CENTRAL FILES

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION I
631 PARK AVENUE
KING OF PRUSSIA, PENNSYLVANIA 19406

NOV 16 1979

Docket No. 50-272

Public Service Electric and Gas Company
ATTN: Mr. F. W. Schneider
Vice President - Production
80 Park Place
Newark, New Jersey 07101

Gentlemen:

The enclosed IE Information Notice No. 79-27 provides information with regard to the sequence of events that followed incidents involving steam generator tube ruptures at two PWR units. If you have any questions regarding this matter, please contact this office.

Sincerely,

James M. Allan

for Boyce H. Grier
Director

Enclosures:
1. IE Information Notice No. 79-27
2. List of Recently Issued Information Notices

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ENCLOSURE 1

SSINS No.: 6870
Accession No.:
7910250488

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON D.C. 20555



# STEAM GENERATOR TUBE RUPTURES AT TWO PWR PLANTS

## Description of Circumstances:

In recent months two incidents involving steam generator tube ruptures have occurred. In both instances, the units were cooled down and placed in the residual heat removal mode with existing procedures.

## Event of June 25, 1979 at the Doel 2 Nuclear Power Plant in Belgium

The first event occurred on June 25, 1979, at the Doel 2 nuclear power plant in Belgium. The Doel unit is a 390 Mwe Westinghouse two-loop reactor. The event consisted of a rupture of several tubes in the loop B steam generator. The resultant leakage between the primary and secondary systems was estimated to be 125 gpm. The event started when the plant was heated up after a shutdown caused by a malfunction of the main steam isolation valve. At the time of the incident the primary coolant pressure was: 2233 psi and the temperature: 491°F. The reactor remained subcritical throughout the event.

The first indication of abnormal behavior was a rapid decrease of the primary system pressure (approximately: 28 psi/min.). This was followed by the sequence of events listed below:

| | | Time, min. |
|---|---|---|
| 1. | Increase of charging flow demand, requiring startup of a second charging pump. | 1.8 |
| 2. | Automatic isolation of the CVCS letdown line. | 2.4 |
| 3. | Shut off of the pressurizer heaters due to low liquid level in the pressurizer. | 2.4 |

| | | Time, min. |
|---|---|---|
| 4. | Closing of block valves in the pressurizer relief line. | 4.6 |
| 5. | Rapid increase of water level in the damaged steam generator (loop B). The steam generator was isolated. | 9.4 |
| 6. | Startup of the third charging pump and realignment of the suction of all charging pumps from the CV tank to the refueling water storage tank. | - |
| 7. | Shut off of the main coolant pump in loop B. This was done in order to reduce heat generation in the primary coolant system. | 17.4 |
| 8. | Safety Injection Signal on low pressure in pressurizer followed by: startup of diesels, containment isolation, and high pressure safety injection, resulting in increase of the primary system pressure. | 19.2-19.5 |
| 9. | Manual startup of the pressurizer spray in an attempt to decrease primary system pressure. | 28 |
| 10. | Pressurizer fills up solid with water. Level indicator off scale. There was no release of primary coolant from the pressurizer because the block valve was closed and the pressurizer did not exceed safety valve settings. | 33 |
| 11. | Automatic startup of auxiliary feedwater flow to both steam generators. | 44 |
| 12. | Flow of auxiliary feedwater to the damaged steam generator is stopped. | 50 |
| 13. | Beginning of depressurization of the primary coolant system. SI pumps are stopped and the isolation valves in the CV letdown line are opened. | 60-88 |
| 14. | Startup of the residual heat removal system. | 195 |

## Discussion

The operator's action during the accident were directed towards:

a. maintaining primary coolant subcooled,
b. minimizing leakage rate between the primary and secondary coolant system.
c. preventing radioactive fluid from escaping from the damaged steam generator.

Sufficiently high degree of subcooling in the primary coolant system was achieved by reducing heat generation in the primary system (switching off one (1) main coolant pump "B") and by controlling, to the extent possible, primary coolant pressure.

Two actions were taken to prevent radioactive fluid from escaping from the leaky steam generator. As soon as the leak was detected, the secondary side of the steam generator was isolated and the setpoints of the safety valves were raised to their maximum value.

In general, the accident was handled in accordance with the existing procedures and no radioactive releases or equipment damage was experienced.

All safety systems functioned as designed with exception of the air operated valves in the CV letdown line and in the line to the cooling system of the main pump thermal shields. The cause of this problem was that the containment isolation signal interrupted the supply of compressed air to these valves and rendered them inoperative until the air was manually restored. This malfunction of the valves resulted in a delay of primary system cooldown and depressurization (item 13) and caused the primary coolant pumps to operate for a while without proper cooling. However, none of these events produced any detrimental consequences.

Conclusions

The accident was successfully terminated using the presently existing procedures which, with only one exception, proved to be adequate. In the future, the procedure dealing with containment isolation will have to be revised.

The leak was reported to be located in the U-Bend of the first row tubes. The suspected cause was stress corrosion due to ovalization of the short bend radius tubes.

Event of October 2, 1979 at the Prairie Island 1 Nuclear Power Plant

The second event occurred on October 2, 1979 at Prairie Island Nuclear Generating Plant Unit No. 1, a 530 Mwe Westinghouse two-loop reactor. The event consisted of mechanical wear due to a foreign object until a tube failure occurred in the "A" Steam Generator; the resultant leakage was calculated to be about 390 gpm. At the time of the incident, the plant was operating at 100% power. The following information was taken from the licensee's event report No. 79-27 dated October 16, 1979 and from NRC inspections of the event.

| Date | Time (CDT) | Event |
|---|---|---|
| Oct 2 | 1414 | High Radiation alarm on the air ejector discharge gaseous radiation monitor |
| | 1420 | Overtemperature ΔT Turbine Runback due to decreasing pressure (Maximum rate was approximately 100 psi/minute.) |
| | 1421 | Low Pressurizer pressure (< 2139.9 psig) |
| | 1421 (approx) | Commenced load reduction |
| | 1422 | Low pressurizer level (< 18.3%) |
| | 1423 | Started second charging pump (#11) |
| | 1424 (approx) | Started third charging pump (#13) |
| | 1424:09 | Reactor trip for "Low Pressurizer Pressure" (< 1900 psig) |
| | 1424:14 | Safety injection (SI) occurred due to "Low Pressurizer Pressure (< 1815 psig) |
| | 1424:33 | Minimum RCS water inventory; RCS pressure begins increasing |
| | 1426 | 11 Reactor Coolant Pump stopped |
| | 1427 | 12 Reactor Coolant Pump stopped |
| | 1430 | Emergency Alert declared |
| | 1432:29 | 11 Steam Generator level increased above the "Lo Lo Level" setpoint (13%) on the narrow range after having gone off-scale low after the trip (It is normal for SG Level to go offscale low on a trip; recovery in this case was much more rapid than usual) |

| Date | Time | Event |
|---|---|---|
| | 1438 | SI Reset |
| | 1441 | Loop A MSIV closed to isolate No. 11 Steam Generator |
| | 1456 | Pressurizer Level returned on scale |
| | 1456 | Stopped 12 SI pump |
| | 1456-57 | Began depressurization of the RCS using the pressurizer PORV. (The valve was cycled 6 to 8 times to reduce pressure to required value) |
| | 1500 (approx) | Site Emergency declared |
| | 1502 | Pressurizer level reached the high level setpoint (> 55%) |
| | 1506 | 11 SI Pump stopped |
| | 1507 | Pressurizer Relief Tank rupture disc relieved |
| | 1515 | RCS pressure at 910 psig (same as 11 SG pressure ) Leak apparently stopped |
| | 1550 | Commenced normal cooldown |
| | 2200 | Site Emergency terminated. |
| Oct 3 | 0640 | RHR placed in service to continue cooldown to cold shutdown |
| | 1300 | RCS at cold shutdown |

The radiological aspects of the event are summarized below:

RADIOACTIVE RELEASED FROM THE PLANT

Airborne

The monitor on the exhaust of the steam jet air ejectors (SJAE) alarmed at 1514 hours EDT about 10 minutes prior to the reactor trip. The monitor was off-scale shortly thereafter; the highest range of the monitor is equivalent to approximately 0.004 Ci/sec release rate at an exhaust flow of about 20 cfm. The monitor was thought to have been filled with water.

Based on the initial full-scale reading of the SJAE monitor, and analysis of several grab samples taken from the SJAE exhaust, it is estimated that approximately 30 curies of noble gases (primarily xenon) were released throughout the incident with the majority of the release being within the first 2 hours. No iodine levels were measured.

The airborne releases do not appear to have exceeded the applicable Technical Specification limit (120 Ci/Hr) on maximum allowable release rate averaged over an hour period. The release rate decreased after the isolation of the steam generator, continuing to decrease with time. After the first hour the release rate was ∿ 0.002 Ci/sec and was in the range of 2-500 μCi/sec after the second hour.

Liquid

Analysis of samples of water from the turbine building sumps showed only one isotope detectable, Xe-133 at the concentration of ∿ $5 \times 10^{-5}$ uCi/ml. During the course of the incident, water was pumped from the sumps for offsite release at a rate of about 250 gallons per minute for approximately 3 minutes, resulting in a total release of about 140 uCi of noble gases (Xe-133) dissolved in water. No regulatory limits were exceeded for this release, considering an MPC of about $2 \times 10^{-4}$ uCi/ml normally used for noble gases dissolved in water.

OFFSITE RADIOLOGICAL IMPACT

During the first 4 hours after the steam generator tube rupture, the winds were blowing generally from the east to the west. Using site meterological data, the dispersion factor (X/Q) at the site boundary was estimated to be $4 \times 10^{-5}$

sec/$M^3$. Conservatively assuming the total estimated release of ~30 curies of noble gases over the 4-hour period, the dose to an individual continuously present at the site boundary would be about 0.05 millirem, slightly above the normal background dose rate.

After the first 2 hours, the release rate had dropped to the point where calculated dose rates offsite were well below natural background radiation levels.

Environmental surveys were carried out by licensee and State teams operating out to a distance of about 5 miles from the site. Air samples and direct radiation surveys made by these survey teams yielded negative results (i.e., background readings). Surveys performed by the NRC inspectors at the site confirmed the licensee and State results.

At ~ 2000 hours EDT, the State of Minnesota conducted an aerial survey over the site at altitudes from 400 to 2000 feet. The survey detected only background levels using a portable survey instrument (CDV-700).

RADIOACTIVITY IN THE PLANT

Area direct radiation monitors in the plant and direct radiation surveys showed no significant increase in radiation levels.

Analysis of air samples taken in the turbine building showed concentrations of krypton and rubidium daughters in the range of $10^{-10}$ to $10^{-9}$ uCi/cc (MPC of $10^{-6}$ uCi/cc) and xenon at a concentration of $10^{-6}$ uCi/cc (MPC of $10^{-5}$ uCi/cc).

The direct radiation monitor in containment (instrument seal table) showed no increase after the trip (~2 mrem/hr). The noble gas monitor in containment increased by a factor of ~10 (from 1000 to 12,000 cpm) indicating 3 X $10^{-3}$ uCi/cc gaseous activity in containment.

PLANT PERSONNEL EXPOSURES

No personnel overexposures resulted from the occurrence. A total of about 200 plant contractor personnel were involved in evacuation from the site as a result of the declaration of a site emergency condition. These personnel were working in the auxiliary building and turbine building. All personnel had been "badged" with personnel monitors and were surveyed for contamination before they departed the site.

Cause of Event

Licensee examination of the steam generator tube determined that a single tube (out of 3388 in the steam generator) had ruptured. The size of the rupture was 2 inches long and 3/8 inches wide in the wall of the 7/8-inch diameter tube.

Plant personnel found a coil spring lodged near the ruptured tube. The spring apparently had rubbed against the tube during operation, causing the tube to wear away and eventually rupture. An adjacent tube was also worn by the spring vibration.

The spring is believed to have been part of a hose used to loosen and remove sludge products from the tube support sheet during an early refueling outage.

Action Taken to Prevent Recurrence

The ruptured tube, the additional worn tube and surrounding tubes have been plugged. The spring has been removed from the steam generator.

The licensee has completed eddy current examination of approximately 6 per cent of the tubes in the steam generator with failed tubes and approximately 3 per cent of the second Unit 1 steam generator. Both steam generators were examined to assure there are no other visible objects that could cause tube damage. While in both events a cold shutdown was achieved with existing procedures, there was a common concern expressed on the effects of isolating the air supplies to valves inside containment on the maintenance of reactor coolant inventory and pressure.

This IE Information Notice is provided as an early notification of a possibly significant matter that is still under review by the NRC staff. It is expected that recipients will review the information for possible applicability to their facilities. No specific action or response is requested at this time. If NRC evaluations so indicate, further licensee actions may be requested or required.

No written response to this IE Information Notice is required. If you have any questions regarding this matter please contact the Director of the appropriate NRC Regional Office.

ENCLOSURE 2



RECENTLY ISSUED INFORMATION NOTICES

| Information Notice No. | Subject | Date Issued | Issued to |
|---|---|---|---|
| 79-19 | Pipe Cracks in Stagnant Borated Water Systems at PWR Plants | 7/17/79 | All power reactor facilities with an OL or CP |
| 79-20 | NRC Enforcement Policy - NRC Licensed Individuals | 8/14/79 | All Holders of Reactor OLs and CPs and Production Licensees with Licensed Operators |
| 79-20 (Revision No. 1) | Same Title as 79-20 | 9/7/79 | Same as 79-20 |
| 79-21 | Transportation and Commercial Burial of Radioactive Material | 9/7/79 | All power and research reactors with OLs |
| 79-22 | Qualification of Control Systems | 9/14/79 | All power reactor facilities with an OL or CP |
| 79-23 | Emergency Diesel Generator Lube Oil Coolers | 9/26/79 | All power reactor facilities with an OL or CP |
| 79-24 | Overpressurization of Containment of a PWR Plant After a Main Steam Line Break | 10/1/79 | All power reactor facilities with an OL or CP |
| 79-25 | Reactor Trips at Turkey Point Unit 3 and 4 | 10/1/79 | All Power Reactor Facilities with an OL or CP |
| 79-26 | Breach of Containment Integrity | 11/5/79 | All Power Reactor Facilities with an OL or CP |
| 79-12A | Attempted Damage to New Fuel Assemblies | 11/9/79 | All Fuel Facilities, Research Reactors and Power Reactors with an OL or CP |