U. S. NUCLEAR REGULATORY COMMISSION

REGION III

Report No.: 50-461/90023(DRS)

Docket No.: 50-461 License No.: NPF-62

Licensee: Illinois Power Company
500 South 27th Street
Decatur, IL 62525

Facility Name: Clinton Power Station

Inspection At: Clinton Site, Clinton, IL 61727

Inspection Conducted: November 7-8, December 12-13, 1990, and January 9-10, 1991

Inspector: D H Danielson for J. F. Schapker 1/18/91 Date

Approved By: D H Danielson 1/18/91
D. H. Danielson, Chief Date
Materials and Processes Section

Inspection Summary

Inspection on November 7-8, December 12-13, 1990, and January 9-10, 1991 (Report No. 50-461/90023(DRS)).

Areas Inspected: Routine announced inspection of inservice inspection activities including review of program (73051), procedures (73052), observation of work activities (73753), data review and evaluation (73755), and review of the licensee's action on open items and Part 21 reports (92701).

Results: Of the areas inspected, no violations or deviations were identified. During the course of the inspection, the following were noted:

- The licensee adequately demonstrated the ability to properly implement the inservice inspection (ISI) program including Generic Letter (GL) 88-01, Augmented Inspection of Intergranular Stress Corrosion Cracking (IGSCC) Susceptible Materials.

- Licensee and contractor ISI personnel were knowledgeable, utilized state of the art equipment and were qualified to applicable Code requirements.

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DETAILS

1. Persons Contacted

Illinois Power Company (IP)

*J. Perry, Vice President
*F. Spangenberg, Manager, Licensing and Safety
*J. Cook, Manager, Clinton Power Station
*S. Bell, Supervisor, Inservice Inspection (ISI)
*R. Phares, Director, Licensing
*J. Palchar, Manager, Nuclear Planning and Support
*R. Wyatt, Manager, Quality Assurance
*D. Gill, Manager, Nuclear Training
*J. Miller, Manager, Nuclear Safety Engineering Department
*R. Kereutes, Director, Nuclear Safety Engineering Department
*K. Moore, Director, Plant Technical
*S. Rassr, Director, Plant Maintenance
*K. Graf, Director, Plant Radiation Protection
*H. Nodinc, Supervisor, Procedures
*S. Huntington, Supervisor, Maintenance Services
*J. Sipek, Supervisor, Regional Regulatory Interface
D. Anthony, Level III, Nuclear Safety Engineering
M. Baig, Project Engineer, Nuclear Safety Engineering Department
T. Wilmoth, Supervising Specialist, Nuclear Safety Engineering Department
T. Elwood, Licensing Specialist, Nuclear Safety Engineering Department

EBASCO Services, Inc. (EBASCO)

J. Harrison, Level III, UT Inspector
D. Robbins, Level II, UT Inspector
B. Focer, Level II, UT, MT, PT Inspector
B. Lukawsky, Level II, UT, MT, PT Inspector

Rockwell International Corporation (RIC)

R. Hardy, Level III, UT
W. Johnson, Level III, UT
R. Macshell, Level III, UT
C. Richards, Level III, UT

U. S. Nuclear Regulatory Commission (NRC)

*P. Brochman, Senior Resident Inspector
F. Brush, Resident Inspector

*Denotes those attending the exit meeting conducted on January 10, 1991.

Other members of the plant staff and contractors were contacted and interviewed during the course of this inspection.

2. Followup of Open Items (92701)

a. (Closed) 10 CFR Part 21 Item (4§2/86006-PP): Two motor-operated valves failed during preoperational testing of the High Pressure Core Spray System (HPCS). Valve No. 1E22-F010 experienced a sheared stem and valve No. 1E22-F011 experienced a separation of the stem from the disc. The original valve stems supplied by the manufacturer had high hardness values with resultant high residual stress, typical of Type 410 stainless steel.

The licensee's corrective action included hardness testing of all 166 safety-related valves with Type 410 stainless steel stems or check valve pins. Fifteen valves were found to have stems or pins with a hardness in excess of the General Electric recommendation. General Electric (GE) performed a safety evaluation which demonstrated that a common mode failure of the valves in question would not impact safe shutdown and accident response functions.

Since Type 410 material cracking requires a combination of high hardness and high applied stress, GE had concluded that it was acceptable to leave the fifteen Type 410 items in service until the first refueling outage.

During the first refueling outage, the licensee replaced the following valve stems which exceeded GE's recommended hardness values:

	Valve No.	
(1) 1E22-F001	(6) 1E22-F015	(11) 1E12-F041C
(2) 1E22-F004	(7) 1E51-F010	(12) 1E51-F066
(3) 1E22-F010	(8) 1G33-F040	(13) 1B21-F032B
(4) 1E22-F011	(9) 1SX-105A	(14) 1E12-F050B
(5) 1E22-F012	(10) 1E21-F006	(15) 1B21-F032A

The NRC inspector reviewed the licensee's work packages for the above valve stem replacements and concluded the licensee had taken the appropriate corrective action and replaced the suspect valve stems with new valve stems with GE's recommended material properties. This item is closed.

b. (Closed) Open Item (461/88003-01): Generic Letter (GL) 84-11 response did not commit to the leakage detection and leakage limits described in Action Item Number 4 which states:

A. The leakage detection system shall be sufficiently sensitive to detect and measure small leaks in a timely manner and to identify the leakage sources within practical limits. Particular attention should be given to upgrading and calibrating those leak detection systems that will provide prompt indication of an increase in leakage rates.

Other equivalent and/or local leakage detection systems will be reviewed on a case-by-case basis.

B. Plant shutdown shall be initiated for inspection and corrective action when any leakage detection system indicates, within any period of 24 hours, an increase in rate of unidentified leakage in excess of 2 gpm or its equivalent, whichever occurs first. For sump level monitoring systems with a fixed-measurement interval method, the level shall be monitored at 4-hour intervals or less.

C. At least one of the leakage measurement instruments associated with each sump shall be operable, and the outage time for inoperable instruments shall be limited to 24 hours or immediately initiate an orderly shutdown.

D. Unidentified leakage should include all leakage other than:

(1) leakage into closed systems, such as pump seal or valve packing leaks that are captured, flow metered, and conducted to a sump or collecting tank, or

(2) leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operations of unidentified leakage monitoring systems, or not to be from a through-wall crack in the piping within the reactor coolant pressure boundary.

E. A visual examination for leakage of the reactor coolant piping shall be performed during each plant outage in which the containment is deinerted. The examination will be performed consistent with the requirements of IWA-5241 and IWA-5242 of the 1980 Edition of Section XI of the ASME Boiler and Pressure Vessel Code. The system boundary subject to this examination shall be in accordance with IWA-5221."

The licensee's response to Item 4 of GI 1 stated:

"Clinton Power Station's Code Class 1, 2, and 3 pressure boundary piping meets the guidelines of Part III of NUREG 0313, Revision 1. CPS Technical Specification limits on unidentified leakage are sufficiently restrictive to ensure timely investigation of unidentified leakage."

The licensee's Technical Specification requires:

Leak Detection and Leakage Limits

The licensee's technical specifications requires reactor coolant system leakage be limited to:

(1) No Pressure Boundary Leakage

(2) 5 gpm Unidentified Leakage

(3) 25 gpm Identified Leakage (averaged over any 24-hour period)

(4) 0.5 gpm leakage per nominal inch of valve size up to a maximum of 5 gpm from any reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1, at rated reactor pressure.

Action required:

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

(2) With any reactor coolant system leakage greater than the limits in (2) and/or (3) 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.

(4) 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 two other closed manual or deactivated automatic valves, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Surveillance requirements

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

(1) Monitoring the drywell atmospheric particulate and gaseous radioactivity at least once per 12 hours,

(2) Monitoring the drywell floor and equipment drain sump level and sump flow rate at least once per 12 hours,

(3) Monitoring the drywell air coolers condensate flow rate at least once per 12 hours, and

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

The NRC inspector previously informed the licensee (NRC Inspection Report No. 50-461/88003) that the licensee's Technical

Specifications (TS) requirements did not fulfill the guidance addressed in GL 84-11, Action Item No. 4. The licensee concurred with this finding and committed to submit a revision to the TS which reflects the GL 84-11 guidance. Subsequently, the NRC issued GL 88-01 which superceded GL 84-11. However, the guidance for leak detection and leakage limits remained the same.

The licensee is preparing changes in the TS to satisfy the GL 88-01 guidance. This TS change submittal is scheduled to be in place by April 1991. The NRC inspector reviewed the proposed TS change which includes the following:

Inserts for Technical Specification 3.4.3.2 (pg. 3/4 4-13)

(LIMITING CONDITION FOR OPERATION)

e. No greater than a 2 gpm increase in UNIDENTIFIED LEAKAGE within a 24-hour period or less during OPERATIONAL CONDITION 1

(ACTION)

d. With any reactor coolant system UNIDENTIFIED LEAKAGE increase greater than 2 gpm within any 24-hour period or less (during OPERATIONAL CONDITION 1), within 4 hours from the time of discovery isolate the source of increased leakage or verify that the source of increased leakage is not associated with service sensitive Type 304 or 316 austenitic stainless steel; otherwise be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

This item is currently being reviewed by NRR. The NRC inspector will review this item in a future inspection during routine GL 88-01 reviews.

3. Inservice Inspection (ISI) (73051, 73052, 73753, and 73755)

a. General (73051)

This was the second outage of the first period in the first ten year plan. The ISI Plan conforms to the requirements of the ASME Section XI, 1980 Edition, Winter 1981 Addenda. The services of an Authorized Nuclear Inservice Inspector (ANII) were procured and the ISI procedures and personnel certifications have been reviewed by the ANII.

The licensee contracted Rockwell International (RIC) to perform the reactor vessel mechanized ultrasonic examination (UT), and EBASCO, Inc., to perform UT, Liquid Penetrant (PT), Magnetic Particle (MT), and Visual (VT) examinations of the remainder of the planned ISI examinations. The licensee's inspection plan of intergranular stress corrosion cracking (IGSCC) susceptible components complies with the guidance provided in GL 88-01.

b. Review of Procedures (73052, 73753)

The NRC inspector reviewed the following nondestructive examination procedures:

Rockwell International Procedures

- 205ISI000001, Revision 0 - ISI-UT Examination of Boiling Water Reactor (BWR) Vessel Shell Welds.
- 205ISI00000, Revision 0 - ISI-UT Examinations of BWR Nozzle Inner Radius Section.

EBASCO Procedures

- CPS-UT-W81, Revision 0 - UT Examination for Reactor Vessel Nozzle Inner Radius.
- CPS-MT-W81-1, Revision 0 - MT Exam of Welds and Bolting.
- CPS-PT-W81-1, Revision 0 - PT exam - Solvent Removable Method.
- CPS-VT-W81-1, Revision 0 - VT-1 Visual Examination.
- CP-UTRF-1, Revision 1 - Performance of RF Waveforms for KrautKramer USK Series UT Scopes.
- CPS-UT-W81-1, Revision 0 - UT Exam of Class 1 and 2 Piping Welds Similar and Dissimilar Materials.
- UT-CP-2, Revision 3 - Procedure for Inspection System Performance Checks.
- CPS-UT-W81-P2, Revision 0 - Automated UT Exam of Piping (P-Scan Detection).
- CPS-UT-W81-3, Revision 0 - UT Exam of Class 2 Vessel Welds Less Than 2".
- CPS-UT-W81-4, Revision 0 - UT Exam for the Detection of IGSCC.
- CPS-UT-W81-7, Revision 1 - UT Exam of Pressure Retaining RPV Studs 2" or Greater With Bore Holes.
- CPS-UT-W81-10, Revision 0 - UT Manual Exam of Class 1 RV Welds Covered by RG 1.150.

- CPS-UT-W81-12, Revision 0 - UT Exam for Detection of Cracking in Alloy 182 Nozzle Weldments.

- NDE-1, Revision 13 - Procedure for Training, Exam and Certification of NDE Personnel.

c. <u>Review of ISI Data, Material, Equipment, and NDE Personnel Certifications (73753, 73755)</u>

The NRC inspector reviewed the following documents and determined the applicable Code and QC requirements were met:

- ISI, nondestructive examination (NDE) reports.

- UT instruments, transducers and couplant certifications.

- PT penetrant, cleaner and developer certifications.

- MT equipment calibration.

- NDE personnel compliance to SNT-TC-1A certification requirements and EPRI certifications for IGSCC examinations where indicated by GL 88-01.

d. <u>Observation of Work and Work Activities (73753)</u>

The NRC inspector observed the following work activities in progress:

- Ultrasonic examination of reactor vessel shell welds performed by RIC.

- Ultrasonic examination of reactor vessel nozzle safe end welds performed by Ebasco utilizing automatic (P-Scan) and manual (A-Scan) procedures.

- Magnetic particle and ultrasonic examination of Residual Heat Removal (RHR) pipe to pipe nozzle welds.**

- Liquid penetrant examination of reactor vessel nozzle safe end weld.

- Visual examination of reactor vessel (RV) internals.*

<u>*Visual Inspection of RV Steam Dryer</u>

During the previous refueling outage, a crack was discovered on the rightside vertical weld of Steam Dryer Drain Channel #8. The crack started approximately 1/4" from the bottom end of the weld and extended about 6 7/8" upward. This crack was evaluated and approved for one additional cycle of operation per GE FDDR LH1-5842, Revision 0. A repair of the cracked area was recommended to be performed during refueling outage (RF) 2.

The licensee requested that GE Nuclear Energy provide criteria to allow evaluation of the crack for continued operation following RF 2. The criteria was provided per a letter, GE Nuclear Energy to L. H. Larson titled "Clinton 1 Steam Dryer Drain Channel Cracking Criteria," dated November 2, 1989. This letter specifies a crack length during RF-2 inspection should be less than 11.5" and anything greater than or equal to 11.5" should be repaired.

Visual examination of the cracked area was performed on October 25, 1990, to determine the overall length, the amount of growth during the second cycle, and the general condition of the weld. The results are as follows:

1) Crack length = 7 1/2"
2) The 1/4" ligament at the bottom of the weld (noted during RF-1) is now cracked through.
3) The upper extension of the crack is approximately 3/8" longer than RF-1.
4) There is more separation of the channel plate from the skirt.

Steam Dryer Evaluation

The crack on Steam Dryer Drain Channel #8 has grown approximately 5/8" in length. The 1/4" bottom area is cracked completely through plus the 3/8" additional extension on the upper end of the crack.

The additional plate separation is apparently due to the cleavage of the last 1/4" weld segment at the bottom. The area of separation should be completely submerged during power operation.

Total measured length is now 7 1/2". The 8" specified in FDDR LH1-5842 was apparently a conservatism used to account for measurement uncertainties during the RF-1 inspection. Since the measured length of the RF-2 inspection is still less than 8", the crack growth data given in the criteria letter should be applicable to RF-3 and beyond.

The licensee plans to inspect the steam dryer during the next refueling outage to assure crack growth has been arrested. The steam dryer is not safety-related and the present condition does not pose a safety concern to the RV internals. The NRC inspector will observe the visual inspection during the next refueling outage.

**The magnetic particle examination procedure the licensee utilized allowed the examination to be performed through paint. The NRC inspector verified that the licensee demonstrated the procedure to the ANII who approved the procedure. The NRC inspector cautioned the licensee that the sensitivity of the examination can be reduced performing MT examination through paint and that criteria for the paint thickness, type of paint, and methods for determining this criteria should be part of the examination procedure. The NRC inspector's observation of MT in progress did not include welds with paint except for paint in as-welded bead valleys.

Work activities were performed in accordance with approved procedures, utilizing calibrated NDE equipment, and certified personnel. Detection and resolution of flaws detected by NDE procedures were completed in accordance with ASME Code and regulatory requirements.

No violations or deviations were identified.

4. Exit Meeting

The inspector met with licensee representatives (denoted in Paragraph 1) at the conclusion of the inspection on January 10, 1991. The inspector summarized the scope and findings of the inspection activities. The licensee acknowledged the inspection findings. The inspector also discussed the likely informational content of the inspection report with regard to documents or processes reviewed by the inspector during the inspection. The licensee did not identify any such documents/processes as proprietary.