

SSINS No.: 6870
Accession No.:
7910250488

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

November 16, 1979

IE Information Notice No. 79-27

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:

	<u>Time, min.</u>
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

	<u>Time, min.</u>
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 the 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.

<u>Date</u>	<u>Time (CDT)</u>	<u>Event</u>
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)

<u>Date</u>	<u>Time</u>	<u>Event</u>
	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:

RADIOACTIVITY RELEASED FROM THE PLANT

Airborne

The monitor on the exhaust of the steam jet air ejectors (SJAЕ) 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 μ Ci/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 $\sim 5 \times 10^{-5}$ 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 2×10^{-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 meteorological data, the dispersion factor (X/Q) at the site boundary was estimated to be 4×10^{-5}

sec/M³. 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 3×10^{-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 percent of the tubes in the steam generator with failed tubes and approximately 3 percent 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.

LISTING OF IE INFORMATION NOTICES
ISSUED IN THE LAST SIX MONTHS

Information Notice No.	Subject	Date Issued	Issued To
79-27	Steam Generator Tube Ruptures at Two PWR Plants	11/16/79	All power reactor facilities holding OLs and CPs
79-26	Breach of Containment Integrity	11/5/79	All power reactor facilities holding OLs and CPs
79-25	Reactor Trips at Turkey Point Units 3 and 4	10/1/79	All power reactor facilities holding OLs and CPs
79-24	Overpressurization of Containment of a PWR Plant After a Main Steam Line Break	10/1/79	All power reactor facilities holding OLs and CPs
79-23	Emergency Diesel Generator Lube Oil Coolers	9/26/79	All Holders of CPs and OLs
79-22	Qualification of Control Systems	9/17/79 9/14/79	All Holders of CPs All Holders of OLs
79-21	Transportation and Commercial Burial of Radioactive Materials	9/11/79	All Licensees as Supplemental Information to IE Bulletin Nos. 79-19 & 79-20
79-20 (Rev. 1)	NRC Enforcement Policy NRC Licensed Individuals	9/7/79	All Holders of Reactor OLs and CPs and Production Licensees with Licensed Operators
79-20	NRC Enforcement Policy NRC Licensed Individuals	8/14/79	All Holders of Reactor OLs and CPs and Production Licensees with Licensed Operators
79-19 (Correction - Enclosure)	Pipe Cracks in Stagnant Borated Water Systems at PWR Plants	7/18/79	All Holders of Reactor OLs and CPs
79-19	Pipe Cracks In Stagnant Borated Water Systems At PWR Plants	7/17/79	All Holders of Reactor OLs and CPs

LISTING OF INFORMATION NOTICES
ISSUED IN THE LAST SIX MONTHS

Information Notice No.	Subject	Date Issued	Issued To
79-18	Skylab Reentry	7/6/79	All Holders of Reactor OLS
79-17	Source Holder Assembly Damage From Misfit Between Assembly and Reactor Upper Grid Plate	6/20/79	All Holders of Reactor OLS and CPs
79-16	Nuclear Incident at Three Mile Island	6/22/79	All Research Reactors and Test Reactors with OLS
79-15	Deficient Procedures	6/7/79	All Holders of Reactor OLS and CPs
79-14	NRC Position of Electrical Cable Support Systems	6/11/79	All Power Reactor Facilities with a CP
79-12A	Attempted Damage to New Fuel Assemblies	11/9/79	All Fuel Facilities, Research Reactors, and OLS and CPs