Wayne H. Jens
Vice President
Nuclear Operations

Detroit Edison

Fermi-2
6400 North Dixie Highway
Newport, Michigan 48166
(313) 586-4150

July 5, 1985
VP-85-0134

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
NRC License No. NPF-53

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

(3) Detroit Edison Letter to NRC, "Detroit Edison Response to NRC Generic Letter 83-28", EF2-66117, dated November 3, 1983

(4) Detroit Edison Letter to NRC, "Clarification of Detroit Edison's Response to Generic Letter 83-28", EF2-72014, dated November 29, 1984

Subject: Detroit Edison Updated Status to NRC Generic Letter 83-28

Detroit Edison is providing updated information concerning commitments made in Reference 3. The subject items are identified below, with proper reference, along with supplemental information on the updated Fermi 2 response.

Item 1: Page 1 of Reference 3, Item 1.1

Fermi 2 Response: The recently issued INPO "Good Practice, OP-211, Post Trip Reviews" document was reviewed by Detroit Edison and its recommendations were incorporated, where appropriate, into Operations Procedure - Administrative, Number 21.000.03, "Post-Scram Evaluation and Re-Start Authorization". This procedure controls the post-scram review program used at Fermi 2. A copy of this procedure is attached to this report.*

A055
1/40

* 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.

Item 2: Page 4 of Reference 3, Item 1.1.3

Fermi 2
Response: Training is currently being provided to those personnel responsible for conducting post-scram reviews at Fermi 2. This training provides familiarization with the techniques used during post-scram reviews including the use of the sequence of events recorders, strip charts, the plant process computer, and other devices providing important information.

Item 3: Page 10 of Reference 3, Item 1.2.4

Fermi 2
Response: Detroit Edison requested in a letter to NRC, EF2-71999, dated November 12, 1984 that the Operating License reflect December, 1985 as the date required for the Emergency Response Information System (ERIS) to be functional. The date was incorporated in the Fermi 2 OL for the Safety Parameter Display System (SPDS) only. The SPDS is substantially functional at this time, including appropriate personnel trained. All that remains for the SPDS to be formally declared functional to satisfy the License Condition is the resolution of a few minor open items concerning the completion of acceptance testing. This is expected to be completed by the end of July, 1985. Detroit Edison expects to have the entire ERIS system functional by December 1985.

Item 4: Page 12 of Reference 3, Item 2.1.1

Fermi 2
Response: Detroit Edison has completed a review of the Reactor Trip System (RTS) components to ensure that these components are appropriately identified as safety-related. These components include all active components of existing plant systems that function to implement a reactor scram. The results of this review indicate that Fermi 2 has in place sufficient administrative controls and procedural practices to meet this position.

Item 5: Pages 12 and 13 of Reference 3, Item 2.1.2.1
Page 14 of Reference 3, Item 2.1.2.2
Pages 19 through 21 of Reference 3, Item 2.2.2.1
Pages 21 and 22 of Reference 3, Item 2.2.2.2
Pages 24 and 25 of Reference 3, Items 3.1.2 and 3.2.2

Fermi 2
Response: Detroit Edison has in place a vendor interface and information program meeting the requirements of Items 2.1.2.1, 2.1.2.2., 2.2.2.1, 2.2.2.2, 3.2.1, and 3.2.2. See also Item 2 of Reference (4). It should be noted that the Vendor Equipment Technical Information Program (VETIP) as defined in the March 1984 NUTAC document is considered a valid response to Section 2.2.2 of NRC Generic Letter 83-28. Detroit Edison has implemented the program as described therein.

Item 6: Page 28 of Reference 3, Item 4.5.3

Fermi 2
Response: Detroit Edison is actively involved in the BWROG Technical Specification group. This group recently completed a special study of the on-line testing intervals in Technical Specifications which is contained in Topical Report NED C-30844, January 1985. Letter BWROG-8505, January 31, 1985, provided the results of the Group's study to the NRC for its review. When the review is complete, 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.

The updated information in this letter is being provided to demonstrate that Detroit Edison has addressed all items of concern and describes the manner in which all commitments made in Reference 2 have been completed.

If you have any further questions on this matter, please contact Mr. O. K. Earle at (313) 586-4211.

Sincerely,

Wayne H. Jens

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

SR
Safety Classification

FERMI 2 PROCEDURE - OPERATIONS - ADMINISTRATIVE

TITLE: POST SCRAM EVALUATION AND RESTART AUTHORIZATION

PROCEDURE NUMBER: 21.000.03

REVISION NUMBER: 8

INFORMATION ONLY

Name of preparer: F. E. Abramson /s/

Technically reviewed by: Guy Reece /s/ Date: 03/07/85

Reviewed/concurred by: W. E. Miller /s/ Date: 05/07/85
Supv - Operational Assurance

Approved by: F. E. Abramson /s/ Date: 05/07/85
Responsible Section Head or OSRO Member/Alt

Further Approval Required for Safety-Related or Superintendent-Designated Procedures:

Recommended by: R. S. Lenart /s/ Date: 05/07/85
OSRO Chairman/Alternate

Approved by: R. S. Lenart /s/ Date: 05/07/85
Superintendent - Nuclear Production

The following approved procedure Change Requests are incorporated in this revision: M3094

This revision |X| does | | does not constitute periodic review.

TABLE OF CONTENTS

1.0 Purpose

1.1 This procedure is to provide a systematic method diagnosing the causes of unscheduled Reactor Scrams, ascertaining the proper functioning of safety-related and other important equipment prior to reactor restart and making the determination that the plant can be restarted safely.

18

2.0 Discussion

2.1 Post-scram reviews establish a consistent, comprehensive and systematic method to determine the causes and conditions associated with reactor scrams. This documented review will help ensure events that may have had an impact on the cause of the scram and subsequent equipment responses are identified and thoroughly understood. The review results will permit a determination to be made as to the readiness of the plant to safely return to operation.

3.0 References

3.1 Nuclear Operations Interfacing Procedures (NOIP) 11.000.49, Document Control and Records Management

3.2 NOIP, 11.000.52, Deviation and Corrective Action Reporting

8

3.3 Plant Operations Manual (POM) Procedure, 21.000.01, Shift Operations and Control Room

3.4 POM Procedure, 21.000.06, Documentation of Allowable Operating Transients

3.5 Institute of Nuclear Power Operations (INPO) Good Practice, OP-211, Post-Trip Reviews

4.0 Definitions

4.1 Cause - The root initiator of an event (usually an equipment malfunction, procedural error or personnel error). When the cause is corrected, the possibility of the event recurring is minimized.

4.2 Diagnostics - A systematic approach to the identification of problems and selection of appropriate measures. This may include a method such as the "K-T" (Kepner-Tregoe) method.

4.3 Post-Scram Data - A collection of information used to conduct an investigation and review of an unscheduled reactor scram. The data includes a completed Post-Scram Data and Evaluation form (Attachment 1), hard-copy recorded data and written statements from personnel involved in the event.

condition as prescribed by the NSS. He shall also aid the NSS is determining the required Technical Specification compliance, applicable procedure compliance, and event reportability.

He shall also sign the Post-Scram Data and Evaluation form (Attachment 1) as being in agreement with the class determination event.

The OE shall also review and sign the Post-Scram Data and Evaluation package before the package is submitted to the Technical Engineer as specified in Section 6.10 of this procedure.

8

5.1.4 Plant Personnel

Plant personnel involved in the unscheduled reactor scram are responsible for providing the NSS and the STA objective comments that describe their observations and their actions relating to the reactor scram. Objective comments regarding the cause of the Reactor Scram are particularily important.

5.1.5 Plant Superintendent - Nuclear Production

The plant superintendent or his delegate is responsible for authorization of a plant restart following any unscheduled Reactor Scrams.

5.1.6 Shift Technical Advisor (STA)

The STA is responsible for compiling the data required on the Post-Scram Data and Evaluation form (Attachment 1). The STA is also responsible with the NSS for the investigation phase of the post-scram review.

5.1.7 Technical Engineer

The Technical Engineer is responsible for generating the necessary documents and reports to ensure that lessons learned from unscheduled reactor scrams are used to improve plant safety and reliability and to transfer in-house experience of generic interest to the industry.

2. The STA shall collect written statements from plant personnel who were involved in or who observed the reactor scram. The written statements should include the following type of information:

   a. plant conditions prior to the reactor scram (for Maintenance and I&C personnel this will include the status of maintenance or testing),

   b. first indication that a problem existed,

   c. individual action as a result of the indication,

   d. subsequent indications and plant response including manual actions,

   e. noted equipment malfunctions or inadequacies,

   f. procedure deficiencies identified during the situation.

3. The STA shall complete the Post-Scram Data and Evaluation form with the aid of the control room Nuclear Supervising Operator (NSO) and the Nuclear Assistant Shift Supervisor (NASS).

   a. The STA shall assign an in-sequence number to each Post-Scram Data and Evaluation form used. The number shall start with the present year and then a sequential number. At the beginning of each new year, the numbering sequence shall be zeroed. (e.g. 85-001, 85-002) 8

6.4 Post-Scram Investigation - The NSS and the STA are responsible for the initial post-scram investigation. The purpose of this investigation is to determine the cause of the scram and to assess the plant readiness to return to operation. The return to operation assessment must include verification of the following:

6.4.1 Systems/Functions

1. The Reactor Protection System operated properly.

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

3. The initiating scram signal has been identified.

6.5 Scram Classification

6.5.1 Based on the results of the analysis and evaluation of the reactor scram and subsequent plant response, the NSS or his delegate and the STA shall classify the scram as one of the following:

1. Class I - The cause of the scram is positively known and has been corrected; all safety-related and other important equipment functioned properly during and after the scram.

2. Class II - The cause of the scram is positively known and has been corrected; some safety-related and/or other important equipment did not function properly. However, the malfunction has been corrected or Technical Specification constraint does not prohibit a reactor start-up.

3. Class III - Any items specified in 6.4.1 of this procedure have not been satisfied or some safety-related and/or other important equipment functioned in an abnormal or degraded manner during the event which has not been corrected or prevents reactor start-up due to technical specifications constraints.

6.5.2 If the NSS and the STA cannot agree on the scram classification, the Post-Scram Data and Evaluation form and associated data will be forwarded to the OSRO for evaluation and classification.

6.6 Notifications

6.6.1 Once the scram has been classified, the NSS or his delegate shall notify the Operations Engineer or his delegate and the Technical Engineer or his delegate to allow them an opportunity to determine thoroughness, technical accuracy and consistency of the scram investigation and sign the agreement with the scram classification.

The notification of the Technical Engineer is in addition to the notification specified in 6.3.2 of this procedure.

6.6.2 If the scram has been classified as a Class III, the OSRO chairman must also be notified.

6.7 Investigation Review

6.7.1 Class I and Class II scrams shall be reviewed by the OSRO during the next regularly scheduled meeting. This review is not required prior to reactor restart.

6.9.2 If the cause of the scram has not been positively identified, the Superintendent - Nuclear Production shall determine if the cause and the circumstances surrounding the cause have been analyzed adequately. He shall ensure adequate measures are taken to prevent repetitive challenges to safety systems during future power operations.

6.10 In-House Operating Experience Review

6.10.1 The completed Post-Scram Data and Evaluation form and all associated data shall be reviewed by the OE or his delegate and sent to the Technical Engineer. The Technical Engineer shall review the data and determine if the event has generic significance to plant safety and reliability. The data should also be evaluated to determine if a requirement exists for additional corrective actions such as procedure changes or design modifications. Information copies of the Post Scram Data and Evaluation form will be forwarded to Nuclear Engineering and the Nuclear Safety Review Group for their review.

The data should also be reviewed to determine if any lessons learned are applicable for incorporation into Plant Staff training or for industry dissemination.

8

6.11 Retention

6.11.1 After the post-scram data review and evaluation has been completed, the data must be forwarded to the Record Center per reference 3.4 for record retention. The Post-Scram Data and Evaluation form with all associated data shall be retained for the life of the plant.

POST-SCRAM DATA AND EVALUATION No.____________

1.0 Reactor Scram Data:

1.1 Time and date of reactor scram, __________ / __________.

1.2 Control Room NSO on duty, ____________________.

1.3 Initiating scram signal, ____________________.

1.4 Parameter value at which initiating scram signal occurred, __________________.

1.5 Turbine trip time/date __________ / __________.

1.6 Recirculation pump runback YES _____ NO _____.

1.7 Recirculation pump trip YES _____ NO _____.

Cause: ____________________________________
____
____
____
____

8

1.8 Were any control stations taken from AUTO to MANUAL YES _____ NO _____.

1.8.1 If YES, specify station and time.

____
____
____
____
____

2.0 Initial Condition Prior to Scram:

2.1 Reactor Mode Switch Position:

Shutdown _____ Refuel _____.
Startup/Hot Standby _____ Run _____.

2.2 Reactor Power, ________%.

2.3 Generator Gross Load, ________ Mwe.

2.4 Total Core Flow, ________ M#/hr.

2.5 Reactor Pressure, ________ PSIG.

POST-SCRAM DATA AND EVALUATION (Continued)

2.6 Reactor Water Level,_______ IN.

2.7 Reactor Recirculation Loop A Flow _______ GPM.

2.8 Reactor Recirculation Loop B Flow _______ GPM.

2.9 RHR Division I Mode/Status _______________.

2.10 RHR Division II Mode/Status _______________.

2.11 Reactor Feedwater Control:

2.11.1 Master Control, MAN _____ AUTO _____.

2.11.2 Elements selected, SINGLE _____ THREE _____.

2.11.3 Reactor Feed Pump A, MAN _____ AUTO _____.

2.11.4 Reactor Feed Pump B, MAN _____ AUTO _____.

2.12 Reactor Pressure Regulator in Service, A _____ B _____.

2.13 CRD Pump in Service, A _____ B _____.

8

2.14 Off normal status of any trains/portions of a safety systems:

		Details
2.14.1	RPS	____________________
		____________________
2.14.2	ECCS	____________________
		____________________
2.14.3	SBGTS	____________________
2.14.4	Emergency Buses/ (Diesels)	____________________
		____________________
		____________________
2.14.5	DC Buses	____________________
		____________________

2.15 Testing/Surveillances in Progress

Test Number	Status/Step

POST-SCRAM DATA AND EVALUATION (Continued)

3.0 Post-Scram Data

3.1 Did all operable control rods fully insert? YES _____ NO _____.

3.1.1 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 SRMs fully inserted YES _____ NO _____.

3.3 IRMs fully inserted YES _____ NO _____.

3.4 SRM Count Rate and *Time:

3.4.1 SRM A __________ CPS, __________
3.4.2 SRM B __________ CPS, __________
3.4.3 SRM C __________ CPS, __________
3.4.4 SRM D __________ CPS, __________

3.5 Did any SRVs open? YES _____ NO _____.

3.5.1 List Safety Relief Valve letter, opening mode, lift pressure and reseat pressure for any SRVs that opened, if known.

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

*Include date if different from scram date.

POST-SCRAM DATA AND EVALUATION (Continued)

3.5.2 List SRVs which cycled and number of cycles, if known.

____
____
____
____
____
____
____
____

3.6 Did any isolations occur? YES _____ NO _____.

3.6.1 List any isolations which occurred by group number.

____
____
____
____
____
____
____
____

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

* ____
____
____
____
____
____
____
____
____
____
____
____
____
____
____

*Use additional attached description, if necessary.

POST-SCRAM DATA AND EVALUATION (Continued)

3.8 Did any process radiation monitors indicate increases in reading?
YES _____ NO _____.

If YES, list parameter and attach recording chart.

____
____
____

3.9 The following hard copy data is attached:

NOTE: Where data is not device recorded, a logged record of observation may be used.

		(check)
3.9.1	Sequence recorder printout.	_______
3.9.2	Process computer rod position printout.	_______
3.9.3	Copy of the applicable pages of the NSO log.	_______
3.9.4	Copy of the applicable pages of the NSS log.	_______
3.9.5	APRM - A, B, C, D, E, F.	_______
3.9.6	Reactor vessel water level.	_______
3.9.7	Recirculation pump section temperature.	_______
3.9.8	Core flow.	_______
3.9.9	Reactor vessel pressure.	_______
3.9.10	Main steam flow.	_______
3.9.11	MTG control valve position.	_______
3.9.12	Drywell pressure.	_______
3.9.13	Torus water level.	_______
3.9.14	Torus water temperature.	_______
3.9.15	LPCI, CS, HPCI, RCIC flow.	_______
3.9.16	Condenser vacuum.	_______
3.9.17	Vessel metal temperature.	_______
3.9.18	Reactor coolant chemistry sample results.	_______

POST-SCRAM DATA AND EVALUATION (Continued)

4.0 Post-Scram Evaluation

4.1 Chronological Series of Events

4.2 Probable Cause of Trip

8

POST-SCRAM DATA AND EVALUATION (Continued)

4.3 Unexpected Aspect of Transient Behavior (If event compared with previous similar transient, note the transient with which compared.)

Compared With

FSAR transient page number____________________

Previous trip on__________/__________

____
____
____
____
____

4.4 Identification of Systems with Inadequate Performance

System/Component	Description of Problem
______________	______________________________
______________	______________________________
______________	______________________________
______________	______________________________
______________	______________________________
______________	______________________________
______________	______________________________

____ / ________
NSS Signature Date Time

____ / ________
STA Signature Date Time

POST-SCRAM DATA AND EVALUATION (Continued)

5.0 Preliminary Safety Assessment

5.1 Transient Data for Pertinent Plant Parameters

		Maximum	Minimum
5.1.1	RCS pressure as measured in steam dome.	Loop A___ B___	Loop A___ B___
5.1.2	Reactor vessel water level.	______in.	______in.
5.1.3	Reactor coolant flow.	Loop A___ B___	Loop A___ B___
5.1.4	Reactor core thermal power.		_______

5.2 Preliminary Safety Assessment

5.2.1	RPV pressure remained above 825 psig.	YES _____ NO _____
5.2.2	Reactor isolation occurred.	YES _____ NO _____
5.2.3	RCS pressure increased to safety/relief valve operating pressure.	YES _____ NO _____
5.2.4	RCS temperature decrease less than 100° F/hr.	YES _____ NO _____
5.2.5	HPCI/RCIC initiated.	YES _____ NO _____
5.2.6	ADS timer initiated.	YES _____ NO _____
5.2.7	Primary containment.	press _____ temp (ave) _____
5.2.8	Torus water.	level _____ temp (ave) _____

5.3 Scram Class

Classify scram as I, II or III according to 6.5 in this procedure.

The scram on ____________________ (Date) at ___ : ___ (Time) is a Class _________ (I, II, III).

POST-SCRAM DATA AND EVALUATION (Continued)

Signature indicates agreement with class.

NSS	Date /	Time
STA	Date /	Time
OE/Delegate	Date /	Time
TECH. ENG./Delegate	Date /	Time

Notification

Superintendent - Nuclear Production notified of event classification.

Person notified	Date /	Time

8

6.0 Permission to Start-up

6.1 Class I, II Events

6.1.1 Superintendent - nuclear production notified and permission granted to start-up the reactor.

Nuclear Shift Supervisor	Date /	Time
STA	Date /	Time
OE/Delegate	Date /	Time

COMMENTS ______________________________

POST-SCRAM DATA AND EVALUATION (Continued)

6.2 Class III Event

6.2.1 OSRO review of event on ____________, meeting number ______________.

6.2.2 Permission is granted to start-up the reactor.

_______________________________ ______/______
OSRO Chairman Date Time

_______________________________ ______/______
Supt. -Nuclear Production Date Time

6.2.3 COMMENTS ____

____

____

____

7.0 OE Review

_______________________________ ______/______
OE/Delegate Date Time

8

POST-SCRAM INVESTIGATION STATEMENT SHEET

Name ______________________________

Time ______________________________

1. Plant conditions prior to the reactor scram (for maintenance and I&C personnel. This will include the status of maintenance or testing.)
2. First indication that a problem existed
3. Individual action as a result of the indication
4. Subsequent indications and plant response including manual actions
5. Noted equipment malfunctions or inadequacies
6. Procedure defficiencies identified during the situation
7. Additional comments

8

______________________________ ____________________
Signature   Time - Date

If additional space is required to answer the above questions, use blank paper and number the answers.