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Iowa Electric Light & Power Co. Cedar Rapids, Iowa Mr. ELL. Hammond	4-19-74	4-25-74	X			

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J.G. Keppler	1 signed			XXX	XXX

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DESCRIPTION:
Ltr reportingaan abnormal Occurrence at the Duane Arnold Plant...concerning...Violation of a limiting condition for operation.... resulting from the initiation of the Core Spray System by a simulated Hi Drywell Pressure Signal....trans the following....

PLANT NAME: Duane Arnold

ENCLOSURES:
Abnormal Occurrence Report No. DPR-49-74-2

ACKNOWLEDGED

(1 cy encl rec'd)

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FOR ACTION/INFORMATION 50-4-25-74 JB

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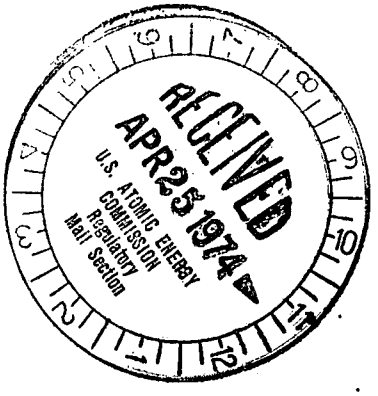
IOWA ELECTRIC LIGHT AND POWER COMPANY

General Office

CEDAR RAPIDS, IOWA

DUANE ARNOLD ENERGY CENTER
PALO, IOWA
APRIL 19, 1974
DAEC - 74 - 141



Mr. James G. Keppler
Regional Director
Directorate of Regulatory Operations, USAEC
Region III
799 Roosevelt Road
Glen Ellyn, Illinois 60137

50-331

SUBJECT: Abnormal Occurrence No. DPR 49/74-2
FILE: A-118a

Dear Mr. Keppler:

In accordance with Appendix A to Operating License DPR-49, Technical Specifications and Bases for Duane Arnold Energy Center, please find enclosed a written report on the subject abnormal occurrence.

Yours very truly,

Ellery L Hammond

Ellery L. Hammond
Assistant Chief Engineer
DAEC

ENCLOSURE

DLW/ELH/mrm
CC: ✓John O'Leary
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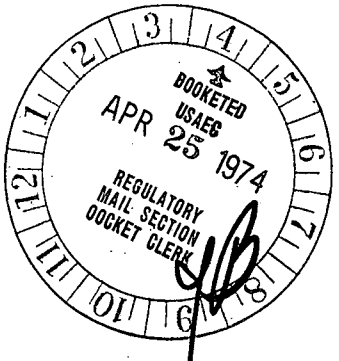

3659

IOWA ELECTRIC LIGHT AND POWER COMPANY

General Office

CEDAR RAPIDS, IOWA

Subject: Abnormal Occurrence

Report Number: DPR-49/74-2

Report Date: April 19, 1974

Occurrence Date: April 10, 1974

Facility: Duane Arnold Energy Center, Unit No. 1, Palo, Iowa

Identification of Occurrence

Violation of a limiting condition for operation established in the Technical Specifications, Section 3.6.A.2, resulting from the initiation of the Core Spray System by a simulated Hi Drywell Pressure Signal.

Conditions Prior to Occurrence

1. Reactor in cold shutdown mode.
2. Reactor vessel filled with water.
3. Reactor vessel head vents open.
4. Reactor vessel pressure - atmosphere (20 psig on recorder due to water column above normal vessel water level.) (See item 2 above.)
5. Reactor vessel wall temperature - 170^{0}F.
6. Reactor flange temperature - 155^{0}F.
7. Core Spray System in automatic mode. (Core Spray System not required to be operable for plant conditions at the time of the occurrence.) Core Spray System "B" Inboard Throttle Valve MOV 2137 and Outboard MOV 2135, breakers open.
8. ECCS fill pump 1P-70 not operable.
9. RHR System in shutdown cooling mode with MOV 1908 and MOV 1909 closed.

10. Diesel generator 1G-21 was running in preparation for load testing. Diesel 1G-31 in automatic. (not running)

11. Test switch installed to simulate a high drywell pressure signal for automatic start of Standby Diesel Generators for testing purposes.

Description of Occurrence

At the time of the occurrence (0959), Iowa Electric Light and Power Company's DAEC Operations personnel were preparing to line up the RHR System for commencement of Startup Test Instruction - 34, Recirculation System Vibration Test.

At time 0959 and 20 seconds, the occurrence was detected by Alarm "B507 Drywell Pressure ECCS on AA, High", Alarm "B576 Drywell Pressure ECCS on AB High," and operator observed Core Spray Pump 211A flow indicator (FI-2110) surged momentarily to approximately 1500 gpm.

The following items describe the events during and following the occurrence:

1. 0959:04 Computer logged "Reactor Pressure 26 psig" (Routine Output)

2. 09hours:59minutes:19seconds:51cycles: Computer logged "Diesel 1G-31 Cranking"

3. 0959:20 Computer logged "Drywell Pressure High on Channel AA"

   "Drywell Pressure High on Channel AB"

   "Core Spray Pump 1P-211B Breaker 404, closed"

   "Core Spray Pump 1P-211A Breaker 304, closed"

4. 0959:21 Computer logged "Drywell Pressure ECCS on AA, reset"

   "Drywell Pressure ECCS on AB, reset"

   "Core Spray Pump 1P-211B Discharge Pressure, High"

   Core Spray Pump 1P-211B started on initiation signal but inboard throttle valve MOV 2137 and outboard MOV 2135 were electrically deenergized so the pump did not actually put any water into the reactor vessel.

5. 0959:22 Computer logged "Diesel 1G-31 completed cranking"

   Diesel 1G-21 was already running for test purposes.

6. 0959:24 Computer logged "Core Spray Pump 1P-211A Discharge Pressure, High"

7. 0959:26 Computer logged, "HPCI Main Steam Control Valve opened"

   The High Drywell pressure signal initiated the HPCI System but steam was not available.

8. 0959:31 Operator manually tripped core spray pump 211A, and reset initiation signal.

9. 0959:32 Computer logged, "Core spray pump 1P-211A Breaker, Trip"

10. 0959:33 Operator manually tripped core spray pump 211B and reset initiation signal.

11. 0959:34 Computer logged, "Core spray pump 1P-211B Breaker, Trip"

    "HPCI main steam turbine stop valve, not closed."

12. 1000:04 Computer logged, "Reactor Pressure 292 psig"

    This was the maximum pressure indicated by the one minute computer scan pickup.

    Drywell floor drain sump pump cycled automatically.

13. 1001:04 Computer logged, "Reactor pressure 179 psig"

14. 1001:48 Computer logged, "Core spray pump 211A discharge pressure normal"

15. 1002:04 Computer logged, "Reactor Pressure 114 psig"

16. 1003:04 Computer logged, "Reactor Pressure 75 psig"

17. 1004:04 Computer logged, "Reactor Pressure 55 psig"

18. 1005:04 Computer logged, "Reactor Pressure 42 psig"

19. 1006:04 Computer logged, "Reactor Pressure 35 psig"

20. 1007:04 Computer logged, "Reactor Pressure 31 psig"

21. 1008:04 Computer logged, "Reactor Pressure 28 psig"

22. 1014:25 Computer logged, "HPCI Pump Low Flow"

23. 1015:15 Operator manually reset HPCI initiation signal.

24. 1015:16 Computer logged, "HPCI Pump Low Flow, Reset"

25. 1017:49 Computer logged, "HPBI Main Steam Control Valve, closed"

26. 1017:56 Computer logged, "HPCI Main Steam Turbine Stop Valve, not open."

27. While recovering from the occurrence, the test switch was noticed to be out of the neutral position and it was immediately returned to neutral.

28. The IEL&P General Production Manager was notified of the occurrence at 1330 on April 10, 1974.

29. The DAEC Chief Engineer was notified of the occurrence at 0830 on April 11, 1974.

30. The occurrence was discussed at a DAEC Operations Committee meeting on April 10, 1974.

31. At 0945, on April 11, 1974, the Office of the Directorate of Regulatory Operations for Region III was notified by telephone. The same office was notified via telecopier on April 11, 1974.

Designation of Apparent Cause of Occurrence

The cause of the abnormal occurrence was the inadvertent activation of the High Drywell Pressure test switch.

An Operator was approaching the console for preparation of STI-34. A temporary test switch for simulating a high drywell pressure signal in accordance with Procedure EM-3, was lying on the floor in front of the console and the Operator apparently accidentally activated the switch with his foot.

The test switch was a GE type SB-1 Control Switch. It was connected as below:

1C32	Terminals CC61 and CC62
1C32	Terminals CC63 and CC64
1C43	Terminals BB4 and BB13 (Core Spray Logic A)
1C43	Terminals BB5 and BB14 (Core Spray Logic A)

The test switch was part of test #EM-3, Standby Diesel Generators "Test Ability of Diesel Generators to Pickup Emergency Core Cooling Systems sequentially following inadvertent closing of Diesel Generator Breakers with generators out of synchronism."

There was no other apparent cause for the high drywell pressure signal.

Analysis of Occurrence

The subject abnormal occurrence has been analyzed in detail and it has been determined that the transient did not have a deleterious effect on the pressure vessel.

The following is a summary of the information used in determining the above conclusion:

1. Computer printout of Reactor Pressure

Time	Pressure (psig)
0959:04	26
1000:04	292
1001:04	179
1002:04	114
1003:04	76
1004:04	55
1005:04	42
1006:04	35
1007:04	31
1008:04	28
1009:04	27
1010:04	25

Analysis of the computer printout indicates a maximum pressure of 292 psig. (Prior to the occurrence, an Operator had selected Reactor Pressure to be read by the computer and logged at one minute intervals.) The pressure was greater than the technical specification limit of 250 psig for less than one (1) minute.

2. Analysis of the reactor pressure chart recorder (PI4563 and 4564) indicates a maximum pressure of 370 psig. The pressure transient was an instantaneous spike on the recorder, and a time difference cannot be detected between the increase and decrease in pressure above 250 psig. (Recorder speed 1"/hour). The slope of pressure line is asymptotic above an indicated pressure of 250 psig.

3. The General Electric Company (NSSS supplier) has advised that the reactor vessel design is such that the vessel temperature should be above 120°F. (actual indicated was 155°F.), for vessel pressures in the range of 312-1000 psig.

4. Reactor vessel wall temperature, reactor vessel head temperature, and moderator temperature did not vary from pre-transient values.

5. Following the occurrence, the Core Spray Systems were inspected for possible mechanical damage. During the inspection, the anchors on one seismic restraint (6BB-14-SR18) were found to be pulled from the concrete wall on Core Spray System B. All remaining pipe restraints and hangers on both Core Spray Systems were inspected and found to be satisfactory. An investigation into the cause of the restraint movement is in progress. A tentative cause for the excessive discharge pipe movement has been attributed to the fact that the fill pump was not operating prior to the occurrence and that the discharge piping may not have been filled with water. (See item 9 of the section on "Conditions Prior to Occurrence").

Correction Action

1. Immediate corrective action consisted of securing the core spray pumps and resetting the initiation trip signals.

2. Followup corrective action consisted of placing the temporary High Drywell Pressure Test Switch in an area where accidental activation is not possible.

3. The Maintenance Department was directed to ensure that temporary test switches and other devices utilized during plant maintenance and testing are placed in areas where this type of occurrence cannot happen again.

4. Operations personnel were cautioned to take additional measures to ensure their moves in the Control Room are deliberate and safe.

5. Since the Core Spray Systems were not required to be operable for plant conditions at the time of the occurrence, an Unusual Event Report will be submitted within 30 days of the abnormal occurrence to document the Core Spray seismic restraint movement.

Conclusion

The contents of this report including corrective actions were reviewed by the DAEC Operations Committee on April 18, 1974. The Committee concluded that the corrective action taken will help prevent recurrence of this type of occurrence in the future. The Committee also concluded that the health and safety of the public and plant personnel was not impaired. This report was reviewed and approved by the General Production Manager.

Ellery L. Hammond

Ellery L. Hammond
Assistant Chief Engineer
Duane Arnold Energy Center

GAE/DLW/ELH/mrm