50-249

Commonwealth Edison Company

ONE FIRST NATIONAL PLAZA ★ CHICAGO, ILLINOIS

Address Reply to:
POST OFFICE BOX 767 ★ CHICAGO, ILLINOIS 60690

December 17, 1971

Dr. Peter A. Morris, Director
Division of Reactor Licensing
U.S. Atomic Energy Commission
Washington, D.C. 20545

Subject: Report of Safety Valve Operation following a Feedwater Transient - Dresden Unit 3 (DPR-25)

Dear Dr. Morris:

This is to report a condition in which a condensate booster pump tripped, which caused a reactor water level transient. This water level transient resulted in filling the main steam line with water, opening of a safety valve, and pressurization of the drywell. The following information pertaining to this occurrence is submitted pending completion of the investigation which is currently in progress.

Summary

A 1413 on December 8, 1971, as a result of a condensate booster pump trip on Dresden Unit 3, the reactor feed pumps tripped. Tripping of the feed pumps resulted in a reactor water level transient. This eventually resulted in filling the main steam lines with water, opening of a safety valve for approximately 1½ minutes and pressurization of the drywell. Pressurization of the drywell resulted in a high drywell signal which initiated starting of emergency diesels, low pressure core cooling pumps, and HPCI.

During the transient, drywell pressure reached a maximum of 20 psig. The maximum and minimum reactor pressures were 1050 psig and 795 psig, respectively, and the reactor water level reached a minimum of -20 inches and a maximum of +130 inches. With water level at -20 inches, there is more than 9 feet of water above the fuel. A detailed sequence of events is attached.

All safety systems functioned as designed. No significant radioactivity was released to the environment as a result of the incident. During post incident recovery, both the primary system and the primary containment were maintained in a "bottled up" condition until analysis of reactor water and containment atmosphere could be made.

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Damage Assessment

A preliminary inspection in the drywell following the incident revealed damage to the following equipment:

1 - The rupture discs on all the safety valves showed cracks. This may not be related to the incident since this condition has been encountered previously on normal shutdowns.

2 - The 3A electromatic valve was damaged by the steam jet from the "F" safety valve. One steam discharge rams horn on the "F" safety valve was directed towards the electromatic valve. The cover on the solenoid assembly of this valve was blown off. The holding coil portion of the solenoid assembly was found open. Wiring to a position indicating limit switch was also damaged rendering the position indication circuit inoperative.

3 - Miscellaneous thermal insulation was damaged and requires repair.

4 - The top coat of paint on the containment wall over an area about 3' x 3' was removed by the steam jet from the "F" safety valve impinging on the surface of the containment.

5 - Sections of ventilating duct in the vicinity of the steam jet were dislodged and require repair.

6 - The LPRM cables were found damaged. This cable must be replaced in total.

7 - The SRM/IRM cables were tested and found to be good. Following the Dresden Unit 2 June 5 incident, this cable was replaced on Dresden Units 2 and 3 and Quad-Cities Units 1 and 2 with cable having a higher temperature (302°F 10 hour rating) rating.

8 - One containment cooling fan motor was found to have a ground caused by moisture in the containment. This motor will require drying out. The other six cooling fan motors were found to be in good condition.

Preliminary Conclusions

The following conclusions have been reached regarding the Dresden Unit 3 incident of December 8, 1971:

1 - There were no radiological consequences since no significant release to the environment resulted from the incident.

2 - No compromise of the health and safety of the public resulted from the incident.

3 - All operations during the incident and post incident recovery period were within Technical Specifications.

4 - All safety systems functioned as designed including High Pressure Coolant Injection (HPCI), Low Pressure Coolant Injection (LPCI), Core Spray, Main Steam and Containment Isolations, Standby Gas Treatment System, Pressure Suppression System, and Standby Diesel Generators.

5 - Feedwater control system performance during the transient was deficient, in that, the control system locked out on low air pressure, probably during rapid valve movement. Previous experience has demonstrated the inability of the feedwater control system to automatically control level below the high water level trip point for main steam isolation valves during a system transient. This was the primary reason for the need to take operator action.

6 - Operator response was in accordance with operating procedures throughout the incident and post incident recovery with two exceptions. The operator did not reset the feedwater regulating valve lockout condition when it occurred, and he did not trip the feed pump when the water level reached +60 inches. Had he done so, the incident may have been prevented. It is important to place these actions in proper perspective, and it should be emphasized that he did take a number of steps to control feedwater input to the vessel. The operator actions were:

(a) He reduced the master controller set point to minimize the error signal between the actual level and set point level.

This response was previously established on shift by General Electric during the startup program to compensate for the known overshoot which has been experienced following scrams. While not specifically called for by the station operating procedure 600-ANI, it is consistent with the intent of the procedure to keep the level on scale.

(b) He closed the minimum flow feedwater valve. — what effect does this have

(c) He reduced the manual output control potentiometer on the "manual-auto" controller to zero and transferred them to manual to terminate feedwater input.

7 - The "F" safety valve lifted at approximately 1020 psig reactor pressure. The safety valve set point is 1240 psig. — TEST RESULTS

The lifting of this valve was probably caused by some mechanism resulting from the effects of feedwater flooding the main steam line.

The pressurization of the drywell could probably have been avoided if this valve had not lifted. — NO Kidding

Corrective Actions

The following corrective actions will be accomplished prior to startup:

DETERMINE CAUSE OF SAFETY VALVE LIFTING?

1 - Safety evaluations.

(a) Effects on fuel
(b) Vessel internals
(c) Performance of suppression pool
(d) Effects of pressure, temperature and steam impingement on primary containment
(e) Differential temperature on vessel

2 - LPRM repair

3 - Thermal insulation repair

4 - Replace "F" safety valve with tested valve

5 - Check operability of 3A electromatic valve

6 - Investigate reorientation of safety valve discharge — ?

7 - Check calibration of feedwater control system and verify feedwater regulating valve response to control signal. leak through

8 - Increase torque setting on feedwater regulator isolation valve so that it will close under pressure and seat properly.

9 - Modify and emphasize procedure for handling water level transients. Leave control system in automatic, reduce feedwater controller set point and trip feed pump at 45".

10 - Evaluate the need for non-destructive inspection of the main steam lines.

11 - Check calibration of head-to-flange temperature indicator. ?

12 - Test all electrical penetrations and main steam line bellows.

13 - Test all motors in drywell.

14 - Repair damaged paint on drywell wall.

15 - Perform functional test on all equipment exposed to drywell incident environment.

16 - Check system for freedom of movement during startup. ?

17 - Hydrostatic test of 1000 psig.

Additionally, changes to the feedwater control system to improve its operation are under consideration. When our investigation is complete, we will file a final report with you.

Very truly yours,

Byron Lee Jr.

Byron Lee, Jr.
Assistant to the President

cc: Mr. Boyce H. Grier, Director
Region III Compliance

SEQUENCE OF EVENTS

DECEMBER 8, 1971 INCIDENT

Initial Conditions

a) Load - 792 MWe; approximately 2300 MWt; the base load condition was being maintained for determination of chimney monitor sampling system plate out.

b) Steam flow - 9×10^6 lb/hr.
Feedwater flow - 9×10^6 lb/hr.

c) Equipment in-service - 3B, 3C, 3D condensate booster pumps (3A pump was out of service for alignment and gland leak); 3A & 3C reactor feed pumps (3B pump in standby); 3B feedwater regulating valve in service (3A feedwater regulating valve in standby); low flow feedwater regulating valve open approximately 20%.

d) Drywell - inerted and at normal pressure of approximately 0.25 psig.

NOTE: A graph of significant parameters during the initial transient is attached.

Sequence of Events

1) 14:13:08

3C condensate-booster pump tripped.

NOTE: Cause of trip unknown. The pump can only be tripped on under-voltage or overcurrent. Prior to the pump trip, Commonwealth Edison Company and General Electric Company personnel were present in the condensate pump room working on a circulating water hydraulic unit. They noted a very loud noise coming from 3C pump. It was presumed the pump tripped on overcurrent as a result of mechanical binding. Neither the overcurrent or under-voltage relays at the pump breaker showed a trip target. It has not been determined if the condensate or the booster pump was the cause of the noise.

2) 14:13:09

3A and 3C reactor feed pumps tripped on low suction pressure.

3) 14:13:10

Standby reactor feed pump 3B started.

4) 14:13:21 - 22

a) Reactor water level decreased rapidly.

b) Reactor scrammed on low water level.

c) Reactor water level continued downward rapidly to approximately - 20". (ECCS Initiation is -59".)

d) The operator observed the following indications:

1) The "Flow On" light on the benchboard, indicating a runout condition, blinked on and off.

2) The "Feedwater Pump Max Capacity" annunciator came up. This is also an indication of a runout condition.

3) The "Loss of Air to F. W. Reg. Valve A" and "Loss of Air to F.W. Reg. Valve B" annunciators. This alarm occurs at 85 psig. Lockout occurs until 75 psig.

e) Vessel level started to increase and the operator took manual action between -20" and -12" in anticipation of a rapid level increase, as follows:

1) Reduced "master feedwater controller" set point, to close the feedwater regulating valve.

2) Reduced manual output control potentiometers on "manual/auto" controllers to zero.

3) Transferred from "auto" to "manual" on "manual/auto" stations.

4) Manually closed low flow feedwater regulating valve.

f) At -12" vessel level hesitated and the operator started opening the low flow F.W. valve to increase level.

g) Reactor water level began to increase rapidly.

h) As soon as the operator verified that level was increasing, he closed the low flow feedwater valve again. (Estimated at approximately zero inches reactor water level)

i) As vessel level came through zero, the operator started closing the feedwater regulating valve motor operated isolation valve, again in anticipation of a rapid level increase.

j) As the F.W. isolation valve closed, the feedwater flow was reduced from 5.7×10^6 to 2.3×10^6 lb/hr. At some time during the closure of this valve, it stalled due to high differential pressure. Flow levelled out at approximately 2.3×10^6 lb/hr.

5) 14:14:13

Main Steam Line Press. low

Group I Isolation on Main Steam Line Low Press.

Main Steam Isolation Valves closed.
Main Steam Line Drains closed.
Recirc. Loop Sample Line closed.
Isolation Condenser Vent to Main Steamline closed.

6) 14:14:14

Reactor pressure reached a low point of 795 psig after Main Steam Line isolation.

7) 14:14:29 - 32

a. Low reactor water level trip reset (+17")

b. Level continued up to 80" and steam line started filling.

NOTE: It is surmised that when the standby reactor feed pump started, flow increased to the point where the feedwater regulating valve went into a runout condition. (Flow control mode). This is substantiated by the fact that the pump flow corresponded to a runout flow control point of approximately 5.7×10^6 lb/hr., and the observation of the annunciator which indicates a runout condition. The operator stated that the "Flow On" light on the benchboard, which would also indicate a runout condition, only blinked on and then off (The "Flow On" light should have been on steady for the runout condition). At some time during the reactor level transient "3B" feedwater regulating valve locked out in an open position. Subsequent testing has shown that the lockout condition was probably caused by low air pressure which resulted from rapid movement of the valve. It is believed that the lockout occurred when the runout condition was reset on increasing level. At this time, the valve tried to close rapidly causing a lockout on low pressure.

c. Turbine tripped on high reactor water level.

d. Operator broke condenser vacuum manually.

e. On increasing reactor pressure, the operator put the isolation condenser in service. At this time, the reactor level was above the isolation condenser supply line and the operator noted very little effect on pressure with it in operation.

8) 14:18:19

a) Steam line filled and pressure increased to 1020 psig.

b) Safety valve "3F" lifted.

c) High drywell press. (2 psig) was reached.

d) Diesel generator 3 and 2/3 started.

e) Core Spray Pumps started.

f) LPCI pumps started.

g) Reactor Recirculation Pumps tripped.

h) HPCI received an initiation signal, but tripped on high level.

i) Containment isolation. (Group II)

9) 14:18:39

Drywell press. - 5 psig. Drywell pressure continued to increase and peaked at 20 psig, then began to decrease.

It is estimated that the safety valve remained open for approximately 1½ minutes, based on drywell pressure information and reactor vessel pressure data.

Following closure of the safety valve, water level continued to increase in the reactor vessel due to feedwater input at a rate of approximately 2.3×10^6 lb/hr.

Reactor pressure increased gradually due to decay heat input and increased vessel water inventory.

10) 14:26:13

a) Reactor feedwater "3B" pump tripped manually by operator.

b) Reactor water level at 130".

11) 14:20

Suppression chamber cooling placed in service.

12) 14:35

Drywell pressure had decreased to 13.5 psig.

13) 15:17

Drywell pressure had decreased to 10.5 psig.

14) 18:00*

Drywell pressure has decreased to 4.5 psig.

15) 18:00*

Installed jumper to allow opening the reactor water sample valves.

NOTE: All systems were maintained in a "bottled-up" condition until sampling of reactor water and containment environment could be accomplished.

16) 19:16

Reactor water sample collected. Analysis indicated the activity to be normal (2.7×10^{7} p C/l.)

17) 20:00*

Drywell pressure had decreased to 4.0 psig.

18) 20:00*

Following reactor water analysis, the cleanup system was placed in service and reactor blowdown was established for water level control. At this point, the level had reached approximately 145" due to control rod drive cooling water input.

December 10, 1971

19) 04:00*

A drywell atmosphere sample was obtained in preparation for venting and entry. The results were:

I 131	1.3×10^{-9} uc/cc
Beta-Gamma	6.3×10^{-11} uc/cc
Alpha	1.1×10^{-14} uc/cc

20) 07:15

Started purging the drywell to the Reactor Building Ventilation Stack in accordance with station operating procedures. Reactor water temperature was 166° F.

21) 10:45

Initial entry into the drywell was made to obtain atmosphere samples. Oxygen concentration was satisfactory. Analysis of airborne activity:

I 131	8.5×10^{-10} uc/cc
Beta-Gamma	2.3×10^{-11} uc/cc
Alpha	1.8×10^{-12} uc/cc

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799 ROOSEVELT RD GLEN ELLYN ILL

DPR-25 DRESDEN NUCLEAR POWER STATION UNIT 3. THIS WILL CONFIRM A CONVERSATION AT 1945 HOURS THIS DATE WITH MR E JORDAN REGARDING AN APPEARANCE AT 1413 HOUR ON DECEMBER 8 1971 WHICH RESULTED IN A PRIMARY CONTAINMENT PRESSURE OF 20 PSIG. THE UNIT WAS OPERATING AT 792 MWE STEDY STATE WHEN A CONDENSATE-BOOSTER PUMP TRIPPED. THIS CAUSED LOSS OF THE FEEDWATER PUMPS ON LOW SUCTION PRESSURE AND A RESULTANT LOW REACTOR WATER LEVEL SCRAM. APPROXIMATELY 50

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SECONDS AFTER THE SCRAM THE MAIN STEAM ISOLATION VALVES CLOSED ON THE LOW STEAM LINE PRESSURE. WATER LEVEL ROSE RAPIDLY IN THE REACTOR VESSEL DURING THIS TIME UNDER CONTROL OF THE ~~SPEED~~ FEED WATER CONTROL SYSTEM AND REACHED THE LEVEL OF THE STEAM LINE. AT ABOUT 1418 HOURS PRIMARY CONTAINMENT PRESSURE REACHED TO TWO PSIG AND ALL REQUIRED ECCS SYSTEMS STARTED. CONTAINMENT PRESSURE CONTINUED TO RISE, PEAKED AT 20 PSIG, AND BEGAN TO DROP. TORUS SPRAYS WERE PLACED IN SERVICE AT THIS POINT TO ASSIST IN THE HEAT REMOVAL FROM THE CONTAINMENT. AT 2000 HOURS CONTAINMENT PRESSURE WAS 3.8 PSIG AND DECREASING. VENTING OF THE PRIMARY CONTAINMENT HAS NOT BEEN EMPLOYED. SERVAILANCE

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