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

APR 2 5 1980

Docket No: 50-341

Dr. Wayne H. Jens
Assistant Vice President
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Detroit Edison Company
2000 Second Avenue
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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 June 4, 1980. 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:
See next page

APR 25 1980

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# ENCLOSURE

## REQUESTS FOR ADDITIONAL INFORMATION

## ENRICO FERMI ATOMIC POWER PLANT UNIT 2

## DOCKET NO. 50-341

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

| Branch | Page No. |
|---|---|
| Reactor Systems Branch | 212-28 through 212-36 |

212.0 REACTOR SYSTEMS BRANCH

212.67A (6.3) Review procedure III.5 of SRP Section 6.3 recommends that prior to installation, representative active components used in the ECCS will be proof-tested under environmental conditions and for time periods representative of the most severe operating conditions to which they may be subjected.
Insufficient information has been presented in the FSAR or in the response to Q212.67 to determine that proof-testing has been completed for ECCS pumps. Provide a description of vendor in-shop testing and testing after installation.
Provide the expected service life of the ECCS pumps and supply the maximum expected accumulated operating time for the ECCS pumps during the life of the plant as indicated below:
a) In-shop testing
b) Pre-operational testing
c) Monthly testing
d) Yearly testing
e) Post-LOCA
f) Shutdown

212.75 (6.3) Provide characteristic curves for the RHR, HPCI, and core spray pumps. Brake horsepower, efficiency, and NPSH should be included on these curves.

212.76 (6.3) Section 6.3.2.14 addresses NPSH considerations for the LPCI and the core spray systems, but only mentions the HPCI system. Provide NPSH calculations to support your claim that adequate NPSH is provided for HPCI operation. Assurance should be provided that adequate NPSH will be available during switchover from the condensate storage tank to the suppression pool.

212.77 (6.3) Describe any precautions (mechanical or administrative) taken to prevent vortex formation and air ingestion during operation of the ECCS pumps.
The description should include both primary and secondary water supplies (i.e., HPCI uses both the condensate storage tank and the suppression pool).

212,34A (6.3) The response to Q212.34 is incomplete. Section 6.3.2.2.7 describes the leak detection system provided for the ECCS suction lines but does not address the possibility of ECCS impairment from a reduced pool level. Discuss preoperational tests to be performed to demonstrate that there is no impairment of ECCS function due to lowered suppression pool level.

212.78 (6.3) The response to Q212.39 requires additional clarification. We are concerned that administrative controls alone may not be adequate to ensure that a system is operational.

Provide a list of those manual valves critical to the operation of the ECCS and indicate which valves have position indication in the control room and which valves are controlled only by administrative procedures.

212.79 Table 6.3-10 indicates that the maximum peak cladding temperature for
(6.3) the DBA is 2084°F. Figure 6.3-9a is intended to be a graphical representation of Table 6.3-10, but the PCT for the DBA is shown on the figure to be about 2150°F. Resolve this discrepancy.

212.80 Section 8.2.8 of the FSAR states that the suppression pool water is
(6.3) normally maintained at condensate quality by circulating about 500 gpm through the Torus Water Management System and can be entirely replaced within a 48 hour period.
What administrative controls or other constraints are provided to ensure that the suppression pool is not drained when the need for the ECCS systems could be required?

212.81 Section 6.3.2.14.1 of the FSAR addresses the subject of pump run-out
(6.3) for the RHR pumps.
Provide assurances that the HPCI and/or core spray pumps have been analyzed for pump run-out conditions. Describe any tests and associated possible ECCS lineups to demonstrate design adequacy for pump run-out.

212.56A The response to Q212.56 is not complete. Provide calculations to support
(6.3) the sizing of the relief valves installed on the RHR and the Core Spray piping to protect against over-pressurization.

212.82 Several plants have used sand-bags or sand-filled tanks as biological
(6.3) shielding inside containment. In the event of a LOCA, these tanks or bags could be damaged and sand would be released. The release of sand inside containment could result in damage to the ECCS pumps. Identify any areas where sand-bags or sand-filled tanks are used for biological shielding. What precautions would be taken to prevent ECCS damage if sand, insulation, paint chips, or similar debris material were released within containment?

212.83 Section 7.3.1.2.2.9 states that ADS safety/relief valve operability
(6.3) will be monitored by a temperature element installed on the valve discharge piping. Operating experience has shown that a "false" temperature increase may be indicated even though the valve has not operated. Justify use of the temperature element over a direct valve position indication to assure safety/relief valve operability. Discuss other instrumentation possibilities (e.g., ΔP or acoustic).

212.24A The FSAR states that each of the safety/relief valves provided for
(5.2.2) automatic depressurization is equipped with an accumulator and check valve arrangement. The response to previous request 212.24 implies that all fifteen safety/relief valves have accumulators. This is not clear from P&I diagram 5.1-3. Please clarify whether or not all safety/relief valves have accumulators.

212.84 (5.2.2) Provide the calculations to support your relief valve discharge coefficients and flow capacities.

212.85 (5.2.2) Provide a discussion of the number and type of operating cycles for which each component (such as valves, solenoids, vacuum breakers, etc.) in the overpressure protection system is designed.

212.86 (5.2.2) Provide the results of the hydraulic calculations that show the Mach number, pressure, and temperature at various locations from upstream of the safety/relief valves to the suppression pool at maximum flow conditions. The concern is related to the potential for the development of damaging shock waves to the discharge piping. Include the effects of suppression pool swell variations during the operation of the safety/relief valves.

212.87 (15B.4.4) The narrative on page 15B.4-24 states that the water level does not reach either the high or low level setpoints. Curve 1 of Figure 15B.4.4-1 indicates the low level trip point is reached at time T=22 seconds. Explain.

212.88 (15B.4.4) (15B.4.5) Identify the units of flow in sector 2 and sector 3 of Figures 15B.4.4-1 and 15B.4.5-1 and in sector 2 and sector 3 of all figures of Appendix 15B.

212.89 (15B.6.5) You state that the quantitative analyses of the spectrum of pipe breaks is covered in Sections 6.2, 6.3, 7.1, 7.3, and 8.3. However, most of the information provided applies only to the DBA line break. Provide a list of the pipe size and break locations that were analyzed for LOCA inside containment.

212.90 (5.2.7) Describe what procedures are available to the control room operator to convert various indicator readouts to a common leakage equivalent, such as gallons per minute, or pounds per hour.

212.91 (5.2.7) Do the primary containment radiation monitors have a built-in check source to permit operability testing? Describe your method of calibrating the containment radiation monitoring system.

212.92 (5.2.7) Describe your method of testing and calibrating the drywell atmospheric cooler condensate monitoring instrumentation.

212.93 (5.2.7) The drywell equipment drain sump collects identified leakage from "hot sources, such as the reactor vessel head flange, valve stem packings, and pump packing glands. This leakage may flash into steam which must be condensed to reach the sump in order to be measured. What assurance is there that the steam will be condensed for leak detection monitoring purposes?

212.94 (5.2.7) Verify that the calibration of the leak detection level sensors is performed during normal plant operation. Note that per SRP 5.2.5, the leakage detection systems should be equipped with provisions to permit calibration and operability tests during plant operation. Also, the testing and calibration should be in compliance with IEEE Standard 279-1971. Discuss how you intend to comply with the above requirements.

212.95 (5.5.6) Is the RCIC turbine speed control system a safety grade design (i.e., Seismic Category I)?

212.96 (5.5.6) Show that the preoperational initial startup test programs for the RCIC system meet the intent of Regulatory Guide 1.68.

212.97 (5.5.6) Describe the design features and operating procedures that preclude water hammer effects at the pump discharge of the RCIC system.

212.98 (5.5.6) In the steam condensing operation of the RCIC system, turbine speed control is transferred from the flow control mode to the heat exchanger level control mode. Is this transfer automatic or manual?

212.99 (15B.0) Provide a realistic range and a permitted operating band for the exposure dependent parameters in Tables 4.4-1 and 15B.0-1. In Table 15B.0-1, provide assurance that values of parameters selected yield the most conservative results.

212.100 (15B.0) The correct value of APRM neutron flux scram setpoint to be used in transient analyses is not clear. The value indicated as input for transient analysis in Table 15B.0-1 is 122.4% NBR. However, a value of 120% NBR is indicated in Tables 7.2-1 and 7.6-10. Explain this discrepancy. For the correct value of setpoint used in transient analyses, provide a breakdown of any uncertainty allowances that are added to the nominal value.

212.101 (15B.0) For each transient and accident analyzed in Section 15B.0 include sections on the following items:

a) Identification of Operator Actions
b) Effect of Single Failures or Operator Errors

Specify the transients and accidents in Section 15B.0 for which operator action is required in order to mitigate the event consequences. In those cases, provide justification for any corrective actions by the operator after transient initiation.

212.64A (15B.0) In Question 212.64 and its response, we noted that some analyzed transients and accidents take credit for nonsafety-grade systems or components. The main concern is that these events are using less reliable systems to show that the acceptance criteria are met. If only safety-grade systems or components were used, analysis results may show that acceptance criteria are not met.

a) Identify each normally operating system for which credit has been taken for each transient and accident analyzed in Section 15B.0.
b) Provide a table of the nonsafety-grade systems and components assumed to mitigate consequences for each transient and accident. For each system or component, reference or provide a description of its design basis and features.
c) Modify NSCA drawings in Appendix F.6 to include nonsafety-grade systems or components which mitigate consequences of transients and accidents.
d) Provide the Δ(ΔCPR) and Δ(Δ peak vessel pressure) for each transient and accident that takes credit for specific nonsafety-grade systems or components that would result if only safety-grade systems or components were assumed in the analysis.

212.102 (15B.0) In the analysis of transients and accidents in Section 15B.0, there is no description of the RPS time response delays used in the REDY analytical model (NEDO-10802). For each transient and accident in Section 15B, specify whether the sensor or overall delay time is used in the analysis and why the specified delay time is conservative.

212.103 (15B.0) Confirm the following items for all transients and accidents in Section 15B.0 which require control rod insertion to prevent or lessen plant damage:

a) All calculations were performed with the conservative scram reactivity curve No. 2 in Figure 15B.0-2.
b) The slowest allowable scram insertion speed was used.

212.104 (15B.0) For transients and accidents analyzed in Section 15B.0, credit was taken for safety/relief valve (SRV) actuation in the safety mode, based on the response to Q212.20. For the pilot-operated Target Rock safety/relief valves used in the Enrico Fermi 2 design, credit for only 3/4 of the rated capacity (11 or less valves) is allowed for safety mode actuation in accordance with ASME B & PV Code Section III, NB-7000. Credit for more than 3/4 of rated capacity is assumed for some transients in Table 15B.0-2.

a) Re-analyze all transients and accidents in Section 15B.0 that take credit for more than 3/4 of rated SRV capacity taking credit for only 3/4 of the rated capacity.
b) Explain why SRV actuation in the safety mode is indicated as relief valve flow instead of safety valve flow in the figures associated with each transient and accident.

212.105 (15B.0) Discuss how the pre-operational and startup tests will be used to confirm flow parameters used in Section 15B.0 analyses. Provide details of any previous test of components in test facilities conducted to show satisfactory performance of the recirculation and feedwater flow control systems and respective pumps. Describe how this information was used in Section 15B.0 analyses.

212.106 (15B.0) On page 4-7 of NEDC-10802, it is stated that the difference in trend of flow coastdown versus initial power between the analytical and experimental coastdown curves for Dresden Unit No. 2 (a BWR/3) in Figure 4-11 was due in part to differences between actual and computed jet pump efficiencies.

a) How has this effect been treated in the analysis of Enrico Fermi 2 transients involving flow coastdown with two recirculation pump trips?

b) Explain why this treatment is applicable to Enrico Fermi 2 which is a BWR/4.

212.107 (15B.0) Table 15B.0-1 does not contain all of the input parameters used in the REDY computer code. For each transient and accident analyzed in Section 15B.0, provide the following:

a) A list of all input parameters.

b) Justification that the input parameters are conservative.

212.108 (15B.1.2.3.2) The assumed feedwater flow controller failure at 120% flow appears low compared to a failure value of up to 146% flow used in other FSARs. Explain the basis for the assumed feedwater flow controller failure at 120% flow.

212.109 (15B.1.3.3.3) Safety/relief valve (SRV) actuation for this transient is not included in Tables 15B.0-2 and 15B.1.3-1 and Section 15B.1.3.5 of the FSAR for decay heat removal. However, it is included in Figure 15B.1.3-1. Explain this discrepancy.

212.110 (15B.1.3.3.3) The steam flow increase of 130% in Section 15B.1.3.3.2 is not simulated for the "pressure regulator failure—open to 130%" transient at time = 0 in Figure 15B.1.3-1. In addition, a pressure increase is indicated during the initial portion of the transient when a pressure decrease should first occur. Please explain.

212.111 (15B.1.3.3.2) With regard to the "pressure regulator failure open to 130%" transient:

a) Specify the assumed operating mode (manual or automatic) of the recirculation flow control system and provide justification that the most conservative results on core thermal margins are obtained with the assumed operating mode.

b) Explain the discrepancy between the high flux peaks shown in Figure 15B.1.3-1 and the small value on Table 15B.0-2.

212.112 (15B.1.3.2.1) The sequence of events in Table 15B.1.3-1 is not present in sufficient detail to follow the process variations in Figure 15B.1.3-1. For example, neither safety/relief valve actuation nor recirculation pump runback, nor initial core cooling are included. Provide a more detailed Table 15B.1.3-1 between 0 and 40 seconds.

212.43A (15B.1.1.1.1) The response to Q212.43 is not acceptable. Provide a quantitative analysis for operation with partial feedwater heating.

212.42A (15B.2.2.3.2) The response to Q212.42 is not acceptable. Provide the following information:

a) Confirm quantitatively that evaluation of this transient at lower power with full stroke turbine control valve (TCV) closure does not result in a more restrictive $\Delta$MCPR and peak vessel pressure.

b) In Section 10.2, a TCV full arc closure time of 0.18 to 0.22 seconds is indicated. The maximum full arc closure time should not exceed 0.18 seconds for transient analysis. Explain why the previous analysis used a conservative 0.15 seconds for full stroke closure and the current analysis uses a non-conservative value of 0.20 seconds. Provide an analysis with a full stroke TCV closure time of 0.18 seconds.

c) Confirm quantitatively that evaluation of this transient at full power with partially open TCVs does not result in a more restrictive $\Delta$MCPR and peak vessel pressure.

212.113 (15B.2.3.3.2) For the "turbine trip" and "MSIV closure" transients, a full stroke closure time of 0.20 seconds was used for the turbine stop valves (TSV) in Section 15B.2.3. There is no description of an acceptable TSV full stroke closure time span in Section 10.2. Similar information was presented for the turbine control valves. Provide the following information:

a) The acceptable full stroke closure time span for the TSV.

b) Verification that the TSV full stroke closure time used for transients in Section 15B.2.3 and 15B.2.5 was the minimum value of the time span or less. If not, re-analyze these transients with a TSV full stroke closure time less than or equal to the minimum value of the acceptable time span.

212.114 (15B.2.3.3.3) Provide the following information regarding the "turbine trip with partial steam bypass failure" transient:

a) The MCPR for the "turbine trip with partial steam bypass failure" transient is specified as 1.14. However, this does not agree with a value of 1.18 in Table 15B.0-2. Explain this discrepancy.

b) Provide analytical results (figures) for the "turbine trip with steam bypass" transient. Revise Table 15B.2.3-1 to include the same level of detail as included in Table 15B.2.3-2.

212.115 (15B.2.3.3.3.2) During the "turbine trip with partial steam bypass" transient, explain why feedwater flow in Figure 15B.2.3-1 decreases to zero prior to the L8 vessel level setpoint trip of the feedwater pumps at 31.7 seconds in Table 15B.2.3-2.

212.116 (15B.2.5.2.1) Include the time at which the following occur in Table 15B.2.5-2

a) Turbine stop valves are closed.

b) Feedwater pumps are tripped.

212.117 (15B 2.7 .2.2) It is indicated that credit is taken for safety/relief valve operation with "low setpoints" to remove decay heat since bypass valves become ineffective with MSIV isolation. Specify the values of these "low setpoints" and provide justification if they are lower than the values in Table 15B.0-1.

212.118 (15.3.1 .2.1.1) For the "trip of one recirculation pump" transient, the text indicates that no scram occurs and core flow and power level should stabilize at a new equilibrium condition. In Table 15B.3.1-1 and Figure 15B.3.1-1, a scram is indicated via an L8 initiated turbine trip. Correct this discrepancy and update Table 15B.3.1-1 and Figure 15B.3.1-1 accordingly.

212.119 (15.3.1 .2.1.2) For the "trip of both recirculation pumps" transient, an L8 initiated turbine trip scram occurs. Typically, this results in the pressure relief, reactor vessel isolation, and initial core cooling safety actions.

a) Explain why these safety actions do not occur for the Enrico Fermi 2 design.

b) Since the scram safety action occurs for this transient, provide an NSOA figure for event 27 in Appendix F.6. Include the safety actions in step a) above if they occur and update the text description for event 27.

212.120 (15B.3.3 .2.2) For the "recirculation pump seizure" accident, coincident loss of offsite power is not simulated with the assumed turbine trip and coastdown of the undamaged pump. The current analysis also takes credit for nonsafety-grade equipment (L8 trip) to terminate the event. Reanalyze this transient assuming coincident loss of offsite power and the use of only safety-grade equipment (see Q212.64A).

212.121 (5.2.7) Regulatory Guide 1.45 recommends that "The leakage detection systems should be capable of performing their functions following seismic events that do not require plant shutdown. The airborne particulate radioactivity monitoring system should remain functional when subjected to the safe shutdown earthquake." It is not clear from the FSAR whether the above criteria are met or not. Discuss the seismic capabilities of the leakage detection system components.

212.122 (5.5.7) On Figure 5.5-16, sheet 6 of 6 (Amendment 3), the following valve positions are stated for RCIC operation in the steam condensing mode:

Valves F051, F052, F053, F026, F607 opened
Valves F003 and F047 closed

However, the figure itself shows:
Valve F003 open. Valves F047 and F607 are not indicated on the figure. Further, the positioning of Valve F606 is not addressed.

Correct the discrepancies.

212.123 Amend the FSAR as indicated below.

1. Figure 6.3-3 sheet 3 shows a pressure switch on the discharge of each of
(6.3) the core spray pumps. These pressure switches are used as a permissive in the ADS logic. Sheets 4, 5, and 6 of Figure 6.3-3 show only one pressure switch per loop. The pressure switch identification numbers also changed in going from sheet 3 to sheets 4, 5, and 6. Clarify.

2. Table 6.3-12 states that the low water level (L2) initiation signal for
(6.3) HPCI activation is set at 10.26 feet above the top of the active core. Figure 5.1-3 states that HPCI initiation occurs at water level L2 (10.75 feet above the top of the active core). Resolve this inconsistency.

3. The following errors were found in the FSAR and should be corrected:
(6.3)

a) Normalized core average flow should be a dimensionless quantity; Figure 6.3-12 specifies psia units for the normalized core average flow.

b) Section 6.3.3.7.3 of the FSAR incorrectly specifies that the maximum permitted peak cladding temperature is 220°F rather than 2200°F.

c) Section 6.3.2.4 of the FSAR identifies Dresser as the safety/relief valve supplier. Fermi safety/relief valves are Target Rock two-stage valves.

4. Correct the subsection references on page 15.1.6-1. Inadvertent HPCI
(15.1) start is covered in Subsection 15.B.5.1 (not 15.B.4.2). Startup of an idle recirculation pump is covered in Subsection 15.B.4.4 (not 15.B.5.1).

5. Correct the following inconsistency:
(15.1)

On page 15.1-6 you state that, "Maintaining a MCPR greater than 1.07 is a sufficient, but not a necessary condition to ensure that no fuel damage occurs."

On page 15B.0-7 you state that, "Maintaining a MCPR greater than 1.06 is a sufficient, but not a necessary condition to ensure that no fuel damage occurs."

6. Add the initial core cooling safety actions to Tables 15B.2.5-2 and
(15B.2.5.2) 15B.2.4-1 to be consistent with Figures F.6-9 and F.6-7, respectively.

7. Correct the following:
(15B.2.6.2)

a) Add the reactor vessel and containment isolation safety actions to Figure F.6-15 for the "loss of all grid connections."

b) Add the following items to Table 15B.2.6-1 to be consistent with the proposed revision to Figure F.6-15 for the "loss of all grid connections" transient:

1) Restoration of AC power

2) Extended core cooling

3) Containment isolation