PDR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

JUL 28 1980

Docket No. 50-461

Illinois Power Company
ATTN: Mr. W. C. Gerstner
Executive Vice President
500 South 27th Street
Decatur, IL 62525

Gentlemen:

Thank you for your final report dated July 18, 1980, pursuant to 10 CFR 50.55(e) regarding Control Rod Drive (CRD) penetrations through the drywell wall. We will complete our review of this matter during a future inspection.

Your cooperation with us is appreciated.

Sincerely,

Gaston Fiorelli

Gaston Fiorelli, Chief
Reactor Construction and
Engineering Support Branch

cc: Director, RCI/II
Director, AEQD
Chief, OEB/MPA
IE Files

cc w/ltr dtd 7/18/80:
Central Files
PDR
Local PDR
NSIC
Mr. Dean Hansell, Office of
Assistant Attorney General
Mr. Gary N. Wright, Chief
Director of Nuclear Safety

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Fiorelli

U-0157
L14-80(07-18)-9

ILLINOIS POWER COMPANY

500 SOUTH 27TH STREET, DECATUR, ILLINOIS 62525

July 18, 1980

Mr. James G. Keppler
Director, Region III
Office of Inspection and Enforcement
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Dear Mr. Keppler:

Reference: IP Letter 6/24/80, U-0150, L. J. Koch to J. G. Keppler, NRC Region III; Subject Matter: Extension of Time for 10CFR50.55(e) Final Report

Clinton Power Station Unit 1
Docket No. 50-461
Construction Permit No. CPPR-137

On May 27, 1980, Illinois Power Company verbally notified Mr. H. M. Wescott, NRC Region III, of a potential reportable deficiency per 10CFR50.55(e) associated with the Control Rod Drive (CRD) penetrations through the drywell wall. By the referenced letter, we informed you on June 24 by interim report that our investigation of the deficiency was not completed for final reporting.

We have since concluded our investigation as summarized in the attached report and corrective action is currently underway. We did not undertake a detailed investigation of the "reportability" of the deficiency as defined by 10CFR50.55(e) because: (1) it would entail an extremely large number of analyses to cover all of the possible operating modes to be considered, and (2) the recommended corrective action was taken. As the attached report indicates, CRD tube failure during normal power operation would not prevent normal control rod actuation and reactor shutdown, but under certain part load and reduced reactor pressure conditions the drive system may not function properly. It would require expensive, time consuming analyses to evaluate each possible operating scenario to determine if a reportable situation

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could result. We have concluded that it is more realistic to assume that a reportable set of circumstances could arise and to report the deficiency in accordance with the intent of 10CFR50.55(e) All of the CRD tubes which had been adversely affected by the welding are being replaced as corrective action for the deficiency.

We trust that the information provided in the attached final report is sufficient for your analysis and evaluation of the deficiency and the corrective action.

Sincerely,

L. J. Koch
Vice President

LJK:nh
attach.
cc: Director, Office of I&E, NRC, Washington, D.C.
H. H. Livermore, NRC Resident Inspector

Attachment to: U-0157

Y-5460
L14-80(07-10)-9

Clinton Power Station
10CFR50.55(e) Deficiency Report
Control Rod Drive Penetrations

Deficiency (10CFR50.55(e)(1)(iii))

As a result of a cleanliness inspection of a control rod drive (CRD) penetration of the drywell (Penetration 1MD-37), an area on the internal surface of Control Rod Drive Tube 2-B was found to have an area which appeared to have melted during the fillet welding process of the tube to the outside drywell plate. Since the pipe internals were not purged with an inert gas (a purge was not a requirement), the melted area had a burnt sugar appearance (oxidized). Subsequently, all tubes in the CRD penetrations were inspected. A total of fifteen insert tubes (1 1/4-inch Schedule 80 pipe, ASME SA-312, TP-316L) and twenty-three withdrawal tubes (1-inch Schedule 80 pipe, ASME SA-312, TP-316L) were identified by this investigation to have surface irregularities such as burn through and suck back on the internal surfaces of the tubes at the weld area connecting the tubes to the drywell penetration plate (1/2-inch carbon steel), and several more tubes were found to have evidence of oxidation on the internal surfaces of the tubes in the weld area.

Corrective Action

Nonconformance Report (NCR) No. 3041 was initiated and forwarded to Sargent & Lundy for dispositioning the deficiency. The corrective action was to:

a. Remove all tubes identified as having burn through/suck back conditions utilizing Sargent & Lundy approved Job Instruction P-017.

b. Clean and polish the remaining tubes utilizing a Sargent & Lundy approved Job Instruction P-018. Per the Job Instruction, the tubes were to be cleaned and polished and reinspected after thirty days. If a sugaring or rust-like condition reappeared after cleaning, a nonconformance report was to be initiated.

c. Replace unacceptable CRD tubes in accordance with Sargent & Lundy approved design.

This corrective action is currently being implemented. As of June 27, 1980, thirty-five (35) CRD tubes had been removed and the installation of the replacement tubes had commenced. The NRC resident inspector, H. H. Livermore, has performed surveillance activities on this corrective action.

To prevent recurrence of these problems in the installation of replacement tubes, the following actions will be taken (per Job Instruction M-009, Rev. 0):

a. The inside of the tubes will be purged with inert gas (i.e., argon) during welding operations.

b. During the welding operations, the welding variables (i.e., heat input) will be monitored.

Safety Implications

The effect of the burn through and suck back on the integrity of the tubes over the life of the plant is very difficult to determine. Since the tubes are stainless steel, it is believed that any tube failures would be in the form of cracks rather than a sudden rupture of the tube, and that all of the tubes would not fail simultaneously.

The reactor is designed so that the core system can be made subcritical at any time or at any core condition with the highest worth control rod fully withdrawn. Thus, any CRD tube could fail totally and the reactor could be brought to cold shutdown (the most limiting condition with regard to the shutdowm criteria) conditions.

If the reactor pressure was greater than 600 psig and all of the CRD tubes identified to have burn through or suck back were to fail totally, all control rods would insert on a scram signal. Reactor pressure would cause the control rods with failed tubes to insert. Thus, the reactor could be brought to cold shutdown conditions.

Damaged withdrawal tubes would not prevent a scram at any operating condition. If reactor pressure was less than 600 psig and more than one insert tube failed, the potential exists that the reactor could not be brought to cold shutdown conditions without using the standby liquid control system. If this action is undesirable, the reactor could be maintained in a stable condition at a pressure less than 600 psig and the corresponding saturated temperature. If reactor pressure would rise above 600 psig, the remaining withdrawn control rods could be inserted with reactor pressure and cold shutdown achieved. A detailed engineering analysis has not been performed to determine what, if any, combination of insert tube failures (of those fifteen insert tubes identified) and reactor conditions would not give a cold shutdown condition without the standby liquid control system. This is due to the cost of the analysis, the fact that all the tubes which had been adversely affected by the welding are being replaced, and such an analysis would not serve a useful purpose.

The corrective actions for the deficiency assure that no unsafe installation exists. Because a detailed engineering analysis was not performed, the significance of the deficiency to the safety of the plant has not been quantified. In summary, due to the items list above:

a. Failure mode via cracks is expected and not total failure.

b. Any one rod could be fully withdrawn and the reactor could be made subcritical at all conditions.

c. Above 600 psig reactor pressure, all control rods can be inserted even if the CRD tubes fail.

d. The small fraction of time the reactor is below 600 psig, the reactor could be made subcritical with the standby liquid control system if necessary or reactor pressure could be increased to 600 psig and then all control rods could be inserted.

e. A stable condition below 600 psig could always be reached.

The safety significance of the deficiency is considered minimal.

DWW:dl