Harry Tauber
Vice President
Engineering and Construction

Detroit Edison
2000 Second Avenue
Detroit, Michigan 48226
(313) 237-8000

May 18, 1982
EF2 - 57,885

Mr. L. L. Kintner
U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Division of Licensing
Washington, D. C. 20555

Dear Mr. Kintner:

References: (1) Enrico Fermi Atomic Power Plant, Unit 2
NRC Docket No. 50-341
(2) EF2-54,761 dated August 25, 1981
(3) EF2-54,665 dated September 9, 1981

Subject: Seismic Re-Evaluation of the NSSS Piping

The seismic reassessment of the NSSS piping, as discussed in the referenced letters and the SSER, has essentially been completed. A summary of the analytical results of this reassessment is provided in Attachment I.

The results show that all piping stresses are within ASME Code allowable values and that the loads on pipe mounted equipment are within prescribed limits. However, some snubbers have predicted loads that exceed their rated loads, and three recirculation discharge reactor pressure vessel nozzles have loads that exceed the allowable values.

Detroit Edison is in the process of upgrading these snubbers to accommodate the site-specific seismic loads and expects that the increased stiffness of these supports will result in acceptable loads at the reactor vessel nozzles.

It should be noted that the reactor water cleanup piping inside containment (not required for plant safe shutdown) was also originally analyzed using the center-of-gravity method for seismic response spectra selection. This piping system is currently being analyzed for the final as-built condition using an acceptable response spectra selection method (envelope method).

Sincerely,

C.M. Heidel for H. Tauber

cc: B. Little

SER Section: 3.9.3 SER Page: 3-26

Current SER Discussion

"We have reviewed applicant's description and procedures for the design and mounting of the safety/relief valves for the reactor coolant pressure boundary (Section 5.2.2 of the FSAR). The combination safety/relief valves are made by Dresser Industries. The design employs a spring-actuated pilot for the relief function. Each valve will be removed from service and tested every other refueling outage."

Detroit Edison Comments

Two comments were identified on this passage and they are discussed below.

1) As indicated in FSAR Section 5.2.2.4.1 (per Amendment 12 - June, 1978), GE changed the safety/relief valve supplier from Dresser Industries to Target Rock Corporation. The Target Rock valves are two-stage, pilot-operated safety/relief valves.

2) As indicated in FSAR Section 5.2.2.4.1.3 (per Amendment 12 - June, 1978), the testing interval for the SRVs currently requires fifty percent of the SRVs to be removed from service and tested at each refueling outage. The remaining fifty percent are to be tested during the subsequent refueling outage. [Detroit Edison to NRC letter EF2-65232, dated September 15, 1983 (attached) also indicated this position.]

Harry Tauber
Group Vice President

Detroit Edison
2000 Second Avenue
Detroit, Michigan 48226
(313) 237-8000

September 15, 1983
EF2 - 65,232

Director of Nuclear Reactor Regulation
Attention: Mr. B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Youngblood:

Reference: (1) Enrico Fermi Atomic Power Plant, Unit 2
NRC Docket No. 50-341

(2) Detroit Edison Letter to NRC, EF2 - 53,454, June 4, 1981

Subject: Clarification on SRV Maintenance Interval

The reference (2) Detroit Edison letter to the NRC and section 5.2.2 of the Fermi 2 SER (p. 5-11) discuss a two year maintenance period for Main Steam Safety/Relief Valves (SRV's). Closer scrutiny of this commitment indicates that a clarification is required. The commitment as stated is impractical from a plant operation, maintenance standpoint, and is inconsistent with current industry practices and NRC positions. Accordingly, Detroit Edison intends to do the following in this regard:

1) 50% of the SRV's will be removed from service and tested and serviced at any given refueling outage (nominally 18 months).

2) The remaining 50% will be tested during the subsequent refueling outage.

The testing program is in conformance with Section XI of the ASME Code as stated in section 5.2.2.4.1.3 of the Fermi 2 FSAR. The maintenance performed on the valves is that maintenance or servicing of the valve to correct or prevent abnormal or unsatisfactory SRV operation.

8309200405 PDR

The above position is consistent with current industry practice and is consistent with the NRC Safety Evaluation Report of the BWR Owners Group Response to Item II.K.3.16 of NUREG-0737. The Fermi 2 FSAR will be modified in a forthcoming amendment to remove any ambiguity or confusion. It should also be noted that we are working closely with the BWR Owners Group on the recent hypothesized binding/sticking problem which resulted in a delayed SRV actuation. We will plan to modify our testing and/or maintenance program appropriately, consistent with the group's recommendations.

If you should have any questions, please contact Mr. Larry E. Schuerman on (313) 586-4207.

Sincerely,

Harry Tauber

cc: Mr. P. Byron
Mr. M. D. Lynch

SER Section: 4.5.1 SER Page: 4-20, 21

Current SER Discussion

The last paragraph of page 4-20 discusses an augmented testing program for the control rod drives.

Detroit Edison Comments

This discussion should be modified based upon EF2-68289 dated May 23, 1984.

NRC Chron File

Wayne H. Jens
Vice President
Nuclear Operations

Detroit Edison
2000 Second Avenue
Detroit, Michigan 48226
(313) 586-4150

May 23, 1984
EF2-68,289

Director of Nuclear Reactor Regulation
Attention: Mr. B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Youngblood:

Reference: (1) Fermi-2
NRC Docket No. 50-341

Subject: Request to Modify SER Statement Concerning Collet Retainer Tube

Section 4.5.1 of the Fermi-2 SER discusses the design and inspection requirements for the collet retainer tube in the CRD system. This section states in part "the augmented testing program recommended by the General Electric Company will be carried out." The section also indicates that NRC staff would include the "augmented testing program" as a requirement in Fermi's technical specifications. We believe that the phrase "augmented testing program" refers to Edison's response to question 212.155 (Appendix E.5 of the FSAR) in which Edison described a program consisting of three parts. One of the parts was characterized as "an augmented surveillance and inspection program" which consists of the following actions:

1. Each rod not fully inserted will be tested by inserting one or more notches at least weekly to confirm operability.
2. All CRDs removed for maintenance will have a dye penetrant examination made of the outer surface of the collet retainer tube (CRT). The criteria established by General Electric in Service Information Letter (SIL) 139 will be used to decide rejection. The term CRT refers to a portion of the outer tube, and replacement of a rejected CRT requires a new cylinder, tube, and flange subassembly.

Surveillance 4.1.3.1.5 in the Fermi-2 technical specifications currently includes the requirement to perform the dye penetrant exam.

~~8405300127 PDR~~

Since the other two parts of Fermi-2's response to the CRT issue eliminate unnecessary thermal cycling and provide a source of water with very low oxygen content to the CRD system we have determined that the probability of cracking has been made extremely remote. In view of this and the fact that the dye penetrant examination is not included in the GE Standard Technical Specifications nor any other BWR's technical specifications, we request the staff to revise the SER to eliminate this as a technical specification requirement. It will continue to be a commitment, however, and is included in the Fermi-2 maintenance procedures.

Please coordinate the review of this request with Mr. Don Hoffman of your staff to support the Proof and Review process. Should you have any questions, please contact Mr. Keener Earle (313) 586-4211.

Sincerely,

Wayne H. Jens

cc: Mr. P. Byron
Mr. M. D. Lynch
Mr. D. Hoffman

SER Section: 5.2.2 SER Page: 5-11

Current SER Discussion

"Safety/relief valves are designed for a specific number of actuations between overhauls. By letter dated June 4, 1981, the applicant discussed the expected frequency of SRV actuations and committed to a 2-year maintenance period....The Office of Inspection and Enforcement will verify that this maintenance is included in plant procedures. We find this acceptable."

Detroit Edison Comments

As indicated in FSAR Section 5.2.2.4.1.3 (per Amendment 12 - 1978) and Detroit Edison to NRC letter EF2-65232, dated September 15, 1983, the testing interval for the SRVs currently requires fifty percent of the SRVs to be removed from service and tested at each refueling outage. The other fifty percent are to be tested in the following refueling outage. The valve manufacturer conducted tests (referred to as life cycle tests) which were designed to verify the reliability of SRVs by subjecting the valves to repeated actuations and statistically evaluating its performance. The tests were not destructive tests conducted to determine a specific number of valve actuations that constitute a valve's operable life. Therefore, the current testing program that was discussed above and is designed to provide periodic valve testing, independent of the actual actuation frequency, is deemed acceptable by Edison.

Harry Tauber
Group Vice President

Detroit Edison

2000 Second Avenue
Detroit, Michigan 48226
(313) 237-8000

September 15, 1983
EF2 - 65,232

Director of Nuclear Reactor Regulation
Attention: Mr. B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Youngblood:

Reference: (1) Enrico Fermi Atomic Power Plant, Unit 2
NRC Docket No. 50-341

(2) Detroit Edison Letter to NRC, EF2 - 53,454, June 4, 1981

Subject: Clarification on SRV Maintenance Interval

The reference (2) Detroit Edison letter to the NRC and section 5.2.2 of the Fermi 2 SER (p. 5-11) discuss a two year maintenance period for Main Steam Safety/Relief Valves (SRV's). Closer scrutiny of this commitment indicates that a clarification is required. The commitment as stated is impractical from a plant operation, maintenance standpoint, and is inconsistent with current industry practices and NRC positions. Accordingly, Detroit Edison intends to do the following in this regard:

1) 50% of the SRV's will be removed from service and tested and serviced at any given refueling outage (nominally 18 months).

2) The remaining 50% will be tested during the subsequent refueling outage.

The testing program is in conformance with Section XI of the ASME Code as stated in section 5.2.2.4.1.3 of the Fermi 2 FSAR. The maintenance performed on the valves is that maintenance or servicing of the valve to correct or prevent abnormal or unsatisfactory SRV operation.

8309200165 PDR

The above position is consistent with current industry practice and is consistent with the NRC Safety Evaluation Report of the BWR Owners Group Response to Item II.K.3.16 of NUREG-0737. The Fermi 2 FSAR will be modified in a forthcoming amendment to remove any ambiguity or confusion. It should also be noted that we are working closely with the BWR Owners Group on the recent hypothesized binding/sticking problem which resulted in a delayed SRV actuation. We will plan to modify our testing and/or maintenance program appropriately, consistent with the group's recommendations.

If you should have any questions, please contact Mr. Larry E. Schuerman on (313) 586-4207.

Sincerely,

Harry Tauber

cc: Mr. P. Byron
Mr. M. D. Lynch

SER Section: 5.2.4.1 SER Page: 5-15

Current SER Discussion

"Components (and supports) may be examined to the requirements set forth in subsequent Editions and Addenda of the Code throughout the 1977 Edition, including Summer 1978 Addenda, subject to certain limitations and modifications."

Detroit Edison Comments

The phrase- " throughout the 1977 Edition, including Summer 1978 Addenda" should be deleted. The intent of the sentence is to acknowledge the ability of Edison to comply with later editions of the Code than presently committed to. This compliance option should not be restricted by the identification of any specific editions.

SER Section: 5.2.4.1 SER Page: 5-16

Current SER Discussion

"Although the preservice program has not been performed in its entirety, an access survey was conducted by Southwest Research Institute on Class 1 piping and components to identify welds that either could not be examined ultrasonically or require modification of the examination procedures. Since nine welds could not be examined by ultransonic procedure, an alternate examination of either liquid penetrant or magnetic particle, supplemented by visual examination during hydrostatic testing, will be performed. Of the remaining welds (approximately eight), many may be examined using special procedures, including special calibration, but field verification is required to determine the precise degree of examinability. A list of the welds was provided in the preservice inspection program, including the identification number, required examination, problem restricting strict compliance, and the alternate method of examination."

Detroit Edison Comments

The above discussion is based on a preliminary draft of the Preservice Inspection Program. The Preservice Inspection Program is presently being refined to reflect the as-built configuration of piping systems. A proper description is provided below.

> The Preservice Inspection Program will identify all welds which have access limitations for examination. For all welds which can not be examined ultrasonically, alternate means of examination will be employed (such as radiography, liquid penetrant or magnetic particle, supplemented by visual examination during hydrostatic testing) or a relief request will be prepared and submitted to the NRC for review.

SER Section: 5.2.5 SER Page: 5-18, 19, 20

Current SER Discussion

This section discusses the Fermi 2 systems in place to detect and monitor identified and unidentified leakage.

Detroit Edison Comments

The initial discussion is a narrative that should be revised to read: "Leaks within the drywell can be detected..." versus the current wording of "are detected". The current wording implies that all six monitoring techniques are used at Fermi 2. This conflicts with FSAR Section 5.2.7.1.1 and the SER discussion which follows the narrative which specifically define the Fermi 2 leakage detection system.

The reference to the "drywell floor level monitor" should be revised to reference it as the "drywell floor drain sump level monitor". This revision is required in several instances in this section and is consistent with FSAR Section 5.2.7.1.2.

The following statement should be deleted since the Regulatory Guide does not require these monitors to be used: "Regulatory Guide 1.45 recommended airborne particulate monitors not be used." To accurately describe the Fermi 2 monitoring system, the phrase "along with temperature and particulate radioactivity monitoring" on page 5-19 should be revised to read: "along with temperature and gaseous radiation monitoring." This revision to the leak detection system (i.e., deletion of particulate radiation monitoring capability) was documented in FSAR Section E.5 in response to Item 212.121 [via Amendment 33- March, 1981].

The discussion of the drywell floor drain sump should reflect the fact that only one sump is present and that overflow from the equipment drain sump will be routed to the drywell floor drain sump.

The following sentence from SER page 5-19 should be revised to delete the reference to "humidity measuring devices" since neither the FSAR discussion in Section 5.2.7 or the Fermi 2 design identifies or utilizes this equipment.

> "Pressure, temperature and humidity measuring devices are also used to indicate the existence of leakage." FSAR Section 5.2.7.11 does acknowledge the use of pressure and temperature measuring devices in leakage detection.

The sentence presented below should be deleted since the oxygen and hydrogen monitors are not considered part of the leak detection system. This system is used primarily in post-accident conditions and can be used to verify inert containment.

> "In addition, the use of fully redundant oxygen and hydrogen monitoring systems allows continuous online comparisons."

SER Section: 5.4.1 SER Page: 5-21

Current SER Discussion

The 3rd paragraph states that the RCIC system is capable of delivering rated flow within 30 seconds of initiation.

Detroit Edison Comments

This value has been revised to 50 seconds. See FSAR Section 5.5.6.3, 3rd paragraph.

SER Section: 5.4.1 SER Page: 5-23

Current SER Discussion

The 2nd paragraph states that ambient and differential temperature setpoints will be established for isolation of the RCIC system.

Detroit Edison Comments

The differential temperature isolation has been eliminated because of a history of spurious isolations at other plants. As reflected in Appendix E.5 of the FSAR, Item 212.30, the differential temperature sensor was retained to provide a control room alarm. See also FSAR Section 5.5.6.2.2, Item 9.

SER Section: 5.4.2 SER Page: 5-23

Current SER Discussion

The last sentence of the 5th paragraph should be deleted.

Detroit Edison Comments

Detroit Edison has elected to delete the steam condensing mode of RHR and to remove the associated valves and piping.

SER Section: 5.4.2 SER Page: 5-23, 24

Current SER Discussion

"The two loops also have connections to steam via the high pressure coolant injection system steam line and can discharge condensate to the reactor core isolation cooling system pump suction or to the suppression pool.

"The residual heat removal system operates in five different modes:...
2. Steam condensing"

"Isolation between the reactor coolant system and residual heat removal system is provided by a check valve within containment and closed motor-operated isolation valve outside containment, except for the suction line which draws water from the recirculation line for shutdown cooling."

Detroit Edison Comments

The first two passages reflect hardware and an operating mode which were affiliated with the RHR steam condensing mode of operation. This facet of the Fermi 2 design has been deleted and should not be addressed in the SER.

The third passage should reflect a 1-inch bypass valve that is installed inside containment around the check valve.

SER Section: 5.4.2 SER Page: 5-26

Current SER Discussion

"The Fermi 2 Technical Specifications will require that the low-pressure coolant injection mode operability is verified every 30 days; that every 90 days each pump is shown to start from the control room; and every 18 months that a system functional test is performed without requiring coolant injection into the reactor vessel...."

Detroit Edison Comments

The draft Fermi 2 Technical Specifications, including the most recent draft dated May 8, 1984, indicate in Surveillance Requirements 4.0.5 and 4.5.1 that the operability test is required every 31 days and the pump start test is required every 92 days. We believe that the NRC staff meant "monthly" and "quarterly" surveillance when they wrote the SER. The 31-day and 92-day interpretations are part of the Standard Technical Specification.

SER Section: 6.2.3 SER Page: 6-6

Current SER Discussion

The second paragraph currently reads:

"The standby gas treatment system (SGTS) is an engineered safety feature system which consists of two separate, parallel 100% capacity trains. Each train consists of a moisture separator, a prefilter, an electric heater, a high energy particulate air (HEPA) filter, a deep bed charcoal adsorber, and an exhaust and cooling fan. The SGTS flowrate capability is based upon a secondary containment air volume change rate of once per day, and the maximum expected reactor building inleakage of 3000 scfm while under a partial vacuum of negative one-quarter inch water gauge."

Detroit Edison Comments

The phrase "an electric heater" should be changed to "electric heaters" and "high energy particulate air (HEPA) filter" should be changed to "high efficiency particulate air (HEPA) filter".

The last sentence of the paragraph should be modified to be consistent with the information provided in EF2-68233 dated June 26, 1984. This information will also be incorporated into the FSAR in a future amendment.

Wayne H. Jens
Vice President
Nuclear Operations

Detroit Edison

2000 Second Avenue
Detroit, Michigan 48226
(313) 586-4150

June 26, 1984
EF2-68,233

Director of Nuclear Reactor Regulation
Attention: Mr. B. J. Youngblood, Chief
Licensing Branch No. 1
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Youngblood:

Reference: (1) Fermi 2
NRC Docket No. 50-341

Subject: Secondary Containment Drawdown Time

SER Section 6.2.3 states, in part, that the SGTS will take six minutes to drawdown the secondary containment pressure to minus one-quarter inch of water following a DBA-LOCA with coincidental loss of all off-site power. The six minute drawdown time was based on the response to question 042.28 of FSAR Appendix E.5. The analysis that was performed in the response to question 042.28 assumed a maximum outdoor temperature of 105°F, which was the worst case for the internal environmental profile. However, this is not the most limiting case for drawdown time. As ambient air temperature decreases, more mass leaks into the secondary containment and thus, more mass must be removed to attain design negative pressure.

The secondary containment pressure response analysis has been reperformed using an outdoor temperature of -10°F (Attachment 1). The analysis shows that in order to drawdown the secondary containment in six minutes following a DBA LOCA, the secondary containment would have to be made unreasonably leak tight and the SGTS would have to operate at or beyond its maximum design flowrate of 4000 CFM. However, by operating the SGTS at 3800 CFM, a negative one-quarter inches of water pressure is predicted to be achieved in ten minutes based on the limiting case. At this flow rate, the steady state secondary containment pressure will be well below minus one-quarter inches of water. It should be noted that the drawdown time under non-environmentally extreme conditions will be much less than ten minutes.

~~8407030166~~ PDR

The Fermi 2 SER assumes a secondary containment drawdown time of six minutes (Section 6.2.3). A radiological consequence analysis using this time is discussed in SER Section 15.2.3.1. An extrapolation of this analysis was made to assess the effects of the increase in drawdown time. The post LOCA thyroid dose at the site boundary would be increased approximately thirty percent but would remain well within the guidelines of 10CFR100. This dose is based on very conservative assumptions.

The FSAR is being revised to reflect a maximum ten minute drawdown time to achieve minus one-quarter inch of water pressure in the secondary containment post LOCA. Your expedited review and concurrence is requested. Please coordinate a revision to the SER with our technical specification reviewer to support the proof and review process.

If you have any questions, please contact Mr. O. Keener Earle (313) 586-4211.

Sincerely,

Wayne H Jens

cc: Mr. P. M. Byron
Mr. M. D. Lynch
Mr. D. Hoffman
USNRC, Document Control Desk
Washington, D.C. 20555

Attachment 1 - SECONDARY CONTAIMENT PRESSURIZATION DURING DBA LOCA

The Standby Gas Treatment System (SGTS) provides sufficient flow to maintain the secondary containment pressure at or below -0.25 inches of water, thus ensuring that any airborne radioactive material in the secondary containment is not released to the surrounding atmosphere without passing through the SGTS filters. In the event of a DBA-LOCA, loss of off-site power is assumed; consequently, there is a delay period from the start of the event to the activation of the SGTS and the emergency area coolers.

During the delay period, the secondary containment pressure increases above -0.25 inches of water due to heat generated by emergency equipment and other sources. Upon initiation of the SGTS and emergency area coolers, a short period of time is required to reduce the secondary containment pressure to a negative pressure at or below -0.25 inches of water.

The purpose of this calculation is to generate the secondary containment pressure response during a DBA-LOCA and to determine the period of time where the secondary containment pressure is above -0.25 inches of water.

The method of analysis, assumptions and results are described below.

METHOD OF ANALYSIS AND ASSUMPTIONS

The computer code HVAC (Reference 1) was used to generate the secondary containment pressure response.

All major assumptions are given below:

1. No credit was taken for exfiltration from the secondary containment.

2. Infiltration to the secondary containment was included in the pressure response analysis.

3. No heat transfer was allowed to the outdoor atmosphere.

4. Heat transfer to interior secondary containment walls, floors and ceilings was included.

5. Heat transfer from the torus room to the secondary containment is based on flow through the pressure relieving doors in the corner room basement walls.

6. Only one SGTS filter train is available with a minimum volumetric flow rate of 3800 CFM.

7. Off-site power is lost at the start of the DBA-LOCA event.

8. The activation of the SGTS is delayed by 33 seconds and the activation of the emergency area coolers is delayed by 38 seconds.

9. The RHR pump rooms and the core spray and RCIC pump rooms in the reactor building sub-basement are treated separately from the main secondary containment volume. These rooms have their own emergency coolers to handle emergency equipment and lighting heat loads.

   Because the heat loads and cooling are confined to partially enclosed volumes at the very bottom of the secondary containment, the area coolers will absorb the heat loads within the confines of the corner rooms.

10. The heat loads from the RHR, core spray and RCIC pump rooms will not affect the main secondary containment volume prior to the initiation of the area coolers. The RHR pumps are activated 13 seconds after the start of the DBA-LOCA event. The emergency coolers are activated at 38 seconds. For the heat loads to affect the main volume, the pumps, piping, and subsequently the corner room atmospheres must heat up. After the corner room atmospheres have heated up, the only mode of heat tranfer to the main volume is by natural convection. Considering that natural convection is a rather slow process, no significant heat transfer to the main secondary containment volume from the corner rooms is expected during the 25 seconds from the intiation of the RHR pumps to the initiation of emergency cooling.

11. An outdoor temperature of -10°F was used in the analysis.

Results

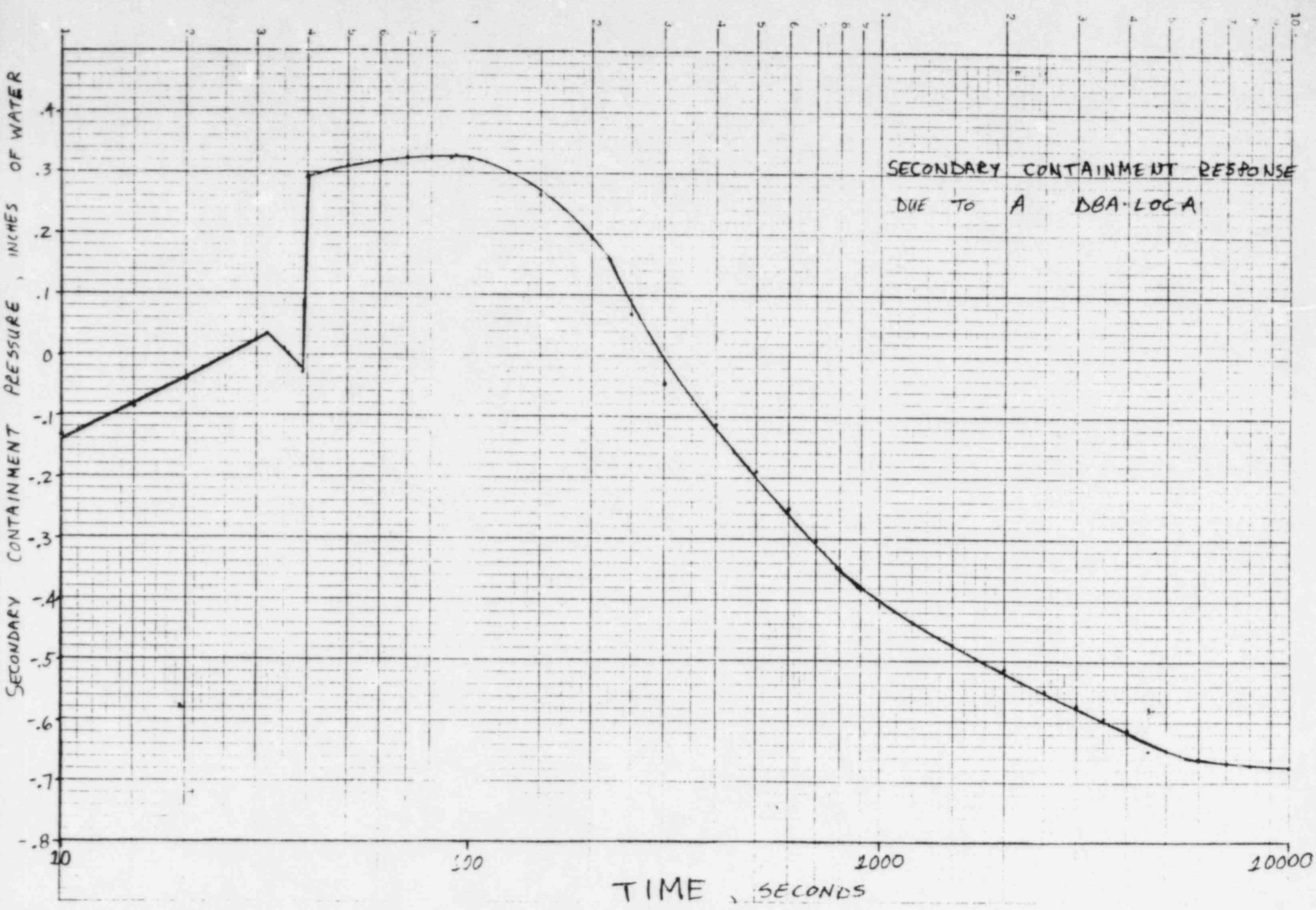
The secondary containment response due to a DBA-LOCA is shown in Figure 1. During the first 33 seconds, the pressure increases to a slightly positive value. With the activation of the SGTS at 33 seconds and the activation of the area coolers at 38 seconds, the pressure decreases to near atmospheric.

At 40 seconds. pressure relieving doors on the common wall between the torus room and the corner rooms open and allow heated torus room air to enter the rest of secondary containment. This step input of heat into the secondary containment appears as a sharp pressure spike on Figure 1.

The pressure then decreases past -0.25 inches of water to a steady state secondary containment pressure. A period of approximately 600 seconds elapses from the start of the DBA-LOCA event to the point where the secondary containment pressure decreases to and subsequently stays below -0.25 inches of water.

References

1. DET-07-035, "HVAC Computer Code for Environmental Response Profiles", Nutech File No. 50.0407.1328, Rev. 0


SECONDARY CONTAINMENT RESPONSE
DUE TO A DBA-LOCA
SECONDARY CONTAINMENT PRESSURE, INCHES OF WATER
.4
.3
.2
.1
0
-.1
-.2
-.3
-.4
-.5
-.6
-.7
-.8
10
100
1000
10000
TIME, SECONDS

SER Section: 6.2.4 SER Page: 6-11

Current SER Discussion

"The purge system meets the requirements of Branch Technical Position CSB 6-4, including the provision to limit purging to less than 90 hours per year. We find this acceptable; however, a confirmatory audit will be performed prior to issuing an operating license. (See Item II.E.4.2, Section 22.2 of this report.) We will include this limit on purging in the plant Technical Specifications."

Detroit Edison Comments

Edison's commitment to limit use of the purge system to less than 90 hours per year while in the start-up, power, hot standby, and hot shutdown modes of operation was made during the review by CSB. The commitment was made since a radiological consequence analysis had not yet been completed demonstrating that accident doses were within 10CFR100 guidelines.

Edison believes the 90 hour limit is no longer justified and requests the SER be amended to delete it. This is needed to permit finalizing the Fermi 2 technical specifications currently scheduled for July.

Edison is presently formulating a letter to the NRC that will provide the background and justification for this position.

SER Section: 6.2.5 SER Page: 6-11

Current SER Discussion

"In Amendment 11, the applicant provided a positive pressure seal type MSIVLCS. The proposed MSIVLCS consists of an inboard and outboard system, only one of which functions at a given time. The inboard system pressurizes the main steam line between the inboard and outboard MSIVs. The outboard system pressurizes the steam lines between the outboard MSIV and the third MSIV."

Detroit Edison Comments

FSAR Section 9A.3.2 (per Amendment 21 - March, 1979) indicates the following:

> "System activation would occur after the reactor pressure falls below approximately 44 psig. Both divisions would automatically maintain a pressure between the MSIVs of two to six psi above the reactor pressure. As the reactor pressure decays further, the air injection pressure would follow while holding the aforementioned constant differential. If there were a failure of an operational division, the failed division would be deactivated."

The SER should, therefore, be revised to reflect the fact that both systems are started initially, with reliance on a single division occurring only when one division is inoperable.

SER Section: 6.2.6 SER Page: 6-13

Detroit Edison Comments

The errata to this page issued with Supplement No. 3 did not clarify the capacity versus initial flow settings of the recombiner. The last sentence of the 1st paragraph should be followed by a new sentence which reads: "The initial recombiner flow settings are 60 scfm inlet flow and 90 scfm recirculation flow based on an inlet gas containing 5% oxygen." This is in addition to the Supplement 3 errata.

SER Section: 6.5.2.1 SER Page: 6-27

Current SER Discussion

The 2nd paragraph states that a removal efficiency of 95% was assumed for organic iodine. Table 15.2 of the SER states that a value of 99% was used.

Detroit Edison Comments

The correct value is 99%.

SER Section: 7 SER Page: Chapter 7

Current SER Discussion

Throughout the SER, and specifically in Chapter 7, statements are made to the effect that a Fermi 2 system is identical to the system installed at another facility. (For example, page 7-5: "The applicant has stated that the reactor trip system is identical to the Hatch 1 design.")

Detroit Edison Comments

As reflected in FSAR Table 7.1-1 (per Attachment 56 - May, 1984), these references should reflect that the systems are functionally identical, but due to inherent plant design, layout and construction difference, are not physically identical.

SER Section: 7.1.2 SER Page: 7-4

Current SER Discussion

"Safety System Setpoints

The applicant has stated that setpoints, sensor ranges, and sensor accuracies will be included in the Technical Specifications. Technical specifications for the setpoints and allowable values are also expected to be made available during the technical specifications review. The resolution of this issue will be provided in the review of the final technical specifications when they are submitted. This is acceptable to the ICSB."

Detroit Edison Comments

The basis for the first sentence in this paragraph is unclear to Detroit Edison. Our records and our review of the FSAR do not indicate that Edison stated that "setpoint, sensor ranges, and sensor accuracies will be included in the Technical Specifications". Nonetheless, if such a statement was made it is no longer considered appropriate. The Fermi 2 Technical Specifications define instrument setpoint values. Sensor ranges and accuracies are not identified in the Technical Specifications, but these factors are included in the selection of setpoint values which appear in the Technical Specifications.

On the basis of the foregoing, Detroit Edison recommends that the first sentence in the paragraph entitled "Safety System Setpoints" be deleted.

SER Section: 7.3.2 SER Page: 7-10

Current SER Discussion

"ADS is interlocked with the core spray (CS) and Residual Heat Removal System (RHR) by means of pressure switches located on the discharge of these pumps. These interlocks are common to the automatic and the manual ADS initiation circuits. However, the independence of the manual and automatic initiation circuits are not compromised because each of the logics are duplicated."

Detroit Edison Comments

The second sentence noted above should be revised to read: "These interlocks are associated with the automatic ADS initiation circuit only."

FSAR Section 7.3.1.2.2 reflects this position. The ADS can be manually initiated at any time regardless of whether or not the automatic initiating signals or interlocks are present.

SER Section: 7.3.2 SER Page: 7-10

Current SER Discussion

"The power supply for the automatic valves in each loop is the same as that used for the corresponding core spray pump in that loop."

Detroit Edison Comments

The power supply for the automatic valves in each loop is supplied from the same division as the corresponding core spray pumps of that loop.

SER Section: 7.3.2 SER Page: 7-11, 12

Current SER Discussion

"The EECW system ensures cooling water to remove heat from emergency equipment upon loss of offsite power or failure of the reactor building closed cooling water (RBCCW) system....Low RBCCW flow automatically isolates the RBCCW and starts one of the EECW loops and places the other EECW loop in standby."

Detroit Edison Comments

FSAR Section 9.2.2.2 (per Amendment 0) indicates the following:

> "Upon loss of offsite power, both divisions of the EECW system are automatically activiated; i.e., pumps start and valves isolate the nonessential portion of the RBCCW system. Upon loss of RBCCW system differential pressure between the supply and return headers, either Division I and/or Division II EECW loops will start automatically, depending on the portion of the RBCCW system affected."

In addition, FSAR Section 7.3.4.2.3 is being revised to reflect this operating mode.

The SER should be revised to reflect this information.

SER Section: 7.4.2 SER Page: 7-14

Current SER Discussion

"Division I logic (A & C sensors) is powered from the 130 volt direct current Bus A and Division II (B & C sensors) from the 130 volt direct current Bus B."

Detroit Edison Comments

In accordance with the discussion of channel independence provided in FSAR Section 7.4.2.2.2.5 (Amendment 0), the above referenced section should be revised to read:

> "Division I logic (A and C sensors) is powered from the 130 volt Division I battery and Division II (B and D sensors) from the 130 volt Division II battery.

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SER Section: 8.2 SER Page: 8-1

Current SER Discussion

"The 120 kilovolt switchyard is also connected to the five peaking units located on the site. The largest of these generators is the 165 megawatt turbine associated with Fermi 1. The Fermi 1 liquid metal fast breeder reactor has been decommissioned and the turbine generator is being supplied with steam from an oil-fired boiler. The remairing four generators are 18.8 megawatt gas turbines. There are two 13.8 kilovolt generator buses for the peaking units."

"One of the 13.8 kilovolt windings is also the feed noted above that supplies power from the 120 kilovolt switchyard to the Division I safety loads and approximately one-half the normal unit loads through 13.8 kilovolt/ 4.16 kilovolt transformers. The other 13.8 kilovolt winding feeds the loads at the Fermi 2 circulating water pumphouse through a 13.8 kilovolt/ 4.16 kilovolt transformer."

Detroit Edison Comments

Main-turbine generator number 1 has been decommissioned. Therefore, this SER section should be revised to reflect only four peaking units onsite. Similarly, the second and third sentences should be deleted to reflect this.

In addition, as reflected in FSAR Section 8.2.1.2 (via Amentment 55 - March, 1984), one 13.8 kV winding is now used as an alternate feed, while the new Transformer no. 1 supplies the loads discussed above. The modification of the 13.8 kV feeds also requires the following phrase from the third paragraph of SER page 8-2 to be deleted:

> "....and the 4.16 kilovolt Fermi 2 buses at the intermediate 13.8 kilovolt level."

Additional description for the Transformer 1 feed to transformers SS64, 66 (CWPH) and 68 (GSWPH) should be reflected in a future SER supplement.

SER Section: 8.2.1 SER Page: 8-4

Current SER Discussion

"The Class 1E bus load shedding scheme should automatically prevent shedding during sequencing of the emergency loads to the bus. The load shedding feature should, however, be reinstated upon completion of the load sequencing action. The technical specifications must include a test requirement to demonstrate the operability of the automatic bypass and reinstatement features at least once per 18 months during shutdown.

In the event an adequate basis can be provided for retaining the load shed feature during the above transient conditions, the setpoint value in the technical specifications for the first level of undervoltage protection (loss of offsite power) must specify a value having maximum and minimum limits. The basis for the setpoints and limits selected must be documented."

Detroit Edison Comments

One load shedding scheme is set to trip on loss of offsite power, while the second load shedding scheme is set to trip when offsite voltage has degraded to the point where safety systems may not operate. The second scheme is provided a short time delay to override motor starting voltage transients that will not adversely affect safety related equipment. Technical Specifications Table 3.3.3.2 provides the setpoint values for the undervoltage load shedding.

Load shedding is by-passed when the Emergency Diesel Generator (EDG) output breaker has closed (i.e., loss of offsite power) and is not reinstated after the load sequencing action is complete. Once the EDG is connected to the bus and load sequencing starts, undervoltage tripping of the loads will not occur. Fermi 2 load shedding in not by-passed during load sequencing action when the associated EDG breaker is open (i.e., offsite power is available).

SER Section: 8.3.1 SER Page: 8-8

Current SER Discussion

"A modular power unit consists of an automatic transfer switch with appropriate sensing devices, three single-phase transformers, and one single-phase voltage regulator."

"The third output supplies regulated power for instrumentation loads, which has an output variation of ±0.5 percent for input variation of +10 percent, -20 percent."

Detroit Edison Comments

FSAR Section 8.3.1.1.9 (via Amendment 55 - March, 1984) reflects a revision which noted that a voltage regulator per division has been added to each modular power unit. This discussion should be revised to reflect the fact that there are now two regulators per modular power unit.

The second paragraph above should also be revised to read: "....output variation of ±1.5 percent for input variations of +10 percent, -10 percent" to be consistent with FSAR Section 8.3.1.1.9.

SER Section: 9.1.1 SER Page: 9-2

Current SER Discussion

The 1st paragraph now reads:

> "....fuel assemblies in an array which will limit the effective multiplication factor to 0.95 in the event that the new fuel area were flooded with water. The outer structure of the rack design precludes the inadvertent placement of a fuel assembly in the rack closer than the design spacing. The new fuel storage racks will be bolted together and fixed to the new fuel storage vault. The new fuel racks and storage vault are designed to seismic Category I requirements."

Detroit Edison Comments

The new fuel storage racks are not bolted together but are individually bolted to the new fuel storage vault. See FSAR Section 9.1.1.3 and Figure 9.1-1 and 9.1-2.

SER Section: 9.1.2 SER Page: 9-2

Current SER Discussion

The 2nd sentence of the last paragraph states that 13 of the acks will each have 169 storage cells on a 13 x 14 array.

Detroit Edison Comments

The storage cells are configured in a 13 x 13 array. See Figure 9.1-4 of the FSAR.

SER Section: 9.14 SER Page: 9-7

The first paragraph states that a provision that the reactor building crane will be inspected and maintained in accordance with the requirements of ANSI Standard B30.2-1967 will be included in the technical specifications.

Detroit Edison Comments

Edison believes that the commitment to inspect and maintain the crane in accordance with ANSI B30.2-1967 is inappropriate for including in the technical specifications although Edison commits to comply with Chapter 2-2 of the ANSI B30.2-1976 and has proceduralized the inspection process.

Because of the means by which technical specifications are implemented, a limiting condition for operation is satisfied by successfully performing the associated surveillance requirements prior to entering the operational condition or other specified applicable condition (Technical Specification 4.0.4). Because of the nature of the requirements in the subject ANSI standard it is difficult to interpret how to apply this in determining operability and many of the provisions are inappropriate to use as a condition for determining operability.

SER Section: 9.1.6 SER Page: 9-10

Current SER Discussion

"Demineralizer resin will be replaced when pool water samples show reduced decontamination effectiveness. To maintain water quality, the demineralizer will also be used when the chloride concentration in the pool water exceeds 0.5 ppm."

Detroit Edison Comments

These statements should be revised as follows to be accurate: "Demineralizer resins will be replaced when demineralizer effluent or differential pressure limits are attained. The chloride concentration in the fuel pool will be maintained at $\leq$ 500 ppb."

SER Section: 9.3.1 SER Page: 9-14

Current SER Discussion

"The compressed air system includes a safety-related seismic Category I control air system and a nonsafety-related station air system. The control air system is located in the auxiliary building and the station air system is in the turbine building.

The control air system consists of two air compressors, air receivers, filters and dryers each having a capacity of 100 standard cubic feet per minute and two distribution systems--noninterruptible and interruptible. Noninterruptible control air is supplied through two separate systems (Division I and Division II) to engineered safety features including the standby gas treatment system, the control center air conditioning system, emergency equipment cooling water system, residual heat removal system, reactor core isolation cooling system, high pressure coolant injection system, the control rod drive system, and the primary containment atmospheric monitoring system. The noninterruptible portion of the control air system is required to mitigate accidents and to effect safe plant shutdown for anticipated operational occurrences. The interruptible portion of control air is used for nonsafety-related purposes.

The station air system consists of three air compressors and associated inlet filters, intercoolers and aftercoolers each having a capacity of 1,225 standard cubic feet per minute. Station air is used for maintenance, operational processes, process instruments and controls. Accumulators are provided to operate safety-related valves supplied by station air in the event the station air system fails.

The control air system is connected to the station air system through intertie valves. If air pressure drops below a set point (80 pounds per square inch gauge) the intertie valves are closed and control air compressors are automatically started. On loss of offsite power, the emergency diesel generators automatically supply power to the compressors."

Detroit Edison Comments

FSAR Section 9.3.1.2 (per Amendment 0) indicates that the compressed air system is comprised of non-safety grade station air and interruptible control air systems and a safety grade noninterruptible control air system. The non- safety grade systems are located in the turbine building, while the safety grade noninterruptible control air system is located in the auxiliary building.

In addition, the description of the components which comprise each of the three systems should be revised to clearly delineate which equipment is tied to each system. The present discussion attributes some components to the interruptible control air system which are used exclusively in the noninterruptible control air system (NIAS). The SER should be revised to reflect the contents of FSAR Section 9.3.1.2.

SER Section: 9.3.1 SER Page: 9-14

Detroit Edison Comments (cont'd)

Station air is not used for process instruments and controls. The interruptible air system will use dried and filtered station air to feed process instruments and controls. Accumulators for safety-related valves are predominately located inside the drywell and are fed from the nitrogen system, with backup provided by NIAS.

As indicated in the FSAR (via Amendment 48 - May, 1983), the intertie valves close at 75 psig (not 80 psig).

The discussion in the fourth paragraph should be clarifed to indicate it pertains solely to the NIAS.

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SER Section: 9.3.2 SER Page: 9-15

Current SER Discussion

"The drain collection system for reactor coolant components include a 500 gallon sump in the drywell with two 50-gallon-per-minute transfer pumps. The pump water is cooled by recirculation through a heat exchanger."

"The turbine building has three 200 gallon sumps each provided with a 200-gallon-per-minute emergency pump to transfer water rapidly to the waste holding pond in the event of a fire."

Detroit Edison Comments

FSAR Section 9.3.3.2 (per Amendment 53 - February, 1984) states that the equipment drain collection system for primary containment components terminates in a 1000 gallon (as opposed to a 500 gallon) sump.

Similarly, Section 9.3.3.2 indicated that the turbine building has seven separate radioactive drain collection systems. The FSAR discussion of the turbine building sumps noted above has been revised to read:

> "Finally, three sumps, with nominal total capacities of 300, 3000, and 4000 gallons, are provided to collect oil-contaminated liquids. These sumps are each provided with twin 50-gpm or 60-gpm pumps as well as a 200 gpm or a 250 gpm emergency pump. The discharge is normally routed to an oil separator prior to treatment in the radwaste building. The emergency pumps empty the sump rapidly, in case of fire, to the liquid waste holding pond."

SER Section: 9.5.2 SER Page: 9-19

Current SER Discussion

"The intra-plant public address system is a independent page-party communication system which consists of loud speakers, permanent stations and portable telephone handset locations throughout the plant. The system provides two-way communication for speech at all handset stations. The public address system is powered from the AC emergency system with backup provided by the emergency diesel generators."

Detroit Edison Comments

As discussed in FSAR Section 9.5.2.2.2 (per Amendment 0), the public address system is comprised of loudspeakers and permanent telephone handset locations. Fermi 2 does not use portable telephone handsets on the public address system.

SER Section: 9.5.3 SER Page: 9-21

Current SER Discussion

"The emergency lighting is provided by self contained (including charger) battery powered units capable of seven hours of continuous operation. The panel incandescent lighting for auxiliary power panels, standby core cooling system panels, and supplementary emergency sealed beam units in the main control room are normally powered from the normal and essential AC lighting system. On failure of these systems power is automatically provided by the 130-volt station battery."

Detroit Edison Comments

As indicated in FSAR Section 9.5.3, the emergency lighting units are capable of eight (not seven) hours of continuous operation. In addition, the last sentence from the SER section above should be deleted, since no throw over to the station battery is provided for the control room lighting. Control room lighting is ensured to be reliable by the physical separation and redundancy in the lighting system supplies. The emergency lighting portion of the control room lighting power supplies are reenergized from the EDGs by automatic digital load sequencer action.

SER Section: 9.5.5 SER Page: 9-26

Current SER Discussion

The 5th paragraph now reads:

"Except for the two jacket water system vent lines and the equalizing line to the expansion tank, the diesel engine cooling water system piping and components up to the diesel engine interface, including auxiliary skid mounted piping are designed to Seismic Category I, ASME Section III, Class 3 (Quality Group C) requirements and meet the recommendations of Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam- and Radioactive Waste Containing Components of Nuclear Power Plants", and Regulatory Guide 1.29, "Seismic Design Classifications."

Detroit Edison Comments

There are additional lines which are not designed to Class 3 requirements. FSAR Appendix E.5, Items 222.55 and 222.62 reflect the quality classifications to which the various portions of the diesel generator and its support systems were designed and constructed.

SER Section: 9.5.5 SER Page: 9-26

Current SER Discussion

The 3rd paragraph states that when the engine is idle, the engine coolant is heated to a temperature of 120°F to 130°F.

Detroit Edison Comments

The statement should be revised to state that the coolant is heated to approximately 110°F. FSAR Section 9.5.5.2 is being revised accordingly.

SER Section: 9.5.6 SER Page: 9-28

Current SER Discussion

The first paragraph states that each air starting subsystem has sufficient capacity to provide a minimum of five consecutive cold engine starts.

Detroit Edison Comments

FSAR Section 9.5.6.1 has been modified to indicate that the combined capacity of the two air receivers in each air starting subsystem provides sufficient capability to achieve five cold starts.

SER Section: 10.1 SER Page: 10-1

Current SER Discussion

"A turbine bypass system is provided to discharge directly to the condenser up to 25% of the main steam flow around the turbine during transient conditions. This bypass capacity together with a 40% reactor automatic step load reduction capability is sufficient to withstand a 65% generator load loss without tripping the turbine or causing control rod movement or tripping the reactor."

Detroit Edison Comments

Detroit Edison does not consider the 40% reactor step load capability to be linearly additive to the 25% bypass capacity because of the different response times of these systems. Detroit Edison believes it sufficient and more accurate to say simply that Fermi 2 has a 25% bypass capacity. In SSER 3, the NRC deleted a nearly identical description in SER Section 10.4.2 in response to Detroit Edison's comment number 19, in letter EF2-56767, dated August 2, 1982.

SER Section: 10.2.1 SER Page: 10-1

Current SER Discussion

"Overspeed protection is accomplished by three independent systems; i.e., normal speed governor, electric, overspeed and mechanical backup overspeed control systems. The normal speed governor...."

Detroit Edison Comments

There should be no comma between the words "electric" and "overspeed".

SER Section: 10.2.1 SER Page: 10-2

Current SER Discussion

"In order to protect the turbine-generator, the following signals will shutdown the turbine: . . . . .(10) loss of both speed signals, . . . (12) mechanical trip via manual trip handle at the front standard,..."

Detroit Edison Comments

Item 10 in the discussion above implies two speed controls are present for turbine-generator. As indicated in FSAR Section 10.2.2.4, there are three speed sensing channels, but loss of two of them will trip the emergency trip system.

Item 12 in the above discussion should be deleted since it is not present in the Fermi 2 design. FSAR Section 10.2.2.4 is being modified to similarly delete reference to this mechanism.

SER Section: 10.3.1 SER Page: 10-4

Current SER Discussion

"The main steam isolation valves are designed to provide positive isolation against steam flow associated with a main steam line break. They are pneumatic operated, fast-closing valves. Operating air is supplied to the valves from the station air system and a seismic Category I air accumulator provides backup operating air for each valve."

Detroit Edison Comments

FSAR Section 5.5.5.2 (per Amendment 0) indicates that control air, not station air, and the nitrogen system supply the main steam isolation valves. Nitrogen is supplied to the inboard MSIV, while interruptible control air is supplied to the outboard MSIV.

SER Section: 10.4.3 SER Page: 10-6

Current SER Discussion

The first paragraph of SER Section 10.4.3 indicates that the Circulating Water System flow rate to the main condenser is 180,000 gallons per minute.

Detroit Edison Comments

As indicated in FSAR Section 10.4.1.1.1, the Circulating Water System flow rate is 836,700 gallons per minute.

Other corrections suggested are:

In line six of that paragraph, delete the phrase "of 200,000 gpm". As indicated in FSAR Section 10.4.5.3, rupture of a circulating water line expansion joint would result in forcing water out the resulting gap at an estimated rate of about 200,000 gpm.

In line eight of that paragraph, change the phrase "pressure differential transmitters" to "pressure switches", and at the end of that sentence delete the phrase "in each line". This would more accurately describe the instrumentation described in the FSAR.

In line nine of that paragraph, change the phrase "After such an alarm..." to "On continuing low pressure..." to reflect the actual instrumentation installed.

SER Section: 10.6

SER Page: 10-8

Current SER Discussion

The next to last sentence in the 7th paragraph states "The conductivity is continuously monitored for the system influent, effluent and demineralizer tank effluent".

Detroit Edison Comments

This sentence should be revised to read "The conductivity is continuously monitored for the system influent, effluent and individual demineralizer effluent".

SER Section: 11.2.2 (Table 11-1) SER Page: 11-9

SSER 3 Section: 11 (Table 11-2) SSER 3 Page: 11-6

Current SER Discussion

Table 11-1 in SER Section 11, and Table 11-2, SSER 3, Section 11, list parameters used by NRC for the calculation of releases of gaseous radioactive wastes.

Detroit Edison Comments

Detroit Edison used the residence times identified in FSAR Section 11.3.2.7.3.1 and indicated below:

Air ejector off gas	4.9 minutes
Charcoal delay - Krypton	1 day
Charcoal delay - Xenon	16 days

SER Section: 11.3 SER Page: 11-6

Current SER Discussion

"Scintillation detectors are used for monitoring liquids and radioactive gases in gaseous effluents. Particulates will be collected on replaceable filters which will be routinely monitored. Gaseous iodine will be collected in replaceable, impregnated charcoal adsorbers which will be routinely monitored".

Detroit Edison Comments

As reflected in FSAR Section 11.2 and 11.3, the reference to "scintillation detectors" in the SER should be revised to read: "beta and gamma sensitive detectors".

SER Section: 12.4 SER Page: 12-4

Current SER Discussion

"The Radiation Chemistry Engineer has the responsibility for administering this program."

"The radiation protection program is designed to ensure that: ...(7) personnel access to high radiation areas and maintenance work in radiation areas are controlled by use of a radiation work permit, which must be approved by the Radiation Chemistry Engineer or Supervisor, ..."

"The radiation protection facilities include an access control point, high and low level laboratories, counting room, spectrometer room, calibration room, offices, decontamination and laundry area, and change room."

Detroit Edison Comments

FSAR Section 12.3.1.1 indicates that the General Supervisor, Health Physics, not the Radiation Chemistry Engineer, has the responsibility for adminstering the radiation protection program.

Similarly, FSAR Section 12.3.1.3.L states that radiation work permits are "prepared by and require approval of Health Physics", not the Radiation Chemistry Engineer.

FSAR Section 13.3.2.1.1 indicates that multiple access control points and change areas (as opposed to change rooms) are available as needed.

SER Section: 12.4 SER Page: 12-5

Current SER Discussion

"Additional whole-body counting will be provided on a random basis during high maintenance activity and if a person shows evidence of surface contamination or is exposed to concentrations in excess of 10 CFR Part 20 values."

Detroit Edison Comments

Additional whole-body counting will be provided in instances of significant skin contamination. Minor surface contamination will not necessarily mandate a whole-body count as is implied by the current SER wording.

SER Section: 13.1 SER Page: Section 13

Current SER Discussion

Conduct of Operations - Organizational Structure of Applicant

Detroit Edison Comments

The text of the SER reflects the NRC's perception of the organization and staffing in place when the SER was written (July 1981). In the intervening three years the organization has evolved and personnel have been promoted. We believe that these changes will enhance the safe operation of Fermi 2 and so do not invalidate the NRC's earlier conclusions. Some of these changes include:

a) The position previously called "Supervisor, Engineering Assurance" is now called "Principal Engineer, Engineering Quality Assurance" (SER page 13-3, second paragraph).

b) The current Manager-Nuclear Operations has experience in nuclear startup, operation and construction of fossil-fired power plants (SER page 13-3, third paragraph).

c) The position previously called "Health Physics Monitor" is now called "Health Physics Technician" (SER page 13-4, first paragraph).

d) Detroit Edison now projects Nuclear Operations staffing at 664 persons for 1984 and 670 for 1985, rather than 490 persons given in Table 13.1-1 of the SER.

e) The current Superintendent-Nuclear Production has experience in nuclear startup and extensive commercial fossil experience (SER page 13-6, third paragraph).

f) The position previously called "Training Superintendent" is now called "Director Nuclear Training" (SER page 13-10, Section 13.2.1).

g) There have been minor changes to the organizations depicted in SER Figures 13.1-1 through 13.1-7.

h) The organization referred to as the "Independent Review and Audit Group (IRAG)" in the SER should be revised to read "Nuclear Safety Review Group". The discussion of this group should be revised to be consistent with FSAR Section 13.4.3.2.

i) Reference to Procedure 12.000.05 (SER page 13-14) is incorrect and should be deleted. Edison recommends not referencing a procedure, but if a reference is deemed necessary, the proper reference is Nuclear Engineeirng Procedure NE 1.4.

SER Section: 14 SER Page: 14-1

Current SER Discussion

(Paraphrase) This section indicated that the staff reviewed the Fermi 2 test program for conformance with applicable Regulatory Guides, including:

Regulatory Guide 1.20 (Revision 2, October 1975)

Regulatory Guide 1.41 (March 1973)

Regulatory Guide 1.52 (June 1973)

Regulatory Guide 1.56 (June 1973)

Regulatory Guide 1.68 (November 1973)

Regulatory Guide 1.68.1 (January 1977)

Regulatory Guide 1.68.2 (Revision 1, July 1978)

Regulatory Guide 1.80 (June 1974)

Regulatory Guide 1.108 (Revision 1, August 1977)

Detroit Edison Comments

As indicated in Appendix A of the Fermi 2 FSAR, Detroit Edison has committed to the Regulatory Guides cited by NRC except in three cases in which Edison has committed to more recent revisions. These three cases are:

Regulatory Guide 1.20 (Revision 2, May 1976)

Regulatory Guide 1.52 (Revision 2, March 1978) (with exceptions)

Regulatory Guide 1.56 (Revision 1, July 1978)

SER Section: 15.1 SER Page: 15-5

Current SER Discussion

The third paragraph on page 15-5 indicates that the Technical Specifications will not allow operation with partial feedwater heating.

Detroit Edison Comments

It is our understanding that this restriction has been imposed as a license condition for recently licensed plants rather than a technical specification. In discussions with the NRC technical specification reviewer it was agreed that a properly worded license condition more simply achieves the purpose. Accordingly we propose revising the statement to reflect a license condition.

SER Section: 15.2.3.3 SER Page: 15-16

Current SER Discussion

"Detection of high radiation signal in the main steam lines automatically closes the main steam line isolation valves, shuts down the mechanical vacuum pump and closes the isolation valve downstream of the pump."

Detroit Edison Comments

The discussion above is incorrect. This section should be reworded to read:

> Detection of a high radiation signal in the main steam lines automatically closes the main steam line isolation valves. The exhaust path from the condenser is automatically isolated by tripping of the mechanical vacuum pumps when the radiation level exceeds the trip setpoint for the exhaust monitor in the two-minute holdup pipe.

SER Section: Table 20-2 SER Page: 20-5

Current SER Discussion

The text on this page refers to a "30-year expected life of the facility".

Detroit Edison Comments

In amendment 50, Detroit Edison requested the NRC to issue the operating license to be effective for 40 years. (See attached)

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August 31, 1983

RECEIVED
AUG 31 1983

Mr. Harold Denton
Director
Office of the Nuclear
Reactor Regulation
U.S. Nuclear Regulatory
Commission
Washington, D.C. 20555

Re: The Detroit Edison Company
Enrico Fermi Atomic Power Plant
Unit 2 - Docket No. 50-341

Dear Mr. Denton:

As counsel for Detroit Edison, we enclose three (3) originals and nineteen (19) copies of Amendment No. 50 to the Amended and Substituted Application for Licenses, including certain pages which are to be substituted in the Application. These materials are being submitted for the purposes of

1) updating certain general information set forth in the Application;

2) amending the Application to reflect the addition of Co-Applicant Wolverine Power Supply Cooperative, Inc., which previously has been approved by the Commission as a co-holder of the Construction Permit issued in this proceeding;

3) revising the Application to request that the Commission issue the Applicant an Operating License for a term of forty (40) years from the date of issuance of the Operating License;

4) referencing the submittal on June 29, 1983 of Applicants' revised Physical Security Plan; and

5) transmitting revisions to Applicants' Final Safety Analysis Report.

Also enclosed are sixty-three (63) copies, including three (3) originals, of the modifications to the Company's Final Safety Analysis Report referred to in the Amendment.

Very truly yours,

LeBOEUF, LAMB, LEIBY & MacRAE

By L. Charles Landgraf
L. Charles Landgraf
Attorneys for The Detroit Edison Company

Enclosures

TO: Walter Pike

c/o: LORI STADLER

6/13/84 3453.31

TO: MATT RAGER
FROM: Hazel Jordan

BEFORE THE UNITED STATES

NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF:	)	
	)	
THE DETROIT EDISON COMPANY	)	
(Enrico Fermi Atomic Power	)	
Plant Unit No. 2)	)	Docket No. 50-341

AMENDMENT NO. 50
TO AMENDED AND
SUBSTITUTED APPLICATION FOR LICENSES

THE DETROIT EDISON COMPANY, Applicant in the above captioned proceeding, hereby files Amendment No. 50 to its Amended and Substituted Application for Licenses for the purposes of (1) updating certain general information set forth in the Application, (2) amending the Application to reflect the addition of Co-Applicant Wolverine Power Supply Cooperative, Inc., which previously has been approved by the Commission as a co-holder of the Construction Permit issued in this proceeding, (3) revising the Application to request that the Commission issue the Applicant an Operating License for a term of forty (40) years from the date of issuance of the Operating License, (4) referencing the submittal of Applicants' revised Physical Security Plan, and (5) transmitting revisions to Applicants' Final Safety Analysis Report.

Background

On April 29, 1969, Applicant Detroit Edison filed with the Atomic Energy Commission ("AEC") an Application for Licenses requesting authorization to construct and operate Enrico Fermi Atomic Power Plant Unit No. 2 ("Fermi 2"), a utilization facility for the generation of commercial power. Twenty amendments to this application were subsequently filed, and on September 26, 1972, the Commission issued Detroit Edison a construction permit (CPPR-87) for Fermi 2.

On October 22, 1974, Detroit Edison filed with the AEC its Amended and Substituted Application for Licenses for the purpose of transmitting the Final Safety Analysis Report (FSAR) and to bring up to date other information contained in the original Application for Licenses.

On May 6, 1977, Applicant filed with the Nuclear Regulatory Commission ("NRC" or the "Commission") an Application to Amend its Construction Permit to add Northern Michigan Electric Cooperative, Inc., and Wolverine Electric Cooperative, Inc., as co-owners of the Fermi 2 facility. After public notice and hearing, the Commission approved Amendment No. 1 to the Construction Permit on July 5, 1978 adding the two cooperatives as co-holders of the Construction Permit.

On August 13, 1982, Detroit Edison filed its second request to amend its Construction Permit to reflect a statutory merger between the two electric cooperatives and substitution of the resulting new entity: Wolverine Power Supply Cooperative, Inc. ("WPSC"). The NRC approved Amendment No. 2 to the Fermi 2 Construction Permit on December 1, 1982.

Proposed Amendment

The 1974 Amended and Substituted Application for Licenses should be revised to conform to the amended Construction Permit reflecting substitution for WPSC as a Co-Applicant.

In addition, in view of the considerable time which has passed since issuance of the Construction Permit, Detroit Edison amends its 1974 Application to request an Operating License with a term of forty years from the date of issuance of the Operating License. The general information in the Application should also be changed to conform with the above revision and with information contained in annual reports and other materials previously submitted. The attached revised pages replace the correspondingly numbered pages in Applicant's 1974 Amended and Substituted Application for Licenses.

In this Amendment, Detroit Edison is also requesting that its Application be amended to include the filing of its revised Physical Security Plan which was made separately on June 29, 1983 (reference: EF2-64,443).

Finally, Applicant is filing modifications to Chapter 7 and Appendix 4A of its FSAR along with several minor technical and administrative corrections and clarifications.

THE DETROIT EDISON COMPANY

By C M Heidel
Charles M. Heidel
President

Subscribed and sworn to before me
this [illegible] day of [illegible], 1983.

[illegible]
Evelyn M. Tracy, Notary Public
Oakland County, Acting in Wayne
County, Michigan

My Commission expires 4/30/84

SER Section: 22, Item I.A.1.2 SER Page: 22-6

Current SER Discussion

Item 5 under the description of typical duties and responsbilities of the Nuclear Shift Supervisor (NSS) reads as follows:

"5. Is responsible to determine the circumstances, analyze the cause and correct the fault before directing the return of the reactor to power after an explained trip, runback or power reduction."

Detroit Edison Comments

Some clarification of Item 5 is warranted. Under Fermi 2 Procedure 21.000.03, "Post Scram Evaluation and Restart Authorization", the NSS is responsible with the Shift Technical Advisor (STA) for completing the post scram data and evaluation form after an unscheduled reactor scram. The NSS and the STA are responsible for the initial post scram investigation. The Operations Engineer or his delegate and the Technical Engineer or his delegate verify the thoroughness, technical accuracy and consistency of the scram investigation. The Superintendent-Nuclear Production, or delegate, has the responsibility to grant permission to commence reactor startup following an unscheduled reactor scram.

This is consistent with Section 1.1 of EF2-66117, dated November 3, 1983, which provided Edison's response to Generic Letter 83-28.

Wayne H. Jens
Vice President
Nuclear Operations

Detroit Edison

2000 Second Avenue
Detroit, Michigan 48226
(313) 237-8612

November 3, 1983
EF2 - 66,117

Director of Nuclear Reactor Regulation
Attention: Mr. B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Youngblood:

Reference: (1) Enrico Fermi Atomic Power Plant, Unit 2
NRC Docket No. 50-341

(2) Letter, NRC to Detroit Edison, Generic Letter 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events", July 8, 1983

Subject: Detroit Edison Response to NRC Generic Letter 83-28

Attached please find our response to your Generic Letter 83-28. We have reviewed your positions and have summarized the Detroit Edison program relative to the positions on an item by item basis. Often we have referenced Detroit Edison procedures to demonstrate implementation of the program. Where a program is still being developed, we provide a description of the program and have included an estimated implementation date.

Should you have any questions regarding the above, please contact Mr. O. Keener Earle, (313) 586-4211.

Sincerely,

Wayne H. Jens

Attachment

cc: Mr. P. M. Byron
Mr. M. D. Lynch

~~8311080302 PDR~~

Mr. B. J. Youngblood

EF2 - 66,117
Page 2

I, WAYNE H. JENS, do hereby affirm that the foregoing statements are based on facts and circumstances which are true and accurate to the best of my knowledge and belief.

Wayne H. Jens
WAYNE H. JENS
Vice President - Nuclear Operations

On this 3rd day of November 1983, before me personally appeared Wayne H. Jens, being first duly sworn and says that he executed the foregoing as his free act and deed.

James J. Morgan
Notary Public

JAMES J. MORGAN
Notary Public, Oakland County, MI
My Commission Expires Jan. 3, 1987
Acting in Monroe County Michigan

DETROIT EDISON

ENRICO FERMI 2

RESPONSE TO GENERIC LETTER 83-28

NOVEMBER 1983

DETROIT EDISON
ENRICO FERMI 2

RESPONSE TO GENERIC LETTER 83-28

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DETROIT EDISON
ENRICO FERMI 2

RESPONSE TO GENERIC LETTER 83-28

ITEM 1.1 POST-TRIP REVIEW (PROGRAM DESCRIPTION AND PROCEDURE)

NRC Position - Licensees and applicants shall describe their program for ensuring that unscheduled reactor shutdowns are analyzed and that a determination is made that the plant can be restarted safely.

Fermi 2 Response

The Detroit Edison Company has a post-trip review program that will be used during the operation of Fermi 2 to ensure that unscheduled reactor shutdowns are analyzed to determine if the plant can be restarted safely. The controlling procedure for this program is draft Operations Procedure - Administrative, Number 21.000.03, "Post-Scram Evaluation and Re-Start Authorization". A copy of this procedure is attached to this report.* This procedure is consistent with the Nuclear Operations Directive Number 21, "Effective Problem Solving", also attached to this report. (A Nuclear Operations Directive is a policy document, issued by the Vice President of Nuclear Operations, communicating policy to Detroit Edison managers, supervisors and employes.) The recently issued INPO "Good Practice" document on post-trip reviews is being reviewed and its recommendations will be incorporated, where appropriate, into the present Fermi 2 procedure.

The following is an item-by-item summary of the Fermi 2 post-trip program compared to NRC Generic Letter 83-28 positions.

ITEM 1.1.1 NRC Request - Describe the criteria for determining the acceptability of restart.

Fermi 2 Response

The criteria for determining the acceptability of restart is defined in draft Operations Procedure-Administrative, 21.000.03, "Post-Scram Evaluation and Restart Authorization." The specific procedural requirements satisfy the following three basic criteria:

- Has the reactor plant responded properly with all applicable safety systems functioning as required?

*All of the Detroit Edison procedures referenced in this response to NRC Generic Letter 83-28 are referenced to demonstrate implementation of the responses, but they are not referenced to document commitments to the NRC. These procedures are controlled, living documents that may change depending on Fermi 2 operational and organizational needs.

- Has the cause of the reactor scram been determined and adequately explained?

- Are shift supervisory personnel satisfied that no unreviewed safety questions exist?

If responses to any of the above criteria are negative, an independent engineering analysis and a thorough administrative review and reporting process is required prior to any restart authorization.

ITEM 1.1.2 NRC Request - Describe the responsibilities and authorities of personnel who will perform the review and analysis of these events (unscheduled reactor shutdowns).

Fermi 2 Response

The Nuclear Shift Supervisor (NSS) has the following responsibilities for the post-trip review program (as identified in Operations Administrative Procedure 21.000.03):

- Ensure that the plant is stable and in a safe condition.

- Complete the Post-Scram Data and Evaluation Form.

- Consult with the Shift Technical Advisor (STA) in making the restart determination and ensure that the criteria of Item 1.1.1 are met.

- Contact the Technical Engineer as required by procedure.

- Provide documentation of the restart authorization.

The NSS has the authority to initiate a restart only if all criteria are met. The NSS has other recording, reporting and informing responsibilities in accordance with the overall Fermi 2 operations administrative program which compliment these efforts and provide for management review of his decisions.

The Shift Technical Advisor (STA) has the following responsibilities for the post-trip review program:

- To consult with the Nuclear Shift Supervisor on determining the acceptability of a plant restart based on his review of the Post-Scram Data and Evaluation Form.

- To provide input to the Nuclear Shift Supervisor concerning any unreviewed safety question that he believes may exist.

The Shift Technical Advisor reports by a matrix organization to the Nuclear Engineering department from which he can obtain additional technical assistance.

The Technical Engineer has the responsibility to perform a post-scram engineering review and issue a report of this review to the Superintendent-Nuclear Production to determine that the cause of any failure to meet the restart criteria (improper system response, inability to determine the cause of the scram, or an unreviewed safety question) has been thoroughly analyzed, determined, corrected and documented.

The Technical Engineer will draw on all available resources; informational and personnel, as necessary, to thoroughly address the technical issues raised. Informational resources available are parameters recorded in the Post-Scram Data and Evaluation Form by the Nuclear Shift Supervisor, as well as other information sources such as printouts from: sequence of events recorders, the process computer, and strip chart, as indicated in the response to Item 1.2, "Post-Trip Review Data and Information Capability." Personnel resources available are the operations, technical, and maintenance sections of the Nuclear Production department, and the Nuclear Engineering and Nuclear Administration departments.

The Superintendent - Nuclear Production has the responsibility for restart approval when any of the criteria of Item 1.1.1 are responded to negatively. He is to ensure that the cause of the failure to meet the restart criteria (improper system response, inability to determine the cause of the scram, or an unreviewed safety question) has been thoroughly analyzed, determined, corrected, and documented. Following this review and after consultation with the Technical Engineer and other personnel, as necessary, the Superintendent-Nuclear Production has the authority to approve a reactor plant restart.

The Operations Engineer has the following administrative responsibilities concerning the post-trip review effort:

- To conduct a post-event review of the Post-Scram Data and Evaluation Form.

- To ensure proper documentation of the authorization for plant restart, whether by Nuclear Shift Supervisor or Superintendent-Nuclear Production.

These specific responsibilities are included in the general responsibilities of the Operations Engineer which are defined in the overall operations administrative program. These responsibilities ensure that the Operations Engineer is actively involved in the review of any abnormal plant responses, corrective actions, and all decisions for a plant startup or restart.

In addition to these pre-restart activities, there are several follow-on analysis and review activities conducted following restart. Any scram requiring a post-scram engineering review by the Technical Engineer will also require an Internal Incident Report to be written and reviewed under the guidelines contained in the Administrative Procedure - General, Number 12.000.47, "Incident Reporting System." This procedure requires formal review of the Internal Incident Report by the Technical Engineer and by the

On-Site Safety Review Organization (OSRO). All Internal Incident Reports are also reviewed within the Fermi 2 Nuclear Operating Experience Reviews program. This program is described in the Nuclear Operations Program Description NOP-105, "Nuclear Operating Experience Reviews." Additionally, both the Nuclear Engineering department and the Nuclear Safety Review Group will receive copies of the post-scram evaluation and will selectively review the evaluation. When determined appropriate, these groups will conduct a detailed re-evaluation of the scram.

ITEM 1.1.3 NRC Request - Describe the necessary qualifications and training for the responsible personnel.

Fermi 2 Response

The qualifications and training of personnel responsible for the review, analysis, and restart authorization are presented in the FSAR, Sections 13.1 and 13.2. This training will be augmented to include special training on the conduct of post-scram reviews at Fermi 2 including the use of the sequence of events recorders and other devices providing important information.

ITEM 1.1.4 NRC Request - Describe the sources of plant information necessary to conduct the review and analysis. The sources of information should include the measures and equipment that provide the necessary detail and type of information to reconstruct the event accurately and in sufficient detail for proper understanding. (See Item 1.2)

Fermi 2 Response

The Post-Scram Data and Evaluation Form provides the Nuclear Shift Supervisor and the Shift Technical Advisor with the plant parameters and equipment status indications that are necessary to determine if the plant can meet the following basic restart criteria:

- Has the reactor plant responded properly with all applicable safety systems functioning as required?
- Has the cause of the reactor scram been determined and adequately explained?
- Are shift supervisory personnel satisfied that no unreviewed safety questions exist?

Additional sources of plant information are made available to the Technical Engineer for his detailed engineering analysis, if the restart criteria of the Post-Scram Data and Evaluation Form cannot be met. Additional instrumentation and sources of plant information are specified in the response to Item 1.2, "Post-Trip Review-Data and Information Capability."

ITEM 1.1.5 NRC Request - Describe the methods and criteria for comparing the event information with known or expected plant

behavior (e.g., that safety-related equipment operates as required by the Technical Specifications or other performance specifications related to the safety function).

Fermi 2 Response

The Fermi 2 post-trip review program compares actual event information with expected system response or behavior. The criteria for "expected" system or plant behavior is determined through the overall Fermi 2 operations program.

The training received by Fermi 2 operators, Nuclear Shift Supervisors, and Shift Technical Advisors includes general operating, operating surveillance, abnormal operating, and alarm response procedures. These procedures are written to satisfy Technical Specifications and in accordance with system design specifications. The procedures identify the proper system response and behavior criteria. The operating logs and an operational experience assessment program provide additional specific value criteria for both normal and experienced abnormal plant behavior.

ITEM 1.1.6 NRC Request - Describe the criteria for determining the need for independent assessment of an event (e.g., a case in which the cause of the event cannot be positively identified, a competent group such as the Plant Operations Review Committee, will be consulted prior to authorizing re-start) and guidelines on the preservation of physical evidence (both hardware and software) to support independent analysis of the event.

Fermi 2 Response

As previously described in the responses to Item 1.1.1 and Item 1.1.2, the Fermi 2 post-trip review program always requires an independent assessment if the Nuclear Shift Supervisor and the Shift Technical Advisor concur that any of the following basic criteria cannot be met:

- Has the reactor plant responded properly with all applicable safety systems functioning as required?
- Has the cause of the reactor scram been determined and adequately explained?
- Are shift supervisory personnel satisfied that no unreviewed safety questions exist?

The direct involvement of the Technical Engineer, the Superintendent-Nuclear Production, and the resources available to them such as the Nuclear Engineering department, provide the necessary independent assessment. In addition, an Internal Incident Report would have to be documented, (as described under Item 1.1.2), and reviewed by the Technical Engineer and the On-Site Safety Review Organization (OSRO).

The completed Post-Scram Data and Evaluation Form along with the printouts, graphs and recordings discussed in Item 1.2, includes the essential physical evidence necessary for an independent analysis of an event.

Item 1.1.7 NRC Request - Items 1.1.1 through 1.1.6 above are considered to be the basis for the establishment of a systematic method to assess unscheduled reactor shutdowns. The systematic safety assessment procedures compiled from the above items, which are to be used in conducting the evaluation, should be in the report.

Fermi 2 Response

Operations Administrative Procedure, 21.000.03, "Post-Scram Evaluation and Re-Start Authorization" contains the Fermi 2 post-trip review safety assessment method. As part of the Plant Operating Manual, any personnel responsibilities, authorities, or functions specified in Procedure 21.000.03, are consistent with and subject to plant administrative policies and practices.

ITEM 1.2 POST-TRIP REVIEW - DATA AND INFORMATION CAPABILITY

NRC Position - Licensees and applicants shall have or have planned a capability to record, recall and display data and information to permit diagnosing the causes of unscheduled reactor shutdowns prior to restart and for ascertaining the proper functioning of safety-related equipment.

Adequate data and information shall be provided to correctly diagnose the cause of unscheduled reactor shutdowns and the proper functioning of safety-related equipment during these events using systematic safety assessment procedures (Action 1.1). The data and information shall be displayed in a form that permits ease of assimilation and analysis by persons trained in the use of systematic safety assessment procedures.

Fermi 2 Response

The Detroit Edison Company has installed the necessary data and information systems at Fermi 2 to permit diagnosing the causes of unscheduled reactor shutdowns and determining the proper functioning of safety-related equipment. The Fermi 2 systems used to provide the diagnoses and determinations as required by draft Operations Procedure - Administrative Number 21.000.03, "Post-Scram Evaluation and Re-Start Authorization" include printouts from two sequence of events recorders, strip charts, and the plant process computer. The data and information provided by these systems allow for a complete systematic assessment of unscheduled reactor shutdowns. The following is an item-by-item summary of the Fermi 2 data and information systems compared to NRC positions.

ITEM 1.2.1 Capability for assessing sequence of events (on-off indicators).

ITEM 1.2.1.1 NRC Request - Provide a brief description of equipment.

Fermi 2 Response

Two dedicated sequence of events recorder systems have been provided for assessing the sequence of events on Fermi 2. The primary sequence of events recorder has a capacity of 2200 inputs and includes both nuclear steam supply (reactor protection system trip logic) and balance-of-plant (BOP) signals. The second smaller sequence of events recorder has a capacity of 120 inputs and is dedicated to monitoring the reactor protection system trip logic only. Each system shares the same input logic contacts, but are isolated from each other by optical coupling devices. The primary recorder displays the recorded sequence on two printers located on the operators record desk in the main control room. The smaller recorder is located in the equipment cabinet in the relay room.

ITEM 1.2.1.2 NRC Request - Discuss parameters monitored.

Fermi 2 Response

The primary trip variables for each scram channel of the Reactor Protection System (RPS) are monitored by both sequence of events recording systems. The resulting RPS sequence data set currently consists of approximately 54 inputs. A summary of the monitored reactor protection system variables is included in Table 1.2.1.2. Each variable generally requires several inputs.

ITEM 1.2.1.3 NRC Request - Describe time descrimination between events.

Fermi 2 Response

Both dedicated sequence of events recording systems have the ability to resolve events to one millisecond.

ITEM 1.2.1.4 NRC Request - Describe the format for displaying data and information.

Fermi 2 Response

The format of the data and information printed on the primary sequence of events recorder includes: the type of event; the time of event in hours, minutes, seconds and milliseconds of real time; a four digit point identification; and an alpha-numeric description of the event. The format for the smaller recorder, which only prints the RPS trip logic data, is similar but without the alpha-numeric description.

ITEM 1.2.1.5 NRC Request - Discuss capability for retention of data and information.

Fermi 2 Response

Both sequence of events recording systems provide infinite retention capability since the final records are printed on hard copy.

ITEM 1.2.1.6 NRC Request - Describe the power sources.

Fermi 2 Response

Both sequence of events recording systems are powered directly from the plant BOP battery. All of the associated AC operated devices are supplied by battery inverters making both sequence of events recorders independent of AC power supplies.

ITEM 1.2.2 Capability for assessing the time history of analog variables needed to determine the cause of unscheduled reactor shutdowns, and the functioning of safety-related equipment.

ITEM 1.2.2.1 NRC Request - Provide a brief description of equipment (e.g., plant computer, dedicated computer, strip charts).

Fermi 2 Response

The ability to record the important analog variables needed to determine the cause of unscheduled reactor scrams has been provided by two distinct techniques at Fermi 2. The first method is through the use of dedicated strip chart recording devices located on the control room operating panels. The second method provided is the post-scram log generated by the plant process computer.

ITEM 1.2.2.2 NRC Request - Describe parameters monitored, sampling rate, and basis for selecting parameters and sampling rate.

Fermi 2 Response

Reactor parameters which are used to determine the cause of scrams and the proper functioning of safety-related equipment are pressure, water level, and neutron flux level which are continuously recorded on strip chart recorders. The computer post-scram log of the process computer is triggered into operation by a reactor scram, and will record 15 preselected analog variables at a rate which samples each point every 5 seconds. Parameters are selected to allow rapid determination that the reactor safety analysis limits were not exceeded and include: neutron flux, reactor pressure, core pressure, feedwater flow, reactor water level, steam flow, recirculation flow, and feedwater temperature.

ITEM 1.2.2.3 NRC Request - Describe the duratation of the time history (minutes before trip and minutes after trip).

Fermi 2 Response

The recordings produced by the dedicated strip chart recorders are continuous, and therefore the entire time history is available. The post-scram log on the plant process computer provides the values of the variables for a period of 5 minutes before and after the scram occurs.

ITEM 1.2.2.4 NRC Request - Describe the format for displaying data including scale (readability) of time histories.

Fermi 2 Response

The format of the dedicated recorders are major divisions linearly spaced over the range of the instrument. Intermediate range neutron flux is a manually ranged variable and is scaled 0 to 40 and 0 to 125 percent; power range neutron flux is scaled from 0 to 125 percent, reactor pressure is scaled from 0 to 1500 psig and the wide range water level is scaled from 10 to 220 inches above the top of active fuel. Flux recorders have a readability of 1 percent, pressure 20 psig, and level 2 inches.

The plant computer system will provide a table of point identification numbers, and point descriptions followed by the pre-scram and post-scram values of the variables.

ITEM 1.2.2.5 NRC Request - Describe the capability for retention of data, information, and physical evidence (both hardware and software).

Fermi 2 Response

For both types of analog recording, the use of a printed record results in infinite retention capability. The process computer log is automatically archived on magnetic tape for future use by the plant staff.

ITEM 1.2.2.6 NRC Request - Describe the power source(s) (e.g., class IE, non-class IE, noninterruptible).

Fermi 2 Response

Power is supplied to the level and pressure recorders by Class 1E battery inverters. A BOP uninterruptible power supply provides the power for the neutron monitor recorders. The plant process computer is supplied by a highly reliable non-class 1E AC power source.

ITEM 1.2.3 NRC Request - Describe other data and information provided to assess the cause of unscheduled reactor shutdowns.

Fermi 2 Response

Fermi 2 will have an additional system that can also be used for post-scram logging of transient and accident events. This is the Emergency Response Information System (ERIS) computer system described in Appendix H.III.A.1.2.7 of the Fermi 2 FSAR.

ITEM 1.2.4 NRC Request - Provide the schedule for any planned changes to existing data and information capability.

Fermi 2 Response

No changes are planned for the existing Fermi 2 data and information systems. The ERIS system is expected to be functional by September, 1984, as described in Detroit Edison letter EF2-62,262 to the NRC dated June 23, 1983.

Table 1.2.1.2

Reactor Protection System Variables Monitored by the Fermi 2 Sequence of Events Recorders

1. APRM Upscale Trip on Level.
2. Scram Discharge Volume High Water Level.
3. IRM Upscale Trip on Level.
4. Reactor Neutron Monitor System Trip.
5. Reactor Vessel Low Water Level.
6. Main Steam Line Isolation Valve Closure.
7. Reactor Vessel High Pressure.
8. Primary Containment High Pressure.
9. Manual Scram.
10. Reactor Scram.
11. Turbine Control Valve Fast Closure.
12. Turbine Stop Valve Closure.
13. Main Steamline High Radiation.

ITEM 2.1 EQUIPMENT CLASSIFICATION AND VENDOR INTERFACE (REACTOR TRIP SYSTEM COMPONENTS)

ITEM 2.1.1 Equipment Classification (Reactor Trip System Components).

NRC Position - Licensees and applicants shall confirm that all components whose functioning is required to trip the reactor are identified as safety-related on documents, procedures and information handling systems used in the plant to control safety-related activities, including maintenance, work orders, and parts replacement.

Fermi 2 Response

Detroit Edison has identified all components of the Reactor Trip System (RTS) which should be classified as safety-related for Fermi 2. These components include all active components of existing plant systems that function to implement a reactor scram. The following documents and procedures used in the plant to control safety-related activities, including maintenance, work orders and parts replacement, are being reviewed to ensure that these components are appropriately identified as safety-related:

- Documents - Drawings (P&ID's, Schematics) and Equipment History Folders (where applicable), Master Instrument List, Mechanical Equipment List, QA1 Major Electrical Equipment List, QA Level 1 Electrical Cables List, QA Level 1 Valves List, and QA1-Motor List.

- Procedures - Surveillance and Maintenance Administrative Controls.

The preliminary results of this review indicate that Fermi 2 has already established sufficient administrative controls and procedural practices to meet this position.

Detroit Edison intends to complete this review and correct any deficiencies to ensure that all documents and procedures are complete, accurate, and identified as safety-related for all Reactor Trip System components. It is estimated that this task will be completed by April 1, 1984.

Detroit Edison also is an active participant in a BWR Owners Group considering special programs in this area. Detroit Edison will use the results of these programs, as appropriate, to check its equipment classification and safety-related document identification program.

ITEM 2.1.2 Vendor Interface (Reactor Trip System Components).

ITEM 2.1.2.1 NRC Position - For these components, applicants shall establish, implement and maintain a continuing program to ensure that vendor information is complete, current and controlled throughout the life of the plant, and appropriately referenced or incorporated in plant instructions and procedures.

Fermi 2 Response

Detroit Edison's current program to control vendor information including Reactor Trip System (RTS) components is discussed in Item 2.2.2.1.

The experience gained from this current program will be used to establish an improved vendor information program, as discussed in Item 2.2.2.1, to be used during the operation of Fermi 2. The Reactor Trip System is included in this program and will be the first part of the program implemented. For the Reactor Trip System, the program will meet the following requirements:

1. The responsibilities for the receipt, review, approval, distribution, and use of vendor manuals and related vendor information pertinent to the Reactor Trip System (RTS) components will be established.

2. Specific administrative controls for the receipt, storage and distribution of vendor information pertinent to RTS components will be established.

3. Technical controls necessary to provide for the technical review, approval, and use of vendor information, including the control of revisions or changes to the vendor information pertinent to RTS components, initiated either by Detroit Edison or the vendor, will be established.

Detroit Edison will establish the appropriate arrangements to ensure that information for the RTS components is complete, current, and its use controlled throughout the life of the plant. The estimated schedule for implementation of this improved vendor information program for the RTS is June 1, 1984.

ITEM 2.1.2.2 NRC Position - Vendors of these components should be contacted and an interface established. Where vendors cannot be identified, have gone out of business, or will not supply the information, the licensee or applicant shall assure that sufficient attention is paid to equipment maintenance, replacement, and repair, to compensate for the lack of vendor backup, to assure reactor trip system reliability. The vendor interface program shall include periodic communication with vendors to assure that all applicable information has been received. The program should use a system of positive feedback with vendors for mailings containing technical information. This could be accomplished by licensees acknowledging receipt of technical mailings. The program shall also define the interface and division of responsibilities among the licensees and the nuclear and nonnuclear divisions of their vendors that provide service on reactor trip system components to assure that requisite control of and applicable instructions for maintenance work are provided.

Fermi 2 Response

The existing interface between Detroit Edison and General Electric (our prime RTS component supplier) includes GE initiated Service Information Letters (SIL's), Application Informations Document (AID's) and other specific GE technical letters directed to Detroit Edison. Detroit Edison presently has a controlled process to receive, review, approve, control, and utilize such information. The Operating Experience Review (OER) Program at Detroit Edison includes GE originated SIL's and AID's as well as INPO originated reports (SER, SOER, AND O&MR's), NRC I&E Bulletins, Circulars, and Notices, and other miscellaneous documents including INPO "NOTEPAD" generated questions or items applicable to Detroit Edison.

In support of this ongoing effort, Detroit Edison in 1982, backordered all SIL's designated by General Electric to be potentially applicable to Fermi 2, to assure that all such SIL's have been addressed. A system will be established to ensure receipt of all applicable SIL's. This review program is described in Nuclear Operations Program Description NOP-105, "Nuclear Operating Experience Reviews."

To further enhance the vendor interfaces, Detroit Edison will be contacting RTS component suppliers to update vendor information pertinent to RTS components. The schedule for the completion of this RTS vendor interface activity is June 1, 1984. Detroit Edison is an active participant in the BWR Owners Group Committee and the Nuclear Utility Task Action Committee (NUTAC) Group on Generic Letter 83-28. Detroit Edison will consider Owners Group and NUTAC recommendations as they are developed and will modify its vendor interface program based on these recommendations, as appropriate.

The primary source of RTS components vendor information are the operational and/or maintenance manuals provided to Detroit Edison by General Electric or other vendors. These documents generally contain: component or system operating procedures, preventive maintenance requirements, calibration procedures, removal/replacement instructions, post-maintenance test procedures, component parts lists, and related drawings as appropriate. The use of this vendor information by plant personnel in conducting the required maintenance, operations, calibration, parts replacement, and other related activities will be accomplished as described in Item 2.2.2.1.

ITEM 2.2 EQUIPMENT CLASSIFICATION AND VENDOR INTERFACE (PROGRAMS FOR ALL SAFETY-RELATED COMPONENTS)

ITEM 2.2.1 Equipment Classification (Programs For All Safety-Related Components).

<u>NRC Position</u> - For equipment classification, licensees and applicants shall describe their program for ensuring that all components of safety-related systems necessary for accomplishing required safety functions are identified as safety-related on documents, procedures, and information handling systems used in the plant to control safety-related activities, including maintenance, work orders and replacement parts.

ITEM 2.2.1.1 <u>NRC Request</u> - Describe the criteria for identifying components as safety-related within systems currently classified as safety-related. This shall <u>not</u> be interpreted to require changes in safety classification at the systems level.

<u>Fermi 2 Response</u>

The general basis used for identifying safety-related structures, equipment and components is described in the FSAR, Section 3.2. If credit is taken for operation of any system or component to (a) prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary (RCPB), (b) permit shutdown of the reactor and maintain it in the safe shutdown condition, and (c) contain radioactive material; then that system, component, or structure is designated safety-related.

Many systems and components were identified by the NSSS vendor (General Electric) as safety-related in the original design. Systems were also developed by Edison for which Design Instructions and P&ID's were prepared. The Design Instructions and P&ID's were prepared utilizing input from General Electric and the Fermi 2 PSAR. The Design Instructions provide essential information describing the system function, which would include the safety-related status. The Design Instructions were written based upon a generic guide so that all essential information is provided. The P&ID's augment the information of the Design Instructions, showing all major components of the system, also including the safety-related system classification. In general, all components associated with a system designated to be safety-related are, in fact, safety-related. The designer made this assumption unless there was concrete evidence that the component does not perform a safety-related function.

Additions or modifications to systems were made during the design and construction phase of Fermi-2. Revisions or additions to systems, including classification of added or changed components, were controlled utilizing procedure based multiple levels of review.

To aid in component identification, various lists were prepared as part of the design process. The lists identify components by Plant Identification System (PIS) number and include a safety classification. Procedures were developed to control the information on the lists. These equipment lists have been subject to review and audit.

For maintenance and surveillance, procedures have been, and continue to be developed for identification of safety-related components. The procedures generally require reference to design documents, drawings or lists for classifications of components.

For procurement of spare parts for maintenance, procedures have been written requiring technical review of all requisitions. The technical reviewer's procedure includes guidance for determining the safety classification of a sub-component in accordance with the definition referenced above. Review and signature by the Procurement Quality Assurance section and the responsible Section Head is also required.

The criteria and methodology described above adequately and conservatively identify safety-related components because:

1. Adequate direction in the form of Design Specifications was obtained from the NSSS vendor to identify systems and components in vendor supplied systems as safety-related.

2. P&ID's and Design Instructions were prepared by Detroit Edison which include identification of safety-related status (subject to multi-level review and approval).

3. Within safety-related systems, designers designated components and sub-components as safety-related unless there was justification that the component or sub-component did not perform a safety function.

4. Any change addition or deletion affecting safety-related components is subject to multi-level review.

5. For maintenance, surveillance and parts procurement, procedures are prepared, or in the process of being prepared, which require either: the careful review of existing Fermi 2 documents to obtain the predetermined safety classification, or the evaluation of the component function to determine the safety-related status.

ITEM 2.2.1.2 <u>NRC Request</u> - Provide a description of the information handling system used to identify safety-related components (e.g., computerized equipment list) and the methods used for its development and validation.

<u>Fermi 2 Response</u>

The information handling system for Fermi 2 includes equipment and components identified in FSAR Section 3.2 (Table 3.2-1), electrical diagrams, P&ID's and equipment lists at the component level. The Fermi 2 information

handling system was developed using the methodology described in Item 2.2.1.1 and identifies safety-related equipment on a component level.

Detroit Edison procedures require that these documents be reviewed and approved by several levels within the Fermi 2 organization, and revision control is required for future changes.

These documents, which are available to plant personnel, contain the predetermined safety classification of plant components. The equipment and components are identified by Plant Identification System (PIS) numbers, which is a numbering system that station personnel are familiar with and use routinely. This system, developed by the Fermi 2 Project, has been validated by review and audit. Provisions within Detroit Edison's Quality Assurance Program assures that the information handling system is maintained current, and that revisions are controlled.

ITEM 2.2.1.3 NRC Request - Provide a description of the process by which station personnel use this information handling system to determine that an activity is safety-related and what procedures for maintenance, surveillance, parts replacement and other activities defined in the introduction to 10CFR50, Appendix B, apply to safety-related components.

Fermi 2 Response

As outlined below, Fermi 2 has approved procedures controlling activities for safety-related components during maintenance, surveillance, parts replacements and other activities as defined in the introduction to 10CFR50 Appendix B. These approved procedures assure that safety-related components are treated as such during plant activities. The predetermined safety classification minimizes the potential for errors which might result from determinations made on a case-by-case basis. The process pertaining to these activities is summarized below:

Procurement, Storage, and Spare-Parts Replacement

When a replacement component is ordered, the component is evaluated to determine whether or not it is safety-related. A technical evaluation is performed using approved procedures. In accordance with these procedures, the design, qualification, and quality assurance requirements are specified for safety-related components. This information is applied to the purchase order, receipt inspection, storage, and issuance of safety-related components. The user of a spare or replacement component is required to specify the safety classification of the component based on its application, and on the predetermined classification in the information handling system.

Maintenance and Surveillance

Prior to the commencement of maintenance and surveillance activities, Work Orders are prepared and processed in accordance with the approved Plant Procedure 12.000.15, "PN-21 Work Order Processing." During Work Order

preparation and review, approved procedures are used to determine a component's safety classification. At a minimum, the contents of a Work Order considers and documents the disposition of the following: (1) safety classification; (2) applicable plant procedures; (3) controlled drawings; (4) quality assurance requirements; and (5) reviews and approvals pertinent to the maintenance and/or surveillance of the component.

Approved plant procedures (as designated within the Work Order) govern the actual performance of: (1) routine and non-routine preventative maintenance; (2) non-routine corrective maintenance; (3) routine surveillance; and (4) post-maintenance testing (see Item 3.2).

ITEM 2.2.1.4 NRC Request - Describe the management controls utilized to verify that the procedures for the preparation, validation, and routine utilization of the information handling system have been followed.

Fermi 2 Response

Administrative procedures and the Detroit Edison quality assurance program for Fermi 2, control activities and procedures related to the information handling system. These controls govern the preparation, validation and routine use of the information handling system. The controls provide for checks, reviews, approvals, controlled documents and QA audits related to safety-related activities. These provisions help assure that approved procedures are followed. Furthermore, a complete review of the adequacy of the administrative controls is performed by the Onsite Review Organization (OSRO). This review will assist in ensuring the routine utilization of specified management controls by plant personnel.

ITEM 2.2.1.5 NRC Request - Demonstrate that appropriate design verification and qualification testing is specified for procurement of safety-related components. The specifications shall include qualification testing for expected safety service conditions, and provide support for the licensee's receipt of testing documentation to support the limits of life recommended by the supplier.

Fermi 2 Response

The program for component procurement includes a technical evaluation which assures that the appropriate design verification and qualification testing is specified for procurement of safety-related components. This program includes: approved procedures which require a determination of the safety classification of the component (MI-245, Maintenance Instruction - "Criteria for Technical Review"), the environmental conditions associated with the in-plant application of the component, and the qualification testing requirements for the component.

Plant personnel perform these activities using approved procedures. These procedures include the use of predetermined information contained in the information handling system. This process is subject to audit under the Detroit Edison quality assurance program for safety-related components.

ITEM 2.2.1.6 NRC Request - Licensees and applicants need only to submit for staff review the equipment classification program for safety-related components. Although not required to be submitted for staff review, your equipment classification program should also include the broader class of structures, systems, and components important to safety required by GDC-1 (defined in 10CFR Part 50, Appendix A, "General Design Criteria, Introduction").

Fermi 2 Response

The Fermi 2 program for classification of safety-related components is described above in Items 2.2.1.1 through 2.2.1.5. Detroit Edison, in addition, has generally applied design and quality standards to nonsafety-related structures, systems, and components in a manner comensurate with the functions of such items in the overall safety and operation of the plant. Detroit Edison is also an active member of the Utility Safety Classification Group and will specifically respond to the NRC on this issue based on the Group's recommendation. Detroit Edison is confident that the quality and design standards which were used for Fermi 2 adequately ensure nonsafety-related equipment will perform its intended function.

ITEM 2.2.2 Vendor Interface (All Safety-Related Components).

ITEM 2.2.2.1 NRC Request - For vendor interface, licensees and applicants shall establish, implement and maintain a continuing program to ensure that vendor information for safety-related components is complete, current and controlled throughout the life of their plants, and appropriately referenced or incorporated in plant instructions and procedures.

Fermi 2 Response

Detroit Edison's current program to control vendor information is documented in project procedures used for the design and construction of Fermi 2. These project procedures provide the following:

1. The administrative procedures necessary to receive, control, store and distribute vendor information (drawings and documents, exclusive of manuals).

2. The administrative procedures necessary to receive and distribute vendor operations and maintenance manuals.

3. The procedures for technical review, approval and control of the use of vendor drawings and documents and any revisions to them (initiated either by Detroit Edison or the vendor).

Detroit Edison is currently establishing an improved vendor information program to be used during the operation of Fermi 2. This program will be based on the experience gained during the construction of the plant.

The vendor information program at Fermi 2 will include:

1. Specific identification of responsibilities for the receipt, review and approval, distribution, and use of vendor manuals and related vendor information pertinent to safety-related components.

2. Establishment of the administrative controls necessary to provide for the receipt, storage and distribution of vendor information pertinent to safety-related components.

3. Provisions for the technical review, approval, and use of vendor information, including the control of revisions or changes to the vendor information.

Procedures are being established to define, implement, document, and maintain a program to ensure that vendor supplied information of safety-related components is complete, current, and their use controlled throughout the life of the plant. The schedule for the implementation of this vendor information program is June 1, 1983.

The organizational responsibilities for the implementation of the vendor information program will include the following activities by the organizational units of Nuclear Operations:

1. Nuclear Administration:

   Information Systems - Receive and process all manuals, supplements, revisions, and Engineering Change Notices. Nuclear Administration's Automated Records Management System (ARMS) will contain applicable information necessary for identification, control, and retrieval. The ARMS listing will show:

   a. Document status

   b. Document revision level

   c. Document number

   d. Originator

   e. Reference to the component or sub-system

   Information Systems shall record, film and establish controlled files in the Production Information Center, from which all vendor information is checked out. Vendor information will be available to all users on an around the clock basis. Only "approved for use" materials (or copies of) will be distributed to users. Attached to each document will be a cover sheet clearly stating its review and revision status and the statement "controlled."

   Nuclear Procurement - Order new, lost or replacement vendor information as requested by Nuclear Engineering, Nuclear Production or

Nuclear Administration. Nuclear Procurement will initiate contact with vendors as required to obtain updates or new information pertinent to safety-related vendor supplied components.

NOTE: This process is subject to considerations and actions of the Nuclear Utility Task Action Committee (NUTAC) on Generic Letter 83-28 and the related BWR Owners Group Committee.

2. Nuclear Engineering:

Will be responsible, with support from Nuclear Production personnel, as appropriate, for the technical review, evaluation and approval of vendor supplied information. Nuclear Engineering is also responsible for verification of assigned document numbers, and for approving and/or initiating and approving required Engineering Change Notices generated by any user.

3. Nuclear Production:

Will support Nuclear Engineering in the technical review and evaluation of vendor supplied information when requested. Additionally, Nuclear Production will be responsible for implementing the use of approved and controlled vendor supplied information. Nuclear Production will have access to the Production Information Center from which they will obtain the applicable, controlled information as necessary. The use of vendor information will be in accordance with approved plant procedures and instructions. Nuclear Production initiated modifications or changes to vendor supplied information will be controlled and approved by Nuclear Engineering, and documented as being approved, prior to use by plant personnel.

ITEM 2.2.2.2 NRC Request - Vendors of safety-related equipment should be contacted and an interface established. Where vendors cannot be identified, have gone out of business, or will not supply information, the licensee or applicant shall assure that sufficient attention is paid to equipment maintenance, replacement, and repair, to compensate for the lack of vendor backup, to assure reliability commensurate with its safety function (GDC-1). The program shall be closely coupled with action 2.2.1 above (equipment qualification). The program shall include periodic communication with vendors to assure that all applicable information has been received. The program should use a system of positive feedback with vendors for mailings containing technical information. This could be accomplished by licensee acknowledgement for receipt of technical mailings. It shall also define the interface and division of responsibilities among the licensee and the nuclear and nonnuclear divisions of their vendors that provide service on safety-related equipment to assure that requisite control of and applicable instructions for maintenance work on safety-related equipment are provided.

Fermi 2 Response

As discussed in Item 2.2.2.1, Detroit Edison has a program for interfacing with vendors during the construction phase of Fermi 2. The experience gained from this interfacing during construction will be used to establish the program for the operation of Fermi 2. Detroit Edison is also an active participant in a NUTAC group created to address this item. Detroit Edison intends to incoporate into its vendor interface program the results of the NUTAC group. These results are expected to be available for approval during February, 1984.

ITEM 3.1 POST-MAINTENANCE TESTING (REACTOR TRIP SYSTEM COMPONENTS)

ITEM 3.1.1 NRC Request - Licensees and applicants shall submit the results of their review of test and maintenance procedures and Technical Specifications to assure that post-maintenance operability testing of safety-related components in the reactor trip system is required to be conducted and that the testing demonstrates that the equipment is capable of performing its safety functions before being returned to service.

Fermi 2 Response

The Detroit Edison Company's committment to operate Fermi 2 in accordance with Plant Technical Specifications mandates the Fermi 2 post-maintenance test program for safety-related equipment. Periodic equipment and instrumentation operability testing is required by Plant Technical Specifications; Section 4.0, "Surveillance Requirements." These surveillance requirements call for a variety of tests to demonstrate the functional OPERABILITY of the associated equipment, system, or instrumentation channel and are required to be performed following any RTS maintenance.

The Plant Operations Manual (POM) includes the Nuclear Operations and I&C surveillance program procedures that implement the Technical Specification surveillance requirements and establish OPERABILITY of the associated equipment, system, or instrumentation channel. Plant Procedure 12.000.15, "PN-21 Work Order Processing," provides for specification of these post-maintenance testing requirements.

Prior to declaring a component OPERABLE (returning it to service) to meet a particular Limiting Condition for Operation (LCO), all the applicable surveillance requirements for the LCO will have been met. A computerized system correlating the specific surveillance procedure(s) to the specific surveillance requirement has already been established and will be operational prior to fuel loading.

The Nuclear Operations and I&C surveillance programs have been designed to facilitate post-maintenance testing. The divisional and channelized features of these programs will aid in the accurate identification of specific post-maintenance testing requirements. All components whose functioning is required to trip the reactor are demonstrated operable in these programs. These procedures are all safety-related and are approved by the On-site Safety Review Organization (OSRO).

The Fermi 2 Technical Specifications are still in the review and approval stage. If, during the Detroit Edison review, any changes are identified as necessary for the RTS, the changes and justification will be submitted for NRC review.

ITEM 3.1.2 NRC Request - Licensees and applicants shall submit the results of their check of vendor and engineering recommendations to ensure that any appropriate test guidance is included in the test and maintenance procedures or the Technical Specifications, where required.

Fermi 2 Response

Detroit Edison has endeavored to include applicable vendor and engineering recommendations in the development of its various procedures, programs and plant Technical Specifications. All such procedures reference the appropriate source material. This includes the updated material contained in General Electric's SIL's and AID's, as well as other experience related information, as it is processed through the Nuclear Operating Experience Reviews program described under Item 2.1.2.2. Moreover, the existing administratively required periodic review of procedures (Administrative Procedure - General, Number 12.000.24, "Periodic Review of Plant Procedures") will be augmented in conjunction with the improved vendor information program, discussed under Items 2.1.2.1 and 2.2.2.1, to include a check to assure that current vendor and engineering recommendations are appropriately included in the relevant safety-related test and maintenance procedures. For the RTS related procedures, this will begin as soon as the relevant vendor information is updated as described under Item 2.1.2.1.

ITEM 3.1.3 NRC Request - Licensees and applicants shall identify, if applicable, any post-maintenance test requirements in existing Technical Specifications which can be demonstrated to degrade rather than enhance safety. Appropriate changes to these test requirements, with supporting justification, shall be submitted for staff approval.

Fermi 2 Response

The Fermi 2 Technical Specifications are still in the review and approval stage. If, during the Detroit Edison review and use of the Fermi 2 Technical Specifications, any requirements are discovered that degrade rather than enhance safety, the appropriate changes and justification will be submitted for NRC review.

ITEM 3.2 POST-MAINTENANCE TESTING (ALL OTHER SAFETY-RELATED COMPONENTS)

ITEM 3.2.1 NRC Request - Licensees and applicants shall submit a report documenting the extending of test and maintenance procedures and Technical Specifications review to assure that post-maintenance operability testing of all safety-related equipment is required to be conducted and that the testing demonstrates that the equipment is capable of performing its safety functions before being returned to service.

Fermi 2 Response

As discussed in our response to Item 3.1.1, post-maintenance testing is inherently required by plant Technical Specifications for all equipment, systems, or instrumentation channels covered by a Secton 4.0 surveillance requirement and a Limiting Condition for Operation.

The existing Fermi 2 computerized system that correlates specific surveillance requirements to the procedures that fulfill those requirements already extends to all systems covered by plant Technical Specifications. In addition, a prioritized Preventative Maintenance Program includes all other Technical Specifications related components*, not specifically required by Technical Specifications or covered by an individual surveillance procedure, and assigns them the highest priority category.

ITEM 3.2.2 NRC Request - Licensees and applicants shall submit the results of their check of vendor and engineering recommendations to ensure that any appropriate test guidance is included in the test and maintenance procedures or the Technical Specifications where required.

Fermi 2 Response

Detroit Edison has endeavored to include applicable vendor and engineering recommendations in the development of its various Nuclear Operations procedures, programs and plant Technical Specifications. All such procedures reference the appropriate source material. This includes appropriate vendor manuals as well as updated material contained in General Electric's SIL's and AID's and other experience related information as it is processed through the Nuclear Operating Experience Reviews Program described under Item 2.1.2.2. Moreover, the existing administratively required periodic review of all procedures (Administrative Procedure - General, 12.000.24, "Periodic Review of Plant Procedures") will be augmented in conjunction with the improved vendor information program discussed under Items 2.1.2.1 and 2.2.2.1. The improved vendor information program includes a check to assure that current vendor and engineering recommendations are appropriately included in the relevant safety-related test and maintenance procedures.

*Such as an instrument necessary in performing Technical Specification surveillance, but not germane to the Technical Specification itself.

ITEM 3.2.3 NRC Request - Licensees and applicants shall identify, if applicable, any post-maintenance test requirements in existing Technical Specifications which are perceived to degrade rather than enhance safety. Appropriate changes to these test requirements, with supporting justification, shall be submitted for staff approval.

Fermi 2 Response

The Fermi 2 Technical Specifications are still in the review and approval stage. If, during the Detroit Edison review of the Fermi 2 Technical Specifications, any requirements are discovered that degrade rather than enhance safety, the appropriate changes and justification will be submitted for NRC review.

(Items 4.1 thru 4.4 do not apply to boiling water reactors)

ITEM 4.5 REACTOR TRIP SYSTEM RELIABILITY (SYSTEM FUNCTIONAL TESTING)

NRC Position - On-line functional testing of the reactor trip system, including independent testing of the diverse trip features, shall be performed on all plants.

Fermi 2 Response

At Fermi 2 detailed surveillance requirements and sufficient administrative programs are "in place" to ensure that thorough on-line functional testing of the reactor trip system is performed. The following are responses to the specific requests of the NRC concerning this issue:

ITEM 4.5.1 NRC Request - The diverse trip features to be tested include the scram pilot valve and the backup scram valves (including all initiating circuitry) on GE plants.

Fermi 2 Response

The reactor trip system components at Fermi 2 that are required to function to cause a reactor scram fall into two categories:

1. Components required to function for the insertion of all rods (common).

2. Components required to function for the insertion of each individual rod (185 sets of these).

The components whose function is common to all rods are the initiating circuitry and the final output relays. All the diverse initiating circuits and final output relays are on-line functionally tested in accordance with plant Technical Specifications, Section 3.3.1, Reactor Protection System Instrumentation.

The components required to function for the insertion of the individual control rod (185 sets of these) are on-line functionally tested by a sample group in accordance with plant Technical Specifications, Section 3.1.3.2, "Control Rod Maximum Scram Insertion Times". This is accomplished by individually scramming at least 10% of the control rods, on a rotating basis, every 120 days of power operation. The 185 sets of pilot scram valves are included in this group of components.

The backup scram valves and associated logics are tested at each refueling outage (or every 18 months) in the Reactor Protection System Logic Functional Test in accordance with plant Technical Specifications, Section 3.3.1, "Reactor Protection System". Fermi 2 will also administratively require that the "low scram header pressure" alarm be acknowledged after each scram occurrence prior to resetting the scram logic. (This will

confirm that at least one of the backup scram valves has functioned properly.) The NRC has indicated that this is an adequate method to ensure the operability of the backup scram valves in NUREG-0979, Safety Evaluation Report related to the final design approval of the GESSAR II BWR/6 Nuclear Island Design (April 1983).

It should be noted that possible modifications to the RTS based on the NRC's final ATWS rule could change this response.

ITEM 4.5.2 NRC Request - Plants not currently designed to permit periodic on-line testing shall justify not making modifications to permit such testing. (Remainder of item applicable to licensees only.)

Fermi 2 Response

As indicated in the response to Item 4.5.1 above, Fermi 2 is designed to permit on-line testing of the reactor trip system. Therefore, this item is not applicable to Fermi 2.

ITEM 4.5.3 NRC Request - Existing intervals for on-line functional testing required by Technical Specifications shall be reviewed to determine that the intervals are consistent with achieving high reactor trip system availability when accounting for considerations such as:

1. uncertainties in component failure rates
2. uncertainty in common mode failure rates
3. reduced redundancy during testing
4. operator errors during testing
5. component "wear-out" caused by the testing

Licensees currently not performing periodic on-line testing shall determine appropriate test intervals as described above. Changes to existing required intervals for on-line testing as well as the intervals to be determined by licensees currently not performing on-line testing shall be justified by information on the sensitivity of reactor trip system availability to parameters such as the test intervals, component failure rates, and common mode failure rates.

Fermi 2 Response

Detroit Edison is an active member of the BWR Owner's Group currently undertaking a special study of the on-line testing intervals in Technical Specifications. Detroit Edison plans to use the results of this study as a basis for requesting/or not requesting changes to the existing on-line testing intervals in the Fermi 2 Technical Specifications. As Detroit Edison gains operational experience with Fermi 2, changes to testing intervals will also be considered, based on this operational experience.

ATTACHMENTS

Operations Procedure - Administrative Number 21,000.03
Post-Scram Evaluation and Re-Start Authorization

Nuclear Operations Directive Number 21
Effective Problem Solving

DRAFT

21.000.03

ENRICO FERMI ATOMIC POWER PLANT
UNIT NO. 2

Type: OPERATIONS PROCEDURE - ADMINISTRATIVE

INFORMATION ONLY

Title: POST-SCRAM EVALUATION AND RE-START AUTHORIZATION

DRAFT

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DRAFT

1.0 Purpose

1.1 The purpose of this procedure is to provide guidelines to the plant operating authority in defining the post-scram data requirements and the criteria for reactor re-start authorization.

2.0 References

*2.1 Administrative Procedure 12.000.10, "Plant Reporting Requirements".

*2.2 Administrative Procedure 12.000.47, "Incident Reporting System".

2.3 Operations Administrative Procedure 21.000.01, "Shift Operations and Control Room".

*2.4 Operations Administrative Procedure 21.000.06, "Documentation of Allowable Operating Transients".

4

3.0 Functions and Responsibilities

3.1 In the event of a Reactor Scram it shall be the responsibility of the Nuclear Shift Supervisor to assure that the Reactor Protection Systems and Reactivity Control Systems have operated properly to place the reactor in the required shutdown condition.

3.2 Following a Reactor Scram, the Nuclear Shift Supervisor or his delegate must notify the On-Call Plant Supervisor and provide information regarding the occurance of the scram and the status of the plant. This notification should be made as soon as practical but no later than thirty (30) minutes after the scram has occurred.

3.3 After the plant has been placed in a safe, stable condition following a Reactor Scram, the Nuclear Shift Supervisor must assure completion of the Post-Scram Data and Evaluation Form (Attachment 1).

4

3.4 If the information recorded on the Post-Scram Data and Evaluation Form indicates that:

3.4.1 The Reactor Protection Systems operated properly.

3.4.2 The Reactivity Control Systems operated properly.

3.4.3 No Emergency Core Cooling Systems were actuated with injection into the reactor vessel.

3.4.4 The initiating scram signal has been identified.

*Denotes "Use" Reference

3.4.5 The reason for the initiating scram signal has been clearly explained.

3.4.6 No automatic initiations which were required to function during the transient, failed to initiate.

Then the Nuclear Shift Supervisor, after consultation with the Shift Technical Advisor, has the authority to order a re-start of the plant.

The order for plant re-start shall be documented by the signature of the on-duty Nuclear Shift Supervisor on the Post-Scram Data and Evaluation Form.

3.5 If the information recorded on the Post-Scram Data and Evaluation Form does not confirm all the items listed in 3.4 above or, if advised by the Shift Technical Advisor that an unreviewed safety question may exist, then a Post-Scram Engineering Review by the Technical Engineer is required Upon completion of the engineering review, a plant re-start must be authorized by the Superintendent - Nuclear Production or his delegate.

The order for plant re-start shall be documented by the approval signature of the Superintendent - Nuclear Production or his delegate on the Post-Scram Engineering Review report prepared by the Technical Engineer.

4.0 Administrative Controls

4.1 It shall be the responsibility of the Operations Engineer or his delegate to review the Post-Scram Data and Evaluation Form for completeness and proper signature application.

4.2 The Operations Engineer or his delegate shall evaluate the data recorded on the Post-Scram Data and Evaluation Form and assure that either the Nuclear Shift Supervisor's authorization to re-start the plant is in accordance with Section 3.4 of this procedure, or, that the Technical Engineer or his delegate has been notified that a Post-Scram Engineering Review is required.

4.3 When required, the Technical Engineer shall conduct a Post-Scram Engineering Review, soliciting the resources from Nuclear Engineering if necessary to determine the cause of the reactor scram. He shall have a Post-Scram Engineering Review report prepared for approval of the Superintendent-Nuclear Production or his delegate.

4.4 At the completion of the Post-Scram Engineering Review, the Operations Engineer, or his delegate, shall assure that the Superintendent - Nuclear Production or his delegate has authorized a plant re-start prior to directing the Nuclear Shift Supervisor to begin a plant startup.

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DRAFT

4.5 The Operations Engineer or his delegate shall assure that the appropriate information derived from the circumstances prior to and following the reactor scram are documented and processed in accordance with References 2.1, 2.2 and 2.4.

4

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POST-SCRAM DATA AND EVALUATION

1.0 Initial Condition Prior to Scram:

1.1 Reactor Mode Switch Position:

Shutdown ☐ Refuel ☐
Startup/Hot Standby ☐ Run ☐

1.2 Reactor Power, ________%.

1.3 Generator Gross Load, ________ Mwe.

1.4 Total Core Flow, ________ M#/hr.

1.5 Reactor Pressure, ________ PSIG.

1.6 Reactor Water Level,________ IN.

1.7 Reactor Recirculation Loop A Flow _______ M#/hr.

1.8 Reactor Recirculation Loop B Flow _______ M#/hr.

1.9 RHR Division I mode/status ________________.

1.10 RHR Division II mode/status ________________.

1.11 Reactor Feedwater Control:

1. Master Control, MAN ☐ AUTO ☐
2. Elements selected, SINGLE ☐ THREE ☐
3. Reactor Feed Pump A, MAN ☐ AUTO ☐
4. Reactor Feed Pump B, MAN ☐ AUTO ☐

1.12 Reactor Pressure Regulator in Service, A ☐ B ☐

1.13 CRD Pump in service, A ☐ B ☐

2.0 Reactor Scram Data:

2.1 Time and Date of Reactor Scram, __________ / __________.

2.2 Control Room NSO on duty, ____________________.

2.3 Initiating Scram signal, ____________________.

2.4 Parameter value at which initiating scram signal occurred, ____________________.

4

POST-SCRAM DATA AND EVALUATION (con't)

3.0 Post-Scram Data

3.1 Did all operable control rods fully insert? YES ☐ NO ☐

1. List Control Rod number and notch for all operable control rods not fully inserted.

Rod ____________, Notch ____________
Rod ____________, Notch ____________
Rod ____________, Notch ____________
Rod ____________, Notch ____________
Rod ____________, Notch ____________
Rod ____________, Notch ____________
Rod ____________, Notch ____________
Rod ____________, Notch ____________

3.2 SRM's fully inserted YES ☐ NO ☐

3.3 SRM Count Rate and *Time:

1. SRM A ________ CPM, ________
2. SRM B ________ CPM, ________
3. SRM C ________ CPM, ________
4. SRM D ________ CPM, ________

3.4 Did any SRV's open? YES ☐ NO ☐

1. List Safety Relief Valve letter, opening mode, lift pressure, and reseat pressure for any SRV's that opened.

Valve ____, Mode ____, lift ____ PSIG, Reseat ____ PSIG
Valve ____, Mode ____, lift ____ PSIG, Reseat ____ PSIG
Valve ____, Mode ____, lift ____ PSIG, Reseat ____ PSIG
Valve ____, Mode ____, lift ____ PSIG, Reseat ____ PSIG
Valve ____, Mode ____, lift ____ PSIG, Reseat ____ PSIG
Valve ____, Mode ____, lift ____ PSIG, Reseat ____ PSIG
Valve ____, Mode ____, lift ____ PSIG, Reseat ____ PSIG
Valve ____, Mode ____, lift ____ PSIG, Reseat ____ PSIG

2. List SRV's which cycled and number of cycles, if known.

____
____
____
____
____
____
____
____

4

*Include date if different from scram date.

POST-SCRAM DATA AND EVALUATION (con't)

3.5 Did any isolations occur? YES ☐ NO ☐

1. List any isolations which occurred by group number.

____
____
____
____
____
____
____
____

3.6 Describe, if any, the actuation of any Safety Systems and the reason for their actuation.

* ____
____
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____
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____
____

4.0 Post-Scram Evaluation

4.1 Did Reactor Protection Systems operate properly? YES ☐ NO ☐

1. If NO, describe what improper operation was observed.

____
____
____
____
____
____
____
____
____

*Use additional attached description if necessary.

4

DRAFT

POST-SCRAM DATA AND EVALUATION (con't)

4.2 Did Reactivity Control Systems operate properly? YES ☐ NO ☐

If NO, describe what improper operation was observed.

4.3 Did any Emergency Core Cooling System actuate and inject into the reactor vessel? YES ☐ NO ☐

1. If YES, describe what system(s) actuated and what signals initiated the actuation.

4.4 Has the initiating scram signal as listed in 2.3 of this attachment been confirmed as the initiating scram signal? YES ☐ NO ☐

1. If NO, describe the reasons for the non-confirmation.

4

DRAFT

POST-SCRAM DATA AND EVALUATION (con't)

4.5 Has the reason for the confirmed initiating scram signal been clearly explained? YES ☐ NO ☐

1. If NO, descfibe the reasons for the non-explanation.

4.6 Did all automatic initations which were required to function during the transient, initiate properly? YES ☐ NO ☐

1. If NO, describe which automatic initiation that failed to function and the corrective action taken.

4

4.7 Describe any plant response which appeared to be abnormal either before, during, or after the scram.

POST-SCRAM DATA AND EVALUATION (con't)

5.0 Post Scram Action

5.1 The review of the data and evaluation sections of this attachment indicate that:

1. No unreviewed safety question exists.

(NSS Initial)

2. The criteria specified in Section 3.4 of this procedure has been met.

(NSS Initial)

3. No transient related plant responses were determined to be abnormal.

(NSS Initial)

4

5.2 Based on the information provided in this Post-Scram Data and Evaluation form and after consultation with the Shift Technical Advisor, authorization is given to re-start the plant.

Shift Technical Advisor   Nuclear Shift Supervisor   Date

5.3 Based on the information provided in this Post-Scram Data and Evaluation form and after consultation with the Shift Technical Advisor, an engineering review is ordered and plant re-start must be authorized by the Superintendent - Nuclear Production.

Shift Technical Advisor   Nuclear Shift Supervisor   Date

6.0 Post-Scram Administration

6.1 The information provided in the Post-Scram Data and Evaluation form has been reviewed and all sections are complete as required.

Operations Eng./delegate   Date

6.2 The Technical Engineer has been notified of the re-start decision in either section 5.2 or section 5.3 of this Attachment and the following documents are attached:

POST-SCRAM DATA AND EVALUATION (con't)

		(check)
1.	Sequence Recorder printout.	______
2.	Process Computer Rod position printout.	______
3.	Copy of the applicable pages of the NSO log.	______
4.	Copy of the applicable pages of the NSS log.	______

4

______________________________ ______
Operation's Engineer/delegate Date

6.3 The Post-Scram Data and Evaluation Form has been forwarded to the Technical Engineer for review and file.

______

END

FOR INFORMATION ONLY

NUCLEAR OPERATIONS DIRECTIVE NO.21

EFFECTIVE PROBLEM SOLVING

PURPOSE

The purpose of this directive is to assure that the cause of a problem is accurately determined and properly resolved prior to continuing a safety-related activity.

GENERAL

It is fundamental to identify a problem before working on its solution. (Detroit Edison provides supervisors and management personnel with training in the use of Kepner-Tregoe problem solving techniques.)

After an incident or apparent problem occurs, no safety-related activity should be resumed until the problem has been identified, its cause determined and a solution formulated and implemented. (Example: In the case of a plant trip, the reason for the trip must be determined by careful analysis of the data. After the problem has been identified, its solution should be formulated and implemented. Startup must be properly authorized before the reactor is again started.)

It is vital that this directive be followed to the fullest extent.

Wayne H. Jens

Wayne H. Jens
Vice President - Nuclear Operations

SER Section: 22, Item I.C.6 SER Page: 22-22

Current SER Discussion

The SER described a draft administrative procedure prepared by Detroit Edison for verifying correct performance of operating activities. This procedure was identified as 21.000.12.

Detroit Edison Comments

The procedure number was changed from 21.000.12 to 12.000.43. The title did not change.

SER Section: 22, Item I.D.1 SER Page: 22-26

Current SER Discussion

The 2nd paragraph references a June 9, 1981 letter.

Detroit Edison Comments

The date of the letter is actually June 4, 1981.

SER Section: 22.2, Item I.G.1 SER Page: 22-27
Apendix C.5, Item A-44 C-13

Current SER Discussion

The current discussion reflects the original commitment by Detroit Edison to conform to the NRC guidance concerning conduct of a station blackout test.

Detroit Edison Comments

Detroit Edison revised its commitment in EF2-65898, dated October 5, 1983. This letter responded to NRC Generic Letter 83-24 and committed to comply with the BWR Owners Group position on NUREG 0737, Item I.G.1 instead of performing the loss of AC power test. This should either be corrected in the SER or a future supplement should make clear that Detroit Edison is no longer committed to perfom the loss of AC power test.

FSAR Section H.I.G.1.3 has also been updated to reflect this revised position as indicated below.

"H.I.G.1.3 Detroit Edison Position

Detroit Edison commits to comply with the BWR Owners' Group position on this issue as detailed in Reference 1. This commitment was contained in Detroit Edison letter EF2-65898 to the NRC dated October 5, 1983, and supersedes all previous commitments regarding this issue.

H.I.G.1.4 Reference

1. Letter from D. B. Waters (BWR Owners' Group) to D. G. Eisenhut (NRC), BWROG-8120, dated February 4, 1981, "BWR Owners' Group Evaluation of NUREG-0737 Requirement I.G.1, Training During Low Power Testing."

100/LIC11/1.55
062084

Wayne H. Jens
Vice President
Nuclear Operations

Detroit
Edison

2000 Second Avenue
Detroit, Michigan 48226
(313) 237-8612

October 5, 1983
EF2 - 65,898

Director of Nuclear Reactor Regulation
Attention: Mr. B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Youngblood:

References:
(1) Enrico Fermi Atomic Power Plant, Unit 2 NRC Docket No. 50-341
(2) Letter H. Tauber to R. L. Tedesco, EF2-53285, dated May 14, 1981, "TMI Task Action Item I.G.1, Special Low Power Test Program for BWR's"
(3) Letter W. F. Colbert to L. L. Kintner, EF2-53430, dated June 3, 1981, "Simulated Loss of AC Power Special Test"
(4) Letter W. F. Colbert to L. L. Kintner, EF2-53825, dated June 22, 1981, "Position on I.G.1 Special Test"
(5) Letter H. Tauber to B. J. Youngblood, EF2-60717, dated February 14, 1983, "Simulated Loss of AC Power Test"
(6) Letter D. B. Waters to U. S. Nuclear Regulatory Commission, Attn: D. G. Eisenhut, BWROG-8120, dated February 4, 1981 "BWR Owners' Group Evaluation of NUREG-0737 Requirement I.G.1, Training During Low Power Testing"
(7) Letter D. G. Eisenhut to all BWR Operating License Applicants, Generic Letter 83-24, dated June 29, 1983, "TMI Task Action Plan Item I.G.1, 'Special Low Power Testing and Training,' Recommendations for BWR's"

Subject: Commitment to BWR Owners' Group Position in Accordance with Generic Letter 83-24

The following is submitted in response to Generic Letter 83-24 (Reference 7) dealing with TMI Action Plan I.G.1, Special Low Power Testing and Training.

Detroit Edison has confirmed an earlier statement (Reference 5) that the station blackout test would pose a risk to drywell equipment. Calculations indicate that the non-safety related equipment in the drywell would reach a limiting temperature (185 degrees F) within approximately three (3) minutes. Therefore, since the test would provide little if any meaningful data but has a high potential of damaging equipment, Detroit Edison will comply with the BWR Owners' Group recommendation to constitute compliance with Item I.G.1. This position has been confirmed in analysis performed for the Susquehannah and LaSalle plants.

As noted on page 20 of Reference 6, Detroit Edison has supported the BWR Owners' Group position as being necessary and sufficient to meet the requirement of Item I.G.1. Detroit Edison reaffirms this position and commits to comply with the Owners' Group position detailed in Reference 6. The special tests so defined will be incorporated into our present programs for preoperational and startup testing and will be handled in the same manner as the other tests and test procedures within this programming in accordance with the requirements of Regulatory Guide 1.68.

It is the position of Detroit Edison that the above commitment supersedes previous commitments made in References 2, 3, 4, and 5. Section H.I.G.1 in appendix H of our FSAR will be revised to replace all previous commitments related to the Station Blackout test with a commitment to comply with the BWR Owners' Group position.

If you have any questions, please contact Mr. O. Keener Earle (313) 586-4211.

Sincerely,

Wayne H Jens

cc: Mr. P. Byron
Mr. M. D. Lynch

SER Section: 22.2, Item II.F.1 SER Page: 22-57

Current SER Discussion

"The applicant has replaced analog effluent monitors with Eberline Company SPING-3 digital monitors on each gaseous effluent discharge point in the Fermi 2 plant."

Detroit Edison Comments

FSAR Appendix H.II, Item H.II.F.1.3 (Amendment 33 - March, 1981) states that the monitors are SPING-3/4 (not SPING-3) as indicated below.

> "Extended range requirements fc- noble gas effluent monitors have resulted in the removal of the existing analog effluent monitors installed in the Fermi 2 plant and replacement with an Eberline Company SPING-3/4 series digital monitor on each gaseous effluent discharge point."

SER Section: 22.2, Item II.F.1 SER Page: 22-64

Current SER Discussion

"The Fermi 2 drywell pressure monitoring system measures pressures in both a "narrow" and a "wide" range. The narrow range instrumentation measures pressures from -280 to +120 inches water gauge and the wide range instrumentation measures pressures from 0 to 250 psig... The response time of the -280 to +120 inch water gauge channel...."

Detroit Edison Comments

The range for the narrow range drywell pressure instrumentation has been revised to be -5 psig to +5 psig. This revision was incorporated into FSAR Section 7.6.1.11.3.1 via Amendment 56 (dated April, 1984).

SER Section: 22.2, Item II.F.1 SER Page: 22-65

Current SER Discussion

"The containment water level monitoring system monitors the water level in the containment between the range from 56 inches above normal water level to 19 feet below normal."

Detroit Edison Comments

The range of the suppression pool water level instruments has been revised to reflect a range of -144 inches to +56 inches (i.e., a total range of 200 inches). This revision is reflected in FSAR Table 7.5-1 (per Amendment 56, dated April, 1984).

SER Section: 22, Item II.K.1.22 SER Page: 22-73

Current SER Discussion

The third paragraph under "Discussion and Conclusions" reads in part:

> "....Reactor vessel heat removal may also be accomplished while the vessel is isolated by operator action to align the residual heat removal system for the steam condensing mode of operation. This also involves remote valve alignments and startup of the residual heat removal service water system."

Detroit Edison Comments

The steam condensing mode of RHR will not be used at Fermi 2. As stated in FSAR Section 5.5.7.3.4, "Detroit Edison has elected to delete this mode of RHR and has removed the associated piping and valves".

SER Section: 22, Item II.K.1.23 SER Page: 22-75

Current SER Discussion

"All of the level sensors are located on spatially separated divisional safety grade instrument racks located in the Reactor Building approximately 150 feet from the drywell wall".

Detroit Edison Comments

The correct distance is 15.0 feet. The value of 150 feet referenced from FSAR Section H.II.K.1.23 is based on the allowable run length. The FSAR will be revised to reflect a value of 15.0 feet.

SER Section: 22, Item II.K.3.3 SER Page: 22-76

Current SER Discussion

"By letter dated May 29, 1981, the applicant has committed to report failure of a safety relief valve to open or close when called upon, within 24 hours by phone, confirmed the first working day following the event by telegraph (or similar transmission) and followed up with a written report in 2 weeks. This written report will be in the form of a License Event Report.

The Detroit Edison annual report to the NRC will list each safety relief valve which is challenged during the year and will include the number of times each is challenged. This is acceptable to us. The Fermi 2 Technical Specifications will include this reporting requirement."

Detroit Edison Comments

The reporting requirements on which our commitment was based have been superseded by revisions of 10 CFR 50.72 and 50.73. Consequently, Detroit Edison informed the NRC via letter EF2-68193, dated June 13, 1984, that FSAR Section H.II.K.3.3 was revised in Amendment 57 (May, 1984) to reflect a 30-day schedule for submittal of the written report (consistent with the new LER rule).

Wayne H. Jens
Vice President
Nuclear Operations

Detroit Edison

2000 Second Avenue
Detroit, Michigan 48226
(313) 586-4150

June 13, 1984
EF2-68193

Director of Nuclear Reactor Regulation
Attention: Mr. B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Youngblood:

Reference: (1) Fermi 2
(2) Letter from Detroit Edison to NRC, EF2-53421, dated May 29, 1981
(3) Safety Evaluation Report for Fermi 2 (SER), NUREG-0798, July 1981

Subject: Reporting Failures of Safety/Relief Valves

Fermi 2 FSAR, Section H.II.K.3.3, presently requires the following:

"Detroit Edison will report a failure of a safety/relief valve to open or close when called upon within 24 hours by phone; the report will be confirmed by telegraph (or similar transmission) the first working day following the event and followed up with a written report in 2 weeks. This written report will be in the form of a Licensee Event Report ...."

This position is consistent with both Reference 2 and Section II.K.3 of Reference 3.

However, as you are aware 10CFR50.73 was recently added to the regulations. Consequently, the above referenced FSAR section will be revised in a forthcoming amendment and the applicable implementing documents (e.g., Technical Specifications, plant procedures) have been or will be revised to reflect a 30-day schedule for submittal of the subject report. This report will be in the form of a Licensee Event Report. In addition, the reporting requirements of the newly amended 10CFR50.72 will apply if the situation warrants.

~~8406200055 PDR~~

The attachment provides the proposed FSAR revision.

If you have any questions, please contact Mr. Keener Earle at (313) 586-4211.

Sincerely,

Wayne H. Jens

cc: Mr. P. M. Byron*
Mr. M. D. Lynch*
USNRC, Document Control Desk*
Washington, D.C. 20555

*With attachment

SER Section: 22, Item II.K.3.18 SER Page: 22-81

Current SER Discussion

"The applicant has committed to a design modification recommended by the BWR Owner's Group Study with regard to automatic depressurization system initiation. The Owner's Group Study was submitted by letter dated March 31, 1981.... The ADS logic is to be modified by the addition of a bypass to the high drywell pressure trip if the reactor water level remains below the low pressure ECCS initiation setpoint for a setpoint for a sustained period....

We have reviewed the applicant's design and compared it with the NUREG-0737 position and classification and conclude that the design is acceptable. By letter dated June 24, 1981, applicant has committed to complete modifications prior to fuel load, which meets our requirement for this item."

Detroit Edison Comments

FSAR Section H.II.K.3.18 was recently revised (per Amendment 55 - March, 1984) to reflect the following:

> "In Detroit Edison letter EF2-53873, dated June 24, 1981, Edison committed to complete and test the modifications before fuel loading. In letter EF2-56943, dated April 26, 1982 (attached), Detroit Edison indicated the GE and the BWR Owner's Group were in the process of identifying the optimum logic design modifications to resolve the conflict between the BWR Emergency Procedure Guidelines and the requirements of Item II.K.3.18. The BWR Owners Group resolution of this issue has been completed and submitted to the NRC....
>
> The descriptions of proposed modifications will be provided on a reasonable schedule after receipt of the NRC's evaluation of the referenced letter...
>
> It is anticipated that the modifications can be completed by the end of the first refueling."

Edison is currently developing a letter that will commit to implement Option 4 of the October 28, 1982, BWROG report on this issue. Implementation involves the installation of an ADS manual inhibit switch, in conjunction with a timer which bypasses the high drywell pressure permissive signal after a sustained low water level indication. This commitment to Option 4 is consistent with other NTOL plants that have received NRC approval to implement this modification.

Harry Tauber
Vice President
Engineering and Construction

Detroit Edison
2000 Second Avenue
Detroit, Michigan 48226
(313) 237-8000

FCN 82-32

For Information Only
NOT PART OF
FSAR REVISION

April 26, 1982
EF2 - 56,943

Mr. R. L. Tedesco
Assistant Director for Licensing
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Tedesco:

References: (1) Enrico Fermi Atomic Power Plant, Unit 2 NRC Docket No. 50-341

(2) Letter from T. J. Dente (BWROG) to D. G. Eisenhut (NRS), "Schedule for BWR Owners Group Compliance with NUREG-0737, Item II.K.3.18, BWROG-8204, dated February 5, 1982.

Subject: Schedule for Detroit Edison's Compliance with NUREG-0737 Item II.K.3.18, "ADS-Logic Modification"

Reference 2 provides the technical basis for a need to re-evaluate the ADS logic change proposed by the Owners Group in response to NUREG-0737, Item II.K.3.18.

GE and the BWR Owners Group are in the process of identifying the optimum logic design modifications which would resolve the conflict between the operator action specified in the Emergency Procedure Guidelines and the requirement of NUREG-0737, Item II.K.3.18.

The Owners Group resolution of this issue is scheduled for completion by September 30, 1982 (Reference 2). Following NRC review and approval of final design descriptions, individual utilities are to select one of these options and proceed with the design description and implementation for its plant.

In view of these developments, Detroit Edison has put the ADS logic change modification (FSAR H.II.K.3.18) on hold.

~~8205030176 PDR~~

Detroit Edison is hereby requesting the NRC to grant an extension on completion of modifications to the ADS logic until the first refueling outage.

Sincerely,

cc: L. L. Kintner
B. Little

For Information Only
NOT PART OF
FSAR REVISION

SER Section: 22, Item III.D.1.1 SER Page: 22-99

Current SER Discussion

"In Section H.III.D.1.1 of Appendix H to the Final Safety Analysis Report the applicant has provided a description of leak reduction measures and of a preventive maintenance and monitoring program for minimizing leaks from the systems outside the containment that would or could contain highly radioactive fluids during serious transient or accident conditions. The applicant has committed to determining actual leakage rates during pre-operational testing at the time of fuel load and reporting the results to the NRC. This meets the full power requirement of Enclosure 2 of NUREG-0737.

We have reviewed the proposed leak reduction, preventive maintenance and leak testing program and find the applicant's program to be in compliance with the requirements set forth in NUREG-0737, and therefore it is acceptable. The Office of Inspection and Enforcement will verify that leakage testing has been completed prior to issuing the operating license."

Detroit Edison Comments

Detroit Edison transmitted a revised description of its leakage reduction program in letter EF2-67742, dated March 27,1984 (attached). As noted in that letter, Edison will complete this program by full power operation, not by fuel load. This is necessary because some systems cannot be tested until the reactor is operating.

Wayne H. Jens
Vice President
Nuclear Operations

Detroit Edison

2000 Second Avenue
Detroit, Michigan 48226
(313) 586-4150

March 27, 1984
EF2 - 67,742

Director of Nuclear Reactor Regulation
Attention: Mr. B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Youngblood:

Reference: Fermi-2
NRC Docket No. 50-341

Subject: Leakage Reduction Program

A revised Leakage Reduction Program is attached for your review. It has been formatted for incorporation into Section H.III.D.1.1 of the Fermi-2 FSAR in a forthcoming amendment. The program description has been revised to more clearly define the program and its implementation. It should be noted that Fermi will be submitting leakage reduction test results after fuel load. This is due to the fact that some systems cannot be tested until the reactor is operating. Consultation with other utilities indicates that this approach has been previously accepted by the NRC.

Should you have any questions concerning the above, please contact Mr. O. Keener Earle, (313) 586-4211.

Sincerely,

Wayne H Jens

cc: Mr. P. M. Byron
Mr. M. D. Lynch

H.III.D.1.1 Primary Coolant Outside Containment

H.III.D.1.1.1 Statement of Concern

Parts 20 and 100 of Title 10 of the Code of Federal Regulations specify radiation limits and guidelines for licensed facilities to ensure the protection of public health and safety. In a power reactor, many systems that may or will contain significant radioactive liquid and/or gas inventories after a serious transient or accident have components located outside containment. At TMI-2, the major radioactive releases appear to have come from leaks in such systems. Leakage from the systems must be maintained as low as practical to prevent releases of significant quantities of radioactive material when the systems are operated. The plant operating staff should know the leakage rate of each system and have positive control over them to ensure the maximum availability of the equipment.

H.III.D.1.1.2 NRC Position

H.III.D.1.1.2.1 Full Power License Requirement

Applicants shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during or after a serious transient or accident to as-low-as-practical levels. This program shall include the following:

(1) Immediate leak reduction

(a) Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.

(b) Measure actual leakage rates with system in operation and report them to the NRC.

(2) Continuing Leak Reduction - Establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at intervals not to exceed each refueling cycle.

H.III.D.1.1.2.2 Dated Requirement

Applicants shall submit the information requested in the "Clarification" section of this position at least 4 months prior to issuance of a fuel-loading license.

This requirement shall be implemented by applicants for operating license prior to issuance of a full-power license. (See Section III.D.1.1 of Ref. 4).

H.III.D.1.1.2.3 Clarification

Applicants shall provide a summary description, together with initial leak-test results, of their program to reduce leakage from systems outside containment that would or could contain primary coolant or other highly radioactive fluids or gases during or following a serious transient or accident.

(1) Systems that should be leak tested are as follows (any other plant system which has similar functions or post-accident characteristics even though not specified herein, should be included):

   a. Residual heat removal (RHR)

   b. Containment spray recirculation

   c. High-pressure injection recirculation

   d. Containment and primary coolant sampling

   e. Reactor core isolation cooling

   f. Waste gas (including headers and cover gas system outside of containment in addition to decay or storage system).

(2) Testing of gaseous systems should include helium leak detection or equivalent testing methods.

(3) Should consider program to reduce potential release paths due to design and operator deficiencies as discussed in NRC letter to all operating nuclear power plants regarding North Anna and related incidents, dated October 17, 1979.

H.III.D.1.1.2.4 Applicability

This requirement applies to all operating license applicants.

H.III.D.1.1.3 Detroit Edison Position

Detroit Edison has developed a Leakage Reduction Program to reduce and maintain leakage to as-low-as-practical from systems outside primary containment that could or would contain highly radioactive fluids during and/or after a serious transient or accident. This program is based on Requirement 2.1.6a of NUREG-0578 (Reference 1) and the requirements of item III.D.1.1 of NUREGs 0660, 0694 and 0737 (References 2, 3 and 4 respectively).

H.III.D.1.1.3.1 Program Scope

Table H.III.D.1.1-1 identifies systems included in the Leakage Reduction Program. Table H.III.D.1.1-2 lists systems to which the Leakage Reduction Program is not applicable and futher provides the justification for their exclusion. Only the systems listed in Table H.III.D.1.1-1 are included in the program.

H.III.D.1.1.3.2 Program Description

The Detroit Edison Leakage Reduction Program includes the following features:

a. A combination of periodic visual inspections on accessible portions of the systems and detailed system walkdowns to identify leakage into secondary containment out of components such as valve stems, pump seals, fittings, relief valve discharge lines, drains, vents and instrument loops. When possible, these inspections are performed with the systems at approximately operating pressure in a normal or test condition.

b. An aggressive maintenance program is utilized to correct identified leakage problems and assign a high priority to leakage related work requests for systems in this program. Essentially all leakage on concerned (i.e., those identified in Table H.III.D.1.1-1) systems will be addressed. These preventive and corrective maintenance measures ensure minimum leakage on a continuing basis.

c. Periodic leak rate testing of systems (those listed in Table H.III.D.1.1-1) and system components such as valves at intervals not to exceed each refueling outage. The general test methods used to determine leakage from systems within the scope of this Leakage Reduction Program are provided in paragraph H.III.D.1.1.3.3.

d. Records are maintained on inspections and tests performed and are used to identify chronic or generic leakage problems in order to implement modifications and/or corrective maintenance measures. These records are also made available to the plant operators.

Approximately about the time full power is achieved, Detroit Edison will have collected the necessary data and will submit to the NRC staff a report of the recorded leakage and preventive/corrective maintenance performed as the direct result of the evaluation of this leakage. The report will also identify general leakage criteria to be applied during the first fuel cycle as the basis for instituting corrective action in the form of preventive maintenance. Prior to the start of the second fuel cycle, Detroit Edison will revise the general criteria to the extent necessary based on the

experience gained during the first operating cycle of Fermi 2. These revised criteria will be used as the basis for the long term leakage reduction/monitoring program for EF-2.

NOTE: In addition to this testing program, system leakage tests will be performed on many of these systems as part of the 10CFR50, Appendix J leakage testing program. The systems and components that are subject to this testing and which comprise the containment boundary are identified in Table 6.2-2 of this FSAR.

H.III.D.1.1.3.3 Test Methods

a) Liquid Systems - Systems or portions of systems that could contain radioactive liquids during and/or after an accident are periodically placed into normal operation or a testing mode. During these test conditions the systems are visually inspected for leakage with all results being recorded. Leakage detected during the periodic visual inspections or the less frequent integrated leakrate test, will be measured where possible, and recorded. Techniques used for leakage measurement include collection into a graduated container and estimation by equating drops per unit of time to a standard volume.

b) Gaseous Systems - For systems or portions of systems that may contain radioactive gases during and/or after an accident, a pressure drop or make-up gas rate test is used. Clean air or nitrogen is used for these tests. When leakage is indicated by a pressure drop or excessive make-up, visual inspection techniques are applied to components during pressurization. The most common method of visual inspection will be the application of leak-detection fluid to suspected points of leakage (i.e., valve stem packings & air pump seals). The application of the helium leak detection method of inspection may be considered for some gaseous systems.

H.III.D.1.1.3.4 Test Procedures

Each system identified in Table H.III.D.1.1-1 will have a surveillance testing procedure(s). These test procedure will contain the following elements as applicable:

a) A description of system and plant operating conditions necessary to conduct each leak test. Test boundaries are identified and include only those portions of the system that could contain radioactive fluids during and/or after an accident. For example, the Core Spray suction piping from the condensate storage tank would not be inspected as this suction line is used for test purposes only and would not contain radioactive fluid during or after an accident.

b) Elaboration of special test methods necessary to supplement general test methods.

c) Data sheets listing the specific areas to be inspected. These data sheets will identify isometric drawing numbers and provide spaces to record inspection results.

H.III.D.1.1.3.5 References

1. U.S. Nuclear Regulatory Comission, TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations, NUREG-0578, July 1979.

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

3. U.S. Nuclear Regulatory Commission, TMI-Related Requirements for New Operating Licenses, NUREG-0694, June 1980.

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

5. ASME Boiler and Pressure Vessel Code, Section XI.

TABLE H.III.D.1.1-1 SYSTEMS OUTSIDE PRIMARY CONTAINMENT THAT COULD CONTAIN HIGHLY RADIOACTIVE FLUIDS

Reactor core isolation cooling
Residual heat removal
  Containment Spray
  Suppression pool cooling
  Low-pressure coolant injection
  Shutdown cooling
Core spray
Reactor water sample
Reactor water cleanup
Combustible gas control
High-pressure coolant injection
Standby gas treatment
Control rod drive discharge headers
Containment sampling system

TABLE H.III.D.1.1-2 SYSTEMS OUTSIDE PRIMARY CONTAINMENT THAT WOULD NOT CONTAIN HIGHLY RADIOACTIVE FLUIDS

System	Comment
RHR fuel pool cooling	Not directly affected by accident.
Standby liquid control	Injects fluid and does not circulate reactor coolant.
General service water/emergency equipment service water	Does not circulate reactor coolant and could become contaminated only due to system leaks.
Reactor building closed cooling water/emergency equipment cooling water	Does not circulate reactor coolant and could become contaminated only due to system leaks.
Condensate storage	Could become contaminated only due to isolation valve leakage.
Demineralized water makeup	Could become contaminated only due to isolation valve leakage.
Torus water management	Isolated during LOCA and not required for accident mitigation.
Control air/station air	Would require system failure.
Fuel-pool cooling and cleanup	Not directly affected by accident.
Main steam lines	Would require failure of MSIVs and failure of MSIV leakage control system.
Feedwater lines	Would require failure of isolation valves.
Drywell cooling system	Uses RBCCW or EECW and is not needed for safe shutdown of plant.
RHR steam condensing	Not required for accident Mitigation
Reactor building floor/equipment	Not required for accident mitigation. Minimizing leakage from systems in Table H.III.D.1.1-1 minimizes input to this system.
Radwaste	Not required for accident mitigation.

SER Section: 22, Item III.D.3.3 SER Page: 22-100

Current SER Discussion

"To take into account the fact that the portable unit will occasionally be out of service, and that situations may well arise when rapid determinations will be necessary at more than one location in an emergency, the applicant has committed to provide a minimum of one such portable unit for each vital area."

Detroit Edison Comments

The SER should be clarified to indicate that portable units will be provided to at least the four vital areas identified in FSAR Section H.III.D.3.3.3 (per Amendment 37-June, 1981).

SER Section: Appendix C SER Page: C-14, 15

Current SER Discussion

In the 5th paragraph on page C-14 and the 1st and 2nd paragraph on page C-15, the term "high pressure core injection" is used.

Detroit Edison Comments

The correct term is "high pressure coolant injection".

SER Section: Appendix C   SER Page: C-15

Current SER Discussion

The 2nd paragraph states that the RCIC system has been upgraded to safety grade quality.

Detroit Edison Comments

Edison believes this statement could be misleading and suggests the sentence be revised to specifically state the upgrade as follows:

> "The reactor core isolation cooling system has been upgraded to automatically restart on level 2 after a level 8 trip and automatically transfer suction from the condensate storage tank to the suppression pool on a low water level signal in the condensate storage tank."

SSER 1 Section: 3.9.3 SSER 1 Page: 3-5

Current SER Discussion

"The staff has reviewed the applicant's procedures used on the design of SRV systems. The SRV discharge piping system has been upgraded to ASME Code Class 2."

Detroit Edison Comments

FSAR Section 5.2.2.6 (Amendment 57 - May, 1984) has been revised, as noted below, to more clearly delineate the code class breaks present on the SRV discharge line.

> "The portion of the lines inside the drywell and the torus are designed and classified as Quality Group B, Category I, QA Level I. (The protion of the lines in the vent line was originally installed as Quality Group D. This portion of the lines has been upgraded to include the requirements of Quality Group B components and is classified as Quality Group D+, Category I, QA Level I.) The T-quenchers are designed and classified as Quality Group C, Category I, QA Level I."

SSER 1 Section: 6.2.7 (2) (b) SSER 1 Page: 6-1, 6-2

Current SSER 1 Discussion

"The applicant proposes to perform hydrostatic testing to determine the leak tightness of isolation valves in the following system valves:

(a) Torus pressure and liquid level instrumentation and torus water management system suction and injection.

(b) Residual heat removal (RHR) minimum flow, RHR heat exchanger relief and thermal relief, steam condensing mode header relief, steam condensing mode test line, RHR heat exchanger vent line, liquid sample return, RHR pump suction and pump suction header thermal relief.

(c) High-pressure coolant injection (HPCI), reactor core isolation cooling (RCIC) and core spray pump suction, core spray pump suction thermal relief, pump discharge header relief, pump minimum flow and pump test line, HPCI, and RCIC minimum flow."

Detroit Edison Comments

In Amendment 51 of the Fermi 2 FSAR, Detroit Edison revised Section 3.1.2.4.5 and supplemented the responses to NRC questions E.5.212.23 and E.5.212.56 to indicate that Detroit Edison has elected not to use the steam condensing mode of RHR and has chosen to remove the equipment necessary for that mode of RHR. The "steam condensing mode header relief, steam condensing mode test line and RHR heat exchanger vent line" mentioned in (b) above have been removed from the plant and are no longer subject to the hydrostatic testing of isolation valves described in the SER.

SSER 1 Section: 6.2.7 (5) SSER 1 Page: 6-3

Current SSER 1 Discussion

"(5) Purge Valve Testing

Drywell air purge inlet and exhaust containment isolation valves will be Type C tested (Appendix J) at least every 2 years. However, recent reports have indicated that resilient seats in the large butterfly valves of the purge system may deteriorate unacceptably. As a result, the staff has determined that more frequent periodic leakage tests are required for butterfly valves which have these resilient seats. The applicant has agreed to test the butterfly isolation valves in the purge penetrations once every 90 days."

Detroit Edison Comments

The draft Fermi 2 Technical Specifications, including the most recent draft dated May 8, 1984, indicate in Surveillance Requirement 4.6.1.8.2 that purge valves with resilient material seals will be tested every 92 days.

SSER 2 Section: 22, Item II.K.3.22 SSER 2 Page: 22-3

Current SSER 2 Discussion

"By letter dated January 6, 1982, the applicant provided description showing how the switchgear instrumentation meets the Instrumentation and Control System Branch (ICSB) position on the inoperability of instrumentation as a result of extreme cold weather, a justification for this instrumentation's nonseismic location, and a description of the quality level for this instrumentation. The applicant also committed to lock the door on the cabinet enclosing the instrumentation. The staff will include in the plant Technical Specifications that the cabinet door be kept locked."

Detroit Edison Comments

As reflected in both the referenced January 6, 1982, letter (EF2-55977) and FSAR Section H.II.K.3.22.4, Edison has committed to incorporate into a procedure the administrative controls required to maintain the subject cabinet locked. Fermi Plant Order EFO-8002, is currently being revised to implement this commitment. In view of this, and the fact that a control room alarm alerts the operator to low temperature in the cabinet, Edison believes a technical specification is unnecessary.

Harry Tauber
Vice President
Engineering and Construction

Detroit Edison

2000 Second Avenue
Detroit, Michigan 48226
(313) 237-8000

2390-016

January 6, 1982
EF2 - 55,977

NRC Ven

Mr. B. J. Youngblood, Chief
U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Division of Licensing
Washington, D. C. 20555

DOCUMENT CONTROL
H19000
... 1982
STATUS
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E41000
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Dear Mr. Youngblood:

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

Subject: Automatic Switchover of the Suctions for the HPCI and RCIC Systems

Detroit Edison has carefully reviewed your recent request for clarification of certain design features incorporated as part of the equipment used for automatic switchover of the suctions for the HPCI and RCIC systems. Attached please find a description of the system which meets the guidance of the ICSB position on freeze protection. Secondly, a discussion and justification of the seismic design adequacy of the level instrumentation is included. Finally, a statement of the quality design basis for the system is provided. Detroit Edison believes this supplemental material addresses the identified technical concerns.

If further clarification is required, please contact Mr. L. E. Schuerman, 313-649-7562.

Sincerely,

Harry Tauber

~~8201110015 PDR~~

016 - 015

CL & DVT
EF2 55977

Enclosure

cc: Mr. L. L. Kintner
Mr. B. Little

2380-017

Attachment to
EF2-55,977

AUTOMATIC HPCI/RCIC SUCTION SWITCHOVER SYSTEM

As discussed in the Fermi 2 Final Safety Analysis Report on page H.II.K.3.22-1, a redundant pair of analog level transmitters (E41-N061B & D) provide an automatic transfer of the RCIC and HPCI suction valves on low condensate storage tank level. The condensate storage tank level instrumentation is designed to meet the ICSB position with respect to freeze protection as described in the following summary:

A single source connection penetrates the tank. This source connection is common to both the analog transmitters which monitor tank level for the purpose of transferring the RCIC/HPCI pump suction and the transmitter associated with the continuous wide range tank level indication provided in the main control room. This equipment is contained within a large insulated steel cabinet (H21-P492) welded directly to the exterior of the condensate storage tank about three feet above ground level. The access doors to the cabinet are locked, and administratively controlled by the plant operating staff. The environment within the cabinet is maintained at a temperature of approximately 80 degrees Fahrenheit by a 100 watt radiant strip heater and a local control thermostat. A temperature sensing device which is independent of the strip heater and its associated control thermostat is also located within the cabinet. This sensor produces a visual and audible alarm in the main control room whenever the temperature in the transmitter cabinet falls below thirty five degrees Fahrenheit. The cabinet temperature control and the low temperature alarm are electrically independent and powered from completely independent and diverse power sources. A failure of either would not affect the ability of the other to perform its function. In order to guarantee the continued performance of the environmental control and monitoring systems, Edison will perform a yearly functional surveillance of the systems prior to the advent of freezing weather.

Edison has justified the non-seismic location of the transmitters used in the suction transfer system based primarily on the degree of conservatism in instrumentation seismic design. The level transmitters used in this transfer application were seismically qualified as described in the licensing topical report NEDO-21617.

Fermi site ground response spectra applicable to a transmitter mounting on the tank located at grade level would fall well below the values used for qualification of the transmitters in the reference document. As a result, the transmitters are expected to operate properly during and after a seismic event. As an added degree of conservatism, a complete failure of the tank and/or transmitter system would result in an automatic suction transfer since the loss of the current signal from either transmitter will cause the trip units (E41-N661B and D) and associated trip relays to transfer the RCIC and HPCI suction valves to the suppression pool. These trip units and relays are located on the first floor of the reactor building (reference C-9) in panel H21-P081. These devices and cabinet are located within the seismically qualified portion of the plant and meet the environmental and seismic qualification requirements for Class 1E electrical equipment.

All of the equipment which accomplishes the automatic suction valve transfer on low condensate tank level is classified as quality level 1. The transmitters were purchased as qualified instruments along with the balance of the transfer system and are included with the trip units and relays in the plant technical specifications since the surveillance requirement includes the entire measurement loop.

SSER 3 Section: 13.3.2.1 SSER 3 Page: 13-3

Current SER Discussion

The last paragraph on page 13-3 states that Canada is not considered to be within the EPZ for planning purposes.

Detroit Edison Comments

A small parcel of land in Ontario, Canada does lie within the 10 mile EPZ and is treated as such. Emergency Procedure EP-290 contains notification requirement of Canadian authorities.

SSER 3 Section: 13.3.2.3 SSER 3 Page: 13-6

Current SER Discussion

Item 5 (a) indicates that Detroit Edison has a mutual assistance agreement with the Cincinatti Gas and Electric Co.

Detroit Edison Comments

This agreement will be canceled as a result of the termination of the Zimmer project.

SSER 3 Section: 13.3.2.14 SSER 3 Page: 13-21

Current SER Discussion

The 7th paragraph on this page states that communication drills will be conducted quarterly with the NRC.

Detroit Edison Comments

10 CFR 50 Appendix E.IV.E.9 (d) requires these drills to be conducted monthly and Detroit Edison has modified the Radiological Emergency Response Plan accordingly.

SSER 3 Section: 22, Item II.E.4.2 SSER 3 Page: 22-6

Detroit Edison Comments

In the sixth paragraph valve VR3-3015 should be inserted after valve VR3-3014 in conformance with Figure 22.1.

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