U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No. 50-316/88019(DRS)

Docket No. 50-316 License No. DPR-74

Licensee: American Electric Power Service Corporation
Indiana and Michigan Power Company
1 Riverside Plaza
Columbus, OH 43216

Facility Name: D. C. Cook Nuclear Plant, Unit 2

Inspection At: D. C. Cook Site, Bridgman, Michigan

Inspection Conducted: June 21, 24, 28, 30; July 7, 13, 14, 28; and August 1, 4, 9, 17, 24, and 25, 1988

Inspectors: for J. M. Jacobson — 9/1/88 Date

J. A. Gavula — 9/1/88 Date

for J. F. Schapker — 9/1/88 Date

Approved By: D. H. Danielson, Chief, Materials and Processes Section — 9/1/88 Date

Inspection Summary

Inspection on June 21, 24, 28, 30; July 7, 13, 14, 28; and August 1, 4, 9, 17, 24 and 25, 1988 (Report No. 50-316/88019(DRS))

Areas Inspected: Special, announced safety inspection of previous inspection findings concerning design control and welding (92701); review of the ISI program, procedures, and results for the reactor vessel UT examination (73051, 73052, 73753, 73755); review of procedures, qualifications, and in-process work activities associated with the Steam Generator Replacement (37702).

Results: No violations or deviations were identified. Based on the results of the inspection, the inspectors reached the following conclusions:

- Considering the licensee actions, the concern related to design control is apparently resolved; however, additional issues relating to welding control and documentation were identified.
- The reactor vessel ISI was found to be in accordance with applicable codes, standards, and licensee commitments.
- The Steam Generator Replacement Project is progressing ahead of schedule with minimal problems.

DETAILS

1. Persons Contacted

American Electric Power Service Corporation (AEP)
Indiana Michigan Power Company (I&M)

W. Smith, Plant Manager
*J. White, Site Project Manager
B. Rarrick, QA Supervisor
M. Bruce, Engineering Supervisor
T. McElroy, Plant Project Engineer
T. Ossmundsen, Construction Supervisor
J. Winckel, Construction QA Coordinator
*S. Brewer, Manager, Nuclear Safety and Licensing
T. Postlewait, Technical Engineering Supervisor
J. Kauffman, Construction Manager
S. Heinecke, Project Manager
G. Caple, Administrative Compliance Coordinator
J. Sampson, Safety and Assessment Superintendent
R. Tella, Maintenance Engineer
E. Morse, Safety and Assessment Supervisor
J. Kobyra, Engineering Manager
D. Powell, Metallurgist
P. Barrett, Director, QA
J. Droste, Maintenance Superintendent
T. Beilman, I&C Superintendent

*Denotes those attending the exit meeting on August 25, 1988.

2. Licensee Action on Previous Inspection Findings

(Open) Violation (316/88012-01): Failure to execute adequate design control in that a thorough engineering review was not documented with regard to Condition Report 12-04-85-792. This condition report documented the inadvertent use of a welding electrode not specified in the procedure. Calculation No. DCCPV12PWOIN Revision 0, April 13, 1988, was performed to evaluate the use of this welding electrode. The NRC inspector reviewed this calculation and noted the following:

- The formula for calculating the longitudinal pressure stress was incorrect. This was apparently a transcription error since the stresses were eventually shown to be correctly calculated.

- The material maximum allowable stress for 308L was supposedly taken from USAS B31.1, 1967 Edition. No such material exists in that reference nor in any later editions of that code.

Based on the above comments, the calculation was revised and resubmitted for NRC review. AEPSC Calculation No. DCCPV12PW01N, "RFC-DC-12-2665", Revision 2, July 1, 1988, was reviewed and the following comments were noted:

- The design pressure of 2475 psi for the CVCS Crosstie System is inconsistent with the attached Boron Injection System. At least a portion of the CVCS crosstie will see the charging pump design pressure of 2550 psi. However, additional reviews indicated that the maximum discharge pressure for the centrifugal charging pumps is 2740 psi and safety valves on the boron injection tank are set for 2735 psi. Additional work by the licensee is required to evaluate the potential discrepancies in the design bases of the CVCS crosstie modification.

- The maximum design basis earthquake stress shown on Sheet 2 of 6 of calculation Reference 3 is invalid. The maximum stress point was not utilized in the calculation causing an approximate error of 2000 psi. Although the resulting stresses were still within code allowables, additional reviews by the licensee may be required to determine the extent of this type of problem.

Members of the Region III staff attended a presentation at the D.C. Cook site on July 28, 1988. This presentation was intended to demonstrate to the NRC, corrective actions taken by the licensee, both on this specific example and with respect to assuring complete engineering reviews in general.

With regard to the use of a welding electrode not specified in the procedure, the licensee reviewed 247 completed work packages covering the time frame 1982 till present. This sampling resulted in finding 112 welding related discrepancies. The majority of the discrepancies appear, at this time, to be related to failure to follow procedures and failure to provide accurate documentation. The NRC inspector reviewed the documentation for the CVCS cross-tie modification (RFC-DC-12-2665) and found similar discrepancies. The licensee is currently evaluating this issue.

With regard to the issue of assuring complete, well documented engineering reviews in the future, the licensee effected changes to the applicable AEP and D.C. Cook site procedures. The NRC inspector reviewed AEP General Procedures 3.1 and 15.1 and the site procedure PMP 5040.MOD.004, and found them to adequately address this issue.

3. Steam Generator Replacement Project

a. Review of Procedures

In preparation for the steam generator (SG) replacement operations, the following MK-Ferguson procedures were reviewed:

Welding Procedure Specification (WPS):
WPS N-8-8-A, Procedure No. FWP-15.01
WPS M-8-8-BF, Procedure No. FWP-15.02
WPS M-8-8-AB, Procedure No. FWP-15.03
WPS M-1-1-AB, Procedure No. FWP-15.04
WPS M-1-1-BF, Procedure No. FWP-15.05
WPS M-8-8-BF, Procedure No. FWP-15.02
WPS M-3-3-AB, Procedure No. FWP-15.06
WPS N-3-3-C1, Procedure No. FWP-15.07
WPS M-3-1-AB, Procedure No. FWP-15.09
WPS M-1-1-B, Procedure No. FWP-15.11
WPS N-1-1-C, Procedure No. FWP-15.12
WPS M-3-1-B-F, Procedure No. FWP-15.13
WPS M-1-8-AB, Procedure No. FWP-15.15
WPS M-1-1-A, Procedure No. FWP-15.16
WPS SPOT-1, Procedure No. FWP-15.17
Cad welding, Procedure No. SQP-10.8

In addition, the following MK-Ferguson work packages were reviewed:

Reactor Coolant Pipe Cuts	1220
Reactor Coolant Pipe Installation	1430
Main Steam and Feedwater Installation	1531
SG Restraint Removal (Temporary)	1532
SG Support Removal (Temporary)	1210
Interference Installation - Mechanical	1740
Interference Installation - Structural	1720
Lower Assembly Preparation	1442
Interference Removal - Mechanical	940
Interference Removal - Structural	920
Pipe Restraint Installation	1421
SG Support Installation (Temporary)	1422
Main Steam and Feedwater Pipe Cuts	1120
Concrete Removal	1600
Steam Drum Removal	1140

All procedures and work packages reviewed were found to comply with applicable codes, standards, and licensee commitments.

b. Cutting and Weld Prep of Piping

The mechanical and thermal cutting and weld prep machining was performed by Power Cutting Inc. Portions of the cutting operations were observed by the NRC inspector and found to be in accordance with Work Package Nos. 1220 and 1120. In an effort to assure that adequate material remained on the coolant elbows, a decision to perform the thermal cut on the carbon steel SG nozzle was made. Since the carbon steel is clad with stainless steel, a technique for plasma cutting was developed. The NRC inspector observed portions of the plasma cutting development. To facilitate performing the coolant loop pipe weld prep operation, templates fabricated off of the new lower assemblies (LA) were used to precisely identify the

nozzle locations. Portions of the weld prep machining operation were observed by the NRC inspector.

c. Concrete Removal

The concrete SG enclosure cutting operation was performed by Trentec. The NRC inspector observed portions of the core drilling and concrete cutting and found the operation to be in accordance with Work Package No. 1600. After the concrete removal, rebar placement anomalies were noted by the licensee. The licensee is currently evaluating the rebar location issue. The NRC inspector observed portions of the concrete chip-back operation and found it acceptable. The NRC inspector expressed some concern associated with the concrete dust and debris collecting in the pool where control rod components are temporarily stored.

d. Lower Assembly (LA) Preparation and Handling

The NRC inspector observed portions of the LA off-loading, preparation and handling. The off-loading of the LA's and transportation to the Aux Building went smoothly and was performed in accordance with Work Package Nos. 1440 and 1441. During the preparation of the LA's, which included the removal of all hydro caps, the staging and temporary enclosure on the lower end of one of the LA's caught fire. The fire was caused by sparks from the torch cutting operation on an adjacent LA. The NRC inspector examined the LA after the fire was extinguished and found no evidence of heat damage. The licensee's evaluation of the fire concluded that no degradation of the LA occurred. The Aux Building cranes and containment polar crane performed well during the lifting operation. The NRC inspector observed portions of the rigging and movement of the LA's. The removal and transportation of the old LA's to the storage vault was observed in part by the NRC inspector and found to be in accordance with Work Package No. 1230.

The NRC inspector reviewed the ASME Code Data Reports for two of the new LA's. The shell sections were fabricated by Westinghouse (Board Nos. 37 and 38). The channel heads (Board Nos. 893 and 892) were fabricated by Ishikawajima-Harima Heavy Industries. All documentation appeared to be in order.

e. Welding

The NRC inspector reviewed the qualification records for the following welders:

Name	Stamp Number
Sawdo	B3
Mattner	B5
Hook	I1
Womack	I4
Stapleton	P4
Hunter	P6

All qualification records were found to be in accordance with MK-Ferguson Procedure FWP 2.1 and ASME Section IX.

The NRC inspector also observed portions of the cadweld training and qualification effort. This operation was found acceptable and in accordance with MK-Ferguson Procedure SQP 10.8, "Cadwelding".

The in-process radiographs of the loop welds on steam generator (SG) Nos. 1 and 4 were reviewed. The cold leg weld on SG No. 1 required repair due to lack of fusion at location 7-8. The hot leg weld on SG No. 1 displayed ID indications which require further blending. The NRC inspector found the contractor's radiographic evaluations to be conservative.

4. <u>Inservice Inspection (ISI)</u>

a. <u>General</u>

The NRC inspector reviewed the ISI plan for the Ultrasonic examination (UT) of the Unit 2 reactor vessel. A relief request applicable to this examination was also reviewed. The licensee applied for and was granted exemptions to certain requirements of 10 CFR 50.55a(g) to allow common start dates for the next inspection interval for Units 1 and 2. Unit 2's first interval was shortened by two years to accommodate the common start dates for the second interval. The change of dates resulted in a need for a relief request for volumetric examinations of the reactor vessel for Unit 2. The Unit 2 reactor vessel examination for the first ten-year inspection interval was conducted during this outage, maintaining the 1974 Edition with Addenda through Summer 1975 of the ASME B&PV Code Section XI as the governing code.

b. <u>Review of ISI Plan, Procedures, Personnel Qualifications and Equipment Certifications</u>

The NRC inspector reviewed the ISI plan to perform mechanized ultrasonic examinations of selected areas of the reactor pressure vessel and adjacent piping and remote visual examinations of selected reactor pressure vessel internals and interior surfaces of the reactor vessel.

The following Southwest Research Institute procedures were reviewed:

- NDT-700-11 "Mechanized Ultrasonic Inside Surface Examination of Ferritic Vessels Greater than 2.5 Inches in Thickness", Revision 9.
- NDT-700-11 "Mechanized Ultrasonic Inside Surface Examination of Ferritic Vessels Greater than 2.0 Inches in Thickness", Revision 10.
- NDT-700-10 "Mechanized Ultrasonic Examination of Austenitic Pressure Piping Welds", Revisions 5 and 6.

- NDT-900-2 "Visual Examination of Nuclear Reactor Internals by Direct or Remote Viewing", Revision 12.

The following UT Calibration standards were inspected:

Identification	Drawing	Vessel weld
11-CSCL-5-DCC	D-3378055(A)	Upper Shell
NS-CSCL-7-DCC	D-3378058(B)	Nozzle Bore
IR-CSCL-8-DCC	D-3378059(A)	Inlet IRS

The calibration standards comply with the requirements of ASME Section V, Article IV.

The NRC inspector reviewed the qualification certifications of the Southwest Research Institute (SWRI) ultrasonic and visual examination personnel. Certifications verified the SWRI inspectors were qualified to SNT-TC-1A as required by applicable code requirements.

c. Observation of Work Activities

The NRC inspector observed the UT mechanized scanning of the reactor vessel circumferential welds and nozzle-to-shell welds, data acquisition, and data evaluation in progress. The UT method for the reactor vessel welds performed from the ID surface was as follows:

(1) The circumferential, meridional, and longitudinal welds were examined from the vessel wall with search units producing 0-degree longitudinal waves (UTOL) for detection of laminar reflectors which affect interpretation of the angle-beam results.

(2) The circumferential, meridional, and longitudinal welds were examined from the vessel wall with search units producing 0-degree longitudinal waves (UTOW) for detection of reflectors in the weld and adjacent base material.

(3) The circumferential, meridional, and longitudinal welds were examined from the vessel wall with search units producing 45 ±2-degree and 60 ±2-degree shear waves (UT45 and UT60) for detection of reflectors oriented parallel with the weld and adjacent base material.

(4) The circumferential, longitudinal, meridional, and nozzle-to-shell welds were examined from the vessel wall with search units producing 45 ±2-degree and 60 ±2-degree shear waves (UT45T and UT60T) for detection of reflectors oriented transverse to the weld and adjacent base material.

(5) The circumferential, longitudinal, meridional, and nozzle-to-shell welds were examined from the vessel wall

with 50/70-degree refracted longitudinal waves to detect underclad cracking and flaws in the near surface area of the weld and adjacent base material. All scanning was done parallel with the beam component direction for the circumferential, meridional, and longitudinal welds.

(6) The inlet nozzle-to-shell welds were examined from the nozzle bore utilizing search units producing 6-degree refracted longitudinal waves and 45 ±2-degree shear waves for the detection of reflectors in the weld and adjacent base material.

(7) The outlet nozzle-to-shell weld and the integral extension areas were examined from the nozzle bore utilizing search units producing 10-degree longitudinal waves and 45 ±2-degree shear waves for the detection of reflectors in the weld, adjacent base material, and the nozzle integral extension region.

(8) The inlet nozzle inside radius section were examined from the nozzle bore with 50/70-degree refracted longitudinal waves to detect underclad cracking and flaws in the near surface area between the shell and nozzle bore tangent points (High Strain Region) as defined by Section XI, 1974 Edition with Addenda through Summer 1975.

(9) The outlet nozzle inside corner region was examined from the nozzle bore with 50/70-degree refracted longitudinal waves to detect underclad cracking and flaws in the near surface area between the shell tangent point and the point along the nozzle bore (M-N-O-P) as defined by Section XI, 1983 Edition with Addenda through Summer 1983.

(10) The inlet and outlet nozzle-to-safe end welds were examined from the nozzle bore with search units producing 0-degree longitudinal waves (UTOL) for detection of laminar reflectors which affect interpretation of the angle-beam results.

(11) The inlet nozzle-to-safe end welds were examined from the nozzle bore with search units producing 45 ±2-degree and 60 ±2-degree shear waves (UT45 and UT60) for detection of reflectors oriented parallel with the weld and adjacent base material.

(12) The outlet nozzle-to-safe end welds were examined from the nozzle bore with search units producing 45 ±2-degree shear waves for the detection of reflectors oriented parallel to the weld and adjacent base material.

(13) The vessel-to-flange weld was examined from the vessel seal surface with search units producing -3.5-degree, +4-degree, and +12.5-degree refracted longitudinal waves for detection of reflectors in the weld and adjacent base material.

The NRC inspector had discussions with the NDE inspectors concerning UT methodology, inspection parameters, calibration, and data analysis throughout the inspection. The NRC inspector observed calibration of the mechanized UT system, observed "A" scans during the data acquisition, and observed the remote visual examination of the reactor vessel core barrel.

The ultrasonic examination and visual examinations observed by the NRC inspector of the reactor vessel fulfills the requirements of Reg Guide 1.150 for inservice examination and provides adequate assurance that no significant degradation of the reactor vessel and internals has occurred.

5. Exit Interview

The Region III inspector met with the licensee representatives (denoted in Paragraph 1) at the conclusion of the inspection on August 25, 1988. The inspector summarized the purpose and findings of the inspection. The licensee representatives acknowledged this information. The inspector also discussed the likely informational content of the inspection report with regard to documents or processes reviewed during the inspection. The licensee representatives did not identify any such documents/processes as proprietary.