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U.S. NUCLEAR REGULATORY COMMISSION

June 4, 1981
EF2 - 53454

Mr. L. L. Kintner
U. S. Nuclear Regulatory Commission
Division of Project Management
Office of Nuclear Reactor Regulation
7920 Norfolk Avenue
Bethesda, Maryland 20014

Dear Mr. Kintner:

Reference: Enrico Fermi Atomic Power Plant, Unit 2
NRC Docket No. 50-341

Subject: Responses to NRC Questions and Requirements

Please find enclosed Detroit Edison's responses to several NRC requests. These items will be included in a forthcoming FSAR amendment.

Item I

CPB Q241.5 Seismic and LOCA Loads in Fuel

Detroit Edison's response to this question is enclosed as Attachment 1.

Item 2

RSB H.II.B.2.4,2 CS Definition

Detroit Edison will voluntarily amend page H.II.B.2-3 of the FSAR to define the Core Spray System as a low-pressure system, as requested at the April 23, 1981 NRC/Edison meeting in Bethesda. Refer to Attachment 2.

Item 3

RSB FSAR5.4.6, 5.4.7 Valve Categorizatior

Detroit Edison's response to this item is enclosed as Attachment 3.

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Item 4

RSB	FSAR 6.3	HPCI Pump Reliability

Detroit Edison's response to this item is enclosed as Attachment 4.

Item 5

PSB	BTP-PSB1	Degraded Grid Voltage

Detroit Edison's response to this Branch Technical Position is enclosed as Attachment 5.

Item 6

RSB	FSAR 5.2.2	S/R Valve Maintenance

Detroit Edison's response to this item is enclosed as Attachment 6.

Item 7

RSB	FSAR 15	Recirculation Pump Coastdown

Detroit Edison's response to this item is enclosed as Attachment 7.

Item 8

RSB	FSAR 6.3	Assurance of Filled ECCS Lines

Detroit Edison's response to this item is enclosed as Attachment 8.

Item 9

RSB	Draft Position 5/27/81	ECCS Pumps NPSH

Detroit Edison's response to this draft RSB Position is enclosed as Attachment 9.

Item 10

ICSB	II.K.3.21	CS/LPCI Modifications

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Detroit Edison was requested to amend page H.II.K.3.21-1 to state that HPCI will automatically restart, following manual termination by the operator, should low water level again be reached. Attachment 10 includes this proposed change.

Sincerely,

William F. Colbert

William F. Colbert
Technical Director
Enrico Fermi 2

RMB/kw

Attachment

bcc: R. M. Berg
F. E. Gregor
J. W. Honkala
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A. E. Wegele

Document Control

ATTACHMENT 1
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RESPONSE TO QUESTION NO. 241.5 (CPB)

In response to question 241.-5 (Core Performance Branch), transmitted via NRC letter dated February 18, 1981 (R. L. Tedesco to W. H. Jens), requesting documentation of the combined seismic and LOCA load analysis for the fuel assembly, the following is provided:

The fuel assembly was analyzed for the seismic and LOCA loads, including the Annulus Pressurization Loads and the results were documented via FSAR Amendment 29 in April, 1980. The results are presented in FSAR Table 3.9-40, showing a significant safety margin of a calculated maximum acceleration of 1.04 g versus the allowable acceleration of 3.1[illegible] g. The method of analysis and load combinations are discussed in FSAR Section 3.9.1.5.6.

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Section 6.4 of NUREG 75/087 (Reference 2) provided the guideline for modeling the SGTS effluent plume.

(a low-pressure system)

H.II.B.2.4.2 Radioactive Systems

The systems assumed to contain radioactive liquids include the high-pressure injection system, the core spray system, the reactor core isolation coolant system, and the residual heat removal system, as well as portions of the control rod hydraulic system, sample lines, and all piping and equipment in communication with the primary coolant system out to the second isolation valve. A design review has been performed to ensure that no systems other than those mentioned above would become contaminated with post-accident primary coolant. In particular, design corrections have been made to ensure that the reactor building sumps (which could contain postaccident primary coolant) would not be pumped out of the reactor building. The radwaste system, therefore, would not be contaminated by postaccident sources.

Systems determined to contain postaccident primary containment atmosphere are the drywell, the torus free air volume, the hydrogen recombiner system, all piping and equipment connected to the drywell, and torus free air volume out to the second isolation valve. The reactor building atmosphere is assumed to be contaminated as a result of primary containment leakage. Steam lines are assumed to contain the core release fractions for airborne sources outlined in Subsection H.II.B.2.4.1. It is assumed that these sources are restricted to the vapor-containing areas of the primary coolant system. A design review shows that the gaseous radwaste system is not exposed to postaccident source terms.

H.II.B.2.4.3 Radiation Environment

The determination of the total radiation environment at any location includes the consideration of all of the many potentially contributing sources. The sources considered include the following:

a. Direct radiation shine from the airborne and liquid radiation sources in the drywell and torus

b. Direct radiation shine from essential safety feature (ESF) equipment and piping circulating postaccident contaminated liquids or gases in the reactor building (e.g., RHR, HPCI, RCIC, CSS, and hydrogen recombiners)

c. Immersion in and inhalation of the airborne sources within the reactor building HVAC boundary, resulting in gamma whole-body doses, beta skin doses, and thyroid doses due to iodine inhalation.

d. Direct radiation shine of the reactor building and refueling floor atmospheres to surrounding areas

RSB Draft Question of 4/23/81

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QUESTION:

8. <u>Valve Categorization (5.4.6, 5.4.7)</u>

   - We require that motor operated valves which isolate the residual heat removal system from the reactor coolant system, or the RCIC system from the reactor coolant system be classified Category A in accordance with Section XI of the ASME code. Check valves performing this function are to be classified A/C.

RESPONSE:

Table 1 (attached) contains the list of valves performing an isolation function between high pressure and low pressure portions of systems connected to the reactor coolant system. These valves will be incorporated into the ASME Section XI Pump and Valve Testing Program and categorized as Type A or Type AC.

These valves shall not be routinely exercised every three months during plant operation as required by IWV-3410 because:

1. Such tests would remove one of the two barriers protecting the low pressure portion of emergency core cooling systems.

2. The operators on testable check valves cannot overcome the force on the valve with reactor pressure on one side.

The testing program for these valves will be:

EF-4 Exercise valve and verify valve position during refueling and after valve maintenance prior to return to service in accordance with IWV-3300 or IWV-3522-(b)

EF-2 Exercise valve (full stroke) for operability during cold shutdown mode as time permits but not more frequently than once every three months.

ET Measure the full-stroke time in conformance with IWV-3410. (Not for check valves.)

SLT-4 Seat leak test the valve prior to reaching power operation following refueling and after valve maintenance prior to return to service.

M L Batch

M. L. Batch

4/29/81

TABLE 1

List of Pressure Isolation Valves

SYSTEM	P&ID	VALVE NUMBERS	TYPE	SIZE	FUNCTION
RHR	6M721-2083 6M721-2084	E11-F015A, B	Gate	24	Discharge to Recirc. System
		F050A, B	Check	24	Discharge to Recirc. System
		F023	Globe	6	Discharge to Head Spray
		F022	Gate	6	Discharge to Head Spray
		F008	Gate	20	Suction from Recirc. System
		F009	Gate	20	Suction from Recirc. System
		F608	Gate	20	Suction from Recirc. System
CS	6M721-2034	E21-F005A, B	Gate	12	Discharge to Core Spray Sparger
		-F006A, B	Check	12	Discharge to Core Spray Sparger
HPCI	6M721-2035	E41-F006	Gate	14	Discharge to FW Line
		E41-F005	Check	14	Pump Discharge
RCIC	6M721-2044	E51-(V8-2229)	Check	6	Pump Discharge
		E51-F013	Gate	6	Discharge to FW Line

ATTACHMENT 4
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SAVANNAH RIVER QUESTION NO. 1 (6.3)

The applicant must provide data to demonstrate HPCI pump reliability.

SPECIFIC CONCERN: Basis for expected operating time of 500 hours in Detroit Edison response to Q.212.67A.

RESPONSE:

The Figure of "500 hours" for HPCI Pump was derived from the following:

Reference HPCI Pump Purchase Specification Data Sheet 21A9243AR, Rev. 4, Para. 4.3.1, "The unit will be tested once a month and may undergo several real starts during its 40-year lifetime." Surveillance tests are performed once a month, times 12 months, times 40 years equals 480 tests, plus a possible 20 real starts equals 500 operating hours.

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EF2 FSAR

BRANCH TECHNICAL POSITION PSB1 - Adequacy of Station Electrical Distribution System Voltage

RESPONSE

Fermi 2 has committed to install a second level of undervoltage relaying that addresses the concerns of the subject Branch Technical Position. Specific features of the design are outlined below.

1. The undervoltage relays are set in accordance with design calculations to preclude damage to Class 1E equipment. A time delay setting was chosen to avoid operation of the relay for motor starting conditions.

2. Alarm relaying has been provided to alert the operators that a low voltage condition exists. The setpoint of the alarm relay is above that of the degraded grid trip setting. This was done to give the operators advanced indication of system degradation. It also eliminates any possibility that setpoint drift would permit the trip function to be actuated ahead of an alarm. It does not in any way affect the time delay of the degraded grid relaying.

3. The time delay for actuation of the degraded grid undervoltage relay has been selected to be as short as possible without encountering spurious trips due to motor starting.

4. The degraded grid voltage protection system at Fermi 2 meets all applicable requirements of IEEE Standard 279-1971 "Criteria for Protection Systems for Nuclear Power Generating Stations," as outlined in BTP PSB1.

5. Upon loss of offsite power, the emergency diesel generators start and, upon achieving synchronous speed, the automatic sequencer begins to add loads as required. If a safety injection actuation signal is received, the sequencer will automatically shed all loads from the emergency diesel generators. The sequencer will then begin adding engineered safety feature equipment as needed to mitigate the consequences of the accident. The degraded grid relaying is not designed to operate during sequencer operation. (Refer also to Question 222.33 of the Fermi 2 FSAR, Appendix E).

6. The Class IE buses have been analyzed for all anticipated operating situations. Refer to Chapter 8 of the Fermi 2 FSAR for a description of the Class IE Distribution System.

7. Measurements will be made prior to full power operation to verify the Class IE buses analysis techniques employed.

ATTACHMENT 6
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SAVANNAH RIVER QUESTION NO. 4

ADDITIONAL QUESTION:

It was stated that up to 60 cycles are allowed between overhauls. Can GE verify this?

RESPONSE:

The 60 cycle test described under "Life Cycle Test" in response to Q.212.174 was to verify the ability of the design to meet its performance objectives. This test was not a life duration test to establish period between the overhauls

Response to 212.85, last sentence, commits to a 2 year maintenance period.

Based on operating experience, about 45 SRV actuations per average year are predicted for the valve that actuates most frequently. The other SRV's would cycle much less frequently; less than 10 actuations per average year is predicted. These predicted actuations should be compared with the life cycling testing described in response to Q.212.174 when establishing the period between overhauls. The period between overhauls will also be a function of the problems encountered with the SRV's both during operation and during overhauls. A two year period between overhauls appears reasonable, initially.

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ATTACHMENT 7
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SAVANNAH RIVER QUESTION NO.13

A) In the applicant's proposed preoperational test of the recirculation system, the coastdown characteristics of the recirculation pumps apparently would be obtained by tripping the breakers of the drive motors for the motor generator sets. This would result in a slower coastdown which would not represent the ATWS recirculation pump trip in which circuit breakers open the generator field. We require that the preoperational test of recirculation pump coastdown include a test involving the ATWS trip to verify that sufficiently conservative values of pump inertia were used in the analyses of the consequences and mitigation effects of this type of trip. The time delay for recirculation pump trip should also be verified during the test.

B) What is the Level-2 trip for? (For NPSH protection only?)

RESPONSE

A) Detroit Edison Company will perform an RPT pump coastdown test to verify that sufficiently conservative values of pump inertia were used in the analyses where the trip occurs. The time delay for the trip will also be confirmed.

B) The primary function of the low-water level (L2) trip of the recirculation pumps is to provide the NPSH protection for the recirculation pumps.

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ATTACHMENT 8
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Non-TMI Open Item #9

Assurance of Filled ECCS Lines (6.3)

The static head keep full system for the RCIC and HPCI injection lines is not sufficient. We require provisions such as those used to maintain the low pressure ECCS injection lines full to be made for the HPCI and RCIC discharge piping. In addition, the applicant must provide for incorporation into the technical specifications a schedule for periodic high point venting of all the ECCS injection lines and the RCIC injection line.

Respor e

The RCIC and HPCI systems are normally lined up with the pump taking suction from the condensate storage tank. All valves between the storage tank and the first isolation valve on the pump discharge lines are open. This allows communication between the condensate storage tank and discharge line, through the RCIC and HPCI pumps. The water level elevation in the condensate storage tank ranges from 615 feet to a minimum of about 594 feet. Elevations of the first isolation valves in the RCIC and HPCI discharge lines are about 587 feet. No portion of the HPCI and RCIC pump suction or discharge lines is higher in elevation than the minimum water level in the condensate storage tank. Therefore, the elevation head of the condensate storage tank will ensure that the discharge line remains completely filled with water up to the isolation valves.

The discharge lines connect to the bottom of the feedwater line. Therefore, the remainder of the discharge lines will be maintained full by feedwater flow.

Vent and drain connections are incorporated at high and low points in the RCIC and HPCI piping. Before initial start of the RCIC and HPCI system, the discharge lines are checked to make sure that they are filled with water by manually venting the high-point vents to avoid trapped air pockets in these lines.

Detailed filling and venting procedures are included in the Operating Procedure Manual and the preoperational test procedures and checks. Strict administrative procedures for manually venting the discharge lines provide the main basis of assurance that these lines are full of water.

Periodic high-point venting of all the ECCS injection lines and RCIC injection line is already a part of the plant technical specification requirement. According to the plant technical specification, high-point venting of the discharge lines is required once every 31 days to assure that the discharge lines on the ECCS and RCIC are full of water.

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ATTACHMENT 9
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June 3, 1981

NRC QUESTION

Calculations of NPSH available to ECCS pumps in BWRs are normally provided with reference to the pump suction. We are concerned that under certain post accident conditions the potential may exist for damage to ECCS pumps from cavitation because of local flashing in the system suction lines. The potential can result for example from local elevation changes in the piping runs. Calculations of NPSH available at the pump suction may erroneously assume liquid continuity up to the point of pump suction. We require therefore that the applicants provide calculations demonstrating that all points in all safety related suction piping, the NPSH available is adequate to preclude local flashing under the worst postulated conditions.

RESPONSE

The pump suction piping isometrics for the HPCI, LPCI, and Core Spray systems were reviewed for possible local elevation changes. In all cases, the piping is routed so that the pipe elevation decreases from the suction source to the ECCS pumps, without any loops of increased elevation.

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ATTACHMENT 10
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H.II.K.3.21 Restart of Core Spray and LPCI Systems on Low Level

H.II.K.3.21.1 Statement of Concern

Operator action could prevent the core spray and the low-pressure coolant injection (LPCI) systems from functioning when required, resulting in inadequate core cooling.

H.II.K.3.21.2 NRC Position

The core spray and LPCI systems may be stopped by the operator. These systems would not restart automatically on loss of water level if an initiation signal is still present. The core spray and LPCI system logic should be modified so that these systems will restart, if required, to ensure adequate core cooling. Because this design modification affects several core-cooling modes under accident conditions, a preliminary design should be submitted for NRC review and approval before making the actual modification. The modification of system design should be made in accordance with those requirements set forth in Sections 4.12, 4.13, and 4.16 of IEEE 279-1971 with regard to protective function bypasses and completion of protective action once initiated. Refer to NUREG-0660 and NUREG-0737 (References 1 and 2).

H.II.K.3.21.3 Detroit Edison Position

Detroit Edison endorses the BWR Owners' Group position that the current BWR ECCS design, when coupled with rigorous and continuous operating staff training programs, represents the optimum approach to BWR safety. General Electric and the BWR Owners' Group have reviewed the modifications suggested in the NRC position, above. Their review concluded that the current emergency core cooling system (ECCS) design is adequate and that the proposed changes would have a negative impact on the overall safety of the plant. The negative impacts include a significant escalation of control system complexity, restricted operator flexibility when dealing with anticipated events, and reduced system reliability. The conclusion that the current ECCS design is adequate is based on the comprehensive nature of BWR operator training, the emphasis placed in this training on reactor water level control, the Emergency Procedure Guidelines, the relatively long time an operator has to correct errors, and the extent to which low reactor water level conditions are displayed and alarmed in the control room.

As a result of this study by General Electric and the BWR Owners' Group, Detroit Edison has concluded that the above modifications suggested by the NRC should not be included in the Fermi 2 design.

Following manual termination, HPCI will restart automatically on Level 2. | Amend

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H.II.K.3.21.4 References

1. U.S. Nuclear Regulatory Commission, NRC Action Plan Developed as a Result of the TMI-2 Accident, NUREG-0660, May 1980; Revision 1, August 1980.

2. U.S. Nuclear Regulatory Commission, Clarification of TMI Action Plan Requirements, NUREG-0737, October 1980.