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Closeout of IE Bulletin 86–02: Static "O" Ring Differential Pressure Switches

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Washington, DC 20555
NRC FIN B8729

ABSTRACT

Documentation is provided in this report for the closeout of IE Bulletin 86-02 regarding static "O" ring differential pressure switches Series 102 or 103 supplied by SOR, Incorporated, and defined per 10 CFR 50.49(b) as electrical equipment important to safety. Closeout is based on the implementation and verification of either one or six actions required by the bulletin for holders of an operating license or a construction permit for a nuclear power plant. All six actions are required when the facility is equipped with the switches of concern in systems subject to limiting conditions for operations in technical specifications. Evaluation of utility responses and NRC/Region inspection reports in accordance with two criteria indicates that the bulletin is closed for 116 (97%) of the 119 nuclear power facilities to which it was issued for action. Followup items are proposed for the three (3) facilities with open status, for the use of NRC regional inspectors in ensuring successful completion of required and corrective actions. A conclusion based on the utility responses is presented. Background information is supplied in the Introduction and Appendix A.

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CLOSEOUT OF IE BULLETIN 86-02: STATIC "O" RING DIFFERENTIAL PRESSURE SWITCHES

INTRODUCTION

This report provides documentation for the closeout status of IE Bulletin 86-02 in accordance with the Statement of Work in Task Order 37 under NRC Contract 05-85-157-02. The documentation is based on the records obtained from the NRC Document Control System.

A low reactor water level incident at LaSalle 2 on June 1, 1986 and a similar event at Oyster Creek 1 on January 17, 1986 led to issuance of IE Bulletin 86-02. Erratic operation of SOR, Incorporated Series 102 or 103 switches was found to be the basic cause of these incidents.

The NRC issued Information Notice 86-47 on June 10, 1986 and IE Bulletin 86-02 on July 18, 1986, to all holders of an operating license or a construction permit for a nuclear power reactor. The bulletin required owners of facilities equipped with affected switches in systems subject to limiting conditions for operations of the plant technical specifications to take six actions in order to assure reliability of operation. Other owners were required to make negative declarations or report usage of the SOR switches in other systems important to safety as described in 10 CFR 50.49(b).

The bulletin and the applicable information notice are included in Appendix A for background information. Evaluation of utility responses and NRC/Region inspection reports is documented in Appendix B as the basis for bulletin closeout. Followup items are proposed in Appendix C for the use of NRC/Region inspectors in assuring that required and corrective actions are completed satisfactorily. Abbreviations used in this report and associated documents are listed in Appendix D.

SUMMARY

1. The bulletin is closed per Criterion 1 (see page B-7) for the following eight (8) facilities for which corrective actions were completed satisfactorily:

Browns Ferry 2	Sequoyah 1,2	WNP 2
LaSalle 1,2	South Texas 1,2	

2. The bulletin is closed for the following 108 facilities which do not have the switches of concern covered by 10 CFR 50.49(b), per Criterion 2 (see page B-7):

Arkansas 1,2	Fort St. Vrain	Prairie Island 1,2
Beaver Valley 1,2	Ginna	Quad Cities 1,2
Bellefonte 1,2	Grand Gulf 1	Rancho Seco 1
Big Rock Point 1	Haddam Neck	River Bend 1
Braidwood 1,2	Harris 1	Robinson 2
Brunswick 1,2	Hatch 1,2	Salem 1,2
Byron 1,2	Hope Creek 1	San Onofre 1,2,3
Callaway 1	Indian Point 2,3	Seabrook 1
Calvert Cliffs 1,2	Kewaunee	Shoreham
Catawba 1,2	Limerick 1,2	St. Lucie 1,2
Clinton 1	Maine Yankee	Summer 1
Comanche Peak 1,2	McGuire 1,2	Surry 1,2
Cook 1,2	Millstone 1,2,3	Susquehanna 1,2
Cooper Station	Monticello	TMI 1
Crystal River 3	Nine Mile Point 1,2	Trojan
Davis-Besse 1	North Anna 1,2	Turkey Point 3,4
Diablo Canyon 1,2	Oconee 1,2,3	Vermont Yankee 1
Dresden 2,3	Palisades	Vogtle 1,2
Duane Arnold	Palo Verde 1,2,3	Waterford 3
Farley 1,2	Peach Bottom 2,3	Watts Bar 1,2
Fermi 2	Perry 1	Wolf Creek 1
FitzPatrick	Pilgrim 1	Yankee-Rowe 1
Fort Calhoun 1	Point Beach 1,2	Zion 1,2

3. The bulletin is open for the following three (3) facilities:

Browns Ferry 1,3	Oyster Creek 1

4. The following facilities which have been shut down indefinitely or permanently (SDI) or have had construction halted indefinitely (CHI) are not included in Table B.1 for documentation of bulletin closeout status:

Dresden 1	SDI	Perry 2	CHI
Humboldt Bay 3	SDI	Seabrook 2	CHI
Indian Point 1	SDI	TMI 2	SDI
La Crosse	SDI	WNP 1,3	CHI

CONCLUSION

The majority of licensees have not installed the specified SOR differential pressure switches, series 102 or 103, as electrical equipment important to safety. The following brief summaries of actions in response to the bulletin at the 11 facilities, which have SOR Model 102 or 103 switches in critical systems, (see summary items 1 and 3) indicate that the concerns of the bulletin have been resolved.

Browns Ferry 1,2,3
(Unit 2 closed, Units 1 and 3 open)
The licensee has committed to implement maintenance instructions for the two affected switches (FS-74-50 and FS-74-64). The maintenance instructions were prepared for performing and checking calibration of the switch setpoints. The NRC/Region inspection report closed the bulletin for Unit 2 on the basis of commitments made by the licensee and held it open for Units 1 and 3. The switches had not yet been installed in Units 1 and 3 of this plant; these units are in extended shutdown (see the followup item on page C-1).

LaSalle 1,2
(Closed)
Because the problem was identified initially at LaSalle Unit 2, corrective actions were planned and checked with particular care. Each unit had about 60 affected switches. The NRC/Region safety evaluation concluded that short-term actions were acceptable for LaSalle Unit 2. The NRC/Region inspectors accepted the utility's commitments to a long-term corrective action plan and closed the bulletin.

Oyster Creek 1
(Open)
See the followup item on page C-1.

Sequoyah 1,2
(Closed)
See note 7 on page B-7. The ten SOR switches of concern in each unit were modified. On the basis of the NRC safety evaluation dated June 23, 1988, the staff finds that the licensee has satisfactorily addressed all issues identified by the bulletin.

South Texas 1,2
(Closed)
See note 5 on page B-7. Bulletin corrective requirements do not apply to the SOR switches at this plant because these switches are not covered by the plant technical specification. The two safety-related switches per unit were tested and were found not to drift out of tolerance.

WNP 2
(Closed)

Four SOR switches were installed at this facility. No significant setpoint shift was found. Augmented testing of these switches was planned and was to be included in the training program. This testing at reactor pressure showed that the instruments exhibited a well-defined shift that could be compensated for during instrument calibration. The NRC/Region inspectors approved the testing but required a sample test in about a year to ensure the drift was constant.

APPENDIX A

Background Information and Required Actions

Note: For required actions, see pages A-7 and A-8.

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IEB 86-02

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

July 18, 1986

IE BULLETIN NO. 86-02: STATIC "O" RING DIFFERENTIAL PRESSURE SWITCHES

Addresses:

All power reactor facilities holding an operating license (OL) or a construction permit (CP).

Purpose:

The purpose of this bulletin is to request that boiling water reactor (BWR) and pressurized water reactor (PWR) licensees determine whether or not they have Series 102 or 103 differential pressure switches supplied by SOR, Incorporated (formerly Static "O" Ring Pressure Switch Company), installed as electrical equipment important to safety. Those licensees that have SOR Series 102 or 103 differential pressure switches installed in systems subject to Technical Specifications are requested to take certain actions to assure that system operation is reliable.

Description of Circumstances:

SOR Series 103 differential pressure switches were installed in LaSalle 2 in mid 1985 as part of an environmental qualification modification which was performed after initial operation of the unit. Identical switches were also installed in LaSalle 1. LaSalle 1 and 2 each have about 60 of these switches in various systems, including the reactor protection system and the emergency core cooling system.

On June 1, 1986, LaSalle 2 experienced a feedwater transient that resulted in low water level in the reactor vessel. One of four low level trip channels actuated, resulting in a half scram. The operator recovered level and power operation was continued. However, subsequent reviews by the Licensee's personnel raised concerns that the level apparently had gone below the scram setpoint and that a malfunction of the reactor scram system may have occurred. Based on this concern, the Licensee declared an "Alert," shut the plant down, notified the NRC, and subsequently informed SOR of possible switch malfunctions. (This incident is described in greater detail in IE Information Notice 86-47).

NRC dispatched an augmented inspection team to the site on June 2 to investigate the root cause and significance of the feedwater transient, the performance of the differential pressure switches in the low level trip channels, the response of the reactor protection system, and related matters.

After recalibrating the level switches on June 1, the Licensee tested the performance of the level switches by lowering water level (drop test) in the reactor and reading the levels indicated on level transmitters when each of the four level switches tripped. The results were erratic with the switches tripping at levels between 2.4 inches and 12.2 (plus or minus about 1.5 inches, depending on the transmitter read). These measurements are relative to instrument zero which is at 161.5 inches above the top of active fuel. The technical specifications require that level channels be declared inoperable if the actual trippoint is below 11.0 inches.

As of June 9, 1986, the Licensee had tested differential pressure switches in the residual heat removal systems and the high pressure core spray system. These switches open valves in minimum flow recirculation lines so that adequate cooling to pump seals and bearings is provided when system flow is low. One of the switches actuated within the range permitted by technical specifications; the others did not. The switch for the high pressure core spray system was calibrated to actuate at 1300 gpm but did not actuate until flow decreased to 530 gpm. The switches for the two residual heat removal systems should have actuated at 1000 gpm but did not actuate until flow decreased to the 480 to 800 gpm range. On the basis of these results, the Licensee declared all emergency core cooling systems for Units 1 and 2 to be inoperable. Both units remain in cold shutdown.

Information Notice 86-47 was issued by the Office of Inspection and Enforcement on June 10, 1986 to inform licensees of the erratic behavior of SOR differential pressure switches during the incident at LaSalle 2 on June 1 and during subsequent testing. An attachment to the information notice listed licensees to which SOR had supplied Series 103 differential pressure switches. That list has been revised (Attachment 1) to include Series 102 differential pressure switches which have important similarities to Series 103 switches. It should be noted that the list of affected licensees is not believed to be fully accurate. The information notice also announced a public meeting of representatives from NRC, General Electric Company, SOR, and interested licensees to discuss the application and performance of Series 102 and 103 switches in safety related systems, which was held on June 12, 1986.

Testing at LaSalle of other Model 103 SOR differential pressure switches used to actuate emergency core cooling system, primary containment isolation system, and other engineered safety feature systems revealed that these switches displayed the same types of behavior as the switches used for reactor scram.

During the vessel water level drop tests at LaSalle 1 on June 2, one of two Series 103 switches used to provide a confirmatory water level input signal to the automatic depressurization system failed to function. On June 17, 1986, testing showed that the trippoint had shifted nonconservatively by 25 inches. In this application, the relative locations of the instrument taps are such that the system could not produce sufficient differential pressure to actuate the switch. Therefore, this amount of shift constitutes a functional failure of the switch. On June 25, the switch was disassembled and inspected. Rust (severe corrosion) was found inside the switch assembly and probably caused a cross shaft bearing, which is outboard of the O-rings, to seize.

A similar event (Licensee Event Report 86-001-00) occurred at Oyster Creek 1 on January 17, 1986, during monthly surveillance of four SOR differential pressure switches which detect low water level in the reactor vessel. The "as-found" setpoints for three of the switches had drifted downward as much as 6 inches. During the subsequent 11 weeks, the level switches continued to perform erratically, each switch was replaced one or more times, and modified switches were installed. On April 7, after a modified switch had nonconservative setpoint drift, the Licensee performed daily surveillance until about April 12 when the reactor was shutdown for a six month outage. Increased surveillance frequency did not resolve the problem.

Earlier concern for mechanical level indication equipment was expressed in NRC Generic Letter No. 84-23 which addressed water level instrumentation for BWR reactor vessels. The generic letter was based on NRC's evaluation of a report by S. Levy, Incorporated, which had been commissioned by a BWR Owner's Group. The generic letter addressed the need for BWR licensees to review plant experience related to mechanical level indication equipment, indicated that analog trip units have better reliability and greater accuracy than mechanical level indication equipment, and stated that BWR licensees should replace such equipment with analog transmitters unless operating experience indicates otherwise.

Responses to Generic Letter No. 84-23 show that 80% of BWR licensees have replaced or plan to replace their mechanical level instrumentation with analog level transmitters. Recipients of this bulletin should recognize that while this bulletin focuses on more immediate problems with two similar models of mechanical differential pressure switches manufactured by SOR, Incorporated, the reliability of other mechanical instrumentation is also in question because it may be vulnerable to similar problems. Because the same urgency has not been demonstrated for other mechanical differential pressure switches, the NRC plans to address that matter separately.

Discussion:

DESCRIPTION OF SERIES 102 AND 103 DIFFERENTIAL PRESSURE SWITCHES

The Series 102 and 103 differential pressure switches consist of a piston (Series 102) or a diaphragm (Series 103) which moves a lever that rotates a cross shaft. These components are contained in a steel case designed to withstand system pressure. Both ends of the cross shaft extend out of the wetted volume and O-ring seals are provided to form the pressure boundary and prevent leakage along the cross shaft. The condition of these surfaces and the O-rings will determine the extent to which frictional forces cause a torque which opposes rotation of the cross shaft. A lever is attached to each end of the cross shaft. When the cross shaft rotates, one lever moves to actuate a microswitch. The other lever bears on a helical spring. An adjusting screw is used to change the compression of the spring and thus change the setpoint of the differential pressure switch.

The case contains two ports on either side of the piston or diaphragm. The lower port on one side is connected to the system reference leg, and the lower port on the other side is connected to the lower instrument tap (i.e. variable leg). The upper ports on both sides are used as vents and are plugged when the switch is in service.

The design of the cavity containing the diaphragm (or piston) is such that motion of the diaphragm is limited to 0.015 inch. Most of the time, the diaphragm is against one or the other of the mechanical stops which limit motion of the diaphragm. Thus the sum of the unbalanced hydraulic forces across the diaphragm is supported by one stop or the other except when the microswitch is forced to change position. This occurs when the absolute value of the torque caused by the unbalanced hydraulic forces changes from a value less than to a value greater than the torque caused by the helical spring. This movement causes the cross shaft and the levers to rotate 1.8 degrees.

PROBLEM AREAS

Differential pressure switches are often calibrated in situ after isolating them from the reactor system. A test rig consisting essentially of two bottles each containing water and air or nitrogen are connected to the differential pressure switch with one bottle on either side of the diaphragm. The differential pressure for calibration is established by adjusting the gas pressures in the bottles. Often, the lower pressure is at or near atmospheric pressure.

When the SOR Model 103 differential pressure switch is calibrated to a setpoint at atmospheric pressure and then connected to a system operating at a static pressure of about 1000 psig, the actual setpoint shifts in most cases in the conservative direction toward less differential pressure required to trip. In other cases, the offset of setpoint due to calibration at atmospheric pressure has been found to be in the opposite direction. The manufacturer has stated that each switch has unique characteristics and that switches with the same model number do not all behave in the same way. It has been postulated that this may be caused by deformation or movement of the O-rings on the cross shaft when system pressure is applied. For water level applications and depending on the location of the lower instrument tap relative to the required setpoint, offset may be so large that the switch will not actuate before the level drops below the tap. In this case, the switch would not actuate no matter how low the level dropped. The vendor has indicated to the staff that factory tests showed an offset between behavior at atmospheric pressure and behavior at system pressure and that this information is provided to all customers. It has been the practice at LaSalle to calibrate at atmospheric pressure without compensating for errors due to static pressure effects.

For minimum flow applications where it is necessary to open a valve in a recirculation line to protect a pump in an emergency core cooling system, assurance is needed that offset will not delay that action and result in pump damage.

Testing of Series 103 differential pressure switches at LaSalle showed that application of a static pressure to the switch for a period of time also resulted in a significant shift in the setpoint of the switch, and that the shift due to prolonged pressure was generally in the opposite direction from the shift due to the initial application of static pressure. After being calibrated at atmospheric pressure, a static pressure of 1000 psig was maintained. A recheck of the setpoint of one switch at the end of 24 hours showed that the setpoint had shifted by a net amount that was nonconservative by about 10 inches. Subsequent rechecks continued to show shifting but in lesser amounts. To be valid, it appears that calibration and tests would need to be rechecked after static pressure has been maintained for at least 48 hours.

Recent testing at LaSalle has also shown that the point at which trip occurs depends on whether the switch setpoint is being approached from low differential pressure or high differential pressure. This is particularly important for automated blocking valves in the recirculation lines which protect emergency core cooling pumps from damage when system flow is low. When flow decreases to a value below the setpoint, the switches should actuate to open the valves. Conversely, when flow increases, the switches should deactuate to provide maximum flow to the core.

In addition to showing offset problems, some of the Series 103 switches evidence sticky behavior, i.e. a larger change in differential pressure is required to actuate the switch on the first demand than on subsequent operations and on subsequent tests actuation may be erratic. It is believed that starting friction and the condition of the cross shaft surfaces may cause these problems. If the O-rings stick, then the torque that they apply is added to the torque applied by the calibration spring. SOR is conducting a long range test with switches that have more highly polished finishes on those parts of the cross shafts that are in contact with O-rings.

It has been common practice at LaSalle to actuate the switches several times and then to record the differential pressures required for the third or fourth actuation. It appears that the Licensee has not emphasized that the "as-found" condition of the switch is the value of differential pressure required to actuate the switch during the first demand. It is this value that must be used to determine whether the switch and its system would have performed their intended functions if called upon to do so.

The life of Series 103 switches has been said to be 20 to 40 years. However, the shelf life of the elastomeric material used in the O-rings is considerably less than 40 years. The O-rings may need to be changed several times during the life of the plant. Further, there is some concern for the effect of reactor water on the O-rings, cross shaft surfaces bearing on the O-rings, and on the diaphragm material, and possible corrosion of the cross shaft bearings.

OBJECTIVES OF REQUIRED ACTIONS

General Design Criterion 21 "Protection System Reliability and Testability" requires that the protection system be highly reliable. It is clear that the SOR differential pressure switches that have been tested carefully to date have not performed reliably.

A significant uncertainty exists as to where SOR differential pressure switches are currently installed or planned to be installed. A list provided by SOR, Inc. included one utility that had ordered the switches but later decided not to install the switches. The NRC later learned that another utility that was not on the SOR list had installed SOR switches. It is important to assessing the safety impact to know with certainty which plants have SOR switches installed and in what plant systems. Since these switches were installed predominately as environmentally qualified electrical equipment important to safety, as described in 10 CFR 50.49(b), this Bulletin requests all licensees to identify each such installation. The NRC intends to evaluate this information, in combination with the results of other actions required by this Bulletin, to determine if further actions should be required.

Licensees, who have SOR switches installed, are requested to determine which of those switches are installed in systems which are subject to Limiting Conditions for Operations of the plant Technical Specifications. For SOR differential switches that are not in systems subject to Technical Specifications, licensees are expected to review the information in this Bulletin and consider actions, if appropriate, to preclude problems similar to those discussed in this Bulletin from occurring. For SOR differential pressure switches that are installed in systems subject to Technical Specifications, the Bulletin requests licensees to take certain actions to assure that these switches and systems will be capable of performing acceptably, if called upon during an actual plant transient or accident.

First, each licensed reactor operator (and senior reactor operator) on duty should be made aware of the potential problem that may occur at his/her plant. This information should include a knowledge of the incident at LaSalle, where SOR differential pressure switches are installed in his/her plant, how to detect a malfunction or failure of any of these switches, and the remedial actions that he/she should be prepared to take if a malfunction were to occur.

Second, the Bulletin requests licensees to conduct special operability tests of each system that is subject to Technical Specifications that involve SOR differential pressure switches. Special tests are necessary to determine the actual trippoint of the switches and the operability of the systems since tests of the type typically conducted may not be adequate to reveal the type of problems that have been revealed at the LaSalle station.

It is important that the tests simulate the conditions of the operation of the system. Further, the test results from LaSalle suggest that the system operating conditions should be maintained for at least 48 hours before attempting to measure the performance of the SOR switches. For those systems that are not testable during plant power operations, it is anticipated that licensees will use test rigs in order to to simulate operating conditions and not impact plant operations. It is also expected that licensees may take credit for the 48 hours that the switch was at system operating conditions prior to connecting the test rig, in order to minimize the time the switch/system is bypassed or tripped. If the test rig can be connected to the switch so as to make a virtually "bumpless" transfer from the system to the test rig without tripping the switch, such credit may be appropriate. A primary objective of the special tests is to determine how the switch will respond to its first demand after being at system operating conditions for a period of time. Special care may be necessary to assure that the first actuation is measured.

If one channel of a system of redundant channels (or similar equipment in redundant safety systems) is found to have an actual trippoint that is outside the Technical Specifications or otherwise unacceptable for adequately reliable system operation, then the redundant channels (or similar equipment) should be tested as soon thereafter as practical. The short term corrective actions to be taken to return the set of channels to operable status should be based on an analysis that conservatively considers the performance of the set of redundant channels (or similar equipment). In view of the generic safety concerns and the possibility of common mode failures, unacceptable performance of an SOR differential pressure switch should be reported to the NRC in accordance with 10 CFR 50.72 and 10 CFR 50.73.

Since the conduct of any special test could have potential adverse affects, the requirements for followup tests to verify continuing proper functioning of the switches and systems have been minimized to the extent possible consistent with the safety objective. The Bulletin requests that licensees propose an interim performance monitoring program that would cover the time between the special tests and full implementation of long term corrective actions. The objectives of the program are to detect any instance of unacceptable performance, to provide for timely initiation of additional corrective action, and to gather additional switch performance data.

The Bulletin requests licensees to determine what long term corrective actions may be appropriate and will be taken. Part of this determination would include considering the potential effects of common mode failures. This determination should be based upon an analysis using the worst observed shift of the actual trippoint from the calibration setpoint for SOR switches in each general type of application, e.g., water level measurement or main steam flow measurement. The purpose of the analysis is to determine if improvements in calibration and testing methods, improvements in setpoint methodology, additional safety analysis to establish a revised licensing basis for the plant, change in the Technical Specifications, repair, modifications, or replacement, or other improvements are needed in order to meet existing regulatory requirements (e.g., General Design Criterion 21 or plant technical specifications). The analysis should demonstrate that the long term corrective action will provide an adequate margin for safety so as to assure high functional reliability.

Actions Required of All Licensees:

1. Within 7 days, submit a report on the extent to which SOR Model 102 or 103 differential pressure switches are installed (or planned) as electrical equipment important to safety, as defined in 10 CFR 50.49(b). Include in the report: the model number of the switch, the system in which it is installed (e.g., low pressure safety injection), the application of the switch (e.g., water level measurement, system flow measurement), and the

function of the switch (e.g., control of minimum flow recirculation valve). A negative report, if appropriate, is required.

Actions Required of Licensees That Have SOR Model 102 or 103 Differential Pressure Switches Installed in Systems That Are Subject to Limiting Conditions for Operation in Technical Specifications:

2. Within 7 days, take positive action to assure that licensed reactor operators on duty are prepared for potential malfunctions of SOR switches.

3. Within 30 days, conduct a special test of each SOR switch to determine if the switch and system function properly or if short term corrective actions are necessary. The tests are to determine if the switches/systems will respond acceptably on the first demand after being at system operating conditions for a period of time. The tests should be planned and conducted so as to minimize any potential adverse affects of the testing. If any corrective action includes the replacement of SOR switches with mechanical differential pressure switches by another manufacturer, the licensee should submit a technical justification, including a reliability demonstration. Repeat the special tests on a monthly basis until two consecutive successful tests are attained.

4. Report failures in accordance with 10 CFR 50.72 and 10 CFR 50.73.

5. Within 60 days, develop, implement and submit a written report describing your interim performance monitoring program to provide continuing assurance that the performance of the switches and plant systems remains acceptably reliable until long term corrective actions are fully implemented.

6. Within 60 days, submit a written report which describes the margin and basis for switch actuation. The report should also describe the long term corrective actions to be taken, including the implementation schedule, the impacts of potential common mode failures, and an analysis to demonstrate that the system involved will meet regulatory requirements and function reliably. The report should include specific information on the installed SOR switches: the manufacturer's specified range for the switch, the nominal and allowable values for the calibration setpoint in the Technical Specifications in the same terms as the manufacturer's specified range for the switch, the relative locations of the instrument taps for water level monitoring applications, sources of systematic errors such as the differences in elevations of the installation of condensing pots, and "as found" and any subsequent test data for any switch that does not conform to the Technical Specifications or is otherwise unacceptable.

Recipients of this Bulletin who hold construction permits and licensees of plants that are shutdown for an extended period (e.g., Browns Ferry) are not required to complete the actions of this Bulletin on the schedule shown. In each case, compliance with this Bulletin should be addressed prior to the next critical operation of the plant or within 1 year, whichever occurs first.

If, because of plant unique conditions, a licensee should determine that any action requested by this Bulletin jeopardizes plant safety, the action should not be initiated and the NRC should be notified as soon as practical. This notification should include the basis for the determination. Further, if a licensee determines that, even with "best efforts," an action requested by this Bulletin can not reasonably be completed within the prescribed schedule, the NRC should be notified within 7 days of receipt of the Bulletin.

The written reports shall be submitted to the appropriate Regional Administrator under oath or affirmation under the provisions of Section 182a of the Atomic Energy Act of 1954, as amended. Also, the original copy of the cover letters and a copy of the reports shall be transmitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC, 20555 for reproduction and distribution.

The request for information was approved by the Office of Management and Budget under blanket clearance number 3150-0012. Comments on burden and duplication may be directed to the Office of Management and Budget, Reports Management, Room 3208, New Executive Office Building, Washington, DC, 20503.

If you have any questions regarding this matter, please contact the Regional Administrator of the appropriate NRC Regional Office or one of the technical contacts listed below.

James M. Taylor

James M. Taylor, Director
Office of Inspection and Enforcement

Attachments:
1. Plants with Similar SOR Switches
2. List of Recently IE Bulletins

Technical Contacts: J. T. Beard, NRR
(301) 492-4415

Roger W. Woodruff, IE
(301) 492-7205

PLANTS WITH SERIES 102 OR 103 DIFFERENTIAL PRESSURE SWITCHES

Series 102:

Florida Power and Light

Series 103:

Commonwealth Edison
General Public Utilities - Nuclear Corporation
Houston Lighting & Power Company
Northeast Utilities
Pennsylvania Power & Light
Southern California Edison
Tennessee Valley Authority
Washington Public Power Supply System

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.

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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.

E. L. Jordan

Edward L. Jordan, Director
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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

APPENDIX B

Documentation of Bulletin Closeout

TABLE B.1 BULLETIN CLOSEOUT STATUS

Facility	Utility	Docket	Facility Status 07-18-86	NRC Region	NSSS	Utility Response Date	Inspection Report and Date	Closeout Status and Criterion
Arkansas 1	AP&L	50-313	OL	IV	B&W	07-31-86	86-40(01-27-87)	Closed 2
Arkansas 2	AP&L	50-368	OL	IV	C-E	07-31-86	86-40(01-27-87)	Closed 2
Beaver Valley 1	DLC	50-334	OL	I	W	08-05-86	86-18(09-04-86)	Closed 2
Beaver Valley 2	DLC	50-412	CP	I	W	12-23-86		Closed 2
Bellefonte 1	TVA	50-438	CP	II	B&W	11-20-86		Closed 2
Bellefonte 2	TVA	50-439	CP	II	B&W	11-20-86		Closed 2
Big Rock Point 1	CPC	50-155	OL	III	GE	07-24-86	86-14(12-11-86)	Closed 2
Braidwood 1	CECO	50-456	CP	III	W	07-25-86 10-02-86	86-50(11-20-86)	Closed 2
Braidwood 2	CECO	50-457	CP	III	W	07-25-86 10-02-86	86-37(11-20-86)	Closed 2
Browns Ferry 1	TVA	50-259	OL	II	GE	07-20-87	88-28(12-09-88)	Open
Browns Ferry 2	TVA	50-260	OL	II	GE	07-20-87	88-28(12-09-88)	Closed 1
Browns Ferry 3	TVA	50-296	OL	II	GE	07-20-87	88-28(12-09-88)	Open
Brunswick 1	CP&L	50-325	OL	II	GE	07-28-86	87-17(07-28-87)	Closed 2
Brunswick 2	CP&L	50-324	OL	II	GE	07-28-86	87-17(07-28-87)	Closed 2
Byron 1	CECO	50-454	OL	III	W	07-25-86	86-33(10-08-86)	Closed 2 (Note 4)
Byron 2	CECO	50-455	CP	III	W	07-25-86	86-24(09-23-86)	Closed 2
Callaway 1	UE	50-483	OL	III	W	07-25-86	86-20(12-12-86)	Closed 2
Calvert Cliffs 1	BG&E	50-317	OL	I	C-E	07-29-86	86-11(09-18-86)	Closed 2
Calvert Cliffs 2	BG&E	50-318	OL	I	C-E	07-29-86	86-11(09-18-86)	Closed 2
Catawba 1	DUPCO	50-413	OL	II	W	07-28-86	87-08(04-06-87)	Closed 2
Catawba 2	DUPCO	50-414	OL	II	W	07-28-86	87-08(04-06-87)	Closed 2
Clinton 1	IP	50-461	CP	III	GE	08-26-86	86-64(10-15-86)	Closed 2
Comanche Peak 1	TUGCO	50-445	CP	IV	W	05-11-87	87-36(02-09-88)	Closed 2
Comanche Peak 2	TUGCO	50-446	CP	IV	W	05-11-87	87-27(02-09-88)	Closed 2

See notes and criteria for closeout of bulletin at end of table.

TABLE B.1 (contd)

Facility	Utility	Docket	Facility Status 07-18-86	NRC Region	NSSS	Utility Response Date	Inspection Report and Date	Closeout Status and Criterion
Cook 1	IMECO	50-315	OL	III	W	07-28-86	86-30(10-02-86)	Closed 2
Cook 2	IMECO	50-316	OL	III	W	07-28-86	86-30(10-02-86)	Closed 2
Cooper Station	NPPD	50-298	OL	IV	GE	07-25-86	87-06(03-11-87)	Closed 2
Crystal River 3	FPC	50-302	OL	II	B&W	07-29-86	86-31(10-30-86)	Closed 2
Davis-Besse 1	TECO	50-346	OL	III	B&W	07-28-86	86-23(11-04-86)	Closed 2
Diablo Canyon 1	PG&E	50-275	OL	V	W	07-29-86		Closed 2
Diablo Canyon 2	PG&E	50-323	OL	V	W	07-29-86		Closed 2
Dresden 2	CECO	50-237	OL	III	GE	07-25-86	88-26(01-23-89)	Closed 2
Dresden 3	CECO	50-249	OL	III	GE	07-25-86	88-26(01-23-89)	Closed 2
Duane Arnold	IELPCO	50-331	OL	III	GE	07-30-86	86-12(09-17-86)	Closed 2
Farley 1	APCO	50-348	OL	II	W	07-25-86 08-08-86	86-19(10-21-86)	Closed 2
Farley 2	APCO	50-364	OL	II	W	07-25-86 08-08-86	86-19(10-21-86)	Closed 2
Fermi 2	DECO	50-341	OL	III	GE	07-28-86 10-03-86	86-26(11-04-86)	Closed 2
FitzPatrick	NYPA	50-333	OL	I	GE	07-28-86		Closed 2
Fort Calhoun 1	OPPD	50-285	OL	IV	C-E	07-29-86	87-10(05-15-87)	Closed 2
Fort St. Vrain	PSCC	50-267	OL	IV	GA	07-28-86 02-20-87	87-08(04-06-87)	Closed 2
Ginna	RG&E	50-244	OL	I	W	07-28-86		Closed 2
Grand Gulf 1	MP&L	50-416	OL	II	GE	07-30-86	87-17(07-28-87)	Closed 2
Haddam Neck	CYAPCO	50-213	OL	I	W	07-25-86	86-27(11-25-86)	Closed 2
Harris 1	CP&L	50-400	CP	II	W		87-26(08-03-87)	Closed 2
Hatch 1	GPC	50-321	OL	II	GE	07-25-86	87-12(07-14-87)	Closed 2
Hatch 2	GPC	50-366	OL	II	GE	07-25-86	87-12(07-14-87)	Closed 2
Hope Creek 1	PSE&G	50-354	CP	I	GE	07-30-86		Closed 2
Indian Point 2	ConEd	50-247	OL	I	W	07-30-86	88-26(11-02-88)	Closed 2
Indian Point 3	NYPA	50-286	OL	I	W	07-30-86		Closed 2

See notes and criteria for closeout of bulletin at end of table.

TABLE B.1 (contd)

Facility	Utility	Docket	Facility Status 07-18-86	NRC Region	NSSS	Utility Response Date	Inspection Report and Date	Closeout Status and Criterion
Kewaunee	WPS	50-305	OL	III	W	07-28-86	86-07(11-03-86)	Closed 2
LaSalle 1	CECO	50-373	OL	III	GE	07-25-86 08-29-86 09-08-86 01-21-87 02-06-89	SE (08-07-86) 86-46(02-14-87)	Closed 1
LaSalle 2	CECO	50-374	OL	III	GE	07-25-86 08-29-86 09-08-86 01-21-87 02-06-89	SE (08-07-86) 86-46(02-14-87)	Closed 1
Limerick 1	PECO	50-352	OL	I	GE	07-25-86	86-27(02-26-87)	Closed 2
Limerick 2	PECO	50-353	CP	I	GE	07-08-88		Closed 2
Maine Yankee	MYAPCO	50-309	OL	I	C-E	07-24-86		Closed 2
McGuire 1	DUPCO	50-369	OL	II	W	07-28-86	87-27(09-15-87)	Closed 2
McGuire 2	DUPCO	50-370	OL	II	W	07-28-86	87-27(09-15-87)	Closed 2
Millstone 1	NU	50-245	OL	I	GE	07-25-86	86-13(08-18-86)	Closed 2
Millstone 2	NU	50-336	OL	I	C-E	07-25-86	86-13(08-18-86)	Closed 2
Millstone 3	NU	50-423	OL	I	W	07-25-86		Closed 2
Monticello	NSP	50-263	OL	III	GE	07-28-86	86-07(11-10-86)	Closed 2
Nine Mile Point 1	NMP	50-220	OL	I	GE	07-29-86		Closed 2
Nine Mile Point 2	NMP	50-410	CP	I	GE	08-19-86		Closed 2
North Anna 1	VEPCO	50-338	OL	II	W	07-25-86	88-08(04-28-88)	Closed 2
North Anna 2	VEPCO	50-339	OL	II	W	07-25-86	88-08(04-28-88)	Closed 2
Oconee 1	DUPCO	50-269	OL	II	B&W	07-28-86	87-25(07-30-87)	Closed 2
Oconee 2	DUPCO	50-270	OL	II	B&W	07-28-86	87-25(07-30-87)	Closed 2
Oconee 3	DUPCO	50-287	OL	II	B&W	07-28-86	87-25(07-30-87)	Closed 2

See notes and criteria for closeout of bulletin at end of table.

TABLE B.1 (contd)

Facility	Utility	Docket	Facility Status 07-18-86	NRC Region	NSSS	Utility Response Date	Inspection Report and Date	Closeout Status and Criterion
Oyster Creek 1	GPUN/ JCP&L	50-219	OL	I	GE	07-30-86 09-23-86 10-14-86	SE (12-15-86) 89-14(08-03-89)	Open
Palisades	CPC	50-255	OL	III	C-E	07-24-86	86-23(09-25-86)	Closed 2
Palo Verde 1	APSCO	50-528	OL	V	C-E	07-31-86	86-30(10-23-86)	Closed 2
Palo Verde 2	APSCO	50-529	OL	V	C-E	07-31-86	86-29(10-23-86)	Closed 2
Palo Verde 3	APSCO	50-530	CP	V	C-E	07-31-86	86-22(10-23-86)	Closed 2
Peach Bottom 2	PECO	50-277	OL	I	GE	07-25-86		Closed 2
Peach Bottom 3	PECO	50-278	OL	I	GE	07-25-86		Closed 2
Perry 1	CEI	50-440	CP	III	GE	07-30-86	86-23(10-10-86)	Closed 2
Pilgrim 1	BECO	50-293	OL	I	GE	07-31-86		Closed 2
Point Beach 1	WEPCO	50-266	OL	III	W	07-25-86	86-15(10-08-86)	Closed 2
Point Beach 2	WEPCO	50-301	OL	III	W	07-25-86	86-14(10-08-86)	Closed 2
Prairie Island 1	NSP	50-282	OL	III	W	07-25-86 09-30-86	86-10(10-27-86)	Closed 2
Prairie Island 2	NSP	50-306	OL	III	W	07-25-86 09-30-86	86-12(10-27-86)	Closed 2
Quad Cities 1	CECO	50-254	OL	III	GE	07-25-86	88-28(02-14-89)	Closed 2
Quad Cities 2	CECO	50-265	OL	III	GE	07-25-86	88-29(02-14-89)	Closed 2
Rancho Seco 1	SMUD	50-312	OL	V	B&W	01-09-87	86-42(02-18-87)	Closed 2
River Bend 1	GSU	50-458	OL	IV	GE	07-28-86	88-25(11-29-88)	Closed 2
Robinson 2	CP&L	50-261	OL	II	W	07-28-86	86-17(08-19-86)	Closed 2
Salem 1	PSE&G	50-272	OL	I	W	07-30-86		Closed 2
Salem 2	PSE&G	50-311	OL	I	W	07-30-86		Closed 2
San Onofre 1	SCE	50-206	OL	V	W	07-28-86	86-49(02-26-87)	Closed 2
San Onofre 2	SCE	50-361	OL	V	C-E	07-28-86	86-38(02-26-87)	Closed 2
San Onofre 3	SCE	50-362	OL	V	C-E	07-28-86	86-38(02-26-87)	Closed 2
Seabrook 1	PSNH	50-443	CP	I	W	09-15-86	86-47(12-10-86)	Closed 2

See notes and criteria for closeout of bulletin at end of table.

TABLE B.1 (contd)

Facility	Utility	Docket	Facility Status 07-18-86	NRC Region	NSSS	Utility Response Date	Inspection Report and Date	Closeout Status and Criterion
Sequoyah 1	TVA	50-327	OL	II	W	08-13-86 11-07-86 03-18-88 09-26-88	88-19(05-27-88) SE (06-23-88)	Closed 1 (Note 7)
Sequoyah 2	TVA	50-328	OL	II	W	08-13-86 11-07-86 03-18-88 09-26-88	88-19(05-27-88) SE (06-23-88)	Closed 1 (Note 7)
Shoreham	LILCO	50-322	LPTL	I	GE	07-30-86		Closed 2
South Texas 1	HL&P	50-498	CP	IV	W	07-17-87	87-39(07-30-87)	Closed 1 (Note 5)
South Texas 2	HL&P	50-499	CP	IV	W	07-17-87	87-39(07-30-87)	Closed 1 (Note 5)
St. Lucie 1	FPL	50-335	OL	II	C-E	07-28-86 08-05-86	87-14(07-28-87)	Closed 2
St. Lucie 2	FPL	50-389	OL	II	C-E	07-28-86 08-05-86	87-13(07-28-87)	Closed 2
Summer 1	SCE&G	50-395	OL	II	W		86-15(09-18-86)	Closed 2
Surry 1	VEPCO	50-280	OL	II	W	07-25-86	87-04(03-06-87)	Closed 2
Surry 2	VEPCO	50-281	OL	II	W	07-25-86	87-04(03-06-87)	Closed 2
Susquehanna 1	PP&L	50-387	OL	I	GE	07-28-86	86-14(09-24-86)	Closed 2 (Note 6)
Susquehanna 2	PP&L	50-388	OL	I	GE	07-28-86	86-14(09-24-86)	Closed 2 (Note 6)
TMI 1	GPUN/Met-Ed	50-289	OL	I	B&W	07-29-86	87-24(02-17-88)	Closed 2
Trojan	PGE	50-344	OL	V	W	07-28-86	86-39(11-04-86)	Closed 2
Turkey Point 3	FPL	50-250	OL	II	W	07-28-86 08-06-86	87-20(05-20-87)	Closed 2
Turkey Point 4	FPL	50-251	OL	II	W	07-28-86 08-06-86	87-20(05-20-87)	Closed 2

See notes and criteria for closeout of bulletin at end of table.

TABLE B.1 (contd)

Facility	Utility	Docket	Facility Status 07-18-86	NRC Region	NSSS	Utility Response Date	Inspection Report and Date	Closeout Status and Criterion
Vermont Yankee 1	VYNP	50-271	OL	I	GE	07-24-86		Closed 2
Vogtle 1	GPC	50-424	CP	II	W	10-15-86 01-09-87	87-17(03-24-87)	Closed 2 (Note 4)
Vogtle 2	GPC	50-425	CP	II	W	10-15-86 01-09-87	87-30(08-21-87)	Closed 2 (Note 4)
WNP 2	WPPSS	50-397	OL	V	GE	07-29-86 11-07-86	86-34(12-03-86)	Closed 1
Waterford 3	LP&L	50-382	OL	IV	C-E	07-25-86	86-16(10-10-86)	Closed 2
Watts Bar 1	TVA	50-390	CP	II	W	11-20-86		Closed 2
Watts Bar 2	TVA	50-391	CP	II	W	11-20-86		Closed 2
Wolf Creek 1	KG&E	50-482	OL	IV	W	07-28-86 07-30-86		Closed 2
Yankee-Rowe 1	YAECO	50-029	OL	I	W	07-23-86	86-08(10-24-86)	Closed 2
Zion 1	CECO	50-295	OL	III	W	07-25-86	86-22(12-19-86)	Closed 2 (Note 4)
Zion 2	CECO	50-304	OL	III	W	07-25-86	86-20(12-19-86)	Closed 2 (Note 4)

See notes and criteria for closeout of bulletin on the next page.

Notes:

1. Facility status is based on Reference 1 below.

2. The following abbreviations apply to facility status:
CP, construction permit; LPTL, low-power testing license; OL, operating license.

3. For bulletin closeout criteria see below.

4. The response is clarified per the listed inspection report.

5. The response of 07-17-87 for South Texas 1,2 reported that the switches in the only safety-related system (Auxiliary Steam) were found not to drift out of tolerance. The applications were not covered by the plant technical specifications.

6. The response of 07-28-86 for Susquehanna 1,2 reported that no SOR switches of concern were installed as safety-related equipment and that 14 modified switches would not be installed.

7. The verification requested per the SE (06-23-88) is provided in the response of 09-26-88 for Sequoyah 1,2. The resident inspector has assured the NRC project manager that a favorable inspection report is forthcoming.

CRITERIA FOR CLOSEOUT OF BULLETIN

Criterion 1: The utility response and an NRC/Region inspection report or an NRC safety evaluation indicate that corrective actions required by the bulletin (see pages A-7 and A-8) have been completed satisfactorily.

Criterion 2: The utility response or an NRC/Region inspection report indicates that there are none of the subject switches installed (or planned) as electrical equipment important to safety, as defined in 10 CFR 50.49(b) (see Reference 2 below).

REFERENCES

1. United States Nuclear Regulatory Commission, Licensed Operating Reactors. Status Summary Report. Data as of 03-31-89, NUREG-0020, Volume 13, Number 4, April 1989.

2. United States Nuclear Regulatory Commission, Code of Federal Regulations, Energy, Title 10, Chapter 1, January 1, 1987, cited as 10 CFR 0.735-1.

APPENDIX C

Proposed Followup Items

Region I

Oyster Creek 1
According to Inspection Report 89-14, the systems containing two of the affected SOR switches were replaced with analog trip systems, and a monthly test program was initiated for the remaining SOR switches. The bulletin is held open by the NRC inspectors pending the licensee's submittal of the requested information and implementation of the training requirement.

Region II

Browns Ferry 1,3
The utility's response of 07-20-87 was evaluated in Inspection Report 88-28 (12-09-88) for all three units. The bulletin is closed for Unit 2 only. A later inspection report is needed to close out the bulletin for Units 1 and 3 before startup because these units are in an extended shutdown.

APPENDIX D

Abbreviations

ANPP	Arizona Nuclear Power Project
APCO	Alabama Power Company
AP&L	Arkansas Power and Light Company
APSCO	Arizona Public Service Company
BECO	Boston Edison Company
BG&E	Baltimore Gas and Electric Company
B&W	Babcock & Wilcox Company
BWR	Boiling Water Reactor
C-E	Combustion Engineering Incorporated
CECO	Commonwealth Edison Company
CEI	Cleveland Electric Illuminating Company
CFR	Code of Federal Regulations
CHI	Construction Halted Indefinitely
ConEd	Consolidated Edison Company of New York, Inc.
CP	Construction Permit
CPC	Consumers Power Company
CP&L	Carolina Power and Light Company
CR	Contractor Report
CYAPCO	Connecticut Yankee Atomic Power Company
DECO	Detroit Edison Company
DLC	Duquesne Light Company
DUPCO	Duke Power Company
FPC	Florida Power Corporation
FPL	Florida Power & Light Company
GA	General Atomic
GAO	Government Accounting Office
GE	General Electric Company
GPC	Georgia Power Company
GPUN	GPU Nuclear Corporation
GSU	Gulf States Utilities Company
HL&P	Houston Lighting and Power Company
IE	(See NRC/IE)
IEB	Inspection and Enforcement Bulletin (NRC)
IELPCO	Iowa Electric Light and Power Company
IMECO	Indiana and Michigan Electric Company
IP	Illinois Power Company
IR	Inspection Report (NRC/Region)
JCP&L	Jersey Central Power and Light Company
KG&E	Kansas Gas and Electric Company
LER	Licensee Event Report
LILCO	Long Island Lighting Company
LP&L	Louisiana Power and Light Company
LPTL	Low Power Testing License

MP&L	Mississippi Power and Light Company
MYAPCO	Maine Yankee Atomic Power Company
NMP	Niagara Mohawk Power Company
NPPD	Nebraska Public Power District
NRC/IE	Nuclear Regulatory Commission/ Office of Inspection & Enforcement
NRR	Office of Nuclear Reactor Regulation (NRC)
NSP	Northern States Power Company
NU	Northeast Utilities
NYPA	New York Power Authority
OL	Operating License
OPPD	Omaha Public Power District
PECO	Philadelphia Electric Company
PGE	Portland General Electric Company
PG&E	Pacific Gas and Electric Company
PP&L	Pennsylvania Power and Light Company
PSCC	Public Service Company of Colorado
PSE&G	Public Service Electric and Gas Company
PSNH	Public Service Company of New Hampshire
PWR	Pressurized Water Reactor
R	Region (NRC)
RG&E	Rochester Gas and Electric Corporation
SCE	Southern California Edison Company
SCE&G	South Carolina Electric and Gas Company
SDI	Shut Down Indefinitely
SE	Safety Evaluation
SMUD	Sacramento Municipal Utility District
STP	Surveillance Test Procedure
TECO	Toledo Edison Company
TMI	Three Mile Island
TUGCO	Texas Utilities Generating Company
TVA	Tennessee Valley Authority
UE	Union Electric Company
VEPCO	Virginia Electric and Power Company
VYNP	Vermont Yankee Nuclear Power Corporation
<u>W</u>	Westinghouse Electric Corporation
WEPCO	Wisconsin Electric Power Company
WNP	Washington Nuclear Project
WPPSS	Washington Public Power Supply System
WPS	Wisconsin Public Service Corporation
YAECO	Yankee Atomic Electric Company