CFR Part 50. This allows for a one-time exemption to the testing of the interior containment isolation valves for the RHR system (three valves).
November 13, 1987	Exemption to GDC 56, Appendix A of 10 CFR Part 50, for Primary Containment Radiation Monitor Isolation Valves.
July 31, 1986	Exemption to GDC 56, Appendix A of 10 CFR Part 50. This exemption permits postponement of full compliance with GDC 56 for the traversing in-core probe (TIP) nitrogen purge line until the first scheduled refueling outage.
July 30, 1986	Exemption to 10 CFR 50.44. Permits postponement of the inerting of the Fermi-2 primary containment from December 21, 1985, until either completion of the startup test program or until the reactor has operated for 120 effective full power days.
July 24, 1986	Exemption to GDC 56, Appendix A of 10 CFR Part 50. Limited period to allow a single penetration to have two isolation valves outside containment rather than one valve inside and one valve outside containment. Exemption would extend until the first scheduled refueling outage.
July 11, 1985	Exemption to 10 CFR Part 50, Appendix E. This exemption would delay the conduct of a full participation offsite emergency planning exercise. (Section IV.F of Appendix E requires that this exercise be conducted within one year before the issuance of the first operating license at the site for full power and prior to operation above 5% of rated power. The exercise was scheduled for October.
April 11, 1985	Partial exemptions from Appendix J of 10 CFR Part 50. The exemptions would (1) allow Type C testing of the main steam isolation valves to be conducted at a differential pressure less than that required by Paragraph III.C.2 of Appendix J, and (2) eliminate the full pressure test required by Paragraph III.D.2(b)(ii) of Appendix J for normal air lock opening and substitute a seal leakage test to be conducted at a pressure specified in the Technical Specifications.