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

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MAY 5 1981

Docket No.: 50-341

Mr. Harry Tauber
Vice President
Engineering & Construction
Detroit Edison Company
2000 Second Avenue
Detroit, Michigan 48226

Dear Mr. Tauber:

Subject: Requests For Additional Information In Fermi 2 Operating License Application

As a result of our continuing review of the operating license application for the Enrico Fermi Atomic Power Plant Unit 2, we have developed the enclosed requests for additional information.

Please amend your application to comply with the requirements listed in the enclosure. Our review schedule is based on the assumption that the additional information will be available for our review by May 18, 1981. If you wish clarification of the requests or if you cannot meet these dates, please telephone the Licensing Project Manager, L. Kintner, within 7 days after receipt of this letter.

Sincerely,

Robert L. Tedesco, Assistant Director
for Licensing
Division of Licensing

Enclosures:
Requests for Additional
Information

cc w/encl.: See next page

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ENCLOSURE 1

REQUESTS FOR ADDITIONAL INFORMATION IN THE SAFETY REVIEW

ENRICO FERMI ATOMIC POWER PLANT UNIT 2

DOCKET NO. 50-341

Requests by the following branches in NRC are included in this enclosure. Requests and pages are numbered sequentially with respect to previously transmitted requests.

BRANCH	PAGE NO.
Materials Engineering Branch	121-16 through 121-22

121.0 Materials Engineering Branch

Compliance with Appendix G, 10 CFR Part 50

121.16 Section I.B of Appendix G, 10 CFR Part 50 requires that all ferritic welds and weld heat-affected zones in the primary coolant pressure boundary be impact tested according to the requirements of Appendix G. Fermi Unit 2 nonbeltline welds and all heat-affected zone materials have not been qualified by impact tests. The applicant has estimated an RT_{NDT} of 0° F for nonbeltline welds based on weld qualification test results and has assumed RT_{NDT} values for heat-affected zone materials to be the same as for the base materials. In order to demonstrate compliance with the requirements of Appendix G, Section I.B., the applicant should provide the following information:

(a) Impact test data on nonbeltline welds from weld qualification tests to indicate that the estimation of 0° F for RT_{NDT} is conservative.

(b) Impact test data on heat-affected zone materials from similar BWR reactors or general literature to demonstrate that RT_{NDT} values for heat-affected zone materials are the same as the values for base materials.

121.17 Section III.A of Appendix G requires that both unirradiated and irradiated ferritic materials should be tested for fracture toughness properties by means of the Charpy V-notch test specified by paragraph NB-2321.2 of the ASME Code. In addition, when required by the ASME Code, unirradiated ferritic materials should be tested by means of the drop weight test specified by paragraph NB-2321.1 of the ASME Code. Reactor vessel beltline welds, closure head flange, vessel flange and feed waternozzle forging in Fermi Unit 2 were not qualified by drop weight tests. But RT_{NDT} values have been reported in

the FSAR for these components. The applicant is required to explain the basis for these estimations since the ASME Code procedure for evaluating RT_{NDT} requires both NDT temperatures from drop weight tests and Charpy V-notch test results.

121.18 Fracture toughness data presented in Table 5.2-6a and Table 1 of Amendment 23 of Fermi Unit 2 FSAR are not adequate enough to determine if the fracture toughness requirements of Appendix G, 10 CFR Part 50 are met or to determine the degree of conservatism involved in the alternate procedures employed for evaluating the impact test data. The applicant should supply the following additional information:

(a) Identify each ferritic material used in a pressure-retaining component of the reactor coolant pressure boundary, as defined by paragraphs I.A, I.B and I.C of Appendix G, 10 CFR Part 50, by material specification and heat number and by location in the component.

(b) Provide the existing impact test data for each material listed in (1) as required by the testing programs of Section III of the ASME Code and Appendix G, 10 CFR Part 50. The results to be reported should include Charpy V-notch energy and laterial expansion values from each test, the test temperature, Charpy specimen location and orientation and T_{NDT} temperature from the drop weight test. List also the calculated RT_{NDT} from the above data.

Also, supply the following additional information for those materials listed in a) that are in the reactor vessel beltline region, as defined by Paragraph II.H, Appendix G, 10 CFR Part 50:

(c) Full Charpy V-notch curves, including data points, reported in impact energy and lateral expansion as a function of temperature

(d) The minimum upper shelf energy

(e) Chemical analyses, particularly Cu and P

(f) Estimated change in RT_{NDT} and upper shelf energy as a function of neutron fluence.

From the data requested in items (a) to (f), identify the most limiting material in the reactor coolant pressure boundary at the beginning-of-life and at the end-of-life and the procedure used to determine the limiting materials.

For those components that were not tested in full compliance with the requirements of Appendix G, 10 CFR Part 50, state the alternate procedures adopted in estimating RT_{NDT}, provide a technical justification to indicate that these RT_{NDT} estimations are conservative, and identify any impact test data from WRC Bulletin 217 or other reactors that were used in developing any correlations for defining RT_{NDT}.

121.19 Section III.C.2 of Appendix G requires, in part, that the materials used to prepare test specimens for the reactor vessel beltline region should be taken directly from excess material in the vessel shell. The Fermi Unit 2 impact test program does not comply with this requirement explicitly since beltline test welds were not made on the same heat of the base plate as the vessel shell. The applicant is required to demonstrate that his test welds are metallurgically equivalent to the vessel beltline welds with equivalent impact properties. The information to be provided should include the types of weld wire and flux materials used and the postweld heat treatment given to the test welds and the reactor vessel beltline welds.

121.20 In order to demonstrate compliance with the fracture toughness requirements of 10 CFR Part 50, Appendix G, Section IV.A.3, it will be necessary to demonstrate that the main steamline piping, the main steamline isolation valves, and the safety/relief valves of Fermi Unit 2 comply with paragraph NB-2300 "Fracture Toughness Requirements for Materials", of the 1971 ASME Code, Section III, Summer 1972 Addenda.

121.21 Table 5.2-9 of the Fermi FSAR states that current toughness requirements for closure head studs are met at 10° F and that this temperature would be the lowest service temperature. The current requirements for bolting materials of nominal diameter over 4" are 25 mils lateral expansion and 45 ft-lbs Charpy energy at a temperature no higher than the preload temperature or the lowest service temperature whichever is less. The reported minimum values for the Fermi vessel closure head studs are 50 ft-lbs energy and 3 mils lateral expansion at 10° F and hence the current requirements have not been met. The applicant should provide a clarification of this discrepancy and also provide all existing impact test data for the bolting materials in terms of Charpy energy and lateral expansion values obtained in each test and the test temperature.

121.22 Section IV.B of Appendix G, 10 CFR Part 50 requires that the reactor vessel beltline materials should have a minimum upper shelf energy, as determined from Charpy V-notch tests on unirradiated specimens in accordance with paragraph NB-2322.2(a) of the ASME Code, of 75 ft-lbs. Minimum upper shelf energy values are not available for Fermi Unit 2 vessel welds since no vessel weld impact test requirements were in existence at the time the vessel was ordered. It is stated in the Fermi FSAR, however, that the beltline welds in the Fermi vessel have a lowest measured Charpy energy of 47 ft-lbs at +10° F. Based on weld toughness data obtained on similar BWR reactors it is claimed that when a value

121.22 (cont'd) of 47 ft-lbs energy is obtained at +10° F, the upper shelf energy exceeds 75 ft-lbs. The applicant should present all the existing impact test data for the beltline welds and supply or identify the impact test data from other BWR reactors used to develop the above correlation. This information is required to determine whether the applicant's evaluation of the weld test data is as conservative as that required by a strict compliance with Appendix G.

Compliance with Appendix H, 10 CFR Part 50

121.23 As a supplement to the table given in response to Question 121.11, provide a sketch of the beltline of the reactor vessel showing the location of all of the beltline plates, welds, and surveillance capsules. Provide an EOL isofluence map of the reactor vessel.

121.24 Additional information is required in the response to Question 121.12; specifically, provide an explanation detailing why limiting material was not used for the surveillance specimens and justify the selection of the representative material actually used. Furthermore, indicate why the location from where the surveillance material was removed is technically equivalent to that required by Appendix G, 10 CFR Part 50.

Also submit details of the test specimen fabrication, including the justification for the use of the simulated stress relief (40 hours at 1150 F).

In addition to the above, include in the Fermi Unit 2 FSAR and Technical Specifications a series table that provide the following information for each surveillance specimen capsule:

(a) The actual surveillance materials in the capsule.

121.24 (cont'd) (b) The beltline material including the plate and/or weld numbers, from which each surveillance material was obtained,

(c) The test specimen type(s), and their orientation, for each surveillance material, (Note that the base metal tensile specimen orientation relative to the plate rolling direction, and the HAZ impact and tensile specimens' orientation relative to the weld and plate thickness, are not currently specified in the FSAR. This information will be required.)

(d) The actual location and attachment method of the capsule in the reactor vessel.

(e) The lead factor for the capsule calculated with respect to the 1/4 wall thickness location.

(f) The proposed loading schedule of the capsule into the Fermi 2 reactor vessel, and

(g) The proposed time of capsule withdrawal calendar years and effective full power years.

121.25 The predicted shifts in RT_{NDT} for the Fermi Unit 2 beltline welds exceed 100° F. Following ASTM E185-73, Annex A1. Surveillance Material Selection Procedures, the limiting material is the one which has the highest Cu content when P contents and initial RT_{NDT} values are equivalent.

Detroit Edison has used the weld with the highest initial RT_{NDT} as the limiting material which is inconsistent with the recommended procedure. Applicant should explain why the procedure recommended in ASTM E185-73 was not used and should provide a technical justification for his selection procedure for weld specimens.

121.26 The minimum number of serveillance specimens for each exposure set (capsule) is defined in ASTM E185-73. Twelve (12) Charpy V-notch specimens each of base metal, weld metal and heat-affected zone material are required for each capsule. The required total number of Charpy specimens for the Fermi Unit 2 surveillance program is 108 for 3 capsules. According to the Fermi Unit 2 FSAR, only 84 specimens are included in the surveillance program. The applicant is required to explain the reason for this discrepancy and provide a justification for the smaller number of specimens employed.

121.27 Indicate the type and number of dosimeters and temperature monitors included in each surveillance capsule. This information is required to show compliance with ASTM E185-73.

ENCLOSURE 2

REQUEST FOR ADDITIONAL INFORMATION
FERMI 2
DOCKET NO. 50-341

Our review of your "Responses to NUREG-0737" in Appendix H to the FSAR has resulted in the need for additional information. Pages are numbered sequentially. The alpha numeric item designations correspond to the items in NUREG-0737. The following requests are included in this enclosure.

II.B.3 Post Accident Sampling

This section requires operating license applicants to provide a description of the implementation of the position and clarification including P&IDs, together with either (a) a summary description of procedures for sample collection, sample transfer or transport, and sample anslysis, or (b) copies of procedures for sample collection, sample transfer or transport, and sample analysis.

Sufficient information has not been provided for the staff to evaluate compliance with the following:

1. II.B.3 clarifications 2.a., 2.b., 2.c., 2.d., 4, 5, 7, 8, 10, 11.a. and 11.b.

2. Summary of procedures for sample collection.

3. Description of methods of sample transfer or transport.

4. Summary of procedures for sample analysis.

Specific information needed to determine compliance with the requirements of II.B.3 is as follows:

1.) 2a - A discussion of what radionuclides will be looked for that may be indicators of the degree of core damage. Information on the methodology/procedures that will be used to relate specific radionuclide analysis inforamtion to the degree of core damage. (e.g., fuel clad perforation, abnormally high fuel temperature or onset of fuel melt)

2.) 2b., 11 a - A discussion of why the sample locations which were selected are representative of containment hydrogen. Included should be statements on why the selected location reasonably represents the containment and what assures that the sample nozzle will not become blocked. A description of how recirculation is accomplished. A description of how a representative sample is obtained and analyzed.

3.) 2c, 4, 5, 7, 11a - A discussion of why the liquid sample locations which were selected are representative of coolant in the core.

4.) 2d, 8 - A description of any automatic on-line monitors used and their accuracy and reliability.

5.) 10 - A description of the reactor coolant analytical procedures and accuracy for:

- Total dissolved gases or hydrogen
- Chloride
- Boron
- pH
- Dissolved oxygen - in the event chloride exceeds technical specification limits.

6.) 11b - A description of provisions to preclude contamination spread from the sample sink during sampling operations.

7.) Verify that automatic containment isolation valves in all post accident sampling systems close on contamination isolation or safety injection signals. Verify that all remotely operated valves have assured power supplies so they can be reopened after an accident, without clearing the isolation signal. Verify that the sample system remotely operated valves which are located in the secondary containment are environmentally qualified to assure operability as long as secondary containment is unaccessible.

8.) Provide P&ID's or flow diagrams of the piping station inside of the secondary containment and for the post accident sample station which is outside of the secondary containment.