Central files

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

APR 1 8 1975

Commonwealth Edison Company
ATTN: Mr. R. L. Bolger
Assistant Vice President
P. O. Box 767
Chicago, Illinois 60690

Docket No. 50-237
Docket No. 50-249

Gentlemen:

This refers to the inspection conducted by Messrs. Johnson, Brown, and Shafer of this office on March 11 - 14, 1975 of activities at Dresden Units No. 2 and No. 3 authorized by Licenses No. DPR-19 and No. DPR-25 and to the discussion of our findings with Messrs. A Roberts and D. Butterfield at the conclusion of the inspection.

A copy of our report of this inspection is enclosed and identifies the areas examined during the inspection. Within these areas, the inspection consisted of a selective examination of procedures and representative records, interviews with plant personnel, and observations by the inspectors.

During this inspection, it was found that certain of your activities appear to be in noncompliance with NRC requirements. The items and reference to the pertinent requirements are listed under Enforcement Action in the Summary of Findings Section of the enclosed inspection report.

This notice is sent to you pursuant to the provisions of Section 2.201 of the NRC's Rules of Practice," Part 2, Title 10, Code of Federal Regulations. Section 2.201 requires you to submit to this office within twenty days of your receipt of this notice, a written statement or explanation in reply, including: (1) corrective steps which have been taken by you, and the results achieved; (2) corrective steps which will be taken to avoid further items of noncompliance; and (3) the date when full compliance will be achieved. Except for Infractions No. 2 and No. 3, such a statement or explanation should be provided for each noncompliance item listed. Prior to the conclusion of the inspection, the inspectors determined that corrective action had been taken with respect to Infractions No. 2 and No. 3, and we have no further questions regarding these items at this time.

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Based upon discussions with your station management, as described in the enclosed report, it is our understanding that you plan to provide increased training for licensed personnel in the considerations involved when performing control rod movements. We will examine this matter further during a future inspection.

In accordance with Section 2.790 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this notice, the enclosed inspection report, and your response to this notice will be placed in the NRC's Public Document Room. If this report contains any information that you or your contractors believe to be proprietary, it is necessary that you make a written application to this office, within twenty days of your receipt of this notice, to withhold such information from public disclosure. Any such application must include a full statement of the reasons for which it is claimed that the information is proprietary, and should be prepared so the proprietary information identified in the application is contained in a separate part of the document. Unless we receive an application to withhold information or are otherwise contacted within the specified time period, the written material identified in this paragraph will be placed in the Public Document Room.

Should you have any questions concerning this inspection, we will be glad to discuss them with you.

Sincerely yours,

Gaston Fiorelli, Chief
Reactor Operations Branch

Enclosure:
IE Inspection Reports
No. 050-237/75-07 and
No. 050-249/75-07

cc: Mr. B. B. Stephenson,
Superintendent, w/encl

bcc: IE Chief, FS&EB
IE:HQ (4)
Licensing (4)
Central Files
IE Files
PDR
Local PDR
NSIC
TIC
A. Roisman

U. S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

REGION III

Report of Operations Inspection

IE Inspection Report No. 050-237/75-07
IE Inspection Report No. 050-249/75-07

Licensee: Commonwealth Edison Company
P. O. Box 767
Chicago, Illinois 60690

Dresden Nuclear Power Station
Units 2 and 3
Morris, Illinois

License No. DPR-19
License No. DPR-25
Category: C

Type of Licensee: GE BWR, 810 MWe

Type of Inspection: Routine, Unannounced

Dates of Inspection: March 11-14, 1975

Dates of Previous Inspection: January 27, 29, 30 and February 3, 10, 14, 18-19, 1975 (Operations)

Principal Inspector: PH Johnson
P. H. Johnson

4/17/75
(Date)

Accompanying Inspectors: C. H. Brown

W. D. Shafer

Other Accompanying Personnel: None

Reviewed By: R C Knop
R. C. Knop
Senior Reactor Inspector
Projects Unit 1

4/17/75
(Date)

SUMMARY OF FINDINGS

Enforcement Action

Violations: None.

Infractions

1. Contrary to Paragraph 3.10.D of the Dresden 2 Technical Specifications, two adjacent control rods were fully withdrawn on January 25, 1975, while conducting control rod drive maintenance activities. (Paragraph 3.a, Report Details)

   This infraction was identified as a result of an event which constituted an occurrence with safety significance.

2. Contrary to Paragraph 3.9.B.2 of the Dresden 3 Technical Specifications, both diesel generators which provide emergency electrical power to Unit 3 were inoperable for a period of approximately 23 hours on March 4 and 5, 1975. (Paragraph 3.q, Report Details)

   This infraction was identified by the licensee and had the potential for causing or contributing to an occurrence with safety significance. Corrective actions were noted to have been taken or were scheduled by the licensee, and a response to this item of noncompliance is not required.

3. Contrary to Paragraph 3.3.C of the Dresden 3 Technical Specifications, Unit 3 was operated for approximately 40 hours after the performance of control rod scram timing tests on March 2, 1975, before a review determined that test results did not satisfy the limiting conditions for operation. (Paragraph 7.a, Report Details)

   This deviation was identified as a result of an event which had the potential for causing or contributing to an occurrence with safety significance. Corrective actions were noted to have been taken by the licensee, and a response to this item of noncompliance is not required.

Deficiencies

1. Contrary to Paragraph 6.6.A of the Dresden 2 Technical Specifications, the discovery on October 17, 1974, that a reactor low pressure ECCS initiation instrument had a trip setting outside Technical Specifications limits was not reported to the NRC. (Paragraph 6.a, Report Details)

   This deficiency was identified by the inspector.

2. Contrary to Paragraph 6.2.F of the Dresden 2 Technical Specifications temporary hand-written changes were made to the high pressure coolant injection system surveillance procedure No. 2300-S-1 without the concurrence of two individuals holding senior operator licenses. (Paragraph 6.b, Report Details)

   This deficiency was identified by the inspector.

Licensee Action on Previously Identified Enforcement Matters

Not reviewed.

Unusual Occurrences

Selected unusual occurrences were reviewed during the inspection, as identified in Paragraph 3 of the Report Details.

Other Significant Findings

A. Current Findings

   Additional training on limitations and considerations involved when performing control rod movements is needed for licensed personnel. (Management Interview, Paragraph I)

B. Status of Previously Reported Unresolved Matters

   Not reviewed.

Management Interview

The inspectors conducted a management interview with Messrs. Roberts (Assistant Superintendent) and Butterfield (Administrative Assistant) at the conclusion of the inspection. The principal findings of the inspection, as known at that time, were also discussed with Mr. Stephenson (Station Superintendent) on March 13. Matters discussed during the management interview were as follows:

A. The inspector noted that Unit 3 was operated for approximately 40 hours after the performance of scram time measurements of 25 control rods on March 2, 1975, before evaluation of the results showed some of the scram times to be outside Technical Specifications limits. The inspector stated that this represented an item of noncompliance with Paragraph 3.3.C of the Technical Specification, although corrective actions were noted to have been taken by the licensee. The licensee acknowledged the inspector's comment. (Paragraph 7.a, Report Details)

B. The inspector noted that the general type calibration data sheets currently in use do not facilitate prompt identification of incomplete information. He stated that he had seen during the inspection a revised data sheet format which would facilitate supervisory review, although the new format had not yet been incorporated into all calibration data sheets for Units 2 and 3. The licensee stated that implementation of the revised format was being followed to completion.

C. The inspector stated that during his review of surveillance testing he noted that the diesel generator shutdown procedure following the monthly surveillance test had been revised to require readjustment of the speed droop and governor settings prior to engine shutdown although a similar revision had not been made to the surveillance procedure used for the refueling outage auto-initiation test. The licensee stated that the revision would be made in this procedure before it is used during this refueling outage for Unit 2.

D. The inspector stated that one reactor low pressure switch had been noted to have an as-found setpoint outside Technical Specifications limits, but that the condition was not reported to the NRC as required by Paragraph 6.6.A of the Technical Specification.

The inspector stated that this represented an item of noncompliance with Technical Specifications. The licensee acknowledged the inspector's comment. (Paragraph 6.a)

E. The inspector stated that handwritten temporary changes were noted to have been made to the HPCI surveillance test for Unit 2 on two occasions without the supervisory review required by Section 6.2.F of the Technical Specifications. This was noted to represent an item of noncompliance. The licensee acknowledged the inspector's comment. (Paragraph 6.b, Report Details)

F. The inspector commented that several surveillance and calibration data sheets showed evidence of supervisory review although supervisory signatures were not present. The inspector asked that supervisory personnel be reminded of their sign-off requirements. The licensee made no comment.

G. The inspector stated that review of the training program for instrument mechanics showed it to consist primarily of on-the-job training, supplemented with some vendor-procured programs and periodic supervisory lectures. He noted that training records do not identify individual skills but record only lecture attendance. The inspector identified a concern that a "B" instrument mechanic could be promoted to an "A" instrument mechanic without satisfying prerequisites other than on-the-job training. The licensee stated that skill levels were a factor in promotions to a higher level, and that if an individual next in line for a promotion were not considered by management to possess the required competence, promotion could be withheld. The licensee also stated that management is working on a program to provide improved training to the Instrument Maintenance Department in view of the high turnover rate experienced during the recent past. (Paragraph 6.d, Report Details)

H. The inspector stated that the abnormal occurrence involving withdrawal of two adjacent control rods during rod drive maintenance represented an item of noncompliance with Paragraph 3.10.D of the Technical Specifications. He also noted that corrective actions to date were inadequate; in particular, nothing had been done to increase the concern for (1) proper review of procedures and (2) alertness when performing control rod movements. He stated that a majority of the licensed personnel interviewed were not aware that the event had occurred. At this time the licensee presented the inspector a copy of a directive issued earlier in the day by the Station Superintendent (following discussions with the inspector) which advised all licensed personnel of the event and its significance. (Paragraph 3.a, Report Details)

I. The inspector stated that the event involving withdrawal of two adjacent control rods on Unit 2 and the October 31, 1974, rod movements on Unit 3 which resulted in high offgas release rates and apparent fuel damage had identified a need for additional training of licensed personnel in the concerns, considerations, and limitations involved in performing control rod movements. The licensee stated that additional training in these areas would be provided to licensed personnel.

J. The licensee was informed that the inadvertent disabling of No. 3 diesel-generator involved noncompliance with Paragraph 3.9.B of the Technical Specifications. The inspector stated that enforcement action was still under review, although the licensee's corrective actions had been reviewed, and that he had no further questions related to the occurrence. (Paragraph 3.q, Report Details)

K. The inspector noted that one hydraulic pipe restraint had been found with pinched O-rings, and stated that additional attention should be given to methods for verifying proper installation of

O-rings. He stated that maintenance personnel had expressed an intention to evaluate the feasibility of a Zerk fitting check following addition of hydraulic fluid to an installed pipe restraint. (Paragraph 3.e, Report Details)

L. The licensee was advised that additional investigation into the possible introduction of liquid nitrogen into the inerting piping appeared to be in order in view of the recent failure of valve 3-1601-21 and the recent crack in the Unit 2 inerting piping. The licensee was also informed that local leak rate data being maintained for bellows penetrations was not adequate. The inspector stated that the composite abnormal occurrence report for local leak rate test results should be submitted no later than 10 days following completion of related maintenance and retests, and no later than 10 days prior to startup of Unit 2. The licensee acknowledged the inspector's comments. (Paragraph 3.f, Report Details)

M. The inspector stated that he had examined the reactor feed pump minimum flow piping removed from Unit 2. He noted that the new material was expected to be more resistant to erosion but that nothing had been done to change the flow characteristics which caused the erosion, such that recurrence of the erosion could be expected, although at a slower rate. The inspector questioned the need for redesign of the piping run in the vicinity of the observed erosion. The licensee acknowledged the inspector's comments. (Paragraph 3.g, Report Details)

N. The inspector noted that followup reports related to uncoupling of Unit 2 control rod drives and failure of control rod drives to fully insert on scram were still outstanding. The licensee acknowledged the inspector's comment.

O. The inspector noted that water hammer in a core spray line had resulted in a Unit 3 scram on November 27, 1974, apparently as a result of inadequate venting of the core spray line during a refilling operation following valve maintenance. The licensee stated that procedures used for filling the core spray lines following maintenance would be reviewed to insure proper venting of the piping. (Paragraph 3.o, Report Details)

REPORT DETAILS

PART I

Prepared By: P. H. Johnson

1. Persons Contacted

B. Stephenson, Station Superintendent
A. Roberts, Assistant Superintendent
D. Butterfield, Administrative Assistant
G. Abrell, Unit 2 Operating Engineer
D. Adam, Radiation Chemistry Supervisor
J. Bauer, Shift Engineer
G. Bergau, Chemist
L. Dimmock, Engineer
J. Dodge, Nuclear Station Operator
J. Dolter, Leading Nuclear Engineer
R. Dyer, Job Planner
H. Hentschel, Engineer
W. Hildy, Instrument Engineer
B. Jaicomo, Nuclear Station Operator
R. Jolley, Nuclear Station Operator
T. Lang, Unit 3 Leading Engineer
E. Meintel, Maintenance Engineer
E. Petrowski, Engineer
G. Reimer, Engineer
C. Schiavi, Engineering Assistant
E. Seckinger, Engineer
M. Turbak, Engineer
J. Uremovic, Nuclear Station Operator
H. Whitehead, Shift Engineer
R. Williams, Unit 2 Leading Engineer

2. General

This unannounced inspection was conducted to perform portions of the periodic Inspection and Enforcement inspection program. At the time of the inspection Unit 2 was in a cold shutdown condition for core spray piping repairs. Unit 3 was operating at approximately 50% power with a stack offgas release rate of approximately 45,000 uCi/Sec.

3. Abnormal Occurrence Review

A review of reporting corrective actions, licensee review and evaluation, and compliance with regulatory requirements was conducted for the following abnormal occurrences and unusual events (UE's):

	Event Title	Event Date	Licensee Report Date
Unit 2			
1.	Primary Containment Penetration X-105C Leakage	12/12/74	12/20/74
2.	Withdrawal of two adjacent control blades during rod drive maintenance.	1/25/75	2/3/75
3.	Disabled overcurrent relay on Bus 23 to Bus 32-1 4KV Breaker.	1/20/75	1/30/75
4.	Unit 2/3 Diesel Generator Trip on Mechanical Overspeed	1/20/75	1/30/75
5.	Feedwater Sparger Failures (UE)	12/27/74	1/27/75
6.	Four-inch Recirculation Bypass Line Crack	12/13/74	12/20/74
7.	Leak on Gland Leak-Off Line for Valve 2-1201-99A	10/22/74	10/31/74
8.	Failure of Hydraulic Piping Restraints	10/21/74	10/31/74
Unit 3			
9.	Failure of Hydraulic Piping Restraints	1/4/75	1/16/75
10.	Failure of Hydraulic Pipe Restraint (UE)	2/16/75	3/3/75
11.	Excessive Leakage Through Valve 3-1601-21	2/14/75	2/24/75
12.	Reactor Feed Pump Minimum Flow Line Leak	1/7/75	1/17/75
13.	Failure of Cleanup System Inboard Isolation Valve to Open (UE)	1/3/75	1/30/75

	Event Title	Event Date	Licensee Report Date
14.	Cloride Concentration in Reactor Coolant Greater Than 0.5 ppm.	1/3/75	1/10/75

Specific aspects were reviewed for the following additional abnormal occurrences:

Unit 2

15.	Failure of 2B Core Spray Pump Manual Discharge Stop Check Valve	1/23/75	1/31/75
16.	Failure of 2B Core Spray Pump Manual Discharge Stop Check Valve	1/12/75	1/21/75
17.	Failure of Control Rod Drives to Fully Insert on Scram	11/2/74	11/12/74
18.	Uncoupling of Control Rod Drive K-11	8/2/74	8/9/74
19.	Uncoupling of Control Rod Drive P-12	10/23/74	10/29/74
20.	Uncoupling of Control Rod Drive N-10	11/2/74	11/12/74
21.	HPCI System Area Temperature Switch Setpoint Drift	11/9/74	11/18/74
22.	Two-Thirds Core Height Level Switch Setpoint Drift	1/22/75	1/31/75
23.	Failure of No. 4 Control Valve to Give Half Scram	11/2/74	11/12/74

Unit 3

24.	Isolation Level Switch Setpoint Drift	2/18/75	2/27/75
25.	Failure of Core Spray Valve 1402-24A to Open (UE)	11/27/75	12/17/75
26.	Failure of Core Spray Valve 1402-38A to Close (UE)	8/27/74	9/19/74

Event Title	Event Date	License Report Date
Unit 3 cont		
27. Secondary Containment Blowout Panel Failure	11/30/74	12/12/74
28. Unit 3 Diesel Generator Inoperable	3/10/75	3/18/75

The inspector's review included discussion of each event with a licensee representative and an examination of the report referenced above and other documents related to the particular areas reviewed. The following comments resulted from the inspector's review:

a. Event No. 2: This event represented noncompliance with Paragraph 3.10.D of the Technical Specifications, which allows a maximum of 2 non-adjacent control rods to be withdrawn from the core for rod drive maintenance. Other related findings were as follows:

   (1) The event was discussed with one of the senior operators who had approved the temporary procedure change which resulted in the withdrawal of two adjacent control rods. He stated that he had approved the procedure change based on his understanding that the adjacent control rod had been inserted. The licensed operator who was at the controls at the time of the event was not available for discussion at the time of the inspection.

   (2) The inspector considered the licensee's corrective actions related to the event to be inadequate; in particular, licensed personnel had not been advised of the occurrence. One senior operator and three of four licensed operators interviewed were not aware that an event involving the withdrawal of two adjacent control rods had occurred.

   (3) Licensed personnel interviewed stated that a nuclear engineer is normally present in the control room for all control rod movements. The discussions showed this practice to have resulted in decreased awareness and understanding (on the part of licensed personnel) of the considerations involved in control rod manipulations. The inspector also noted that some licensed operators regarded rod movement recommendations from the nuclear engineer as directions to be followed with little or no question. Additional training was noted to be needed in this area, as further discussed during the Management Interview.

b. Event No. 4: The inspector's review showed the event and corrective actions to have been as described in the licensee's report. The procedure change referenced in the licensee's report requires the governor droop setting to be reestablished at 5 and engine speed readjusted to 61 Hz prior to engine shutdown.

c. Event No. 5: At the time of the inspection, all four feedwater sparger assemblies had been prepared for installation in the reactor vessel, which was to proceed after removal (by grinding) of remaining PT indications on the blend radius of the feedwater nozzles. The inspector examined a sparger assembly and observed related activities on the refueling floor without comment.

d. Event No. 6: A followup report for this event is outstanding from the licensee. IE review of the related repairs is discussed in previous inspection reports.1/

e. Event Nos. 8, 9 and 10: The inspector reviewed a revised overhaul procedure for pipe restraints which was stated to include all O-rings and other parts used in reassembly. Licensee representatives stated that the geometry related to the installation of some O-rings is such that definite assurance of proper O-ring installation can not be provided (Event No. 9). Although the licensee's report related to Event No. 10 stated that zerk fittings would be checked for leakage following the addition of hydraulic fluid, licensee representatives did not know how this was to be accomplished. The shop maintenance procedure provides a leak check of each snubber following overhaul, although procedures do not discuss an in-place leak check. Licensee representatives stated that a check would be performed to determine whether the fittings could be tested for leakage by applying force to the indicator rod.

f. Event No. 11: Reviewed of local leak test measurements and discussions with a licensee representative resulted in the following comments:

(1) The information presented in the "IDENTIFICATION OF OCCURRENCE" section of the licensee's report was partially incorrect. This report indicated that a total integrated leak rate of 350 SCFH had been observed, and that this was below the Technical Specifications limit of 588 SCFH. The observed total leakage of 350 SCFH was measured at

1/ IE Inspection Rpts No. 050-237/75-01 and No. 050-237/75-04.

5.30 P.M. on February 14. Two earlier tests had been conducted, at 11:00 a.m. and 2:00 p.m. on the same date. The initial (as-found) leak rate measured at 11:00 a.m. was 1074 SCFH for the local leak rate measurement, which corresponded to an integrated containment leakage of 689 SCFH. The leakage was also reported as an unusual event, although it should have been reported as an abnormal occurrence, in that it represented an abnormal degradation of a boundary. These comments were discussed by phone with Mr. Stephenson (Station Superintendent) on March 26. The inspector noted that noncompliance was not involved since the licensee had made the initial notifications and the 10-day report as required for an abnormal occurrence.

(2) The local leak rate test performed at 2:00 p.m. gave a result of 536 SCFH, which corresponded to an integrated containment leak rate within the total allowable leakage. A blank flange was installed in the outboard flange of 1601-21 at 5:00 a.m. on February 15. Several additional tests were performed on February 14 and 15 during maintenance efforts to reduce the local leak rate using blank flanges, with a final LLRT result of 165 SCFH at 9:05 p.m. on February 15. The reactor was shutdown and cooled down on February 16 for other maintenance activities. Valve 1601-21 was replaced prior to startup, with a LLRT result of 4.26 SCFH following valve replacement.

(3) The licensee's report stated that inspection of the valve following removal showed it to have a cracked (rubber) valve seat, and stated that liquid nitrogen was a possible cause. A licensee representative demonstrated temperatures he had recorded on the associated piping on a previous occasion while inerting the containment with nitrogen. Contact temperature readings as low as 10°F (using a surface pyrometer) were observed. The inspector noted that previous instances of leakage of similar valves and cracks in associated ventilation piping had been experienced, possibly as a result of wide temperature variation. This matter will be reviewed further during a subsequent inspection.

g. Event No. 12: This was the most recent of several similar events involving erosion of feed pump minimum flow piping at a point just upstream of the maintenance valve which isolates it from the condenser. Examination of the piping arrangement in Unit 2 showed a vertical 3" piping run to a 90° elbow followed immediately by a 3" to 6" reducer. Erosion and leakage have been experienced at the top surface of the 6" piping adjacent to the reducer. The piping section between the reducer and the maintenance valve (approxiamtely 1" in length) was observed to have been replaced in the 3 minimum flow lines in Unit 2. Examination of the removed piping sections showed considerable wall thinning on the A and C lines, with actual penetration plus partial penetration of a welded patch on the B minimum flow line. The corrective action section of the licensee's report describes replacement of the mild steel 6" portion of the line with a more erosion-resistant chrome-moly section. The inspector noted that although the new material would be more resistant to erosion nothing had been done to eliminate the principal cause of the erosion - this is, that the flow characteristics which have been causing the erosion are still present.

h. Event No. 14: Licensee representatives stated that no recurrences of the abnormal chloride concentration had been observed. Conductivity recorder charts examined by the inspector showed normal conductivity in the reactor coolant prior to the cleanup system outage described in the licensee's report. The chart showed that cleanup system flow had been reestablished at approximately 0300 on January 5, with cleanup system outlet conductivity returning to a consistent 0.2 umho/cm thereafter, and inlet conductivity returning to normal over the next several hours. Recorder traces showed condensate demineralizer outlet conductivity to have remained essentially unchanged during the period of the cleanup system outage, supporting the licensee's conclusion that the cause of the high cloride concentration was the inoperability of the reactor coolant cleanup system.

i. Event Nos. 15 and 16: These events involved failure of the 2B core spray pump manual discharge stop check valve to reseat after termination of pump operation on two occasions in January 1975. As noted in the licensee's reports, the valve had been disassembled and inspected following the first occurrence. A licensee representative stated that further operation of the system had not been possible due to piping replacement still in progress, and that surveillance testing will be continued at an increased frequency following system repairs in an attempt to duplicate the problem. It was noted that the failure is not one which would affect initiation of system operation.

j. Event No. 17: A licensee representative stated that the size of the scram discharge volume was being increased in accordance with GE recommendations to correct the cause of the occurrence. It was also noted that the excessive rod drive seal leakage being experienced at the time of the occurrence was corrected by overhaul of most of the rod drives during the current outage. The licensee's abnormal occurrence report stated that additional corrective actions would be reported in a followup letter.

k. Event Nos. 18, 19, and 20: Licensee representatives stated that the cause of control rod uncouplings was determined to be due to dislodgement of inner filter in the rod drive, as determined during overhaul of the affected drives. The inspector noted that the referenced reports committed the licensee to a followup letter which was still outstanding.

l. Event No. 21: The licensee's report noted that calibration problems have been experienced with the Fenwal series 1700 switches. A licensee representative stated during the inspection that the 16 temperature switches associated with the Unit 2 HPCI system had been replaced with a liquid filled temperature sensor, and that replacement of remaining switches was being held in abeyance until the performance of the new switches has been evaluated.

m. Event No. 22: This report discussed setpoint drift of reactor level indicating switches apparently caused by loose mounting screws. A licensee representative stated that related corrective actions for Unit 2 had been completed, with screw tightening required on 9 of 20 similar switches inspected. The inspector examined a letter from the licensee to the Yarway Corporation questioning the suitability of "Loctite" on the screws as a resolution of the problem. A licensee representative stated that a representative of the Yarway Corporation had subsequently indicated during a phone conversation that the use of Loctite should solve the problem. The licensee representative stated that switch mountings on Unit 3 would be checked during the forthcoming refueling outage.

n. Event No. 23: The licensee's report stated that the cause of this occurrence was determined to be a broken lead on the amphenol plug from the fast acting solenoid initiating the scram function. A licensee representative confirmed during the inspection that all amphenol plugs associated with the Unit 2 control valves had been replaced with solder connections. He stated that no plans had as yet been made for Unit 3.

o. Event No. 25: The referenced report discussed a reactor scram experienced on November 27, 1974, due to a water hammer in the core spray system, the vibrations from which actuated the main steam line flow switches. The inspector questioned licensee representatives concerning the source of the core spray system water hammer. The representatives stated that a portion of the core spray piping had been drained for valve maintenance and had apparently not been properly vented during the refilling operation. Licensee management stated during the management interview that the procedure for refilling the core spray piping would be reviewed to insure that proper venting of the piping is provided.

p. Event No. 27: A licensee representative stated that a permanent repair for this occurrence is scheduled to be accomplished during the Unit 3 refueling outage.

q. Event No. 28: Review of records and discussions with licensee representatives showed the occurrence to have been as described in the licensee's report. The inspector noted that the occurrence represented noncompliance with Technical Specifications requirements, although corrective actions to prevent recurrence had been taken or were planned by the licensee as indicated in the referenced report.

REPORT DETAILS

PART II

Prepared By: C. H. Brown H C Dance/for 4/17/75
(Date)

W. D. Shafer H C Dance/for 4/17/75
(Date)

Reviewed By: H. C. Dance H C Dance 4/17/75
(Date)

4. Persons Contacted

A. Roberts, Assistant Superintendent
W. Hildy, Instrument Engineer
T. Watts, Technical Staff Supervisor
D. Adam, Radiation Chemistry Supervisor
R. Ragan, Unit 3 Operating Engineer
G. Abrell, Unit 2 Operating Engineer
R. Cozzi, Surveillance Engineer
M. Wright, Quality Control Engineer
G. Bergan, Chemist
J. Dolter, Leading Nuclear Enginer
E. Petrowski, Nuclear Engineer
R. Cambell, Instrument Maintenance Foreman
L. Dimmock, Engineer
N. Scott, Shift Engineer
T. Lang, Unit 3 Leading Engineer
R. Williams, Unit 2 Leading Engineer

5. Logs and Records

The following logs and records were reviewed with specific comments as noted in Paragraphs 6 and 7 of this report:

a. Shift Engineer's Log - 1/30/75 - 3/10/75, 6/1 - 6/30/75, and 11/1 - 11/30/74.

b. Deviation Log, Unit 2 - June and July 1974.

c. Tag Out Sheets, Unit 3 - December 1974 and February 1975

d. Surveillance Records - as shown in Table I.

e. Calibration Records - as shown in Table I.

f. Unit 2 Operating Log - April, August, November, 1974.

g. Unit 3 Operating Log - August, July, 1974.

6. Surveillance and Calibration

The inspectors reviewed surveillance and calibration procedures and completed data sheets for Unit 2 and Unit 3 as listed in Table 1 for the following information: prerequistes; test and calibration; operational checks prior to returning equipment to service; acceptance criteria; review and approval; and detailed instructions. The following comments related to the surveillance and calibration program resulted from the inspectors' review:

a. In reviewing the test data taken on October 17, 1974 for the Unit 2 Reactor Low Pressure LPCI Start Test, the inspector noted that the "as found" reading for instrument number 263-52-A1 was recorded as 306 psig. The Technial Specifications limits are 300 $\leq$ P $\leq$350 (reactor pressure) but for the instrument readout a 10 psig instrument head must be included, shifting the Technical Specifications limits to 310 $\leq$ P $\leq$ 360 (instrument pressure). The inspector observed that the 306 psig setpoint had been recognized and corrected by licensee personnel, but had not been identified as being outside Technical Specifications limitations. Consequently, it was not reported as required by Technical Specification 6.6.A.

b. The inspector identified penciled-in changes (involving required valve positions) on surveillance data sheets completed on April 13, 1974 and August 1, 1974 for the HPCI Pump and Valve Operability Procedure. Review of the Temporary Change Log showed that no temporary changes had been approved, contrary to Technical Specification 6.2.F.

c. With respect to calibration standards, the inspector reviewed the calibration frequency, traceability and storage control for (1) a Pyrotest instrument (2) an oscilloscope, and (3) a Weston D. C. Ammeter. The overall program used to monitor instruments needed for calibrating plant equipment appeared to be adequate.

d. The inspector reviewed the qualification records of two instrument mechanics and noted that the information consisted of the title of each training course attended, the date, and the instructor's signature. In discussion with a licensee representative the inspector established that individual assignment to instrument repair work is dependent upon the supervisor's confidence in the instrument mechanic's ability. For the most part, experience is obtained through on-the-job training with some training bought from outside vendors.

e. The scram reset time delay relay was verified to have operated at greater than 10 seconds on Unit 3 on March 12, 1974. Unit 2 relay reset time is scheduled to be checked before restart from the current outage.

7. Limiting Conditions for Operation, Limiting Safety System Settings and Safety Limits

The operation of selected systems was reviewed for compliance with Technical Specifications requirements, with the following comments:

a. Unit 3 Control Rod Drive Slow Scram Times. On March 2, 1975, a 25 rod scram test was performed on Unit 3. The results indicated that rod drive J-9 had a 90% scram insertion time more than 0.75 seconds greater than the 25-rod average. The required action of scramming the eight rods around the slow rod was taken and drive J-8 also exceeded the 25-rod average plus 0.75 seconds. The scram times were examined at the time and average 90% insertion times for 2 x 2 arrays were found acceptable. However, during further review of the test results on March 4 it was determined that three 2 x 2 arrays exceeded the 5% and 20% average scram time limits. Because of the delay in evaluating the test results, the reactor was operated at approximately 50% power for 40 hours with scram times exceeding those specified in Paragraph 3.3.C of the Technical Specifications. The licensee was informed that this was an item of noncompliance. Subsequent to the determination, the test was repeated and all 2 x 2 arrays were found to be within Technical Specifications limits. Additional testing was performed on March 9, 1975 and the same two rods were found to be slow. The 2 x 2 array was found to have its 5% and 20% insertion times slow. Control rods J9 and K8 were declared inoperable as required by Technical Specifications. Rod J9 was left at position 48 and K8 was fully inserted and disarmed within three hours after the 25 rod scram test was commenced. This event was reported by the licensee as an abnormal occurrence.2/ Licensee representatives stated that the high scram times resulted from wearing of rod drive seals, and would be corrected by maintenance to be performed during the forthcoming refueling outage.

b. Unit 2 Control Rod Drives. The licensee stated that all but 23 control rod drives in Unit 2 had been modified by March 1975. This modification involves the placement of the inner

2/ Ltr. Stephenson to Keppler, dtd 3/12/75 (AO Rpt 75-13).

filter on the stop piston. The 23 drives that require modification were stated to be peripheral drives with the exception of N-8 and are scheduled for overhaul during a subsequent outage.

TABLE 1

Listing of Calibration and Surveillance Records Reviewed

Unit	Tech. Spec. Reference	Procedure	Title	Dates Reviewed	Comments
2	3.1	33-200-VII	Reactor Water Level Low	4/22/74 7/23/74 10/4/74 11/21/74 1/15/75	Data sheets were marked to indicate supervisory review but no signature was present on data sheets completed on 4/2 & 11/21/74.
3	3.1	33-200-VII	Reactor Water Level Low	5/10/74 8/11/74 11/6/74 2/10/75	
2	3.6.C.1	37-3-8 Rev 1	Chemistry	8/1-31/74 11/1-30/74	
3	3.6.C.1	37-3-8 Rev 1	Chemistry	5/1-31/74 8/1-31/74 9/1-30/74	
2	3.2.b	33-200-V	Sustained High Rx Pressure	4/25/74 7/28/74 8/18/74 10/15/74	
3	3.2.b	33-200-V	Sustained High Rx Pressure	5/12/74 6/10/74 9/10/74 11/11/74 1/5/75	
2	3.2	33-200-IV	Rx Low Pressure, LPCI Start	4/19/74 7/28/74 10/17/74 12/15/74	See Paragraph 6.a.
3	3.2	33-200-IV	Rx Low Pressure, LPCI Start	5/12/74 9/30/74 11/11/74 2/24/75	
2	3.5	1500-S-IV	Containment Cooling	4/14/74 7/18/74 10/10/74 1/6/75	
3	3.5	1500-S-IV	Containment Cooling	4/13/74 7/17/74 10/3/74 1/12/74	

TABLE 1

Listing of Calibration and Surveillance Records Reviewed

Unit	Tech. Spec. Reference	Procedure	Title	Dates Reviewed	Comments
2	4.5.C	2300-S-1 2300-S-III	HPCI System Valve and Pump Operability and Flow Rate Test	4/13/74 5/21/74 7/20/74 8/1/74	Revision change in March 1974. See Paragraph 6.b.
3	4.5.C	2300-S-1 2300-S-III	HPCI System Valve and Pump Operability and Flow Rate Test	6/21/74 7/20/74 8/24/74 11/21/74	
2	3.2.3	33-700-II	IRM Upscale & Inop	5/8/74 6/12/74 7/24/74 8/29/74 9/25/74	
3	3.2.3	33-700-II	IRM Upscale & Inop	5/1/74 6/27/74 8/20/74 10/10/74 12/24/74 1/2/75 2/19/75	
2	4.2.1	33-700-IV	APRM Upsacle, Inoperable, & Downscale Scram.	4/74-2/75	Last test completed 10/5/7 Unit in scheduled refueling.
3	4.2.1	33-700-IV	APRM Upscale, Inoperable, & Downscale Scram.	4/74-2/75	
2	3.3.C & 4.3.C		CRD Scram Time, 25 and 50 Rod	5/2/74 6/2/74 8/3/74 9/1/74	
3	3.3.C & 4.3.C		CRD Scram Time, 25 and 50 Rods	2/74-3/74	See Paragraph 7.a.

TABLE 1

Listing of Calibration and Surveillance Records Reviewed

Unit	Tech. Spec. Reference	Procedure	Title	Dates Reviewed	Comments
2	Tables 4.1.1 and 4.1.2	33-500-IV	Loss of Turbine Control oil pressure	4/27/74 10/19/74	
3	Tables 4.1.1 and 4.1.2	33-500-IV	Loss of Turbine Control oil pressure	4/74 - 2/75	
2	Tables 4.1.1 and 4.1.2	500-S-X	Turbine Stop Valve Closure	4/74 - 10/74	
3	Tables 4.1.1 and 4.1.2	500-S-X	Turbine Stop Valve Closure	4/74 - 2/75	
2	Tables 4.1.1 and 4.1.2	500-S-1X	Generator Load Rejection Scram	4/74 - 10/47	
3	Tables 4.1.1 and 4.1.2	500-S-1X	Generator Load Rejection Scram	4/74 - 2/75	
2	Tables 4.1.1 and 4.1.2	33-500-III	Turbine 1st stage pressure scram @ 45%	5/74 - 10/74	
3	Tables 4.1.1 and 4.1.2	33-500-III	Turbine 1st stage pressure scram @ 45%	5/74 - 2/75	
2	4.5.A	6600-S-V	Under Voltage Emergency Bus (ECCS Test)	-	Not yet completed for this refueling.
3	4.5.A	6600-S-V	Under Voltage Emergency Bus (ECCS Test)	5/12/74	6600-S-V did not reflect revision in 6600-S-1
2	4.9.A.1	6600-S-1	Diesel Full Load Test	4/74 - 2/75	
3	4.9.A.1	6600-S-1	Diesel Full Load Test	4/74 - 2/75	See Paragarph 3.q.
2/3	4.9.A.1	6600-S-1	Diesel Full Load Test	4/74 - 2/75	

TABLE 1

Listing of Calibration and Surveillance Records Reviewed

Unit	Tech. Spec. Reference	Procedure	Title	Dates Reviewed	Comments
2	4.3.D	-	CRD Accumulator Pressure and Level Alarm Checks	5/74 8/74 10/74	
3	4.3.D	-	CRD Accumulator Pressure and Level Alarm Checks	6/74 12/74 1/75	
2	4.7.A.4.d	1600-S-XII	Torus to Drywell Vacuum Breaker Cycling	4/74 - 10/75	
3	4.7.A.4.d	1600-S-XII	Torus to Drywell Vacuum Breaker Cycling	4/74 - 2/75	