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

MAR 2 1979

Docket No. 50-341

Dr. Wayne H. Jens
Assistant Vice President
Engineering & Construction
The Detroit Edison Company
2000 Second Avenue
Detroit, Michigan 48226

Dear Dr. Jens:

SUBJECT: REQUESTS FOR ADDITIONAL INFORMATION IN FERMI 2 FSAR

As a result of our continuing review of the Final Safety Analysis Report (FSAR) for the Enrico Fermi Atomic Power Plant Unit 2, we have developed the enclosed requests for additional information.

Please amend your FSAR 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 April 27, 1979. This is the latest date for filing information to be considered in our Safety Evaluation Report for Fermi 2. If you cannot meet this date, please inform us within 7 days after receipt of this letter so that we may revise our scheduling.

Sincerely,

John F. Stolz

John F. Stolz, Chief
Light Water Reactors Branch No. 1
Division of Project Management

Enclosure:
Requests for Additional
Information

cc w/enclosure:
See page 2

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ENCLOSURE

REQUESTS FOR ADDITIONAL INFORMATION

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.

110.0 MECHANICAL ENGINEERING BRANCH

The following include information requests which were verbally transmitted by the Mechanical Engineering Branch staff in the February 6 and 7, 1979 meeting with Detroit Edison in Bethesda.

110.11 Postulated Primary System Pipe Breaks (LOCA) in the Reactor Vessel Cavity Resulting in Asymmetric Pressure Loads on the Reactor Vessel, Internals, and Vessel Supports. Responses to Asymmetric LOCA Loads to be Combined with Responses to SSE Loads.

The results of analyses performed to assess the effects of exposure to load effects resulting from these postulated breaks were summarized in the February 6, 1979 meeting. Results of these analyses to be formally submitted on the Fermi 2 docket should include:

(1) Design buckling stress criteria used for the Reactor Vessel Support Skirt together with the calculated stress level in buckling under the combination of responses from LOCA and SSE. The method of analysis used to determine the skirt buckling stress level should be described in detail.

(2) Calculated and allowable stress level for the support skirt to vessel interface weld.

(3) Provide a detailed description of the method used to combine dynamic responses resulting from the various LOCA induced load effects. The description should specifically address responses to all LOCA related loads i.e., annulus pressurization, pipe rupture reaction forces, and asymmetric pressure forces on the reactor internals. Provide justification for the method of combination used.

(4) Provide a description (tensile or compressive) of the stresses induced in the reactor vessel pedestal bolts resulting from exposure to cavity pressurization forces.

110.12 Mark I Containment Modifications - Short Term Program.

Results of analyses performed were summarized in the February 7, 1979 meeting and a topical report documenting these results has been submitted. Additionally as requested in the February 7, 1979 meeting, describe how the operability of active pumps and valves and the functional capability of all essential piping that can be affected by containment vibratory or uplift loads has been evaluated and is assured by the containment modifications which were described in the meeting.

110.13 (3.9) As discussed in the February 6, 1979 meeting in Bethesda, provide confirmation that the reactor internals preoperational vibration assurance program will be in accordance with the G.E. Topical Report NEDE 24057-P.

110.14 (3.10) As stated in the February 7, 1979 meeting in Bethesda, a review of the design adequacy of your safety-related electrical, mechanical, and instrumentation equipment will be performed by our Seismic Qualification Review Team (SQRT). A site visit will be necessary to inspect and evaluate selected pieces of equipment. The SQRT effort will be primarily focused on two subjects. The first is the adequacy of the original single axis-single frequency tests or analyses of equipment qualified per the criteria of IEEE Std. 344-1971.

The second subject is the qualification of equipment for the combined seismic and hydrodynamic vibratory loadings. The frequency of the hydrodynamic vibratory loadings, resulting from the response of the Mark I containment torus to safety/relief valve actuation and postulated Loss of Coolant accidents, may exceed 33 Hz and negate the original assumption of a component's rigidity in some cases.

In order to assist the SQRT in its preparation for the site visit, we request that you transmit to NRC a list of the safety related electrical, mechanical, and instrumentation equipment that can be exposed to Mark I containment vibratory loads. It is expected that the majority of this equipment will be located in or adjacent to the torus.

Based upon this equipment list and information already contained in the Fermi 2 FSAR, the SQRT will determine which equipment will be reviewed in detail during the site visit. Prior to the site visit you will be advised as to which pieces of equipment the SQRT will include in its review.

413.0 QUALITY ASSURANCE BRANCH - INITIAL TESTS AND OPERATION

413.16(1)A (RSP) Your response to item 413.16 (part 1) is not acceptable. The test description does not provide assurance that the total reactor protection system response time is consistent with your accident analysis assumptions. It is our position that you modify your description of the preoperational test to include the following:

(1) measure the response time of each RPS trip comparator;

(2) account for process-to-sensor hardware (e.g., instrument lines, hydraulic snubbers) delay times; and

(3) provide assurance that the response time of each primary sensor is acceptable.

Note: Item 3 can be accomplished by measuring the response time of each sensor during the preoperational test, stating that the response time of each sensor will be measured by the manufacturer within two years prior to fuel loading, or describing the manufacturer's certification process in sufficient detail for us to conclude that the sensor response times are in accordance with design.

413.16(3)A (RSP) Your response to item 413.16 part (3) is not acceptable. It states that the DC system components required to operate at the end of the battery design basis event load period will be demonstrated operable at that voltage. It is our position that you also demonstrate the operability of other loads on the 130/260 Vdc system at the lowest voltage at which they may be called upon to operate.

413.16(4)A (RSP) Your response to item 413.16 part (4) is not acceptable. It is our position that you provide acceptance criteria and their bases for turbine stop valve/control valve closure and turbine bypass valve opening response times for the turbine trip and generator load rejection test. It is also our position that you (1) maintain the loss of offsite power in test No. 29 for at least 30 minutes in order to demonstrate proper operation of equipment and support systems (e.g., ventilation and pump seal water systems) that are powered from onsite emergency power sources, and (2) provide acceptance criteria and their bases for start and load times of the emergency diesel generators.

413.17 (RSP) Our review of event reports has disclosed many failures of the high pressure coolant injection (HPCI) and the reactor core isolation cooling (RCIC) systems. It appears that many of the causes for these failures should have been detected and corrected during initial testing of the systems. It is our position that you should consider these failures and their causes in developing the initial test programs for the HPCI and RCIC systems in order to establish the necessary confidence in the reliability of these systems. One way of demonstrating the reliability of these systems acceptable to us is to conduct at least five successful, consecutive, cold, quick starts of each system. Modify the descriptions of the preoperational and/or startup tests of the HPCI and RCIC systems to show that the above described concern is addressed.

413.18 (RSP) For startup test No. 23, "Main Steam Isolation Valves," modify the test method to measure the full travel of the valves or provide technical justification for extrapolating the full closure time when only measuring 90 percent.

413.19 For startup test No. 24, "Relief Valves," provide acceptance criteria for valve capacity (total and ADS capacities) and describe how they relate to Chapter 15 assumptions.

413.20 The description of your requirements to review and approve the results of power ascension tests at "preselected test conditions" as presented in Amendment 12 is not adequate. Modify Section 14.1.4.7 to state which test conditions they are.

413.21 Appendix A of your FSAR states that initial tests will be conducted in accordance with Regulatory Guide 1.68.2, Revision 1, July 1978, "Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants" and Regulatory Guide 1.108, Revision 1, July 1977, "Periodic Testing of Diesel Generator Units Used as Onsite Power Systems at Nuclear Plants." Modify the test descriptions of your remote shutdown test and diesel generator test to describe how you will perform preoperational and startup tests in conformance with these regulatory guides.

413.22 Clarify the description of your initial test program to specifically identify any startup tests that are not considered "essential" to demonstrate the operability of structures, systems, and components that meet any of the criteria listed below.

(1) Those that will be used for safe shutdown and cooldown of the reactor under normal plant conditions and for maintaining the reactor in a safe condition for an extended shutdown period; or

(2) Those that will be used for safe shutdown and cooldown of the reactor under transient (infrequent or moderately frequent events) conditions and postulated accident conditions and for maintaining the reactor in a safe condition for an extended shutdown period following such conditions; or

(3) Those that will be used for establishing conformance with safety limits or limiting conditions for operation that will be included in the facility technical specifications; or

(4) Those that are classified as engineered safety features or will be used to support or ensure the operations of engineered safety features within design limits; or

(5) Those that are assumed to function or for which credit is taken in the accident analysis for the facility, as described in the FSAR; or

(6) Those that will be used to process, store, control, or limit the release of radioactive materials.

413.23 (RSP) Main Steam Line Isolation Valve Leakage Control System Preoperational Test. It is our position that the system be tested under conditions that approximate actual service conditions. Therefore, modify the abstract to include final system testing under hot conditions.

413.24 (RSP) Control Rod Drive System. Provide technical justification for the acceptance criteria for control rod scram times for vessel dome pressure $\leq$ 950 psig.

122.0 MATERIALS ENGINEERING BRANCH - METALLURGY SECTION

122.2 Recent operating experience at BWR plants has indicated degraded performance of the design and materials in the safe ends and thermal sleeves of the recirculation nozzles of the reactor vessel. Provide a sketch of the design and the materials used in the Enrico Fermi Atomic Power Plant No. 2 for these areas. Provide an evaluation of the design that will give reasonable assurance that these items will not degrade in service.

122.3 Recent operating experience at BWR plants has indicated degraded performance of the design and materials in the collet retainer tube, index tube, and piston tubes of the control rod drive (CRD) mechanisms. Provide a sketch of the design, materials used and an inspection program for these items used in the Enrico Fermi No. 2 plant. Provide an evaluation of the design, materials and inspection program that will give reasonable assurance that these items will not degrade in service.

122.4 Provide a description of the implementation of NUREG-0313 (MTEB BTP-7), stainless steel cracking by IGSCC, including the isolation condenser lines and shut down heat exchanger lines. The response should include the materials of construction and the methods used for mitigating stress corrosion cracking in the referenced lines. Based on the incidence of IGSCC in recirculation-riser piping in Japan, an augmented inservice inspection program should be developed for these lines (recirculation-riser) if they do not meet the guidelines stated in Part II of NUREG-0313. We recommend that the augmented inservice inspection program conform to that described for nonconforming, service sensitive lines in NUREG-0313. The augmented inservice inspection program should be described and be made a part of the complete inservice inspection program for the plant.

042.0 CONTAINMENT SYSTEMS BRANCH

042.12A The response to our request 042.12 regarding the annulus pressurization analysis for the sacrificial shield is insufficient. The effects of the higher initial temperature in the annulus space and the difference between the peak calculated shield wall differential pressure and the design pressure of 50 psid were not addressed in the reports NUS-3129 and SL-3647. Therefore, provide the following information:

a. The basis as indicated in the meeting held on February 6, 1979, for concluding that the effect of an increase in the lateral load and in the overturning moment due to a higher initial annulus temperature would not significantly affect the reactor pressure vessel support design.

b. The documentation of the results of a dynamic analysis using the actual pressure transient provided in NUS-3129 for the shield wall design as indicated in the February meeting; the difference between the local peak calculated shield wall differential pressure of 85 psid and the design pressure of 50 psid should be addressed.

c. The transient uplift force on the vessel for the reactor vessel supports since it appears that your presentation did not include the contribution of the uplift force in the vessel support design.

042.14A The response to our request 042.14 regarding the recombiner system is inadequate. The Fermi-2 containment will not be inerted with nitrogen, although it was originally designed to be inerted. Therefore, provide the following information:

(a) Update the inconsistency in the FSAR regarding the primary containment environment which is currently described as a non-inerted environment; an inerted containment was previously described in Section 6.2.5 of Fermi-2 FSAR.

(b) The assumption and the analysis to show the steaming rate and the maximum (conservative) time at which steam addition to the drywell can be considered to occur.

(c) A conservative estimate of the time periods when the pressure exceeds the recombiner operating pressure of 20 psig given in the reference Report AI-77-55 of FSAR, since there exists phases of the transient where containment pressure exceeds 20 psig.

(d) An analysis of the post-LOCA combustible gas concentration based on the same assumptions described in Section 6.2.5 of FSAR. Provide all the necessary input data which are consistent with those provided in the EF2-16014 report (Post-LOCA Hydrogen Control System dated November 20, 1973).

(e) It is our position that the leak tight integrity of the recombiner system should be demonstrated since following the postulated LOCA the recombiner system becomes an extension of the containment. Therefore, provide the appropriate commitment to demonstrate the leak tight integrity.

042.15A Regarding our request 042.15, relating to the suppression pool temperature limit, you stated that the information will be submitted after your selection of the type of quencher device to be installed in your facility. We believe, however, that the information related to the suppression pool temperature transient following SRV operation is independent of the type of quencher device that is used. Since this information is needed for our evaluation on the capability of Fermi 2 plant regarding the concern of steam quenching instability, you are required to provide this information.

042.18A The response to our question 042.18 regarding the testing of the isolation valves is inadequate. It is our position that those isolation systems given in Table 6.2.2 with Note 9 be tested in accordance with the requirements specified in Appendix J to 10 CFR 50, since the test is needed to verify the analytical approach and the condition of any potentially degradable component.

042.27 Provide the following information with respect to the hydrogen monitoring system.

(1) Verification on whether the hydrogen/oxygen analyzer is still performing its monitoring function as the result of the change from an inerted to a non-inerted containment system.

(2) Discussion on the environmental qualification tests performed to assure that the equipment is capable of monitoring the post accident containment atmosphere.

(3) Discussion on the distribution of hydrogen/oxygen sampling points in the containment. Justify that the system is adequate to detect potential non-uniform hydrogen concentration within these volumes.

042.28 Section 6.2.3.3 of the FSAR states that the pressure of the secondary containment volume after a LOCA has been studied. Therefore, provide the analysis and the results of the pressure transient for the secondary containment. Describe the planned leak test for verifying the inleakage assumption and the drawdown time for reestablishing -0.25 inches of water gauge pressure. Provide the appropriate commitments for conducting these tests. Identify the location of pressure sensors in the secondary containment volume during the testing of the SGTS. Discuss the adequacy of the sensor distribution to identify regions of potential exfiltration.

042.29 Provide an evaluation of the containment purge system based upon the provisions of enclosed Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operation."

Branch Technical Position CSB 6-4

CONTAINMENT PURGING DURING NORMAL PLANT OPERATIONS

A. BACKGROUND

This branch technical position pertains to system lines which can provide an open path from the containment to the environs during normal plant operation; e.g., the purge and vent lines of the containment purge system. It supplements the position taken in Standard Review Plan 6.2.4.

While the containment purge system provides plant operational flexibility, its design must consider the importance of minimizing the release of containment atmosphere to the environs following a postulated loss-of-coolant accident. Therefore, plant designs must not rely on its use on a routine basis.

The need for purging has not always been anticipated in the design of plants, and therefore, design criteria for the containment purge system have not been fully developed. The purging experience at operating plants varies considerably from plant to plant. Some plants do not purge during reactor operation, some purge intermittently for short periods and some purge continuously.

The containment purge system has been used in a variety of ways, for example, to alleviate certain operational problems, such as excess air leakage into the containment from pneumatic controllers, for reducing the airborne activity within the containment to facilitate personnel access during reactor power operation, and for controlling the containment pressure, temperature and relative humidity. However, the purge and vent lines provide an open path from the containment to the environs. Should a LOCA occur during containment purging when the reactor is at power, the calculated accident doses should be within 10 CFR 100 guideline values.

The sizing of the purge and vent lines in most plants has been based on the need to control the containment atmosphere during refueling operations. This need has resulted in very large lines penetrating the containment (about 42 inches in diameter). Since these lines are normally the only ones provided that will permit some degree of control over the containment atmosphere to facilitate personnel access, some plants have used them for containment purging during normal plant operation. Under such conditions, calculated accident doses could be significant. Therefore the use of these large containment purge and vent lines should be restricted to cold shutdown conditions and refueling operations.

The design and use of the purge and vent lines should be based on the premise of achieving acceptable calculated offsite radiological consequences and assuring that emergency core cooling (ECCS) effectiveness is not degraded by a reduction in the containment backpressure.

Purge system designs that are acceptable for use on a non-routine basis during normal plant operation can be achieved by providing additional purge and vent lines. The size of these lines should be limited such that in the event of a loss-of-coolant accident, assuming the purge and vent valves are open and subsequently close, the radiological consequences calculated in accordance with Regulatory Guide 1.3 and 1.4 would not exceed the 10 CFR 100 guideline values. Also, the maximum time for valve closure should not exceed five seconds to assure that the purge and vent valves would be closed before the onset of fuel failures following a LOCA.

The size of the purge and vent lines should be about eight inches in diameter for PWR plants. This line size may be overly conservative from a radiological viewpoint for the Mark III BWR plants and the HTGR plants because of containment and/or core design features. Therefore, larger line sizes may be justified. However, for any proposed line size, the applicant must demonstrate that the radiological consequences following a loss-of-coolant accident would be within 10 CFR 100 guidelines values. In summary, the acceptability of a specific line size is a function of the site meteorology, containment design, and radiological source term for the reactor type; e.g., BWR, PWR or HTGR.

B. BRANCH TECHNICAL POSITION

The system used to purge the containment for the reactor operational modes of power operation, startup, and hot standby; i.e., the on-line purge system, should be independent of the purge system used for the reactor operational modes of hot shutdown, cold shutdown, and refueling.

1. The on-line purge system should be designed in accordance with the following criteria:
   a. The performance and reliability of the purge system isolation valves should be consistent with the operability assurance program outlined in MEB Branch Technical Position MEB-2, Pump and Valve Operability Assurance Program. (Also see Standard Review Plan 3.9.3.) The design basis for the valves and actuators should include the buildup of containment pressure for the LOCA break spectrum, and the purge line and vent line flows as a function of time up to and during valve closure.
   b. The number of purge and vent lines that may be used should be limited to one purge line and one vent line.
   c. The size of the purge and vent lines should not exceed about eight inches in diameter unless detailed justification for larger line sizes is provided.
   d. The containment isolation provisions for the purge system lines should meet the standards appropriate to engineered safety features; i.e., quality, redundancy, testability and other appropriate criteria.
   e. Instrumentation and control systems provided to isolate the purge system lines should be independent and actuated by diverse parameters; e.g., containment pressure, safety injection actuation, and containment radiation level. If energy is required to close the valves, at least two diverse sources of energy shall be provided either of which can affect the isolation function.
   f. Purge system isolation valve closure times, including instrumentation delays, should not exceed five seconds.
   g. Provisions should be made to ensure that isolation valve closure will not be prevented by debris which could potentially become entrained in the escaping air and steam.

2. The purge system should not be relied on for temperature and humidity control within the containment.

3. Provisions should be made to minimize the need for purging of the containment by providing containment atmosphere cleanup systems within the containment.

4. Provisions should be made for testing the availability of the isolation function and the leakage rate of the isolation valves, individually, during reactor operation.

5. The following analyses should be performed to justify the containment purge system design:
   a. An analysis of the radiological consequences of a loss-of-coolant accident. The analysis should be done for a spectrum of break sizes, and the instrumentation and setpoints that will actuate the vent and purge valves closed should be identified. The source term used in the radiological calculations should be based on a calculation under the terms of Appendix K to determine the extent of fuel failure and the concomitant release of fission products, and the fission product activity in the primary coolant. A pre-existing iodine spike should be considered in determining primary coolant activity. The volume of containment in which fission products are mixed should be justified, and the fission products from the above sources should be assumed to be released through the open purge valves during the maximum interval required for valve closure. The radiological consequences should be within 10 CFR 100 guideline values.
   b. An analysis which demonstrates the acceptability of the provisions made to protect structures and safety-related equipment; e.g., fans, filters and ductwork, located beyond the purge system isolation valves against loss of function from the environment created by the escaping air and steam.
   c. An analysis of the reduction in the containment pressure resulting from the partial loss of containment atmosphere during the accident for ECCS backpressure determination.

d. The allowable leak rates of the purge and vent isolation valves should be specified for the spectrum of design basis pressures and flows against which the valves must close.

310.0 ACCIDENT ANALYSIS BRANCH

310.25 (2.1) Identify the local, State, or federal agency which has jurisdiction over the waters of Lake Erie within the exclusion area and discuss what arrangements have been made to control the movement of people in this area in case of an emergency, as required by 10 CFR Part 100.

310.26 (2.1) The population center distance of 5.5 miles is measured to the nearest corporate boundary of the city of Monroe, Michigan. Discuss whether the corporate boundary is a reasonable approximation of the boundary of the densely populated area based on the actual population distribution. Our position is that the population center distance should be determined by considering population distribution rather than political boundaries.

412.0 QUALITY ASSURANCE BRANCH - CONDUCT OF OPERATIONS

412.17 (13.1.2.1) Revise Figure 13.1-2 to show the number of persons assigned to the position of Chemical Technicians.

412.18 (13.1.2.1) Provide the qualification requirements for the position of Performance Engineer shown in Figure 13.1-2; and provide the resumes of the persons filling that position and the positions of Assistant Operations Engineer, Chemical Engineer, and Reactor Engineer.

412.19 (13.4.3.1) It is our position that Section 13.4.3.1 should specifically describe the review responsibilities of the Onsite Review Organization (OSRO). These review responsibilities should include those described in Section 6.5.2.7 of the Standard Technical Specifications.

412.20 (13.4.3.2) It is our position that Section 13.4.3.2 should specifically describe the audit responsibilities of the Independent Review and Audit Group (IRAG). The IRAG responsibilities should include provisions for assuring audits as described in Section 6.5.2.8 of the Standard Technical Specifications.