U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-341/84-62

Docket No. 50-341 License No. CPPR-87

Licensee: Detroit Edison Company
2000 Second Avenue
Detroit, MI 48224

Facility Name: Enrico Fermi Nuclear Power Plant, Unit 2

Inspection At: Enrico Fermi 2 Site, Monroe, Michigan

Inspection Conducted: November 28-30, December 5 and 10, 1984

Inspectors: Anil S. Gautam A. S. Gautam	1/23/85 Date
K. Tani K. Tani	01/23/85 Date
R. Mendez for Z. Falevits	1/23/85 Date
R. Mendez R. Mendez (In-Office Review)	1/23/85 Date
C. Williams for J. Norton	1/23/85 Date
Approved By: Cordell C. Williams C. C. Williams, Chief Plant Systems Section	1/23/85 Date

Inspection Summary

Inspection on November 28-30, December 5 and 10, 1984 (Report No. 50-341/84-62(DRS))

Areas Inspected: Routine, announced inspection by regional inspectors of licensee action on previous inspection findings; implementation of an as built program; and independent inspection of the RHR system relay logic. The inspection involved a total of 88 inspector-hours onsite by four inspectors including 7 inspector-hours during off-shifts, 3 inspector-hours in the Regional Office, and 15 inspector-hours during meetings with the licensee at Region III.

Results: Of the two functional areas inspected no items of noncompliance or deviation were identified in one area. Three items of noncompliance were identified in the second area (failure to implement an as built program which provides timely access to as built design documents in the control room - paragraph 3.b.; failure to assure that design documents and as built conditions are in mutual agreement - paragraph 3.c; failure to control components which do not conform to requirements, failure to establish a procedure - paragraph 3.c).

DETAILS

1. Persons Contacted

Detroit Edison Company (DECo)

#°W. H. Jens, Vice President Nuclear Operations
#W. R. Holland, Vice President
#W. J. Fahrner, Project Manager
°S. H. Noetzel, Assistant Project Manager
#F. Agosti, Manager Nuclear Operations
#°G. M. Trahey, Director Nuclear QA
#°W. M. Street, Supervising Engineer - Civil
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*°L. Bregni, Licensing Engineer
°M. J. Gavin, General Supervisor
°L. G. Ferguson, Resident Engineer I&C
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#R. A. Bryer, Principal Engineer
#°T. J. O'Keefe, Nuclear Engineer
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°R. W. Barr, Supervising Engineer
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R. H. Bense, Engineer
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°E. P. Griffing, Assistant Manager Nuclear Operations
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Sargent and Lundy, Chicago, IL

#V. S. Shastri, Head, Water Res. Division

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NRC Region III

#A. B. Davis, Deputy Regional Administrator
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#R. F. Warnick, Branch Chief, DRP
*#C. Williams, Chief, Plant Systems Section, DRS
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*°#K. Tani, Reactor Inspector, PSS
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*°#A. Gautam, Reactor Inspector, PSS

°Denotes persons present at the interim exit meeting on November 30, 1984.

#Denotes persons present at the meeting with the licensee (Detroit Edison) in Region III on December 5, 1984.

*Denotes persons present at the meeting with Detroit Edison in Region III on December 10, 1984.

2. Action on Previous Inspection Findings

a. (Closed) Open Item (341/84-17-07): This item addressed the seismic bolting of 480V Motor Control Centers (MCCs) in regard to the size and torquing of their anchor bolts. A letter from the manufacturer, Gould Electronics, dated November 9, 1984, was reviewed for seismic data. It was stated by the manufacturer that 5/16" was the correct size of the required seismic bolts and that standard workmanship values for torquing were sufficient for installation. The licensee reported that the industry standard practice for tightening 5/16" bolts was 11 ft lb. Torquing of 5/16" anchor bolts in the field was reviewed and documented by the licensee to be a minimum of 11 ft lbs. and consistent among MCC's in the plant. Based on this review this item is closed.

b. (Closed) Unresolved Item (341/83-07-06): This item was closed on previous report 84-57 dated December 4, 1984, in which this item was identified in error as 341/82-07-06. Report 84-57 also identified block valve V10-2003 in error as VID-2003. This paragraph corrects these errors.

c. (Closed) Open Item (341/84-17-03): This item was closed in previous report 84-45 dated November 19, 1984, where it was identified in error as 341/84-17-02. This paragraph corrects that error.

d. (Closed) Unresolved Item (341/84-35-02): This item concerned discrepancies in the insulation and jacket wall thickness values of coaxial cables received with Receipt Inspection Report (RIR) 4-19-79-3. Further, in a subsequent follow-up inspection it was determined that wall thicknesses of some cables had been calculated instead of being physically measured. The licensee subsequently, obtained four foot cable samples from six of the reels received with RIR No. 4-19-79-3. These samples were sent to the manufacturer where physical measurements were performed. The results of the tests indicate that given the single worst case (three cables out of forty-eight measured less than the minimum acceptance criteria), the cable's calculated characteristic impedance of .2 ohms was within the permissible variance of ± 3.0 ohms for coaxial cables. Based on the above, this matter is considered resolved.

e. (Closed) Noncompliance item (341/84-17-01B): This item concerns an inadequate design review performed on penetration backup fuses DCP T2301E01. Wrong fuse sizes were identified as installed in the fuse cabinets which did not conform to design drawings or design calculations. As a result of the NRC findings the licensee initiated a comprehensive review and replacement program to assure that all safety-related fuses are properly sized and labeled and that the proper fuses are installed in the safety-related components as specified by design calculations spec. 3071-128-EJ.

   Licensee letter EF2-72433 dated October 12, 1984 indicated that specification 3071-128-EJ was issued for exclusive use with fuse applications, and that the specification would provide Nuclear Operations Department with a verified list of fuses to be used for a field inspection prior to fuel load to assure that the proper fuses were installed in the plant's safety-related electrical circuits. The letter also stated that in the future this specification would assure that all fuses are controlled and consistent with design requirements.

   The letter further stated that "It is Engineering requirement that operations will walkdown all fusing that is listed in the referenced specification. The purpose of the walkdown is to compare and verify that the installed fuses agree with the fuse data furnished in the specification. If the installed fuse agrees with the Specification; make, type and size then it need not be changed out. If however, the installed fuses differs from the specification it must be changed out to agree with the specification."

   The licensee indicated that the fuse verification program has been started in November, 1984 and will be completed by fuel load. As of the date of this inspection, two 480V switchgears and three 4160V switchgears had been done. The inspector inspected a sample of the completed work to ascertain whether all fuse specification requirements had been met, that is; fuse identification, size, quantity, manufacturer, and type. The inspector also reviewed the available licensees fuse inspection results which were compiled during the licensee's inspections of the installed fuses in the switchgears prior to implementation of corrective action. The results indicated the following:

(1) 480V switchgear 72E inspected November 14, 1984 contained a total of 42 fuses, 37 of the 42 were found to be "wrong", (not in agreement with specification 3071-128-EJ) and will be replaced.

(2) 480V switchgear 72F inspected November 14, 1984 contained a total of 42 fuses, 37 of the 42 were found "wrong" and will be replaced.

(3) 480V switchgear 72B inspected November 1, 1984 contained 30 fuses, 24 of the 30 were found wrong and will be replaced.

(4) 480V switchgear 72C contained a total of 34 fuses, 31 of the 34 were found "wrong" and will be replaced.

No apparent deficiencies were identified by the inspector with the new fuse program and its implementation.

In view of the comprehensive program being implemented by the licensee to resolve the fuse deficiencies, which appears to be adequately implemented and will be completed prior to fuel load, this item is considered closed.

f. (Closed) Unresolved Item (341/84-14-02): This item addresses qualifications of main turbine main stop valves limit switches 2N30N165C, -166C, -167C, and -168C. Instrument list specifies these position switches as nonseismic QA Level III. These switches transmit a signal to the scram logic of the reactor protection system.

Subsequently the licensee performed a test and evaluation of these switches documented in Engineering Research Report 84E80.

Two tests were performed on limit switch Namco model SLS-4 to demonstrate its suitability for application in the Reactor Protection Trip System.

(1) The limit switch and G.E. HFA relay circuit as a load (simulating real condition) was cycled for 2000 operations at a rate of 1 to 2 seconds per operation. (Contact A-B was used.)

(2) The limit switch was cycled for 2000 more cycles at the same rate using the G.E. HFA relay as a load plus three additional coils in parallel to increase the inductive load to more than 3.5 times the load current of test 1 (i.e., to 1.03 amperes). Contact A-B was used.

Test results indicated no trend toward increased contact resistance. The SLS-4 limit switch remained in excellent condition after both tests, the limit switch was found suitable for the intended application. The inspector reviewed the post Design Basis Accident environmental conditions for the main turbine stop valve limit switches which indicated that a main steam line break resulting in venting to the turbine building would be the only credible event to be considered. Since these switches are on the third floor of the turbine building the temperature of the steam would be significantly less than 175°F where the limit switches' rated temperature is 194°F. Therefore, environmental degradation of these switches was found not to be credible. Based on test data and qualification data submitted by the licensee this item is considered closed.

g. (Closed) Open Item (341/84-45-01): This item addresses the assumed value of six (6) times the full load amps (FLA), used by the licensee's design engineers in sizing of motor starters, when locked rotor amps (LRA) values were not supplied by the vendor. The inspector reviewed letter EF2-103, 512 dated November 8, 1984 which stated that "Electrical Field Engineering has performed a survey on all QA-1 MOV's to assure that the locked rotor current used in design calculation 968 and shown on MCC frontal are vendor values and the vendor was contacted to confirm few of these values. The corrected data was sent to Project Engineering in Troy to be incorporated in Revision B of design calculation 968 and change document was issued from Field Engineering to update the drawings (ABE-1407)."

   Based on the above this item is considered closed.

h. (Closed) Open Item (341/84-45-03): This item concerns possible valve overcycling (5 cycles/minute or 10 cycles/10 minutes) and subsequent failure of contactors for (E11) RHR system motor operated valves. The licensee performed an evaluation on these valves during various modes of operation of the RHR systems. (LPCI mode, Shutdown Cooling mode). The following valves were evaluated for expected duty (cycling) E11-F017A/B; E11-F024A/B; E11-F048A/B. Letter EF2-72295 dated October 26, 1984 documents the results of this evaluation which concluded that the expected number of operations of these valves will be less than 5 cycles/minute or 10 cycles/10 minutes. Based on the licensees analysis this item is considered closed.

i. (Closed) Open Item (341/84-50-03): This item concerns the outgoing electrical interlocks of Reactor Recirculation Extractor Isolation to RHR valve E1150-F009 into redundant control circuits of valves E1150-F015A (Div. I) and E1150-F015B (Div. II). The concern was that malfunction of valve E1150F009 limit switches will compromise the operation of redundant valves E1150-F015A&B. The inspector reviewed the following documents presented by the licensee, addressing the LPCI mode of the RHR system and which allow the loss of both loops of the RHR LPCI mode.

   (1) NUREG-0138 indicates that for those plants where the swing bus design is permitted, the consequences of complete failure of LPCI coincident with LOCA are analyzed, and the results are acceptable. This was accepted because the remaining ECCS could perform the cooling function.

(2) FSAR Table 6.3-1 - emergency core cooling system single failure evaluation,

Assumed Failure	Suction Break Systems Remaining
LPCI Valve	All ADS, 2CS, HPCI

Table 6.3-11, note 'a' states "This table shows the single, active failures considered in the ECCS performance evaluation. Other postulated failures are not specially considered because they all result in at least as much ECCS capability as one of the assumed failures."

(3) The licensee stated that the loss of the LPCI Loops A & B will not prevent the safe shutdown of the unit, using the ADS and the CS systems, as alternate shutdown systems.

(4) GE report NEDO 10139 Paragraph 3.3.2 states that the LPCI system by itself is not required to meet all the requirements of IEEE-279 since it is backed up by the two core spray systems.

Due to the fact that a complete failure of LPCI is acceptable this item is considered closed.

j. (Open) Noncompliance Item (341/84-17-01C): It was previously identified that the licensee did not calibrate safety-related shutdown instruments to the required accuracy of 0.25%, and that the licensee had no program in place for punch listing these safety-related instruments for recalibration to the required accuracy. During the NRC inspection period of November 7th through the 9th 1984, as documented in Report Number 84-57, the licensee had proposed and took corrective action. However, the licensee elected to change the calibration requirements for a significant portion of the subject instruments. NRC Inspectors requested that the licensee provide a comprehensive engineering analysis on the safety-related instruments that the licensee elected not to calibrate to the required accuracy of 0.25%.

(1) During this inspection period, the requested information in Report Number 84-57 (Comprehensive Engineering Analysis) was presented to the inspector. The Engineering Analysis was contained in Detroit Edison Company's Potential Design Change form No. PDC-1784A, dated November 21, 1984, and Engineering Evaluation Request No. EER-298 dated November 8, 1984.

While reviewing DECo's Potential Design Change form No. PDC-1784A, dated November 21, 1984 on the subject of the Relaxation of Accuracy Requirements for Process Instrumentation, the inspector observed the following on the attached Engineering Evaluation Request (EER) #84 - 298: "Detailed the Description of the Problem: The instruments listed on Attachment 1 of this EER are among those which must be "calibrated to the manufacturer's stated accuracy except as approved by Nuclear Engineering prior to fuel load". These transmitters have been identified by Nuclear Production as those which will be particularly difficult to calibrate to (or maintain at) the tolerances dictated by the manufacturer's stated accuracy.

The manufacturer's stated accuracy for these transmitters exceeds that of our most reliable low-range (i.e., less than 15 PSIG) pressure-measurement M&TE when the degradation of the accuracy of those devices due to ambient temperature is considered.

In some cases, the effective accuracy of our pressure-measurement M&TE is further reduced by the fact that the transmitters being calibrated are adjusted to elevated ranges."

EER-84-298 further stated that "Proposed Solution (if any): Relax the calibrated accuracy requirements of those process instruments listed on Attachment 1 to values which are consistent with our practical ability to calibrate them. (See the "Practical Measurement Tolerance" column on Attachment 1 and the "Notes" column on Attachment 2)."

EER-84-298 also stated that "Expected Savings/Benefits: A significant number of manhours will be saved over the life of the plant (both in the field and the metrology laboratory) due to fewer post-calibration failures and the subsequent investigations and process instrument recalibrations required. Also, less chance for error will result if the need for temperature dependent accuracy calculations is eliminated."

The following information was attached to EER-84-298 and it is quoted as follows: "The following is a list of instruments for which problems in calibration are anticipated due to test equipment limitations:"

Instrument Number	Manf's Accy Stmt	Approx Cal Signal Input Range	Manf's Accy* (Eng'g Units)	"Practical" Measurement Tolerance (M&TE Suited For Field Use)
B21-N080A-D	± .2%	74-32 "WCD	± .084 "WCD	± .5 "WCD
B21-N081A-D	± .2%	220-70 "WCD	± .3 "WCD	± 1 "WCD
B21-N085A, B	± .25%	393-021 "WCD	± .48 "WCD	± 1 "WCD
B21-N091A-D	± .25%	220-70 "WCD	± .37 "WCD	± 1 "WCD
B21-N094A-H	± .25%	0-10 PSIG	± .025 PSIG	± .03 PSIG
B21-N095A-D	± .2%	74-32 "WCD	± .084 "WCD	± .5 "WCD
B21-N450, 51	± .25%	0-110 "WCD	± .275 "WCD	± .5 "WCD
B21-N484, 87	± .25%	0-10 PSID	1 .025 PSID	± .03 PSID
B31-N110A-D	± .2%	0-278 "WCD	± .55 "WCD	± 1 "WCD
B31-N112A, B	± .2%	0-10 PSID	± .02 PSID	± .03 PSID

B31-N113A, B	± .2%	0-10 PSID	± .02 PSID	± .03 PSID
B31-N114A, B	± .2%	0-10 PSID	± .02 PSID	± .03 PSID
B31-N115A, B	± .2%	0-10 PSID	± .02 PSID	± .03 PSID
C71-N050A-D	± .25%	0-10 PSID	± .025 PSIG	± .03 PSIG
G33-N041	± .4%	0-141 "WCD	± .57 "WCD	± 1 "WCD
T48-N164A, B	± .25%	0-15 "WCD	± .0375 "WCD	± .1 "WCD
T48-N175A, B	± .25%	0-15 "WCD	± .0375 "WCD	± .1 "WCD
T50-N401A, B	± .25%	-5-±5 PSIG	± .025 PSIG	± .06 PSIA
T50-N406A, B	± .2%	39-239 "WCD	± .4 "WCD	± 1 "WCD

*Also equals the limiting measurement tolerance based on the "greater than or equal to" accuracy criteria found in IEEE 498 and ISA 67.04.

NRC inspector reviewed DECo's letter #F2E-84-0173 dated November 19, 1984, from the Assistant Project Engineer to the lead Plant Engineer, NSPE on the Subject EER #298 which states, "Worst Probable Error effects caused by relaxation of calibrated accuracies have been calculated and found acceptable under normal plant conditions for instruments T48N164A&B, N175A&B and B21N484, N487, N450 and N451."

Licensee's disposition of PDC-1784A discussed above was as follows:

Nuclear Engineering concurs with the above stated solution for the following instruments:

B21-N080A-D, 81A-D, 85A-B, 91A-D, 94A-H, 95A-D
B31-N110A-D, N112, 13, 14, 15 (A,B)
C71-N050A-D
G33-N041

T50-N401A, B and 406, B are display only instruments and FCN to change the FSAR will be generated by Systems Engineering (P. M. Harrigan) to complete this change.

The following instruments require approval by Fermi 2 Engineering (Troy).

B21-N450, 451, 484, 487
T48-N164 A and B, N175 A and B

NE-84-1574 letter generated requesting approval from Troy. Upon receipt of their response, this PDC will be revised to reflect that response.

F2E-84-0173 letter allows for calibration accuracies as specified above. Nuclear production to change these accuracies by approved nuclear production procedures".

It appears from the above that although the licensee has addressed the effect of less M&TE accuracy on some of the safety-related instruments during normal plant operations as a result of the accuracy requirement relaxation, other safety-related instruments required for safe shutdown of the plant, for example B21-N081A, B, C & D and B21-N091A, B, C & D, do not appear to have been addressed for plant accident conditions.

The inspector requested additional information from the licensee. The inspector was subsequently presented with G. E's Field Deviation Disposition Request (FDDR) #KH1-1053 Rev. 0. The inspector reviewed this FDDR at the regional office and observed that some of the values and assumptions used by DECo (Detroit Edison Company) in the Reactor water level instrument calibration calculation for the wide, narrow, fuel zone and shutdown ranges appeared to be misleading and/or inaccurate.

On November 28 through 30, 1984, during a routine inspection followup on the above noncompliance item, the inspector was presented FDDR KH1-1053, Rev. 1, (not an approved document) which the licensee said will supersede FDDR KH1-1053, Rev. 0. While reviewing FDDR KH1-1053, Rev. 1, the inspector observed the following:

(a) Some of the specific volume values (VT1, VT2, and VR3) used in the water level instrumentation calibration calculation in FDDR KH1-1053, Rev. 0 & 1 (using the same temperature and pressure values of vessel, drywell, containment and calibration) did not appear to agree with specific volume values that the inspector read from the steam tables and FDDR KH1-749, Rev. 0. This matter was discussed with the licensee and resolved.

(b) Lower instrument tap value used in the calculation of h_2 for the top of scale on the wide range (517.00") differs from the value given as a calculation basis on sheet 2 of 15 of FDDR KH1-1053, Rev. 0, and sheet 2 of 18 of FDDR KH1-1053, Rev. 1 (366.00"). This is a subject of continuing discussion with the licensee.

(c) Elevation values of condensing chamber, upper instrument tap and fuel zone lower instrument tap used in FDDR KH1-1053, Rev. 0, differs from values in FDDR KH1-1053, Rev. 1, and elevation values in both FDDRs differ from the as-built drawings presented to the inspector (DWS #232-907-Vessel Assembly, Rev. 6, 232-895-General Arrangement Elevation, Rev. 4, 233-308-As-built Dimensions, Rev. 3). This was discussed and resolved with the licensee and its representatives on December 10, 1984.

(d) Instrument zero value used by the licensee in the water level instrument calibration calculation was 366.31", while Design Spec data sheet 22A2919AB, Sheet 28, note #5, states that "All water levels are referenced to instrument zero which is 525.50 inches above vessel zero except fuel zone." This matter was discussed and resolved with the licensee on December 10, 1984.

(3) A meeting was held between the licensee and the NRC staff on December 5, 1984, where the above observations were brought initially to the attention of the licensee. The licensee expressed

willingness to demonstrate to the NRC staff that the assumptions and values used in the calculations are correct. At the end of the meeting, the licensee and NRC staff came to the conclusion that some of the assumptions and values used in the calculations needed to be better defined and the calculations re-evaluated.

On December 7, 1984, the licensee sent additional information to Region III containing justification for the use of 366.31" as instrument zero versus the use of 525.50" as instrument zero and some corrections to values and terms used in the calculations in FDDR KH1-1053, Rev. 1. The licensee also requested a meeting with the NRC staff on December 10, 1984. This matter has been discussed and resolved with the licensee on December 10, 1984.

(3) During the NRC/licensee meeting on December 10, 1984, the licensee indicated that the total loop accuracy of the process instruments that were calibrated are within the allowable value that General Electric had provided in the Design Spec data sheets.

The NRC staff informed the licensee that during review of FDDR KH1-749, Rev. 0, the inspectors noted that the FDDR-KH1-749, Rev. 0 documents that the licensee has made the following significant changes to the Reactor Vessel that affected the nuclear boiler water level instruments, new instrument scales, calibration data and drawings during March 9, 1981:

(a) Change of fuel which changed the top of active fuel (T.A.F.) from 360.5" to 366.31" from vessel zero.

(b) Implementation of the testability retrofit option purchased by the licensee.

(c) Completion of the technical specification margin program which resulted in different water level set points.

When asked by the NRC staff how item (c) above impacts the licensee's conclusion that the process instrument total loop accuracy was still within the water level set point margin provided in the G.E. Design Spec data sheet, the licensee replied that item (c) was an ongoing program that has not been completed yet.

(4) <u>Summary</u>

At the end of the meeting, the NRC staff requested that the licensee provide the following information for review:

(a) Data indicating that water level analytical limit and other water level set points were evaluated following the fuel change that changed the top of activie fuel (T.A.F.) from 360.50" to 366.31".

(b) As-built drawings reflecting the actual elevation values of all instrument taps used in the water level instrument calibration calculations.

(c) The definition of the value "h_2" for top of scale calculation (wide range) as used in FDDR KH1-1053 Rev. 0 and Rev. 1 (see J.(1)(b) above).

(d) The licensee's documented consideration in determining total loop accuracy.

Pending a review of the request information this item remains open.

k. (Open) Noncompliance item (341/84-30-01): This item concerns certain discrepancies and deficiencies in the as-built configuration and quality records of the plant size shore barrier structure. The details of this concern are outlined in paragraph 5 of this report.

3. Review of As-Built Program (Module 37051B)

a. During this review, the licensee made two major presentations on November 28 and December 5, 1984, regarding their implementation of an as-built program. During both presentations, the licensee proposed taking exception to bringing plant design documents up to date with as-built installations found in the field. In lieu of bringing design documents up to date generally, the licensee proposed implementing a "road map" program which would compensate for current deficiencies in the as-built drawings by defining the current accurate information available on each drawing and providing guidance to the accurate source for the missing information. Consequently, the road maps presented by the licensee to the NRC, stated disclaimers to certain essential information normally found on approrpriate lead drawings and provided references to other lead documents for the same information.

After review of the road map program at the site and the region, the following NRC concerns were identified and conveyed to the licensee at the December 5 and 10, 1984 meetings at the Region III office:

(1) Schematic drawings as called out on the road maps took exception to correct HP and fuse sizes. The road map called for accurate information on the fuses to be found in Spec 3071-128; however, it was not clear how a correct fuse size and type could be traced back to a schematic drawing which had more than one fused circuit. A similar clarification was needed in the prescription of correct HP, disclaimed on the schematic drawings.

(2) The road maps confirmed that the limit switch contacts on the schematic drawings were designed per the intent of the the limit switch development. It was not clear if this meant that other switch or relay developments on schematics would not accurately reflect field installed contacts. In the latter situation it was not clear where the correct developments would be prescribed.

(3) Wiring diagrams as called out on the road maps did not accurately reflect spare conductors, terminals, fuse sizes and the physical layout of components. This was considered a significant exception with a potential of error during maintenance. It was also not clear when this information would be updated.

(4) General arrangement drawings per the road maps were listed as having minor relocation and variances on construction details as prescribed on the current as-built drawings. The road maps, however, did not indicate what the extent of these "minor" relocations and variances would be.

(5) The road maps called for wiring drawings to be the lead document for internal wiring connections. The road maps were taken at face value during the November 28-30, 1984 inspection, and a review performed by NRC on the wiring drawings. Three wiring drawings reviewed were found to have discrepancies in regard to internal wiring connections. This appeared to indicate that the road maps did not comply with their own definition of what accurate information could be found on wiring drawings. Further details of discrepancies in wiring connections are noted in paragraph 3.c. of this report.

Based on the above comments, the licensee's proposals and exceptions suggested by the road maps appeared to be unacceptable. During the December 5, 1984, meeting at Region III, the licensee was told by the NRC that in view of the foregoing concerns, it was expected that conventional as-built design documents would be made available to demonstrate the as-built configuration of the plant.

b. A review of the as-built program was performed in the Fermi 2 control room.

The licensee's as-built program in the control room was reviewed for preparation and timely access of as-built documents essential for the safe operation and maintenance of the plant. The following areas were reviewed.

(1) General arrangement drawing 6I721-2003-1, Rev. C, for Combination Operating Panel H11-P601 was reviewed for as-built data against installed panel H11-P601 in the control room.

The following randomly selected indicators and devices were reviewed:

Annunciator windows - 1D1, 1D5, 1D9, 1D17, 1D18, 1D31, 1D48, and 1D94. The inscriptions on these annunciator windows were verified on drawings 5I721-2083-1, Rev. L and 5I721-2083-2, Rev. J. No discrepancies were found.

Insert assembly H11-P601B513 shown on drawing 6I721-2003-1, Rev. E, consisting of indicators, pushbuttons, switches and indicating lights, was reviewed on detail drawing 6I721-2003-19, Rev. G, and 3I721-2975-45, Rev. A, for markings, nameplates, legends, and colors.

Items reviewed on insert assembly H11-P601B513 included items 405, 406, 408, 412, 413, 415, and 418 as listed and described on drawing 3I721-2975-45, Rev. A. No discrepancies were found in the items reviewed.

It was observed during this review that General Arrangement Drawing 6I721-2003-1, Rev. C, referenced window inscriptions to be on drawings of series 3I721, while during the review, inscriptions were actually found on drawings of series 5I721. The licensee reported that the first numeral of the drawing series referred only to the size of the drawing and 'that an engineer would not be able to use an incorrect drawing.' The licensee also reported that they would correct this discrepancy on all applicable drawings.

(2) Connection drawings 6I721-2004-10, Rev. E, and 6I721-2004-17, Rev. D, were reviewed for termination of wires from connector plate J2 to lighted pushbuttons AW and BD. In this limited review, wires from connection points, 1, 2, 3, 4, 5, 7, 8, on J2 shown on drawing 6I721-2004-10, Rev. E, were partially traced to appropriate termination points 1C, INO, INC, and G on pushbutton BD, both pushbutton terminations shown on 6I721-2004-17, Rev. D.

No discrepancies were identified during the review of above two connection drawings.

(3) A review was performed on the measures established by the licensee to control the issuance of documents in the control room area, to be used by plant operators for the safe operation of the plant.

The following deficiencies were identified:

(a) The licensee had failed to establish a complete set of drawings essential for the safe operation of the plant and to be used in the location of the control room.

The unfinalized list of drawings presented by the licensee listed 98 drawings, most of which were determined to be P&ID drawings. The inspector was concerned about the lack of schematic and logic drawings as well as one line drawing prescribing the power distribution to the safety (essential) buses. The licensee reported the final list to include over 200 drawings. A separate set of drawings were also to be maintained on aperture cards in the tagging center, separated by a key card door from the control room. These drawings are to be used by operators and this set of drawings had also not yet been completed, with aperture cards and drawings remaining to be included.

(b) The licensee had failed to establish a controlled location, and required sizes for the drawings in the control room. Those that were available appeared to be randomly stored. The current sets of drawings were found loose with no controlled location and the licensee had not decided if they needed full size drawings.

(c) Drawings currently placed in the control room and associated tagging center were reviewed for posting of design changes.

Six of the eleven drawings reviewed in the tagging center were found to have incorrect revisions. The discrepant drawings found in the tagging center were:

721I-22250-1, Rev. J	Latest correct revision K
721I-22250-5, Rev. H	Latest correct revision I
721I-2205-2, Rev. I	Latest correct revision J
721I-2205-04, Rev. G	Latest correct revision F
721I-2205-05, Rev. I	Latest correct revision J
721I-2205-07, Rev. G	Latest correct revision F

One of seven drawings reviewed in the control room was found to have an incorrect revision. The discrepant drawing was 721I-2336-26, Rev. E, the latest correct revision was F, issued August 28, 1984, three months prior to this NRC review.

(d) During review of the current set of drawings in the control room, it was observed that due to the reduced size and excessive information, some drawings could not be read. For example, P&ID drawing 6M721-2081, Rev. M, was not clear enough to read various valve numbers. Other drawings in the set had similar problems.

The licensee was informed that the above deficiencies (a) through (d) indicated the licensee's failure to define, control and implement an adequate as-built program that provides for the preparation of as-built drawings and related documentation in a timely manner in the control room. These deficiencies also indicated lack of measures taken by the licensee to control the issuance of documents for the control room and assure that documents, including changes, were reviewed for adequacy and distributed to and used at the location where the prescribed activity is performed. These deficiencies are considered to be an item of noncompliance to 10 CFR 50 Appendix B, Criteria VI, Document Control. (341/84-62-01(DRS))

c. The inspector performed an as-built inspection of randomly selected safety-related 480V siwtchgears. This inspection was made to determine if the as-built system conforms to design drawings and is in agreement with FSAR commitments. The inspector also examined the effectiveness of licensee's road maps which were developed in response to NRC inspection findings in the as-built program.

The undervoltage relay circuits were selected for this sample to ascertain whether the as-installed wiring conforms to the applicable wiring diagrams. The inspector requested that the licensee trace conductors point by point for accurate assessment of the installed component wiring.

The following deficiencies were noted:

(1) 480V safety-related switchgear 72E position 1A (Instrument Compartment) shown on wiring diagram 6SD721-2511-43, Rev. H, indicates that undervoltage protective relay device 27-XY contains a jumper between relay terminals 1 and 5. A visual inspection did not indicate this jumper to be installed on the relay. A review of applicable schematic diagram 6I721-2573-44, Rev. F, did not indicate that this jumper is required on this relay.

(2) 480V safety-related switchgear 72F position 1A (Instrument Compartment) shown on wiring diagram 6SD721-2511-50, Rev. K, reflects the electrical connections for undervoltage relays, devices 27ZX, 27YZ, and 27XY. A visual and point-to-point inspection indicated the following discrepancies between the design drawing and the as-installed condition in field.

Wiring as Reflected on Drawing		Wiring as Found Installed	
From Device	To Device	From Device	To Device
PA11	LA5	PA11	PB7
PB7	PC11	PB7	PA11
PB10	--	PB10	PC11
PC11	PB7	PC11	PB10

Based on the discrepancies identified above, the inspector informed the licensee that an apparent deficiency exists with the as-built program pertaining to internal components wiring diagrams which are required to reflect the as-installed configuration of the electrical components. As stated in 10 CFR 50, Appendix B, Criterion V, "Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions procedures or drawings..."

The licensee's road maps did not lead the inspector to the identification of deficiencies mentioned in items (1) and (2) above, and therefore did not serve their purpose adequately.

The inspector informed the licensee that the as-built discrepancies identified in items (1) and (2), indicate that measures were not implemented to assure that the design basis is correctly reflected in the design drawings and that deviations from such design basis are controlled. This is an item of noncompliance in accordance with 10 CFR 50, Appendix B, Criterion V (341/84-62-02).

(3) During this as-built inspection, the inspector identified missing or burned out breaker position status indicating lights in the following 480V and 4160V switchgear cubicles:

(a) 480V safety-related switchgear 72E, cubicles 1C, 2A, 2C, 4B and 5D.

(b) 480V safety-related switchgear 72C, cubicles 1B, 2C, 2D, 3B, 3C, 4B and 4C.

(c) 480V safety-related switchgear 72B, cubicles 2D and 3D.

(d) 480V safety-related switchgear 72F, cubicle 1A.

(e) 4160V safety-related switchgear 64B, cubicles B5, B10 and B11.

(f) 4160V safety-related switchgear 65F, cubicles F5 and F10.

(g) 4160V safety-related switchgear 65E, cubicle E11.

No procedures were available to address this area. A licensee Operations Engineer stated that these indicating lights are being examined during operator walkdowns three times daily.

Based on the findings outlined in item (3) above, the inspector informed the licensee that failure to identify nonconforming conditions is an item of noncompliance with 10 CFR 50, Appendix B, Criterion XV. (341/84-62-03)

d. A meeting was held on December 5, 1984 in the Region III office between the licensee and the Region III staff. The inspectors indicated that based on NRC findings in the electrical discipline during the last nine months, the licensee appears to have had deficiencies in the as-built program, specifically relating to installed equipment versus design drawings and design calculation requirements. The licensee stated that a program to review a sample of wiring diagrams against the installed safety-related equipment had been started and that a comprehensive review of all safety-related wiring diagrams versus the as-installed wirings would be completed in the future. The licensee also indicated that the road maps as presented during the meeting would be modified to include specific information and guidance in the use of as-built lead drawings. The inspectors will review these items on subsequent inspections.

e. The inspector reviewed selected 480V and 4160V switchgear protective relay and breaker settings against relay and breaker setting sheets. The following devices were reviewed:

(1) Settings of ITE-GR5 overcurrent ground shield type GR-5 relays in 480V switchgear 72E positions 2D, 3B, 3C, 3D, 4D, 5C, and 5D. 480V switchgear 72C positions 4B, 4C, 3B, 3C, 3D, and 2D. All were found set at 50A/2.

(2) Settings of ITE breakers type SS4G in 480V switchgear 72E position 2C, 4B and 5B. Short time, ground, ampere tap and long time settings were reviewed against relay setting sheets.

(3) Setting of undervoltage (27) relays in 480V switchgear 72E and 72F and 4160 switchgear 65E and 65F. Pickup voltage and time dial settings were reviewed against setting sheets.

No apparent deficiencies were identified.

4. Review of RHR Separation Criteria

The inspector reviewed the RHR relay logic loop "A" and loop "B" delineated on schematic drawings 6I721-2205-2, Rev. J and 6I721-2205-5, Rev. J, specifically, "High Drywell Pressure and Reactor Low Level 1 Logic." It appears that Division I contacts of relays K7A and K8A are arranged in a parallel configuration with Division II contacts K7B and K8B in loop A logic. Furthermore, relays K7A and K8A contacts are also utilized in the loop B (Div II) of the RHR logic, K7B and K8B relay contacts are likewise utilized in the loop B logic. This arrangement is typically used throughout the RHR and other safety-related systems (i.e., C.S.). It appears that contacts from the same relay are used in loop A (Div I), as well as in loop B (Div II), while IEEE-279 1971 (criteria for the separation of circuits and equipment that are redundant) states in paragraph 4.2 that "Any single failure within the protection system shall not prevent proper protective action at the system level when required." Paragraph 4.6 states "Channels that provide signals for the same protective function shall be independent and physically separated to accomplish decoupling of the effects of unsafe environmental factors, electrical transients, and physical accident consequences documented in the design basis, and to reduce the likelihood of interactions between channels during maintenance operation or in the event of channel malfunction."

FSAR, paragraph 3.12.3.2.3 (Physical Separation) states "Electrical equipment and wiring for the ESF systems are segregated into separate divisions which are designated I and II, so that no single credible event is capable of disabling sufficient equipment to prevent reactor shutdown.... Arrangement and/or protective barriers are such that no locally generated force or missile can destroy both redundant safe shutdown and ESF system functions."

FSAR, page 3.12-11, Amendment 2, dated January 1976 states, "A specific set of separation criteria must be met by the internal wiring of individual operating panels, ... that contain components (control devices and wiring) of both engineered safety feature divisions. Generally, the criteria specify the use of separate terminal boards, spacing of... and wiring to preclude the possibility of fire propagating from one division of wiring to another. Alternatively, separation of a pair of redundant control devices that must be located in close proximity is achieved by totally enclosing the wiring to one of the two devices within a fire-resistant material. In a few specific cases, the criterion for separation within the metallic enclosure is relaxed. This relaxation of the criterion is allowable since an analysis for the particular system shows that the complete failure of the equipment within the enclosure will not compromise the system's redundant counterpart or the redundant power supply (refer to the single failure analysis in G.E. Report NEDO 10139....)"

The inspector reviewed G.E. Report NEDO 10139, dated June 1970, titled, "Compliance of Protection System to Industry Criteria." Pages 49 to 80 address the LPCI mode of the RHR system. A point-to-point comparison of the LPCI system with the requirements of IEEE-279 is presented. Paragraph 3.3.2.1 lists the general functional requirements and their compliance or noncompliance with IEEE-279. The following are the items that do not comply with IEEE-279:

Item (4)(g) Malfunction - Tolerant to any single component failure to operate or command, but selected shorts can prevent proper loop selection.

Item (4)(i) Fire - Tolerance to most single wireway fires or mechanical damage.

Item (4)(k) Missiles - Tolerance to any single missile destroying no more than one pipe or wireway or cabinet except for injection valve control.

It further states that the LPCI system cannot (by itself) comply to items (4)(g), (4)(i) and (4)(k) as any of these malfunctions or events can disable the LPCI injection valve which is required to operate.

The licensee also indicated that since the battery system is an ungrounded system, a short to ground at one point will not affect the system.

It appears to the inspector that the G.E. analysis addresses this subject. Since the entire LPCI system can be lost with no effect on safe plant shutdown, this concern is resolved. Further, NRR has reviewed this matter.

5. Discussion of Shore Barrier Structure Issues. (NRC III/Lic Meeting December 5, 1984)

   (1) Summary of Meeting

      (a) The specific item addressed by the Region III inspector was the shore barrier structure.

      (b) A chronology of events regarding the shore barrier was presented. A summary of NRC major concerns was included which are briefly stated as follows:

         1 The structure is not in accordance with as-built drawings.

         2 Significant deviations (externally and internally) from design exists.

         3 Deviations from design were documented by QC inspectors, but appropriate corrective action was not taken.

         4 Soft clay defined as unsuitable in the specifications exists in the foundation.

      (c) It was decided that a meeting with NRR personnel would be necessary to address the issues. The meeting will be arranged for the week of December 9, 1984, if possible.

      (d) Following the December 5, 1984, general meeting with the main body of DECo personnel, a technical meeting was held with the following individuals present:

J. Norton, USNRC, Region III, Reactor Inspector
R. Landsman, USNRC, Region III, Reactor Inspector
J. McCormick-Barger, USNRC, Region III, Reactor Inspector
V. Annambhotla, Sargent & Lundy, Hydraulic/Hydrologic Section
*W. Fahrner, DECo, Project Manager of Fermi 2
G. Trahey, DECo, Director of NQA
R. Bryer, DECo, Principle Engineer
*W. Street, DECo, Supervising Engineer/Civil

*Messrs. Fahrner and Street were present for about 20 minutes at the beginning of the meeting.

(e) Several items of NRC concern were discussed during the technical meeting. DECo agreed to furnish additional information related to several technical areas to Region III (NRC) for review. This data will be conveyed to Region III prior to the planned meeting with NRR.

(2) Chronology of Shore Barriers Structure Events

(a) August 1983 - Report No. 83-19

Licensee action was reviewed on Open Item 81-10-01. The licensee committed to performing a survey (SER, pg. 2-12) to assure that the shore barrier construction was in conformance with the design prior to the issuance of an operating license. Visual inspection of the structure revealed several areas of the surface configuration apparently out of design tolerance.

(b) DECo issued DDR No. C-12154 during the NRC inspection (on August 18). The item was left open pending dispositioning of the DDR.

(c) July 1984 - Report N. 84-30.

DDR C-12154 was reviewed. The DDR was dispositioned "use as is", concurred in by the DECo Supervising Civil Engineer. Certain survey data (accumulated during construction) and a letter/report from design engineer, R. Noble, which was from Noble's inspection dated March 10, 1981, during final construction stage, were attached to the DDR.

(d) In the review (Noble's inspection), the Region III inspector noted that certain apparent structure deviations were not accurately and comprehensively recorded. In response to the NRC inspector's concern, the site survey crew was brought in to take cross sections at 2 selected stations. The check revealed deviations on the order of 3 feet in elevation from design values. A violation was written (84-30-01) based on 5 points of deficiencies and omissions in the dispositioning of DDR C-12154.

(e) The Duke CAT team also recommended (July 1984 report) that an engineering evaluation be accomplished to determine the significance of the deviations and how they relate to the structure's intended design function.

(f) On July 11, 1984, the engineer who designed the structure, Mr. Ron Noble, evaluated the structure (See letter/report of July 11, 1984 from Ron Noble and associates).

(g) Detroit Edison answered the violation of 9/10/84. The licensee's an rs did not appear comprehensive. The root cause of the problem and potential effects were not addressed. The NRC is currently in the process of pursuing this issue.

(h) Sargent & Lundy was retained by DECo to perform an independent evaluation of the ability of the structure to perform its design function (SLS No. 2668, Nov. 5, 1984).

(i) NRC Inspection 84-62 began November 28, 1984 and is presently ongoing. Apparent findings to date include, but are not limited to, the following:

<u>1</u> Structure (internally and externally) deviates significantly from design.

<u>2</u> QC inspectors documented deficiencies during construction that no corrective action was taken on. Moreover, QA inspector qualification files were apparently not available on site for review.

<u>3</u> Soft clay defined as unsuitable in the specs exists in the foundation. Pending review and discussion of these issues with NRC staff and further from the licensee, this matter remains open.

(See Paragraph 2.k. for status.)

6. <u>Exit Interview</u>

The inspectors met with the licensee's representatives (denoted in Paragraph 1) on November 30, December 5 and 10, 1984, and summarized the scope and findings of the inspection. The licensee acknowledged the statements made by the inspectors and agreed to take corrective action on all of the outstanding items of concern.