SSINS No.: 6835
IN 86-106

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

December 16, 1986

IE INFORMATION NOTICE NO. 86-106: FEEDWATER LINE BREAK

Addressees:

All nuclear power reactor facilities holding an operating license or a construction permit.

Purpose:

This information notice is to alert addressees of a potentially generic problem with feedwater pipe thinning and other problems related to this event. Recipients are expected to review the information for applicability to their facilities and consider actions, if appropriate, to preclude similar problems occurring at their facilities. However, suggestions contained in this information notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

On Tuesday, December 9, 1986, at 2:20 p.m., both units at the Surry Power Station were operating at full power when the 18-inch suction line to the main feedwater pump A for Unit 2 failed catastrophically. Eight workers who were replacing thermal insulation on a nearby line were burned by flashing feedwater. All were transported to area hospitals. Two workers were treated and released. Four other workers subsequently died.

Units 1 and 2 are identical. In each unit, feedwater flows from a 24-inch header to two 18-inch suction lines that each supply one of two main feedwater pumps. At maximum load under normal conditions, feedwater flow through each pump is 5 million lb/hr. Feedwater temperature, pressure, and enthalpy are 370°F, 450 psig, and 346 Btu/lb, respectively. At these conditions the fluid is in the single phase, liquid only regime. That is, the piping does not see a mixture of liquid and vapor.

The event was initiated by the main steam isolation valve on steam generator C failing closed. Because of the increased pressure in steam generator C that collapsed the voids in the water, the reactor tripped on low-low level in that steam generator. A 2-by-4 foot section of the wall of the suction line to the A main feedwater pump was blown out and came to rest in an overhead cable tray. The break was located in an elbow in the 18 inch line about one foot from the 24-inch header. The lateral reactive force generated by escaping feedwater completely severed the suction line. The free end whipped and came to rest against the discharge line for the other pump.

Steam flashing from the break and condensing in control cabinets and in open conduit piping apparently caused the fire suppression system to actuate, resulting in release of halon and carbon dioxide in the emergency switchgear room and in various cable tunnels and vaults and in the cable spreading room. Because of the volume of water and steam being released, operators isolated lines carrying high energy fluids to areas inundated by steam. Steam generator water levels were maintained with the auxiliary feedwater system, and system cooling was provided by actuating atmospheric dump valves as necessary.

The primary system responded normally to the loss of load transient with a partial loss of main feedwater. Primary coolant temperature was stabilized at 520°F and pressurizer level was recovered as it reached the low level set point. Primary pressure decreased from 2235 to 2015 psig following the reactor trip. By 2 a.m. on the following day, reactor temperature had been reduced to the point where the residual heat removal system could be put on line. The unit reached cold shutdown that morning. During the recovery effort, the operators and the plant performed as expected.

Discussion:

The pipe material is A-106B carbon steel and the elbow is 18-inch, extra stron A-234 grade WPB carbon steel. Nominal wall thickness of the suction piping is 0.500 inch. Measurements of the wall fragment demonstrated that the wall had been generally eroded to about 0.25 inch and was one of the causes of the failure. Preliminary examination of the 2-by-4 foot section of pipe blown out during the event shows the thinning to be relatively uniform except for some small localized areas. The thinnest areas are localized and appear to be about 1/16 inch thick. Some corrosion pitting is present. A preliminary micro-examination indicated that the pipe surface near the fracture had not been highly strained as with a high stress event, such as a high pressure spike in the system.

It has not been determined at this time whether a pressure spike in the system was a contributor to this event. There was no damage evident in the hanger supports to the condensate system.

Inspection revealed a disabled check valve in the discharge piping of the A main feedwater pump. This check valve was found with its seat displaced and a hinge pin missing.

On December 10, the licensee shut down Unit 1 for examination of the condition of feedwater piping. Inspection of the Unit 1 feedwater piping shows wall thinning similar to but not as severe as that in Unit 2.

The NRC dispatched an augmented investigation team (AIT) to the site . The AIT includes a metallurgist and a water hammer analyst.

The NRC will issue additional information as more inspection and analysis is completed.

No specific action or written response is required by this information notice. If you have questions about this matter, please contact the Regional Administrator of the appropriate NRC regional office or this office.

Edward L. Jordan, Director
Division of Emergency Preparedness
and Engineering Response
Office of Inspection and Enforcement

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Attachment:
List of Recently Issued IE Information Notices

Attachment 1
IN 86-106
December 16, 1986

LIST OF RECENTLY ISSUED IE INFORMATION NOTICES

Information Notice No.	Subject	Date of Issue	Issued to
86-105	Potential for Loss of Reactor Trip Capability at Intermediate Power Levels	12/19/86	All holders of OL or CP for PWR or BWR
86-104	Unqualified Butt Splice Connectors Identified in Qualified Penetrations	12/16/86	All pressurized and boiling-water reactor facilities holding an OL or CP
86-14 Supplement 1	Overspeed Trips Of AFW, HPCI, And RCIC Turbines	12/17/86	All power reactor facilities holding an OL or CP
86-103	Respirator Coupling Nut Assembly Failures	12/16/86	All power reactor facilities holding an OL or CP and fuel facilities
86-102	Repeated Multiple Failures Of Steam Generator Hydraulic Snubbers Due To Control Valve Sensitivity	12/15/86	All power reactor facilities holding an OL or CP
86-101	Loss Of Decay Heat Removal Due To Loss Of Fluid Levels In Reactor Coolant System	12/12/86	All PWR facilities holding an OL or CP
86-100	Loss Of Offsite Power To Vital Buses At Salem 2	12/12/86	All PWRs or BWRs holding an OL or CP
86-99	Degradation Of Steel Containments	12/8/86	All power reactor facilities holding an OL or CP
86-21 Sup. 1	Recognition Of American Society Of Mechanical Engineers Accreditation Program For N Stamp Holders	12/4/86	All power reactor facilities holding an OL or CP
86-98	Offsite Medical Services	12/2/86	All power reactor facilities holding an OL or CP

OL = Operating License
CP = Construction Permit