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IN 86-102

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, DC 20555

December 15, 1986

IE INFORMATION NOTICE NO. 86-102: REPEATED MULTIPLE FAILURES OF STEAM GENERATOR HYDRAULIC SNUBBERS DUE TO CONTROL VALVE SENSITIVITY

Addressees:

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

Purpose:

This notice is provided to alert recipients of a potentially significant safety problem pertaining to recent events in which the steam generator hydraulic snubbers failed to meet their bleed and lockup specifications at two consecutive refueling outages. The primary cause appears to be control valve sensitivity to low hydraulic fluid flow velocity. It is expected that recipients will review the information for applicability to their facilities. However, suggestions contained in this notice do not constitute NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances:

On January 7, 1986, the Portland General Electric Company reported [Licensee Event Report (LER) 85-13] multiple snubbers which failed to meet their bleed and lockup specifications at its Trojan Nuclear Plant. The report, and its supplement dated April 1, 1986, identified three areas of multiple snubber failures that were discovered during the 1985 refueling outage that began in May, 1985. These snubber failures were discovered as a direct result of the expanded inservice testing program which was instituted in accordance with a recent change to the plant's technical specifications. The prior inservice inspection program had not required the testing of these snubbers.

The 16 steam generator hydraulic snubbers at Trojan are 900-Kip Anker-Holth units. Following the failure of the first 2 steam generator snubbers to meet their bleed and lockup specifications, the remaining 14 were declared inoperable because of uncertainty regarding the time required to rebuild the snubbers following testing. All the snubbers were removed and overhauled. During the overhaul, the snubber seals were found to be degraded and the hydraulic fluid was heavily contaminated with seal material and rust. However, as discussed below, this was not the primary cause of the problem detected during the subsequent 1986 outage.

An engineering evaluation of the effect of the failed snubbers on the steam generators was not initiated during the 1985 refueling outage because of the belief that they would not have restricted normal thermal growth. This decision was based on the snubber-testing company's judgment that the foreign material in the hydraulic fluid would not have affected the normal operation of the snubbers because of the relatively large channels through which fluid would flow under thermal growth conditions. However, in the case of a seismic or other severe dynamic event, it was determined that the snubbers would have activated (i.e., locked up*) and the foreign material could have blocked the bleed orifice. Because of the manner in which the snubber control valves are hydraulically interconnected, it is the licensee's belief that this would have to occur to all four snubbers on one steam generator before they would become locked in their current position.** They would remain in this position until a load reversal allowed flow through the main valves or possibly cleared the bleed port in at least one snubber.

The revised technical specifications for testing the snubbers required testing of each snubber that had failed its test during the previous testing program. Therefore, the 16 steam generator hydraulic snubbers were again tested during the refueling outage that began in April, 1986. The results of this testing indicated 12 failures--4 with excessive drag, 4 with high bleed rates at faulted load, 2 with no bleed rate at faulted load, 1 with excessive drag and high bleed rate, and 1 with high bleed in compression and no bleed in tension. The snubbers with no bleed rate cleared themselves upon load reversal.

There also was an issue of unusual movements of the pressurizer surge line that was thought for a while to be related to the snubber problems. This is discussed in Attachment 1.

Discussion:

As a part of its corrective actions during the 1985 refueling outage, the licensee had all the steam generator snubbers overhauled. Following overhaul, the snubbers could not meet their safety analysis acceptance criteria of a

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*Note: Common snubber nomenclature uses the term "lock up" to refer to (1) that point where the main flow path is closed and all flow is forced through the bleed orifice and (2) the condition where all flow is stopped and the snubber becomes a rigid strut. To eliminate any possibility for confusion between the two meanings, the term "activated" will be used for the first definition.

** In their safety evaluation report (Steven A. Varga's May 30, 1986, letter to Bart Withers) the Office of Nuclear Reactor Regulation staff concluded that "...the likelihood of full thermal lock-up occurring would require that the various contributing factors would have to affect three or four of the hydraulic snubbers on a single steam generator."

maximum drag force of 1,000 pounds at a minimum displacement rate of 0.025 in./min. This was because the snubber activated each time the velocity approached 0.025 in./min. Following consultation with the reactor vendor, the acceptance criteria was revised and the snubbers tested satisfactorily.

Because of the reoccurring snubber failures identified during the 1986 refueling outage, the licensee contracted for a detailed root cause analysis of the snubber failures. This analysis indicated that the low activation velocity (0.025 in./min) of the steam generator snubbers caused them to activate at very low fluid velocity through the main flow port. Once the snubber had activated, all flow was forced through the bleed port. Because of its extremely small size, this port acted much like a fine sieve. Apparently the first particle of foreign material would block the port causing the snubber to lock up. Thus, although contamination of the hydraulic fluid was a contributor to the problem, it was not the primary cause.

Based on this root cause analysis, the licensee decided to continue with its previously made plan to change out the control valves on the steam generator snubbers. The new snubber control valves have a much higher activation velocity (6 to 9 in./min) which is still acceptably small compared with that expected during any significant seismic event. In addition, the new snubber control valves incorporate a widely used "self-cleaning" poppet valve design as opposed to the original spring-ball check valve design. In the new design, the bleed orifices are grooves on the main poppet valve. In this way, the bleed orifices tend to be self-cleaning whenever there is flow through the main poppet valve.

All of the Anker-Holth steam generator snubbers were initially designed with relatively low activation velocities. Therefore, they are suspected of having the same type of problems as encountered at Trojan. In addition to Trojan, three other utilities have modified their steam generator snubbers so that they have activation velocities in the 6 to 10 in./min range.

However, since the root cause of the problem is the selection of an extremely low activation velocity, as opposed to a design flaw in the snubbers themselves, the problem may not be limited to only the facilities having Anker-Holth snubbers.

Attachment 2 to this information notice describes other multiple snubber failures found at Trojan during the 1985 refueling outage.

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

E L Jordan

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Attachments:
1. Pressurizer Surge Line Movements
2. Other Multiple Snubber Failures
3. List of Recently Issued IE Information Notices

Pressurizer Surge Line Movements

In what had been (until midway through the 1985-86 fuel cycle) a separate issue, the licensee had been monitoring the unusual movements of the pressurizer surge line since 1982. A walk-down of this line at the beginning of the 1985 refueling outage revealed additional movement had occurred during the last fuel cycle. A consultant was hired to evaluate and analyze these movements, and had determined that none of the previously identified potential causes, whether singly or combined, could have produced the observed movement. However, when he was advised of the possible problems with the steam generator snubbers, his worst case analysis (i.e., all snubbers on one steam generator were locked-up) indicated that locked-up snubbers could have produced the observed movement. This discovery delayed the submittal of LER 85-13, which was being prepared at the time.

Testing associated with the root cause analysis demonstrated that the snubbers on a particular steam generator would not restrict growth of that loop unless all four snubbers lock-up because the snubber hydraulic lines were connected in parallel. In addition, based on the results of the thermal expansion monitoring program conducted during the startup from the 1986 refueling outage, the licensee has determined that most, if not all, of the observed movement of the pressurizer surge line is expected due to normal thermal transients experienced by this line during heatups and cooldowns. Based on these findings, the licensee further concluded that the most likely cause of the reactor coolant system thermal restraint was due solely to the inadequate size of the gaps between system components and associated seismic or pipe whip restraints.

Other Multiple Snubber Failures

In addition to the steam generator hydraulic snubber failures, the Trojan LER identified two other areas of multiple snubber failures. Although not the subject of this information notice, they are briefly discussed to assist in identifying all the safety-related failures discussed in the LER.

1. The first additional area of multiple snubber failures was a 25 percent overall failure rate of small mechanical snubbers [Pacific Scientific models PSA-1/4 (36 percent failure rate) and PSA-1/2 (17.6 percent failure rate)].

2. The second additional area of multiple snubber failures involved the four main steam line hydraulic snubbers (two 70-Kip and two 130-Kip Bergen-Paterson units). The snubbers were declared inoperable without testing upon discovery of the steam generator hydraulic snubber failures.

Additional discussions of multiple snubber failures can be found in IE Information Notice 84-67, "Recent Snubber Inservice Testing with High Failure Rates," LER 84-079 for San Onofre Nuclear Generating Station Unit 2 (dated January 25, 1985, and revised March 12, 1985), and LER 85-027 for San Onofre Nuclear Generating Station Unit 3 (dated May 16, 1985).

Attachment 3
IN 86-102
December 15, 1986

LIST OF RECENTLY ISSUED IE INFORMATION NOTICES

Information Notice No.	Subject	Date of Issue	Issued to
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
86-97	Emergency Communications System	11/28/86	All power reactor facilities holding an OL or CP and fuel facilities
86-96	Heat Exchanger Fouling Can Cause Inadequate Operability Of Service Water Systems	11/20/86	All power reactor facilities holding an OL or CP
86-95	Leak Testing Iodine-125 Sealed Sources In Lixi, Inc. Imaging Devices and Bone Mineral Analyzers	11/14/86	All NRC licensees authorized to use Lixi, Inc. imaging devices
86-94	Hilti Contrete Expansion Anchor Bolts	11/6/86	All power reactor facilities holding an OL or CP
86-93	IEB 85-03 Evaluation Of Motor-Operators Identifies Improper Torque Switch Settings	11/3/86	All power reactor facilities holding an OL or CP

OL = Operating License
CP = Construction Permit