UNITED STATES
ATOMIC ENERGY COMMISSION
DIVISION OF COMPLIANCE
REGION I
970 BROAD STREET
NEWARK, NEW JERSEY 07102

201 645

June 11, 1971

J. B. Henderson, Chief, Reactor Construction
Branch, Division of Compliance, HQ

CONSOLIDATED EDISON COMPANY OF NEW YORK
INDIAN POINT NO. 3
Docket No. 50-286

The enclosed report of the inspection of Consolidated Edison's Indian Point No. 3 reactor construction site, conducted on May 24, 1971, by R. F. Heishman is forwarded for information.

This inspection was limited to a review of the documentation relative to the refurbishing of the polar crane and inspection of the reactor pressure vessel following the lifting incident which occurred on January 12, 1971.

Records pertaining to the NDT testing of components, replacement of components, and reassembly of the units were reviewed and appeared to be comprehensive and complete. Load testing of the crane in accordance with test procedures was completed on May 23, 1971.

The final report on the reactor pressure vessel had not been completed; however, the Oak Ridge National Laboratory has completed their stress analysis and critical flaw size calculations.

The reactor vessel was positioned on May 27, 1971.

/S/ JBH 6/23/71
DEW
GCG
7/JHP

E. M. Howard
E. M. Howard
Senior Reactor Inspector

Enclosure:
CO Report No. 286/71-3, by R. Heishman, dtd 6/7/71

cc: E. G. Case, DRS (3)
P. A. Morris, DRL
R. S. Boyd, DRL (3)
R. C. DeYoung, DRL (2)
D. J. Skovholt, DRL (3)
H. R. Denton, DRL (2)
A. Giambusso, CO
L. Kornblith, Jr., CO
R. H. Engelken, CO
Regional Directors, CO
REG File

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U. S. ATOMIC ENERGY COMMISSION
REGION I
DIVISION OF COMPLIANCE

Report of Inspection

CO Report No. 286/71-3

Licensee: Consolidated Edison Company of New York
Indian Point No. 3
License No. CPPR-62
Category A

Date of Inspection: May 24, 1971

Date of Previous Inspection: April 6-7, 1971

Inspected by: RF Heishman 6/7/71
R. F. Heishman, Reactor Inspector (Principal) Date

Reviewed by: M. C. Moseley for 6/7/71
E. M. Howard, Senior Reactor Inspector Date

Proprietary Information: None

SCOPE

An announced limited inspection of the 3023 MWt pressurized water reactor under construction near Buchanan, New York, was conducted on May 24, 1971. The inspection was limited to a review of the QC records of the polar crane refurbishing and reactor vessel inspection following the lifting incident* which occurred on January 12, 1971. Results of crane load testing were also reviewed.

SUMMARY

Safety Items - None

Nonconformance Items - None

Unusual Occurrences - None

Status of Previously Reported Problems

Reactor Vessel Lifting Incident (Inquiry Memo 286/71-A) (CO Report 286/71-1 and 286/71-2)

N-3 The records of refurbishing of the polar crane were reviewed by the inspector. Detailed QC check list and marked-up drawings were available to indicate the

*Inquiry Memo 286/71-A

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NDT testing of components, replacement of components and reassembly of the units. Completed check lists were reviewed and indicated functional, mechanical and electrical checks were satisfactorily completed. Records indicated minor design changes had been made in order to strengthen the supports in the gear train assembly. The crane was successfully load tested in accordance with WEDCo procedure E-EVP-42, "Unit #3 Polar Crane Load Test Procedure", dated May 3, 1971. This test was completed on May 23, 1971. Post load testing visual inspection of the crane was conducted by WEDCo QC personnel and the manufacturer (Whiting Company).

The reactor vessel preliminary inspection reports from Con Ed's consultant, Oak Ridge National Laboratory, were reviewed by the inspector. ORNL has performed a stress analysis and critical flaw size calculations. The conclusions of these analyses are as follows:

> "Two types of fracture analyses, both believed to be conservative, have been made for the purpose of estimating the critical size of a flaw that would have been extended by the stresses induced in the incident. In one analysis the induced stress at the vessel surface was simply and conservatively assumed to be the yield stress, and realistic values of fracture toughness were used. The smallest flaw depth calculated by this method is 2.16 in. In the other analysis the load on the vessel was first calculated on the basis of observed damage (and lack thereof) in supporting structures. Stresses thus determined were combined with conservative dynamic fracture toughness values with the conclusion that a flaw of depth less than 3.72 in. would not have propagated.
>
> The critical size of a flaw would vary according to location and orientation in the vessel, and in both analyses these values represent the smallest sizes with respect to this factor. It should also be noted that the yield stress levels assumed in one analysis (Canonico) are much higher than the maximum stress calculated on the other analysis.
>
> We conclude that should the on-site nondestructive examination reveal only flaws less than two in., that such flaws would not have been extended as a consequence of the incident."

The final reports had not been received at the time of the inspection; however, review by Con Ed of the preliminary information from all sources involved in the review, has led to Con Ed's conclusion that no damage to the pressure vessel occurred as a result of the incident. The licensee stated the engineering evaluation would support the conclusions and would be available to AEC Regulatory as requested.

The inspector was informed by the licensee by telephone on May 27, 1971, that the reactor vessel was successfully positioned on May 26, 1971 with alignment being within tolerances. No shiming was necessary.

Other Significant Items

The inspector reviewed records of NDT examinations performed by WEDCo of the Unit 3 Steam Generator No. 32 (Westinghouse Spin No. INTRCPCSG2, S/N 8004). The NDT examinations were performed at the request of Westinghouse, Tampa Division. The inspection was conducted using procedures certified to the requirements of Appendix IX, Section III, ASME B&PV Code. The MP examinations provided coverage of the entire lower head assembly and approximately two inches of the lower shell course adjacent to the lower head circumferential weld. One linear indication was found in the lower shell course. This indication was removed by grinding and the area examined by liquid penetrant and magnetic particle methods. The reinspection established that the linear indication was removed.

The channel head nameplate of the steam generator reads:

FASMA	10004
ESSINGTON	PA
N2	
X5721	

Persons Contacted

The following personnel, in addition to those listed in the exit interview, were contacted during the inspection.

WEDCo

Mr. M. Snow, Manager, Reliability and QA
Mr. W. Diebler, Manager, Site QC
Mr. C. Bliesener, Quality Planning Engineer

Status of Construction

The licensee reported the status of construction to be approximately 50% complete.

Management Exit Interview

A management interview was conducted on May 24, 1971, at the construction site. The following personnel were in attendance.

Mr. J. A. Corcoran, Resident Engineer, Con Ed
Mr. E. J. Dadson, QA Engineer, Con Ed
Mr. F. M. Matra, IP-3 Project Superintendent, Con Ed

The following items were discussed.

The inspector stated the records of polar crane repair had been reviewed and no deficiencies identified.

The inspector also stated the results of the load testing of the polar crane were reviewed and no significant deficiencies identified.

Mr. Corcoran acknowledged the comments.

The inspector quiered Mr. Corcoran regarding the steam generator head problem identified by Westinghouse.

Mr. Dadson stated the testing conducted by WEDCo was considered adequate and no further action was anticipated.

The inspector asked when the final report of the reactor vessel inspection would be available.

Mr. Dadson stated the report was expected on or about July 1, 1971.

The inspector stated the final report would be reviewed when available.