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REVIEW OF THE OPERATING EXPERIENCE FOR SOUTH TEXAS 1 AND 2
FROM JANUARY 1991 - DECEMBER 1992

Engineering Technology Division
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1.0 INTRODUCTION

The Nuclear Operations Analysis Center (NOAC) was requested by NRC's Office for Analysis and evaluation of Operational Data (AEOD) to review the operating experience from January 1991 through December 1992 for the South Texas 1 and 2 plants. This review will assist NRC staff in preparing for a Diagnostic Team evaluation of the South Texas 1 and 2 plants.

As compared to other operating experience reviews conducted by NOAC, this review focused on selected areas and will not provide overall findings regarding plant operations. Any findings or observations are relevant only to the specific area analyzed.

Tables 1.1 through 5.2 in the report reflect the same information normally compiled for a comprehensive review of operating experience. Based on a review of this data, the following areas were chosen for further analysis:

- Licensee Event Reports (LERs) involving reportability criterion 50.73(a)(2)(iv) - ESF actuations (Table 2.1)
- LERs involving reportability criterion 50.73(a)(2)(ii) - Unanalyzed conditions (Table 2.1)
- Personnel errors involving intrinsic human error associated with operations activities (Tables 3.1 through 3.3)
- Personnel errors involving task description inadequacies associated with testing/calibration and operations activities (Tables 3.1 through 3.3)
- Component failures involving AC circuit breakers, toxic gas primary elements, cables and wires, isolation valves, and fasteners (Table 4.1)
- Train and system occurrences involving the residual heat removal, primary coolant, auxiliary feedwater, and chilled water systems (Tables 5.1, 5.2).

The operating performance of South Texas 1 and 2 is compared to other plants similar in design. Table 1.1 describes all of the plants in the peer group of new 3- and 4-loop Westinghouse reactors. All peer group data presented excludes the contribution of South Texas 1 and 2 to the peer group averages.

The data in the tables was derived from LER information contained in the Sequence Coding and Search System (SCSS). The indicated number of personnel errors, component failures, system occurrences, etc., presented in the tables reflects actual numbers of errors or failures as encoded in SCSS, not a count of LERs involving those failures. Note that a single LER may involve multiple errors or failures, resulting in more errors and failures than LERs.

Appendix A lists the abstracts of events for South Texas 1 and 2 which were included in this review. Three LERs which occurred in 1992 were not yet available in the SCSS database (498/92-021, 499/92-009, 499/92-010). LER 498/92-021 describes a technical specification violation caused by a failure to properly perform response time testing of the main steam isolation bypass valves. LER 499/92-009 describes a missed surveillance caused by a faulty modem from a toxic gas monitor. LER 499/92-010 describes a reactor trip caused by failure of a driver card in the control system for a feedwater control valve.

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Table 1.1 New 3- and 4-Loop Westinghouse Plant Peer Group

Plant Name	Docket	Initial Criticality	Commercial Operation	Electrical Rating
Beaver Valley 2	412	8/4/87	11/17/87	833
Braidwood 1	456	5/29/87	7/29/88	1120
Braidwood 2	457	3/8/88	10/17/88	1120
Byron 1	454	2/2/85	9/16/85	1120
Byron 2	455	1/9/87	8/21/87	1120
Callaway 1	483	10/2/84	12/19/84	1171
Catawba 1	413	1/7/85	6/29/85	1145
Catawba 2	414	5/8/86	8/19/86	1145
Comanche Peak 1	445	4/3/90	8/13/90	1150
Diablo Canyon 1	275	4/29/84	5/7/85	1086
Diablo Canyon 2	323	8/19/85	3/13/86	1119
Harris 1	400	1/3/87	5/2/87	900
McGuire 1	369	8/8/81	12/1/81	1180
McGuire 2	370	5/8/83	3/1/84	1180
Millstone Point 3	423	1/23/86	4/23/86	1154
Seabrook 1	443	6/13/89	8/19/90	1200
Sequoyah 1	327	7/5/80	7/1/81	1148
Sequoyah 2	328	11/5/81	6/1/82	1148
South Texas 1	498	3/8/88	8/25/88	1250
South Texas 2	499	3/12/89	6/19/89	1250
Summer 1	395	10/22/82	1/1/84	900
Vogtle 1	424	3/9/87	6/1/87	1101
Vogtle 2	425	3/28/89	5/20/89	1101
Wolf Creek 1	482	5/22/85	9/3/85	1170

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2.0 ANALYSIS OF LERS AS A FUNCTION OF REPORTABILITY CODES

Table 2.1 compares the percentage of LERS in various reportability categories for events occurring at South Texas 1 and 2 to the peer group percentages. This table indicates that both South Texas 1 and 2 reported a higher percentage of events resulting in ESF actuations than the peer group. South Texas 2 also reported a higher percentage of LERS classified as unanalyzed conditions than the peer group. Events at both South Texas 1 and 2 in all other categories were at or below peer group percentages. The following sections provide detailed analyses of events which: (1) resulted in ESF actuations, or (2) were classified as unanalyzed conditions.

2.1 10 CFR 50.73(a)(2)(iv) ESF Actuations

As shown in Table 2.2, South Texas 1 and 2 ranked fourth and second in the peer group, respectively, in the number of events that resulted in ESF actuations. During the review period, South Texas 1 reported 22 events, and South Texas 2 reported 15 events. Included in the ESF actuations are events that resulted in RPS actuations. Table 2.3 compares the number of events that resulted in RPS actuations while the reactor was critical for each plant in the peer group. South Texas 2 ranked second in the peer group with 7 RPS actuations, and South Texas 1 was average with 4 RPS actuations.

2.1.1 South Texas 1. Three of the four RPS actuations at South Texas 1 involved personnel errors (498/91-021, 91-022, 92-003). These include operations, maintenance and administrative errors. Personnel errors are further described in Section 3. The fourth RPS actuation (498/91-012) involved random failure of a timer relay.

Three events involved spurious high readings on a toxic gas analyzer for the control room (498/91-003, 91-010, 91-017). Two events (498/92-001, 91-008) involved spurious actuation of the containment ventilation isolation system. One event (498/91-015) involved failure in a sequencer test circuit, which resulted in an inadvertent start of an auxiliary feedwater pump. One event involved breaker phase-to-ground flashover caused by a failure of a snap ring which held the connecting pin in place (498/91-007). The remaining 11 events (498/91-002, 91-004, 91-008, 91-013, 92-005, 92-007, 92-009, 92-010, 92-014, 92-015, 92-016) involved personnel errors, including operations, maintenance and administrative errors. These errors are further described in Section 3.

2.1.2 South Texas 2. Three of the seven RPS actuations at South Texas 2 involved personnel errors (499/001, 91-007, 91-010). These included operations, maintenance and administrative errors, which are further described in Section 3. Two RPS actuations were caused by a difference in saturation rates of the current transformer associated with relay 87-1/G1 (499/91-003, 91-004). One event (499/92-001) was associated with a failed diode, which resulted in dropping a control rod into the reactor core. One event (499/92-003) involved the loss of all three turbine-driven steam generator feedwater pumps due to rain leakage into the electrohydraulic control cabinet that housed the controls for the three pumps.

Three of the ESF actuations involved spurious high readings of the toxic gas monitor for the control room (498/91-005, 91-006, 92-008). Two events involved spurious actuation of the containment ventilation isolation system (499/91-008, 92-005). One event (499/91-009) involved failure in a sequencer test circuit, which resulted in an inadvertent start of an auxiliary feedwater pump. This event was similar to 498/91-015. One event (499/92-006) involved the failure of both power supplies for the digital rod position indication panel. One event (499/92-007) involved inadequate training to prevent "burping" of solenoid operated valves.

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Table 2.1 Comparison of Reportability Codes at South Texas 1 and 2 and Other Peer Group Plants

Reportability Category	Percentage of all Peer Groups LERs	Percentage of South Texas 1 LERs	Percentage of South Texas 2 LERs
10 CFR 50.73(a)(2)(iv) ESF Actuations	35	50	83
10 CFR 50.73(a)(2)(i) Shutdowns or Technical Specification Violations	51	39	6
Other: Voluntary report, special report, Part 21 report, etc.	8	9	--
10 CFR 50.73(a)(2)(v) Event that could have prevented fulfillment of a safety function	8	7	--
10 CFR 50.73(a)(2)(vii) Single failure criteria	7	5	--
10 CFR 50.73(a)(2)(ii) Unanalyzed condition	7	5	11

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Table 2.2 Number of LERs Reporting ESF Actuations at South Texas 1 and 2 and Other Peer Group Plants

Plant	Docket	Number of LERs Reporting ESF Actuations
Comanche 1	445	24
South Texas 2	499	22
Vogtle 2	425	16
South Texas 1	498	15
Diablo Canyon 1	275	13
McGuire 2	370	12
Vogtle 1	424	12
Seabrook 1	443	12
Millstone 3	423	10
Catawba 2	414	9
Braidwood 1	456	9
Beaver Valley 2	412	9
Shearon Harris 1	400	9
Braidwood 2	457	8
Catawba 1	413	8
Callaway 1	483	8
Sequoyah 2	328	8
Wolf Creek 1	482	8
McGuire 1	369	7
Sequoyah 1	327	7
Byron 2	455	6
Summer 1	395	5
Diablo Canyon 2	323	4
Byron 1	454	3

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Table 2.3 Number of LERs reporting RPS Actuations While Critical at South Texas 1 and 2 and Other Peer Group Plants

Plant	Docket	Number of LERs Reporting RPS Actuations While Critical
Comanche 1	445	10
South Texas 2	499	7
McGuire 2	370	7
Seabrook 1	443	7
Diablo Canyon 1	275	6
Braidwood 2	457	6
Vogtle 2	425	5
South Texas 1	498	4
McGuire 1	369	4
Catawba 1	413	4
Millstone 3	423	4
Callaway 1	483	4
Shearon Harris 1	400	4
Sequoyah 2	328	4
Sequoyah 1	327	3
Catawba 2	414	2
Byron 2	455	2
Wolf Creek 1	482	2
Summer 1	395	2
Braidwood 1	456	1
Byron 1	454	1
Vogtle 1	424	1
Beaver Valley 2	412	1

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2.2 10 CFR 50.73(a)(2)(i) Unanalyzed Conditions

South Texas 1 and 2 each had 2 events classified as unanalyzed conditions. The events at South Texas 1 included improper design of the pressurizer safety relief valves (498/91-024) and the power operated relief valve on the cold overpressure mitigation system (498/92-019). These events were discovered through NRC information notices and through industry notifications. A justification for continued operation was issued in each case.

The events at South Texas 2 included a failure to properly update the technical specification for an overtemperature delta temperature trip setpoint (499/92-002) and identification of a need to revise technical specification 3.6.3 to help prevent unnecessary shutdowns due to loss of containment isolation valves (499/092-004).

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3.0 ANALYSIS OF PERSONNEL ERRORS

Summaries of personnel errors reported at South Texas 1 and 2 are presented in Tables 3.1 and 3.2, respectively. A summary of average personnel errors for the peer group is presented in Table 3.3 for comparison. The tables indicate that South Texas 1 has a much higher number of personnel errors reported than the peer group average during the review period. Intrinsic human errors and task description inadequacies were particularly numerous at South Texas 1. South Texas 2 reported a lower number of personnel errors than the peer group average in every category.

3.1 Intrinsic Human Errors

Eleven events at South Texas 1 involved intrinsic human errors involving administrative activities. Of these, eight were directly related to task description inadequacies, and are discussed in Section 3.2. One event not related to task description inadequacies involved misunderstanding of the requirements of the containment integrity technical specification, resulting in a violation (498/92-002). The other two events involved late reporting of technical specification violations due to inadequate understanding of the reporting requirements (498/91-010, 92-009).

Five events involved intrinsic human errors involving operations activities. Two of these events involved poor communication: one during performance of the ESF power availability surveillance (498/91-006), and the other during addition of corrosion inhibitor, which resulted in failure to reopen the makeup water valve to the component cooling water surge tank (498/92-016).

The other three events involved inattention to detail, including:

- delay in noticing improper sequencing of loads following startup of a diesel generator (498/91-008)
- failure to notice improper positioning of the auxiliary feedwater flow control valves (498/92-006)
- failure to properly follow procedures, resulting in actuation of a component cooling water pump (498/92-015).

3.2 Task Description Inadequacy

Twelve events at South Texas 1 involved testing/calibration procedural deficiencies (four events had inadequacies which applied to both South Texas 1 and 2, which resulted in a total of 16 deficiencies counted in the database). These procedural inadequacies included the following:

- three events associated with surveillance procedures developed by someone unfamiliar with the systems (498/92-004, 92-013, 92-017). Procedural reviews were not adequate to detect these procedural errors
- lack of controls to ensure that the fuel handling building truck door remains closed during refueling (498/91-005)

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Table 3.1 Personnel Activity Versus Cause For Personnel Errors at South Texas 1

Personnel Activity	Intrinsic Human Error	Task Description Inadequacy	Unknown Cause	Inadequate Man-Machine Interface	Proper Human Action	Total
Maintenance	4	7	1	0	0	12
Testing/Calibration	6	16	0	0	0	22
Design	4	0	0	0	0	4
Administrative	11	2	0	0	0	13
Operations	6	6	0	0	2	14
Installation	0	0	0	0	0	0
Fabrication	4	0	0	0	0	4
Radiation Protection	0	0	0	0	0	0
Construction	0	0	0	0	0	0
Unknown	0	0	0	0	0	0
Total	35	31	1	0	2	69

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Table 3.2 Personnel Activity Versus Cause For Personnel Errors at Other Peer Group Plants (average number of errors per plant)

Personnel Activity	Intrinsic Human Error	Task Description Inadequacy	Unknown Cause	Inadequate Man-Machine Interface	Proper Human Action	Total
Maintenance	4	3	0	0	0	7
Testing/Calibration	4	9	0	0	0	13
Design	4	0	0	0	0	4
Administrative	5	1	0	0	0	6
Operations	3	3	0	0	0	6
Installation	0	0	0	0	0	0
Fabrication	1	0	0	0	0	1
Radiation Protection	0	0	0	0	0	0
Construction	0	0	0	0	0	0
Unknown	0	0	0	0	0	0
Total	21	16	0	0	0	37

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Table 3.3 Personnel Activity Versus Cause for Personnel Errors at South Texas 2

Personnel Activity	Intrinsic Human Error	Task Description Inadequacy	Unknown Cause	Inadequate Man-Machine Interface	Proper Human Action	Total
Maintenance	2	1	0	1	0	4
Testing/Calibration	0	1	0	0	0	1
Design	0	0	0	0	0	0
Administrative	4	0	0	0	0	4
Operations	3	1	0	0	2	6
Installation	0	0	0	0	0	0
Fabrication	0	0	0	0	0	0
Radiation Protection	0	0	0	0	0	0
Construction	0	0	0	0	0	0
Unknown	0	0	1	0	0	1
Total	9	3	1	1	2	16

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- lack of controls to ensure surveillance testing is performed, when not completed within one shift (498/91-016)
- post maintenance test procedure inadequate to identify operability of the rod position deviation monitor (498/91-020)
- insufficient emphasis on the risk associated with performance of a calibration of the reactor coolant flow transmitter (498/92-003)
- lack of guidance for performing a surveillance test of a component cooling water pump (498/92-005)
- lack of distinction between steps which verify equipment startup and steps which require an attempted startup (498/92-009)
- lack of complete procedures for changing component cooling water pump configurations (498/92-010)
- improper sequence of steps in the containment ventilation isolation actuation and response time test (498/92-014)
- poor administrative review of the power operated relief valve setpoint curves for the cold overpressure mitigation system (498/92-019).

Six events involved operational procedure inadequacies. These events included:

- lack of controls to ensure that the fuel handling building truck door remains closed during refueling (498/91-005)
- using the wrong procedure during the performance of the ESF power availability surveillance (498/91-006)
- lack of proper procedures following a partial loss of offsite power (498/91-008)
- lack of controls to ensure proper positioning of the auxiliary feedwater flow control valves (498/92-006)
- poorly written procedures requiring extra operator attention (498/92-015)
- lack of procedural step to verify valve position during injection of corrosion inhibitor to the component cooling water surge tank (498/92-016)

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4.0 ANALYSIS OF COMPONENT FAILURES

Single failures of components are not generally required to be reported in LERs. However, component failures are frequently initiators of reportable events. These failures can be analyzed to determine trends, however this analysis should not be confused with a comprehensive component failure analysis.

Table 4.1 presents the dominant component failures at South Texas 1 and 2 and compares these to peer group averages. Failures are defined as actual or potential undesired equipment performance which would result in a repair action. A repair action would include replacing a power supply, rebuilding a pump, or repacking a valve. Resetting switches or manipulating valves do not constitute repair actions.

4.1 AC Circuit Breakers

South Texas 1 reported four occurrences involving AC circuit breakers, as compared to none for South Texas 2 and an average of one for the peer group. These occurrences were reported in two LERs. One event (498/91-008) involved improper lubrication of a load center feeder breaker. Not only was the breaker not greased as needed, but an improper grease was used. The other event (498/91-007) involved the failure of a snap ring which resulted in a phase-to-ground flashover.

4.2 Toxic Gas Primary Elements

South Texas 2 reported three occurrences involving toxic gas primary elements, as compared to none at South Texas 1 and an average less than one for the peer group. These occurrences include two spurious actuation signals due to a failed circuit board (499/91-006) and one spurious signal due to a failed infrared source (499/92-008).

4.3 Cables and Wires

South Texas 1 reported three occurrences involving cables and wires, as compared to none at South Texas 2 and an average of one for the peer group. Two occurrences involved cracked insulation on the leads to all of the residual heat removal motors (498/91-023). These cracks did not result in a failure of the motors. The other occurrence involved a random failure of a connection to a radiation monitor, which resulted in a spurious actuation of the containment ventilation isolation system at South Texas 1 (498/92-008).

4.4 Isolation Valves

South Texas 2 reported three occurrences involving isolation valves, as compared to none at South Texas 1 and an average of one for the peer group. One occurrence involved a leaking pressurizer spray valve which was blocked in during a pressure transient, thus contributing to the opening of the pressurizer relief valves (499/91-007). The other occurrences involved the failure of both containment isolation valves for penetration M-86. The cause of the failures was not identified in the LER.

4.5 Fasteners

South Texas 1 reported two occurrences involving fasteners, and South Texas 2 reported three occurrences, as compared to an average of one for the peer group. One occurrence involved loose screws on a fuse block (499/91-001). Two occurrences involved disengaged linkage arms on spray valves in the safety injection system (499/91-010). Both occurrences at South Texas 1 involved a failed snap ring on a breaker, which resulted in a breaker phase-to-ground flashover (498/91-007).

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Table 4.1 Dominant Component Failures at South Texas 1 and 2 and Other Peer Group Plants

Component	Peer Group Plants (avg)	South Texas 1	South Texas 2
AC Circuit Breaker	1	4	0
Toxic Gas Primary Element	0	0	3
Cable/Wire	1	3	0
Isolation Valve	1	0	3
Fastener	1	2	3

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5.0 ANALYSIS OF SYSTEM AND TRAIN OCCURRENCES

Table 5.1 summarizes the dominant train failures at South Texas 1 and 2 during the review period, and compares these to the peer group averages. Table 5.2 summarizes the dominant system occurrences and compares South Texas 1 and 2 to the peer group averages. These failures and occurrences are defined as undesired performance, alignments, or configurations of systems, not just catastrophic failures or instances of systems not performing when called upon.

5.1 Residual Heat Removal System

For South Texas 1, a total of five residual heat removal train failures, and 16 system occurrences, were counted in the SCSS database. While these figures are much higher than peer group averages, note that these occurrences were reported in three events (still a higher figure than the peer group average). Two events were caused by tripping electrical breakers during refueling outages (498/91-007, 91-008). One event involved the discovery of cracking of the motor lead insulation on all residual heat removal pumps (498/91-023). The cracks did not lead to a loss of the pumps.

5.2 Primary Coolant System

For South Texas 1, a total of four train failures were reported in three separate LERs. Two of the events involved intrinsic human error which resulted in reactor trips. One event (498/91-022) involved an operator failing to properly perform a functional test of the solid state protection system logic train, the other event (498/91-021) involved an electrician misapplying multimeter test leads resulting in actuation of a lockout relay and loss of power. The third event involving train failure resulted from review of NRC information notice 89-90, which indicated that the pressurizer safety relief valves were improperly designed (498/92-024).

5.3 Auxiliary Feedwater System

South Texas 1 reported one auxiliary feedwater train failure, and South Texas 2 reported two, as compared to a peer group average of two failures. These events include mispositioning the four auxiliary feedwater flow control valves at South Texas 1 (498/92-006), and failure to perform a required pressure test before placing a steam supply line to a turbine-driven pump in service at South Texas 2 (499/91-002). Neither event resulted in a loss of auxiliary feedwater.

5.4 Chilled Water System

South Texas 1 reported a technical specification violation when one chiller was declared inoperable due to a low oil level indication while another chiller was also inoperable. This event lasted for less than 10 minutes (498/92-001). Note that investigation of this event revealed that other occurrences similar to this event were not properly identified as violations.

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Table 5.1 Summary of Train Failures at South Texas 1 and 2 and Other Peer Group Plants

System	Average number of Train Failures at Peer Group Plants	Number of Train Failures at South Texas 1	Number of Train Failures at South Texas 2
Residual Heat Removal	1	5	0
Primary Coolant	1	4	2
Auxiliary Feedwater	2	1	2
Chilled Water	0	1	1

Table 5.2 Summary of System Occurrences at South Texas 1 and 2 and Other Peer Group Plants

System	Average Number of system failures at Peer Group Plants	Number of system failures at South Texas 1	Number of system failures at South Texas 2
Residual Heat Removal	3	16	0
Nonnuclear Instrumentation	10	14	11
Containment Isolation	6	4	14
High Voltage AC	3	13	4
Reactor Protection	3	12	2
Low Voltage AC	4	11	4
Component Cooling Water	2	11	2

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APPENDIX A: LISTING OF ABSTRACTS FOR SOUTH TEXAS 1 AND 2 LERS

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Table A.1 Listing of LERs in Analyzed Categories

LERs for South Texas 1 ESF Actuations

1 498/91-001	2 498/91-002	3 498/91-004	4 498/91-007
5 498/91-003	6 498/91-008	7 498/91-010	8 498/91-012
9 498/91-013	10 498/91-015	11 498/91-017	12 498/91-022
13 498/91-021	14 498/92-003	15 498/92-005	16 498/92-007
17 498/92-008	18 498/92-009	19 498/92-010	20 498/92-014
21 498/92-015	22 498/92-016		

LERs for South Texas 2 ESF Actuations

1 499/91-001	2 499/91-003	3 499/91-004	4 499/91-005
5 499/91-006	6 499/91-007	7 499/91-008	8 499/91-009
9 499/91-010	10 499/92-001	11 499/92-003	12 499/92-005
13 499/92-006	14 499/92-007	15 499/92-008	

LERs for South Texas 1 RPS Actuations

1 498/91-012	2 498/91-022	3 498/91-021	4 498/92-003
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LERs for South Texas 2 RPS Actuations

1 499/91-001	2 499/91-003	3 499/91-004	4 499/91-007
5 499/91-010	6 499/92-001	7 499/92-003	

LERs for South Texas 1 Intrinsic Administrative Errors

1 498/91-002	2 498/91-006	3 498/91-010	4 498/91-013
5 498/91-020	6 498/92-002	7 498/92-004	8 498/92-009
9 498/92-013	10 498/92-017	11 498/92-019	

LERs for South Texas 1 Intrinsic Operations Errors

1 498/91-006	2 498/91-008	3 498/92-006	4 498/92-015
5 498/92-016			

LERs for South Texas 1 Testing/Calibration Task Description Inadequacies

1 498/91-005	2 498/91-016	3 498/91-020	4 498/92-003
5 498/92-004	6 498/92-005	7 498/92-009	8 498/92-010
9 498/92-013	10 498/92-014	11 498/92-017	12 498/92-019

LERs for South Texas 1 Operations Task Description Inadequacies

1 498/91-005	2 498/91-006	3 498/91-008	4 498/92-006
5 498/92-015	6 498/92-016		

LERs For South Texas 1 Component Failures

AC Circuit Breakers

1 498/91-007	2 498/91-008
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Cable/wire

1 498/91-023	2 498/92-008
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Fastener

1 498/91-007

LERs for South Texas 2 Component Failures

Toxic Gas Primary Element

1 499/91-006	2 499/92-008
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Isolation Valve

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1 499/91-007 2 499/92-004

Fastener

1 499/91-001 2 499/91-010

LERs for South Texas 1 Train Failures

Residual Heat Removal

1 498/91-007 2 498/91-008 3 498/91-023

Primary Coolant

1 498/91-021 2 498/91-022 3 498/91-024

Auxiliary Feedwater

1 498/92-006

Chilled Water

1 498/92-001

LERs for South Texas 2 Train Failures

Primary Coolant

1 499/91-001

Auxiliary Feedwater

1 499/91-002

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Table A.2 Abstracts of LERs Reported at South Texas 1 and 2

FORM 1 LER SCSS DATA 02-23-93

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
499 1991 001 0 9102110269 221012 01/09/91

ABSTRACT

POWER LEVEL - 100%. ON JANUARY 9, 1991, UNIT 2 WAS IN MODE 1 AT 100% POWER. AT 2207 HOURS, FEEDWATER ISOLATION VALVE (FWIV) 2C CLOSED DURING THE INVESTIGATION FLOW NITROGEN AND LOW HYDRAULIC PRESSURE ALARMS FOR FWIV 2C. THE RESULTANT LOSS OF FEEDWATER FLOW CAUSED A DECREASE IN STEAM GENERATOR (SG) LEVEL AND THE REACTOR WAS MANUALLY TRIPPED. THE CAUSE OF THE MANUAL REACTOR TRIP WAS A FAILED- CLOSED FEEDWATER ISOLATION VALVE. THE FEEDWATER ISOLATION VALVE CLOSED WHEN AN OPERATOR INCORRECTLY REMOVED A POWER SUPPLY FUSE TO THE TRIP SOLENOID. THE FUSE WAS REMOVED WHEN TRYING TO DETERMINE THE SOURCE OF POWER LOSS TO THE FWIV HYDRAULIC SKID. THIS WAS CAUSED BY FAILURE TO COORDINATE OPERATIONAL PROBLEM INVESTIGATION AND THE USE OF INFORMATION WITHOUT PROVIDING NECESSARY VERIFICATION; ANNUNCIATOR RESPONSE PROCEDURES DID NOT PROVIDE DIRECTION PERTAINING TO A LOSS OF POWER; AND LACK OF FORMAL TRAINING ON THE INVESTIGATION OF POWER SUPPLIES. CORRECTIVE ACTIONS INCLUDE: TRAINING OF LICENSED AND NON-LICENSED OPERATORS; REVISION OF ANNUNCIATOR RESPONSE PROCEDURES; AS WELL AS OTHER RECURRENCE MEASURES.

FORM 2 LER SCSS DATA 02-23-93

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
498 1991 001 0 9102260312 221075 01/22/91

ABSTRACT

POWER LEVEL - 000%. ON JANUARY 22, 1991, UNIT 1 WAS IN ITS THIRD REFUELING OUTAGE WITH NO FUEL IN THE REACTOR VESSEL. AT 1520 HOURS, A CONTAINMENT VENTILATION ISOLATION ACTUATION OCCURRED. OPERATIONS PERSONNEL VERIFIED THAT ALL EQUIPMENT ACTUATED AS DESIGNED. THE RADIATION MONITORING SYSTEM DID NOT INDICATE ANY HIGH RADIATION CONDITIONS. RADIATION LEVELS IN THE REACTOR CONTAINMENT BUILDING WERE DETERMINED TO BE NORMAL PRIOR TO AND FOLLOWING THE ACTUATION. THE CONTAINMENT VENTILATION ISOLATION ACTUATION APPEARS TO BE THE RESULT OF A SPURIOUS ACTUATION OF THE RADIATION MONITORING SYSTEM. HOWEVER, THE CAUSE OF THE SPURIOUS SIGNAL FROM THE RADIATION MONITORING SYSTEM COULD NOT BE DETERMINED.

FORM 3 LER SCSS DATA 02-23-93

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
498 1991 002 1 9111250173 223568 01/26/91

ABSTRACT

POWER LEVEL - 000%. ON JANUARY 26, 1991, UNIT 1 WAS IN ITS THIRD REFUELING OUTAGE WITH NO FUEL IN THE REACTOR VESSEL AND THE REACTOR COOLANT SYSTEM VENTED TO ATMOSPHERE. AT 0850 HOURS, DURING THE FIRST PERFORMANCE OF A PREVENTIVE MAINTENANCE (PM) WORK ACTIVITY, AN AUTOMATIC ACTUATION OF THE SAFETY INJECTION (SI) SYSTEM OCCURRED IN ONE OF THREE TRAINS (TRAIN C) AS A RESULT OF LESS THAN ADEQUATE PM WORK INSTRUCTIONS. ALL ASSOCIATED ENGINEERED SAFETY FEATURES (ESF) EQUIPMENT OPERATED AS EXPECTED. THE CAUSE OF THE LESS THAN ADEQUATE WORK INSTRUCTIONS WAS PERSONNEL ERROR IN THAT TWO SUPERVISORS FAILED TO REQUIRE FURTHER REVIEW OF WORK INSTRUCTIONS WHICH THEY BELIEVED HAD POTENTIAL FOR CAUSING AN UNPLANNED ESF ACTUATION. CORRECTIVE ACTIONS INCLUDE INACTIVATING THE SUBJECT PM AND THE ASSOCIATED PMS FOR THE OTHER ACTUATION TRAINS IN BOTH UNITS. THESE PMS WILL BE CORRECTED PRIOR TO FUTURE USE. FURTHER CORRECTIVE ACTIONS WERE TAKEN TO ISSUE A TRAINING BULLETIN TO APPROPRIATE OPERATIONS AND MAINTENANCE SUPERVISORS DESCRIBING THE EVENT, AND TO COUNSEL THE TWO SUPERVISORS ON THE NECESSITY OF PERFORMING THOROUGH REVIEWS OF PROCEDURES AND WORK INSTRUCTIONS THAT HAVE THE POTENTIAL TO CAUSE UNPLANNED ESF ACTUATIONS.

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FORM 4 LER SCSS DATA 02-23-93

 DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
 499 1991 002 0 9103120274 221189 01/31/91

ABSTRACT

POWER LEVEL - 100%. ON JANUARY 31, 1991, UNIT 2 WAS IN MODE 1 AT 100% POWER. DURING A REVIEW OF A COMPLETED WORK PACKAGE FOR WELD REPAIRS ON THE TURBINE-DRIVEN AUXILIARY FEEDWATER (AFW) PUMP 24 STEAM SUPPLY LINE, IT WAS DISCOVERED THAT THE ASME SECTION XI PRESSURE TEST REQUIRED BY TECHNICAL SPECIFICATION 4.0.5 HAD NOT BEEN PERFORMED PRIOR TO RETURNING THE SYSTEM TO SERVICE. THIS RESULTED IN THE AFW PUMP 24 BEING ADMINISTRATIVELY INOPERABLE FROM DECEMBER 5, 1990 TO FEBRUARY 3, 1991. THE CAUSES OF THIS EVENT ARE LESS THAN ADEQUATE PROCEDURAL CONTROLS WHICH ALLOWED THE PLANNER TO DEFER COMPLETION OF THE PRESSURE TEST DATA SHEET, LESS THAN ADEQUATE REVIEW OF THE REVISED WORK PACKAGE BY THE COGNIZANT SYSTEM ENGINEER AND LESS THAN ADEQUATE REVIEW OF THE POST MAINTENANCE TEST REQUIREMENTS PRIOR TO RETURN TO SERVICE. CORRECTIVE ACTIONS INCLUDE SUCCESSFUL PERFORMANCE OF THE CODE PRESSURE TEST, REVISION OF APPROPRIATE PROCEDURES AND TRAINING OF APPROPRIATE MAINTENANCE PLANNERS AND SYSTEM ENGINEERS.

FORM 5 LER SCSS DATA 02-23-93

 DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
 498 1991 004 0 9103260059 221277 02/15/91

ABSTRACT

POWER LEVEL - 000%. ON FEBRUARY 15, 1991, UNIT 1 WAS IN ITS THIRD REFUELING OUTAGE WITH NO FUEL IN THE REACTOR VESSEL. AT 0259 HOURS, A PARTIAL LOSS OF OFFSITE POWER OCCURRED DURING MAINTENANCE OF AN OVERCURRENT PROTECTION RELAY. THE SUPPLY BREAKER TO 13.8 KV STANDBY BUS 1H TRIPPED WHICH SUPPLIES POWER TO THE 4.16 KV ENGINEERED SAFETY FEATURES (ESF) BUS ETC. STANDBY DIESEL GENERATOR #13 LOADED AS REQUIRED, RESTORING POWER TO TRAIN C. THE CAUSE OF THIS EVENT WAS DETERMINED TO BE LACK OF ATTENTION TO WORK PERFORMANCE METHODS. AN ELECTRICIAN INADVERTENTLY TOUCHED THE TRIP CONTACT ON THE PROTECTIVE RELAY IN THE PROCESS OF INSERTING THE CONTACT PLUG. CORRECTIVE ACTIONS INCLUDE TRAINING OF MAINTENANCE PERSONNEL, REVISION OF APPROPRIATE GE RELAY CALIBRATION PROCEDURES AND ADDITION OF A TRAINING OBJECTIVE ON THE PROPER METHOD FOR INSTALLING RELAY CONTACT PLUGS.

FORM 6 LER SCSS DATA 02-23-93

 DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
 498 1991 005 0 9103270165 221278 02/18/91

ABSTRACT

POWER LEVEL - 000%. ON FEBRUARY 18, 1991, UNIT 1 WAS IN MODE 6, AT 1334 HOURS. THE FUEL HANDLING BUILDING (FHB) TRUCK DOOR WAS OPENED WHILE OPERATIONS PERSONNEL WERE INVOLVED IN FUEL HANDLING AS PART OF THE CORE RELOAD. AT APPROXIMATELY 1340 HOURS, AN OPERATOR ON THE FHB FUEL BRIDGE NOTED THAT THE DOOR WAS OPENED AND SECURED ALL FUEL MOVEMENT. THE DOORS WERE CLOSED AT 1359 HOURS. THE FHB EXHAUST AIR SYSTEM WAS RENDERED INOPERABLE WHEN THE FHB TRUCK DOORS WERE OPENED. FUEL MOVEMENT WAS SUSPENDED IMMEDIATELY UPON DISCOVERY AS REQUIRED BY TECHNICAL SPECIFICATION 3.9.12 UNTIL THE FHB VENTILATION SYSTEM WAS RESTORED TO AN OPERABLE CONDITION. THE CAUSE OF THIS EVENT WAS INCOMPLETE ADMINISTRATIVE CONTROLS ON THE FHB TRUCK DOOR. THERE WERE NO CONTROLS IN PLACE TO ENSURE THE APPROPRIATE TECHNICAL SPECIFICATION REQUIREMENTS FOR THE FHB EXHAUST AIR SYSTEM WERE FOLLOWED. CORRECTIVE ACTIONS INCLUDE PLACEMENT OF ADDITIONAL LOCKS ON THE FHB TRUCK DOORS, ISSUANCE OF A BULLETIN/NIGHT ORDERS TO SECURITY AND HEALTH PHYSICS PERSONNEL TO ENSURE THAT IN ADDITION TO SECURITY AND HEALTH PHYSICS THAT OPERATIONS PERSONNEL ARE ALSO PRESENT AT THE DOOR PRIOR TO OPENING, ISSUANCE OF A MEMORANDUM TO LICENSED OPERATORS DISCUSSING THIS INCIDENT, AN EVALUATION TO DETERMINE THE EFFECT OF THE FHB DOORS ON THE OPERABILITY OF THE FHB EXHAUST AIR SYSTEM, AND ESTABLISHMENT OF APPROPRIATE ADMINISTRATIVE CONTROLS.

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FORM 7 LER SCSS DATA 02-23-93

 DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
 498 1991 006 0 9103280307 221315 02/22/91

ABSTRACT

POWER LEVEL - 000%. ON FEBRUARY 24, 1991, UNIT 1 WAS IN MODE 6 IN ITS THIRD REFUELING OUTAGE. AT 0603 HOURS DURING PERFORMANCE OF A SURVEILLANCE TEST, IT WAS DISCOVERED THAT THE CLASS 1E 120 VOLT DISTRIBUTION PANEL DPO02 WAS ENERGIZED FROM ITS ALTERNATE POWER SUPPLY IN VIOLATION OF TECHNICAL SPECIFICATION 3.8.3.2. IMMEDIATE ACTIONS WERE TAKEN TO RESTORE THE DISTRIBUTION PANEL TO ITS PROPER ALIGNMENT. THE CAUSES OF THIS EVENT WERE FAILURE TO COORDINATE THE TRANSFER OF POWER TO THE DISTRIBUTION PANEL DUE TO INADEQUATE VERBAL COMMUNICATIONS AND FAILURE TO MONITOR THE ASSOCIATED ALARMS WHICH ANNUNCIATE IN THE CONTROL ROOM DURING AN UNDERVOLTAGE CONDITION. CORRECTIVE ACTIONS INCLUDE TRAINING OF LICENSED AND NON-LICENSED OPERATORS, AND AN EVALUATION OF THE PLANT'S NUMBERING SCHEME FOR ELECTRICAL PANELS.

FORM 8 LER SCSS DATA 02-23-93

 DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
 498 1991 007 1 9110180008 223158 03/09/91

ABSTRACT

POWER LEVEL - 000%. ON MARCH 9, 1991, UNIT 1 WAS IN COLD SHUTDOWN DURING A REFUELING OUTAGE, AND UNIT 2 WAS OPERATING AT 100 PERCENT POWER. WHILE RETURNING A TRANSMISSION LINE TO SERVICE, A SWITCHYARD BREAKER EXPERIENCED A PHASE TO GROUND FLASHOVER CAUSED BY A DISLODGED CONNECTING PIN IN THE INTERRUPTER LINKAGE MECHANISM. THE SNAP RING WHICH HOLDS THE CONNECTING PIN IN PLACE HAD FALLEN OUT. THE BREAKER FAULT CAUSED PROTECTION CIRCUITRY TO CLEAR THE SOUTH BUS AND OFFSITE POWER WAS LOST TO SEVERAL ENGINEERED SAFETY FEATURES (ESF) BUSES. ONE UNIT 1 STANDBY DIESEL GENERATOR (SBDG) AND TWO UNIT 2 SBDGS STARTED AND CARRIED THEIR LOADS. ALTHOUGH THE IMMEDIATE CAUSE OF THIS EVENT IS THE DISLODGED CONNECTING PIN, THE CAUSE FOR THE SNAP RING FALLING OUT OF PLACE IS NOT KNOWN. AS A RESULT OF THIS EVENT, THE BREAKER HAS BEEN REPAIRED AND A MODIFIED PIN DESIGN HAS BEEN INSTALLED IN ALL BREAKERS OF THE SAME MODEL AS THE BREAKER WHICH HAD A FAULT.

FORM 9 LER SCSS DATA 02-23-93

 DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
 498 1991 003 0 9102250306 221511 01/27/91

ABSTRACT

POWER LEVEL - 000%. ON JANUARY 27, 1991, UNIT 1 WAS IN ITS THIRD REFUELING OUTAGE WITH NO FUEL IN THE REACTOR VESSEL. AT 0335 HOURS A RADIATION MONITOR FOR THE CONTROL ROOM ENVELOPE WENT INTO HIGH ALARM AND ACTUATED THE CONTROL ROOM VENTILATION SYSTEM TO THE RECIRCULATION WITH FILTERED MAKEUP MODE. THE ALARM CLEARED AFTER APPROXIMATELY TWO MINUTES. SAMPLES OF THE CONTROL ROOM ATMOSPHERE DID NOT IDENTIFY ANY ACTIVITY. THE REDUNDANT MONITOR REMAINED IN THE NORMAL RANGE THROUGHOUT THIS PERIOD. MAINTENANCE WAS PERFORMED ON THE ACTUATED MONITOR AND THE MONITOR WAS SUCCESSFULLY CALIBRATED. THE CAUSE OF THIS EVENT IS UNKNOWN.

FORM 10 LER SCSS DATA 02-23-93

 DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
 498 1991 008 1 9110240262 223253 03/15/91

ABSTRACT

POWER LEVEL - 000%. ON MARCH 15, 1991, UNIT 1 WAS IN MODE 5 DUE TO A REFUELING OUTAGE. THE UNIT EXPERIENCED A PARTIAL LOSS OF OFFSITE POWER (LOOP) TO TRAIN A AT 1313 HOURS DUE TO ACTUATION OF THE UNIT AUXILIARY TRANSFORMER PILOT WIRE RELAY WHICH OPENED A SWITCHYARD BREAKER. DURING RECOVERY FROM THE FIRST LOOP, A LOOP OCCURRED ON TRAIN B OF UNIT 1 AT 1328 HOURS WHEN A 13.8 KV STANDBY BUS FEEDER BREAKER WAS OPENED BY A CONTROL ROOM OPERATOR. BOTH LOOP EVENTS WERE DUE TO INADEQUATE PROCEDURES. THE SUBJECT PROCEDURES HAVE BEEN REVISED APPROPRIATELY. IN ADDITION, A LOAD CENTER FEEDER BREAKER FAILED TO CLOSE DUE TO INADEQUATE LUBRICATION. WORK REQUESTS HAVE BEEN ISSUED TO ADDRESS PROPER LUBRICATION.

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FORM 11 LER SCSS DATA 02-23-93

 DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
 498 1991 009 1 9108050280 222690 03/11/91

ABSTRACT

POWER LEVEL - 000%. ON MARCH 10, 1991, A CRACKED FUEL INJECTOR NOZZLE TIP FROM LOT 150010 WAS FOUND IN STANDBY DIESEL GENERATOR (SDG) 12. ON MARCH 13, 1991, A CRACKED NOZZLE TIP FROM LOT 150006 WAS FOUND IN SDG 13. HOUSTON LIGHTING & POWER (HL&P) CONDUCTED EDDY CURRENT EXAMINATION OF 151 INJECTOR NOZZLE TIPS FROM THESE AND OTHER SDGS AS WELL AS SPARES, AND IDENTIFIED SEVERAL ADDITIONAL CRACKED NOZZLE TIPS FROM LOT 150006. COOPER-BESSEMER (THE SDG SUPPLIER) NOTIFIED THE NRC PURSUANT TO 10CFR21 AND HL&P FILED LER 91-009 REV. 0 ACCORDINGLY. ADDITIONAL INVESTIGATIONS RESULTED IN THE CONCLUSIONS THAT INADEQUATE LIGAMENT THICKNESS AND EXCESSIVE NITRIDING DEPTH ARE THE PROBABLE CAUSES OF THE FAILURES. HL&P ALSO REMOVED LOT 150009 FROM SERVICE, WHICH SHOWED CRACKING IN LABORATORY EXAMINATION, AND, AS A CONSERVATIVE MEASURE, ALL LOTS OF THE 1500XX SERIES MANUFACTURED BY ALLIED SIGNAL WERE ALSO REMOVED. A REVIEW OF CURRENT MANUFACTURING METHODS SHOWED THAT IMPROVEMENTS HAD BEEN MADE IN DIMENSIONAL CONTROL. THE INFORMATION DEVELOPED HAS BEEN SHARED WITH COOPER-BESSEMER AND MPR ASSOCIATES (PROJECT MANAGER OF THE COOPER-BESSEMER OWNER'S GROUP). ADDITIONAL RECURRENCE CONTROLS HAVE BEEN ADDED AT STPEGS INCLUDING EXAMINATION FOR DEPTH OF NITRIDING.

FORM 12 LER SCSS DATA 02-23-93

 DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
 498 1991 010 1 9111250183 223569 04/04/91

ABSTRACT

POWER LEVEL - 013%. ON APRIL 4, 1991, UNIT 1 WAS IN MODE 1 AT 13 PERCENT POWER. AT 0843 HOURS, THE MAIN CONTROL ROOM RECEIVED A TOXIC GAS HIGH CONCENTRATION ALARM. THE CONTROL ROOM VENTILATION SYSTEM WAS MANUALLY PLACED INTO THE RECIRCULATION MODE AS A CONSERVATIVE RESPONSE. NO TOXIC GAS WAS DETERMINED TO BE PRESENT AFTER AN IMMEDIATE INVESTIGATION. THE ALARM OCCURRED AS A RESULT OF A FAILURE IN THE EMERGENCY RESPONSE FACILITIES DATA ACQUISITION AND DISPLAY SYSTEM COMPUTER. THE CAUSE OF THE ALARM WAS A FAILED FIBER OPTIC DATA ACQUISITION CONTROLLER SUBSYSTEM PRINTED CIRCUIT BOARD. THE FAILED PRINTED CIRCUIT BOARD HAS BEEN REPLACED AS A RESULT OF THE EVENT.

FORM 13 LER SCSS DATA 02-23-93

 DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
 499 1991 003 0 9104220136 221975 03/14/91

ABSTRACT

POWER LEVEL - 100%. ON MARCH 14, 1991, UNIT 2 WAS OPERATING AT 100% WHILE UNIT 1 WAS IN MODE 5. AT 1810 HOURS, UNIT 1 CONTROL ROOM PERSONNEL CLOSED THE SWITCHYARD BREAKER TO ENERGIZE THE UNIT 1 MAIN AND AUXILIARY TRANSFORMERS. IMMEDIATELY FOLLOWING THIS BREAKER CLOSURE, THE UNIT 2 B PHASE GENERATOR ISOPHASE BUS DIFFERENTIAL RELAY ACTUATED. THIS CAUSED THE GENERATOR LOCKOUT RELAY TO ACTUATE WHICH RESULTED IN A TURBINE TRIP AND REACTOR TRIP. DURING THE RECOVERY PROCESS THE MAIN STEAM ISOLATION VALVES (MSIV) WERE CLOSED. A STEAM GENERATOR (SG) MSIV WAS SUBSEQUENTLY REOPENED WHILE A SG LEVEL WAS NEAR THE LOW-LOW SETPOINT AND CAUSED AN AUXILIARY FEEDWATER ACTUATION. THE PROTECTIVE RELAY ACTUATION WAS CAUSED BY DIFFERENCES IN THE SATURATION RATES OF THE TWO CURRENT TRANSFORMERS THAT SUPPLY THE DIFFERENTIAL RELAY. THE AFW ACTUATION WAS CAUSED BY OPERATING PROCEDURES THAT FAILED TO PROVIDE GUIDANCE REGARDING MINIMUM SG LEVELS DURING MSIV MANIPULATIONS. THE CORRECTIVE ACTIONS RELATIVE TO THE CURRENT TRANSFORMERS WILL BE REPORTED IN LER 91-004, WHICH DESCRIBES A SIMILAR SUBSEQUENT REACTOR TRIP EVENT. PROCEDURES WILL BE REVISED AND THIS EVENT WILL BE INCLUDED IN REQUALIFICATION TRAINING TO MINIMIZE THE POTENTIAL FOR UNNECESSARY AFW ACTUATIONS.

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FORM 14 LER SCSS DATA 02-23-93

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
499 1991 004 0 9105060214 221976 03/30/91

ABSTRACT

POWER LEVEL - 100%. ON MARCH 30, 1991, UNIT 2 WAS OPERATING AT 100% WHILE UNIT 1 WAS IN MODE 3. UNIT 1 CONTROL ROOM PERSONNEL CLOSED THE SWITCHYARD BREAKER TO ENERGIZE THE UNIT 1 MAIN AND AUXILIARY TRANSFORMERS. IMMEDIATELY FOLLOWING THIS BREAKER CLOSURE, THE UNIT 2 B PHASE GENERATOR ISOPHASE BUS DIFFERENTIAL RELAY ACTUATED. THIS CAUSED THE GENERATOR LOCKOUT RELAY TO ACTUATE WHICH RESULTED IN A TURBINE TRIP AND REACTOR TRIP. THE PROTECTIVE RELAY ACTUATION WAS CAUSED BY DIFFERENCES IN THE SATURATION RATES OF THE TWO CURRENT TRANSFORMERS THAT SUPPLY THE DIFFERENTIAL RELAY. AN EVALUATION IS UNDERWAY TO ESTABLISH THE FEASIBILITY OF HARDWARE CHANGES TO ADDRESS THIS PROBLEM. AS AN INTERIM MEASURE, A TEMPORARY MODIFICATION HAS BEEN INSTALLED THAT REMOVES THE PROTECTIVE FUNCTION FROM THE AFFECTED DIFFERENTIAL RELAY. REDUNDANT PROTECTION IS PROVIDED BY OTHER PROTECTIVE RELAYS.

FORM 15 LER SCSS DATA 02-23-93

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
498 1991 011 0 9105140266 222021 04/08/91

ABSTRACT

POWER LEVEL - 077%. ON 4/8/91, UNIT 1 WAS IN MODE 1 AT 77% POWER. AT 2205 HOURS, AN OPERABILITY TEST WAS PERFORMED ON THE TRAIN D FEEDWATER ISOLATION VALVE (FWIV). THE VALVE STROKED AS REQUIRED; HOWEVER, ONE OF THE TWO REDUNDANT SOLENOID VALVES WHICH ACTUATES THE FEEDWATER VALVE FAILED. SINCE THE CONDITIONS OF TECH SPEC 3.7.1.7 FOR MODES 1 AND 2 COULD NOT BE MET, A PLANT SHUTDOWN WAS INITIATED AND A NOTIFICATION OF UNUSUAL EVENT (NOUE) WAS DECLARED. THE NRC WAS NOTIFIED AT 0023 HOURS ON 4/9/91. THE FWIV WAS SECURED AND TAGGED AT 0650 HOURS AND UNIT 1 WAS BROUGHT TO MODE 3 AT 0803 HOURS. THE CAUSE OF THIS EVENT WAS FAILURE OF ONE OF TWO REDUNDANT FWIV SOLENOID VALVES TO OPERATE DUE TO HYDRAULIC FLUID POLYMERIZATION. CORRECTIVE ACTIONS INCLUDE ELIMINATING THE MAJOR SOURCE OF MOISTURE ENTRY INTO THE HYDRAULIC SYSTEM, FLUSHING THE HYDRAULIC SYSTEM AND REPLACING THE HYDRAULIC FLUID, REVISION OF PREVENTIVE MAINTENANCE ACTIVITIES AND PLANT MODIFICATIONS TO ADD CLEAN-UP SKIDS AND RELOCATE THE SOLENOID VALVES.

FORM 16 LER SCSS DATA 02-23-93

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
498 1991 012 1 9111070005 223301 04/12/91

ABSTRACT

POWER LEVEL - 040%. ON 4/12/91, AT 0418, THE UNIT 1 REACTOR TRIPPED FROM 40% POWER. A TURBINE TRIP, FEEDWATER ISOLATION AND AUXILIARY FEEDWATER ACTUATION OCCURRED AS A RESULT OF THE REACTOR TRIP. SYSTEMS OPERATED AS DESIGNED IN RESPONSE TO THE REACTOR TRIP. IT WAS DETERMINED THAT ROD DRIVE MOTOR GENERATOR (RDMG) SET #11 TRIPPED DUE TO A TRANSIENT INDUCED BY RDMG #12 WHICH WAS FOUND RUNNING WITH ITS MOTOR AND GENERATOR BREAKERS CLOSED WITH NO OUTPUT VOLTAGE TO THE REACTOR TRIP SWITCHGEAR. IT IS BELIEVED THAT INTERMITTENT PICK-UP AND DROP-OUT OF THE 2R RELAY, WHICH ACTUATES CONTACTS TO SUPPLY POWER TO THE RDMG SET #12'S GENERATOR VOLTAGE REGULATOR, CAUSED INSTABILITY IN THE VOLTAGE REGULATOR OPERATION. THE 2R RELAY MALFUNCTION WAS DUE TO A DEFECTIVE OUTPUT SWITCH. THE INSTABILITY OF THE VOLTAGE REGULATION RESULTED IN TRANSIENTS THAT CAUSED A REVERSE CURRENT TO THE RDMG SET #11 AND A SUBSEQUENT TRIP OF THE GENERATOR OUTPUT BREAKER. IT IS ALSO BELIEVED THAT THE 2R RELAY CONTACTS SUPPLYING POWER TO THE VOLTAGE REGULATOR EVENTUALLY REMAINED OPEN LONG ENOUGH TO ALLOW A LOSS OF THE GENERATOR FIELD IN THE RDMG SET #12. A LOSS OF THE GENERATOR FIELD RESULTS IN ZERO OUTPUT VOLTAGE FROM THE GENERATOR. THE LOSS OF BOTH OF THE POWER SOURCES TO THE REACTOR TRIP SWITCHGEAR RESULTED IN A REACTOR TRIP. THE 2R RELAY'S TIMER AND CONTROL RELAY WERE REPLACED AND A PROCEDURAL CHANGE HAS BEEN MADE TO ENHANCE DETECTION OF MALFUNCTION.

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FORM 17 LER SCSS DATA 02-23-93

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
498 1991 013 0 910513J376 222015 04/12/91

ABSTRACT

POWER LEVEL - 000%. ON APRIL 12, 1991, UNIT 1 WAS IN MODE 3 AT NORMAL OPERATING PRESSURE AND TEMPERATURE. AT 1321 HOURS, DURING TROUBLESHOOTING OF AN ENGINEERED SAFETY FEATURE (ESF) SEQUENCER AUTOMATIC TESTING FAILURE, A MODE III (SAFETY INJECTION COINCIDENT WITH LOSS OF OFFSITE POWER) SEQUENCER ACTUATION WAS INITIATED IN TRAIN B. THE ACTUATION RESULTED FROM LESS THAN ADEQUATE TROUBLESHOOTING INSTRUCTIONS. PLANT EQUIPMENT OPERATED AS DESIGNED AND THERE WERE NO SIGNIFICANT TRANSIENTS AS A RESULT OF THE ESF SEQUENCER ACTUATION. TROUBLESHOOTING PROGRAM PROCEDURES WILL BE REVISED AS A CORRECTIVE ACTION.

FORM 18 LER SCSS DATA 02-23-93

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
499 1991 005 1 9111180286 223454 04/11/91

ABSTRACT

POWER LEVEL - 100%. ON APRIL 11, 1991, UNIT 2 WAS IN MODE 1 AT 100 PERCENT POWER. AT 1130, AN AUTOMATIC ENGINEERED SAFETY FEATURES (ESF) ACTUATION OF CRE HVAC TRAINS B AND C TO EMERGENCY MODE OCCURRED. CONTROL ROOM ENVELOPE (CRE) HVAC TRAIN A HAD BEEN MANUALLY ACTUATED TO THE EMERGENCY MODE IN SUPPORT OF A SURVEILLANCE PROCEDURE. NO INDICATION OF A HIGH RADIATION OR SAFETY INJECTION SIGNAL WAS FOUND. THERE HAS BEEN NO CAUSE ESTABLISHED FOR THIS ACTUATION.

FORM 19 LER SCSS DATA 02-23-93

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
498 1991 014 2 9201220010 223809 04/20/91

ABSTRACT

POWER LEVEL - 100%. ON APRIL 20, 1991, UNIT 1 WAS IN MODE 1 AT 100% POWER. AT 0406 HOURS, WHILE CONDUCTING A CONTAINMENT SUPPLEMENTAL PURGE TO LOWER THE CONTAINMENT PRESSURE IN RESPONSE TO A CONTAINMENT HIGH PRESSURE ALARM, CONTAINMENT EXTENDED RANGE PRESSURE CHANNEL 9759 WAS FOUND TO READ 5 PSIG WHILE CHANNEL 9760 READ 0 PSIG. CHANNEL 9759 WAS DECLARED INOPERABLE AT 0407 HOURS. REVIEW OF HISTORICAL COMPUTER RECORDS INDICATED THAT THE CHANNEL HAD BEEN INOPERABLE IN EXCESS OF THE SEVEN-DAY ALLOWED OUTAGE TIME. AFTER INITIAL RECALIBRATION, SUBSEQUENT CHANNEL CHECK SURVEILLANCE REVEALED AN ADDITIONAL ERRATIC OUTPUT SIGNAL BY THE TRANSMITTER. THE TRANSMITTER CONTROL CARD WAS REPLACED AND THE TRANSMITTER WAS CALIBRATED. CHANNEL CHECKS WERE PERFORMED WEEKLY FOR ONE MONTH TO CONFIRM THE CHANNEL WAS REPAIRED. ALTHOUGH NO GENERIC FAILURE MECHANISM HAS BEEN ESTABLISHED, THE FAILURE RATES ARE CONSISTENT WITH INDUSTRY EXPERIENCE. THESE TRANSMITTERS ARE BEING MONITORED UNDER THE FACILITY TREADING PROGRAM.

FORM 20 LER SCSS DATA 02-23-93

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
498 1991 015 0 9105300179 222149 04/22/91

ABSTRACT

POWER LEVEL - 100%. ON APRIL 22, 1991, UNIT 1 WAS IN MODE 1 OPERATING AT 100% POWER. AT 0200, DURING PERFORMANCE OF A TRAIN "C" ENGINEERED SAFETY FEATURE SEQUENCER SURVEILLANCE TEST, THE TRAIN "C" AUXILIARY FEEDWATER (AFW) PUMP INADVERTENTLY STARTED. THE PUMP WAS SECURED AT 0208. THE CAUSE OF THIS EVENT WAS FAILURE OF A LIGHT EMITTING DIODE (LED) IN THE SEQUENCER TEST CIRCUITRY. THE LED HAS BEEN REPLACED. AN EVALUATION HAS DETERMINED THAT A SIMILAR FAILURE OF AN LED IN THE SEQUENCER ACTUATION CIRCUITRY, RATHER THAN THE TEST CIRCUITRY, WOULD PREVENT ACTUATION OF THE ASSOCIATED ESF COMPONENT. THE FUNCTIONALITY OF THE SEQUENCER IS TESTED QUARTERLY. IN ADDITION, IF SUCH A FAILURE OCCURRED, AN ALARM WOULD INDICATE THE AFFECTED COMPONENT HAD FAILED TO START AND OPERATOR ACTION COULD BE TAKEN TO START THE COMPONENT. THEREFORE, SINCE THERE HAS BEEN ONLY ONE SUCH FAILURE AT STP, NO ADDITIONAL CORRECTIVE ACTION IS PLANNED.

DRAFT

FORM 21 LER SCSS DATA 02-23-93

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
498 1991 016 0 9106160160 222192 05/13/91

ABSTRACT
POWER LEVEL - 100%. ON MAY 13, 1991, AT APPROXIMATELY 2230 HOURS, UNIT 1 WAS IN MODE 1 AT 100 PERCENT POWER. IT WAS DISCOVERED THAT THE TECHNICAL SPECIFICATION 3/4.7.1.4 REQUIREMENTS FOR DETERMINING THE SPECIFIC ACTIVITY OF THE SECONDARY COOLANT SYSTEM HAD NOT BEEN PERFORMED WITHIN THE REQUIRED SURVEILLANCE INTERVAL. THIS IS A VIOLATION OF TECHNICAL SPECIFICATION 3/4.7.1.4 AND IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(1)(B). STEAM GENERATOR BLOWDOWN RADIATION MONITOR DATA WAS CHECKED, AND IT WAS VERIFIED THAT SECONDARY ACTIVITY HAD NOT EXCEEDED NORMAL VALUES OR THE TECHNICAL SPECIFICATION LIMIT DURING THIS PERIOD. THE CAUSE OF THIS EVENT WAS FAILURE TO ENSURE TESTING WAS PERFORMED BEFORE EXCEEDING THE SURVEILLANCE INTERVAL. CORRECTIVE ACTIONS INCLUDED ISSUANCE OF SPECIAL ORDERS, AND CHANGING PROCEDURE AND LABORATORY SCHEDULES TO IMPROVE VISIBILITY AND INCREASE AWARENESS OF SURVEILLANCE TIMES.

FORM 22 LER SCSS DATA 02-23-93

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
499 1991 006 0 9106180374 222193 05/16/91

ABSTRACT
POWER LEVEL - 100%. ON MAY 16, 1991 UNIT 2 WAS IN MODE 1 AT 100 PERCENT POWER. AT 0558 HOURS, THE CONTROL ROOM VENTILATION SYSTEM ACTUATED TO THE RECIRCULATION MODE AS A RESULT OF A SPURIOUS TRIP FROM A TOXIC GAS ANALYZER. THE SPURIOUS ACTUATION SIGNAL SELF-RESET AT 0559 HOURS. ALSO AT 0042 HOURS ON MAY 21, 1991 ANOTHER SIMILAR ACTUATION OCCURRED FROM THE SAME ANALYZER AS THE FIRST EVENT. THE EXACT CAUSE OF BOTH EVENTS COULD NOT BE DETERMINED BUT HAS BEEN ATTRIBUTED TO POOR ELECTRICAL CONNECTION ON ONE OR MORE PLUG-IN INTEGRATED CIRCUIT CHIPS IN THE ANALYZER. CORRECTIVE ACTIONS INCLUDE TROUBLESHOOTING OF THE FAILED ANALYZER, FURTHER DESIGN IMPROVEMENTS TO MINIMIZE FALSE ACTUATION SIGNALS, AND DEVELOPMENT OF PREVENTIVE MAINTENANCE TASKS TO PERIODICALLY RESEAT INTEGRATED CIRCUIT CHIPS IN THE TOXIC GAS ANALYZERS.

FORM 23 LER SCSS DATA 02-23-93

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
499 1991 007 1 9202030253 223871 05/22/91

ABSTRACT
POWER LEVEL - 100%. ON 5/22/91, UNIT 2 WAS IN MODE 1 AT 100% POWER. AT APPROXIMATELY 2220 HRS, WHILE WAITING IN THE AREA OF THE MAIN GENERATOR BREAKER TO UNLOCK A LOCAL CABINET FOR AN ELECTRICAL MAINTENANCE INDIVIDUAL, A NON-LICENSED OPERATOR INADVERTENTLY ACTUATED THE LOCAL GENERATOR BREAKER EMERGENCY TRIP PUSHBUTTON. THE SUDDEN LOSS OF SECONDARY LOAD CAUSED AN AUTOMATIC OVER TEMPERATURE DELTA TEMPERATURE (OTST) REACTOR TRIP. PRESSURIZER SPRAY WAS UNABLE TO REDUCE THE PRESSURE BEFORE THE PRESSURIZER PORVS OPENED AT APPROXIMATELY 2335 PSIG. STEAM GENERATOR 2C POWER-OPERATED RELIEF VALVE (PORV) FAILED TO OPEN EVEN THOUGH THE PRESSURE EXCEEDED THE LIFT SETPOINT. THE NON-LICENSED OPERATOR RESPONSIBLE FOR THE TRIP WAS COUNSELLED WITH REGARDS TO PAYING STRICT ATTENTION TO PERFORMANCE OF OPERATIONS ACTIVITIES. THE STEAM GENERATOR 2C PORV HAS BEEN REPAIRED. OTHER SWITCH DESIGNS HAVE BEEN REVIEWED TO IDENTIFY CHANGES THAT CAN PREVENT SIMILAR INADVERTENT ACTUATIONS.

FORM 24 LER SCSS DATA 02-23-93

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
499 1991 008 0 9107020306 222393 05/25/91

ABSTRACT
POWER LEVEL - 000%. ON MAY 25, 1991, UNIT 2 WAS IN MODE 3 AT 2235 PSIG AND 567 DEGREES. AT 0107 A CONTAINMENT VENTILATION ISOLATION (CVI) ACTUATION OCCURRED. ON MAY 26, UNIT 2 WAS IN MODE 1 AT 75% POWER WHEN AT 0558 A SECOND CVI ACTUATION OCCURRED. TROUBLESHOOTING FOLLOWING THE ACTUATIONS INDICATED THAT A FAULTY RM-23 MODULE ASSOCIATED WITH ONE OF THE TWO PURGE EXHAUST RADIATION MONITORS (RT-8012) CAUSED THE TWO SPURIOUS ACTUATIONS. THE FAULTY MODULE HAS BEEN REPLACED. AN ANALYSIS IS BEING PERFORMED TO DETERMINE THE FAILURE MODE.

DRAFT

FORM 25 LER SCSS DATA 02-23-93

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
498 1991 017 0 9107020304 222516 05/26/91

ABSTRACT

POWER LEVEL - 100%. ON MAY 26, 1991, UNIT 1 WAS IN MODE 1 AT 100 PERCENT POWER. AT 1534 HOURS, THE CONTROL ROOM VENTILATION SYSTEM ACTUATED TO THE RECIRCULATION MODE AS A RESULT OF A SPURIOUS TRIP FROM A TOXIC GAS ANALYZER. THE EXACT CAUSE OF THE EVENT COULD NOT BE DETERMINED BUT HAS BEEN ATTRIBUTED TO POOR ELECTRICAL CONNECTION ON ONE OR MORE PLUG-IN INTEGRATED CIRCUIT CHIPS IN THE ANALYZER. CORRECTIVE ACTIONS INCLUDE TROUBLESHOOTING OF THE FAILED ANALYZER, FURTHER DESIGN IMPROVEMENTS TO MINIMIZE FALSE ACTUATION SIGNALS, AND DEVELOPMENT OF PREVENTIVE MAINTENANCE TASKS TO PERIODICALLY RESEAT INTEGRATED CIRCUIT CHIPS IN THE TOXIC GAS ANALYZERS.

FORM 26 LER SCSS DATA 02-23-93

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
498 1991 018 0 9108050243 222691 07/02/91

ABSTRACT

POWER LEVEL - 100%. ON JULY 2, 1991, UNITS 1 AND 2 WERE IN MODE 1 AT 100 PERCENT POWER. AT ABOUT 1300 HOURS, AN ENGINEER REALIZED THAT THE ALARM ASSOCIATED WITH THE CONDENSER AIR REMOVAL SYSTEM (CARS) WIDE RANGE NOBLE GAS ACTIVITY MONITOR WAS NOT FUNCTIONING. AS A RESULT THE CARS NOBLE GAS MONITOR WAS DECLARED INOPERABLE. IT WAS DETERMINED THAT THIS CONDITION HAS EXISTED SINCE THE STARTUP OF EACH UNIT. THE CAUSE OF THE EVENT HAS BEEN ATTRIBUTED TO MISUNDERSTANDING OF THE INTERNAL FUNCTIONS OF THE MONITOR WHEN PROCESS FLOW IS BELOW DESIGN VALUES. CORRECTIVE ACTIONS INCLUDE CHANGING AND REVIEWING THE DATABASE CONFIGURATIONS OF THE GAS ACTIVITY MONITORS, AND VERIFICATION OF THE PROCESS FLOW SUBSTITUTE VALUE FUNCTION.

FORM 27 LER SCSS DATA 02-23-93

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
499 1991 009 0 9108290165 222831 07/07/91

ABSTRACT

POWER LEVEL - 100%. ON JULY 7, 1991, UNIT 2 WAS IN MODE 1 AT 100% POWER. DURING PERFORMANCE OF AN A TRAIN ENGINEERED SAFETY FEATURE (ESF) SEQUENCER SURVEILLANCE TEST THE A TRAIN AUXILIARY FEEDWATER (AFW) PUMP INADVERTENTLY STARTED. THE TEST WAS SECURED AT 0220 HOURS. THE CAUSE OF THE EVENT APPEARS TO BE A FAILED OPEN LIGHT EMITTING DIODE (LED) IN THE CIRCUIT ASSOCIATED WITH THE BLOCKING RELAY FOR THE AFW PUMP. THIS CONCLUSION IS BASED ON INDICATIONS NOTED DURING ESF SEQUENCER TROUBLESHOOTING. THE CIRCUIT BOARD ASSOCIATED WITH THE AFW PUMP BLOCKING RELAYS AND THE BLOCKING RELAY CIRCUITS WERE TESTED SATISFACTORILY. AN ENGINEERING REVIEW WILL BE CONDUCTED TO DETERMINE IF A GENERIC PROBLEM MAY EXIST WITH THE LEDS USED IN SEQUENCER CIRCUITS.

FORM 28 LER SCSS DATA 02-23-93

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
498 1991 019 0 9110110115 223107 09/05/91

ABSTRACT

POWER LEVEL - 100%. ON SEPTEMBER 5, 1991, UNIT 1 WAS IN MODE 1 AT 100% POWER. AT 1806 HOURS, THE CONTROL ROOM RECEIVED A REACTOR COOLANT DRAIN TANK (RCDT) LEVEL HI-HI/LO-LO ALARM. AT 1838 HOURS, REACTOR COOLANT SYSTEM (RCS) LEAKAGE WAS DETERMINED TO BE APPROXIMATELY 15 GALLONS PER MINUTE (GPM), WHICH IS GREATER THAN THE TECHNICAL SPECIFICATION 3.4.6.2 LIMITS. AS A RESULT, THE PLANT DECLARED AN UNUSUAL EVENT. AT 1954 HOURS, PLANT PERSONNEL ENTERED THE REACTOR CONTAINMENT BUILDING (RCB) TO INVESTIGATE. BY ISOLATING NORMAL LETDOWN WITH EXCESS LETDOWN IN SERVICE AND OBSERVING THE LEAK RATE DECREASE, THE LEAKAGE WAS IDENTIFIED TO BE IN THE RCS LETDOWN VALVE A1CVLCV0465. THIS EVENT RESULTED FROM DAMAGED VALVE PACKING. THE VALVE WAS INSPECTED AND NO EVIDENCE WAS FOUND TO INDICATE A CAUSE FOR THE PACKING FAILURE. THE VALVE WAS REPACKED AND RETURNED TO AN OPERABLE STATUS. FURTHER CORRECTIVE ACTION WILL INVOLVE DISASSEMBLING THE VALVE DURING THE NEXT REFUELING OUTAGE FOR UNIT 1 TO ATTEMPT TO LOCATE AND REPAIR THE CAUSE FOR PACKING FAILURES.

DRAFT

FORM 29 LER SCSS DATA 02-23-93

 DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
 498 1991 020 0 9110250041 223254 09/14/91

ABSTRACT

POWER LEVEL - 100%. ON 9/14/91, AT 1439, UNIT 1 WAS IN MODE 1 AT 100% POWER WHEN THE ROD POSITION DEVIATION MONITOR WAS INCORRECTLY DECLARED OPERABLE BY THE SHIFT SUPERVISOR. THE ERROR WAS DISCOVERED ON 9/15/91, AT 0415 WHEN THE "ROD DEVIATION" ANNUNCIATOR WAS RECEIVED. DURING THE TIME THE MONITOR WAS INCORRECTLY CONSIDERED OPERABLE, TWO INCREASED FREQUENCY SURVEILLANCES WERE MISSED, RESULTING IN A TECH SPEC VIOLATION. THE CAUSE OF THE EVENT WAS THAT ERRORS WERE MADE BY THREE SHIFT SUPERVISORS IN IMPLEMENTING THE PROCEDURAL REQUIREMENTS REGARDING THE OPERABILITY TRACKING LOG SYSTEM. ALSO, THE SHIFT SUPERVISOR WHO INCORRECTLY DECLARED THE MONITOR OPERABLE DID NOT CONSULT ALL REFERENCES PRIOR TO MAKING AN OPERABILITY DETERMINATION. AN ADDITIONAL CAUSE WAS INADEQUATE IDENTIFICATION OF THE EFFECT OF THE TEMPORARY MODIFICATION PACKAGE WHICH DOCUMENTED THE MONITOR INOPERABILITY. A BRIEFING WILL BE GIVEN TO THE LICENSED OPERATORS STRESSING THE IMPORTANCE OF THE OPERABILITY TRACKING LOG SYSTEM AS DESCRIBED IN THE CONFIGURATION MANAGEMENT PROCEDURE. ALSO, A MEMO HAS BEEN SENT TO ALL SENIOR REACTOR OPERATORS STRESSING THE NEED TO EXHAUST ALL AVAILABLE REFERENCES PRIOR TO MAKING OPERABILITY DETERMINATIONS. THE TEMPORARY MODIFICATION REQUEST FORM HAS BEEN REVISED TO PROVIDE A CLEARER OPERABILITY DETERMINATION REMINDER.

FORM 30 LER SCSS DATA 02-23-93

 DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
 498 1991 022 0 9111190266 223453 10/14/91

ABSTRACT

POWER LEVEL - 100%. ON OCTOBER 14, 1991, AT 2304 HOURS, UNIT 1 WAS IN MODE 1 AT 100 POWER. SOLID STATE PROTECTION SYSTEM (SSPS) LOGIC TRAIN R FUNCTIONAL TEST WAS IN PROGRESS WHEN THE LICENSED OPERATOR PERFORMING THE SURVEILLANCE MISUNDERSTOOD THE INTENT OF A NOTE IN THE PROCEDURE AND FAILED TO BLOCK THE TURBINE TRIP SIGNAL BEFORE PROCEEDING TO THE NEXT STEP. THE "MEMORIES" TEST SWITCH WAS PLACED IN POSITION 16 AND AN AUTOMATIC TRAIN R TRIP SIGNAL WAS GENERATED. TRAIN R TRIP SIGNAL GENERATED A "TURBINE TRIP UPON REACTOR TRIP" SIGNAL WHICH HAD NOT BEEN BLOCKED AND THE "MEMORIES" TEST SWITCH ALSO MALFUNCTIONED, WHICH IF IT HAD FUNCTIONED PROPERLY SHOULD HAVE ALSO BLOCKED THE TRIP SIGNAL. SUBSEQUENTLY, THE MAIN TURBINE TRIPPED AND, COINCIDENT WITH A "REACTOR POWER ABOVE 50" SIGNAL, A VALID TRAINS REACTOR TRIP SIGNAL WAS GENERATED TRIPPING THE REACTOR. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR BY THE LICENSED OPERATOR WHO EXERCISED POOR JUDGEMENT WHILE PERFORMING THE TEST. CONTRIBUTING FACTORS WERE A LESS THAN IDEAL PROCEDURE AND THE MALFUNCTION OF THE "MEMORIES" TEST SWITCH. CORRECTIVE ACTIONS INCLUDE SITE-WIDE TRAINING SESSIONS FOR APPROPRIATE PLANT PERSONNEL STRESSING THE APPLICATION OF SELF VERIFICATION DURING WORK PERFORMANCE, COUNSELING OF THE LICENSED OPERATOR INVOLVED IN THE EVENT, REVISION OF ALL SP SERIES SURVEILLANCES PERFORMED AT POWER THAT HAVE THE POTENTIAL TO TRIP THE UNIT/MAIN TURBINE.

FORM 31 LER SCSS DATA 02-23-93

 DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
 498 1991 024 0 9111270005 223450 10/31/91

ABSTRACT

POWER LEVEL - 100%. DURING A REVIEW OF NRC INFORMATION NOTICE 89-90 SUPPLEMENT 1, DATED SEPTEMBER 5, 1991 AND WCAP-12910, WITH UNIT 1 IN MODE 1 AT 100 PERCENT POWER AND UNIT 2 IN A MODE 6 REFUELING OUTAGE, IT WAS DISCOVERED THAT THE UFSAR CHAPTER 15 SAFETY ANALYSIS DID NOT CONSIDER THE TIME REQUIRED TO PURGE THE LOOP SEAL FOR THE PRESSURIZER SAFETY RELIEF VALVES (SRVS). IMMEDIATE ACTIONS TAKEN TO INVESTIGATE THE PROBLEM CONFIRMED THAT THE CALCULATED PEAK RCS PRESSURE FOR THE LOCKED ROTOR EVENT WITH THE PRESSURIZER SRV LOOP SEAL DELAY TIME WOULD EXCEED THE NRC SAFETY LIMIT OF 110% DESIGN PRESSURE. ON OCTOBER 30, 1991, A STATION PROBLEM REPORT WAS ISSUED IDENTIFYING THE DEFICIENCY IN THE SAFETY ANALYSIS. ON NOVEMBER 5, 1991, A JUSTIFICATION FOR CONTINUED OPERATION (JCO) WAS ISSUED. THE JCO CONCLUDED THAT THE CONDITION DOES NOT RESULT IN EITHER SPEGS UNITS 1 OR 2 BEING IN AN UNSAFE CONDITION. THE CAUSE OF THE EVENT WAS THAT THE NSSS VENDOR DID NOT CONSIDER THE DELAY TIME ASSOCIATED WITH PURGING THE PRESSURIZER SRV LOOP SEALS IN THE SAFETY ANALYSIS. SINCE THE JCO FOR THIS ISSUE IDENTIFIED THAT NO UNSAFE CONDITION EXISTS, NO IMMEDIATE CORRECTIVE ACTIONS ARE PLANNED. AFTER THE NRC APPROVAL OF WCAP-12910 AND WESTINGHOUSE OWNERS GROUP (WOG) RESOLUTION OF THIS ISSUE, ADDITIONAL ACTIONS WILL BE DEVELOPED AS NECESSARY.

DRAFT

FORM 32 LER SCSS DATA 02-23-93

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
498 1991 021 0 9111250263 223570 10/10/91

ABSTRACT

POWER LEVEL - 100%. ON OCTOBER 10, 1991, AT 2056, UNIT 1 WAS IN MODE 1 AT 100% POWER WHEN POWER FROM THE 1J BUS WAS LOST. DURING THE PERFORMANCE OF WORK ACTIVITIES, A MAINTENANCE ELECTRICIAN MISAPPLIED MULTIMETER TEST LEADS IN AN ENERGIZED CIRCUIT WITH THE MULTIMETER SET TO READ "RESISTANCE". THE MISAPPLIED TEST LEADS ENERGIZED RELAY 2A WHICH ACTUATED THE 86X LOCKOUT RELAY CAUSING BREAKER P150 TO TRIP AND DE-ENERGIZE THE 1J BUS. UPON LOSS OF POWER ON 1J BUS, REACTOR COOLANT PUMP (RCP) 1D TRIPPED AND CAUSED A REACTOR TRIP DUE TO LOW COOLANT FLOW. THE BUS WAS RE-ENERGIZED AT 2059 FROM THE UNIT AUXILIARY TRANSFORMER WITH NO FURTHER INCIDENTS. THE PRIMARY CAUSE OF THIS EVENT WAS PERSONNEL ERROR. THE MAINTENANCE ELECTRICIAN'S ATTENTION TO DETAIL DURING WORK PERFORMANCE, ELEMENTARY DRAWING READING AND TROUBLESHOOTING TECHNIQUES WERE LESS THAN ADEQUATE. A TRAINING SESSION IS BEING HELD FOR APPROPRIATE PLANT PERSONNEL STRESSING THE APPLICATION OF SELF VERIFICATION DURING WORK PERFORMANCE. A TESTING PROGRAM WILL BE IMPLEMENTED TO ENSURE THAT APPLICABLE PERSONNEL ARE QUALIFIED TO USE ELEMENTARY DRAWINGS TO AID IN PERFORMANCE OF MAINTENANCE ACTIVITIES.

FORM 33 LER SCSS DATA 02-23-93

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
498 1991 023 0 9111270022 223571 10/20/91

ABSTRACT

POWER LEVEL - 000%. ON OCTOBER 20, 1991 UNIT 1 WAS IN MODE 4 AND UNIT 2 WAS IN NO-MODE DURING A SCHEDULED REFUELING OUTAGE, WHEN THE DETERMINATION WAS MADE THAT CRACKS FOUND ON THE RESIDUAL HEAT REMOVAL (RHR) MOTOR "T" LEADS EPOXY INTERFACE WERE REPORTABLE. ON OCTOBER 11, 1991, WHILE PERFORMING WORK ON UNIT 2 RHR MOTOR "A", LEADS FOR THE MOTOR WERE DISCOVERED TO BE DAMAGED. EXAMINATION OF THE REMAINING UNIT 2 RHR PUMP MOTORS AND UNIT 1 TRAIN A AND TRAIN C, REVEALED SIMILAR MOTOR LEAD INSULATION CRACKING ON ALL OF THESE RHR PUMPS. THE APPARENT CAUSE OF THIS CRACKING IS THAT DIFFERENCES IN THE FLEXIBILITY OF THE MOTOR LEAD BETWEEN THE ORIGINAL INSULATION AND INSULATION USING RAYCHEM SLEEVES, CONCENTRATED THE BENDING STRESS IN THE CABLE IN THE AREA ADJACENT TO THE EPOXY CAUSING THE CRACKING. THE UNIT 1 RHR PUMP MOTOR "T" LEAD INSULATION CRACKS HAVE BEEN REPAIRED, AND UNIT 2 RHR MOTOR "T" LEADS WILL BE REPAIRED DURING THE CURRENT REFUELING OUTAGE. THE REPAIRS ARE DESIGNED TO PREVENT RECURRENCE OF THE CRACKING.

FORM 34 LER SCSS DATA 02-23-93

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
499 1991 010 1 9209300002 0 12/24/91

ABSTRACT

POWER LEVEL - 016%. On December 24, 1991, at 1644 hours, Unit 2 was operating at 30% Rated Thermal Power (RTP) when pressurizer spray valve PCV-655C failed open. This ultimately caused an automatic reactor trip and Safety Injection (SI) actuation on low pressure at 1648 hours from 16% RTP. Three Reactor Coolant Pumps (RCPs) were secured to terminate the transient. All available safety equipment performed as designed and no actual injection to the reactor occurred. The cause was disengagement of the feedback arm linkage to the valve stem connecting plate on the pressurizer spray valve controller. Locking nuts were added to the spray valve feedback arm linkage connecting screws. Corrective actions included improving maintenance work instructions, conducting plant management reviews with personnel to discuss the event, and providing training on lessons learned from the event. LER92227001.U2

DRAFT

FORM 35 LER SCSS DATA 02-23-93

 DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
 498 1992 002 0 9202250231 224104 10/18/91

ABSTRACT

POWER LEVEL - 000%. ON JANUARY 24, 1992, UNIT 1 WAS IN MODE 1 AT 100% WHEN IT WAS DISCOVERED THAT CONTAINMENT INTEGRITY REQUIREMENTS WERE VIOLATED BEGINNING ON OCTOBER 18, 1991, AND LASTING APPROXIMATELY 47 HOURS. REPAIRS WERE MADE TO A LEAKING HANDHOLE COVER ON THE SECONDARY SIDE OF STEAM GENERATOR 1C, WHILE THE UNIT WAS IN MODE 4, IN VIOLATION OF THE CONTAINMENT INTEGRITY TECHNICAL SPECIFICATION. THIS EVENT WAS CAUSED BY A MISINTERPRETATION OF THE REQUIREMENTS OF THE CONTAINMENT INTEGRITY TECHNICAL SPECIFICATIONS. CORRECTIVE ACTIONS INCLUDE DISSEMINATION OF INFORMATION REGARDING THIS EVENT TO PLANT MANAGEMENT AND APPROPRIATE OPERATIONS, LICENSING, AND SCHEDULING PERSONNEL. THIS EVENT WILL ALSO BE REVIEWED WITH APPROPRIATE PLANT PERSONNEL DURING LICENSED OPERATOR REQUALIFICATION TRAINING AND THROUGH A MANAGEMENT AND TECHNICAL STAFF TRAINING BULLETIN. ADDITIONALLY, MAINTENANCE WILL ADD GUIDANCE TO APPROPRIATE PROCEDURES, THAT CONTAINMENT INTEGRITY IS REQUIRED IN MODES 1 THROUGH 4 AND THAT OPENING SECONDARY STEAM GENERATOR COVERS BREACHES CONTAINMENT INTEGRITY.

FORM 36 LER SCSS DATA 02-23-93

 DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
 499 1992 001 0 9202250110 224105 01/22/92

ABSTRACT

POWER LEVEL - 100%. ON JANUARY 22, 1992, UNIT 2 WAS IN MODE 1 AT 100% POWER. AT 0909 HOURS, UNIT 2 EXPERIENCED A REACTOR TRIP DUE TO POWER RANGE HIGH NEUTRON FLUX NEGATIVE RATE. THE PLANT WAS BROUGHT TO A STABLE CONDITION IN MODE 3 WITH NO UNEXPECTED POST-TRIP TRANSIENTS. THE CAUSE OF THE POWER RANGE HIGH NEUTRON FLUX NEGATIVE RATE TRIP WAS DROPPING OF CONTROL ROD H-6 INTO THE REACTOR CORE. THE CONTROL ROD DROPPED WHEN THE BLOCKING DIODE AND ITS ASSOCIATED STATIONARY GRIPPER COIL'S POWER CIRCUIT FAILED OPEN, RESULTING IN AN INTERRUPTION OF CURRENT TO THE STATIONARY GRIPPER COIL. THE CAUSE OF THE DIODE FAILURE REMAINS UNKNOWN. THE FAULTY DIODE WAS REPLACED, ALONG WITH ALL OTHER BLOCKING DIODES SHARING THE FAULTY DIODE'S MANUFACTURER'S DATE CODE. HL&P HAS SENT THE FAULTY DIODE AND THE OTHER SELECTED DIODES TO AN INDEPENDENT LABORATORY FOR ANALYSIS. HL&P WILL EVALUATE THE RESULTS OF THE ANALYSIS AND INITIATE FURTHER CORRECTIVE ACTIONS AS NEEDED. ADDITIONALLY HL&P, IN COOPERATION WITH WESTINGHOUSE, WILL PERFORM TESTING TO DETERMINE IF THE BLOCKING DIODES CAN BE ELIMINATED FROM THE PRESENT ROD-CONTROL SYSTEM DESIGN.

FORM 37 LER SCSS DATA 02-23-93

 DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
 499 1992 002 0 9202250104 224106 01/22/92

ABSTRACT

POWER LEVEL - 100%. ON JANUARY 22, 1992, IT WAS DETERMINED THAT STP UNIT 2 HAD BEEN OPERATED IN A CONFIGURATION WHICH RESULTED IN AN OVER TEMPERATURE DELTA TEMPERATURE (OTDT) TRIP SETPOINT WHICH WAS NOT CONSERVATIVE RELATIVE TO THE UFSAR SAFETY ANALYSIS. FOR A PERIOD OF APPROXIMATELY ONE MONTH BEGINNING ON SEPTEMBER 19, 1990, UNIT 2 WAS OPERATED WITH A FAILED THOT RESISTANCE TEMPERATURE DETECTOR (RTD) WHICH WAS BYPASSED UNTIL THE UNIT ENTERED A REFUELING OUTAGE. ALTHOUGH WITHIN THE LIMITS OF THE TECHNICAL SPECIFICATIONS, OPERATION WITH THE FAILED RTD COINCIDENT WITH THE NONCONSERVATIVE OTDT SETPOINT, WHICH SHOULD HAVE INCORPORATED VERITRAK TRANSMITTER UNCERTAINTIES, REPRESENTED A REPORTABLE CONDITION PURSUANT TO 10CFR50.73 FOR OPERATION IN AN UNANALYZED CONDITION. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR THROUGH A LACK OF ATTENTION TO DETAIL IN THE REVIEW AND RESOLUTION OF NSSS VENDOR RECOMMENDATIONS. ADMINISTRATIVE COMPENSATORY ACTIONS ALLOW STP UNITS 1 AND 2 TO CONTINUE NORMAL OPERATION WITHIN THE PRESENTLY DEFINED SAFETY LIMITS UNTIL THE PLANT SAFETY ANALYSIS IS REVISED AND ANY NECESSARY TECHNICAL SPECIFICATION CHANGES ARE APPROVED.

DRAFT

FORM 38 LER SCSS DATA 02-23-93

DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
499	1992	003	0	9203310183	224301	02/24/92

ABSTRACT

POWER LEVEL - 100%. ON FEBRUARY 24, 1992 AT 1515 HOURS, UNIT 2 WAS IN MODE 1 AT 100% POWER. FEEDWATER FLOW OSCILLATIONS WERE OBSERVED ON THE STEAM GENERATOR FEEDWATER PUMP - TURBINE DRIVEN (SGFPT) #23. AT 1703 HOURS, THE LINEAR VARIABLE DIFFERENTIAL TRANSFORMER FOR THE HIGH PRESSURE GOVERNOR VALVE FOR SGFPT #22 FAILED LOW AND THE TURBINE SUBSEQUENTLY TRIPPED ON OVERSPEED. AT 1810 HOURS, SGFPT #21 WAS OBSERVED TO HAVE DECREASING SPEED. THE SGFPT #21 WAS PLACED IN MANUAL AND GIVEN A 100% DEMAND SIGNAL BUT THE SPEED CONTINUED TO DECREASE. SUBSEQUENTLY, MANUAL TURBINE LOAD REDUCTION BEGAN AND CONTROL RODS WERE PLACED IN AUTOMATIC. AT 1811 HOURS, THE REACTOR WAS MANUALLY TRIPPED WITH STEAM GENERATOR WATER LEVELS AT 47% (NARROW RANGE) AND DECREASING. THE CAUSE OF THIS EVENT WAS RAIN WATER LEAKING THROUGH EXPANSION JOINTS IN THE TURBINE GENERATOR BUILDING (TGB) ROOF AND INTO THE ELECTROHYDRAULIC CONTROL (EHC) CABINET, WHICH IS THE COMMON CONTROL FOR ALL THREE SGFP'S. THE EH ELECTRONIC CONTROL SYSTEM WAS DRIED OUT AND SGFPT #21 AND #22 CONTROLS WERE RECALIBRATED. BELZONA FLEXIBLE MEMBRANE WAS APPLIED TO THE LEAKING EXPANSION JOINTS ON THE ROOF OF THE TGB. MODIFICATIONS WILL BE IMPLEMENTED TO SEAL THE TGB ROOFS OF BOTH UNITS.

FORM 39 LER SCSS DATA 02-23-93

DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
498	1992	001	0	9203060094	224399	01/30/92

ABSTRACT

POWER LEVEL - 100%. ON JANUARY 22, 1992, UNIT 1 WAS IN MODE 1 AT 100 PERCENT POWER. ESSENTIAL CHILLER 11C WAS INOPERABLE FOR MAINTENANCE. DUE TO AN OBSERVED LOW OIL LEVEL ON ESSENTIAL CHILLER 11B, OPERATIONS DECLARED THE CHILLER OPERABLE. THIS CONSTITUTED TWO TRAINS OF ESSENTIAL CHILLERS BEING OPERABLE AND REQUIRED ENTRY INTO TECHNICAL SPECIFICATION 3.0.3. THE PERIOD OF TIME DURING WHICH TWO TRAINS OF ESSENTIAL CHILLERS WERE OPERABLE WAS LESS THAN ONE HOUR. APPLICABLE OPERATING AND MAINTENANCE PROCEDURES ADDRESSING THE EFFECT OF OIL LEVEL ON ESSENTIAL CHILLER OPERABILITY, WILL BE REVISED.

FORM 40 LER SCSS DATA 02-23-93

DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
498	1992	003	0	9204210378	224501	03/14/92

ABSTRACT

POWER LEVEL - 100%. ON MARCH 14, 1992, UNIT 1 WAS IN MODE 1 AT 100% POWER. A REACTOR TRIP OCCURRED AT APPROXIMATELY 1108 HOURS FROM A MOMENTARY FALSE REACTOR COOLANT LOW FLOW TRIP SIGNAL. INSTRUMENTATION & CONTROL TECHNICIANS CALIBRATING THE REACTOR COOLANT FLOW TRANSMITTER REVERSED THE PROCEDURAL SEQUENCE OF RESTORING THE TRANSMITTER CAUSING A MOMENTARY LOW (BELOW SETPOINT) DIFFERENTIAL PRESSURE TO BE DETECTED BY THE TWO ADJACENT FLOW TRANSMITTERS. THIS EVENT COMPLETED THE LOGIC IN THE SOLID STATE PROTECTION SYSTEM TO TRIP THE REACTOR. THE CAUSE OF THIS EVENT WAS FAILURE TO FOLLOW PROCEDURES WHICH RESULTED FROM INSUFFICIENT SUPERVISORY AND MANAGEMENT EMPHASIS ON THE RISK ASSOCIATED WITH THE TASK, AND A LIMITED SENSE OF RESPONSIBILITY BY THE TECHNICIANS TO ENSURE PROPER TASK COMPLETION. THE ACTIONS BEING TAKEN TO CORRECT THIS EVENT ARE: SUPERVISION IS REQUIRED TO BE PRESENT TO ENSURE EMPHASIS IS PLACED ON COMPLETING THE ACTIVITY CORRECTLY WHEN A POTENTIAL REACTOR TRIP COULD OCCUR; CLEAR DIRECTION FOR USE AND PHYSICAL PRESENCE OF PROCEDURES HAS BEEN PROVIDED TO MAINTENANCE CRAFTSMEN; AND A MEMORANDUM FROM MANAGEMENT WAS ISSUED EMPHASIZING THE SELF-CHECKING PRINCIPLE.

DRAFT

FORM 41 LER SCSS DATA 02-23-93

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
499 1992 005 1 9209150399 0 05/08/92

ABSTRACT

POWER LEVEL - 100%. On May 8, 1992, Unit 2 was in Mode 1 at 100% power. At approximately 1324 hours a Containment Ventilation Isolation (CVI) actuation occurred. Operations personnel verified that all equipment actuated as designed. The radiation monitoring system did not indicate any high radiation conditions. The Containment Ventilation Isolation actuation appears to be the result of an equipment failure in a radiation monitoring RM-23A module. Troubleshooting of the suspect RM-23A module and maintenance history evaluations have been performed. LER92233001.U2

FORM 42 LER SCSS DATA 02-23-93

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
499 1992 004 0 9206020382 224995 04/28/92

ABSTRACT

POWER LEVEL - 100%. ON APRIL 28, 1992, AT 1730 HOURS, UNIT 2 WAS IN MODE 1 AT 100% POWER WHEN AN UNUSUAL EVENT WAS DECLARED. UNIT 2 COMMENCED THE PLANT SHUTDOWN DUE TO AN ENTRY INTO TECHNICAL SPECIFICATION (TS) 3.0.3. THE ENTRY INTO TS 3.0.3 WAS REQUIRED WHEN THE ACTION STATEMENT OF TS 3.6.3 COULD NOT BE MET. THE ACTION STATEMENT REQUIRES THAT AT LEAST ONE ISOLATION VALVE BE OPERABLE IN EACH AFFECTED PENETRATION THAT IS OPEN. IN THIS CASE, BOTH CONTAINMENT ISOLATION VALVES (SB-FV-4187 AND SB-FV-4187A) FOR PENETRATION M-86 WERE DECLARED INOPERABLE AFTER ATTEMPTS WERE MADE TO CLOSE EACH VALVE WITHOUT SUCCESS. THE CAUSE OF THE VALVE FAILURES HAS NOT BEEN DETERMINED. THE CORRECTIVE ACTIONS TO PREVENT RECURRENCE ARE BEING EVALUATED. A TS CHANGE IS BEING EVALUATED TO ALLOW CREDIT FOR THE STEAM GENERATOR TUBES, TUBESHEET AND SHELL AS AN ISOLATION BARRIER.

FORM 43 LER SCSS DATA 02-23-93

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
499 1992 006 0 9206240312 225109 05/22/92

ABSTRACT

POWER LEVEL - 100%. ON MAY 22, 1992, UNIT 2 WAS IN MODE 1 AT 100%, WHEN THE COMPONENT COOLING WATER (CCW) OUTLET VALVE FROM RESIDUAL HEAT REMOVAL HEAT EXCHANGER 2C OPENED FOR NO APPARENT REASON. AS A RESULT, CCW HEADER PRESSURE DECREASED AND CCW PUMP 2A AUTOMATICALLY STARTED DUE TO THE TRANSIENT. THE CAUSE OF THIS EVENT IS NOT KNOWN AT THIS TIME. PLANT OPERATORS PERFORMED A VISUAL INSPECTION OF THE VALVE AND STROKED THE VALVE WITH NO ADVERSE FINDINGS. ADDITIONALLY, THE OPERATORS SATISFACTORILY TESTED THE FUNCTION OF THE SLAVE RELAYS AND THE VALVE RESPONSE. THE MOST LIKELY CAUSE OF THIS EVENT IS A LOSS OF POWER TO THE SOLANOID VALVE SINCE NO LEAKS WERE DETECTED AND THE VALVE STROKED SATISFACTORILY. ADDITIONAL TROUBLESHOOTING WOULD NOT RESULT IN A CONCLUSIVE CAUSE OF THE EVENT, THEREFORE, NO ADDITIONAL CORRECTIVE ACTIONS ARE NECESSARY.

DRAFT

FORM 44 LER SCSS DATA 02-23-93

 DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
 498 1992 004 0 9206290075 0 05/19/92

ABSTRACT

POWER LEVEL - 100%. ON MAY 19, 1992, UNITS 1 AND 2 WERE IN MODE 1 AND AT 100% POWER. A SYSTEM ENGINEER PERFORMING A BIENNIAL REVIEW OF A SURVEILLANCE TEST PROCEDURE USED TO TEST THE MANUAL REACTOR TRIP FUNCTION, IDENTIFIED THAT THE TEST DID NOT ADEQUATELY TEST ALL CONTACTS ASSOCIATED WITH THE HANDSWITCHES USED TO INITIATE A MANUAL REACTOR TRIP VIA THE SHUNT TRIP DEVICE. THE LACK OF THIS TESTING RENDERED BOTH CHANNELS OF THE MANUAL REACTOR TRIP FUNCTION INOPERABLE. TECHNICAL SPECIFICATION 3.0.3 WAS ENTERED AND AN UNUSUAL EVENT WAS DECLARED. THE UNUSUAL EVENT WAS TERMINATED FOLLOWING VERBAL AUTHORIZATION FROM THE NRC THROUGH A TEMPORARY WAIVER OF COMPLIANCE. THE CAUSE OF THE EVENT WAS UNFAMILIARITY OF THE INDIVIDUAL RESPONSIBLE FOR DEVELOPING THE ORIGINAL PROCEDURE WITH THE REACTOR TRIP FEATURE. A CONTRIBUTING CAUSE WAS INADEQUATE REVIEW OF THE PROCEDURE DURING VARIOUS REVIEW CYCLES. CORRECTIVE ACTIONS INCLUDE: DEVELOPING A TEMPORARY PROCEDURE TO TEST THE MANUAL REACTOR TRIP VIA THE SHUNT TRIP DEVICE, REVISING THE PERMANENT PROCEDURE TO ADDRESS THE NECESSARY TESTING, AND REVIEWING SURVEILLANCE PROCEDURES TO ENSURE THAT THEY MEET TECHNICAL SPECIFICATION REQUIREMENTS.

FORM 45 LER SCSS DATA 02-23-93

 DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
 498 1992 005 0 9207150030 0 06/08/92

ABSTRACT

POWER LEVEL - 100%. On June 8, 1992 Unit 1 was in Mode 1 at 100% power, when an inadvertent start of a Component Cooling Water (CCW) pump occurred. This event occurred when the discharge header pressure went below the setpoint for starting the standby pump. The discharge header pressure decreased because of a high flow condition when one of two running pumps was manually shut down during performance of a surveillance test. The flow condition was caused by inadvertently leaving a large valve open. The cause of this event was that inadequate procedural guidance was available for performance of the test lineup. The possibility of this type of actuation was not recognized and was not incorporated in procedures. corrective actions include performing an evaluation to determine which plant procedures need to be reviewed for insufficient procedural steps to operate plant equipment, revising the appropriate procedures to incorporate appropriate guidance for proper system configurations and to support the conduct of testing, and developing a clear plant directive, for Operations personnel, emphasizing that safety equipment manipulations must be governed by written guidance and that procedural changes must be implemented before work proceeds when written guidance is lacking. LER92183001.U1

FORM 46 LER SCSS DATA 02-23-93

 DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
 498 1992 006 0 9207270063 0 03/18/92

ABSTRACT

POWER LEVEL - 033%. On March 18, 1992, with Unit 1 in Mode 1 at 33% power, the Shift Supervisor discovered that all four Auxiliary Feedwater (AFW) flow control valves were in the closed position following a reactor trip on March 14, 1992, contrary to the normal position as specified per procedures. The correct position for the AFW flow control valves is specified as open in plant procedures and plant drawings. The cause of this event was less than adequate procedures. A contributing cause was inattention to detail by the operating crews in not detecting the mispositioned valves for four days. Corrective actions included immediately opening the AFW flow control valves, revising the Reactor Trip Response procedure to require opening the AFW control valves after securing the AFW pumps, and revising the Plant Startup to 100% procedure so that verification of AFW system alignment for automatic operation prior to Mode 1 is not a conditional step. Additionally, this event will be added to the Licensed Operator Regualification Training. In addition, the Independent Safety Engineering Group (ISEG) will perform an in-depth review of causal factors for valve mispositioning. Additional corrective actions will be developed based on the results of ISEG's review. LER92195001.U1

DRAFT

FORM 47 LER SCSS DATA 02-23-93

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
498 1992 007 0 9208240342 0 07/10/92

ABSTRACT

POWER LEVEL - 095%. On July 10, 1992, at approximately 0917 hours Unit 1 was in Mode 1 at 95 percent power. An unplanned Engineered Safety Features (ESF) actuation occurred during the performance of the Spent Fuel Pool Exhaust Monitor surveillance test. Instrumentation and Control (I&C) Technicians were performing the surveillance test as required by Technical Specifications. An erroneous value was entered into the RM-23A module. With the erroneous value being present when the conversion factor was subsequently entered, the RM-23A immediately processed the data and prematurely actuated the Fuel Handling Building isolation equipment. The cause of the event is attributed to lack of attention to detail and not using effective self-verification methods. Corrective actions included restarting the surveillance test without further incident, and providing the technician involved with a written reminder under the STPEGS Constructive Discipline Program. HLEP will also perform an evaluation to determine which procedures need to be revised to ensure a dual verification is performed for those actions where incorrect data entry errors could cause ESF actuations. LER92203001.U1

FORM 48 LER SCSS DATA 02-23-93

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
498 1992 008 0 9209020007 0 07/31/92

ABSTRACT

POWER LEVEL - 100%. On July 31, 1992, Unit 1 was in Mode 1 at 100% power. At 1048 hours, a Containment Ventilation Isolation (CVI) actuation occurred. Control Room personnel verified that all equipment actuated as designed. The Containment Vent Isolation radiation monitors did not indicate any high radiation conditions. The most likely cause of this event is a momentary variance in current to the remote control unit (RM-23A) associated with radiation monitor (RI-8013B) for the Containment Purge System sufficient to cause an actuation. Troubleshooting of both radiation monitors for the Containment Purge System will be performed. Additionally, hot connections identified in the Unit 1 RM-23 module cabinets will be repaired. LER92231002.U1

FORM 49 LER SCSS DATA 02-23-93

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
498 1992 009 0 9209040194 0 08/01/92

ABSTRACT

POWER LEVEL - 100%. On August 1, 1992, Unit 1 was in Mode 1 at 100% power. Testing of the Solid State Protection System (SSPS) actuation train "C" slave relays was in progress. At approximately 2049 hours, the operator performing the Auxiliary Feedwater (AFW) portion of the test misread a procedure step which directed him to verify that the #13 AFW pump did not start following a relay actuation. Rather than verify the pump did not start, the operator turned the control switch on in an attempt to verify that the pump would not start. The #13 AFW pump started and discharged into "C" Steam Generator. The operator quickly realized the error and stopped the pump. The cause of this event was inattention to detail, in that the operator misread the test procedure. Corrective actions include revising the SSPS Actuation Train Slave Relay Test procedures to provide more distinction between steps which verify equipment startup and steps which require an attempted component startup and including this event into the Licensed Operator Requalification training. Additionally, other surveillance procedures were identified to ensure that equipment actuations are clearly defined and a plan of action was developed to enhance these procedures. LER92231001.U1

DRAFT

FORM 50 LER SCSS DATA 02-23-93

DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
498	1992	010	0	9209110174	0	07/08/92

ABSTRACT

POWER LEVEL - 100%. On August 8, 1992, Unit 1 was in Mode 1 at 100% power. Operators began a surveillance to verify acceptable Component Cooling Water (CCW) flow to the Reactor Containment Fan Coolers (RCFCs). While establishing flow via the running "B" Train CCW pump to the RCFC the standby train "A" CCW pump started due to a sensed low pressure on the miscellaneous supply header. The "A" train CCW pump started and operated properly and was shut down when it was verified not to be required. The cause of this event was lack of adequate procedural guidance. Corrective actions include revising the surveillance procedure to require the operator to place the other pump selector switches in off during the surveillance and revising the system operating procedure to include guidance for changing pump configurations. LER92239001.U1

FORM 51 LER SCSS DATA 02-23-93

DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
498	1992	011	0	9209300276	0	08/24/92

ABSTRACT

POWER LEVEL - 093%. On August 24, 1992, Units 1 and 2 were in Mode 1, with Unit 1 at 93% power and coasting down, and Unit 2 at 100% power. The Surveillance Review Task Force identified that the performance of the Reactor Coolant Pump (RCP) Undervoltage (UV) and Underfrequency (UF) Trip Actuating Device Operability Test (TADOT) surveillance procedures did not verify the bistable status monitoring (BSM) lights operability. The cause of this event is due to the writers and authorities who approve Field Changes (FCs) not identifying the need to verify the BSM lights, which were required to be tested per the Technical Specifications. This was due to inadequate understanding of the definition of TADOT by the individuals involved. This event occurred as a result of FCs in the Spring of 1990. The FCs (also a contributing factor) allowed the removal of verification of a portion of the RCP UV and UF circuitry and the BSM lights from the test procedure. This allowed the surveillance test to be incomplete and allowed entry into Mode 1, following the outage, with only a partially proven channel. Corrective actions include: verification of operability of BSM lights in both units, revision of BSM acceptance criteria of the surveillance procedure, performance of RCP TADOTS that are scheduled during outages while the plant is in Mode 5 and prior to Mode 1, a clear definition of TADOT will be formally documented and presented to appropriate personnel for training, and revision of the procedure to limit the use of FCs for changing acceptance criteria. LER9226100.U1

FORM 52 LER SCSS DATA 02-23-93

DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
498	1992	012	0	9210090281	0	09/03/92

ABSTRACT

POWER LEVEL - 086% On September 3, 1992, Unit 1 was in Mode 1 at 86% power (coastdown). Operations personnel and the system engineer noted an unusual condition on the Digital Rod Position Indication (DRPI) panel. conditions deteriorated to where it was impossible to determine control rod positions. At 1049 hours, both channels of DRPI were declared inoperable and an entry into Technical Specification 3.0.3 was made. At 1149 hours, an Unusual Event was declared due to being in a condition where a shutdown was required by the Technical Specifications. Accordingly, at 1352 hours, with DRPI still inoperable, a shutdown of the unit was commenced. At 1415 hours, I&C Technicians completed the replacement of one of the redundant power supplies and DRPI indication was recovered. The power reduction was immediately terminated and following an assessment DRPI was declared operable at 1426 hours. The cause of this event was the failure of one of the DRPI control module power supplies coupled with an apparent unknown failure of the redundant power supply. Corrective actions include replacing one of the two power supplies and returning DRPI to an operable status, replacing the remaining failed power supply during the upcoming Unit 1 outage, and developing testing for both units for the DRPI system that will include an assessment of the control system power supplies. The test will be implemented during the next Unit 2 refueling outage. LER92266001.U1

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FORM 53 LER SCSS DATA 02-23-93

 DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
 499 1992 007 1 9212210226 0 09/12/92

ABSTRACT

POWER LEVEL - 100%. On September 12 1992, Unit 2 was in Mode 1 at 100% power. Operators were performing quarterly Main Steam system valve operability testing of the solenoid operated containment isolation valve. An operator was dispatched to the Isolation Valve Cubicle (IVC) building to open the Main Steam upstream manual drain isolation valve. At 0535 hours, approximately one minute after the valve was manually opened, the above seat drain line valve on the Main Steam line "D" (MS7903A) indicated open in the Control Room. No intentional action was taken to open MS7903A. The cause of the unexpected opening of the isolation valve is "burping", an undesirable, but avoidable characteristic of piloted SOVs. The slow closure of MS7903A following the "burping" transient was apparently due to a position indication malfunction caused or influenced by unequal temperatures internal to the SOV. Corrective actions include providing training to appropriate plant departments describing the burping characteristics of piloted SOVs including suggested operational means for avoiding the problem. Additionally, a review of other systems containing piloted SOVs will be performed to determine the susceptibility of "burping." System surveillance procedures will be revised as necessary. LER92336001.U2

FORM 54 LER SCSS DATA 02-23-93

 DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
 499 1992 006 0 9210200003 0 09/15/92

ABSTRACT

POWER LEVEL - 100%. On September 15, 1992, Unit 2 was in Mode 1 at 100% power. At 0834 hours a control room toxic gas non-ESF alarm was received. Control room personnel were in the process of verifying the validity of the alarm when the control room envelope Heating Ventilation and Air conditioning system actuated to the recirculation mode on a high toxic gas ESF actuation signal. The redundant analyzer did not actuate. Testing of the analyzer indicated the cause to be a failed infrared source. The analyzer has been repaired and returned to service. The existing toxic gas analyzers are to be replaced with state-of-the-art models. These changes will be made during the current outage for Unit 1 and during the next scheduled refueling outage for Unit 2. LER92273001.U2

FORM 55 LER SCSS DATA 02-23-93

 DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
 498 1992 013 0 9210210031 0 09/15/92

ABSTRACT

POWER LEVEL - 100%. On September 15, 1992, Units 1 and 2 were in Mode 1, with Unit 1 at 79% power and coasting down, and Unit 2 at 100% power. The Surveillance Review Task Force identified that the portion of the Containment Spray (HI-3) channels between the process instrumentation and the Engineered Safety Feature (ESF) actuation and logic instrumentation was not being tested. At 0855 hours, both Units entered Technical Specification 3.0.3, however relief allowed by Technical Specification 4.0.3 was used to delay entry into the 3.0.3 action statements for 24 hours to complete the required testing. The required testing was satisfactorily completed at 1411 hours and at 1335 hours, for Unit 1 and Unit 2 respectively, and Technical Specification 3.0.3 was exited. The cause of this event was that the individual(s) developing the surveillance test procedures did not recognize the significance of the test circuit used to verify continuity of Containment Spray (HI-3) circuitry. Corrective actions included verifying the continuity of the Containment Spray (HI-3) circuitry and revising the procedures governing Containment Pressure Analog Channel Operational Test, to verify continuity of the Containment Spray (HI-3) circuits. LER92283001.U1

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FORM 56 LER SCSS DATA 02-23-93

DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
498	1992	014	0	9211030004	0	09/28/92

ABSTRACT

POWER LEVEL - 000%. On September 28, 1992, Unit 1 was in mode 6 during a refueling outage. The Containment Ventilation Isolation (CVI) Actuation and Response Time Test was in progress. The procedure used verifies the response time for equipment required to actuate on a CVI signal by simulating a high radiation signal to radiation monitors RT-8012 and RT-8013. At 1423 hours, while testing RT-8012, RT-8012 went into alarm and actuated the CVI. This occurred one step earlier than intended in the sequence of the procedure. The cause of this event was attributed to a less than adequate procedure. The CVI actuation was caused by a high radiation signal due to an artificially low high alarm setpoint established during test conditions. The value used was too low for existing radiological conditions. Corrective actions include: 1) revising the Unit 1 surveillance procedure to change the multiplication factor used when calculating the new setpoint for the response time test to increase the test value of the high alarm setpoint and 2) performing an evaluation to determine if the methodology can be improved to reduce the potential for future actuations. LER92293001.U1

FORM 57 LER SCSS DATA 02-23-93

DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
498	1992	015	0	9211120152	0	10/03/92

ABSTRACT

POWER LEVEL - 000%. On October 3, 1992, at 0433 hours, Unit 1 was in Mode 6 while in a refueling outage. The C train (1C) Component Cooling Water (CCW) pump received an automatic actuation from the miscellaneous header low pressure signal. Prior to the start, the operators had filled and vented the Engineered Safety Features (ESF) header of the 1B CCW train per the Component Cooling Water system procedure in order to restore it to an operable status. The miscellaneous header was isolated from the B train pump by closed automatic valves and the 1B pump was not yet running. The static fill and vent was completed satisfactorily and a subsequent action in the procedure was to manually start the 1B pump. When the 1B pump was started, the 1C pump started on low header pressure. The cause of this event was attributed to inattention to operating conditions exacerbated by procedural conditions which required extra attention by the operator. Corrective actions include revising the affected procedure to make the mode selector switch setting mandatory, reviewing and revising additional procedures to incorporate the mandatory mode selector switch setting, counseling the involved Operations personnel, and incorporating this event into Licensed Operator Requalification Training. LER92297001.U1

FORM 58 LER SCSS DATA 02-23-93

DOCKET	YEAR	LER NUMBER	REVISION	DCS NUMBER	NSIC	EVENT DATE
498	1992	016	0	9211100121	0	10/04/92

ABSTRACT

POWER LEVEL - 000%. On October 4, 1992, Unit 1 was in Mode 6 during a refueling outage. The Demineralized Water makeup valve to the Component Cooling Water (CCW) surge tank had been isolated the previous day, in preparation for an addition of corrosion inhibitor. At approximately 0318 hours, Unit 1 experienced an unplanned Engineered Safety Features (ESF) actuation due to an automatic pump start of CCW components caused by a low level in the CCW surge tank. The appropriate off-normal procedure was implemented and level in the surge tank was restored without further incident. All ESF equipment operated as designed. This event was the result of a failure to reopen the CCW surge tank makeup valve following a chemical addition. The immediate cause of this event is less than adequate communications. An additional cause was the lack of a procedural step to verify valve position. This event will be included in requalification training for licensed and non-licensed operators, chemical operators and chemistry technicians. The procedure associated with this surveillance will be revised to include a requirement for verification when manipulating safety related valves. Additionally, a review will be performed of procedures that contains Operations and Chemistry interfaces to ensure adequate independent verification is specified for those systems that require verification of valve positioning. LER92297002.U1

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FORM 59 LER SCSS DATA 02-23-93

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
498 1992 017 0 9212160045 0 11/11/92

ABSTRACT

POWER LEVEL - 000%. On November 1, 1992, at 1506 hours, Unit 1 was defueled during a refueling outage and Unit 2 was in Mode 1 at 100% power. The Surveillance Review Task Force identified that the Feedwater Isolation Actuation and Response Time Testing procedures did not satisfy the requirements for the time-response testing between Safety Injection and Feedwater Isolation because they did not test through the slave relays. It was later discovered that a similar condition existed between Hi-Hi Steam Generator Level and Feedwater Isolation circuitry. Unit 2 entered Technical Specification 3.0.3, however relief allowed by Technical Specification 4.0.3 was used to delay entry into the 3.0.3 action statements for 24 hours to complete the required testing. Unit 1 did not enter any Technical Specification action statements since none were applicable at that time. The required testing was completed for Unit 2 on November 12, 1992 at 0706 hours and for Unit 1 on December 5, 1992 at 2030 hours. The cause of this event was that the individuals involved in developing the original surveillance test procedures did not adequately incorporate the requirements to perform the response time testing. Corrective action included performing the required response time testing using the slave relays and revising the Feedwater Isolation Actuation and Response Time Testing Procedures to accurately test through the slave relays. LER92335001.U1

FORM 60 LER SCSS DATA 02-23-93

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
498 1992 019 0 9301050273 0 12/02/92

ABSTRACT

POWER LEVEL - 000%. On December 2, 1992, Unit 1 was in Mode 5 during a refueling outage and Unit 2 was in Mode 1 at 100% power. At 1500 hrs, while reviewing a Nuclear Network item regarding a calculation error affecting the Power Operated Relief Valve (PORV) setpoint curves for the Cold Overpressure Mitigation System (COMS), it was determined that the same condition existed at South Texas Project. The analysis performed by Westinghouse for the COMS setpoint neglected the pressure loss of the reactor coolant flow through the reactor core. This resulted in a higher pressure at the reactor core midplane elevation than the pressure at the sensing point in the RCS hot leg. Because of the error, COMS has been technically inoperable since the startup of each unit. Corrective actions for this event include issuing a Justification for Continued Operation (JCO), resetting the high PORV COMS setpoint curves to meet the JCO limit and requesting Westinghouse to revise the COMS Safety Analysis as well as providing a root cause analysis on this event to determine the generic implication and corrective actions. LER92356001.U1

FORM 61 LER SCSS DATA 02-23-93

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
498 1992 018 0 9212290250 0 10/21/92

ABSTRACT

POWER LEVEL - 000%. On October 21, 2, Unit 1 was defueled during a refueling outage. After being reset at the third refueling outage for Unit 1 (1RE03), to the specified value of 2485 psig +/- 1.0%, the setpoints for the Unit 1 Pressurizer Safety Valves were found to be 4.7% below to 3.5% above the required setpoint during the fourth refueling outage (1RE04). This is a deviation from the +/- 1.0% Technical Specification requirement. The Unit 1 Pressurizer Safety Valves (PSV-3450, 3451 & 3452) had been sent to Wyle Laboratories for setpoint verification testing. Pressurizer Safety Valve setpoint drift is an industry-wide problem which has been known for some time. The Westinghouse Owners Group (WOG) has addressed this generic problem and WCAP-12910, which makes specific recommendations relative to PSV setpoint verification testing, has been issued. Corrective actions include pursuing efforts to modify the test procedure to test the Pressurizer Safety Valve lift setpoint on saturated steam as recommended by WCAP-12910 which is pending NRC concurrence. LER92346001.U1

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FORM 62 LER SCSS DATA 02-23-93

DOCKET YEAR LER NUMBER REVISION DCS NUMBER NSIC EVENT DATE
498 1992 020 0 9301130191 0 12/08/92

ABSTRACT
POWER LEVEL - 000%. On December 9, 1992, Unit 1 was in Mode 3 at 0% power. While operators were performing control board walkdowns, it was discovered that the Toxic Gas Monitor XE-9326 channel was not in the tripped condition as required by Technical Specification 3.3.3.7. Additionally on December 12, 1992 the monitor was once again found not to be in the required tripped condition. Toxic Gas Monitor XE-9326 had been declared inoperable since November 23, 1992, due to a noisy power supply and the channel was tripped as required by Technical Specifications on November 28, 1992. The cause of this event was less than adequate design of the toxic gas monitors. There is no means to positively place the monitor in trip. A switch to ensure positive control of the trip function on the toxic gas monitors will be installed. LER93006001.U1

DIAGNOSTIC EVALUATION TEAM MEETING

Wednesday, March 10, 1993
MNBB, Room 6507

*Success - Dependent on
- [unclear] Preparation
- [unclear] Team
- [unclear] Team*

1:00 p.m. - Introduction

B. Hehl/S. Rubin

- Team Organization, Areas of Evaluation, Schedule
- Mission of the Diagnostic Evaluation (DE) Program
- DE Process and Methodology
- South Texas Project - Areas of Special Interest

1:30 p.m. - Conduct of the Diagnostic Evaluation

R. Lloyd/H. Bailey

- DE Team Licensee/Counterpart Meetings
- DE Observation Forms
- Formal Management Interviews
- Team Leader and Member Roles and Interfaces
- Onsite Interim Exit Meeting
- Team Member and Contractor Professionalism

2:00 p.m. - Break

2:20 p.m. - Conduct of the Diagnostic Evaluation (Continued)

- Plant Description and System Selected for Verticle Slice

S. Pullani

2:50 p.m. - Evaluation Plans and Report

R. Lloyd/H. Bailey

- Functional Area Evaluation Plan Preparation
- DET Report Format and Schedule

3:20 p.m. - Bagman Trip Debriefing

H. Bailey/M. Smith

- Documents on Hand
- Document Libraries
- Document Control Process
- Information Binders

3:40 p.m. - Functional Area Team Breakout Meetings

- Functional Area Evaluation Plan Review/Assignments
- Performance/Background Material Review

5:00 p.m. - Adjourn

B/9

2

DIAGNOSTIC EVALUATION TEAM MEETING

Wednesday, March 10, 1993
MNBB, Room 6507

1:30 p.m. - Conduct of the Diagnostic Evaluation - Lloyd/Bailey/Pullani

DET Characteristics - Henry

- o During the conduct of the DE (discussed earlier) the DET:
 - Identifies (or confirms) and documents both strengths and weaknesses in safety performance.
 - For weaknesses, the team evaluates and documents the adequacy of associated corrective actions.
 - Identifies the root causes of weaknesses, including any weaknesses in corrective actions.
- o I want to stress again that the evaluation is performance based and can include any safety related, important to safety and/or BOP equipment. Do not evaluate a performance issue in terms of known or potential violations of regulations. If you do, you will be likely miss the big picture on performance issues; issues that clearly impact performance, but are not normally cited as violations.
- o With regard to programmatic issues; do not pursue an issue unless you suspect it to be a root cause of a performance weakness. The STP is believed to have excellent programs on paper.

Data Collection - Henry

- o One important feature of a DE is the large amount of document review completed before the team goes onsite. This review allows you to rapidly come up to speed on the issues after you reach the site and contribute fully to the DET.
- o Weaknesses in performance are not hard to find. The licensee has many of them documented in miscellaneous deficiency tracking systems such as station problem reports (SPRs), service requests (SRs), and QA Audits, surveillances and assessment reports. We have stacks of these documents in our DET library.
- o Despite all these documented performance problems, in many cases the licensee doesn't know: 1) the full extent of the problem, 2) how to temporarily fix the problem with the available resources, and 3) what the root causes are (that would identify a permanent fix). So these deficiency tracking system documents I have mentioned are a good place to get started before we get to the site.

B/10

- o Documents that are relevant to your functional area should be read while the DET is still in Bethesda. Michelle will discuss our DET library later.
- o I have beaten on document reviews pretty good, but the other methods of data collection are also a little different for a DE. We do a large number of interviews. To really understand how an organization functions (or fails to function), interviews are essential. To understand in the shortest time what the problems are, interviews are invaluable. You will find that the workers know what the problems are if you will just ask them and it can save you from drilling a lot of dry holes. And some of them have been just hoping that you would ask them. Other people on the DET will need to know what you find out in interviews, so we ask you to document the interviews. Enough for now on interviews. Ron will be talking more on this later.
- o The last data source I will mention is observations. We do observations as the opportunity arises, but don't count on the opportunity arising as much as you may be accustomed for activities such as maintenance, testing and plant evolutions. DETs have a high profile onsite and for whatever reason, we haven't been able to observe a lot of maintenance, for example. We also believe what you do observe may be skewed quite a bit from the normal routine. We believe that if the document reviews mentioned earlier show, for example, that there have been repeat failures of a major piece of equipment, then the DET's time onsite might better be spent understanding why the repeat failures occurred than on witnessing another corrective maintenance on this equipment that might cover 3-4 days and would most likely be done strictly by the book while the DET observes (due to high DET profile and lack of time for maintenance to "lapse back" into their normal habits). Of course, the ideal situation is for the DET to do both activities.

Team Leader and Member Roles and Interfaces: Henry

- o The team leaders will schedule and assign members work onsite.
- o Team members should not represent an issue as a finding to the licensee until it has been discussed with the team leader and he has agreed.

Interviews: Ron

- o Two types - structured and unstructured.

Interview Preparation and Conduct:

1. Determine what it is you need to know, and who you should talk to gain the information before you schedule an interview.
2. Clearly write down your interview questions in a logical order. This will help you to control the interview, and at least appear coherent in front of the licensee. To open the interview, allow a

few minutes to introduce yourself and to explain what the interview is about. For structured interviews, prepare 10 to 15 questions that you would like answered. Generally, "open" type questions are preferred. Ask your questions and let the interviewee talk. Be professional, do not use "leading" or "loaded" questions, and don't make any snide remarks. Each time you finish an interview, the licensee will get together and talk about what kinds of questions were asked and the individual's responses. Mix things up.

3. For a structured interview, plan on spending at least one hour and not more than two hours to complete your interview.
4. Provide for at least a 1/2 hour block of time following your interview to expand on the notes that you took on questions asked.

Interview Closing:

1. Recap areas covered by the interview with the interviewee.
2. Ask the interviewee if he/she has any additional questions to ask.
3. Recap what the interviewee owes you in terms of documents requested, unanswered questions, etc. If a document is requested, fill out a document request form so it can be properly requested and tracked by Michelle.

Interview Documentation:

1. Immediately following your interview, go over your interview notes to fill in additional detail while your memory is still fresh. If you can type, use the word processor to save you valuable time.
 2. Transfer interview notes to a DEO form on a disk. Underline what you feel is important information, including strengths or weaknesses.
- o Structured interviews should always be summarized and documented in a Diagnostic Evaluation Observation (DEO) format. Each functional area (FA) team (except M&O) should try to conduct 3-4 structured interviews each day. The M&O team will conduct many more.
 - o Inform Michelle of interviews that you plan to do. The schedule of first week's interviews should be set prior to arrival onsite. Look at the organization chart to pick your interviewees.
 - o Each functional area (FA) team should start with the department/section head, both as a matter of courtesy and also to understand the big picture on how the organization is supposed to be operating, and where the interfaces are with other departments. Some senior managers are eager to indicate where they think the problems are and their plan for correction. They will also indicate what they believe are organizational strengths.

- o Early on, the interviews should shift more to a "bottom up" approach, i.e. nonsupervisory personnel, foremen, 1st line supervisors, etc.

DE Observation Forms (DEOs): Ron

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- o Used to record both the results of interviews and functional area team findings. You will receive a copy of a file with blank DEO forms.
 - o DEOs include statement of the issue, substantiating information, an assessment of the root cause for findings, and licensee actions being taken to address your concern.
 - o The potentially significant portions of all interviews are recorded on DEOs as soon as practicable after the interview.
 - o In addition to interviews, those findings expected to be discussed in the DET report are included on DEOs.
 - o DEOs are predecisional information. Neither DEOs or any other written [draft] information is to be given to licensee.
 - o Give a copy of your DEO file (disk) to Michelle every couple of days, so she can print and merge the files as necessary. This way, all DEOs can be read by the entire DET.

Functional Area Team Meetings With Licensee Counterparts: Henry

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- o Team leaders meet daily preferably just before DET meeting.
 - o Purpose - to keep licensee appraised of findings of fact, clear up any mistakes in the facts, coordinate future activities.
 - o Make your licensee counterparts aware of all your DEOs as they are written. This will allow the licensee the opportunity to understand the concern and rebut each DEO.
 - o Have a mini closeout meeting with your counterpart no later than the 9th of April and the 30th of April. Go through each DEO and come to an understanding of the validity of the concern.

Note: Scheduling interviews and requesting documents should be coordinated through Michelle to avoid duplication and schedular conflicts.

DET Meetings: Henry

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- o All team members are expected to attend the daily DET meetings. The team leaders will be the spokespersons at the meetings unless either a member's team leader or the DET manager asks a member to address an issue.
 - o Items discussed at these DET meetings should be limited to those of

general interest and should not include of detailed expose or your team's itinerary for the day.

Onsite Interim Exit Meeting: Henry

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- o DET Leader will present team observations for each functional area except M&O on April 30. Your specific functional area observations will be turned in to the DET Leader by noon on April 29.

Team Member and Contractor Professionalism: Henry/Bill

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- o No prospecting for future business with the licensee
 - o No exchanging of business cards with the licensee
 - o No fraternizing with the licensee
 - o No shop talk in restaurants and bars that can be heard by anyone outside the DET
 - o Any questions, discuss them with your team leader

2:00 p.m. - Break - All

2:20 p.m.- Plant Description and System Selection S. Pullani

2:50 p.m. - Functional Area Evaluation Plan Preparation: Ron

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- o Each functional area team leader (with support from their team members) is responsible for producing an evaluation plan to be reviewed by the DET Leader during the second meeting March 24-25.
 - o Keep the evaluation plan concise, not exceeding 4-6 pages single spaced.
 - o Allow for contingencies in your plan. Don't continue to beat a dead horse just because you have an assignment to look at a particular area. If you find a dry hole, move on to something that would be productive.
 - o The format for your evaluation plans should mimic the report format (see the FitzPatrick DET report). Assign responsibility for each section of your plan. This process will save time during the report writing phase of the DET.

Report Format and Schedule: Ron

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- o Report format and level of detail should resemble the FitzPatrick

report. The first sentence in each paragraph/section should be written in conclusion form. The remainder of the paragraph/section should provide the details to support the conclusion made.

- o Use the King's english in past tense.
- o Initial draft of DET report due May 12.
- o Your report section should be 99% complete by the week of May 24, since this is the week of the formal licensee exit. Each team will be required to produce final findings and conclusion slides to be used at the exit.
- o Final report due to the EDO by June 11.

3:20 p.m. - Bagman Trip Debriefing

Prescott/Smith

DIAGNOSTIC EVALUATION TEAM MEETING

Thursday, March 11, 1993
MNBB, Room 6507

- 8:00 a.m. - Introduction E. Jordan/B. Hehl
- 8:30 a.m. - Region IV Briefing
- Director Reactor Projects B. Beach
 - Deputy Director Reactor Safety A. Howell
 - Senior Resident Inspector J. Tapia
- 10:00 a.m. - Break
- 10:20 a.m. - NRR Briefing
- South Texas Project Manager G. Dick
 - LPEB Performance Evaluation P. Ray
- 11:10 a.m. - AEOD Performance Indicators D. Hickman
- Plant PIs
 - Maintenance PIs
- 11:30 a.m. - DET Administrative Requirements M. Smith
- Travel Arrangement
 - Rental Cars
 - Lodging Accommodations
 - Site Access Training
 - Working Hours/Timekeeping
- 12:00 p.m. - Lunch
- 1:00 p.m. - Breakout Meetings
- Team Manager/Team Leader Interface Meetings
 - Team Member Performance/Background Material Review
- 3:00 p.m. - STP Badging Activities A. Woods
- 5:00 p.m. - Adjourn
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Friday, March 12, 1993
MNBB, Room 6507

- 8:00 a.m. - 2:00 p.m. Continue Team Breakout Meetings
- Functional Area Evaluation Plan Review/Assignments
 - Performance/Background Material Review

B/11