(7-77)

LICENSEE EVENT REPORT

CONTROL BLOCK: | | | | | | | (1) (PLEASE PRINT OR TYPE ALL REQUIRED INFORMATION)

0 1 | LICENSEE CODE: P A B V S 1 (2) | LICENSE NUMBER: 0 0 - 0 0 0 0 0 - 0 0 (3) | LICENSE TYPE: 4 1 1 1 1 (4) | CAT (5)

CON'T 0 1 | REPORT SOURCE: L (6) | DOCKET NUMBER: 0 5 0 0 0 3 3 4 (7) | EVENT DATE: 0 5 0 2 8 1 (8) | REPORT DATE: 0 5 1 5 8 1 (9)

EVENT DESCRIPTION AND PROBABLE CONSEQUENCES (10)

0 2 With the reactor at 100% power and a moisture carryover test in progress, high feed
0 3 flow and rising level in the "A" steam generator were accompanied by vibrations in
0 4 the "A" feed line. A load reduction was commenced prior to reactor trip at 83%
0 5 power due to high-high level in the "A" steam generator which occurred after the
0 6 third such vibration. After the trip, excess steam pressure was relieved to the
0 7 atmosphere, resulting in a calculated release concentration less than .001% of the
0 8 10 CFR 20 limits for Na-24, the moisture carryover tracer.

0 9 | SYSTEM CODE: C H (11) | CAUSE CODE: E (12) | CAUSE SUBCODE: B (13) | COMPONENT CODE: V A L V E X (14) | COMP. SUBCODE: X (15) | VALVE SUBCODE: G (16)

(17) LER/RO REPORT NUMBER | EVENT YEAR: 8 1 | — | SEQUENTIAL REPORT NO.: 0 3 2 | / | OCCURRENCE CODE: 0 1 | REPORT TYPE: T | — | REVISION NO.: 0

ACTION TAKEN: D (18) | FUTURE ACTION: Z (19) | EFFECT ON PLANT: A (20) | SHUTDOWN METHOD: C (21) | HOURS (22): 0 2 1 1 | ATTACHMENT SUBMITTED: Y (23) | NPRD-4 FORM SUB.: N (24) | PRIME COMP. SUPPLIER: A (25) | COMPONENT MANUFACTURER: C 6 3 5 (26)

CAUSE DESCRIPTION AND CORRECTIVE ACTIONS (27)

1 0 The feedwater transient was due to the malfun..ion of the "A" feed regulating valve
1 1 due to a disconnected actuator to positioner feed back linkage and a fractured valve
1 2 stem. The valve stem was replaced, the valve was rebuilt, and actions recommended
1 3 by the valve vendor were implemented on all three main feed regulating valves.
1 4

1 5 | FACILITY STATUS: X (28) | % POWER: 1 0 0 (29) | OTHER STATUS (30): Special Test | METHOD OF DISCOVERY: A (31) | DISCOVERY DESCRIPTION (32): Alarm annunciation

1 6 | ACTIVITY RELEASED: M (33) | CONTENT OF RELEASE: P (34) | AMOUNT OF ACTIVITY (35): 12.7 pCuries total | LOCATION OF RELEASE (36): Steam generator to the atmosphere

PERSONNEL EXPOSURES

1 7 | NUMBER: 0 0 0 (37) | TYPE: Z (38) | DESCRIPTION (39): N/A

PERSONNEL INJURIES

1 8 | NUMBER: 0 0 0 (40) | DESCRIPTION (41): N/A

LOSS OF OR DAMAGE TO FACILITY (43)

1 9 | TYPE: D (42) | DESCRIPTION: Damage to "A" feed line snubbers and supports

PUBLICITY

2 0 | ISSUED: Y (44) | DESCRIPTION (45): Press release on May 2, 1981 | NRC USE ONLY

NAME OF PREPARER W. S. Lacey PHONE: 412-643-8525

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Attachment To LER 81-32/01T
Beaver Valley Power Station
Duquesne Light Company
Docket No. 50-334

At about 0041 hours on May 2, 1981, the Reactor Operator (RO) began to reduce Tavg from 576F to 571F as per procedure BVT 1.2 - 2.21.1, "Steam Generator Moisture Carryover Test". After completing the reduction in Tavg, turbine load was being increased to bring the reactor back up to an indicated 100% power for the test.

At approximately 0157 hours, the Plant Operator (PO) acknowledged a feed flow/stem flow mismatch alarm. He noted that the feed flow on the "A" steam generator had gone off scale high on [FR-MS-478] and that the "A" steam generator had a rapidly increasing level. At about this time, the plant experienced a feedwater vibration. The PO took manual control of the "A" main feed regulating valve [FCV-FW-478] and attempted to reduce feed flow to restore proper level in the steam generator.

A power reduction was commenced at 2% per minute while attempting to stabilize "A" steam generator level, in an effort to transfer control from the main feed regulating valve to the "A" feed regulating bypvass valve [FCV-FW-479]. Approximately 1 or 2 minutes after the first vibration, the plant experienced a second feed water vibration while the PO was having problems controlling flow and level to the "A" steam generator.

Operators were sent throughout the plant to determine what, if any, were the effects of the second feed water vibration. A third feedwater vibration was subsequently experienced and line movement was observed by operators in the feed regulating valve room. This vibration was followed shortly by a reactor trip, at 0208 hours, on high-high level in the "A" steam generator. At the time of the trip, reactor power had been reduced to approximately 83%.

After the reactor trip, the Reactor Operator (RO) followed Emergency Operating Procedure E-5, "Reactor Trip". The primary plant behavior was normal and normal pressurizer level and pressure was restored.

The auxiliary feed water pumps were used to restore normal steam generator level. A main feed pump was then started in an attempt to feed the steam generators via the bypass feed control valves. After the auxiliary feed pumps were shut down, decreasing "A" and "C" steam generator levels and zero feed flow were observed. Normal levels were restored using an auxiliary feed water pump.

Meter and Control Repairmen (MCRs) were sent to the feed bypass valves to determine the cause and repair the valves if possible. They discovered that the control air supply lines to the "A" and "C" valves [FCV-FW-479] and [FCV-FW-499] had been broken off. The air lines were repaired but the "A" valve still would not work due to a failed actuator piston seal.

Following the reactor trip, steam pressure in the generators increased to an indicated value of approximately 1080 psig as observed on [PR-MS-474]. At this time, it was suspected that the atmospheric dump or safety valves opened. This pressure increase above normal no-load pressure (1005 psig) was due to the slow response of the condenser steam dump system. About 20 minutes after the reactor trip, steam generator pressure again increased to approximately 1080 psig, possibly reopening the atmospheric dump/safety valves. While the pressure was increasing the third time, the PO switched to the Steam Pressure control mode. The dump system responded properly to control steam generator pressure in this mode. An investigation revealed that the slow condenser steam dump response was due to a failed summator in the condenser dump control loop.

At 0330 hours, The Shift Supervisor initiated the Emergency Preparedness Plan (EPP) and declared that an "Unusual Event" had occurred, under the understanding that a steam generator atmospheric dump/safety valve had opened, causing an unplanned release to the atmosphere of Na-24, a radioactive tracer used in the moisture carryover test. The release concentration was calculated to be 1.0×10^{-13} μCi/cc and the EPP was terminated at 0420 hours. The health and safety of the general public was not jeopardized since the calculated release was less than .001% of 10 CFR 20 limits for Na-24.

No problems were experienced with primary plant systems during this event and there were no releases of radioactivity from the primary plant.

Damages believed to result from the feed water vibrations experienced were confined primarily to the "A" feed line. The only related damage to the "B" feed line is the failure of flow transmitter [FT-FW-486]. The "C" feed line damage was confined to the severed instrument air line on the bypass valve, [FCV-FW-499].

"A" feed line damages were confined to snubber and pipe support damage inside containment, except for the broken instrument air line to the bypass valve, [FCV-FW-479]. The snubber damage involved a broken extension rod and cracked bushing. Remaining support damage was limited to loose bolts and shims shaken from a pipe whip restraint. All were repaired.

The malfunction of the "A" main feed regulating valve [FCV-FW-478] was the cause of the event. This malfunction was caused by a disconnected feed back linkage to the valve positioner, which resulted in a full open positioner demand when a load increase initiated, and a fractured valve stem, which resulted in valve plug motion due to hydraulic forces rather than actuator force.

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Analysis of the valve stem by Westinghouse Electric Corporation concluded that the valve stem was fractured for some time prior to the incident and that the stem fracture was due to fatigue loading - this analysis was conducted on a crack found approximately one inch from the point of fracture. It is believed that in the time period prior to the incident the fractured valve stem was held together by the stem locking anti-rotation device installed on the valve stem at the fracture point.

Further analysis by Westinghouse and the valve vendor, Copes Vulcan, revealed that no similar failures had occurred previously in these valves as long as the valve trim was manufactured by the valve vendor. It was noted that similar failure had occurred with valves whose trim was manufactured by someone other than the vendor.

It was the opinion of the valve vendor that stem failure was enhanced by increased lateral movement due to the fact that the stem locking anti-rotation device was not properly bolted to the valve actuator yoke. This device is intended for damping stem lateral movement near the yoke.

Analysis by Westinghouse determined that no unreviewed safety question was involved since only a calculated value of 5000 lbf was exerted on the feed piping based on an 80% flow change in 1.5 seconds.

All three main feed regulating valves were disassembled, inspected, and rebuilt. The "A" and "C" line valve stems were replaced (the "C" valve stem was found to have a hairline crack). The stem locking anti-rotation devices were properly installed as instructed by the valve vendor and all valve actuator to positioner linkage connections were sealed with Loc-tite.