Wayne H. Jens
Vice President
Nuclear Operations

Detroit Edison

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

July 13, 1984
EF2 - 69,210

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

(2) NRC Generic Letter 84-11, "Inspection of BWR Stainless Steel Piping", April 19, 1984

(3) Letter from Detroit Edison to NRC, "Late Response to Generic Letter 84-11", EF2-69109, May 21, 1984

Subject: Response to Generic Letter 84-11

Reference 2 was received late by Detroit Edison as documented in Reference 3. This response complies with the submittal agreement stated in Reference 3.

Generic Letter 84-11 was written from the standpoint of plants which previously had been extensively operated. Since Fermi 2 is an NTOL, many of the staff recommended actions and questions are not directly applicable. However, Detroit Edison understands the significance of Intergranular Stress Corrosion Cracking (IGSCC) and has implemented, during plant construction, many countermeasures which will help to mitigate and minimize IGSCC. In addition, our future plans for inspections and other actions in this area equally take into account the IGSCC concern. Accordingly, in the attachment to this letter, is documented a summary of

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these actions, as well as applicable responses to your specific requests for information.

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

Sincerely,

Wayne H. Jens

Enclosures

cc: Mr. P. M. Byron*
Mr. M. D. Lynch*
Mr. W. Hazelton (NRR-MTEB)*
USNRC, Document Control Desk*
Washington, D. C. 20555

*With Attachment

Mr. B. J. Youngblood
July 13, 1984
EF2-69210

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 13th day of July, 1984, 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, 1937
Acting in Monroe County Michigan

Response to NRC
Generic Letter 84-11
Inspections of BWR Stainless Steel Piping

A. Introduction

Intergranular Stress Corrosion Cracking (IGSCC) has been an issue on BWR's for approximately ten years. Detroit Edison has recognized the significance of the issue and has taken corrective action whenever feasible and prudent. Part B of this response summarizes these actions. Part C discusses in more detail IGSCC countermeasures directly relevant to Generic Letter 84-11. Part D provides responses to the requests for information contained in Generic Letter 84-11. Part E summarizes the overall project position on IGSCC.

B. Summary of IGSCC Corrective Actions Employed at Fermi 2

The following is a summary listing of steps which have been taken or will be taken to minimize IGSCC at Fermi 2. (Appropriate docket references are given for your information.)

1. Removal of recirculation pump discharge valve bypass line and use of 308L clad closure caps on the 4-inch sweepolets. (FSAR Sections 5.2.3.2.1.1, 5.5.1.3)

2. Replacement of stainless steel reactor core spray line safe end with Inconel; the remainder of the line is carbon steel. (FSAR Section 5.2.3.2.1.1)

3. Special controls on field welding of stainless steel pipe including: a) limitations on heat input, b) weld bead straightening, c) internal grinding and d) ferrite content (FSAR Sections 5.2.3.2.1.1, 5.2.5.5)

4. Solution annealing of the 12-inch recirculation system risers and application of a nonsusceptible inlay to the ends. (FSAR Sections 5.2.3.2.1.1, 5.2.5.2)

5. Solution annealing of the stainless steel vessel nozzle safe ends plus welding to RPV after vessel heat treatment (Detroit Edison to NRC Letter, "Implementation of NUREG-0313, Rev. 1", EF2-53472, 6/5/81)

6. Induction Heating Stress Improvement (IHSI) of welds in the recirculation system, reactor water

cleanup system and the residual heat removal system. (FSAR Section 5.2.3.2.1.2 and Detroit Edison to NRC Letter, "Induction Heating Stress Improvement (IHSI) Program on Fermi 2", EF2-61929, 4/11/83)

7. Providing the CRD system with a source of lower dissolved oxygen, providing for increased inspections and replacement of applicable parts. (FSAR Section 5.2.3.2.1.3, E.5.122.3, E.5.212.155)

8. Capping the CRD hydraulic return line. (FSAR Section E.5.212.69)

9. Reducing the preload and increased surveillance of the jet pump hold down beams. (FSAR Section E.5.110.17)

10. Utilization of special procedures for stainless steel (e.g., cleanliness controls, solution heat treatment requirements, material inspection requirements and welding controls). (FSAR Section 5.2.5)

11. Commitment to augmented ISI in accordance with NUREG-0313, Rev. 1 and Section XI of the ASME Code. (FSAR Sections 5.2.3.2.1.4 and 5.2.8.7.)

12. Control of the chemistry of the reactor water to operational limits of: a) Conductivity < 1.0 umho/cm at 25°C; b) chlorides < 200 ppb; and c) pH 5.6 to 8.6 at 25°C. (FSAR Section 5.2.3.2.1.2)

13. Evaluating the advisability of hydrogen water chemistry at Fermi 2.

C. Details of IGSCC Countermeasures Taken at Fermi 2 Applicable to Generic Letter 84-11

At Fermi 2, there are three systems employing stainless steel piping containing 117 circumferential welds where IGSCC may be a concern: recirculation, residual heat removal and reactor water cleanup.

Countermeasures against IGSCC employed at Fermi 2 fall into three major categories: a) Induction Heating Stress Improvement, b) solution annealing, and c) Inconel buttering plus solution annealing of safe ends.

A breakdown of the countermeasures employed is as follows:

Countermeasure		Number of Welds
IHSI		79
Solution Annealing		22
Inconel Buttering and Solution Annealing		12
	Subtotal	113
Unmitigated		4
	Total	117

The countermeasures using solution annealing are expected to remain effective for the life of the plant since no sensitized material will be exposed to reactor water at these welds.

The IHSI treatment is also expected to remain effective for the life of the plant since it was implemented prior to operation. (See Reference 1) Plants in Japan have been operating for approximately 5 years after having performed IHSI. Edison will monitor the applicable performance of these plants and will make adjustments accordingly.

The four unmitigated welds were not accessible for IHSI purposes from a practical standpoint. The welds are in 28-inch piping and have stress rule index values of less than 1. General Electric reports that no cracks have been found in welds with a stress rule index less than 1.05 and that no cracks have been found in similar welds at other plants. These four welds are included in the ISI program and will be inspected on an augmented inspection cycle of 80 months in accordance with NUREG-0313, Rev. 1 (see Reference 2).

D. Specific Responses to Requested Information from Generic Letter 84-11

i) Scope and Schedule of Planned Inspections

Stainless steel piping in the recirculation, reactor water cleanup and residual heat removal piping systems will be inspected (NDE) in accordance with the rules of the applicable edition of ASME Section XI, to the extent possible and within design limitations. Further, welds selected in accordance with the rules of Section XI will receive an increased frequency of examination in accordance with the requirements of NUREG-0313, Rev. 1.

Included in the Fermi 2 ISI program are 42 circumferential welds from the subject piping. Examination of the welds is scheduled to be completed in an eighty-month cycle. The examination of these welds is to be spaced over

the 80 month cycle. Assuming five examination outages, then 8 - 10 welds would be examined during each outage.

In view of the fact that Fermi 2 is a new plant and the IGSCC countermeasures already taken, the above program is considered acceptable when compared with the suggested acceptable reinspection program for older BWR's.

With respect to IHSI, approximately 28% of the applicable welds were subjected to post-treatment PT and UT. The results from these examinations showed no deleterious effects resulting from the treatment. This inspection was conducted in accordance with Proposed Code Case N-333 which specified that a minimum of 25% of the IHSI-treated welds be given surface and volumetric examinations in accordance with Table IWB-2500-1 or IWC-2500-1, as applicable. As discussed with Mr. Warren Hazelton of your Materials Branch, this is sufficient and acceptable for a new plant.

ii) Availability and Qualification of Examiners

All current ISI contractor(s) for Fermi 2 are qualified to the requirements of I.E. Bulletin 83-02. Several NDE contractors have examiners qualified to the requirements of IEB 83-02 at the EPRI NDE Center and these examiners will be available to Fermi 2.

All future contracts for NDE at Fermi 2 will require that personnel be qualified in accordance with IEB 83-02. The requirements of IEB 83-02 will be factored into the ISI-NDE Program Administrative Procedures.

iii) Description of any Special Surveillance Measures, in Effect or Proposed, for Primary System Leak Detection, Beyond those Measures Already Required by Your Technical Specifications

The leak detection measures employed at Fermi 2 either meet the practical limits or conform to the limits specified in Attachment 1 to Generic Letter 84-11 and ASME Section XI, 1980 Edition, Winter 1981 Addenda. No special surveillance measures are either in effect or proposed.

Fermi 2 has designed and installed a leak detection system which is sufficiently sensitive to detect small leaks in a timely manner. This is accomplished through the use of a rate-of-change level instrument.

The Fermi 2 draft Plant Technical Specification for Reactor Coolant System Operational Leakage (3.4.3.2) conforms with the time intervals and periods specified in Paragraph B of Attachment 1 to Generic Letter 84-11. The Fermi 2 draft Plant Technical Specification for the Reactor Coolant Leakage Detection System (3.4.3.1) allows a reasonable time for equipment repair on a system basis. The inherent redundancy and diversity is more than adequate to permit continued operation during the specified repair/maintenance period. (The draft Technical Specifications are attached for your information.) Fermi 2 also currently employs in its Technical Specifications the definition of unidentified leakage contained in Paragraph D of Attachment 1 to Generic Letter 84-11.

In addition, a visual examination for leakage of the reactor coolant piping is performed during each outage in which the containment is deinerted in accordance with Fermi 2 Plant Technical Specification 4.0.5.

iv) Results of the Bulletin Inspections Not Previously Submitted to NRC

Not applicable to Fermi 2.

v) Remedial Measures, if any, to be Taken When Cracks are Discovered

If IGSCC cracks are discovered at Fermi 2, remedial measures will be conducted in accordance with the applicable ASME Code Section XI rules and NRC regulations. It is anticipated that the steps taken will be similar to those defined in Attachment 2 to Generic Letter 84-11.

E. Summary of Fermi 2 Position on IGSCC

Fermi 2 believes that all reasonable steps which could be taken to minimize IGSCC at Fermi 2 at the current time, have been taken. IGSCC initiation in a BWR environment is known to require years of operation, even under adverse stress and environmental conditions. This time will be spent following developing technology and implementation as it becomes available. Detroit Edison supports EPRI and participated in the original BWR Owners Group on pipe cracking. Currently, Detroit Edison is represented on the Plant Materials and NDE Subcommittees of the EPRI System and Materials Task Force.

In the highly unlikely event that throughwall cracking does occur, leak detection systems are sensitive enough to ensure safe shutdown. Any remedial measures will be conducted in accordance with the applicable ASME Code Section XI rules and NRC regulations.

F. References

1. Detroit Edison to NRC Letter, "Induction Heating Stress Improvement (IHSI) Program on Fermi 2", EF2-61929, April 11, 1983

2. Detroit Edison to NRC Letter, "Implementation of NUREG-0313, Rev. 1", EF2-53472, June 5, 1981

ATTACHMENT

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.3.1 The following reactor coolant system leakage detection systems shall be OPERABLE:

a. The primary containment atmosphere gaseous radioactivity monitoring system.

b. The primary containment sump flow monitoring system consisting of:

1. The drywell floor drain sump level, flow and pump-run-time system, and

2. The drywell equipment drain sump level, flow and pump-run-time system.

c. The drywell floor drain level monitoring system.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With only two of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous radioactive monitoring system is inoperable; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The reactor coolant system leakage detection systems shall be demonstrated OPERABLE by:

a. Primary containment atmosphere gaseous monitoring systems-performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

b. Primary containment sump flow and drywell floor drain level monitoring systems-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION TEST at least once per 18 months.

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OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.3.2 Reactor coolant system leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE.

b. 5 gpm UNIDENTIFIED LEAKAGE.

c. 25 gpm total leakage averaged over any 24-hour period.

d. 1 gpm leakage at a reactor coolant system pressure of 1040 ± 10 psig from any reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1.

e. 2 gpm increase in UNIDENTIFIED LEAKAGE within any 4-hour period.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With any reactor coolant system leakage greater than the limits in b and/or c, above, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With any reactor coolant system pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least one other closed (manual or deactivated automatic) (or check*) valve, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With one or more of the high/low pressure interface valve leakage pressure monitors shown in Table 3.4.3.2-1 inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or verify the pressure to be less than the alarm setpoint at least once per 12 hours; restore the inoperable monitor(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

e. With any reactor coolant system UNIDENTIFIED LEAKAGE increase greater than 2 gpm within any 4-hour period, identify the source of leakage increase as not service sensitive Type 304 or 316 austenitic stainless steel within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Which has been verified not to exceed the allowable leakage limit at the last refueling outage or after the last time the valve was disturbed, whichever is more recent.

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The reactor coolant system leakage shall be demonstrated to be within each of the above limits by:

a. Monitoring the primary containment atmospheric gaseous radioactivity at least once per 4 hours,

b. Monitoring the primary containment sump flow rate at least once per 4 hours,

c. Monitoring the drywell floor drain sump level at least once per 4 hours, and

d. Monitoring the reactor vessel head flange leak detection system at least once per 24 hours.

4.4.3.2.2 Each reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1 shall be demonstrated OPERABLE by leak testing pursuant to Specification 4.0.5 and verifying the leakage of each valve to be within the specified limit:

a. At least once per 18 months, and

b. Prior to returning the valve to service following maintenance, repair or replacement work on the valve which could affect its leakage rate.

The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 3.

4.4.3.2.3 The high/low pressure interface valve leakage pressure monitors shall be demonstrated OPERABLE with alarm setpoints per Table 3.4.3.2-2 by performance of a:

a. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

b. CHANNEL CALIBRATION at least once per 18 months.

TABLE 3.4.3.2-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	VALVE DESCRIPTION
1. RHR System	
E11-F015A (V8-2161)	LPCI Loop A Injection Isolation Valve
E11-F015B (V8-2162)	LPCI Loop B Injection Isolation Valve
E11-F050A (V8-2163)	LPCI Loop A Injection Line Testable Check Valve
E11-F050B (V8-2164)	LPCI Loop B Injection Line Testtable Check Valve
E11-F023 (V8-2171)	RPV Head Spray Outboard Isolation Valve
E11-F022 (V8-2172)	RPV Head Spray Inboard Isolation Valve
E11-F008 (V8-2092)	Shutdown Cooling RPV Suction Outboard Isolation Valve
E11-F009 (V8-2091)	Shutdown Cooling RPV Suction Inboard Isolation Valve
E11-F608 (V8-3407)	Shutdown Cooling Suction Isolation Valve
2. Core Spray System	
E21-F005A (V8-2021)	Loop A Inboard Isolation Valve
E21-F005B (V8-2022)	Loop B Inboard Isolation Valve
E21-F006A (V8-2023)	Loop A Containment Check Valve
E21-F006B (V8-2024)	Loop B Containment Check Valve
3. High Pressure Coolant Injection System	
E41-F007 (V8-2193)	Pump Discharge Outboard Isolation Valve
E41-F006 (V8-2194)	Pump Discharge Inboard Isolation Valve
E41-F023 (V8-3705)	Pump Discharge Check Valve Bypass Valve
4. Reactor Core Isolation Cooling System	
E51-F012 (V8-2227)	Pump Discharge Isolation Valve
E51-F013 (V8-2228)	Pump Discharge to Feedwater Header Isolation Valve

TABLE 3.4.3.2-2

REACTOR COOLANT SYSTEM INTERFACE VALVES

LEAKAGE PRESSURE MONITORS

VALVE NUMBER	SYSTEM	ALARM SETPOINT (psig)
E11-F015A & B, E11-F022, F023, E11-F050A & B	RHR LPCI	482 ± 12
E11-F008, F009, F608	RHR Shutdown Cooling	138 ± 3
E21-F005A & B, E21-F006A & B	Core Spray	440 ± 12
E41-F006, F007	HPCI	70 ± 1
E51-F013, F014	RCIC	70 ± 1

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