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IN 86-47

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

June 10, 1986

IE INFORMATION NOTICE NO. 86-47: ERRATIC BEHAVIOR OF STATIC "O" RING DIFFERENTIAL PRESSURE SWITCHES

Addressees:

All boiling water reactor (BWR) and pressurized water reactor (PWR) facilities holding an operating license (OL) or a construction permit (CP).

Purpose:

This information notice is intended to advise licensees of erratic behavior of certain differential pressure switches supplied by SOR, Incorporated (formerly Static "O" Ring Pressure Switch Company) which apparently caused failure of the LaSalle 2 reactor to scram automatically when it was operating with water level below the low level setpoint. Similar switches are also installed in the high pressure core spray system and the residual heat removal system.

It is expected that recipients will review this information for applicability to their reactor facilities and consider actions, if appropriate, to preclude the occurrence of a similar problem at their facility. Suggestions contained in this notice do not constitute NRC requirements. Therefore, no specific action or written response is required.

The NRC evaluation of this incident is continuing. If specific action is determined to be necessary, a separate notification will be issued.

Summary of Circumstances

On June 1, 1986, LaSalle 2 experienced a feedwater transient that resulted in a low reactor water level. One of the four low level trip channels actuated, resulting in a half scram. The operator recovered level and operation was continued. Subsequent reviews by licensee personnel raised concerns that the level had apparently gone below the scram setpoint and thus a malfunction of the reactor scram system may have occurred. Based on this concern, the licensee declared an "Alert" and shut the plant down. The NRC dispatched an augmented inspection team to the site. Subsequently, the licensee found that the "blind" switches which operate on differential pressure perform erratically. The licensee also found erratic operation for similar switches in the high pressure core spray system and the residual heat removal system which operate valves in the minimum flow recirculation lines. Based on these results, the licensee declared all emergency core cooling systems in LaSalle 1 and 2 to be inoperable. Both units are in cold shutdown pending further evaluation of the problem.

Description of Circumstances:

The following description was constructed from a preliminary sequence of events prepared by the augmented inspection team and from other input by the team.

At 4:20 A.M. on Sunday, June 1, 1986, LaSalle 2 was operating at 93 percent of full power. Both turbine-driven feedwater pumps were operating, with the "A" pump in manual control and the "B" pump in automatic control. The motor-driven feedwater pump was in standby. While a surveillance test was being conducted on feedwater pump "A", the turbine governor valve opened further and caused pump speed and reactor water level to start increasing. At about the same time, the automatic control systems for both turbine-driven pumps locked out. The reactor operator regained control of feedwater pump "A" and ranback feedwater pump speed in an attempt to restore water level to the nominal value (36 inches on the narrow range recorder). A few seconds later when the control system was reset, the "B" feedwater pump controller automatically ranback the pump speed to zero for no apparent reason. Reactor water level started falling at about 2 inches/second.

Subsequently, the reactor protection system responded via separate level switches to the falling reactor water level by reducing recirculation flow to reduce power, and the operator started the motor-driven feedwater pump to increase level. The level continued to fall for a few more seconds before turning around. The minimum reactor scram setpoint required in the technical specification is 11 inches. The level channels are normally set to trip at 13.5 inches, and the operators are trained to expect reactor scram by the time that the water level reaches 12.5 inches. As the level was falling, one of the four reactor scram level switches (the "D" switch) tripped at approximately 10 inches, causing a "half scram." As designed, this did not initiate control rod motion. None of the other three level switches tripped during this transient. No reactor scram occurred during this transient, either automatically or manually.

In the BWR scram system logic, which is one-out-of-two-taken-twice, at least one instrument channel in each scram system must trip to generate a scram demand signal and thereby initiate control rod motion. Preliminary results of the investigation indicate that the reactor water level fell to a minimum value of about 4.5 inches on the narrow range instrumentation, which is several inches below the specified scram setpoint but still 13 to 14 feet above the top of reactor fuel. The period that the water level was below the specified scram setpoint value was approximately 2 seconds. After feedwater flow turned the transient around, the plant stabilized at a power level of about 45 percent. The "B" scram system half scram was manually reset about 30 seconds later. The power level was increased to 60 percent about 3 hours later.

Shortly after the subsequent shift change, the oncoming shift engineer's review was effective in indicating that the reactor water level appeared to have fallen below the scram setpoint and the level switches may not have performed properly. He then requested that an instrumentation technician check the calibration of the switches. The results were that the "A" and "C" switches, which are in the "A" scram system, tripped at 10 and 13.5 inches respectively during the calibration check; the "B" and "D" switches, which are in the "B" scram system, tripped at 11 and 13.5 inches respectively. The switches were readjusted to trip at 13.5 inches. Based on these results, the operating staff believed that a malfunction of the scram system may have occurred. An orderly shutdown of the plant was initiated at 2:00 P.M. (CDT). At 2:30 P.M., the resident inspector was notified, and at 5:30 P.M., the NRC Operations Center was called via the emergency notification system and informed of this event by the licensee.

At 6:20 P.M., the licensee decided that the "A" scram system had failed to perform during the transient. The "A" scram system was manually tripped providing a half scram on the side that had apparently malfunctioned. The orderly shutdown was continued, and an "Alert" was declared. When all the control rods had been fully inserted at 9:22 the next morning, the Alert was terminated.

On Monday, June 2, the NRC determined that the incident warranted a thorough investigation. The NRC Regional Administrator dispatched an augmented inspection team to the plant site.

On Monday evening, June 2, the licensee checked the calibration of the reactor scram water level switches by varying the actual level in the vessel. The results were that the "A" and "C" switches tripped at indicated levels of 9.0 and 6.9 inches respectively and the "B" and "D" switches tripped at 3.9 and 10.2 inches respectively. These data were obtained about 30 hours after the switches had been calibrated according to plant procedures and suggest a non-trivial difference. Additional data obtained over the next two days by varying reactor water level demonstrated continued erratic behavior of switch setpoints.

On Saturday, June 7, after calibrating the Static "O" Ring flow switch which actuates the minimum flow recirculation valve in the high pressure core spray system, the licensee performed a different test using actual system flow. The switch actuated when flow was at 530 gpm instead of 1000 gpm where it had been set to actuate. The licensee found similar performance of flow switches in the residual heat removal system. The licensee now suspects all Static "O" Ring differential pressure switches and has declared all emergency core cooling systems in both units to be inoperable. Both units remain in cold shutdown.

Discussion:

It appears at present that the water level decreased below the scram setpoint for about two seconds and reached a minimum level of about 4.5 inches. This is based on a recording from the narrow range water level instrument and records from the startup testing data acquisition system which recorded levels from the same transmitter. Had the reactor operator been aware of this fact before the water level had increased to a level above the setpoint, the reactor operator would have been expected to scram the reactor manually.

The differential pressure switches which provide the water level trip input to the reactor scram system were provided by SOR, Incorporated. These level switches are not original equipment; but were installed during replacement of equipment in secondary containment. Affected licensees had determined that the original switches were not qualified to operate in the environment created by an accident. Operation of the SOR switches has been demonstrated to be erratic with little correlation between the setpoints established during atmospheric pressure calibrations and switch actuations under system pressure conditions. Exercising the switches by applying successive differential pressure cycles appears to mask erratic setpoint behavior. Similar problems with SOR differential pressure switches have been reported at Oyster Creek.

Per plant procedure, the switches for reactor water level had been exercised prior to calibration following failure of the reactor to scram automatically. For this reason, performance of the level switches may have been different during calibration than during the event. Further, none of the level switches in the LaSalle 2 reactor scram system operate in conjunction with individual level transmitters. Therefore, the calibration and performance of the individual low level trip channels cannot easily be compared to each other. In effect, the operator is blind to switch performance.

The vendor has indicated that those plants identified in Attachment 1 have similar differential pressure switches. This list of plants includes pressurized water reactors as well as boiling water reactors. NRC intends to meet with representatives of General Electric Company, SOR Incorporated, and interested licensees at 10 A.M. on Thursday, June 12, 1986, in Bethesda, Maryland to discuss experience with the switches.

It is suggested that licensees consider advising their reactor operators of the LaSalle incident and providing guidance to them as to how to promptly detect the occurrence of a similar problem at their plants and the proper remedial action to be taken.

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

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Attachments:
1. Plants with Similar Differential Pressure Switches
2. List of Recently Issued IE Information Notices

PLANTS WITH SIMILAR DIFFERENTIAL PRESSURE SWITCHES

PLANT	SOR MODEL NUMBER
Penn. Pwr. & Light/Susquehanna	103/B202
So. Cal. Edison/San Onofre	103/B903
TVA/Brown's Ferry	103/B212
TVA/Sequoyah	103/BB212 103/BB203 103/BB803
WPPS	103/BB203
GPU/Oyster Creek	103/B905 103/BB212 103/B212 103/B202
N.E. Nuc./Millstone	103/B903
South Texas Projects	103/BB212 103/BB803
Commonwealth Edison/LaSalle	103/B202 103/B212 103/B203 103/BB203 103/BB212 103/BB205 103/BB202

Attachment 2
IN 86-47
June 10, 1986

LIST OF RECENTLY ISSUED
IE INFORMATION NOTICES

Information Notice No.	Subject	Date of Issue	Issued to
86-46	Improper Cleaning And Decontamination Of Respiratory Protection Equipment	6/12/86	All power reactor facilities holding an OL or CP and fuel fabrication facilities
86-45	Potential Falsification Of Test Reports On Flanges Manufactured By Golden Gate Forge And Flange, Inc.	6/10/86	All power reactor facilities holding an OL or CP and research and test facilities
86-44	Failure To Follow Procedures When Working In High Radiation Areas	6/10/86	All power reactor facilities holding an OL or CP and research and test reactors
86-43	Problems With Silver Zeolite Sampling Of Airborne Radioiodine	6/10/86	All power reactor facilities holding an OL or CP
86-42	Improper Maintenance Of Radiation Monitoring Systems	6/9/86	All power rector facilities holding an OL or CP
86-41	Evaluation Of Questionable Exposure Readings Of Licensee Personnel Dosimeters	6/9/86	All byproduct material licensees
86-32 Sup. 1	Request For Collection Of Licensee Radioactivity Measurements Attributed To The Chernobyl Nuclear Plant Accident	6/6/86	All power reactor facilities holding an OL or CP
86-40	Degraded Ability To Isolate The Reactor Coolant System From Low-Pressure Coolant Systems in BWRS	6/5/86	All power reactor facilities holding an OL or CP

OL = Operating License
CP = Construction Permit