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Licensee Event Report (LER) Compilation

For month of June 1991

Oak Ridge National Laboratory

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Nuclear Operations Analysis Center
Oak Ridge, TN 37831

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Abstract

This monthly report contains Licensee Event Report (LER) operational information that was processed into the LER data file of the Nuclear Safety Information Center (NSIC) during the one month period identified on the cover of the document. The LERs, from which this information is derived, are submitted to the Nuclear Regulatory Commission (NRC) by nuclear power plant licensees in accordance with federal regulations. Procedures for LER reporting for revisions to those events occurring prior to 1984 are described in NRC Regulatory Guide 1.16 and NUREG-0161, Instructions for Preparation of Data Entry Sheets for Licensee Event Reports. For those events occurring on and after January 1, 1984, LERs are being submitted in accordance with the revised rule contained in Title 10 Part 50.73 of the Code of Federal Regulations (10 CFR 50.73 - Licensee Event Report System) which was published in the Federal Register (Vol. 48, No. 144) on July 26, 1983. NUREG-1022, Licensee Event Report System - Description of Systems and Guidelines for Reporting, provides supporting guidance and information on the revised LER rule.

The LER summaries in this report are arranged alphabetically by facility name and then chronologically by event date for each facility. Component, system, keyword, and component vendor indexes follow the summaries. Vendors are those identified by the utility when the LER form is initiated; the keywords for the component, system, and general keyword indexes are assigned by the computer using correlation tables from the Sequence Coding and Search System. Questions concerning this report or its contents should be directed to

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[1] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 90-004 REV 01
UPDATE ON DEGRADED FIRE BARRIER PENETRATION AS THE RESULT OF PERSONNEL OVERSIGHT.
EVENT DATE: 053190 REPORT DATE: 051091 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: ARKANSAS NUCLEAR 2 (PWR)

(NSIC 221993) ON MAY 31, 1990 AT 1330, WHILE CONDUCTING A FIRE BARRIER PENETRATION SEAL INSPECTION AS PART OF A COMPREHENSIVE INSPECTION PROGRAM INITIATED AS PART OF A GENERIC LETTER 86-10 EVALUATION, A DEGRADED FIRE BARRIER WAS DISCOVERED BY PERSONNEL WITHIN THE FIRE PROTECTION GROUP AT ARKANSAS NUCLEAR ONE. THE DEFICIENT SEAL CONSISTED OF A 2 INCH METAL SLEEVE THROUGH A FLOOR SLAB AND A 1 1/2 INCH CONDUIT CONTAINED WITHIN THE SLEEVE. A REVIEW OF PAST DOCUMENTATION REVEALED THIS CONDITION HAS EXISTED PRIOR TO A GENERAL FIRE BARRIER INSPECTION WALK DOWN CONDUCTED IN 1983. SINCE THIS CONDITION WAS NOT IDENTIFIED DURING THIS WALK DOWN OR SUBSEQUENT TECH SPEC SURVEILLANCES, THE ROOT CAUSE OF THIS CONDITION HAS BEEN DETERMINED TO BE PERSONNEL ERROR AND OVERSIGHT REGARDING INCORRECT PROCEDURE IDENTIFICATION OF PENETRATION NUMBER 97-0038. UPON DISCOVERY OF THIS CONDITION, THE CORRESPONDING FIRE DETECTION SYSTEM WAS VERIFIED OPERABLE, A FIRE WATCH WAS POSTED IN ACCORDANCE WITH TECHNICAL SPECIFICATION REQUIREMENTS, THE FIRE BARRIER WAS SEALED, AND THE APPLICABLE FIRE PRINT AND PENETRATION LOG UPDATED . IN ADDITION, THE FIRE BARRIER INSPECTION PROCEDURE WILL BE REVISED AND A TRAINING PROGRAM WILL BE IMPLEMENTED FOR FIRE BARRIER INSPECTORS. THIS SUPPLEMENTAL REPORT EXTENDS APPLICABILITY TO UNIT 2 BY DOCUMENTING A SIMILAR EVENT DISCOVERED AS PART OF THE SAME INSPECTION PROGRAM.

[2] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 91-001 REV 01
UPDATE ON AUTOMATIC REACTOR TRIP DUE TO A MAIN TURBINE TRIP WHICH WAS CAUSED BY FAILURE OF THE TURBINE GENERATOR EXCITER.
EVENT DATE: 011091 REPORT DATE: 042991 NSSS: BW TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 221932) ON 1/10/91, AT APPROXIMATELY 2326, WITH THE PLANT AT 100 PERCENT OF RATED POWER, A REACTOR TRIP OCCURRED AS A RESULT OF THE TURBINE TRIPPING DUE TO LOSS OF FIELD EXCITATION TO THE MAIN GENERATOR. AN ANTICIPATORY REACTOR PROTECTION SYSTEM (RPS) TRIP WAS INITIATED, AS DESIGNED, WHEN THE MAIN TURBINE TRIPPED WHILE REACTOR WAS GREATER THAN 43 PERCENT. PLANT RESPONSE TO THE TRIP WAS AS EXPECTED. REACTOR COOLANT SYSTEM (RCS) PRESSURE DECREASED TO 1828 PSIG AND WAS QUICKLY RECOVERED INTO THE POST TRIP WINDOW. MINIMUM POST TRIP RCS TEMPERATURE WAS 553 DEGREES. A TEMPORARY EXCITER WAS INSTALLED WHILE THE PLANT REMAINED IN THE HOT SHUTDOWN CONDITION AND THE REACTOR WAS RETURNED TO POWER ON JANUARY 17, 1991. THE TEMPORARY EXCITER WAS REPLACED WITH A PERMANENT EXCITER DURING MID CYCLE OUTAGE 1M91, WHICH BEGAN ON APRIL 7, 1991. THE CAUSE OF THE EXCITER FAILURE WAS INADEQUATE CLEANLINESS/MATERIAL ACCOUNTABILITY CONTROLS. STRENGTHENED CONTROLS WERE IMPLEMENTED DURING THE PERMANENT EXCITER INSTALLATION.

[3] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 91-002
INADVERTENT ACTUATIONS OF THE COMBINED CONTROL ROOM EMERGENCY VENTILATION SYSTEM DUE TO AN INVALID RADIATION MONITOR TRIP WHICH WAS INITIATED BY A TRANSIENT NOISE SPIKE.
EVENT DATE: 041091 REPORT DATE: 051091 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: ARKANSAS NUCLEAR 2 (PWR)

(NSIC 221992) ON 4/10/91, AT APPROX. 0813, AN AUTOMATIC ACTUATION OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM (CREVS) OCCURRED AS A RESULT OF THE TRIPPING OF THE ANO-2 CONTROL ROOM SUPPLY RADIATION MONITOR (2RE-8750-1). THE RADIATION MONITOR WAS RESET AFTER THE ACTUATION AND ITS INDICATION RETURNED TO NORMAL (APPROX. 60 CPM). THE CREVS WAS THEN SECURED AND THE CONTROL ROOM VENTILATION SYSTEM WAS RETURNED TO ITS NORMAL CONFIGURATION. ANOTHER ACTUATION WAS INITIATED BY A NOISE SPIKE ON THE SAME MONITOR AT 2231 ON 5/9/91. CAUSE OF THESE EVENTS WAS A TRANSIENT NOISE SPIKE WHICH CAUSED THE RADIATION MONITOR INDICATION TO INCREASE TO ITS TRIP SETPOINT AND INITIATE THE CREVS ACTUATION. HOWEVER, CAUSE OF THE NOISE SPIKE COULD NOT BE DETERMINED. A MODIFICATION WILL BE COMPLETED BY 5/31/91 TO INSTALL A TIME DELAY IN THE ACTUATION CIRCUITRY OF 2RE-8750-1 TO AID IN PREVENTING INADVERTENT CREVS ACTUATIONS DUE TO SPURIOUS NOISE SIGNALS. THE ANO-2 TECH SPECS REQUIRE THAT 2RE-8750-1 BE CALIBRATED TO TRIP AT TWO TIMES THE

BACKGROUND RADIATION LEVEL. 2RE-8750-1 IS NORMALLY SET TO TRIP AT APPROX. 120 CPM. CONSIDERING THAT THIS SETPOINT IS VERY CONSERVATIVE, AN EVALUATION WAS INITIATED TO DETERMINE IF A PROPOSED TECH SPEC CHANGE SHOULD BE INITIATED TO RAISE THE SETPOINT OF 2RE-8750-1.

[4] BEAVER VALLEY 1 DOCKET 50-334 LER 91-010
INADVERTENT ESF ACTUATION OF SAFETY INJECTION VALVE DURING SURVEILLANCE TESTING.
EVENT DATE: 040591 REPORT DATE: 050291 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 221942) ON 4/5/91, OPERATORS WERE PERFORMING OST 1.1.12 (SAFEGUARDS PROTECTION SYSTEM TRAIN B TEST). AT 1923 HOURS AN UNANTICIPATED CLOSURE OF TRAIN B LOW HEAD SAFETY INJECTION PUMP MINIMUM FLOW LINE ISOLATION VALVE, MOV-SI-885B, OCCURRED. OPERATORS TERMINATED THE TEST AT 2019 HOURS AND MANUALLY OPENED MOV-SI-885B FROM THE CONTROL BOARD. INSTRUMENTATION AND CONTROL TECHNICIANS INVESTIGATED THIS EVENT AND DETERMINED THAT CONTACT PAIR 3-4 OF SOLID STATE PROTECTION SYSTEM RELAY K641B HAD FAILED, ALLOWING CURRENT TO PASS WHILE THE CONTACTS WERE SUPPOSED TO BE OPEN. FAILURE OF THIS CONTACT PAIR CAUSED VALVE MOV-SI-885B TO CLOSE. TECHNICIANS THEN REPLACED CONTACT PAIR 3-4 IN RELAY K641B. THIS FAILURE ONLY AFFECTED TRAIN B LOW HEAD SAFETY INJECTION PUMP RECIRCULATION CAPABILITY. THE PUMP WOULD HAVE BEEN FULLY CAPABLE OF PERFORMING ITS DESIGN FUNCTION IN THE EVENT OF A LARGE BREAK LOCA WHERE RECIRCULATION FLOW IS NOT REQUIRED. THE TRAIN A SYSTEM WAS OPERABLE THROUGHOUT THE EVENT. THE FAILURE OF ONE TRAIN OF SAFETY INJECTION IS BOUNDED BY ANALYSIS IN BEAVER VALLEY'S UFSAR SECTION 6.3.1.2, "ECCS SINGLE FAILURE CRITERION COMPLIANCE."

[5] BRAIDWOOD 1 DOCKET 50-456 LER 91-005
CONTROL ROOM VENTILATION TO EMERGENCY MODE ON A LOSS OF POWER TO OUTSIDE INTAKE RADIATION MONITORS.
EVENT DATE: 032691 REPORT DATE: 042691 NSSS: WE TYPE: PWR

(NSIC 221854) AT 0343 ON 3/26/91, THE MAIN CONTROL ROOM VENTILATION SYSTEM SHIFTED TO THE EMERGENCY MODE OF OPERATION. THE ACTUATION WAS CAUSED BY A MOMENTARY LOSS OF VOLTAGE TO CONTROL ROOM OUTSIDE INTAKE RADIATION MONITORS ORT-PR033 AND ORT-PR034. THE VOLTAGE DROP WAS CAUSED BY A LIGHTNING STRIKE IN THE SWITCHYARD AREA. ORT-PR033 AND ORT-PR034 AUTOMATICALLY RESTARTED AND RETURNED TO NORMAL OPERATION. THE CAUSE OF THE EVENT WAS A MOMENTARY FLUCTUATION IN VOLTAGE AVAILABLE TO THE MONITOR. WHEN LIGHTNING STRUCK THE SWITCHYARD, IT RESULTED IN A DISTURBANCE TO THE ONSITE POWER DISTRIBUTION SYSTEM. WITHIN SECONDS, THE MONITORS REGAINED THE REQUIRED VOLTAGE AND RETURNED TO THEIR PRE-EVENT CONDITION. THE MAIN CONTROL ROOM VENTILATION SYSTEM WAS RESTORED TO A NORMAL ALIGNMENT. THERE HAVE BEEN PREVIOUS OCCURRENCES OF ENGINEERED SAFETY FEATURE (ESF) ACTUATIONS CAUSED BY LIGHTNING. TO REDUCE THE NUMBER OF ESF ACTUATIONS CAUSED BY LIGHTNING, THE LIGHTNING PROTECTION SYSTEM IS PRESENTLY BEING MODIFIED. THE GROUNDING SYSTEM IS BEING IMPROVED BY THE INSTALLATION OF ADDITIONAL GROUND CONDUCTORS. A LIGHTNING DISSIPATION SYSTEM IS BEING INSTALLED TO PROTECT THE CONTAINMENT BUILDINGS AND THE AUXILIARY BUILDING VENTILATION STACKS. THESE LOCATIONS WERE CHOSEN BECAUSE THEY ARE THE TALLEST STRUCTURES AT BRAIDWOOD STATION AND THE MOST LIKELY TO BE STRUCK BY LIGHTNING.

[6] BROWNS FERRY 2 DOCKET 50-260 LER 91-004
REACTOR PROTECTION SYSTEM ACTUATION AS A RESULT OF EXCEEDING THE HI-HI SOURCE RANGE MONITOR CHANNEL SETPOINT DURING TESTING.
EVENT DATE: 032691 REPORT DATE: 042591 NSSS: GE TYPE: BWR

(NSIC 221893) ON MARCH 26, 1991, AT 2024 HOURS, UNIT 2 REACTOR PROTECTION SYSTEM ACTUATION OCCURRED WHEN THE COUNT RATE ON THE SOURCE RANGE MONITOR (SRM) CHANNEL C SPIKED WHILE A TIME DOMAIN REFLECTOMETER (TDR) TRACE WAS BEING PERFORMED ON THE UNIT 2 INTERMEDIATE RANGE MONITOR (IRM) CHANNEL C DETECTOR CABLE SHIELD. TDR TRACE WAS BEING PERFORMED ON IRM CHANNEL C TO DETERMINE THE LOCATION OF A GROUND FAULT IN THE IRM DETECTOR CABLE SHIELD. DURING THE PERFORMANCE OF THE TDR TRACE, SRM CHANNEL C SPIKED CAUSING A FULL SCRAM. THE ROOT CAUSE OF THIS EVENT HAS NOT BEEN DETERMINED AT THIS TIME. THE SCRAM WAS CAUSED BY AN UNEXPECTED HI-HI TRIP

IN THE SRC CABLE DURING THE PERFORMANCE OF TDR TRACING ON A NEARBY IRM CABLE. THIS INTERACTION BETWEEN CHANNELS HAS NOT BEEN OBSERVED BEFORE. THE GROUND FAULT IN THE IRM CHANNEL C DETECTOR CABLE SHIELD WAS REMOVED. A SPECIAL TEAM WILL PERFORM A DETAILED EVALUATION OF THE ENTIRE NEUTRON MONITORING SYSTEM PRIOR TO UNIT 2 RESTART. TVA WILL REPORT THE RESULTS IN A SUPPLEMENT TO THIS LICENSEE EVENT REPORT.

[7] BROWNS FERRY 2 DOCKET 50-260 LER 91-005
UNPLANNED ENGINEERED SAFETY FEATURES ACTUATION DUE TO CLEARING OF A FUSE CAUSED BY A FAILED RELAY COIL.
EVENT DATE: 032791 REPORT DATE: 042591 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 221894) ON MARCH 27, 1991, AT 0322 HOURS, A FUSE IN THE PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) CLEARED, RESULTING IN ENGINEERED SAFETY FEATURE ACTUATIONS. THESE ACTUATIONS INCLUDED CLOSURE OF THE DRYWELL FLOOR DRAIN ISOLATION VALVE, DRYWELL EQUIPMENT DRAIN SUMP ISOLATION VALVE, AND THE SUPPRESSION POOL DRAIN VALVE. THIS EVENT WAS CAUSED BY AN UNEXPECTED FAILURE OF A PCIS RELAY RESULTING FROM A FAULTY RELAY COIL. THE COIL IN A GENERAL ELECTRIC (GE) TYPE CR120 RELAY, USED IN THE NORMALLY ENERGIZED STATE, FAILED. CORRECTIVE ACTIONS INCLUDED REPLACING THE RELAY COIL AND REALIGNING THE AFFECTED SYSTEMS. FURTHER CORRECTIVE ACTIONS WILL INCLUDE REPLACEMENT OF THE RELAY COIL IN THE GE TYPE CR120 RELAYS USED IN NORMALLY ENERGIZED, SAFETY-RELATED APPLICATIONS IN ALL THREE UNITS.

[8] BROWNS FERRY 2 DOCKET 50-260 LER 91-006
UNPLANNED REACTOR PROTECTION SYSTEM ACTUATION RESULTING FROM LOCAL POWER RANGE MONITOR LEAKAGE CURRENT.
EVENT DATE: 032991 REPORT DATE: [illegible]2591 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO. (NUCLEAR ENG DIV)

(NSIC 221895) AT 2139 HOURS ON MARCH 29, 1991, AN UNPLANNED REACTOR PROTECTION SYSTEM ACTUATION OCCURRED ON UNIT 2 WHEN AVERAGE POWER RANGE MONITOR (APRM) B DRIFTED HIGH AND EXCEEDED THE HI-HI SETPOINT. THE INPUT TO THE APRM IS PROVIDED BY THE LOCAL POWER RANGE MONITORS (LPRMS). DURING THIS EVENT, A SINGLE LPRM DRIFTED HIGH DUE TO A HIGH LEAKAGE CURRENT AND THIS RESULTED IN THE ASSOCIATED APRM EXCEEDING ITS SETPOINT. THIS EVENT WAS TERMINATED AT 2330 HOURS ON MARCH 29, 1991 WHEN THE LPRM WAS BYPASSED AND THE REACTOR SCRAM WAS RESET. THE ROOT CAUSE OF THIS EVENT WAS AN UNEXPECTED FAILURE OF THE LPRM DUE TO A HIGH LEAKAGE CURRENT FROM EITHER THE LPRM DETECTOR OR THE LPRM CABLE. THE INITIAL CORRECTIVE ACTION WAS TO TROUBLESHOOT THE LPRM. THIS TESTING SHOWED A HIGH LEAKAGE CURRENT IN EITHER THE LPRM CABLE OR THE LPRM DETECTOR. ADDITIONAL CORRECTIVE ACTIONS INCLUDE DIAGNOSTIC TESTING OF THE NEUTRON MONITORING SYSTEM AND REPAIR OR REPLACEMENT OF ANY AFFECTED COMPONENTS AS REQUIRED. IN ADDITION, BFN WILL SUBMIT A REVISED LICENSEE EVENT REPORT TO ADDRESS THE ROOT CAUSE OF THE LPRM HIGH LEAKAGE CURRENT AND IMPLEMENTATION OF CORRECTIVE ACTIONS WITHIN 30 DAYS AFTER THE RESTART OF UNIT 2.

[9] BRUNSWICK 1 DOCKET 50-325 LER 90-028 REV 01
UPDATE ON UNEXPECTED AUTOMATIC CLOSURE OF THE HPCI EXHAUST VACUUM BREAKER ISOLATION VALVE.
EVENT DATE: 121590 REPORT DATE: 041591 NSSS: GE TYPE: BWR
VENDOR: ROSEMOUNT, INC.

(NSIC 221936) ON 12/15/90, AT APPROX. 0530, THE ALTERNATE BREAKER SUPPLYING POWER TO THE HIGH PRESSURE COOLANT INJECTION SYSTEM TURBINE EXHAUST VACUUM BREAKER, 1-E41-F079, WAS SWITCHED OFF TO RESTORE IT TO ITS REQUIRED POSITION. COINCIDENT WITH THE OPENING OF THE BREAKER THE VALVE STROKED CLOSED. THE VALVE CLOSURE RESULTED FROM A SEALED IN CLOSURE SIGNAL PRESENT IN THE VALVE LOGIC AFTER THE PERFORMANCE OF 1MST-RHR21 ON 12-1-90. THE VALVE DID NOT CLOSE AT THAT TIME BECAUSE, WITH THE ALTERNATE BREAKER SUPPLYING POWER TO THE VALVE MOTOR, NO AUTOMATIC CLOSURE SIGNAL OR MANUAL OPERATION OF THE VALVE FROM THE CONTROL ROOM OR NORMAL SUPPLY BREAKER IS AVAILABLE, PER DESIGN. THE REASON THE ASSD BREAKER

WAS "ON" IS INDETERMINATE. THE VALVE WAS RESTORED TO THE OPEN POSITION. THIS IS AN ISOLATED EVENT WITH MINIMAL SAFETY SIGNIFICANCE. THE UNIT WAS SHUT DOWN, PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) OPERABILITY WAS NOT REQUIRED, AND THE VALVE AND ELECTRICAL LINE-UPS WOULD HAVE RESTORED THE BREAKER TO ITS REQUIRED POSITION PRIOR TO THE UNIT BEING IN A CONDITION WHICH REQUIRED THE PCIS FUNCTION TO BE OPERABLE. HAD PCIS OPERABILITY BEEN REQUIRED, THE REDUNDANT ISOLATION VALVE AND SECONDARY CONTAINMENT WOULD HAVE BEEN AVAILABLE TO MITIGATE THE EVENT.

[10] BRUNSWICK 1 DOCKET 50-325 LER 91-009
DUAL UNIT REACTOR SHUTDOWN AND RESULTING LOW REACTOR VESSEL WATER LEVEL ISOLATION PRIMARY CONTAINMENT ISOLATION SYSTEM GROUPS 2, 6, & 8 AND AUTOMATIC REACTOR SCRAM SIGNAL DURING THE MANUAL SCRAM.
EVENT DATE: 032891 REPORT DATE: 042991 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BRUNSWICK 2 (BWR)

(NSIC 221937) ON MARCH 22, 1991, AT 0530 #1 EMERGENCY DIESEL GENERATOR (DG) WAS REMOVED FROM SERVICE FOR REPAIRS. BY MARCH 28, 1991, AT 1823 PREPARATIONS FOR A DUAL UNIT REACTOR SHUTDOWN WERE INITIATED BECAUSE #1 DG'S CAMSHAFT WAS DAMAGED DURING THE REPAIRS TO ITS CAMSHAFT BEARINGS. THE CAMSHAFT OF #1 DG REQUIRED REPAIRS PROJECTED TO TAKE LONGER THAN THE SEVEN DAYS ALLOWED BY THE TECHNICAL SPECIFICATION (ENDING MARCH 29, 1991, AT 0530). IT WAS PLANNED TO SHUTDOWN UNIT 1 REACTOR BY MANUAL SCRAM FROM 19% REACTOR POWER DUE TO THE INOPERABLE NEUTRON MONITORING, INTERMEDIATE RANGE MONITORS (IRM) "A" & "C", WHICH UPON ENTERING THE START-UP MODE WOULD HAVE REQUIRED A HALF SCRAM SIGNAL IN THE "A" REACTOR PROTECTION SYSTEM (RPS). DURING THE MANUAL SCRAM THE REACTOR VESSEL LEVEL MOMENTARILY SHRANK TO BELOW THE LOW LEVEL 1 SETPOINT FOR PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) GROUPS 2,6,& 8 ISOLATIONS AND THE AUTOMATIC REACTOR SCRAM SETPOINT. THE UNIT 2 REACTOR WAS ALSO SHUTDOWN BY MANUAL SCRAM, WHEN DURING THE SHUTDOWN IRM "B" & "D" FAILED REQUIRING A HALF SCRAM SIGNAL IN THE "B" REACTOR PROTECTION SYSTEM (RPS).

[11] BRUNSWICK 2 DOCKET 50-324 LER 91-002
INTERMEDIATE RANGE MONITOR SPIKES UPSCALE DURING DC GROUND ISOLATION PROCEDURES CAUSING A FULL RPS TRIP.
EVENT DATE: 040291 REPORT DATE: 050291 NSSS: GE TYPE: BWR

(NSIC 221935) AT APPROX. 0959 HOURS (EST) ON 4/2/91, AN UNEXPECTED TRIP IN RPS TRIP SYSTEM B, ACCIDENT WITH AN EXPECTED TRIP ON TRIP SYSTEM A, WAS INCURRED AND A FULL RPS TRIP WAS RECEIVED. THE EXPECTED TRIP WAS INCURRED WHEN POWER TO THE AUTOMATIC SCRAM LOGIC OF RPS TRIP SYSTEM "A" WAS DEENERGIZED DURING GROUND ISOLATION PROCEDURES. THE FULL RPS TRIP WAS RESET SHORTLY THEREAFTER. THE DIRECT CAUSE OF THE UNEXPECTED TRIP WAS INTERMEDIATE RANGE MONITORING (IRM) CHANNEL SPIKING UPSCALE. A SIGNAL NOISE, CREATED BY THE RPS TRIP SYSTEM "A" SCRAM RELAYS DEENERGIZING, WAS INDUCED INTO THE IRM D DETECTOR CIRCUITRY. IRM CHANNEL D WAS DECLARED INOPERABLE AND A WORK REQUEST/JOB ORDER (WR/JO 91-AGPA1) WAS INITIATED TO TROUBLESHOOT IRM D FOR INCOMING NOISE AND REPAIR AS NECESSARY. METAL OXIDE VARISTORS (MOVS) HAVE BEEN INSTALLED ACROSS THE COILS OF THE ASSOCIATED SCRAM RELAYS ON A TEMPORARY BASIS UNTIL PERMANENT INSTALLATION CAN BE EFFECTED. MOVS HAVE PROVEN TO BE EFFECTIVE IN SUPPRESSING HIGH VOLTAGE SPIKING. ADDITIONAL CORRECTIVE ACTION INCLUDED INITIATING AN ADVERSE CONDITION REPORT (ACR B91-177) TO ASSESS THE PERFORMANCE OF IRMS. THE SAFETY SIGNIFICANCE OF THIS EVENT WAS MINIMAL AS CONTROL RODS WERE FULLY INSERTED IN THE CORE PRIOR TO THE FULL RPS TRIP. PREVIOUS SIMILAR OCCURRENCES HAVE BEEN REPORTED IN LERS 1-84-034, 1-85-031, 1-85-045, 2-86-014, 1-87-010, 2-88-009 AND 1-89-002.

[12] CALVERT CLIFFS 2 DOCKET 50-318 LER 91-001
REACTOR COOLANT INADVERTENTLY DRAINED THROUGH CONTAINMENT SPRAY HEADER DUE TO OPERATOR ERROR.
EVENT DATE: 031291 REPORT DATE: 042691 NSSS: CE TYPE: PWR

(NSIC 221933) ON MARCH 12, 1991 AT 12:48 A.M., PLANT OPERATORS INADVERTENTLY DRAINED APPROXIMATELY 1900 GALLONS OF REACTOR COOLANT SYSTEM WATER THROUGH THE CALVERT CLIFFS UNIT 2 CONTAINMENT SPRAY RING AND A DRAIN LINE. THIS INCIDENT

OCCURRED WHILE OPERATORS WERE LINING UP TO FILL THE UNIT 2 SAFETY INJECTION TANKS. THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR IN THAT THE PLANT WATCH SUPERVISOR MISINTERPRETED PROCEDURES, INAPPROPRIATELY PERFORMED PROCEDURE STEPS CONCURRENTLY, AND FAILED TO VERIFY THE POSITION OF AN ISOLATION VALVE. VERBAL COMM[illegible]CATION BY THE OPERATORS IN THE FIELD WAS DEFICIENT. APPROPRIATE PERSONNEL ACTIO[illegible] WERE TAKEN. ALL SHIFT CREWS WERE INSTRUCTED ON MANAGEMENT EXPECTATIONS CONCE[illegible]ING CORRECT PROCEDURE INTERPRETATION, VERBAL COMMUNICATIONS, AND APPROPRIATE LINES OF AUTHORITY. WE ARE REVISING THE PROCEDURAL CONTROLS FOR PERFORMING STEPS CONCURRENTLY. WE ARE ENHANCING CONTINUING TRAINING COVERAGE OF IMPORTANT CONTROL PROCESSES IN THE OPERATIONS AREA WHERE DEVIATIONS IN THE FIELD COULD REDUCE THE DEFENSE-IN-DEPTH INTENDED BY THESE CONTROLS.

[13] CALVERT CLIFFS 2 DOCKET 50-318 LER 91-002
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS) INITIATION DUE TO OPERATOR ERROR WHILE RESTORING ESFAS LOGIC CABINET.
EVENT DATE: 032791 REPORT DATE: 042691 NSSS: CE TYPE: PWR

(NSIC 221857) ON 3/27/91, AT 11:51 A.M., AN INADVERTENT SAFETY INJECTION ACTUATION SIGNAL (SIAS) WAS INITIATED ON CALVERT CLIFFS UNIT 2. THE INCIDENT OCCURRED WHILE UTILITY LICENSED OPERATORS WERE ATTEMPTING TO RE-ENERGIZE THE ESF ACTUATION SYSTEM AL ACTUATION LOGIC CABINET. AT THE TIME OF THE EVENT UNIT 2 WAS IN COLD SHUTDOWN (MODE 5) WITH A REACTOR COOLANT SYSTEM (RCS) TEMPERATURE OF 140F AND A PRESSURE OF 220 PSI. UNIT 1 WAS OPERATING AT 100 PERCENT POWER. THE ROOT CAUSE OF THE EVENT WAS PERSONNEL ERROR BY THE SENIOR REACTOR OPERATOR (SRO) WHO WAS DIRECTING THE EVOLUTION. THE SRO FAILED TO FULLY COMPREHEND TWO CAUTION STATEMENTS CONTAINED WITHIN THE PROCEDURE HE WAS USING AND DIRECTED A CONTROL ROOM OPERATOR TO PERFORM STEPS OUT OF ORDER. THE INADVERTENT SIAS CAUSED THE CLOSURE OF THE CHEMICAL AND VOLUME CONTROL SYSTEM ISOLATION VALVES AND THE START OF A SECOND CHARGING PUMP. SHUTDOWN COOLING WAS IN-SERVICE MAINTAINING RCS TEMPERATURE AND THE HIGH PRESSURE SAFETY INJECTION SYSTEM WAS LINED UP FOR LOW TEMPERATURE OVER PRESSURE PROTECTION. BOTH OF THESE SYSTEMS WERE UNAFFECTED BY THE SIAS ACTIVATION. IMMEDIATE CORRECTIVE ACTIONS INCLUDED COUNSELING THE OPERATORS AND INFORMING OTHER PERSONNEL OF THE EVENT. CONTINUING ACTIONS FOCUS ON IMPROVING PROCEDURE USAGE, PRE-EVOLUTION BRIEFINGS, INVOLVEMENT OF SUPERVISORY PERSONNEL, AND THE CONTENT OF PROCEDURES.

[14] CALVERT CLIFFS 2 DOCKET 50-318 LER 91-003
INADVERTENT ACTUATION OF REACTOR PROTECTION SYSTEM WHILE TROUBLESHOOTING.
EVENT DATE: 040991 REPORT DATE: 050791 NSSS: CE TYPE: PWR

(NSIC 221994) ON APRIL 9, 1991 AT 0137, AN INADVERTENT REACTOR PROTECTION SYSTEM (RPS) ACTUATION OCCURRED AT CALVERT CLIFFS UNIT 2 WHILE THE PLANT WAS CONDUCTING CONTROL ELEMENT ASSEMBLY TESTING. THE PLANT WAS IN HOT SHUTDOWN, AT A REACTOR COOLANT SYSTEM PRESSURE OF 2250 PSIA AND TEMPERATURE OF 532 DEGREES FAHRENHEIT. THE INCIDENT OCCURRED WHEN PERSONNEL WHO INTENDED TO KEY A PORTABLE RADIO AND OBSERVE FOR RPS TRIP ACTUATIONS (IN THE PERFORMANCE OF TROUBLESHOOTING RADIO FREQUENCY INTERFERENCE PROBLEMS) ACCIDENTALLY KEYED THE RADIO OUT OF SEQUENCE. THE ROOT CAUSE OF THE EVENT WAS PERSONNEL ERROR COMPLICATED BY A COMBINATION OF THE PORTABLE RADIO'S DESIGN AND STANDARD CONTROLS FOR MINIMIZING CONTAMINATION. IMMEDIATE CORRECTIVE ACTIONS INCLUDED AN ENTRY ON THE SHIFT SUPERVISOR'S LOG INFORMING PERSONNEL NOT TO TAKE PORTABLE RADIOS IN THE VICINITY OF THE FLOW TRANSMITTERS AND A SIMILAR ENTRY MADE ON THE SHIFT TURNOVER SHEET. ACTIONS TO PREVENT RECURRENCE INCLUDE PERMANENT SIGNS OUTSIDE OF CONTAINMENTS ACCESSES, CHANGES TO SITE PROCEDURES, INFORMING ALL SITE PORTABLE RADIO USERS OF THE RESTRICTIONS INSIDE THE CONTAINMENTS, AND REASSESSING INDUSTRY EXPERIENCES TO ENSURE CURRENT PLANT CONTROLS ARE ADEQUATE.

[15] CATAWBA 1 DOCKET 50-413 LER 91-006
TECHNICAL SPECIFICATION VIOLATION WHEN NUCLEAR SERVICE WATER VALVES WERE LEFT WITHOUT AN EMERGENCY POWER SUPPLY DUE TO INAPPROPRIATE ACTION.
EVENT DATE: 032391 REPORT DATE: 041891 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: CATAWBA 2 (PWR)

(NSIC 221873) ON MARCH 23, 1991 AT 0300 HOURS, UNIT 1 WAS IN MODE 5, COLD SHUTDOWN, IN PREPARATION FOR THE END OF CYCLE REFUELING OUTAGE. UNIT 2 WAS IN MODE 1, POWER OPERATIONS. AT 0337 HOURS, OPERATIONS REMOVED THE 1A DIESEL GENERATOR (D/G) FROM SERVICE USING PROCEDURE OP/1/A/6350/02, DIESEL GENERATOR OPERATION. A NON-LICENSED OPERATOR (NLO) REQUESTED OPERATOR AT THE CONTROLS' (OATC) ASSISTANCE TO ENSURE THAT 1EMXG, ESSENTIAL MOTOR CONTROL CENTER, WAS ALIGNED TO 2ELXA, ALTERNATE BLACKOUT POWER LOADCENTER. THE OATC AND NLO USED THE OPERATOR AID COMPUTER (OAC) GRAPHICS AND APPARENTLY MISREAD THE GRAPHIC TO CONCLUDE THAT 1EMXG WAS POWERED BY 2ELXA. THE NUCLEAR SERVICE WATER SYSTEM (RN) 'A' TRAIN WAS REMOVED FROM SERVICE ON MARCH 23 AT 0500 HOURS AND WAS RETURNED TO SERVICE ON MARCH 26 AT 0445 HOURS. AT 1400 HOURS, AN ENGINEERED SAFEGUARD FEATURE BYPASS PANEL (1.47) ALARM WAS RECEIVED IN THE CONTROL ROOM AND THE OPERATORS DETERMINED THAT 1EMXG WAS WITHOUT EMERGENCY POWER. THE OPERATORS ALIGNED 1EMXG TO 2ELXA. THIS INCIDENT HAS BEEN A TTRIBUTED TO AN INAPPROPRIATE ACTION IN THAT THE NLO AND OATC MISREAD THE OAC GRAPHICS WHILE VERIFYING THE ALIGNMENT OF 1EMXG TO 2ELXA. OPS PERSONNEL HAVE BEEN INFORMED NOT TO USE THE OAC GRAPHICS FOR PROCEDURE SIGN-OFFS AND OPS MANAGEMENT PROCEDURES WILL BE REVISED.

[16] COMANCHE 1 DOCKET 50-445 LER 91-003 REV 01
UPDATE ON LESS THAN ADEQUATE PROCEDURE REVIEW LEADING TO THE FAILURE TO FULLY SATISFY ASME SECTION XI TESTING REQUIREMENTS.
EVENT DATE: 012491 REPORT DATE: 042991 NSSS: WE TYPE: PWR

(NSIC 221874) ON JANUARY 24, 1991, AN ENGINEER WAS PERFORMING A REVIEW OF A RESIDUAL HEAT REMOVAL SYSTEM OPERABILITY TEST PROCEDURE BEING DEVELOPED TO SUPPORT LICENSING OF UNIT 2. THE REVIEWER NOTED THAT THE ASME SECTION XI TEST REQUIREMENTS SPECIFIED IN THE INSERVICE TESTING PLAN WERE NOT SATISFIED BY THE PROCEDURE. THE REVIEWER OBSERVED A SIMILAR DISCREPANCY IN THE UNIT 1 TEST PROCEDURE. INADEQUATE IMPLEMENTATION OF THE SECTION XI TEST REQUIREMENTS CONSTITUTES A FAILURE TO MEET THE OPERABILITY REQUIREMENTS FOR A TECHNICAL SPECIFICATION LIMITING CONDITION FOR OPERATION. THE CAUSE OF THE EVENT WAS DETERMINED TO BE INADEQUATE TECHNICAL REVIEW DURING OR FOLLOWING DOCUMENT REVISION. CORRECTIVE ACTIONS INCLUDE A REQUEST FOR RELIEF FROM THE TESTING REQUIREMENT AND CONTINUED EVALUATION.

[17] COMANCHE 1 DOCKET 50-445 LER 91-006 REV 01
UPDATE ON INADVERTENT OPERATION OF THE CONTAINMENT SUMP SUCTION ISOLATION VALVES CAUSED BY PERSONNEL ERROR.
EVENT DATE: 022891 REPORT DATE: 051091 NSSS: WE TYPE: PWR

(NSIC 222012) ON FEBRUARY 28, 1991, A REACTOR OPERATOR AND AN INSTRUMENT & CONTROL TECHNICIAN WERE PERFORMING QUARTERLY SURVEILLANCE TESTING ON SAFEGUARDS SLAVE RELAYS REQUIRED FOR AUTOMATIC OPERATION OF THE CONTAINMENT SUMP SUCTION VALVES. THE MULTIMETER BEING USED TO MEASURE VOLTAGE AT POINTS IN THE CONTROL CIRCUIT WAS INCORRECTLY SET UP TO MEASURE CURRENT RATHER THAN VOLTAGE. WHEN THE FIRST READING WAS ATTEMPTED, THE CONTAINMENT SUMP SUCTION VALVE RECEIVED AN OPEN SIGNAL ALLOWING WATER TO DRAIN FROM THE REFUELING WATER STORAGE TANK TO THE CONTAINMENT SUMP. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR. CORRECTIVE ACTION INCLUDED EVENT REVIEW, TRAINING, AND PROCEDURE ENHANCEMENT. A VOLUNTARY REPORT IS BEING SUBMITTED BECAUSE OF THE SIGNIFICANCE AND GENERIC INTEREST OF THE EVENT.

[18] COMANCHE 1 DOCKET 50-445 LER 91-009
AUTOMATIC ISOLATION OF STEAM GENERATOR BLOWDOWN DUE TO COGNITIVE PERSONNEL ERROR.
EVENT DATE: 032191 REPORT DATE: 042291 NSSS: WE TYPE: PWR

(NSIC 221968) ON MARCH 21, 1991, COMANCHE PEAK STEAM ELECTRIC STATION (CPSES) UNIT 1 WAS IN MODE 5, COLD SHUTDOWN, WITH REACTOR COOLANT SYSTEM TEMPERATURE OF 130 DEGREES F AND PRESSURE OF 325 PSIG. STEAM GENERATOR BLOWDOWN (SGBD) SYSTEM WAS IN SERVICE AND BEING USED TO DRAIN THE STEAM GENERATORS (SG). AT 0300 DRAINING OF SG-03 BEGAN. THE UNIT SUPERVISOR FOLLOWED THE SYSTEM OPERATING PROCEDURE AS WRITTEN AND KNEW THAT AN ESF ACTUATION WOULD OCCUR AT 25 PERCENT LEVEL IN SG-03. HOWEVER; HE THOUGHT THIS WAS A CONTROLLED EVOLUTION, WITH PROCEDURAL GUIDANCE, AND THEREFORE NOT REPORTABLE. PRIOR TO THE ESF ACTUATION,

THE TURBINE DRIVEN AND BOTH MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS, AS WELL AS SGBD VALVES TO SG-01, 02, AND 04, WERE SECURED TO MINIMIZE THE IMPACT ON PLANT EQUIPMENT. AT 0656, A SG LO-LO LEVEL ESF ACTUATION OCCURRED, AUTOMATICALLY CLOSING THE SGBD VALVE TO SG-03. AT 0657 THE ONCOMING SHIFT, REACTOR OPERATOR, RECOGNIZED THIS AS A REPORTABLE EVENT, STARTED MDAFW PUMP-02 AND RE-FILLED SG-03 TO APPROXIMATELY 30 PERCENT, TO CLEAR THE ESF ACTUATION SIGNAL. THE ROOT CAUSE OF THIS EVENT WAS COGNITIVE PERSONNEL ERROR AND LESS THAN ADEQUATE PROCEDURAL GUIDANCE. CORRECTIVE ACTIONS INCLUDE TRAINING AND PROCEDURE REVISIONS.

[19] COMANCHE 1 DOCKET 50-445 LER 91-010
OPERATED OUTSIDE TECHNICAL SPECIFICATIONS DUE TO AUXILIARY FEEDWATER SYSTEM TEST LINE ISOLATION VALVE NOT FULLY CLOSED.
EVENT DATE: 032291 REPORT DATE: 042291 NSSS: WE TYPE: PWR

(NSIC 221875) ON MARCH 22, 1991, AN AUXILIARY OPERATOR DISCOVERED THE TRAIN B MOTOR-DRIVEN AUXILIARY FEEDWATER PUMP RECIRCULATION TEST LINE ISOLATION VALVE ONE QUARTER TURN OPEN. THE VALVE WAS LAST OPERATED ON MARCH 13, 1991, DURING AN AUXILIARY FEEDWATER OPERABILITY TEST. WITH THE VALVE ONE QUARTER TURN OPEN, AUXILIARY FEEDWATER FLOW IS 8 GALLONS PER MINUTE LESS THAN ASSUMED IN THE SAFETY ANALYSIS; THEREFORE, TRAIN B AUXILIARY FEEDWATER WAS INOPERABLE FROM MARCH 13 UNTIL THE VALVE WAS CLOSED ON MARCH 22. THIS EXCEEDED THE 72 HOUR TECHNICAL SPECIFICATION ACTION STATEMENT FOR ONE AUXILIARY FEEDWATER PUMP OR ASSOCIATED FLOW PATH INOPERABLE. THE ROOT CAUSE FOR WHY THE VALVE WAS NOT FULLY CLOSED COULD NOT BE DETERMINED. POSSIBILITIES INCLUDE THE OPERATOR FAILING TO FULLY CLOSE THE VALVE WHEN RESTORING FROM THE TEST, THE VALVE BEING DISTURBED AFTER IT WAS CLOSED ON MARCH 13, OR VIBRATION CAUSING THE VALVE TO SLIGHTLY BACK OFF ITS SEAT. CORRECTIVE ACTIONS INCLUDE PROVIDING ADDITIONAL GUIDANCE TO OPERATORS ON ENSURING THESE TYPE OF VALVES ARE CLOSED, HAVING OPERATORS MONITOR THESE TYPE OF VALVES DURING THE NEXT PUMP RUNS, AND HAVING ENGINEERING TAKE VIBRATION READINGS ON THE VALVE.

[20] COMANCHE 1 DOCKET 50-445 LER 91-011
OVERSIGHT IN PREPARATION OF A TEMPORARY MODIFICATION RESULTED IN THE FAILURE TO FULLY SATISFY A TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENT.
EVENT DATE: 032291 REPORT DATE: 042291 NSSS: WE TYPE: PWR

(NSIC 221969) ON FEBRUARY 7, 1991, A 48 INCH CONTAINMENT PURGE EXHAUST ISOLATION VALVE FAILED TO DEMONSTRATE OPERABILITY IN ACCORDANCE WITH THE TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENT FOR VERIFICATION OF LEAKAGE RATE. AS A RESULT, A TEMPORARY MODIFICATION WAS INITIATED TO INSTALL A BLIND FLANGE ON THE SYSTEM. THE FLANGE AND ASSOCIATED TEST CONNECTION PLUG ARE SUBJECT TO THE TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS FOR CONTAINMENT INTEGRITY; HOWEVER, THE SURVEILLANCE TEST PROCEDURE WAS NOT CHANGED IN TIME TO SUPPORT THE TEMPORARY MODIFICATION PRIOR TO THE FIRST PERFORMANCE OF THE TEST FOLLOWING IMPLEMENTATION OF THE MODIFICATION. AS A RESULT, THE SURVEILLANCE REQUIREMENT WAS NOT FULLY SATISFIED. THE CAUSE OF THE EVENT IS PERSONNEL ERROR LEADING TO PROTRACTED PROCESSING OF THE PROCEDURE ENHANCEMENT AND EVENT REVIEW.

[21] COMANCHE 1 DOCKET 50-445 LER 91-012
POTENTIAL GAS BINDING OF CENTRIFUGAL CHARGING PUMPS DUE TO VOIDS IN THE BORIC ACID GRAVITY FEEDLINE.
EVENT DATE: 032691 REPORT DATE: 042591 NSSS: WE TYPE: PWR

(NSIC 221876) ULTRASONIC EXAMINATION OF THE CHEMICAL AND VOLUME CONTROL SYSTEM (CVCS) SUCTION PIPING WAS PERFORMED ON MARCH 4, THROUGH MARCH 15, 1991. THESE EXAMINATIONS REVEALED VOIDS IN THE ALTERNATE BORATION LINE AND THE GRAVITY FEED LINE FROM THE BORIC ACID STORAGE TANK (BAT). ENGINEERING EVALUATION SHOWS THAT VOIDS IN THE ALTERNATE BORATION LINE WOULD NOT AFFECT OPERABILITY OF THE CENTRIFUGAL CHARGING PUMPS (CCPS). HOWEVER, ENGINEERING EVALUATION INDICATES THAT THE VOID IN THE GRAVITY FEED LINE FROM THE BAT COULD CAUSE DAMAGE TO OR GAS BINDING OF THE CCPS. THE POTENTIAL ROOT CAUSE WAS IDENTIFIED AS HYDROGEN COMING OUT OF SOLUTION, IN THE LOWER PRESSURE CCP SUCTION HEADER. CORRECTIVE ACTIONS INCLUDE DAILY VENTING OF THE GRAVITY FEED LINE AND FURTHER MONITORING FOR

HYDROGEN ACCUMULATION. BASED ON THE RESULTS OF THIS MONITORING, VENTING REQUIREMENTS WILL BE ESTABLISHED.

[22] COMANCHE 1 DOCKET 50-445 LER 91-013
LOSS OF OFFSITE POWER CAUSED BY GROUNDED TRANSMISSION LINE.
EVENT DATE: 032891 REPORT DATE: 042991 NSSS: WE TYPE: PWR

(NSIC 221877) ON MARCH 28, 1991, AT APPROXIMATELY 2253 CST, COMANCHE PEAK STEAM ELECTRIC STATION EXPERIENCED AN ENGINEERED SAFETY FEATURE ACTUATION AS A RESULT OF A LINE FAULT ON THE PREFERRED SOURCE OF POWER TO THE UNIT 1 SAFETY-RELATED EQUIPMENT. SAFETY-RELATED BUSES TRANSFERRED TO THE ALTERNATE SOURCE OF POWER, THE EMERGENCY DIESEL GENERATOR STARTED, AND ALL SYSTEMS PERFORMED AS EXPECTED. THE EVENT OCCURRED AGAIN AT 0020 HOURS ON MARCH 29. THE CAUSE OF THE TWO EVENTS WAS DETERMINED TO BE THE ACCUMULATION OF BIRD DROPPINGS ON TRANSMISSION LINE INSULATORS, PROVIDING A CURRENT PATH FROM THE LINE TO THE TOWER. CORRECTIVE ACTIONS INCLUDE INCREASED ATTENTION TO IDENTIFY THE NEED FOR CLEANING.

[23] CONNECTICUT YANKEE DOCKET 50-213 LER 91-004 REV 01
UPDATE ON PLANT SHUTDOWN DUE TO INADEQUATE CONTAINMENT AIR RECIRCULATION FAN AIR FLOW.
EVENT DATE: 030191 REPORT DATE: 042491 NSSS: WE TYPE: PWR

(NSIC 221885) ON MARCH 1, 1991, AT 1635 HOURS, WITH THE PLANT IN MODE 1 AT 100 PERCENT POWER, ALL FOUR CONTAINMENT AIR RECIRCULATION (CAR) FANS WERE DECLARED INOPERABLE FOLLOWING AN ENGINEERING REVIEW OF THE SURVEILLANCE TESTING TECHNIQUES USED TO MEASURE CAR FAN AIR FLOW. THE REVIEW INDICATED A POTENTIAL FOR THE AIR FLOW MEASUREMENTS BEING NONCONSERVATIVELY HIGH AND THE ACTUAL AIR FLOWS NONCONSERVATIVELY LOW. A PLANT SHUTDOWN WAS IMMEDIATELY INITIATED IN ACCORDANCE WITH TECHNICAL SPECIFICATION 3.0.3 AND THE PLANT ENTERED MODE 5 (COLD SHUTDOWN) ON MARCH 2, 1991, AT 1343 HOURS. SUBSEQUENT TESTING BY A DIFFERENT METHOD CONFIRMED THAT THE AIR FLOW FOR ALL FOUR CAR FANS WAS BELOW THE AMOUNT REQUIRED BY THE TECHNICAL SPECIFICATIONS. THE ROOT CAUSE OF THE EVENT WAS APPLICATION OF AN INADEQUATE TEST METHOD . CORRECTIVE ACTION CONSISTED OF ADJUSTING THE FANS' FLOW VANES AND SATISFACTORILY REPERFORMING THE TEST WITH A REVISED AND VALIDATED TEST METHOD. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(I)(A) AND 10CFR50.73(A)(2)(I)(B). THIS SUPPLEMENTAL REPORT HAS BEEN ISSUED TO PROVIDE ADDITIONAL INFORMATION ON THE SAFETY ASSESSMENT OF THE EVENT.

[24] CONNECTICUT YANKEE DOCKET 50-213 LER 91-011
CHANNEL CHECK OF SUBCOOLING MARGIN MONITOR DETERMINED INADEQUATE.
EVENT DATE: 032591 REPORT DATE: 042391 NSSS: WE TYPE: PWR

(NSIC 221886) ON MARCH 25, 1991, AT 1800 HOURS, WITH THE PLANT IN MODE 1 AT 100 PERCENT POWER, OPERATIONS DEPARTMENT PERSONNEL, DURING AN ADMINISTRATIVE REVIEW OF TECHNICAL SPECIFICATION SURVEILLANCE PERFORMANCE, DISCOVERED THAT DATA FOR THE MONTHLY CHANNEL CHECK OF THE SUBCOOLING MARGIN MONITOR WAS BEING OBTAINED FROM THE SAFETY PARAMETER DISPLAY SYSTEM (SPDS) INSTEAD OF THE INADEQUATE CORE COOLING (ICC) SYSTEM. IT WAS DETERMINED THAT THIS DID NOT CONSTITUTE A VALID CHANNEL CHECK. THE ROOT CAUSE OF THIS EVENT WAS THE FAILURE TO ADEQUATELY INCORPORATE TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS INTO APPROPRIATE IMPLEMENTING PROCEDURES. CORRECTIVE ACTION CONSISTED OF REVISING THE SURVEILLANCE PROCEDURE TO REQUIRE THAT DATA FOR THE CHANNEL CHECK BE OBTAINED FROM THE ICC CABINETS. THE MONITOR WAS SUBSEQUENTLY CHECKED WITH SATISFACTORY RESULTS. A REVIEW OF ALL OPERATIONS DEPARTMENT TECHNICAL SPECIFICATION REQUIREMENTS IS CONTINUING TO ENSURE THERE ARE NO OTHER SIMILAR PROBLEMS. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(I)(B) SINCE IT RESULTED IN A CONDITION PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATIONS.

[25] COOPER DOCKET 50-298 LER 91-002
REACTOR WATER CLEANUP ISOLATION DUE TO HIGH SYSTEM TEMPERATURE DURING PLANT COOLDOWN CAUSED BY FAILED TEMPERATURE INDICATION AND POTENTIAL EQUIPMENT FAILURE.
EVENT DATE: 032491 REPORT DATE: 042391 NSSS: GE TYPE: BWR

VENDOR: MERCOID CORP.
NUCLEAR ENGINEERING COMPANY, INC.

(NSIC 221906) ON 3/24/91, AT 2:38 AM AND, AGAIN, AT 4:15 AM, REACTOR WATER CLEANUP (RWCU) SYSTEM ISOLATIONS OCCURRED AS A RESULT OF HIGH TEMPERATURE CONDITIONS (140F) DOWNSTREAM OF THE RWCU NON-REGENERATIVE HEAT EXCHANGERS (NRHX). THE REACTOR WAS SHUTDOWN WITH THE RHR SYSTEM IN THE SHUTDOWN COOLING MODE OF OPERATION. RWCU SYSTEM WAS ALIGNED WITH A FLOW PATH OPEN TO THE CONDENSER HOTWELL TO PROVIDE FOR REACTOR VESSEL LEVEL CONTROL. TO FACILITATE FEEDWATER SYSTEM MAINTENANCE, THE PUMPS WERE NOT IN OPERATION. SINCE REACTOR VESSEL LEVEL CONTROL WAS WELL ESTABLISHED, MINIMAL FLOW EXISTED IN THE RWCU SYSTEM. IT IS PROBABLE THAT THE HIGH TEMPERATURE CONDITION OCCURRED DUE TO THE INPUT OF RESIDUAL HEAT FROM RWCU PIPING AND COMPONENTS. THIS CONDITION MAY HAVE BEEN COMPOUNDED HAD THERE BEEN ANY BACKFLOW THROUGH THE CHECK VALVE IN THE 3/4 INCH SUBCOOLING LINE INSTALLED BETWEEN THE NRHX OUTLET AND THE RWCU SYSTEM INLET PIPING. WITH VIRTUALLY NO SYSTEM DIFFERENTIAL PRESSURE, THIS BACKFLOW COULD HAVE EXISTED, ALLOWING DIRECT INPUT OF HOT WATER. NEITHER OF THESE ISOLATIONS WERE PRECEDED BY ANY INDICATION OF SYSTEM HIGH TEMPERATURE. IT WAS DETERMINED THAT THE ALARM SWITCH AND TEMPERATURE INDICATOR WERE DEFECTIVE. ROOT CAUSE OF THESE EQUIPMENT FAILURES IS LACK OF PREVENTIVE MAINTENANCE AND RANDOM EQUIPMENT FAILURE.

[26] COOPER DOCKET 50-298 LER 91-003
UNPLANNED ACTUATION OF GROUP VI ISOLATION DURING SURVEILLANCE TESTING DUE TO PERSONNEL ERROR AND DEFICIENT PROCEDURE.
EVENT DATE: 032591 REPORT DATE: 042491 NSSS: GE TYPE: BWR

(NSIC 221907) ON MARCH 25, 1991, AT 10:59 PM, WITH THE PLANT IN COLD SHUTDOWN FOR A MAINTENANCE OUTAGE, AN UNPLANNED ACTUATION OF THE GROUP VI ISOLATION LOGIC OCCURRED, RESULTING IN ISOLATION OF THE SECONDARY CONTAINMENT AND CLOSURE OF THE DRYWELL VENT AND PURGE VALVES. THE ACTUATION OCCURRED DURING SURVEILLANCE TESTING, WHEN THE I&C TECHNICIAN STATIONED IN THE CONTROL ROOM FAILED TO RESET THE REACTOR BUILDING VENTILATION RADIATION MONITOR PRIOR TO REMOVAL OF A TEST JUMPER. THE TEST IN PROGRESS WAS SURVEILLANCE PROCEDURE (SP) 6.3.7.5, REACTOR BUILDING VENTILATION RADIATION MONITOR SOURCE CHECK. AS THE TECHNICIAN BEGAN TO REMOVE THE JUMPER, ARCING WAS OBSERVED. THE JUMPER WAS IMMEDIATELY RELANDED, RESTORING THE RADIATION MONITOR LOGIC CIRCUIT TO ITS NORMALLY ENERGIZED CONDITION. THIS INTERRUPTED THE GROUP VI ACTUATION SIGNAL. THE MONITOR WAS THEN RESET AND THE VENTILATION SYSTEMS WERE RESTORED TO THEIR NORMAL LINEUP. WHILE STANDBY GAS TREATMENT SYSTEM OPERATION WAS NOT OBSERVED BY THE CONTROL ROOM OPERATOR, THIS WOULD HAVE BEEN EXPECTED DUE TO THE EFFECT OF THE MOMENTARY ACTUATION SIGNAL ON THE CONTROL SYSTEM LOGIC. THE FAILURE TO PROPERLY RESET THE RADIATION MONITOR WAS DUE TO AMBIGUITY IN THE PROCEDURE AND PERSONNEL ERROR. UPON ENCOUNTERING THE AMBIGUOUS STEP, THE TECHNICIAN WAS UNSURE THAT RESET WAS REQUIRED PRIOR TO REMOVING THE JUMPER.

[27] COOPER DOCKET 50-298 LER 91-004
UNPLANNED AUTOMATIC STARTUP OF DIESEL GENERATOR NUMBER 1 CAUSED BY INADEQUATE PLANNING AND POOR COMMUNICATIONS DURING DRAWING VERIFICATION PROJECT ACTIVITIES.
EVENT DATE: 032691 REPORT DATE: 042591 NSSS: GE TYPE: BWR

(NSIC 221855) ON 3/26/91 AT 9:26 AM, DIESEL GENERATOR NUMBER 1 (DG #1) AUTOMATICALLY STARTED, BUT WAS NOT REQUIRED TO LOAD, WHEN THE FUSE BLOCK FOR THE 4160V 1A BUS UNDERVOLTAGE PROTECTION CIRCUIT WAS PULLED DURING DRAWING VERIFICATION PROJECT ACTIVITIES. AS A RESULT OF THE CIRCUIT BEING DEENERGIZED, 4160V BREAKER 1AF OPENED, DEENERGIZING THE 4160V 1F CRITICAL BUS. THIS RESULTED IN THE RECEIPT OF AN AUTOMATIC START SIGNAL BY DG #1, CLOSURE OF SEVERAL PRIMARY CONTAINMENT ISOLATION VALVES ASSOCIATED WITH GROUPS III AND VI, AND LOAD SHEDDING OF PUMPS AND LOAD CENTERS POWERED FROM THE 4160V 1F CRITICAL BUS. THE BUS WAS AUTOMATICALLY REPOWERED FROM THE EMERGENCY TRANSFORMER, AS DESIGNED. AT THE TIME OF THE EVENT, THE PLANT WAS SHUTDOWN, WITH THE RESIDUAL HEAT REMOVAL (RHR) SYSTEM IN THE SHUTDOWN COOLING MODE OF OPERATION. REACTOR WATER TEMPERATURE WAS 137F. THE CAUSE OF THESE UNPLANNED ACTIONS WAS DUE TO PERSONNEL ERROR, COUPLED WITH INADEQUATE COMMUNICATIONS AND HUMAN FACTORS CONSIDERATIONS. THE POTENTIAL FOR, AND IMPACT OF, LOSING THE UNDERVOLTAGE PROTECTION CIRCUITRY HAD NOT BEEN PROPERLY

ASSESSED BY PROJECT PERSONNEL. THE SHIFT SUPERVISOR UNDERSTOOD THAT THE FUSE WAS ASSOCIATED WITH THE 1A CONDENSATE PUMP. THE LOCATION OF THE FUSE IN THE 1A CONDENSATE PUMP SWITCHGEAR COMPOUNDED THE COMMUNICATIONS AND ASSESSMENT DEFICIENCIES.

[28] DIABLO CANYON 1 DOCKET 50-275 LER 91-005
ACTUATION OF WRONG TEST SWITCH CAUSES UNPLANNED DIESEL GENERATOR START (ESF ACTUATION) DUE TO PERSONNEL ERROR.
EVENT DATE: 032391 REPORT DATE: 041991 NSSS: WE TYPE: PWR

(NSIC 221897) ON MARCH 23, 1991, AT 0525 PST, WITH UNIT 1 IN MODE 5 (COLD SHUTDOWN) AT 0 PERCENT POWER, AN UNPLANNED START OF DIESEL GENERATOR 1-1 AND A REALIGNMENT OF SAFETY INJECTION VALVES OCCURRED WHEN A NON-LICENSED OPERATOR INADVERTENTLY ACTUATED THE WRONG SSPS TEST SWITCH. THIS EVENT CONSTITUTES AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION. THE CONTROL ROOM OPERATORS RETURNED ALL ACTUATED EQUIPMENT TO NORMAL STATUS. ON MARCH 23, 1991, AT 0546 PST, A FOUR-HOUR, NON-EMERGENCY REPORT WAS MADE TO THE NRC IN ACCORDANCE WITH 10 CFR 50.72(B)(2)(II). THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR DUE TO FAILURE BY OPERATORS TO FOLLOW ESTABLISHED PROCEDURES AND POLICIES. THE CONCURRENT VERIFICATION PROCESS AND SELF-VERIFICATION PROCESSES REQUIRED BY PROCEDURE NPAP C-104, "INDEPENDENT VERIFICATION OF OPERATING ACTIVITIES," AND OPERATIONS POLICY B-1, "CONDUCT OF OPERATIONS," WERE IMPROPERLY IMPLEMENTED AND RESULTED IN THE INCORRECT SWITCH BEING ACTUATED. AN OPERATIONS INCIDENT SUMMARY WILL BE PREPARED, RE-EMPHASIZING THE IMPORTANCE OF THE SELF-VERIFICATION PROCESS AND REVIEWING THE REQUIREMENTS AND CONTENT OF THE CONCURRENT VERIFICATION PROCESS. THE OPERATORS INVOLVED WERE COUNSELED CONCERNING THEIR FAILURE TO IMPLEMENT THE GUIDANCE IN PROCEDURE NPAP C-104 AND OPERATIONS POLICY B-1.

[29] DIABLO CANYON 1 DOCKET 50-275 LER 91-006
ACTUATION OF CONTAINMENT VENTILATION ISOLATION DUE TO A SPURIOUS HIGH RADIATION ALARM RESULTING FROM RADIO FREQUENCY ENERGY GENERATED BY A FAULTED MOTOR.
EVENT DATE: 032691 REPORT DATE: 042591 NSSS: WE TYPE: PWR
VENDOR: CONDE MILKING MACHINE COMPANY
RELIANCE ELECTRIC COMPANY

(NSIC 221898) ON MARCH 26, 1991, AT 0015 PST, WITH UNIT 1 IN MODE 5 (COLD SHUTDOWN) AT 0 PERCENT POWER, A CONTAINMENT VENTILATION ISOLATION (CVI) ACTUATION OCCURRED. THIS EVENT CONSTITUTES AN ENGINEERED SAFETY FEATURES ACTUATION. THE ACTUATION OCCURRED DUE TO A SPURIOUS HIGH RADIATION ALARM FROM CONTAINMENT AIR PARTICULATE MONITOR RM-11. A FOUR-HOUR, NON-EMERGENCY REPORT WAS MADE TO THE NRC IN ACCORDANCE WITH 10 CFR 50.73(B)(2)(II) ON MARCH 26, 1991, AT 0216 PST. THE ROOT CAUSE FOR THE SPURIOUS HIGH RADIATION ALARM AND RESULTING CVI WAS RADIO FREQUENCY INTERFERENCE (RFI) PRODUCED WHEN THE RM-11 SAMPLE PUMP SEIZED. AFTER THE SAMPLE PUMP SEIZED, THE PUMP MOTOR FAULTED AND ARCED OVER TO A BUS GROUND. THE ARCING PRODUCED SUFFICIENT RFI TO INDUCE THE SPURIOUS HIGH RADIATION ALARM SIGNAL. IMMEDIATE CORRECTIVE ACTIONS WERE TO VERIFY THAT THE HIGH RADIATION ALARM WAS SPURIOUS AND RESET THE CVI LOGIC; THE RM-11 SAMPLE PUMP AND MOTOR WERE ALSO REPLACED.

[30] DIABLO CANYON 2 DOCKET 50-323 LER 88-027
FAILURE TO MEET TECHNICAL SPECIFICATION 3.7.9.4 DUE TO FAILURE OF TWO DAMPERS TO CLOSE AND LATE ISSUANCE OF THE REPORT DUE TO PERSONNEL ERROR.
EVENT DATE: 112788 REPORT DATE: 042491 NSSS: WE TYPE: PWR
VENDOR: RUSKIN MANUFACTURING COMPANY
S-R PRODUCTS, INC.

(NSIC 221859) ON NOVEMBER 23, 1989, DURING PERFORMANCE OF SURVEILLANCE TEST PROCEDURE M-19B, "HALON FIRE SUPPRESSION SYSTEM FUNCTIONAL TEST," TWO OF THE FOUR FIRE/SMOKE DAMPERS FOR THE SOLID STATE PROTECTION SYSTEM ROOM FAILED TO CLOSE ON MANUAL ACTUATION OF THE HALON SYSTEM. THIS FAILURE OF THE TWO DAMPERS TO CLOSE WAS DUE TO THE FUSIBLE LINKS NOT FUNCTIONING AS REQUIRED. SINCE THE TIME OF THE DAMPER FAILURE WAS UNKNOWN, IT WAS CONSERVATIVELY ASSUMED THAT TS 3.7.9.4, WHICH REQUIRES THE HALON SYSTEM TO BE OPERABLE IN MODE 4, WAS NOT MET FOR THE PERIOD OF

NOVEMBER 27, 1988 (MODE 4 ENTRY) UNTIL NOVEMBER 23, 1989. THIS CONDITION COULD HAVE RESULTED IN DILUTION OF THE HALON DISCHARGED IN THE SSPS ROOM TO BELOW THE REQUIRED CONCENTRATION. THE ROOT CAUSE OF THIS EVENT IS STILL UNDER INVESTIGATION. A SUPPLEMENTAL REPORT WILL BE SUBMIT AFTER THE ROOT CAUSE AND APPLICABLE CORRECTIVE ACTIONS ARE DETERMINED. THE L. REPORT IS DUE TO PERSONNEL FAILING TO RECOGNIZE THAT FAILURE OF THE DAMPERS TO CLOSE COULD AFFECT THE OPERABILITY OF THE HALON SYSTEM.

[31] FARLEY 2 DOCKET 50-364 LER 91-001
DROPPED CONTROL ROD CAUSES REACTOR TRIP.
EVENT DATE: 040191 REPORT DATE: 042391 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 221880) AT 1055 ON 4-1-91, WHILE OPERATING AT APPROXIMATELY 100 PERCENT POWER, A REACTOR TRIP OCCURRED WHEN ROD H-10 DROPPED INTO THE CORE. THE REACTOR TRIP OCCURRED DUE TO A HIGH NEGATIVE FLUX RATE AS DETECTED BY THE POWER RANGE NUCLEAR DETECTORS. THE OPERATOR WAS PERFORMING FNP-2-STP-5.0 (FULL LENGTH CONTROL ROD OPERABILITY TEST). WHEN CONTROL ROD GROUP C WAS TESTED, ROD H-10 DROPPED INTO THE CORE. THIS EVENT WAS CAUSED BY DEFECTIVE CIRCUIT CARD(S) IN THE ROD CONTROL SYSTEM. THE SUSPECT CARDS WERE REPLACED AND THE UNIT RETURNED TO POWER OPERATION AT 1208 ON 4-09-91.

[32] FARLEY 2 DOCKET 50-364 LER 91-002
REACTOR MANUALLY TRIPPED FOLLOWING LOSS OF STEAM GENERATOR FEED PUMP.
EVENT DATE: 040991 REPORT DATE: 050291 NSSS: WE TYPE: PWR

(NSIC 221953) AT 2040 ON 4-9-91, THE UNIT 2 REACTOR WAS MANUALLY TRIPPED FOLLOWING THE LOSS OF THE OPERATING (2A) STEAM GENERATOR FEEDWATER PUMP (SGFP). THE LOSS OF THE SGFP WAS CAUSED BY A FAILED ELECTRO-HYDRAULIC (EH) FLUID SUPPLY LINE COUPLING AND SUBSEQUENT LOSS OF EH FLUID. THE FAILURE WAS IN THE HEAT AFFECTED ZONE OF A TUBING-TO-FITTING SOCKET WELD ON THE EH SUPPLY TO THE HIGH PRESSURE GOVERNOR VALVE TO THE 2A SGFP. EXAMINATION OF THE FAILED TUBING REVEALED THAT THE FAILURE RESULTED FROM CYCLIC FATIGUE. THE EH TUBING WAS REPAIRED AND ALL ACCESSIBLE WELDS ON THE EH SYSTEMS ON UNITS 1 AND 2 SGFPS WERE INSPECTED. VIBRATION READINGS WERE TAKEN ON THE EH TUBING CONNECTIONS IN THE VICINITY OF THE UNIT 2 SGFP TURBINES. VIBRATION WAS FOUND ON THE EH TUBING IN THE VICINITY OF THE FAILURE. THE HIGHEST VIBRATION OCCURRED DURING LOW POWER PLANT OPERATION. AN INVESTIGATION IS CONTINUING TO DETERMINE IF ADDITIONAL CORRECTIVE ACTION IS NECESSARY. THE UNIT RETURNED TO POWER OPERATION AT 1648 ON 4-10-91.

[33] FERMI 2 DOCKET 50-341 LER 90-013 REV 01
UPDATE ON INADEQUATE CONTROL DURING THE PRIMARY CONTAINMENT AIR GRAB SAMPLING PROCESS.
EVENT DATE: 122890 REPORT DATE: 050391 NSSS: GE TYPE: BWR

(NSIC 221946) REVIEW DETERMINED THAT PRIMARY CONTAINMENT ATMOSPHERE SAMPLING PRACTICES COULD POTENTIALLY COMPROMISE PRIMARY CONTAINMENT INTEGRITY. THE NEED FOR ISOLATION OF THE SAMPLING LINE IN THE EVENT OF A LOSS OF COOLANT ACCIDENT WAS NOT IDENTIFIED IN THE PROCEDURE. THE ROOT CAUSE WAS A LACK OF PROPER ADMINISTRATIVE CONTROLS. THE TECHNICIAN OR THE OPERATOR COULD HAVE MANUALLY ISOLATED THE SAMPLE LINES IN 60 SECONDS WHICH IS THE TIME ALLOWED BY TECHNICAL SPECIFICATIONS. CORRECTIVE ACTION WAS TO ADD SAMPLE SUCTION AND RETURN TAPS DOWN STREAM OF AUTOMATIC CONTAINMENT ISOLATION VALVES. DURING A TEST IN MARCH OF 1991, IT WAS DETERMINED THAT PORTABLE GRAB SAMPLING HARDWARE COULD HAVE COMPROMISED THE QUALITY OF THE CONTAINMENT AIR SAMPLE BY USING AN OVERSIZED SAMPLE PUMP WHICH CAUSED BACKFLOW THROUGH THE SYSTEM PIPING AND, THEREBY, DILUTING THE SAMPLE. THE ROOT CAUSE WAS A LACK OF PROPER ADMINISTRATIVE CONTROLS OVER SAMPLE HARDWARE AND ITS APPLICATION. SINCE THE CONTAINMENT ATMOSPHERE WAS MONITORED DURING RELEASES, THERE HAS BEEN NO UNMONITORED RELEASE TO THE ENVIRONMENT. IN NOVEMBER 1990, ON THE SUSPICION THAT DILUTED SAMPLES COULD BE OBTAINED, THE PORTABLE SAMPLE RIG'S VACUUM PUMP WAS DOWNSIZED. THE SAMPLE LOCATION CHANGE ASSURES THAT THE SAMPLE OBTAINED IS VALID .

[34] FERMI 2 DOCKET 50-341 LER 91-005
EXCEEDED TECH SPEC ALLOWABLE LIMITS FOR LOCAL LEAK RATE TESTING.
EVENT DATE: 033191 REPORT DATE: 043091 NSSS: GE TYPE: BWR

(NSIC 221947) PERIODIC LEAKAGE RATE TESTING OF PRIMARY CONTAINMENT ISOLATION VALVES AND PENETRATIONS IS BEING PERFORMED IN ACCORDANCE WITH THE REQUIREMENTS OF TECHNICAL SPECIFICATION 3.6.1.2 AND 10 CFR 50, APPENDIX J. DURING THE PERFORMANCE OF THIS TESTING, SEVERAL VALVES HAVE EXCEEDED THEIR ADMINISTRATIVE ALLOWABLE LEAKAGE RATE AND THEIR COMBINED LEAKAGE EXCEEDS THE LIMITS AS DEFINED IN THE SUBJECT TECHNICAL SPECIFICATION LIMITING CONDITION FOR OPERATION. AS NECESSARY, CONTAINMENT ISOLATION VALVES THAT HAVE EXCEEDED THEIR INDIVIDUAL ADMINISTRATIVE ALLOWABLE LEAKAGE RATE HAVE HAD WORK REQUESTS GENERATED TO REPAIR OR REWORK THEM AS CONDITIONS DICTATE. A FULL REPORT WILL BE PROVIDED IN A SUPPLEMENT TO THIS LICENSEE EVENT REPORT TO BE SUBMITTED 30 DAYS AFTER COMPLETION OF THE TYPE B AND C TESTING/RETESTING.

[35] FITZPATRICK DOCKET 50-333 LER 91-004
RADIOACTIVE LIQUID EFFLUENT RELEASE IN EXCESS OF TWO TIMES MAXIMUM PERMISSIBLE CONCENTRATION DUE TO PERSONNEL ERROR, DEFICIENCIES IN PROCEDURES, AND MANAGEMENT OF ENGINEERING.
EVENT DATE: 031891 REPORT DATE: 041791 NSSS: GE TYPE: BWR

(NSIC 221941) AN UNUSUAL EVENT WAS DECLARED AT 1445 ON 3/18/91 AFTER THE UNMONITORED RELEASE OF RADIOACTIVITY WAS DETECTED OUTSIDE THE AUXILIARY BOILER BUILDING. CONTAMINATION WAS DETECTED ON THE GROUND, BUILDING WALLS, AND ROOFS. DUE TO RAIN SHOWER ACTIVITY, RAIN WATER FROM CONTAMINATED BUILDING ROOFS DRAINED TO THE STORM DRAINAGE SYSTEM, RESULTING IN AN UNMONITORED RELEASE INTO LAKE ONTARIO. THIS EVENT RESULTED FROM ACTIVITIES ASSOCIATED WITH THE LIQUID RADIOACTIVE WASTE CONCENTRATOR (WD). THE CONCENTRATOR WASTE LINE WAS PLUGGED AND OPERATORS WERE PERFORMING TROUBLESHOOTING ACTIVITIES TO CLEAR THE LINE. STEAM FROM THE AUXILIARY BOILER (SA) PROVIDES PROCESS STEAM TO THE WASTE CONCENTRATOR AND THE CONDENSATE IS RETURNED TO THE BOILER. DUE TO INAPPROPRIATE ACTIONS BY THE OPERATORS, THE CONDENSATE RETURN BECAME CONTAMINATED WITH RADIOACTIVE CONCENTRATED WASTE. THE RADIOACTIVE WASTE ESCAPED WITH THE VENTING STEAM VIA THE ATMOSPHERIC VENT ON THE AUXILIARY BOILER ROOF. AT APPROXIMATELY 1315 THE VENT WAS CLOSED, WHICH TERMINATED THE SOURCE OF UNMONITORED RELEASE. THE STORM DRAINAGE SYSTEM WAS BLOCKED AT 1939, TERMINATING THE UNMONITORED RELEASE TO LAKE ONTARIO. ACTIVITIES DURING AND AFTER THE UNUSUAL EVENT INVOLVED PREVENTING CONTINUED RELEASE TO THE ENVIRONMENT, CLEAN-UP, DECONTAMINATION, AND ROOT CAUSE OF THE EVENT.

[36] FT. CALHOUN 1 DOCKET 50-285 LER 91-007
480V CIRCUIT BREAKER COORDINATION OUTSIDE DESIGN BASIS.
EVENT DATE: 032091 REPORT DATE: 041991 NSSS: CE TYPE: PWR

(NSIC 221903) DURING RECONSTITUTION OF THE DESIGN BASIS FOR THE ELECTRICAL DISTRIBUTION OVERCURRENT TRIPPING SCHEME, THE PORTION OF A BREAKER/FUSE COORDINATION STUDY DEALING WITH THE 161 KV SYSTEM DOWN TO THE 480V MOTOR CONTROL CENTERS (MCCS) WAS COMPLETED. TWENTY-ONE 480 VOLT MOLDED CASE CIRCUIT BREAKERS WERE DETERMINED TO HAVE OVERLAPPING BREAKER COORDINATION CURVES WITH THE MCC FEEDER CIRCUIT BREAKERS. THE LACK OF COORDINATION COULD RESULT IN THE TRIPPING OF A MCC DUE TO A FAULT ON ONE OF ITS NON-COORDINATED LOADS. ON MARCH 20, 1991, THIS CONDITION WAS DETERMINED TO BE OUTSIDE THE DESIGN BASIS OF THE PLANT. TWO OF THE TWENTY-ONE LOADS WERE DETERMINED TO HAVE AN UNACCEPTABLE PROBABILITY OF FAULTING DURING A DESIGN BASIS ACCIDENT. THIS CONDITION WAS CAUSED BY DEFICIENCIES IN THE ORIGINAL SYSTEM DESIGN AS CONSTRUCTED BY THE PLANT ARCHITECT/ENGINEER. A CONTRIBUTING CAUSE WAS THE LACK OF COMPREHENSIVE DESIGN BASIS DOCUMENTATION TO SUBSTANTIATE THAT COORDINATED BREAKER FAULT PROTECTION EXISTED. THE TWO AFFECTED LOADS WERE ISOLATED FROM THE 480V SYSTEM. PRESENT DESIGN PROCEDURES REQUIRE BREAKER COORDINATION. COMPLETION OF THE BREAKER/FUSE COORDINATION STUDY WILL DETERMINE IF ANY MORE COORDINATION PROBLEMS EXIST WITH THE ELECTRICAL DISTRIBUTION SYSTEM.

[37] GINNA DOCKET 50-244 LER 91-003
FIREWATCH NOT POSTED DURING FIRE SYSTEM ISOLATION DUE TO INADEQUATE ADHERENCE TO PROCEDURES CAUSES A CONDITION PROHIBITED BY PLANT TECHNICAL SPECIFICATION.
EVENT DATE: 031491 REPORT DATE: 041591 NSSS: WE TYPE: PWR

(NSIC 221912) ON MARCH 14, 1991 AT 1045 EST, WITH THE REACTOR AT APPROXIMATELY 89% FULL POWER, FIRE SPRINKLER SYSTEM S-04 WAS RENDERED INOPERABLE WITHOUT A FIREWATCH WITH BACKUP FIRE SUPPRESSION EQUIPMENT BEING ESTABLISHED. THIS PLACED THE PLANT IN A CONDITION PROHIBITED BY PLANT TECHNICAL SPECIFICATIONS. THIS EVENT WAS NOT DISCOVERED UNTIL APPROXIMATELY 1327 EST, MARCH 14, 1991. IMMEDIATE CORRECTIVE ACTION WAS TO ESTABLISH A FIREWATCH WITH BACKUP SUPPRESSION EQUIPMENT FOR FIRE SPRINKLER SYSTEM S-04. SUBSEQUENTLY AT 1510 EST THE SYSTEM WAS AGAIN DECLARED OPERABLE. THE UNDERLYING CAUSE OF THE EVENT HAS BEEN ATTRIBUTED TO INADEQUATE ADHERENCE TO PROCEDURES. CORRECTIVE ACTIONS WILL BE TO ESTABLISH A FORMALIZED METHOD TO VERIFY A FIREWATCH IS POSTED PRIOR TO HOLDING THE FIRE SYSTEM. OTHER CORRECTIVE ACTION PLANNED IS OUTLINED IN SECTION V B.

[38] GINNA DOCKET 50-244 LER 91-004
MANUAL START AND LOADING OF "B" EMERGENCY DIESEL GENERATOR DUE TO POTENTIAL SEVERE WEATHER CONDITIONS WHILE OPERATING ON REDUCED INVENTORY IN THE REACTOR COOLANT SYSTEM.
EVENT DATE: 032891 REPORT DATE: 042691 NSSS: WE TYPE: PWR

(NSIC 221889) ON MARCH 28, 1991 AT 0211 EST, WITH THE REACTOR IN THE COLD SHUTDOWN CONDITION AND REACTOR COOLANT SYSTEM TEMPERATURE AND PRESSURE AT 100F AND ZERO (0) PSIG RESPECTIVELY, THE "B" EMERGENCY DIESEL GENERATOR WAS STARTED DUE TO THE POTENTIAL FOR SEVERE WEATHER IN THE AREA. AFTER THE "B" EMERGENCY DIESEL GENERATOR WAS STARTED THE CONTROL ROOM OPERATORS TIED IT INTO SAFEGUARDS BUS #17. SUBSEQUENTLY, AFTER THE THREAT OF THE SEVERE WEATHER HAD PASSED, THE "B" EMERGENCY DIESEL GENERATOR WAS REMOVED FROM SERVICE AND REALIGNED FOR AUTO STANDBY. THE UNDERLYING CAUSE OF THE EVENT WAS JUSTIFIABLE CONSERVATISM ON THE PART OF THE PLANT STAFF DUE TO THE SENSITIVITY TO LOSS OF RHR WHILE OPERATING IN THE RCS REDUCED INVENTORY CONDITION. SINCE THE POTENTIAL SEVERE WEATHER CONDITIONS WERE A NATURAL PHENOMENON, NO CORRECTIVE ACTIONS TO PREVENT RECURRENCE ARE WARRANTED.

[39] GRAND GULF 1 DOCKET 50-416 LER 90-023
SHUTDOWN COOLING ISOLATION DUE TO A BLOWN FUSE.
EVENT DATE: 110590 REPORT DATE: 120590 NSSS: GE TYPE: BWR
VENDOR: BUSSMANN MFG (DIV OF MCGRAW-EDISON)

(NSIC 221504) ON NOVEMBER 5, 1990, DURING RESTORATION OF THE DIVISION II (16AB) BUS AN ISOLATION OF SHUTDOWN COOLING OCCURRED. DURING THE BUS OUTAGE TEMPORARY POWER WAS SUPPLIED TO VARIOUS LOADS, AND JUMPERS WERE INSTALLED TO PREVENT ADVERSE IMPACTS ON OUTAGE OPERATIONS. UPON REMOVAL OF THE JUMPER, WHICH WAS INSTALLED TO PREVENT THE SHUTDOWN COOLING ISOLATION, A FUSE IN THE ISOLATION CIRCUITRY BLEW RESULTING IN A SHUTDOWN COOLING ISOLATION. THE FUSE WAS REPLACED AND SHUTDOWN COOLING WAS RESTORED AFTER APPROXIMATELY 25 MINUTES. THE CAUSE OF THE BLOWN FUSE, AS DETERMINED BY PLANT PERSONNEL, WAS A SHORT WHICH OCCURRED DURING THE REMOVAL OF THE JUMPER. THE SYSTEM OPERATING INSTRUCTIONS WILL BE CHANGED PRIOR TO THE NEXT PLANNED BUS OUTAGE (15AA AND 16AB). ADDITIONALLY, SURVEILLANCES WHICH TEST THE SHUTDOWN COOLING ISOLATION LOGIC RESPONSE TIMES WILL BE CHALLENGED PRIOR TO PERFORMANCE. THESE PROCEDURES WILL REQUIRE THE BREAKERS TO THE ISOLATION VALVES TO BE OPEN PRIOR TO JUMPER INSTALLATION AND REMOVAL. THE REACTOR COOLANT TEMPERATURE INCREASED FROM APPROXIMATELY 91 TO 94 DEGREES F. THE LACK OF SHUTDOWN COOLING FOR APPROXIMATELY 25 MINUTES CAUSED NO ADVERSE SAFETY CONSEQUENCES.

[40] HATCH 1 DOCKET 50-321 LER 91-008
INADEQUATE PROCEDURE RESULTS IN MISSED TECHNICAL SPECIFICATIONS SURVEILLANCE.
EVENT DATE: 032191 REPORT DATE: 041991 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: HATCH 2 (BWR)

(NSIC 221934) ON 3/21/91 AT APPROXIMATELY 1300 CST, UNIT 1 WAS IN THE RUN MODE AT A POWER LEVEL OF 2436 CMWT (100% RATED THERMAL POWER) AND UNIT 2 WAS IN THE COLD SHUTDOWN MODE. AT THAT TIME, PLANT PERSONNEL PERFORMING A REVIEW OF CHEMISTRY DEPARTMENT TECHNICAL SPECIFICATIONS SURVEILLANCES DETERMINED THAT SOME DAILY CHANNEL CHECKS HAD NOT BEEN PERFORMED AT THE FREQUENCY REQUIRED BY THE UNIT 1 AND UNIT 2 TECHNICAL SPECIFICATIONS. SPECIFICALLY, THE DAILY CHANNEL CHECKS OF THE UNIT 1 RECOMBINER BUILDING VENTILATION NOBLE GAS ACTIVITY MONITOR AND SAMPLE FLOWRATE MEASURING DEVICE AND THE MAIN STACK SAMPLE FLOWRATE MEASURING DEVICE WERE PERFORMED ON 1/26/91 AT A TIME OF DAY WHICH EXCEEDED THE REQUIRED FREQUENCY OF PERFORMANCE. THIS WAS CONTRARY TO THE REQUIREMENTS OF UNIT 1 TECHNICAL SPECIFICATIONS TABLE 4.14.2-1 AND UNIT 2 TECHNICAL SPECIFICATIONS TABLE 4.3.6.10-1. THE CAUSE OF THIS EVENT IS A LESS THAN ADEQUATE PROCEDURE. PROCEDURE 62EV-SAM-003-0S, "GASEOUS WASTE DISCHARGE MONITOR CHECKS," DID NOT CONTAIN ADEQUATE GUIDANCE TO ENSURE THE DAILY CHECKS WERE PERFORMED DURING THE SAME TIME PERIOD EACH DAY SUCH THAT THE REQUIRED FREQUENCY PLUS GRACE PERIOD WAS NOT EXCEEDED. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDE ISSUING DAILY CHANNEL CHECK GUIDANCE TO CHEMISTRY PERSONNEL AND REVISING PROCEDURES 62EV-SAM-003-0S AND 64CH-ADM-001-0S, "CHEMISTRY PROGRAM."

[41] HATCH 2 DOCKET 50-366 LER 91-007
UNKNOWN INADEQUACY IN JUMPER CONNECTION RESULTS IN SCRAM SIGNAL DURING SURVEILLANCE IN COLD SHUTDOWN.
EVENT DATE: 032691 REPORT DATE: 042591 NSSS: GE TYPE: BWR

(NSIC 221864) ON 3/26/91, AT APPROX. 1155 CST, UNIT 2 WAS IN COLD SHUTDOWN WITH REACTOR COOLANT TEMPERATURE AT APPROXIMATELY 100 DEGREES FAHRENHEIT, REACTOR PRESSURE AT ATMOSPHERIC, AND ALL CONTROL RODS FULLY INSERTED. AT THAT TIME, LICENSED PLANT OPERATORS OBSERVED INDICATIONS THAT A FULL SCRAM SIGNAL HAD BEEN RECEIVED. IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS REQUIREMENTS PRIOR TO ENTERING REFUEL MODE, SURVEILLANCE PROCEDURE 34SV-C51-001-2S WAS BEING PERFORMED TO TEST THE NEUTRON MONITORING SYSTEM (NMS, EIIS CODE IG) NONCOINCIDENT SCRAM LOGIC. THIS REQUIRED THE PLACEMENT OF ELECTRICAL JUMPERS IN THE REACTOR PROTECTION SYSTEM (RPS, EIIS CODE JE) LOGIC TO ENSURE A FULL SCRAM WAS NOT GENERATED WHILE TESTING NMS TRIP FUNCTIONS. IN THIS EVENT, IT APPEARS THAT AT LEAST ONE JUMPER FAILED TO MAINTAIN CONTACT RESULTING IN A FULL SCRAM SIGNAL WHEN NMS SIGNALS WERE INDUCED PER THE SURVEILLANCE PROCEDURE. THE CAUSE OF THIS EVENT COULD NOT BE CONCLUSIVELY DETERMINED. HOWEVER, IT APPEARS TO BE AN ISOLATED EVENT RESULTING FROM AN UNKNOWN INADEQUACY IN A JUMPER CONNECTION. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDE COMPLETING THE FUNCTIONAL TEST WHICH VERIFIED OPERABILITY OF THE NMS NONCOINCIDENT SCRAM LOGIC AND REVISING THE SURVEILLANCE PROCEDURE TO REDUCE THE PROBABILITY OF RECURRENCE.

[42] HATCH 2 DOCKET 50-366 LER 91-009
SAFETY RELIEF VALVES EXPERIENCE SETPOINT DRIFT DUE TO CORROSION INDUCED BONDING.
EVENT DATE: 041091 REPORT DATE: 050891 NSSS: GE TYPE: BWR
VENDOR: TARGET ROCK CORP.

(NSIC 222007) ON 4/10/91, AT APPROXIMATELY 0800 CDT, UNIT 2 WAS IN THE REFUEL MODE WITH ALL FUEL HAVING BEEN REMOVED FROM THE VESSEL. AT THAT TIME PLANT ENGINEERING PERSONNEL RECEIVED WRITTEN NOTIFICATION OF THE RESULTS OF OFF-SITE TESTING OF PRESSURE VESSEL SAFETY RELIEF VALVES (SRVS, EIIS CODE RV). THREE OF THE ELEVEN SRVS EXHIBITED DRIFT IN THE MECHANICAL LIFT SETPOINTS IN EXCESS OF THE +/- 3% TOLERANCE SPECIFIED BY IN-SERVICE TESTING (IST) REQUIREMENTS. FOUR SRVS EXPERIENCED SETPOINT DRIFT GREATER THAN THE +/- 1% UNIT 2 TECHNICAL SPECIFICATIONS TOLERANCE REQUIREMENT IN SECTION 3.4.2.1. THIS VOLUNTARY REPORT IS BEING SUBMITTED DUE TO THE POTENTIAL INDUSTRY INTEREST IN THIS EVENT IN VIEW OF THE ONGOING EFFORTS OF THE BOILING WATER REACTOR OWNERS' GROUP (BWROG) TO REDUCE SETPOINT DRIFT. THE SETPOINT DRIFT EXPERIENCED WAS WELL WITHIN THE ANALYTICAL LIMITS EXISTING FOR REACTOR VESSEL OVER-PRESSURE PROTECTION. THE ROOT CAUSE OF THE SRV DRIFT IN EXCESS OF +/- 3% IS CORROSION-INDUCED BONDING OF THE SURFACE BETWEEN THE PILOT VALVE DISC AND SEAT. ONE ADDITIONAL SRV EXPERIENCED NEGATIVE DRIFT IN EXCESS OF -1%. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDE REFURRBISHING THE VALVES AND CONTINUING TO PARTICIPATE IN THE EFFORTS TO RESOLVE THE SRV

SETPOINT DRIFT ISSUE. THE BWROG PROGRAM DIRECTION HAS BEEN CONCURRED WITH BY THE NRC.

[43] HOPE CREEK 1 DOCKET 50-354 LER 90-028 REV 01
UPDATE ON TURBINE TRIP ON MOISTURE SEPARATOR HIGH LEVEL RESULTS IN REACTOR SCRAM DUE TO MOISTURE SEPARATOR LEVEL CONTROL SYSTEM MALFUNCTION.
EVENT DATE: 111790 REPORT DATE: 043091 NSSS: GE TYPE: BWR
VENDOR: VELAN VALVE CORP.

(NSIC 221950) ON 11/17/90 AT 0352, DURING PERFORMANCE OF A SURVEILLANCE PROCEDURE WHICH TESTS THE MAIN TURBINE COMBINED INTERMEDIATE VALVES (CIV), THE "A" MOISTURE SEPARATOR EXPERIENCED A HIGH LEVEL CONDITION. IN RESPONSE TO THIS HIGH LEVEL CONDITION, THE ASSOCIATED DUMP VALVE BEGAN TO OPEN, BUT LEVEL CONTINUED TO RISE, AND THE MAIN TURBINE TRIPPED ON MOISTURE SEPARATOR HIGH LEVEL. FOLLOWING THE TURBINE TRIP, REACTOR SCRAMMED ON A TURBINE CONTROL VALVE CLOSURE SIGNAL FROM THE REACTOR PROTECTION SYSTEM. ALL CONTROLS RODS WERE VERIFIED TO BE INSERTED, AND PLANT SYSTEMS RESPONDED AS EXPECTED. INVESTIGATION SUBSEQUENT TO THE SCRAM DETERMINED THAT INITIATING CAUSE OF THIS EVENT TO BE A MALFUNCTION OF THE LEVEL CONTROL SYSTEM FOR THE "A" MOISTURE SEPARATOR. DURING THE STATIONS THIRD REFUELING OUTAGE, INSPECTION AND TESTING OF "A" MOISTURE SEPARATOR VALVES, CONTROLS, AND INTERNAL PIPING WAS CONDUCTED. RESULTS OF THESE TESTS AND INSPECTIONS CONCLUDED THAT A PRIMARY CONTRIBUTOR TO THIS SCRAM WAS A BROKEN BUSHING ON THE HINGE PIN FOR THE "A" MOISTURE SEPARATOR NORMAL DRAIN LINE CHECK VALVE. THE AS-FOUND CONDITION OF THE CHECK VALVE WAS EVALUATED AS HAVING CAUSED THE CHECK VALVE TO STICK OPEN. WITH THE CHECK VALVE STUCK OPEN, A BACKFLOW OF CONDENSATE FROM THE #5 FEEDWATER HEATER WOULD OCCUR AS PRESSURE IN THE MOISTURE SEPARATOR DECREASED.

[44] HOPE CREEK 1 DOCKET 50-354 LER 91-006
REACTOR WATER CLEANUP ISOLATION WHEN VENTING FLOW TRANSMITTERS DUE TO A DEFICIENT SYSTEM OPERATING PROCEDURE.
EVENT DATE: 040591 REPORT DATE: 050191 NSSS: GE TYPE: BWR

(NSIC 221952) ON APRIL 5, 1991 AT 1113, CONTROL ROOM PERSONNEL RECEIVED INDICATION OF A REACTOR WATER CLEANUP (RWCU) SYSTEM ISOLATION DURING THE COURSE OF VENTING SYSTEM DIFFERENTIAL FLOW TRANSMITTERS. AFTER ASCERTAINING THE INITIATING CAUSE OF THE ISOLATION, THE NUCLEAR SHIFT SUPERVISOR (NSS, SRO LICENSED) DIRECTED THAT THE SYSTEM BE RESTORED TO A NORMAL ALIGNMENT. SUBSEQUENT INVESTIGATION DETERMINED THAT THE PRIMARY CAUSE OF THIS INCIDENT WAS A PROCEDURAL DEFICIENCY, IN THAT THE RWCU PROCEDURE DOES NOT TAKE INTO ACCOUNT THE NEED FOR DEFEATING THE SYSTEM ISOLATION LOGIC WHEN VENTING THE DIFFERENTIAL FLOW TRANSMITTERS. CORRECTIVE ACTIONS CONSIST OF IMPLEMENTING A PROCEDURE CHANGE TO REQUIRE DEFEATING THE RWCU SYSTEM ISOLATION LOGIC WHEN VENTING SYSTEM DIFFERENTIAL FLOW TRANSMITTERS, AND REVIEWING THIS EVENT DURING LICENSED OPERATOR REQUALIFICATION TRAINING.

[45] INDIAN POINT 2 DOCKET 50-247 LER 91-006
LOSS OF 138 KV OFFSITE POWER.
EVENT DATE: 032091 REPORT DATE: 041991 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: INDIAN POINT 3 (PWR)
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 221890) ON MARCH 20, 1991, DURING A COLD SHUTDOWN OUTAGE FOR REFUELING, THE 138 KV NORMAL OFFSITE POWER SUPPLY WAS LOST. THIS RESULTED IN A LOSS OF POWER TO ALL STATION 6.9 KV AND 480V BUSES. DUE TO MAINTENANCE BEING PERFORMED ON TWO EMERGENCY DIESELS, ONLY THE THIRD DIESEL AUTOMATICALLY STARTED. THIS DIESEL WAS MANUALLY CONNECTED TO 480V BUS 6A, AND POWER WAS RESTORED TO TWO OTHER 480V BUSES THROUGH THE TIE BREAKERS WHICH ALLOWED FOR RESTORATION OF SERVICE WATER AND COMPONENT COOLING PUMPS. DUE TO MAINTENANCE BEING PERFORMED ON ONE OF THE BATTERIES, THERE WAS A LOSS OF ONE DC BUS WHEN ITS BATTERY CHARGER LOST POWER. THIS BUS WAS MANUALLY RE-ENERGIZED BY MEANS OF A CONNECTION TO ANOTHER DC BUS THROUGH A TIE BREAKER. AT THE TIME OF THE EVENT THERE WAS NO FUEL IN THE CORE AND THE SPENT FUEL COOLING FLOW WAS NEVER LOST BECAUSE THE SPENT FUEL PUMP WAS

BEING FED FROM AN ALTERNATE SOURCE. ANOTHER 138KV FEEDER AND 13.8KV OFFSITE POWER REMAINED AVAILABLE DURING THIS EVENT. 138KV OFFSITE POWER WAS RESTORED WITHIN HALF AN HOUR, AND ALL 480V BUSES WERE RETURNED TO NORMAL SUPPLIES BY 2100 HOURS.

[46] INDIAN POINT 2 DOCKET 50-247 LER 91-007
AUTO-START OF EMERGENCY DIESEL.
EVENT DATE: 032891 REPORT DATE: 042991 NSSS: WE TYPE: PWR

(NSIC 221911) ON MARCH 28, 1991, DURING A COLD SHUTDOWN REFUELING OUTAGE, WITH FUEL OUT OF THE CORE, EMERGENCY DIESEL NO. 23 AUTOMATICALLY STARTED. THE DIESEL HAD RECEIVED AN UNDERVOLTAGE SIGNAL FROM ONE OF THE 480V BUSES WHICH INITIATED THE START AND STRIPPED THE LOADS ON THE BUS. THERE WAS NO APPARENT ACTUAL UNDERVOLTAGE CONDITION ON THE 480V OR 6.9 KV BUSES. ALL OFFSITE POWER REMAINED AVAILABLE, AND ALL ENGINEERED SAFETY FEATURES PERFORMED AS EXPECTED. A PERTURBATION ON THE DC CIRCUIT ASSOCIATED WITH THE UNDERVOLTAGE RELAYS OR AN INADVERTENT BUMPING OF A RELAY ARE THE MOST PROBABLE CAUSES FOR INITIATING THE UNDERVOLTAGE SIGNAL.

[47] INDIAN POINT 3 DOCKET 50-286 LER 91-006
VOLUNTARY LER DETAILING PROBLEMS IDENTIFIED DURING GENERIC LETTER 89-10 REVIEW.
EVENT DATE: 112090 REPORT DATE: 042691 NSSS: WE TYPE: PWR

(NSIC 221927) THIS LER IS A VOLUNTARY SUBMITTAL DETAILING THE NEW YORK POWER AUTHORITY'S RESPONSE TO GENERIC LETTER 89-10. ON NOVEMBER 20, 1990, WITH THE REACTOR IN COLD SHUTDOWN, THE PLANT ENGINEERS CONDUCTED A DESIGN BASIS REVIEW ON VARIOUS SAFETY-RELATED VALVES. THE ENGINEERS DETERMINED THAT ELEVEN VALVES DID NOT MEET NEW CRITERIA ESTABLISHED IN GENERIC LETTER 89-10. PROBLEMS IDENTIFIED INCLUDED INADEQUATE SPRING PACKS, GEAR RATIOS, TORQUE SETTINGS, AND MOTORS . ALL ELEVEN VALVES WERE MODIFIED TO MEET GENERIC LETTER 89-10 CRITERIA.

[48] INDIAN POINT 3 DOCKET 50-286 LER 91-004
PLANT TRIP RESULTING FROM ELECTRICAL FAULT ON 345 KV SYSTEM.
EVENT DATE: 032091 REPORT DATE: 041591 NSSS: WE TYPE: PWR

(NSIC 221925) ON MARCH 20, 1991 WITH THE REACTOR AT 100 PERCENT POWER, A UNIT TRIP WAS INITIATED AS THE RESULT OF AN ELECTRICAL FAULT ON A 345 KV BUS SECTION, REMOTE TO THE SITE. THE FAULT ISOLATION CIRCUITRY OPENED BOTH GENERATOR OUTPUT BREAKERS AND GENERATED A PLANT TRIP VIA THE GENERATOR LOCKOUT RELAYS. THE COMPONENTS AFFECTED WERE SUBSEQUENTLY REPAIRED AND TESTED. THE PLANT RETURNED TO SERVICE ON MARCH 22, 1991.

[49] INDIAN POINT 3 DOCKET 50-286 LER 91-005
UNIT TRIP CAUSED BY FAULTY CHECK VALVE IN MAIN FEED PUMP LINE.
EVENT DATE: 032291 REPORT DATE: 041991 NSSS: WE TYPE: PWR
VENDOR: CRANE VALVE CO.

(NSIC 221926) ON MARCH 22, 1991, WITH THE REACTOR AT 25 PERCENT POWER, A UNIT TRIP WAS INITIATED AS THE RESULT OF A STEAM GENERATOR LOW-LOW LEVEL TRIP. ALL PLANT SYSTEMS FUNCTIONED PROPERLY FOLLOWING THE TRIP. THE CAUSE OF THIS EVENT WAS DETERMINED TO BE CYCLIC FATIGUE FAILURE OF THE LOCKING PIN ON THE 31 MAIN BOILER FEED PUMP CHECK VALVE. THE DISCHARGE CHECK VALVES ON BOTH MAIN FEED PUMPS WERE OVERHAULED AND RETESTED.

[50] KEWAUNEE DOCKET 50-305 LER 91-001 REV 01
UPDATE ON SECURING OF PRESSURE RELIEF PATH IN THE TURBINE DRIVEN AUXILIARY FEEDWATER PUMP ROOM RESULTS IN POTENTIAL INABILITY TO MEET HIGH ENERGY LINE BREAK CRITERIA.
EVENT DATE: 020191 REPORT DATE: 041891 NSSS: WE TYPE: PWR

(NSIC 221931) AT 1505 CST ON 2/1/91 WITH PLANT AT 100% POWER, A CONCERN WAS

IDENTIFIED REGARDING THE ABILITY TO MEET DESIGN BASIS REQUIREMENTS FOR STEAM EXCLUSION IN THE EVENT OF A HIGH ENERGY LINE BREAK (HELB) AT KNPP. IT WAS DISCOVERED THAT A BLOW-OUT PANEL PREVIOUSLY INSTALLED IN A PENETRATION OF THE WALL IN THE TURBINE DRIVEN AUXILIARY FEEDWATER PUMP (TDAFWP) ROOM DID NOT APPEAR TO BE IN A CONDITION THAT WOULD MEET ITS ORIGINAL DESIGN INTENT. THIS OPENING WAS DESIGNED TO PROVIDE A RELIEF PATH TO LIMIT THE PRESSURE INCREASE IN THE TDAFWP ROOM FOLLOWING A POSTULATED STEAM RELEASE FROM THE STEAM SUPPLY PIPING TO THE TDAFWP. IF PRESSURE RELIEF WERE NOT PROVIDED, OVERPRESSURIZATION OF THE TDAFWP ROOM COULD OCCUR. THIS OVERPRESSURIZATION COULD RESULT IN A RELEASE OF STEAM TO ADJACENT ROOMS WHICH CONTAIN SAFEY-RELATED ELECTRICAL EQUIPMENT. THE AS-FOUND CONDITION OF THE TDAFWP ROOM OPENING INDICATED THAT A MODIFICATION HAD BEEN MADE TO THE OPENING WHICH DEFEATED THE RELIEF CAPABILITY. BASED ON THE AS-FOUND CONDITION AND THE INFORMATION AVAILABLE AT THE TIME, IT WAS CONCLUDED THAT THE BLOW-OUT PANEL COULD NOT PERFORM ITS DESIGN FUNCTION AND THE PLANT WAS IN A CONDITION OUTSIDE OF ITS DESIGN BASIS. CAUSE OF THE EVENT WAS A FAILURE TO PERFORM AN ADEQUATE SAFETY EVALUATION TO SUPPORT A MODIFICATION PERFORMED IN 1978.

[51] LA SALLE 1 DOCKET 50-373 LER 91-003
LOSS OF ENGINEERED SAFETY FEATURE BUS 143 DURING SURVEILLANCE TESTING CAUSED BY DEFICIENT PROCEDURE & PERSONNEL ERROR.
EVENT DATE: 031991 REPORT DATE: 041891 NSSS: GE TYPE: BWR

(NSIC 221954) ON MARCH 19, 1991 AT 1323 HOURS, WITH UNIT 1 DEFUELED, OPERATIONAL ANALYSIS DEPARTMENT (OAD) PERSONNEL WERE PERFORMING 1B DIESEL GENERATOR (DG) PROTECTIVE RELAY CALIBRATIONS IN ACCORDANCE WITH AN APPROVED STATION PROCEDURE WHEN THE NORMAL FEEDBREAKER TO ENGINEERED SAFETY FEATURE (ESF) BUS 143 TRIPPED. THE 1B DG, WHICH PROVIDES EMERGENCY POWER TO ESF BUS 143, WAS OUT OF SERVICE AT THE TIME OF THIS EVENT AND DID NOT START; CONSEQUENTLY ESF BUS 143 REMAINED DEENERGIZED. REVIEW OF THE BREAKER TRIPPING SCHEME INDICATED THAT THE RELAY THAT OAD PERSONNEL HAD JUST TESTED (1B DG OVERCURRENT WITH VOLTAGE RESTRAINT RELAY K35A) WOULD TRIP THE BREAKER. THE CAUSES OF THIS EVENT WERE AN INADEQUATE PROCEDURE, AND PERSONNEL ERROR IN IDENTIFYING ALL TRIPS TO BE DEFEATED. ALTHOUGH THE PROCEDURE USED IDENTIFIES THE NEED FOR DISABLING TRIPS, THE PROCEDURE DOES NOT GIVE ANY SPECIFIC DIRECTION ON WHICH TRIPS ARE REQUIRED TO BE DISABLED. THE SAFETY CONSEQUENCES OF THIS EVENT WERE MINIMAL. UNIT 1 WAS IN COLD SHUTDOWN AND DEFUELED DURING THIS EVENT. THE HIGH PRESSURE CORE SPRAY (HPCS) SYSTEM WAS ALREADY INOPERABLE DUE TO SCHEDULED MAINTENANCE AT THE TIME OF THIS EVENT. NO EMERGENCY CORE COOLING SYSTEMS (ECCS) ARE REQUIRED OPERABLE WITH THE UNIT DEFUELED. ONCE THE CAUSE FOR THE TRIP WAS DETERMINED, ESF BUS 143 WAS REENERGIZED AT 1352 HOURS ON MARCH 19, 1991.

[52] LA SALLE 1 DOCKET 50-373 LER 91-002
CONTROL ROOM VENTILATION ISOLATION DURING RETURN TO SERVICE DUE TO COGNITIVE ERROR.
EVENT DATE: 032691 REPORT DATE: 042591 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LA SALLE 2 (BWR)

(NSIC 221866) ON 3/26/91, AT APPROX. 2024 HOURS WITH UNIT 1 DEFUELED AND UNIT 2 IN OPERATIONAL CONDITION 1 (RUN) AT 99% POWER, A DIVISION 1, GROUP 4 PRIMARY CONTAINMENT (PC) ISOLATION OCCURRED. THIS ISOLATION CAUSED UNIT 2 REACTOR BUILDING VENTILATION (VR) INBOARD DAMPERS 2VR05YA AND 2VR04YB TO CLOSE AND THE VR FANS TO TRIP OFF. IT ALSO CAUSED THE STANDBY GAS TREATMENT SYSTEM (VG) ON BOTH UNITS TO AUTO START. THE CAUSE OF THIS EVENT WAS A MOMENTARY LOSS OF CONTINUITY IN THE REACTOR BUILDING VENTILATION AND FUEL POOL COOLING EXHAUST RADIATION MONITOR TRIP LOGIC. THE LOSS OF CONTINUITY OCCURRED WHEN A TECHNICIAN LOOSENED AN ELECTRICAL CONNECTION IN THE TRIP CIRCUITRY TO REMOVE A PREVIOUSLY INSTALLED JUMPER. THE LOCATION OF THE INTERRUPTED CIRCUIT PATH IS SUCH THAT THE POWER TO THE CONTACTS FORMING THE COMBINATIONAL LOGIC WAS LOST, RESULTING IN THE COMPLETE ISOLATION SIGNAL FROM THE SINGLE OPEN CIRCUIT. THE ROOT CAUSE OF THIS EVENT WAS A COGNITIVE ERROR ON THE PART OF THE SHIFT ENGINEER AND IM SUPERVISOR APPROVING THE REMOVAL OF THE TEMPORARY SYSTEM CHANGES. THE POSSIBILITY OF DISTURBING THE CONTINUITY OF PCIS LOGIC CIRCUITS DURING THE REMOVAL PROCESS SHOULD HAVE BEEN RECOGNIZED, AND THE NEED OF TEMPORARY ALLIGATOR CLIPS SPECIFIED TO THE

TECHNICIAN. THE TEMPORARY SYSTEM CHANGE PROCEDURE WILL BE REVISED AND THE DEPARTMENTS INVOLVED IN THE EVENT WILL BE ENTRAINED ON THE PROCEDURE.

[53] LA SALLE 1 DOCKET 50-373 LER 91-004
PARTIAL GROUP II PRIMARY CONTAINMENT ISOLATION DUE TO BLOWN FUSES.
EVENT DATE: 040191 REPORT DATE: 043091 NSSS: GE TYPE: BWR
VENDOR: BUSSMANN MFG (DIV OF MCGRAW-EDISON)

(NSIC 221955) ON 4/1/91 AT APPROX. 1416 HOURS WITH UNIT 1 DEFUELED AT 0% POWER, THE "B" REACTOR RECIRCULATION (RR) HYDRAULIC POWER UNIT (HPU) INBOARD ISOLATION VALVES 1B33-F338B, 1B33-F340B, 1B33-F342B, AND 1B33-F344B CLOSED. THE ONLY RELATED WORK IN PROGRESS AT THE TIME WAS THE INSTALLATION OF BANANA JACKS BY INSTRUMENT MAINTENANCE DEPARTMENT IN ACCORDANCE WITH WORK REQUEST L94823. UPON INVESTIGATION IT WAS DETERMINED THAT FUSE 1B21H-FU1B IN PANEL 1PA14J HAD BLOWN CAUSING SEVERAL ISOLATION RELAYS TO DE-ENERGIZE, RESULTING IN THE PARTIAL PRIMARY CONTAINMENT ISOLATION SYSTEM (PC) GROUP II AND X ISOLATION SIGNAL WHICH CAUSED THE VALVES TO CLOSE. ALL OF THE OTHER GROUP II AND X VALVES AND ACTUATIONS WERE ALREADY CLOSED, OUT-OF-SERVICE, OR POWERED FROM ANOTHER FUSE. THE CAUSE OF THE BLOWN FUSE COULD NOT BE IDENTIFIED. IT IS BELIEVED THAT CURRENT SURGES DURING THE LIFTING AND RELANDING OF AN ENERGIZED RELAY'S POWER LEAD MAY HAVE WEAKENED THE FUSE SUCH THAT IT BLEW SOME SHORT TIME LATER WITH ONLY NORMAL CURRENT FLOW. THE FUSE WAS REPLACED. THE ISOLATION LOGIC WAS RESET. THE INSTRUMENT MAINTENANCE DEPARTMENT PRACTICE OF INSTALLING BANANA JACKS ON ENERGIZED CIRCUITS WAS HALTED. THE WORK INSTRUCTIONS WERE REWRITTEN TO DE-ENERGIZE THE CIRCUIT PRIOR TO INSTALLING BANANA JACKS TO MINIMIZE POSSIBILITY OF CAUSING AN ACTUATION.

[54] LA SALLE 1 DOCKET 50-373 LER 91-005
MISSED TECHNICAL SPECIFICATION SURVEILLANCE ON CONTAINMENT MONITORING DUE TO INADEQUATE PRE-LICENSE REVIEW.
EVENT DATE: 041091 REPORT DATE: 051091 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LA SALLE 2 (BWR)

(NSIC 222008) ON APRIL 10, 1991 AT 1230 HOURS WITH UNIT 1 IN MODE 5 (REFUEL) AND UNIT 2 IN MODE 1 (RUN) AT 0%/100% POWER RESPECTIVELY, IT WAS DETERMINED THAT LASALLE STATION HAD NOT PERFORMED A TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENT ON SUPPRESSION CHAMBER OXYGEN SAMPLING. TECHNICAL SPECIFICATION 3.6.6.2 WAS CHANGED FROM TAKING AN OXYGEN SAMPLE FROM PRIMARY CONTAINMENT TO TAKING THE OXYGEN SAMPLE FROM THE DRYWELL AND SUPPRESSION CHAMBER DURING PRE-LICENSING (1981). LASALLE OPERATING SURVEILLANCE LOS-AA-W1, "TECHNICAL SPECIFICATION WEEKLY SURVEILLANCES" PERFORMS SAMPLING OF THE DRYWELL ON A WEEKLY BASIS BUT DID NOT GET REVISED TO INCLUDE SAMPLING OF THE SUPPRESSION CHAMBER. THE APPARENT CAUSE OF THE EVENT WAS DUE TO AN INADEQUATE REVIEW OF THE TECHNICAL SPECIFICATION CHANGE TO DETERMINE PROCEDURES THAT REQUIRED REVISIONS. THE CONSEQUENCES OF THIS EVENT ARE MINIMAL SINCE THE DRYWELL AND SUPPRESSION CHAMBER ARE INERTED IN PARALLEL AND THE DRYWELL FREE AIR SPACE IS LARGER THAN THE SUPPRESSION CHAMBERS FREE AIR SPACE, THEREFORE MAKING IT HIGHLY LIKELY THAT A NON-COMBUSTIBLE MIXTURE HAS BEEN MAINTAINED IN THE SUPPRESSION CHAMBER. LASALLE OPERATION SURVEILLANCE LOS-AA-W1, WAS REVISED AND PERFORMED SATISFACTORIY ON APRIL 10, 1991 AND FOUND THE DRYWELL AND SUPPRESSION CHAMBER AT 2.4% AND 3.2% OXYGEN CONCENTRATION RESPECTIVELY.

[55] LA SALLE 2 DOCKET 50-374 LER 91-002
DIVISION 1 REACTOR BUILDING VENTILATION ISOLATION.
EVENT DATE: 031891 REPORT DATE: 041791 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LA SALLE 1 (BWR)

(NSIC 221956) ON MARCH 18, 1991, AT APPROXIMATELY 2044 HOURS WITH UNIT 1 DEFUELED, AND UNIT 2 RUNNING AT 100 PERCENT POWER, A DIVISION 1, GROUP 4 PRIMARY CONTAINMENT ISOLATION OCCURRED. THIS CAUSED THE UNIT 2 REACTOR BUILDING VENTILATION (VR) OUTBOARD DAMPERS 2VR05YB AND 2VR04YA TO CLOSE AND THE VR FANS TO TRIP OFF. IT ALSO CAUSED THE STANDBY GAS TREATMENT SYSTEM ON BOTH UNITS TO AUTO START. THE APPARENT CAUSE OF THIS EVENT IS UNCERTAIN, BUT AT THE TIME OF THE EVENT THE INSTRUMENT MAINTENANCE DEPARTMENT WAS PERFORMING LASALLE INSTRUMENT

SURVEILLANCE LIS-VR-102, "UNIT 1 REACTOR BUILDING FUEL POOL EXHAUST RADIATION MONITOR CALIBRATION" ON THE "A" CHANNEL. DUE TO THIS SURVEILLANCE ON THE FUEL POOL EXHAUST PROCESS RADIATION MONITOR, THERE WAS A ONE-HALF ISOLATION SIGNAL BEING GIVEN BY THE "A" CHANNEL AND A SIGNAL FROM THE "B" CHANNEL IS ALL THAT WOULD HAVE TO BE RECEIVED TO CAUSE THE GROUP 4 ISOLATION. THERE WERE NO INDICATIONS OF AN ISOLATION SIGNAL BEING RECEIVED BY THE "B" CHANNEL, AND IT WAS TESTED SATISFACTORILY WHEN THE CALIBRATION WAS PERFORMED. THE INSTRUMENT MAINTENANCE DEPARTMENT LOOKED FOR LOOSE CONNECTIONS AND FOUND NONE. THE SURVEILLANCE ON THE "A" CHANNEL WAS REPERFORMED, AND TESTED SATISFACTORILY, AND THE ISOLATION PROBLEM DID NOT REOCCUR. THIS EVENT IS REPORTABLE PURSUANT TO THE REQUIREMENTS OF 10CFR50.73(A)(2)(IV) DUE TO THE ACTUATION OF AN ENGINEERED SAFETY FEATURE SYSTEM.

[56]   LA SALLE 2   DOCKET 50-374   LER 91-003
REACTOR BUILDING VENTILATION ISOLATION DUE TO PERSONNEL ERROR DURING RETURN TO SERVICE.
EVENT DATE: 032891   REPORT DATE: 042691   NSSS: GE   TYPE: BWR
OTHER UNITS INVOLVED: LA SALLE 1 (BWR)

(NSIC 221867) ON MARCH 28, 1991, WITH UNIT 2 IN OPERATIONAL CONDITION 1 (RUN) AT 100% POWER, AND UNIT 1 DEFUELED, THE OPERATING DEPARTMENT WAS TO TRANSFER THE REACTOR PROTECTION SYSTEM BUS. THE NUCLEAR STATION OPERATOR REVIEWED THE PROCEDURE AND TRANSFERRED A PART OF IT TO A SEPARATE PIECE OF PAPER WHERE HE WROTE INCORRECTLY THE TERMINAL POINTS TO BE JUMPERED OUT BY THE EQUIPMENT OPERATOR. THE EQUIPMENT OPERATOR FOLLOWED THE INSTRUCTIONS ON THE PIECE OF PAPER FOR JUMPERING OUT THE TERMINAL POINTS, THEREFORE THE WRONG TERMINAL POINTS WERE JUMPERED OUT. (THE UNIT 2 ISOLATION LOGIC FROM UNIT 1 SHOULD HAVE BEEN JUMPERED OUT BUT INSTEAD UNIT 1 ISOLATION LOGIC FROM UNIT 1 WAS JUMPERED OUT.) WHEN THE REACTOR PROTECTION SYSTEM BUS WAS TRANSFERRED TO THE ALTERNATE POWER SUPPLY, THE DIVISION 1, UNIT 2 REACTOR BUILDING VENTILATION ISOLATION DAMPERS ISOLATED AND THE UNIT 1 STANDBY GAS TREATMENT SYSTEM AUTO STARTED. BY USING LASALLE OPERATING ABNORMAL PROCEDURE, LOA-VR-01, "RECOVERY FROM A GROUP IV ISOLATION OR SPURIOUS TRIP OF REACTOR BUILDING VENT", THE SYSTEM WAS UN-ISOLATED. THE INVOLVED PERSONNEL WERE TAILGATED IN THE IMPORTANCE OF FOLLOWING PROCEDURE AND THE USE OF "IN-HAND" PROCEDURES. THIS EVENT IS BEING REPORTED PURSUANT TO THE REQUIREMENTS OF 10CFR50.73(A)(2)(IV) DUE TO THE ACTUATION OF AN ENGINEERED SAFETY SYSTEM.

[57]   LIMERICK 1   DOCKET 50-352   LER 91-002
INOPERABILITY OF THE SEISMIC MONITORING SYSTEM FOR MORE THAN 30 DAYS DUE TO THE MECHANICAL FAILURE OF THE MAGNETIC TAPE PLAYBACK SYSTEM.
EVENT DATE: 010991   REPORT DATE: 011891   NSSS: GE   TYPE: BWR
OTHER UNITS INVOLVED: LIMERICK 2 (BWR)

(NSIC 221011) ON 12/10/90 AT APPROX. 0900 HOURS, THE LIMERICK GENERATING STATION (LGS) COMMON SEISMIC MONITORING SYSTEM (SMS) WAS DETERMINED TO BE INOPERABLE WHEN SURVEILLANCE TEST (ST) PROCEDURE ST-2-036-606-0, "SEISMIC MONITORING - TRIAXIAL TIME-HISTORY ACCELEROMETER/RECORDER FUNCTIONAL TEST (XRSH-VA-107)," COULD NOT BE COMPLETED SATISFACTORILY. THE SMS ENGINEER INITIATED A PURCHASE REQUISITION FOR THE REPLACEMENT OF THE CASSETTE DRIVE UNIT OF XRSH-VA-107. ON 1/9/91, AT 0900 HOURS, THE REPLACEMENT CASSETTE DRIVE UNIT AND ASSOCIATED PRINTED CIRCUIT BOARDS HAD NOT BEEN RECEIVED DESPITE BEING EXPEDITED. THEREFORE, THE SMS BECAME INOPERABLE FOR MORE THAN 30 DAYS REQUIRING THE SUBMISSION OF A SPECIAL REPORT IN ACCORDANCE WITH TECH SPECS SECTION 3.3.7.2. NO SEISMIC EVENTS OCCURRED WHILE THE SMS WAS DECLARED INOPERABLE. THE ACTUAL AND POTENTIAL CONSEQUENCES OF THIS EVENT WERE MINIMAL. THE CAUSE OF THE MALFUNCTION WAS NOISE BEING TRANSPOSED ONTO THE RECORDED SIGNAL ON THE CASSETTE TAPE. XRSH-VA-107 WAS REPAIRED AND RETURNED TO SERVICE ON 1/10/91. THE SMS WAS DECLARED OPERABLE AT 1530 HOURS ON 1/10, FOLLOWING SATISFACTORY COMPLETION OF ST-2-036-606-0. A COMPREHENSIVE REVIEW TO DETERMINE IF THERE ARE ADEQUATE STOCK LEVELS OF SPARE PARTS FOR ALL THE SMS IS BEING PERFORMED.

[58] LIMERICK 2 DOCKET 50-353 LER 91-002
INADVERTENT ISOLATION OF PRIMARY AND SECONDARY CONTAINMENTS DUE TO A BLOWN FUSE DURING TESTING OF A BATTERY CHARGER.
EVENT DATE: 032491 REPORT DATE: 042391 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LIMERICK 1 (BWR)
VENDOR: ASEA ELECTRIC, INC.

(NSIC 221861) ON 3/24/91, WHILE UNIT 2 WAS SHUTDOWN, VARIOUS ACTUATIONS OF THE UNIT 2 PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM AND A UNIT 2 REACTOR ENCLOSURE SECONDARY CONTAINMENT ISOLATION OCCURRED. THESE ARE ENGINEERED SAFETY FEATURE ACTUATIONS. THE ACTUATIONS ARE SUSPECTED TO BE THE TESTING OF THE DIVISION III SAFEGUARD BATTERY CHARGER. THIS BLOWN FUSE CAUSED A TRANSIENT CONDITION IN THE DC CONTROL POWER TO THE SECOND OF TWO SETS OF UNDERVOLTAGE AND OVERVOLTAGE RELAYS FOR THE 2A REACTOR PROTECTION SYSTEM/UNINTERRUPTIBLE POWER SUPPLY (RPS/UPS) DISTRIBUTION PANEL POWER SUPPLY BREAKERS. THE LOSS OF CONTROL POWER IS SUSPECTED OF CAUSING ONE OF THE PROTECTIVE RELAYS TO TRIP RESULTING IN A TRIP OF THE ASSOCIATED RPS/UPS DISTRIBUTION PANEL POWER SUPPLY BREAKER. THE POWER SUPPLY BREAKER WAS RECLOSED AND THE 2B LOOP OF SHUTDOWN COOLING WAS RETURNED TO SERVICE WITHIN 24 MINUTES. ALL SYSTEMS RESPONDED AS DESIGNED DURING THE LOSS OF POWER. THE PRIMARY CAUSE OF THE BLOWN FUSE WAS A LACK OF INFORMATION FROM THE MANUFACTURER OF A BATTERY CAPACITY TEST SYSTEM USED TO TEST BATTERY CHARGERS. THE APPROPRIATE BATTERY CHARGER TEST PROCEDURES WILL BE REVISED TO REQUIRE TESTING TO BE PERFORMED WITH THE TEST EQUIPMENT IN THE PROPER MODE. THE CAUSE OF THE TRIP OF THE 2A RPS/UPS DISTRIBUTION PANEL POWER SUPPLY BREAKER IS STILL UNDER INVESTIGATION.

[59] LIMERICK 2 DOCKET 50-353 LER 91-003
INADVERTENT ACTUATION OF REACTOR PROTECTION SYSTEM DUE TO A PERSONNEL ERROR DURING THE APPLICATION OF A TEMPORARY CIRCUIT ALTERATION.
EVENT DATE: 032491 REPORT DATE: 042291 NSSS: GE TYPE: BWR

(NSIC 221862) ON 3/24/91, AN INADVERTENT REDUNDANT REACTIVITY CONTROL SYSTEM (RRCS) ALTERNATE ROD INSERTION (ARI) ACTUATION WAS INITIATED FROM DIVISION I LOGIC DURING THE APPLICATION OF A TEMPORARY CIRCUIT ALTERATION (TCA) TO SUPPORT UNIT 2 OUTAGE WORK ON DIVISION I AND III INSTRUMENTATION. THE ARI INITIATION CAUSED THE SCRAM INLET AND OUTLET VALVES TO OPEN, AND CAUSED THE SCRAM DISCHARGE VOLUME (SDV) VENT AND DRAIN VALVES TO CLOSE. REPOSITIONING OF THESE VALVES CAUSED LEVEL IN THE SDV TO INCREASE TO ITS HIGH LEVEL TRIP SETPOINT, CAUSING A REACTOR PROTECTION SYSTEM (RPS) FULL SCRAM ACTUATION. THERE WAS NO CONTROL ROD MOTION DURING THIS EVENT SINCE ALL CONTROL RODS WERE FULLY INSERTED PRIOR TO THE EVENT. THE ARI AND THE RPS ACTUATED AS DESIGNED DURING THIS EVENT. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR RESULTING IN LESS THAN ADEQUATE COMMUNICATION DURING INSTALLATION OF THE TCA; AN RRCS RESET WAS REQUESTED WHEN BOTH RRCS AND ARI RESETS WERE REQUIRED. PROCEDURAL GUIDANCE WILL BE DEVELOPED TO ENSURE BOTH RRCS AND ARI RESETS ARE PERFORMED FOR THESE ROUTINE DIVISIONAL TCAS THAT ARE APPLIED DURING OUTAGES. THE LICENSED OPERATOR REQUALIFICATION TRAINING MODULE FOR RRCS WILL BE REVISED TO INCLUDE MORE DETAILED INFORMATION ON RRCS AND ARI LOGIC RESETS.

[60] LIMERICK 2 DOCKET 50-353 LER 91-004
THE INADVERTENT START OF AN EMERGENCY DIESEL GENERATOR AND AN EMERGENCY SERVICE WATER PUMP DUE TO A SPURIOUS LOCA SIGNAL AS A RESULT OF PERSONNEL ERROR.
EVENT DATE: 032791 REPORT DATE: 042691 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LIMERICK 1 (BWR)

(NSIC 221863) ON 3/27/91, AN INADVERTENT LOCA SIGNAL WAS GENERATED WHILE SYSTEMS ENGINEERS (SES) WERE PERFORMING THE DIVISION III LOSS OF COOLANT ACCIDENT (LOCA) - LOSS OF OFFSITE POWER (LOOP) LOGIC SYSTEM FUNCTIONAL SURVEILLANCE TEST (ST) PROCEDURE ASSOCIATED WITH THE D23 EMERGENCY DIESEL GENERATOR (EDG). THIS LOCA SIGNAL CAUSED THE UNIT 2 D23 EDG TO START, AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION. ALSO, THE ASSOCIATED DIVISION III AC SAFEGUARD BUS SHED NON-ESSENTIAL LOADS. AFTER THE D23 EDG START, THE COMMON 'C' EMERGENCY SERVICE WATER (ESW) PUMP STARTED AUTOMATICALLY, ALSO AN ESF ACTUATION. THE ACTUAL CONSEQUENCES OF THIS EVENT WERE MINIMAL IN THAT NO ACTUAL LOCA SIGNAL OCCURRED AND NO ADVERSE CONDITIONS RESULTED FROM THE SPURIOUS SIGNAL. THE CAUSE OF THIS EVENT WAS

PERSONNEL ERROR IN THAT THE SE FUNCTIONING AS THE TEST DIRECTOR IN THE MAIN CONTROL ROOM (MC) DID NOT COMPLETE TWO SUBSTEPS OF A MULTIPLE PART PROCEDURAL STEP DUE TO A LACK OF ATTENTION TO DETAIL. THE SES INVOLVED WERE COUNSELED ON THE NEED FOR ATTENTION TO DETAILS AND THE NEED TO ADEQUATELY COMMUNICATE. A LETTER FROM THE LIMERICK GENERATING STATION VICE PRESIDENT TO ALL SITE PERSONNEL WAS ISSUED ADDRESSING THIS EVENT AND OTHER INCIDENTS INVOLVING PERSONNEL ERROR THAT HAVE OCCURRED DURING THE CURRENT UNIT 2 REFUELING OUTAGE.

[61] LIMERICK 2 DOCKET 50-353 LER 91-005
SPECIAL REPORT FOR DIESEL GENERATOR TEST FAILURE.
EVENT DATE: 040191 REPORT DATE: 050191 NSSS: GE TYPE: BWR

(NSIC 221951) ON APRIL 1, 1991, WITH UNIT 2 IN A REFUELING CONDITION, PLANT PERSONNEL WERE PERFORMING SURVEILLANCE TEST (ST) PROCEDURE ST-1-092-111-2, "D21 DIESEL GENERATOR 4 KV SFGD LOSS OF POWER LSF/SAA AND OUTAGE TESTING." THE D21 EMERGENCY DIESEL GENERATOR (EDG) HAD BEEN DECLARED INOPERABLE TO PERFORM THIS PROCEDURE. THE D21 EDG SUCCESSFULLY REJECTED THE 2A RESIDUAL HEAT REMOVAL (RHR) SYSTEM PUMP MOTOR LOAD OF 992KW. IN ACCORDANCE WITH THE PROCEDURE, THE 2A RHR PUMP WAS RESTARTED; HOWEVER, THE D21 EDG OUTPUT VOLTAGE INCREASED ABOVE THE ACCEPTANCE CRITERIA VALUE TO APPROXIMATELY 5200 VOLTS. THE TEST WAS THEN TERMINATED. THE CAUSE OF THIS EVENT WAS AN IMPROPERLY INSTALLED POTENTIAL TRANSFORMER PRIMARY FUSE THAT MAY HAVE BEEN DUE TO PERSONNEL ERROR. THIS RESULTED IN A LOSS OF GENERATOR OUTPUT VOLTAGE FEEDBACK TO THE AUTOMATIC VOLTAGE REGULATOR. THIS EVENT IS CLASSIFIED AS A VALID FAILURE. IN THE EVENT OF AN ACTUAL LOSS OF OFFSITE POWER, THE D22 EDG AND D24 EDG WERE OPERABLE TO PROVIDE ADEQUATE POWER TO MAINTAIN THE REACTOR IN A SAFE SHUTDOWN CONDITION AS REQUIRED BY TECHNICAL SPECIFICATIONS. ALL OTHER EDG POTENTIAL TRANSFORMER PRIMARY FUSES WERE VERIFIED TO BE PROPERLY INSTALLED. AN OPERATOR AID WILL BE ADDED ONTO EACH EDG CABINET DOOR TO CAUTION OPERATORS TO ENSURE ALL FUSES ARE FIRMLY SEATED IN THEIR FUSE CLIPS. BECAUSE THIS IS THE FIRST FAILURE FOR THE D21 EDG, THE TEST INTERVAL REMAINS UNCHANGED AT 31 DAYS.

[62] MCGUIRE 2 DOCKET 50-370 LER 91-001
FAILURE TO COMPLY WITH ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TECHNICAL SPECIFICATIONS BECAUSE OF AN INAPPROPRIATE ACTION AND DEFECTIVE PROCEDURE.
EVENT DATE: 031191 REPORT DATE: 041091 NSSS: WE TYPE: PWR

(NSIC 221865) ON 3/11/91, AT APPROX. 1400, INSTRUMENT AND ELECTRICAL IAE PERSONNEL WERE PERFORMING PROCEDURE PT/2/A/4201/05B, UNIT 2 CONTAINMENT PRESSURE CONTROL ANALOG CHANNEL OPERATIONAL TEST. CONTAINMENT PRESSURE CONTROL SYSTEM CHANNEL 2NSPT5490 WAS FOUND TO BE FAULTY. AT 1430, THE OPERATIONS SENIOR REACTOR OPERATOR (OPS SRO) DECLARED CHANNEL 2NSPT5490 INOPERABLE AND LOGGED CHANNEL 2NSPT5490 IN THE TECH SPEC ACTION ITEM LOG (TSAIL) BOOK. IAE PERSONNEL WERE DIRECTED TO PLACE CHANNEL 2NSPT5490 IN THE START PERMISSIVE MODE WITHIN ONE HOUR TO COMPLY WITH TECH SPEC (TS) 3.3.2. IAE PERSONNEL DECIDED TO USE PROCEDURE PT/2/A/4201/05B TO PLACE CHANNEL 2NSPT5490 IN START PERMISSIVE MODE AS REQUIRED BY TS 3.3.2. IAE PERSONNEL PROCEEDED TO CONTAINMENT PRESSURE CONTROL CABINET (CPCC) 4 AND PERFORMED PROCEDURE PT/2/A/4201/05B, SECTION 12.1 THROUGH STEP 12.4.8. THIS ACTION HAD IAE PERSONNEL PLACE THE KEY-OPERATED CONTROL SWITCH FOR CHANNEL 2NSPT5490 IN THE TEST POSITION. THEY THOUGHT THIS ACTION PLACED THE CHANNEL IN THE START PERMISSIVE MODE, SATISFYING TS 3.3.2. WHILE TROUBLESHOOTING CHANNEL 2NSPT5490, IAE PERSONNEL DISCOVERED THE CHANNEL WAS NOT IN THE START PERMISSIVE MODE. THIS EVENT IS ASSIGNED A CAUSE OF INAPPROPRIATE ACTION BECAUSE IAE PERSONNEL IMPROPERLY FOLLOWED THE CORRECT PROCEDURE, IN ADDITION TO A CAUSE OF DEFECTIVE PROCEDURE.

[63] MILLSTONE 1 DOCKET 50-245 LER 91-006
UNJACKETED CABLE RESULTING IN LOSS OF ENVIRONMENTAL QUALIFICATION.
EVENT DATE: 030191 REPORT DATE: 041791 NSSS: GE TYPE: BWR

(NSIC 221917) ON 3/19/91, AT 1400 HOURS, WITH THE PLANT AT 89% POWER (530 DEGREES FAHRENHEIT AND 1030 PSIG), THE AS BUILT CONFIGURATION OF THE PIGTAIL TO FIELD

TERMINATION FOR A CABLE IN THE DRYWELL WAS DETERMINED NOT TO BE FULL ENVIRONMENTALLY QUALIFIED. THIS CONDITION IS REPORTABLE UNDER 10CFR50.73. ON 2/25/91, AN EVALUATION WAS INITIATED WHEN A QUESTION AROSE AS TO WHETHER A CABLE JACKET WAS RECONSTRUCTED ON KERITE CABLE AT THE PIGTAIL TO FIELD TERMINATION FOLLOWING THE INSTALLATION OF RAYCHEM HEAT SHRINK TUBING SLEEVE. KERITF CABLE MUST BE FULLY JACKETED TC TAKE CREDIT FOR BETA RADIATION SHIELDING TO THE CABLE INSULATION TO ENSURE ENVIRONMENTAL QUALIFICATION. SPECIFICALLY, WITH THE CABLE INSULATION EXPOSED TO AN ACCIDENT DRYWELL ENVIRONMENT, OPERABILITY OF THE CABLE END DEVICE COULD BE AFFECTED AND THUS THE ABILITY OF THE DEVICE TO MITIGATE THE CONSEQUENCES OF A LOSS OF COOLANT ACCIDENT. AN OPERABILITY EVALUATION WAS PERFORMED ON THE UNJACKETED KERITE CABLE FOUND ON 19 END DEVICES LOCATED IN THE DRYWELL. OF THE 19 END DEVICES EVALUATED, ALL WERE FOUND TO REMAIN OPERABLE TO PERFORM THEIR DESIGN BASIS ACCIDENT (DBA) FUNCTION WITH THE EXCEPTION OF THE FIELD CABLE TO THE AIR SOLENOID ON VALVE 1-CU-2A. THE SOLENOID SUPPLIES AIR TO CONTAINMENT ISOLATION VALVE 1-CU-2A. 1-C1-2A IS A 1/2 INCH BYPASS VALVE AROUND 1-CU-2.

[64] MILLSTONE 1 DOCKET 50-245 LER 91-004
EMERGENCY DIESEL GENERATOR INOPERABILITY RESULTING FROM LOW LUBE OIL PRESSURE.
EVENT DATE: 030791 REPORT DATE: 040591 NSSS: GE TYPE: BWR
VENDOR: FAIRBANKS MORSE

(NSIC 221916) ON MARCH 7, 1991, AT 1140, WITH THE PLANT IN COLD SHUTDOWN (170 DEGREES F, 0 PSIG), THE EMERGENCY DIESEL GENERATOR WAS DECLARED INOPERABLE FOLLOWING AN AUTOMATIC TRIP ON LOW LUBE OIL PRESSURE. FAILURE TO REACH MINIMUM LUBE OIL PRESSURE HAS BEEN ATTRIBUTED TO A LOW-AMBIENT ROOM TEMPERATURE CONDITION WHICH INCREASED THE LUBE VISCOSITY. PLANT PROCEDURAL REVISIONS HAVE BEEN IMPLEMENTED TO ASSURE LUBE OIL TEMPERATURE WILL NOT DECREASE TO A LEVEL AT WHICH A HIGH EMERGENCY DIESEL GENERATOR OPERABILITY COULD BE COMPROMISED. THE EMERGENCY DIESEL GENERATOR IS DESIGNED TO AUTOMATICALLY STAND ON LOSS OF NORMAL POWER, LOW-LOW REACTOR WATER LEVEL OR HIGH CONTAINMENT PRESSURE. THE EMERGENCY DIESEL GENERATOR PROVIDES EMERGENCY POWER FOR NECESSARY AUXILIARIES, IMPORTANT TO THE ENGINEERED SAFEGUARDS SYSTEMS AND POWER NEEDED DURING THE SHUTDOWN MODE OF OPERATION. NO SAFETY SYSTEMS WERE REQUIRED TO ACTUATE AS A RESULT OF THIS EVENT.

[65] MILLSTONE 2 DOCKET 50-336 LER 91-006
MISSED SURVEILLANCE.
EVENT DATE: 033091 REPORT DATE: 042991 NSSS: CE TYPE: PWR

(NSIC 221943) ON 3/30/91, WITH THE PLANT IN MODE 1 AT 100% POWER, IT WAS DETERMINED THAT SURVEILLANCE REQUIREMENT SECTION 4.0.5 WAS NOT PERFORMED WITHIN THE REQUIRED 92 DAY TIME INTERVAL AND THAT SURVEILLANCE REQUIREMENT SECTION 4.0.5 WAS NOT PERFORMED WITHIN THE REQUIRED 92 DAY TIME INTERVAL AND THAT SURVEILLANCE REQUIREMENT 4.0.2.A, "THE MAXIMUM ALLOWABLE EXTENSION NOT TO EXCEED 25% OF THE SURVEILLANCE TIME INTERVAL," WAS NOT SATISFIED. THE MISSED SURVEILLANCE WAS PERFORMED IMMEDIATELY, MET THE ACCEPTANCE CRITERIA, BUT WAS 7 DAYS PAST THE MAXIMUM ALLOWABLE EXTENSION. SURVEILLANCE REQUIREMENT SECTION 4.0.5 STATES: "INSERVICE INSPECTION OF ASME CODE CLASS 1, 2, AND 3 COMPONENTS AND INSERVICE TESTING ASME CODE CLASS 1, 2, AND 3 PUMPS AND VALVES SHALL BE PERFORMED IN ACCORDANCE WITH SECTION XI OF THE ASME BOILER AND PRESSURE VESSEL CODE." THE SPECIFIC SURVEILLANCE MISSED WAS THE QUARTERLY ISI STROKE TESTING OF THE FACILITY II LPSI SYSTEM PROCEDURE SP 21136. THE VALVES WERE 2-SI-451, LPSI PUMP B MINIMUM FLOW CHECK VALVE, 2-SI-635, AND 2-SI-645, LPSI TO LOOP 2A AND 2B STOP VALVES, RESPECTIVELY. THERE ARE NO SAFETY CONSEQUENCES AS A RESULT OF THIS EVENT BECAUSE NO MODIFICATIONS OR MAINTENANCE WAS PERFORMED ON THE THREE VALVES SINCE THE LAST SUCCESSFUL SURVEILLANCE TEST ON NOVEMBER 29, 1990.

[66] MILLSTONE 3 DOCKET 50-423 LER 90-030 REV 01
UPDATE ON MANUAL REACTOR TRIP DUE TO MOISTURE SEPARATOR REHEATER PIPING LINE BREAKS.
EVENT DATE: 123190 REPORT DATE: 050191 NSSS: WE TYPE: PWR

(NSIC 222019) ON 12/31/90, AT 1636 HOURS WITH THE PLANT IN MODE 1 AT 86% POWER,

580F AND 2250 PSIA, A MANUAL REACTOR TRIP WAS INITIATED DUE TO TWO SIX INCH MOISTURE SEPARATOR DRAIN LINE (DSM) PIPING BREAKS IN THE TURBINE BUILDING. FOLLOWING THE TRIP A MAIN STEAM LINE ISOLATION WAS INITIATED TO MINIMIZE THE RELEASE OF STEAM INTO THE TURBINE BUILDING. THE CAUSE OF THE EVENT WAS THE FAILURE OF THE TWO DSM LINES DOWNSTREAM OF THE RESPECTIVE LEVEL CONTROL VALVES. BOTH LINES APPEARED TO BURST, FAIL LONGITUDINALLY, THEN UNZIP CIRCUMFERENTIALLY AT THE MINIMUM WALL THICKNESS LOCATION. WALL THICKNESS AT THE RUPTURE WAS APPROX. 0.020 INCHES. CAUSE OF THE SEVERE WALL LOSS WAS SINGLE PHASE EROSION/CORROSION. THE COMBINATION OF TEMPERATURE, HIGH FLUID VELOCITY AND EXTREMELY LOW OXYGEN CONTENT ARE CAUSATIVE FACTORS. WALL LOSS WAS LOCALIZED. THE MINIMUM THICKNESS OCCURRED ADJACENT TO THE CONTROL VALVE(S) AND INCREASED AT 0.011 INCHES PER INCH DOWNSTREAM FROM THE VALVE(S). AS IMMEDIATE CORRECTIVE ACTION CONTROL ROOM OPERATORS PERFORMED THE ACTIONS REQUIRED BY THE APPLICABLE EMERGENCY OPERATING PROCEDURES. THE RUPTURED PIPES WERE CAPPED AND THE DSM PUMPS AND PIPING WERE ISOLATED TO ALLOW CONTINUED OPERATION. DURING THE THIRD REFUELING OUTAGE, WHICH BEGAN ON 2/2/91, THE PIPING UPSTREAM AND DOWNSTREAM OF THE LEVEL CONTROL VALVES WAS REPLACED WITH PIPING HAVING GREATER EROSION/CORROSION RESISTANT PROPERTIES.

[67] MILLSTONE 3 DOCKET 50-423 LER 91-004
CONTAINMENT LEAKAGE IN EXCESS OF LIMITS DUE TO VALVE LEAKAGE.
EVENT DATE: 020591 REPORT DATE: 030791 NSSS: WE TYPE: PWR
VENDOR: FISHER CONTROLS CO.
PRATT, HENRY COMPANY
WALWORTH COMPANY
WESTINGHOUSE ELEC CORP.-NUCLEAR ENERGY SYS

(NSIC 221518) WHILE SHUTDOWN IN MODES 5 (COLD SHUTDOWN) AND 6 (REFUELING) DURING THE PERFORMANCE OF LOCAL LEAK RATE TESTING (LLRT), THE "AS FOUND" LEAK RATES FOR FOUR CONTAINMENT ISOLATION VALVES EXCEEDED THE TECHNICAL SPECIFICATION TYPE C AND BYPASS LEAKAGE LIMITS OF 0.6 LA AND 0.042 LA. THE LLRT FAILURES OCCURRED ON 2/5/91 AT 1331 (FOR 3RHS*MV8702A), FEBRUARY 7, 1991 AT 2200 (FOR 3RSS*V6), 2/10/91 AT 2200 (FOR 3CDS*CTV91B), AND 2/19/91 AT 1330 (FOR 3RSS*MOV23B). NO IMMEDIATE ACTION WAS REQUIRED. LEAKAGE PAST 3RHS*MV8702A IS BELIEVED TO BE DUE TO DEBRIS OR BORIC ACID CRYSTALS ON THE SEATING SURFACE. THE PENETRATION WAS FLUSHED WITH WATER AND RETESTED SUCCESSFULLY. LEAKAGE PAST 3RSS*V6 WAS DUE TO IMPROPER SEATING CAUSED BY BORIC ACID CRYSTAL PRECIPITATION ON THE SEATING SURFACE. THE VALVE SEAT WAS CLEANED AND AN "AS-LEFT" LLRT WAS SATISFACTORILY PERFORMED. LEAKAGE PAST 3CDS*CTV91B WAS DUE TO FAIURE OF AN ELASTOMER T-RING WHICH HAD PARTIALLY ROLLED OUT OF ITS RETAINING GROOVE. THE T-RING WAS REPLACED AND AN AS-LEFT LLRT WILL BE PERFORMED PRIOR TO STARTUP. LEAKAGE PAST 3RSS*MOV23B WAS CAUSED BY SEPARATION OF THE VULCANIZED RUBBER SEAT FROM THE VALVE BODY MOUNTING SURFACE. THE VALVE WAS REMOVED FROM THE SYSTEM AND HAS BEEN SENT TO THE MANUFACTURER FOR OVERHAUL. IT WILL BE REINSTALLED AND RETESTED PRIOR TO STARTUP.

[68] MILLSTONE 3 DOCKET 50-423 LER 91-008
PRESSURIZER LEVEL INDICATION ERRORS DUE TO INADEQUATE DESIGN.
EVENT DATE: 031891 REPORT DATE: 041791 NSSS: WE TYPE: PWR

(NSIC 221965) ON MARCH 18, 1991, WHILE SHUTDOWN IN MODE 5 (COLD SHUTDOWN), AN ENGINEERING EVALUATION CONCLUDED THAT CERTAIN LEVEL ERRORS INTRODUCED INTO THE THREE REACTOR COOLANT SYSTEM PRESSURIZER LEVEL TRANSMITTERS, CREATED A CONDITION WHICH RESULTED IN TWO INDEPENDENT CHANNELS BEING INOPERABLE IN A SINGLE SYSTEM DESIGNED TO MITIGATE THE CONSEQUENCES OF AN ACCIDENT. THE INSTRUMENT INACCURACIES RESULTED FROM NON-CONDENSABLE GAS ACCUMULATION IN THE CONDENSATE POTS/RESERVOIRS FOR THE PRESSURIZER LEVEL INSTRUMENT REFERENCE LEGS. THE PRESSURIZER LEVEL INSTRUMENTATION PROVIDE THE PRIMARY INDICATION TO THE OPERATOR FOR MANUAL ACTIONS DURING AND FOLLOWING ACCIDENT CONDITIONS. WITH THE ERRORS POSTULATED, THESE TRANSMITTERS WOULD HAVE PROVIDED OPERATORS MISLEADING INFORMATION. THE ROOT CAUSE OF THE ACCUMULATION OF NON-CONDENSIBLE GASES IS INDEQUATE DESIGN. THE PRESSURIZER LEVEL REFERENCE LEGS WERE ANGLED UPWARD FROM THE PRESSURIZER TO THE CONDENSATE POTS, WHICH ALLOWED GASES TO BUILD UP IN THE CONDENSATE POTS. A DESIGN CHANGE COMPLETED DURING THE REFUELING OUTAGE, ELIMINATED THE CONDENSATE POTS. THE PRESSURIZER REFERENCE LEG TAPS WERE MODIFIED

TO RUN STRAIGHT OUT AND THEN ANGLE DOWNWARD VIA ISOLATION VALVES TO THE LEVEL TRANSMITTERS. THIS ALLOWS THE FREE FLOW OF STEAM AND GASES ALONG THE PIPING RUNS TO MINIMIZE BUILDUP OF GASES IN THE REFERENCE LEGS.

[69] MONTICELLO DOCKET 50-263 LER 90-019 REV 01
UPDATE ON POTENTIAL LOSS OF FUEL OIL TRANSFER CAPABILITY DURING EXTERNAL FLOODING DUE TO PROCEDURAL INADEQUACY.
EVENT DATE: 112690 REPORT DATE: 042491 NSSS: GE TYPE: BWR

(NSIC 221896) ON NOVEMBER 26, 1990 DURING DESIGN BASIS REVIEW OF MONTICELLO PLANT EXTERNAL FLOODING REQUIREMENTS, IT WAS DETERMINED THAT THE EMERGENCY PROCEDURE FOR EXTERNAL FLOODING DID NOT INCLUDE PROTECTIVE MEASURES FOR THE EMERGENCY DIESEL GENERATOR FUEL OIL TRANSFER HOUSE. IN THE ABSENCE OF PROTECTIVE MEASURES, THE PROBABLE MAXIMUM FLOOD EVENT COULD PREVENT THE FULFILLMENT OF A SAFETY FUNCTION REQUIRED TO MAINTAIN THE PLANT IN A SAFE SHUTDOWN CONDITION, I.E., DIESEL FUEL OIL TRANSFER CAPABILITY COULD BE LOST RENDERING ONSITE AC SOURCES INOPERABLE. THIS CONDITION WAS REPORTED TO THE NRC ON NOVEMBER 26, 1990 AT 19:59 AS A FOUR HOUR NON-EMERGENCY NOTIFICATION IN ACCORDANCE WITH 10 CFR PART 50, SECTION 50.72, PARAGRAPH (B)(2)(III)(A). SUBSEQUENT TO THE INITIAL NOTIFICATION, IT WAS DETERMINED THAT THE VENT FOR THE DIESEL FUEL OIL STORAGE TANK IS BELOW THE PROBABLE MAXIMUM FLOOD LEVEL AND THAT FLOODING OF THE STORAGE TANK COULD OCCUR DURING THIS EVENT. ON MARCH 26, 1991 THE VERIFICATION WALKDOWN OF THE DESIGN BASIS DOCUMENT FOR EXTERNAL FLOODING IDENTIFIED ADDITIONAL PROCEDURAL DEFICIENCIES. THE CAUSE OF THIS EVENT IS PROCEDURAL INADEQUACY. THE EMERGENCY PROCEDURE FOR EXTERNAL FLOODING HAS BEEN REVISED TO CORRECT ALL IDENTIFIED DEFICIENCIES.

[70] MONTICELLO DOCKET 50-263 LER 91-002 REV 01
UPDATE ON CONTROL POWER CIRCUIT BREAKER FOR FILTRATION UNIT HEATER FOUND OPEN RESULTING IN INOPERABILITY OF CONTROL ROOM EMERGENCY FILTRATION SYSTEM.
EVENT DATE: 010991 REPORT DATE: 043091 NSSS: GE TYPE: BWR

(NSIC 221918) ON JANUARY 9, 1991 WITH THE PLANT OPERATING AT 97% POWER, ENGINEERING PERSONNEL DISCOVERED THE 120VAC CONTROL POWER CIRCUIT BREAKER FOR THE "A" EMERGENCY FILTRATION TREATMENT SYSTEM FILTER HEATER IN THE "OFF" POSITION. THIS EVENT WAS REPORTABLE SINCE IT RESULTED IN OPERATION IN A CONDITION PROHIBITED BY THE TECHNICAL SPECIFICATIONS, WHICH REQUIRE THE EMERGENCY FILTRATION SYSTEM BE OPERABLE WHENEVER REACTOR WATER TEMPERATURE IS GREATER THAN 212 DEGREES FAHRENHEIT. THE CONTROL POWER CIRCUIT BREAKER WAS RETURNED TO THE "ON" POSITION AND THE CONTROL ROOM EMERGENCY FILTRATION TREATMENT SYSTEM RETURNED TO NORMAL OPERATION FOLLOWING SUCCESSFUL COMPLETION OF THE MONTHLY SURVEILLANCE TEST. THE PLANT WORK CONTROL PROCESS WILL BE REVISED TO REQUIRE THAT FOLLOWING WORK IN A SAFETY-RELATED POWER PANEL OR LIGHTING PANEL A VERIFICATION OF ALL BREAKER POSITIONS IN THAT PANEL BE COMPLETED.

[71] MONTICELLO DOCKET 50-263 LER 91-005
PRIMARY CONTAINMENT ISOLATION VALVES LOCAL LEAK RATE TESTING EXCEEDED.
EVENT DATE: 040191 REPORT DATE: 050191 NSSS: GE TYPE: BWR
VENDOR: ATWOOD & MORRILL CO., INC.
ROCKWELL MANUFACTURING COMPANY
VOGT, HENRY MACHINE COMPANY

(NSIC 221919) WHILE CONDUCTING LOCAL LEAK RATE TESTING DURING THE 1991 REFUELING OUTAGE, IT WAS FOUND THAT THE TECHNICAL SPECIFICATION 3.7.A.2.B LEAK RATE ACCEPTANCE CRITERIA WAS EXCEEDED. MAIN STEAM ISOLATION VALVES AO-2-86A, AO-2-86B, AND AO-2-86D WERE LEAKING IN EXCESS OF 11.5 STANDARD CUBIC FOOT PER HOUR (SCFH), AND THE AS-FOUND COMBINED LEAKAGE RATE OF ALL TYPE B AND C PENETRATIONS USING MAXIMUM PATHWAY LEAKAGE WAS IN EXCESS OF 0.6 ALLOWABLE ACCIDENT LEAKAGE (LA). INVESTIGATION AND REPAIR IS IN PROGRESS. DETERMINATION OF ROOT CAUSES IS BEING PURSUED AND WILL BE ADDRESSED IN A SUPPLEMENTAL LICENSEE EVENT REPORT. ALL TYPE B AND C PENETRATIONS WILL BE REPAIRED TO WITHIN REQUIRED ACCEPTANCE CRITERIA.

[72] NINE MILE POINT 1 DOCKET 50-220 LER 89-014 REV 01
UPDATE ON REDUNDANT SAFETY SYSTEMS INOPERABLE DUE TO THE LACK OF A COMPLETE PROGRAM TO CALIBRATE NON-TECHNICAL SPECIFICATION INSTRUMENTATION.
EVENT DATE: 100489 REPORT DATE: 041591 NSSS: GE TYPE: BWR

(NSIC 221914) ON 8/30/89, WITH THE REACTOR SHUTDOWN AND THE CORE OFFLOADED, IT WAS DISCOVERED THAT THE HEATING ELEMENT TEMPERATURE CONTROLLERS FOR THE NINE MILE POINT UNIT 1 (NMP1) REACTOR BUILDING EMERGENCY VENTILATION SYSTEM CHARCOAL FILTER DUCT HEATERS WERE NOT IN A SCHEDULED CALIBRATION PROGRAM. ON 10/4/89, AFTER TESTING, IT WAS DETERMINED THAT THE CONTROLLERS WERE SET BELOW DESIGN CRITERIA WHICH MAY HAVE RESULTED IN THE REDUNDANT COMPONENTS BECOMING INOPERABLE. IMMEDIATE CORRECTIVE ACTION INCLUDED THE COMPLETION OF A PREVIOUSLY DEVELOPED PROGRAM FOR THE CALIBRATION AND PERIODIC TESTING OF NON-TECH SPEC BALANCE OF PLANT INSTRUMENTATION WHICH MAY AFFECT SYSTEM AND COMPONENT OPERABILITY. ALSO A COMPLETE EVALUATION OF THE REACTOR BUILDING EMERGENCY VENTILATION SYSTEM WAS PERFORMED. THE ADDITIONAL CONDITIONS THAT WERE FOUND ARE BEING REPORTED IN THIS SUPPLEMENT. ADDITIONALLY, A LESSONS LEARNED TRANSMITTAL WAS DEVELOPED TO COMMUNICATE THE CONSEQUENCES OF SUCH A CONDITION.

[73] NINE MILE POINT 1 DOCKET 50-220 LER 90-008 REV 01
UPDATE ON REACTOR BUILDING EMERGENCY VENTILATION INITIATION DUE TO PERSONNEL ERROR.
EVENT DATE: 052390 REPORT DATE: 041991 NSSS: GE TYPE: BWR

(NSIC 221888) ON MAY 23, 1990, AT 1313 HOURS, NINE MILE POINT UNIT 1 (NMP1) EXPERIENCED AN ACTUATION OF AN ENGINEERING SAFETY FEATURE (ESF). SPECIFICALLY, INITIATION OF REACTOR BUILDING EMERGENCY VENTILATION (RBEV) AND ISOLATION OF REACTOR BUILDING NORMAL VENTILATION. AT THE TIME OF THE EVENT, THE PLANT WAS IN AN EXTENDED REFUELING OUTAGE WITH THE CORE LOADED AND THE MODE SWITCH IN THE "REFUEL" POSITION. THE ROOT CAUSE OF THE EVENT WAS DUE TO PERSONNEL ERROR IN THAT CARE WAS NOT EXERCISED WHEN WORKING IN CLOSE PROXIMITY TO CONTROL DEVICES. THE IMMEDIATE CORRECTIVE ACTIONS CONSISTED OF VERIFYING THE RBEV INITIATION, RESETTING CONTROL LOGIC AND ALARMS, RETURNING THE REACTOR BUILDING NORMAL VENTILATION TO SERVICE AND RESTORING THE RBEV TO STANDBY. OTHER CORRECTIVE ACTIONS CONSISTED OF COUNSELING THE RESPONSIBLE TECHNICIAN AND DEVELOPMENT OF A LESSONS LEARNED TRANSMITTAL (LLT). LONG TERM CORRECTIVE ACTION IS TO EVALUATE THE REPLACEMENT OF THE MOTOR GENERATOR SETS WITH UNINTERRUPTED POWER SUPPLIES (UPS) TO PROVIDE A RELIABLE POWER SOURCE TO THE MONITORS, ALONG WITH EVALUATING THE NEED TO MODIFY THE RBEV LOGIC.

[74] NINE MILE POINT 1 DOCKET 50-220 LER 91-001 REV 01
UPDATE ON REACTOR SCRAM DUE TO SPURIOUS NON-COINCIDENT LOGIC TRIP SIGNAL.
EVENT DATE: 010891 REPORT DATE: 042591 NSSS: GE TYPE: BWR

(NSIC 221887) ON JANUARY 8, 1991, AT 0327 HOURS, NINE MILE POINT UNIT 1 (NMP1) EXPERIENCED A FULL REACTOR SCRAM AND A MAIN STEAM ISOLATION VALVE (MSIV) ISOLATION. THE MODE SWITCH WAS IN THE REFUEL POSITION, ALL CONTROL RODS WERE AT POSITION 00, REACTOR COOLANT TEMPERATURE WAS 163 DEGREES FAHRENHEIT, AND REACTOR PRESSURE WAS 0 PSIG. WATER LEVEL WAS BEING LOWERED FOLLOWING THE PERFORMANCE OF A REACTOR VESSEL INSERVICE LEAK TEST. THE INSIDE MSIV'S CLOSED DUE TO THE ISOLATION SIGNAL AND THE OUTSIDE MSIV'S WERE ALREADY CLOSED DUE TO THE HYDRO. THE CAUSE OF THE EVENT WAS DETERMINED TO BE A SPURIOUS ACTUATION OF REACTOR PROTECTION SYSTEM (RPS) NON-COINCIDENT LOGIC, SPECIFICALLY CONDENSER LOW VACUUM AND MAIN STEAM ISOLATION CLOSED BYPASS LOGIC. CORRECTIVE ACTIONS TAKEN WERE TO PERFORM CHECKS, TROUBLESHOOTING, AND MONITORING OF COMPONENTS THAT MAKE UP THE CONDENSER LOW VACUUM AND MAIN STEAM ISOLATION CLOSED BYPASS LOGIC. THE RESULTS OF THESE CORRECTIVE ACTIONS DID NOT INDICATE ANY EQUIPMENT MALFUNCTION OR DEGRADATION THAT COULD HAVE CONTRIBUTED TO A SPURIOUS ACTUATION.

[75] NINE MILE POINT 1 DOCKET 50-220 LER 91-004
OPERATION ABOVE MAXIMUM LAKE TEMPERATURE OF CONTAINMENT SPRAY DUE TO INADEQUATE DESIGN BASIS AND CONFIGURATION CONTROL.
EVENT DATE: 031591 REPORT DATE: 041591 NSSS: GE TYPE: BWR

(NSIC 221915) ON MARCH 15, 1991, WHILE NINE MILE POINT UNIT 1 (NMP1) WAS IN COLD SHUTDOWN IT WAS DETERMINED THAT NMP1 HAS OPERATED IN THE PAST WITH LAKE TEMPERATURES ABOVE THE MAXIMUM ALLOWABLE LAKE TEMPERATURE. ALSO, THE VALUES FOR MAXIMUM LAKE TEMPERATURE, FLOW RATE AND FOULING FACTOR USED FOR CALCULATING THE HEAT REMOVAL CAPACITY OF THE CONTAINMENT SPRAY HEAT EXCHANGERS MAY HAVE BEEN NON-CONSERVATIVE. THE CAUSE OF THE EVENTS WERE STANDARDS, POLICIES AND ADMINISTRATIVE CONTROLS THAT IN THE PAST, WERE LESS THAN ADEQUATE TO IMPLEMENTING DESIGN BASIS INFORMATION AND CONFIGURATION CONTROL. ACTIONS TAKEN WERE TO WRITE AND ISSUE A TECHNICAL SPECIFICATION INTERPRETATION ON MAXIMUM LAKE TEMPERATURE AND THE REQUIRED ACTION SHOULD THAT TEMPERATURE LIMIT BE EXCEEDED. PROCEDURES AND INSTRUCTIONS WERE REVISED TO REFLECT THIS LIMIT. SHIFT BRIEFINGS WERE USED TO INFORM CONTROL ROOM OPERATORS OF THE NEW MAXIMUM LAKE TEMPERATURE. THE LONG TERM CORRECTIVE ACTION WILL BE TO COMPLETE THE DESIGN BASIS RECONSTITUTION FOR THE CONTAINMENT SPRAY SYSTEM AND TO INCORPORATE THAT BASIS IN THE REQUIRED DOCUMENTS. THIS WILL ENSURE THAT ALL OF THE REQUIRED VALUES FOR THE CONTAINMENT SPRAY SYSTEM ARE KNOWN AND ARE BEING MET.

[76]   NINE MILE POINT 2   DOCKET 50-410   LER 90-025
REACTOR BUILDING ISOLATION DUE TO LOW EXHAUST VENTILATION FLOW CAUSED BY PERSONNEL ERROR.
EVENT DATE: 121490   REPORT DATE: 011091   NSSS: GE   TYPE: BWR

(NSIC 221502) AT APPROXIMATELY 1200 HOURS ON DECEMBER 14, 1990, NINE MILE POINT UNIT 2 (NMP2) EXPERIENCED AN ACTUATION OF AN ENGINEERED SAFETY FEATURE. SPECIFICALLY, THE SECONDARY CONTAINMENT AUTOMATICALLY ISOLATED WHEN AN OPERATOR MISTAKENLY SECURED THE REACTOR BUILDING VENTILATION SYSTEM ABOVE REFUEL FLOOR EXHAUST FAN 2HVR-FN5B DURING THE PERFORMANCE OF OPERATIONS SURVEILLANCE PROCEDURE N2-OSP-HVR-Q002, "REACTOR BUILDING VENTILATION SYSTEM AUTOMATIC ISOLATION DAMPER OPERABILITY TEST". AT THE TIME OF THE EVENT, THE REACTOR MODE SWITCH WAS IN THE REFUEL POSITION (MODE 5). THE ROOT CAUSE FOR THIS EVENT INCLUDED: 1) SUSPENSION OF THE PERFORMANCE OF PROCEDURE N2-OSP-HVR-Q002; AND 2) RESTORATION OF REACTOR BUILDING NORMAL VENTILATION. ADDITIONALLY, THE NONLICENSED OPERATOR INVOLVED WAS SUSPENDED FROM PERFORMING TASKS OTHER THAN OPERATOR ROUNDS UNTIL AN EVALUATION OF HIS PERFORMANCE IS COMPLETED BY OPERATIONS MANAGEMENT.

[77]   NINE MILE POINT 2   DOCKET 50-410   LER 91-004
FAILURE TO RESTORE STRUCTURAL INTEGRITY OF AN ASME COMPONENT DUE TO INADEQUATE WRITTEN COMMUNICATION.
EVENT DATE: 032291   REPORT DATE: 042291   NSSS: GE   TYPE: BWR

(NSIC 221871) ON MARCH 22, 1991, IT WAS DETERMINED THAT NINE MILE POINT UNIT 2 (NMP2) HAD NOT BEEN IN COMPLIANCE WITH THE PLANT'S TECHNICAL SPECIFICATION (T.S.) REQUIREMENTS FOLLOWING A REPORTED REACTOR CORE ISOLATION COOLING SYSTEM (ICS) DRAIN LINE LEAK. SPECIFICALLY, THE T.S. REQUIRED ACTIONS OF RESTORING THE STRUCTURAL INTEGRITY OF THE AFFECTED ASME III CODE CLASS 2 COMPONENT TO WITHIN ITS LIMITS OR REMOVING IT FROM SERVICE WERE NOT COMPLETED. AT THE TIME THE LEAK WAS DISCOVERED (JUNE 6, 1990), NO IMMEDIATE ACTIONS WERE TAKEN TO EITHER ISOLATE THE SYSTEM LEAK OR REPAIR THE FAILED COMPONENT. THE ROOT CAUSE FOR THIS T.S. VIOLATION WAS DETERMINED TO BE INADEQUATE WRITTEN COMMUNICATION. A CONTRIBUTING CAUSE WAS TRAINING METHODS. REPAIR ACTIVITY CORRECTIVE ACTIONS INCLUDED: GENERATING A PROBLEM REPORT; ISSUING AN OCCURRENCE REPORT; AND COMPLETING DEFECTIVE WELD REPAIRS. ADDITIONAL CORRECTIVE ACTIONS INCLUDE: REVISING ADMINISTRATIVE PROCEDURES; ISSUING A LESSONS LEARNED TRANSMITTAL; AND PROVIDING TECHNICAL TRAINING.

[78]   NINE MILE POINT 2   DOCKET 50-410   LER 91-005
ESF ACTUATION DUE TO LOW FLOW SIGNAL CAUSED BY PROCEDURAL DEFICIENCY.
EVENT DATE: 032691   REPORT DATE: 042391   NSSS: GE   TYPE: BWR

(NSIC 221872) ON MARCH 26, 1991 AT 0755 HOURS, NINE MILE POINT UNIT 2 EXPERIENCED AN ACTUATION OF AN ENGINEERED SAFETY FEATURE (ESF). SPECIFICALLY, THE SECONDARY CONTAINMENT (REACTOR BUILDING) ISOLATED AND AN AUTOMATIC START OF THE REACTOR BUILDING EMERGENCY RECIRCULATION UNIT COOLER AND THE STANDBY GAS TREATMENT SYSTEM

(GTS) WAS EXPERIENCED. THE ACTUATION WAS INITIATED BY A LOW AIR FLOW SIGNAL IN THE REACTOR BUILDING VENTILATION SYSTEM (HVR). AT THE TIME OF THE EVENT, THE REACTOR MODE SWITCH WAS IN THE "RUN" POSITION (MODE 1) WITH THE REACTOR OPERATING AT 100% RATED THERMAL POWER. THE ROOT CAUSE OF THE EVENT HAS BEEN DETERMINED TO BE INADEQUATE WRITTEN COMMUNICATION (DEFICIENT PROCEDURAL STEP). THE IMMEDIATE CORRECTIVE ACTIONS WERE TO DETERMINE THE CAUSE OF THE ESF ACTUATION, AND RESTORE THE HVR SYSTEM TO ITS NORMAL OPERATIONAL STATUS. OTHER CORRECTIVE ACTIONS INCLUDE REVISING PROCEDURES, ISSUING A LESSONS LEARNED TRANSMITTAL, AND FURTHER INVESTIGATION INTO THE NUMBER OF ESF ACTUATIONS ATTRIBUTED TO THIS SYSTEM.

[79]   NINE MILE POINT 2   DOCKET 50-410   LER 91-006
UNUSUAL EVENT CLASSIFICATION AND REACTOR SHUTDOWN DUE TO AN UNISOLABLE REACTOR COOLANT SYSTEM PRESSURE BOUNDARY LEAK.
EVENT DATE: 033091   REPORT DATE: 042991   NSSS: GE   TYPE: BWR
VENDOR: M[illegible] BELLOWS

(NSIC 221964) ON MARCH 30, 1991, WHILE PERFORMING A DRYWELL INSPECTION TO IDENTIFY THE CAUSE FOR AN INCREASED LEAKAGE RATE FROM THE PLANT'S PRIMARY CONTAINMENT DRYWELL FLOOR DRAINS, AN UNISOLABLE REACTOR COOLANT SYSTEM PRESSURE BOUNDARY LEAK WAS DISCOVERED. AT 140[illegible] HOURS, AN UNUSUAL EVENT EMERGENCY CLASSIFICATION WAS DECLARED AND OPERATORS COMMENCED PLANT SHUTDOWN. NINE MILE POINT UNIT 2 (NMP2) WAS OPERATING AT A POWER LEVEL OF APPROXIMATELY 14 PERCENT WITH THE MODE SWITCH IN THE "RUN" POSITION (OPERATIONAL CONDITION 1) AT THE TIME THE EVENT WAS DECLARED. THE CAUSE FOR THE STAINLESS STEEL FLEX HOSE FAILURE IS UNDER INVESTIGATION AND WILL BE SUBMITTED AS A SUPPLEMENT TO THIS LER. CORRECTIVE ACTIONS INCLUDE: GENERATING A MODIFICATION TO REPAIR/REVISE SAMPLE SYSTEM PIPING; REVIEWING SIMILAR PLANT FLEX HOSE APPLICATIONS; AND PERFORMING A FAILURE ANALYSIS ON THE DAMAGED FLEX HOSE.

[80]   NORTH ANNA 1   DOCKET 50-338   LER 91-006
MISSED AC OFFSITE POWER SOURCE SURVEILLANCE DUE TO PERSONNEL ERROR.
EVENT DATE: 040391   REPORT DATE: 041891   NSSS: WE   TYPE: PWR

(NSIC 221944) AT 1440 HOURS, ON APRIL 3, 1991, WITH UNIT 1 OPERATING AT 99.2% POWER (MODE 1), AND 1J DIESEL GENERATOR TAGGED OUT FOR PRE-PLANNED MAINTENANCE, IT WAS DISCOVERED THAT AN 8 HOUR TECHNICAL SPECIFICATION (TS) SURVEILLANCE HAD NOT BEEN PERFORMED WITHIN THE ALLOWED INTERVAL. TECHNICAL SPECIFICATION 3.8.1.1, ACTION STATEMENT (B), REQUIRES THAT WHENEVER ONE DIESEL GENERATOR IS DECLARED INOPERABLE, THE OPERABILITY OF THE A.C. OFF-SITE POWER SOURCES MUST BE VERIFIED WITHIN ONE HOUR, AND AT LEAST ONCE PER 8 HOURS THEREAFTER. THIS EVENT IS REPORTABLE, PURSUANT TO 10 CFR 50.73 (A) (2) (I) (B) AS A CONDITION PROHIBITED BY TECHNICAL SPECIFICATIONS 4.0.3 AND 3.8.1.1. THE CAUSE OF THE EVENT WHICH RESULTED IN THE FAILURE TO PERFORM THE REQUIRED SURVEILLANCE WAS PERSONNEL ERROR. FOLLOWING DISCOVERY OF THE OMISSION AT 1440 HOURS, OPERATIONS IMMEDIATELY INITIATED THE OFFSITE A.C. POWER SOURCE VERIFICATION AND SATISFACTORILY COMPLETED THE SURVEILLANCE AT 1455 HOURS. THIS EVENT POSED NO SIGNIFICANT SAFETY IMPLICATIONS SINCE OFF-SITE POWER SOURCES REMAINED OPERABLE DURING THE PERIOD, AS EVIDENCED BY SUCCESSFUL COMPLETION OF THE A.C. OFF-SITE POWER SOURCES SURVEILLANCE PROCEDURE. ALARMS AND INDICATION WERE AVAILABLE TO CONTROL ROOM PERSONNEL TO PROVIDE STATUS OF ELECTRICAL DISTRIBUTION SYSTEM AT ALL TIMES. THEREFORE, THE HEALTH AND SAFETY OF THE GENERAL PUBLIC WAS NOT AFFECTED AT ANY TIME DURING THIS EVENT.

[81]   NORTH ANNA 2   DOCKET 50-339   LER 91-001
POWER OPERATED RELIEF VALVE CONTROL CIRCUITRY MISSED SURVEILLANCE DUE TO INCORRECT TECHNICAL SPECIFICATION INTERPRETATION AND ADMINISTRATIVE ERROR.
EVENT DATE: 031891   REPORT DATE: 041291   NSSS: WE   TYPE: PWR

(NSIC 221945) ON 3/18/91, AT 1714 HOURS, WITH UNIT 2 OPERATING AT 100% POWER (MODE 1), IT WAS DETERMINED THAT A SET OF CONTACTS AND ASSOCIATED WIRING ON THE CONTROL ROOM BENCH BOARD SWITCH FOR THE TRAIN "A" POWER OPERATED RELIEF VALVE (PORV) OVER PRESSURE CONTROL CIRCUITRY HAD NOT BEEN TESTED AS REQUIRED BY TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENT 4.4.3.2.1.B. THIS EVENT IS

REPORTABLE PURSUANT TO 10 CFR 50.73(A)(2)(I)(B) FOR A CONDITION THAT IS PROHIBITED BY THE TECHNICAL SPECIFICATIONS. THE CAUSE OF THE EVENT WAS THE INCORRECT INTERPRETATION OF TECHNICAL SPECIFICATION 4.4.3.2.1.B. PREVIOUS INTERPRETATIONS DID NOT REQUIRE TESTING OF THE CONTACTS AND ASSOCIATED WIRING FOR THE PORV CONTROL CIRCUITRY. AS AN IMMEDIATE CORRECTIVE ACTION, THE ACTION STATEMENT OF TECHNICAL SPECIFICATION 3.4.3.2 WAS ENTERED AND PCV-2455C WAS DECLARED INOPERABLE AND ITS ASSOCIATED BLOCK VALVE WAS SHUT. THE APPROPRIATE PROCEDURE WAS TEMPORARILY REVISED TO FUNCTIONALLY TEST THE CONTACTS AND ASSOCIATED WIRING AND TESTING WAS SATISFACTORILY PERFORMED. AS AN ADDITIONAL CORRECTIVE ACTION, APPROPRIATE PROCEDURES WILL BE REVISED TO ENSURE EACH CONTACT AND ASSOCIATED WIRING IN THE PORV CONTROL CIRCUITRY IS ADEQUATELY TESTED. THIS EVENT DID NOT POSE ANY SIGNIFICANT SAFETY IMPLICATIONS.

[82] OCONEE 1 DOCKET 50-269 LER 90-015
UNIT OPERATION IN AN UNANALYZED CONDITION DUE TO DESIGN DEFICIENCY, DESIGN OVERSIGHT.
EVENT DATE: 111990 REPORT DATE: 122090 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: OCONEE 2 (PWR)
OCONEE 3 (PWR)

(NSIC 221484) ON 11/16/90, WITH ALL 3 UNITS AT 100% FULL POWER, A DESIGN ENGINEER DISCOVERED THAT THE REQUIREMENTS SPECIFIED BY TECH SPEC 3.3.1 FOR HIGH PRESSURE INJECTION SYSTEM (HPI) OPERATION BELOW 60% FULL POWER COULD POTENTIALLY RESULT IN INSUFFICIENT EMERGENCY CORE COOLING SYSTEM FLOW FOR CERTAIN HPI LINE BREAK ASSUMPTIONS. THE DESIGN ENGINEER, WHILE REVIEWING HPI SYSTEM OPERATION IN ORDER TO RESPOND TO QUESTIONS RAISED ON ANOTHER ISSUE, REALIZED THAT IF AN HPI LINE BROKE AT THE REACTOR COOLANT SYSTEM INJECTION NOZZLE, LESS FLOW WOULD REACH THE REACTOR COOLANT SYSTEM THAN HAD BEEN PREVIOUSLY ANALYZED. DESIGN ENGINEERING CONTACTED BABCOCK AND WILCOX (B&W) AND REQUESTED THEM TO ASSESS THE POTENTIAL FOR REDUCED HPI SYSTEM FLOW DURING THE POSTULATED LINE BREAK. ON 11/19/90, AT 1600 HOURS, B&W CONFIRMED THAT THE CURRENT TECH SPEC ALLOWANCES FOR OPERATION BELOW 60% FULL POWER ARE INADEQUATE UNDER THE ASSUMED CONDITIONS. ADDITIONAL REQUIREMENTS FOR HPI SYSTEM OPERATION BELOW 60% FULL POWER WERE INITIATED TO CONSERVATIVELY ENSURE ADEQUATE HPI FLOW. BECAUSE THE ORIGINAL EVALUATION OF THE EMERGENCY CORE COOLING SYSTEM DID NOT IDENTIFY THE CONSEQUENCES OF THIS POTENTIAL HPI LINE BREAK, THIS EVENT IS ASSIGNED A ROOT CAUSE OF DESIGN DEFICIENCY, UNANTICIPATED INTERACTION OF SYSTEMS, DESIGN OVERSIGHT.

[83] OCONEE 3 DOCKET 50-287 LER 91-004
INAPPROPRIATE ACTION, FAILURE TO FOLLOW PROCEDURE, DURING A NUCLEAR STATION MODIFICATION IMPLEMENTATION RESULTS IN A DEGRADED FIRE BARRIER.
EVENT DATE: 020891 REPORT DATE: 041991 NSSS: BW TYPE: PWR

(NSIC 221928) ON 2/8/91, DURING THE IMPLEMENTATION OF A NUCLEAR STATION MODIFICATION (NSM), A WORK CREW INSTALLED 3/4 INCH CONDUIT THROUGH THE FIREWALL SEPARATING THE EAST AND WEST PENETRATION ROOMS FOR UNIT 3. THE BREACH WAS NOT SEALED NOR WAS A FIRE WATCH ESTABLISHED AS REQUIRED BY TECH SPECS. THE CREW HAD MISTAKENLY ASSUMED THE WALL WAS NOT A FIREWALL AND FAILED TO FOLLOW THE IMPLEMENTATION PROCEDURE FOR THE NSM. ON 3/20/91 , AT 0800 HOURS, A TECHNICAL SUPPORT LEADER WAS REVIEWING THE PAPERWORK DUE TO QUESTIONS BY ONE OF THE CREW MEMBERS CONCERNING THE UNSEALED PENETRATION. IT WAS DETERMINED THAT THE BREACH HAD INDEED BEEN MADE TO A FIREWALL WITHOUT THE REQUIRED COMPENSATORY ACTIONS. THEY THEN INITIATED ACTION TO GET THE BREACH SEALED. THE REPAIRS WERE COMPLETED THAT AFTERNOON AND THE FIRE BARRIER WAS DECLARED OPERABLE AT 1950 HOURS. UNIT 3 WAS AT 100% FULL POWER WHEN THE INCIDENT OCCURRED AND REMAINED THERE UNTIL IT WAS SHUTDOWN FOR A SCHEDULED REFUELING OUTAGE ON 2/13/91. THE UNIT REACHED COLD SHUTDOWN ON 2/15/91, AT 1235 HOURS. THE ROOT CAUSE OF THIS INCIDENT WAS ASSIGNED INAPPROPRIATE ACTION, FAILURE OF THE WORK CREW TO FOLLOW PROCEDURE. A CONTRIBUTING CAUSE WAS MANAGEMENT DEFICIENCY, INADEQUATE PLANNING. CORRECTIVE ACTIONS INCLUDED REPAIRING THE BREACHED FIRE BARRIER, COUNSELING OF THE INDIVIDUALS INVOLVED, AND PROCEDURE REVISION.

[84] OCONEE 3 DOCKET 50-287 LER 91-003
HIGH PRESSURE INJECTION SYSTEM CROSSOVER FLOW TRANSMITTER INOPERABLE DUE TO INSTALLATION DEFICIENCY.
EVENT DATE: 031991 REPORT DATE: 041791 NSSS: BW TYPE: PWR
VENDOR: ROSEMOUNT, INC.

(NSIC 221920) ON MARCH 19, 1991, WITH UNIT 3 IN A REFUELING OUTAGE, IT WAS DISCOVERED THAT THE IMPULSE LINES TO THE HIGH PRESSURE INJECTION (HPI) LOOP 'B' CROSSOVER FLOW INSTRUMENTATION HAD BEEN INSTALLED INCORRECTLY. THE CROSSOVER FLOWPATH CONSISTS OF PIPING AND ASSOCIATED VALVES CONNECTING AT EACH OF THE HPI PUMP NORMAL DISCHARGE HEADERS. THIS PROVIDES A PARALLEL FLOWPATH, BYPASSING THE INSTALLED THROTTLE VALVES, FOR PROTECTION AGAINST A SINGLE FAILURE. A NUCLEAR STATION MODIFICATION (NSM) HAD BEEN IMPLEMENTED TO REPLACE THE CROSSOVER FLOW INSTRUMENTATION WITH A MORE ACCURATE TRANSMITTER. DURING THE POST MODIFICATION TESTING (PMT), NO FLOW WAS INDICATED BY THE FLOW TRANSMITTER. THIS LED TO THE DISCOVERY THAT THE IMPULSE LINES WERE CROSSED AND HAD BEEN SINCE 1984. AFTER A 1984 NSM WAS COMPLETED, WHICH REPLACED THE FLOW INSTRUMENTATION AND THE IMPULSE LINES, A LESS THAN ADEQUATE PMT WAS PERFORMED. THIS INSTRUMENTATION IS NECESSARY IN ORDER TO MONITOR FLOW RATES THROUGH THE HPI CROSSOVER LINE. THE CROSSED IMPULSE LINES WERE CORRECTED AND ANOTHER FUNCTIONAL TEST WAS PERFORMED THAT VERIFIED OPERABILITY OF THE FLOW TRANSMITTER. THIS EVENT IS DESIGNED A ROOT CAUSE OF INSTALLATION DEFICIENCY. THE IMPULSE LINES TO THE CORRESPONDING FLOW INSTRUMENTATION ON THE OTHER TWO OCONEE UNITS WERE INSPECTED AND VERIFIED TO BE INSTALLED CORRECTLY.

[85] OCONEE 3 DOCKET 50-287 LER 91-005
DESIGN DEFICIENCY AND PROCEDURE DEFICIENCY CAUSE SPURIOUS ANTICIPATED TRANSIENT WITHOUT SCRAM SYSTEM ACTUATION RESULTING IN MANUAL REACTOR TRIP.
EVENT DATE: 040191 REPORT DATE: 050191 NSSS: BW TYPE: PWR
VENDOR: SQUARE D COMPANY

(NSIC 221929) ON 4/1/91, AT 0415 HOURS, ONE CHANNEL OF THE UNIT 3 DIVERSE SCRAM SYSTEM (DSS), AN ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS) MITIGATION SYSTEM, SPURIOUSLY ACTUATED AND BOTH CHANNELS WERE BYPASSED. AFTER SOME TROUBLESHOOTING VERIFIED THE INPUT INSTRUMENT WAS NOT AT FAULT, THE OPERATIONS UNIT SUPERVISOR ATTEMPTED TO RETURN BOTH CHANNELS TO SERVICE, WHILE THE UNIT WAS OPERATING AT 87% FULL POWER. AS HE DID SO, AT 1119 HOURS, SPURIOUS SIGNALS ACTUATED BOTH CHANNELS, RESULTING IN THREE GROUPS OF CONTROL RODS AUTOMATICALLY DROPPING INTO THE CORE. PER PROCEDURE, THE CONTROL ROOM OPERATOR MANUALLY TRIPPED THE REACTOR IN RESPONSE TO THE DROPPED RODS. THE TRIP RESPONSE WAS NORMAL AND THE UNIT WAS STABILIZED, WITH NO SAFETY SYSTEM ACTUATIONS EITHER REQUIRED OR RECEIVED. TROUBLESHOOTING DISCOVERED THAT ONE CHANNEL HAD ACTUATED DUE TO EQUIPMENT FAILURE OF A BAD CIRCUIT BOARD (CONTRIBUTING CAUSE) AND THE SECOND ACTUATED DUE TO A DESIGN DEFICIENCY (ROOT CAUSE) WHICH GENERATED A HIGH SIGNAL SPIKE IN THE SECOND CHANNEL WHEN RETURNING THE FIRST CHANNEL TO SERVICE. A SECOND ROOT CAUSE OF DEFICIENT PROCEDURE WAS ASSIGNED DUE TO LESS THAN ADEQUATE STEPS TO RESET THE SYSTEM. CORRECTIVE ACTIONS INCLUDED REPLACING THE BAD CIRCUIT BOARD, MODIFYING THE SYSTEM, AND ENHANCING PROCEDURES.

[86] PALISADES DOCKET 50-255 LER 90-021
DISCREPANCY IN SAFETY INJECTION TANK LEVEL SWITCH SETTINGS.
EVENT DATE: 111690 REPORT DATE: 122090 NSSS: CE TYPE: PWR

(NSIC 221482) ON 11/16/90, THE PLANT WAS IN COLD SHUTDOWN AND THE CORE WAS TOTALLY DEFUELED FOR THE REPLACEMENT OF THE STEAM GENERATORS. DURING A PREVIOUS ROUTINE SAFETY INSPECTION (90-018), THE RESIDENT INSPECTOR HAD QUESTIONED WHY THE SAFETY INJECTION TANKS (T82A-D) LOWER AND UPPER LEVEL SWITCHES, WHICH ARE SET TO ACTUATE AT THE TECHNICAL SPECIFICATIONS LIMITS, WERE NOT UNDER A CALIBRATION PROGRAM. IT HAD BEEN ASSUMED THAT THE SWITCHES COULD ONLY BE SET AT ONE POSITION TO ACTUATE AND THAT NO DRIFT IN THE SET POINT COULD OCCUR. DURING THE PLANT FOLLOW-UP INVESTIGATION TO ASSURE THAT THE LEVEL SWITCHES WERE ACTUATING AT THE PROPER TANK LEVEL, IT WAS DISCOVERED THAT THE LEVEL SWITCHES HAVE THE POTENTIAL TO DRIFT AS MUCH AS TWO INCHES. THUS IF AN INSTRUMENT DRIFT HAD OCCURRED WHILE THE SWITCHES WERE SET AT THE TECH SPEC LIMITS, THE TECH SPECS FOR SAFETY

INJECTION TANK LEVEL COULD HAVE BEEN EXCEEDED WITHOUT AN ASSOCIATED ALARM; AND THE PLANT WOULD HAVE BEEN IN A CONDITION THAT IS OUTSIDE THE TECHNICAL SPECIFICATIONS. NO IMMEDIATE CORRECTIVE ACTIONS WERE REQUIRED AS THE PLANT WAS SHUTDOWN FOR REFUELING AND STEAM GENERATOR REPLACEMENT. THIS EVENT IS ATTRIBUTABLE TO A LACK OF KNOWLEDGE BY THE PLANT PERSONNEL ON SET POINT METHODOLOGY AS APPLIED TO THESE PARTICULAR SWITCHES. A TECHNICAL SPECIFICATIONS CHANGE WAS SUBMITTED ON 12/7/90 TO INCREASE THE OPERATING BAND BETWEEN THE UPPER AND LOWER TECH SPECS SAFETY INJECTION TANK LEVEL LIMITS.

[87] PALO VERDE 1 DOCKET 50-528 LER 91-002 REV 01
UPDATE ON SURVEILLANCE PERFORMANCE PROHIBITED BY TECHNICAL SPECIFICATION.
EVENT DATE: 020691 REPORT DATE: 042491 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: PALO VERDE 2 (PWR)
PALO VERDE 3 (PWR)

(NSIC 221883) ON 2/6/91, AT APPROX. 1200 MST, PALO VERDE UNIT 1 WAS IN MODE 5 (COLD SHUTDOWN) AND PALO VERDE UNITS 2 AND 3 WERE OPERATING AT APPROXIMATELY 100% POWER WHEN COMPLIANCE PERSONNEL DETERMINED THAT THE 18 MONTH TECH SPEC (TS) REQUIRED SURVEILLANCES TO DEMONSTRATE OPERABILITY OF EACH EMERGENCY DIESEL GENERATOR (EDG) WERE CONDUCTED DURING UNIT OPERATION. THIS WAS CONTRARY TO THE TS REQUIREMENT THAT THE SURVEILLANCES BE PERFORMED DURING SHUTDOWN. THE CAUSE OF THE EVENT WAS A MISINTERPRETATION OF THE BASIS FOR PERFORMING THE TS REQUIRED SURVEILLANCE WHILE SHUTDOWN. THE PHRASE "DURING SHUTDOWN" WAS INCORRECTLY INTERPRETATED AS ONLY APPLYING TO THE SURVEILLANCE REQUIREMENTS WHICH COULD NOT BE PERFORMED WITHIN THE 72 HOUR ACTION PERIOD FOR THE EDG BEING OUT OF SERVICE. THE EVENT WAS DISCUSSED WITH RESPONSIBLE PVNGS MANAGEMENT AND A MEMORANDUM WAS ISSUED TO ENSURE THAT TS REQUIRED SURVEILLANCES WOULD ONLY BE PERFORMED DURING MODES 4 (HOT SHUTDOWN), 5, OR 6 (REFUELING - INCLUDING CORE DEFUELED) WHEN TS STATES "DURING SHUTDOWN." THERE HAVE BEEN NO PREVIOUS SIMILAR EVENTS REPORTED PURSUANT TO 10CFR50.73.

[88] PALO VERDE 1 DOCKET 50-528 LER 91-004
ESF ACTUATION DUE TO LOSS OF POWER TO 4.16 KV BUS.
EVENT DATE: 032091 REPORT DATE: 041991 NSSS: CE TYPE: PWR

(NSIC 221967) AT APPROX. 1414 MST ON 3/20/91, PALO VERDE UNIT 1 WAS IN MODE 1 (POWER OPERATION) AT APPROX. 100% POWER WHEN A SUPPLY BREAKER OPENED BETWEEN THE UNIT 1 NON-CLASS 1E 13.8 KV SWITCHGEAR BUSSES RESULTING IN A LOSS OF POWER TO THE UNIT 1 TRAIN "A" CLASS 1E 4.16 KV BUS. THIS RESULTED IN A LOSS OF POWER (LOP) ENGINEERED SAFETY FEATURE (ESF) SIGNAL BEING GENERATED. THE ESF SIGNAL RESULTED IN AUTOMATIC LOAD SHED OF THE CLASS 1E BUS AND STARTED THE TRAIN "A" EMERGENCY DIESEL GENERATOR (DG). THE DG STARTED AND ASSUMED THE LOADS AS DESIGNED. ALL EQUIPMENT FUNCTIONED AS DESIGNED. NO OTHER ESF PROTECTION SIGNALS WERE ACTIVATED AND NONE WERE REQUIRED. UNIT 1 CONTINUED TO OPERATE NORMALLY AT 100% POWER THROUGHOUT THE EVENT. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR BY AN ELECTRICAL MAINTENANCE WORKER WHO WAS PERFORMING PERIODIC MAINTENANCE ON THE NON-CLASS 1E 13.8 KV SWITCHGEAR. THE INDIVIDUAL INADVERTENTLY BRUSHED THE BREAKER TRIP RELAY WHICH RESULTED IN THE ACTUATION. THERE HAVE BEEN NO PREVIOUS SIMILAR EVENTS REPORTED PURSUANT TO 10CFR50.73.

[89] PEACH BOTTOM 2 DOCKET 50-277 LER 90-027 REV 01
UPDATE ON REACTOR WATER CLEANUP ISOLATION AND STANDBY GAS TREATMENT SYSTEM ACTUATION DUE TO LIGHTNING STRIKE.
EVENT DATE: 091690 REPORT DATE: 041691 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: PEACH BOTTOM 3 (BWR)

(NSIC 221923) ON 9/16/90, AT 1805 HOURS, ACTUATIONS OCCURRED ON BOTH THE UNIT 2 AND UNIT 3 PRIMARY CONTAINMENT ISOLATION SYSTEMS (PCIS) DUE TO A LIGHTNING STRIKE WHICH CAUSED THE OPENING OF THE UNIT 3 STARTUP FEED BREAKER. UNIT 2 AND UNIT 3 EMERGENCY BUSSES ASSOCIATED WITH THE UNIT 3 STARTUP FEED TRANSFERRED TO THE ALTERNATE SUPPLY FOLLOWING THE LOSS OF THE FEED AS DESIGNED. THE FAST TRANSFER RESULTED IN ISOLATION OF THE REACTOR WATER CLEANUP (RWCU) SUCTION OUTBOARD ISOLATION VALVES ON BOTH UNITS. ON UNIT 2, THE "B" REACTOR PROTECTION SYSTEM

MOTOR-GENERATOR SET FAILED TO RESTART WHEN POWER WAS RESTORED RESULTING IN 1/2 OF A GROUP III ISOLATION, WHICH CAUSED THE "B" TRAIN OF THE STANDBY GAS TREATMENT ISOLATION SYSTEM TO START ALONG WITH THE LINEUP OF THE APPROPRIATE OUTBOARD ISOLATION VALVES. NO ACTUAL SAFETY CONSEQUENCES OCCURRED AS A RESULT OF THIS EVENT. RWCU WAS OUT OF SERVICE FOR A TOTAL OF APPROXIMATELY 15 MINUTES, WHICH DID NOT PRESENT A REACTOR WATER CHEMISTRY CONCERN. FOLLOWING THE EVENT THE UNIT 3 STARTUP FEED BREAKER WAS CLOSED. THE EMERGENCY BUSSES WERE RETURNED TO THEIR NORMAL FEEDS. THE ISOLATIONS WERE RESET, THE RWCU SYSTEMS WERE PLACED BACK INTO SERVICE, AND THE VENTILATION SYSTEMS AND THE SBGT SYSTEM WERE NORMALIZED. TWO PREVIOUS SIMILAR LERS HAVE BEEN IDENTIFIED.

[90] PEACH BOTTOM 3 DOCKET 50-278 LER 91-004
PRIMARY CONTAINMENT ISOLATION VALVE LOGGING NOT PERFORMED AS REQUIRED BY TECHNICAL SPECIFICATIONS DUE TO PERSONNEL ERROR.
EVENT DATE: 032491 REPORT DATE: 042491 NSSS: GE TYPE: BWR

(NSIC 221899) ON 3/24/91 AND 3/25/91, A TECHNICAL SPECIFICATION (TECH SPEC) VIOLATION OCCURRED WHEN THE UNIT 3 REACTOR OPERATOR (RO) FAILED TO INITIAL SURVEILLANCE TEST (ST) 5.3, "INOPERABLE ISOLATION VALVE POSITION DAILY LOG", SIGNIFYING THAT HE HAD VERIFIED CONTAINMENT PENETRATIONS CONTAINING INOPERABLE ISOLATION VALVES WERE ISOLATED AS REQUIRED BY TECH SPECS. THE CAUSE OF THIS EVENT IS PERSONNEL ERROR DUE TO FAILURE FOLLOW PROCEDURE. ST 5.3 REQUIRES THE RO TO INITIAL DAILY THAT HE HAS VERIFIED THE PENETRATION IS ISOLATED. CORRECTIVE ACTIONS INCLUDE ROUTING THE PERTINENT INFORMATION FROM THIS LER TO THE APPROPRIATE OPERATIONS PERSONNEL AND REVIEWING THE ON THE JOB PORTION OF THE LICENSED OPERATOR TRAINING PROGRAM. THE RO INVOLVED IN THIS EVENT WAS COUNSELLED AND COACHED BY SHIFT MANAGEMENT FOLLOWING THE INCIDENT ON THE PERFORMANCE OF ADMINISTRATIVE TASKS ASSOCIATED WITH THE RO'S RESPONSIBILITIES. THERE WERE NO SAFETY CONSEQUENCES AS A RESULT OF THIS EVENT.

[91] PERRY 1 DOCKET 50-440 LER 91-005
MISAPPLICATION OF TECHNICAL SPECIFICATION ACTION DURING ROD CONTROL AND INFORMATION SYSTEM MAINTENANCE RESULTS IN TECHNICAL SPECIFICATION VIOLATION.
EVENT DATE: 012991 REPORT DATE: 022291 NSSS: GE TYPE: BWR

(NSIC 221505) ON JANUARY 29, 1991, AT 2305, MAINTENANCE ACTIVITIES ON THE ROD CONTROL AND INFORMATION (RC&IS) RESULTED IN A CONDITION PROHIBITED BY TECHNICAL SPECIFICATION 3.1.3.3. CONTROL ROOM OPERATORS WERE MADE AWARE THAT THE REPLACEMENT OF POWER SUPPLIES WOULD RESULT IN THE LOSS OF SCRAM ACCUMULATOR FAULT INDICATION AND ALSO THE LOSS OF RC&IS INTERFACE CAPABILITIES AT THE MAIN CONTROL PANEL. DUE TO A MISAPPLICATION OF TECHNICAL SPECIFICATION 3.1.3.3.A.2, CONTROL ROOM OPERATORS BELIEVED THE PLANT WAS IN A 12-HOUR SHUTDOWN LCO ACTION STATEMENT; HOWEVER, THE ON-COMING SHIFT, AFTER REVIEWING THE SITUATION, DETERMINED THAT THE TECHNICAL SPECIFICATION 3.1.3.3.A.2 ACTION COULD NOT BE AND THAT THE PLANT WAS IN TECHNICAL SPECIFICATION LCO 3.0.3. ACTIONS WERE INITIATED IN ACCORDANCE WITH TECHNICAL SPECIFICATION 3.0.3 TO SHUT DOWN THE PLANT AND THEN RC&IS WAS RETURNED TO SERVICE. OPERATORS DECLARED THE SCRAM ACCUMULATORS AND RC&IS OPERABLE AND TECHNICAL SPECIFICATION 3.0.3 WAS EXITED ON JANUARY 30, 1991, AT 0200. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR, INATTENTION TO DETAIL. THE UNIT SUPERVISOR INTERPRETED THE TECHNICAL SPECIFICATIONS AND INCORRECTLY APPLIED THE TWELVE-HOUR SHUTDOWN ACTION REQUIREMENT. TO PREVENT RECURRENCE, THIS EVENT HAS BEEN DISCUSSED IN DETAIL WITH THE CONTROL ROOM OPERATORS INVOLVED.

[92] PILGRIM 1 DOCKET 50-293 LER 91-004
REACTOR CORE ISOLATION COOLING SYSTEM MADE INOPERABLE PER TECHNICAL SPECIFICATIONS DUE TO AN INOPERABLE AREA TEMPERATURE SWITCH.
EVENT DATE: 031991 REPORT DATE: 041291 NSSS: GE TYPE: BWR
VENDOR: WEIDMULLER TERMINATIONS INC.

(NSIC 221930) ON 3/19/91 AT 2345 HOURS, THE REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM WAS MADE INOPERABLE AND A SEVEN DAY TECH SPEC LIMITING CONDITION FOR OPERATION (LCO) BEGAN. THE SYSTEM WAS MADE INOPERABLE DUE TO AN AREA TEMPERATURE SWITCH TERMINAL (WEIDMULLER SA K6N) BEING STRIPPED DURING A QUARTERLY FUNCTIONAL

TEST AND CALIBRATION. THE CAUSE OF THE STRIPPED TERMINAL WAS REPEATED TORQUING OF THE SCREWS. THE TEMPERATURE SWITCHES ARE REPLACED QUARTERLY WITH A CALIBRATED SPARE. THE TEMPERATURE SWITCH LEADS ARE DISCONNECTED FROM THEIR TERMINAL DURING THIS PROCESS. THIS REPEATED TORQUING WEAKENED THE TERMINAL. THE BROKEN TERMINAL WAS REPLACED WITH AN INSTALLED SPARE TERMINAL FROM THE SAME TERMINAL BLOCK. THE BROKEN TERMINAL WAS INSERTED INTO THE SPARE TERMINAL'S SPACE AND A MAINTENANCE REQUEST WAS WRITTEN TO REPLACE THE BROKEN TERMINAL. LONG TERM CORRECTIVE ACTION INCLUDES EVALUATING THE FEASIBILITY OF CHANGING THE CIRCUITRY TO MAKE IT MORE SUITABLE FOR FREQUENT CALIBRATION. FOLLOWING TERMINAL REPLACEMENT, THE CHANNEL PORTION OF THE LOGIC CIRCUITRY WAS SUCCESSFULLY TESTED. RCIC WAS DECLARED OPERABLE AND THE SEVEN DAY LCO WAS TERMINATED ON MARCH 20, 1991 AT 0450 HOURS. THIS CONDITION OCCURRED DURING POWER OPERATION AT 100 PERCENT REACTOR POWER. THE REACTOR MODE SWITCH WAS IN THE RUN POSITION. THIS REPORT IS SUBMITTED IN ACCORDANCE WITH 10 CFR 50.73(A)(2)(V)(D).

[93] PILGRIM 1 DOCKET 50-293 LER 91-005
LOSS OF AC POWER TO 'B' TRAINS OF SAFETY SYSTEMS DUE TO DIESEL GENERATOR 'B' VOLTAGE REGULATOR FAILURE DURING SURVEILLANCE.
EVENT DATE: 032591 REPORT DATE: 042491 NSSS: GE TYPE: BWR
VENDOR: BASLER ELECTRIC COMPANY

(NSIC 221904) ON MARCH 25, 1991 AT 0610 HOURS, THE EMERGENCY DIESEL GENERATOR (EDG) 'B' BECAME INOPERABLE. A LOSS OF AC POWER TO TRAIN 'B' COMPONENTS OF SAFETY SYSTEMS, AND ACTUATIONS OF PORTIONS OF THE PRIMARY CONTAINMENT AND SECONDARY CONTAINMENT ISOLATION CONTROL SYSTEMS OCCURRED DURING A TECHNICAL SPECIFICATION REQUIRED SURVEILLANCE TEST OF THE EDG 'B'. THE CAUSE FOR THE EVENT WAS A FAILURE OF THE AUTOMATIC VOLTAGE REGULATOR OF THE EDG 'B' THAT WAS FULLY LOADED ON ITS SAFETY BUS AT THE TIME OF THE EVENT. THE VOLTAGE REGULATOR WAS MANUFACTURED BY THE BASLER ELECTRIC COMPANY, MODEL NUMBER SVR01A05B2B1B, SERIAL NUMBER 9047500105. THE CAUSE FOR THE VOLTAGE REGULATOR FAILURE HAD NOT BEEN IDENTIFIED WHEN THIS REPORT WAS SUBMITTED. THE MANUFACTURER'S ROOT CAUSE ANALYSIS IS CURRENTLY EXPECTED TO BE COMPLETED BY MAY 31, 1991. THIS REPORT IS EXPECTED TO BE UPDATED BY JULY 30, 1991. THE AFFECTED SAFETY SYSTEM COMPONENTS WERE RE-ENERGIZED AND RETURNED TO NORMAL SERVICE BY MARCH 25, 1991 AT 2100 HOURS. THE EDG 'B' VOLTAGE REGULATOR WAS REPLACED AND THE EDG 'B' WAS SURVEILLANCE TESTED WITH SATISFACTORY RESULTS. THE SURVEILLANCE WAS COMPLETED ON MARCH 28, 1991 AT APPROXIMATELY 0915 HOURS. THIS EVENT OCCURRED DURING POWER OPERATION WHILE AT 100 PERCENT REACTOR POWER. THE REACTOR MODE SELECTOR SWITCH WAS IN THE RUN POSITION.

[94] PILGRIM 1 DOCKET 50-293 LER 91-006
HIGH PRESSURE COOLANT INJECTION AND REACTOR CORE ISOLATION COOLING SYSTEMS BECAME INOPERABLE DUE TO TRIPPED INVERTERS.
EVENT DATE: 032691 REPORT DATE: 042491 NSSS: GE TYPE: BWR
VENDOR: TOPAZ ELECTRONICS

(NSIC 221905) ON MARCH 26, 1991 WHEN STARTING THE 'B' REACTOR RECIRCULATION PUMP, THE HIGH PRESSURE COOLANT INJECTION (HPCI) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM INVERTERS TRIPPED ON HIGH VOLTAGE. THE HPCI AND RCIC SYSTEMS WERE INOPERABLE FOR NINE MINUTES. THE CAUSE OF THE INVERTER TRIPS WAS A VOLTAGE FLUCTUATION THAT OCCURRED DURING PUMP START. THE LOAD REQUIRED BY THE PUMP START CAUSED THE BATTERY CHARGER THAT SUPPLIES DC VOLTAGE TO THE INVERTERS TO OVERCOMPENSATE RESULTING IN A VOLTAGE SURGE. THE TRIP SETPOINT OF THE INVERTERS WAS EXCEEDED DURING THIS SURGE. CORRECTIVE ACTION WAS TAKEN TO RESET THE INVERTERS. AN ENGINEERING EVALUATION HAS BEEN INITIATED TO INVESTIGATE ENHANCEMENTS TO PRECLUDE THE INVERTERS FROM TRIPPING AS A RESULT OF LARGE PUMP STARTS. INTERIM MEASURES WILL INCLUDE AN ADMINISTRATIVE CHANGE TO CAUTION OPERATIONS PERSONNEL THAT THE POTENTIAL FOR INVERTER TRIPS EXISTS. THE INVERTERS WERE MANUFACTURED BY TOPAZ ELECTRONICS, MODEL NO. 125-GW-125 (60). THE EVENT OCCURRED AT POWER OPERATION WITH THE REACTOR MODE SELECTOR SWITCH IN THE RUN POSITION. THE REACTOR VESSEL (RV) PRESSURE WAS APPROXIMATELY 956 PSIG AND THE RV WATER TEMPERATURE WAS 542 DEGREES FAHRENHEIT. THE REACTOR POWER LEVEL WAS 30 PERCENT. THIS REPORT IS SUBMITTED IN ACCORDANCE WITH 10 CFR 50.73(A)(2)(V)(D) AND THE EVENT POSED NO THREAT TO THE PUBLIC HEALTH AND SAFETY.

[95] PRAIRIE ISLAND 1 DOCKET 50-282 LER 89-010 REV 01
UPDATE ON REACTOR TRIP RESULTING FROM LOSS OF ONE REACTOR COOLANT PUMP DUE TO PERSONNEL ERROR.
EVENT DATE: 072189 REPORT DATE: 050391 NSSS: WE TYPE: PWR

(NSIC 221924) ON JULY 21, 1989 UNIT 1 WAS AT 100% POWER. DURING THE AFTERNOON, A "HOT LACQUER" SMELL WAS NOTICED COMING FROM 4160V BUS 11. 4160V BUS 11 SUPPLIES NO. 11 REACTOR COOLANT PUMP AND NO. 11 FEEDWATER PUMP. THE PROBLEM WAS INVESTIGATED AND DETERMINED TO BE OF NO IMMEDIATE CONCERN BUT WORTHY OF INCREASED AWARENESS. AN "OPERATIONS NOTE" WAS ISSUED TO ALERT OPERATORS OF THE PROBLEM. DURING A SUBSEQUENT INVESTIGATION FOR THE SOURCE OF THE SMELL, AN OPERATOR PULLED OPEN THE POTENTIAL FUSE DRAWER FOR 4160V BUS 11, CAUSING UNDERVOLTAGE RELAYS TO TRIP. AFTER A 5 SECOND TIME DELAY TIMED OUT, THE BREAKER FOR NO. 11 REACTOR COOLANT PUMP TRIPPED AND THE REACTOR TRIPPED AT 2345 ON JULY 21, 1989 DUE TO SINGLE LOOP LOSS OF FLOW REACTOR TRIP SIGNAL. THE UNIT WAS RETURNED TO SERVICE AT 2204 ON JULY 22, 1989. 4160V BUS DOORS HAVE BEEN LABELED, CAUTIONING PERSONNEL OF THE CONSEQUENCES OF OPENING THE POTENTIAL FUSE DRAWERS. POTENTIAL FUSE DRAWER FRONTS WILL ALSO BE LABELED.

[96] PRAIRIE ISLAND 1 DOCKET 50-282 LER 91-002
AUTO-START OF CONTROL ROOM SPECIAL VENTILATION SYSTEM DUE TO SPIKE ON NEWLY INSTALLED RADIATION MONITOR.
EVENT DATE: 032391 REPORT DATE: 042291 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: PRAIRIE ISLAND 2 (PWR)
VENDOR: NUCLEAR MEASUREMENTS CORP.

(NSIC 221902) ON MARCH 23, 1991, BOTH UNITS WERE OPERATING AT FULL POWER. AT 0019 HOURS A SPIKE ON RADIATION MONITOR R-23 CAUSED CONTROL ROOM ANNUNCIATION OF CONTROL ROOM HIGH RADIATION TRAIN A, WHICH RESULTED IN AUTO-START OF NO. 121 CONTROL ROOM CLEANUP FAN AND ISOLATION OF THE CONTROL ROOM OUTSIDE AIR SUPPLY. SEVERAL UNPLANNED ACTUATIONS OF ESF VENTILATION SYSTEMS HAD TAKEN PLACE OVER A PERIOD OF SEVERAL YEARS AS A RESULT OF SPIKING IN THE CIRCUITRY OF RADIATION MONITORS. TO HELP PREVENT RECURRENCES, UPGRADED MONITOR MODULES HAD BEEN ORDERED FROM THE MANUFACTURER. UPGRADED MONITOR MODULES HAD BEEN RECENTLY RECEIVED AND INSTALLED. SPURIOUS HIGH RADIATION ALARMS OCCURRED IN THESE NEW MONITORS IN THE FIRST FEW HOURS OF OPERATION. NO SPURIOUS ALARMS OCCURRED ON ANY OF THE NEW MODULES AFTER 100 HOURS OF OPERATION. IN COMMUNICATION WITH THE MANUFACTURER, IT WAS DETERMINED THAT THE 100-HOUR FACTORY "BURN-IN" TEST SPECIFIED IN THE PURCHASE ORDER HAD NOT BEEN DONE. HAD THE BURN-IN REQUIREMENT BEEN MET, NO SPURIOUS ALARMS WOULD HAVE OCCURRED AFTER INSTALLATION.

[97] PRAIRIE ISLAND 2 DOCKET 50-306 LER 91-001
DISCOVERY THAT CERTAIN PIPING INSPECTIONS ARE NOT BEING MADE AS REQUIRED BY ASME SECTION XI DUE TO PERSONNEL ERROR.
EVENT DATE: 032791 REPORT DATE: 042691 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: PRAIRIE ISLAND 1 (PWR)

(NSIC 221856) ON MARCH 27, 1991, BOTH UNITS WERE OPERATING AT FULL POWER. SP1168.8, COOLING WATER OPERATIONAL PRESSURE TEST, WAS UNDERGOING REVIEW IN PREPARATION FOR ROUTINE PERFORMANCE OF THE PROCEDURE. DURING THE REVIEW, THE SYSTEM ENGINEER NOTICED THAT A PORTION OF THE REQUIRED TESTING WAS MISSING FROM THE PROCEDURE. THE PROCEDURE REQUIRES A WALKDOWN INSPECTION OF COOLING WATER PIPING IN UNIT 1 CONTAINMENT, BUT DOES NOT REQUIRE INSPECTION OF COOLING WATER PIPING IN UNIT 2 CONTAINMENT, AS IT SHOULD. ON MARCH 28, 1991, THE ASME SECTION XI TEST COORDINATOR REVIEWED THE PROGRAM FOR SIMILAR DEFICIENCIES AND FOUND THAT INSPECTION OF AUXILIARY FEED WATER PIPING IN UNIT 2 CONTAINMENT HAD ALSO BEEN OMITTED. THESE TESTS ARE REQUIRED TO BE PERFORMED EVERY 3 1/3 YEARS. AFTER DISCOVERY, WORK REQUESTS WERE WRITTEN TO PERFORM THE OMITTED INSPECTIONS. THESE INSPECTIONS WERE COMPLETED ON APRIL 5, 1991; NO PROBLEMS WERE IDENTIFIED. CAUSE OF THE EVENT WAS PERSONNEL ERROR; THERE WAS INADEQUATE REVIEW OF PROCEDURES. THE PROCEDURE FOR ADMINISTERING THE ASME SECTION XI PROGRAM WILL BE REVISED TO INCLUDE THE REQUIREMENT FOR A SECOND LEVEL REVIEW OF ALL PROGRAM ADDITIONS, DELETIONS AND INTERPRETATIONS.

[98] QUAD CITIES 1 DOCKET 50-254 LER 90-029
TECHNICAL SPECIFICATION LOCAL LEAK RATE TEST LIMIT 0.6LA EXCEEDED WHILE TESTING FEEDWATER CHECK VALVE.
EVENT DATE: 111590 REPORT DATE: 121290 NSSS: GE TYPE: BWR
VENDOR: CRANE VALVE CO.

(NSIC 221480) ON NOVEMBER 12, 1990, QUAD CITIES UNIT ONE WAS SHUTDOWN FOR THE END OF CYCLE 11 REFUELING AND MAINTENANCE OUTAGE. ON NOVEMBER 15, 1990, AT 0700 HOURS WHILE PERFORMING LOCAL LEAK RATE TESTING (LLRT) OF THE DRYWELL PERSONNEL AIR LOCK, IT WAS DETERMINED THAT THE TECHNICAL SPECIFICATION 3.7.A.2.D. LEAKAGE LIMIT OF 18.4 STANDARD CUBIC FEET PER HOUR (SCFH), 0.0375LA, WAS EXCEEDED. ON NOVEMBER 15, 1990, AT 2325 HOURS WHILE PERFORMING LLRT ON THE FEEDWATER CHECK VALVES, VALVE 1-220-62B COULD NOT BE PRESSURIZED. THE TECHNICAL SPECIFICATION 3.7.A.2.A.2, LIMIT OF 293.75 SCFH (0.6LA) WAS EXCEEDED. AN EMERGENCY NOTIFICATION SYSTEM (ENS) PHONE CALL WAS COMPLETED ON NOVEMBER 16, 1990 AT 0318 HOURS IN ACCORDANCE WITH 10CFR50.72(B)(2)(I). THE CAUSE OF THE EXCESSIVE LEAKAGES WILL NOT BE KNOWN UNTIL REPAIRS HAVE BEEN COMPLETED. A SUPPLEMENTAL REPORT WILL BE ISSUED TO ADDRESS THE CAUSES AND CORRECTIVE ACTIONS TAKEN TO BRING THE COMBINED LEAKAGE TO WITHIN TECHNICAL SPECIFICATION LIMITS. THIS REPORT IS BEING SUBMITTED TO COMPLY WITH 10CFR50.73(A)(2)(I)(B).

[99] QUAD CITIES 1 DOCKET 50-254 LER 91-008
REACTOR BUILDING VENTILATION ISOLATION DUE TO LIGHTNING STRIKE.
EVENT DATE: 032291 REPORT DATE: 041991 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: QUAD CITIES 2 (BWR)

(NSIC 221892) ON MARCH 22, 1991 AT 2035 HOURS, UNIT ONE WAS IN THE SHUTDOWN MODE AT 0% OF RATED CORE THERMAL POWER FOR A REFUELING OUTAGE. UNIT TWO WAS IN THE RUN MODE AT 98% OF RATED CORE THERMAL POWER. AT 2035 HOURS, LIGHTNING STRUCK IN A TRANSMISSION SUBSTATION (TSS) WHERE LINE 0402 (BARSTOW) TERMINATES. THE LIGHTNING STRIKE CAUSED THE REACTOR BUILDING (VA) AND CONTROL ROOM VENTS (VI) TO ISOLATE, A CHANNEL 'B' 1/2 SCRAM, AND OIL CIRCUIT BREAKER (OCB) 10-11 TO TRIP. OTHER SYSTEMS ALSO HAD ACTUATIONS CAUSED BY THE LIGHTNING STRIKE. AT 2057 HOURS, THE REACTOR BUILDING VENT FANS WERE RETURNED TO NORMAL. THE CONTROL ROOM VENTS WERE RESET AND TOXIC GAS SAMPLE POINT 'A' WAS SELECTED. LINE 0402 (OCB 10-11) WAS CLOSED BACK IN AT 2141 HOURS. THE 4 HOUR EMERGENCY NOTIFICATION SYSTEM (ENS) PHONE CALL WAS MADE AT 2205 HOURS ON MARCH 22, 1991. THIS EVENT IS BEING REPORTED ACCORDING TO 10CFR50.73(A)(2)(IV).

[100] RIVERBEND 1 DOCKET 50-458 LER 90-037
VIOLATION OF HIGH RADIATION AREA BY CONTRACT EMPLOYEES.
EVENT DATE: 110590 REPORT DATE: 120590 NSSS: GE TYPE: BWR

(NSIC 221507) ON NOVEMBER 5, 1990 AND AGAIN ON NOVEMBER 6, 1990, WITH THE PLANT IN OPERATIONAL CONDITION 5 (REFUELING), TWO WORKERS ENTERED A HIGH RADIATION AREA (HRA) WITHOUT SATISFYING THE REQUIREMENTS OF TECHNICAL SPECIFICATION 6.12.1. THEREFORE, THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(I)(B) AS OPERATION PROHIBITED BY THE TECHNICAL SPECIFICATIONS. THE ROOT CAUSE OF THIS EVENT WAS FAILURE OF THE 2 EMPLOYEES TO OBSERVE THE HRA BOUNDARY. TO PRECLUDE A RECURRENCE OF THIS TYPE, THE CONTRACTOR ISSUED A SELF-IMPOSED STOP WORK ORDER AT APPROXIMATELY 1700 HOURS ON 11/6/90. THE PURPOSE OF WHICH WAS TO ALLOW FOR THE RETRAINING OF ALL CONTRACTOR EMPLOYEES TO THE REQUIREMENTS OF RP BOUNDARIES AND PROCEDURES. THE TWO EMPLOYEES RESPONSIBLE WERE TRAINED, COUNSELED AND TERMINATED. THE TOTAL DOSE FOR BOTH WORKERS FOR BOTH ENTRIES AS INDICATED BY POCKET DOSIMETER WAS 43 MREM. THIS EVENT INVOLVING THE TWO INDIVIDUALS VIOLATING THE HIGH RADIATION BOUNDARY DOES NOT CONSTITUTE PLANT OPERATIONAL SAFETY CONCERN. THEREFORE, THIS EVENT DID NOT ADVERSELY AFFECT THE HEALTH AND SAFETY OF THE PUBLIC.

[101] RIVERBEND 1 DOCKET 50-458 LER 91-004
ISOLATION OF THE RCIC TURBINE MAIN STEAM SUPPLY LINE OUTBOARD CONTAINMENT ISOLATION VALVE.
EVENT DATE: 032191 REPORT DATE: 041991 NSSS: GE TYPE: BWR

VENDOR: ROSEMOUNT, INC.

(NSIC 221879) AT 1420 ON 3/21/91, WITH THE REACTOR IN OPERATIONAL CONDITION 1 (POWER OPERATION) AND THE REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM ISOLATED FOR MAINTENANCE, AN UNPLANNED ENGINEERED SAFETY FEATURE (ESF) ACTUATION OCCURRED WHEN THE RCIC TURBINE MAIN STEAM SUPPLY LINE OUTBOARD CONTAINMENT ISOLATION VALVE, 1E51*MOVF064, ISOLATED. THIS REPORT IS SUBMITTED PURSUANT TO 10CFR50.73(A)(2)(IV) SINCE THIS EVENT CONSTITUTES AN ESF ACTUATION. IT IS POSTULATED THAT AFTER THE RCIC SYSTEM WAS ISOLATED FOR MAINTENANCE BY SHUTTING 1E51*MOVF063, COOLING IN THE STEAM SUPPLY LINE CAUSED STEAM TO CONDENSE AND FLOW INTO THE VARIABLE LEG INSTRUMENT LINE WHICH RESULTED IN NEGATIVE DIFFERENTIAL PRESSURE SURGES IN FLOW TRANSMITTER 1E51*PDTN084A. THE NEGATIVE TRIP SETPOINT OF THE TRIP UNIT WAS REACHED WHICH CAUSED THE ESF ACTUATION. THIS IS THE PROBABLE ROOT CAUSE OF THE EVENT. A CAUTION HAS BEEN ADDED TO RCIC SYSTEM OPERATING PROCEDURES THAT BOTH RCIC TURBINE MAIN STEAM SUPPLY LINE CONTAINMENT ISOLATION VALVES 1E51*MOVF063 AND 1E51*MOVF064 SHOULD BE ISOLATED WHEN THE SYSTEM IS REMOVED FROM SERVICE FOR MAINTENANCE TO PREVENT UNEXPECTED ESF ACTUATIONS. THE ESF ACTUATION OCCURRED PER DESIGN. THE RCIC SYSTEM WAS INOPERABLE AND ISOLATED FOR MAINTENANCE PER THE ACTION STATEMENT OF TECHNICAL SPECIFICATION 3.7.3. THEREFORE, THE HEALTH AND SAFETY OF THE PUBLIC WAS NOT ADVERSELY AFFECTED.

[102] RIVERBEND 1 DOCKET 50-458 LER 91-003
DAMPER ISOLATIONS AND AUTOMATIC SWAP OF DIVISIONAL CONTROL BUILDING VENTILATION/CHILLER TRAINS DUE TO INADEQUATE WORK PLAN.
EVENT DATE: 032291 REPORT DATE: 042291 NSSS: GE TYPE: BWR

(NSIC 221878) AT 1055 ON 3/22/91, DURING MAINTENANCE ON THE DIVISION II CONTROL BUILDING LOCAL AIR INTAKE RADIATION MONITOR 1RMS*RE13B, THE DIVISION II CONTROL POWER CIRCUIT WAS DE-ENERGIZED. THIS RESULTED IN THE DE-ENERGIZATION OF THE DIVISION II CHARCOAL FILTER TRAIN SUCTION DAMPERS 1HVC*AOD19D AND 1HVC*AOD19F, AND ISOLATION OF THE AIR OPERATED DAMPERS (AODS) TO THE DIVISION II AIR HANDLING UNITS, 1HVC*AOD6B AND 1HVC*AOD8B. NOTE THAT DAMPERS 19D AND 19F WERE CLOSED AT THE TIME OF THE EVENT. THE ISOLATIONS RESULTED IN A TRIP OF THE DIVISION II CONTROL BUILDING VENTILATION SYSTEM/CHILLER AND AUTOMATIC SWAP TO THE DIVISION I VENTILATION SYSTEM/CHILLER. THIS REPORT IS SUBMITTED PURSUANT TO 10CFR50.73 TO DOCUMENT THE ENGINEERED SAFETY FEATURE (ESF) ACTUATIONS DESCRIBED ABOVE. THE EVENT OCCURRED DURING THE IMPLEMENTATION OF MODIFICATION REQUEST (MR) 90-0007. THIS MR SPECIFIED THAT THE RM-80 MOTHER BOARD WAS TO BE REMOVED FROM 1RMS*RE13B. THE ROOT CAUSE OF THIS EVENT IS THAT THE MAINTENANCE PLANNER OVERLOOKED THE 115VAC CONTROL POWER TO THE RM-80 MOTHER BOARD AND THUS, THE POTENTIAL FOR THE ESF ACTUATIONS. THIS EVENT CONCERNED THE ENGINEERING/MAINTENANCE PLANNING INTERFACE AND RESPONSIBILITY. AS PREVIOUSLY REPORTED IN LER 90-033, REVISION 2, A TASK FORCE EVALUATION OF THIS ISSUE WAS PERFORMED AND THE TASK FORCE RECOMMENDATIONS ARE UNDERGOING MANAGEMENT EVALUATION.

[103] RIVERBEND 1 DOCKET 50-458 LER 91-005
DESIGN DEFICIENCIES IN FIRE DOORS.
EVENT DATE: 032291 REPORT DATE: 042291 NSSS: GE TYPE: BWR

(NSIC 221882) ON 3/22/91, WITH THE PLANT AT 100% POWER IN OPERATIONAL CONDITION 1 (POWER OPERATION), A DEFICIENT FIRE DOOR WAS DISCOVERED DURING A QUALITY ASSURANCE AUDIT. THE DESIGN CONFIGURATION OF DOOR CB-70-25 DID NOT ASSURE THE PROPER CLOSING SEQUENCE BETWEEN THE TWO LEAVES. THEREFORE, THE DOOR CANNOT BE CONSIDERED TO HAVE BEEN A QUALIFIED FIRE BARRIER SINCE PLANT STARTUP, AND WAS INOPERABLE CONTRARY TO TECHNICAL SPECIFICATION 3.7.7. THIS REPORT IS SUBMITTED PURSUANT TO 10CFR50.73(A)(2)(I)(B) AS OPERATION PROHIBITED BY THE TECHNICAL SPECIFICATIONS. UPON DISCOVERY OF THE CONDITION, THE FIRE DOOR WAS DECLARED INOPERABLE AND ADDED TO THE HOURLY FIRE WATCH LIST. THE INACTIVE LEAF OF DOOR CB-70-25 WAS CLOSED AND THE TOP AND BOTTOM LATCH BOLTS WERE ENGAGED. THIS CONFIGURATION ENSURES OPERABILITY OF THE DOOR AND ENSURES CONFORMANCE TO APPLICABLE NFPA CODE REQUIREMENTS. THE CORRECTIVE ACTIONS SPECIFIED FOR DOOR CB-70-25 WERE ALSO IMPLEMENTED FOR DOOR CB-98-32. THE AREAS ON BOTH SIDES OF EACH DOOR, CB-70-25 AND CB-98-32, ARE PROVIDED WITH AUTOMATIC SPRINKLER SYSTEMS AND AUTOMATIC FIRE DETECTION SYSTEMS. IN THE EVENT OF A FIRE FROM TRANSIENT

COMBUSTIBLE SOURCES, THESE SYSTEMS WOULD AUTOMATICALLY ACTUATE TO CONTAIN THE FIRE AND NOTIFY THE CONTROL ROOM OF THE CONDITIONS.

[104] RIVERBEND 1 DOCKET 50-458 LER 91-006
PERSONNEL FAILURE TO REPLACE HIGH RADIATION AREA BARRIER.
EVENT DATE: 040191 REPORT DATE: 042991 NSSS: GE TYPE: BWR

(NSIC 221970) AT 1530 HOURS ON 4/1/91 WITH THE REACTOR IN OPERATIONAL CONDITION 1 (POWER OPERATION) AN OPERATIONS ENGINEERING COOPERATIVE STUDENT, HAVING ABOUT 3 MONTHS OF EXPERIENCE, FAILED TO REPLACE A HIGH RADIATION AREA (HRA) ROPE BARRICADE. THE ROPE WAS LOCATED AT THE DOORWAY BETWEEN THE RESIDUAL HEAT REMOVAL (RHR) "A" PUMP ROOM AND THE LOW PRESSURE CORE SPRAY (LPCS) PUMP ROOM. THE EVENT OCCURRED DURING VALVE OPERABILITY SURVEILLANCE TESTING. TECHNICAL SPECIFICATION (TS) 6.12.1 REQUIRES THAT HIGH RADIATION AREAS BE BARRICADED. THIS REPORT IS SUBMITTED PURSUANT TO 10CFR50.73(A)(2)(I)(B) AS OPERATION PROHIBITED BY THE TECHNICAL SPECIFICATIONS. AS AN INTERIM MEASURE, A STANDING ORDER WAS ISSUED TO ALL OPERATIONS PERSONNEL TO REQUIRE PERMISSION OF THE SHIFT SUPERVISOR/CONTROL OPERATING FOREMAN AND ESCORT BY RADIATION PROTECTION PERSONNEL PRIOR TO ENTRY INTO HIGH RADIATION AREAS AND VERY HIGH RADIATION AREAS. THESE REQUIREMENTS ARE IN ADDITION TO THOSE IN THE TECHNICAL SPECIFICATIONS. DURING THE PERIOD THAT THE BARRIER WAS DOWN, NO UNAUTHORIZED PERSONNEL ENTERED THE HRA. THEREFORE, THIS EVENT DID NOT HAVE AN ADVERSE IMPACT ON THE SAFE OPERATION OF RIVER BEND STATION OR THE HEALTH AND SAFETY OF THE PUBLIC.

[105] SALEM 1 DOCKET 50-272 LER 91-003
2 STEAM FLOW CHANNELS FOR I STEAMLINE INOPERABLE DUE TO PERSONNEL ERROR.
EVENT DATE: 020991 REPORT DATE: 030691 NSSS: WE TYPE: PWR

(NSIC 221215) ON 2/9/91 AT 0845 HOURS, DURING REACTOR SHUTDOWN (IN SUPPORT OF THE UPCOMING NINTH REFUELING OUTAGE), A NO. 14 STEAM GENERATOR (S/G) STEAMLINE FLOW CHANNEL I TRANSMITTER SENSING LINE WAS ISOLATED DURING INVESTIGATION OF A 14 S/G STEAMLINE FLOW CHANNEL II ERRONEOUS READING. SUBSEQUENTLY, TECH. SPEC. ACTION STATEMENT 3.0.3 WAS ENTERED SINCE THE ACTION STATEMENTS FOR TECH. SPECS. 3.3.2.1 AND 3.3.3.1 DO NOT ADDRESS REQUIRED ACTIONS WITH MORE THAN ONE INOPERABLE STEAMLINE FLOW CHANNEL OR ANY ONE S/G. ROOT CAUSE OF THIS EVENT IS PERSONNEL ERROR AS ATTRIBUTED TO INAPPROPRIATE SUPERVISORY DIRECTION. THE SUPERVISOR INVOLVED ACTED UPON AN INVALID ASSUMPTION WITHOUT FULLY ASSESSING AN UNUSUAL SITUATION. WHEN THE TRANSMITTER SENSING LINE WOULD NOT STOP VENTING (AFTER THE ROOT VALVE WAS CLOSED) THE SUPERVISOR INCORRECTLY ASSUMED EITHER THE SCHEMATIC WAS NOT READ CORRECTLY, THE VALVES WERE MISLABELED OR THE SCHEMATIC WAS INCORRECT. THE SUPERVISOR DID NOT CONSIDER THAT THE CORRECT TRANSMITTER SENSING LINE ROOT VALVE WAS CLOSED BUT WAS LEAKING. CONTRIBUTING TO THIS EVENT WAS THAT THE SCHEMATIC DRAWING, WHICH DETAILS COMPONENT ALIGNMENT, (FOR THE TRANSMITTER SENSING LINES) WAS NOT TAKEN TO THE JOB SITE. UPON NOTIFICATION OF THE ISOLATION OF THE CHANNEL I SENSING LINE, THE SUPERVISOR REOPENED THE WRONG ROOT VALVE AND CLOSED THE CORRECT VALVE.

[106] SALEM 1 DOCKET 50-272 LER 91-014
TWO CHANNELS MADE INOPERABLE IN A SINGLE SYSTEM DUE TO A COMMON EQUIPMENT FAILURE.
EVENT DATE: 032291 REPORT DATE: 041791 NSSS: WE TYPE: PWR
VENDOR: CONDE MILKING MACHINE COMPANY

(NSIC 221908) ON 3/22/91 AT 0112 HOURS, A CONTROL ROOM OPERATOR OBSERVED A LOW FLOW INDICATION FOR THE 1R11A/1R12A/1R12B CONTAINMENT RADIATION MONITORING SYSTEM (RMS) PUMP . THESE 3 CHANNELS ANALYZE A SAMPLE OF CONTAINMENT ATMOSPHERE PROVIDED BY THIS SINGLE PUMP. IT WAS OBSERVED THAT THE LOCAL PUMP FLOW INDICATION WAS LOW AND THAT THE PUMP WAS MAKING "GRINDING" NOISES. SUBSEQUENTLY, THE 1R11A, 1R12A AND 1R12B RMS CHANNELS WERE DECLARED INOPERABLE AND APPLICABLE TECHNICAL SPECIFICATION ACTION STATEMENTS WERE ENTERED. THE ROOT CAUSE OF THIS EVENT IS ATTRIBUTED TO EQUIPMENT DESIGN CONCERNS. PRIOR LERS HAVE IDENTIFIED THE CONCERN FOR THE ADEQUACY OF THE PUMP DESIGN (E.G., LER 272/89-010-00). THE SAMPLE PUMP IS AN INTERNAL RIGID ROTOR VANE PUMP WHICH USES GRAPHITE VANES. INSPECTION OF THE SAMPLE PUMP REVEALED THAT THE FOUR (4) PUMP GRAPHITE VANES HAD EACH CHIPPED. ON

THE SAME OUTSIDE CORNER (APPROXIMATELY 0.5"). THE SPECIFIC CAUSE OF THE VANES CHIPPING COULD NOT BE IDENTIFIED. THE SAMPLE PUMP WAS REPLACED. ON 3/22/91 AT 1101 HOURS THE 1R11A AND 1R12A MONITORS WERE DECLARED OPERABLE AND THE APPLICABLE TECHNICAL SPECIFICATIONS WERE EXITED. THE PUMP IS SCHEDULED TO BE REPLACED WITH A MORE RELIABLE PUMP AS PART OF THE RMS UPGRADE PROJECT. AN INTERIM MEASURE IS TO REPLACE THE PUMP EVERY 6 MONTHS.

[107] SALEM 1 DOCKET 50-272 LER 91-016
HIGH OXYGEN CONCENTRATION IN WASTE GAS SYSTEM FOR GREATER THAN 48 HOURS.
EVENT DATE: 033191 REPORT DATE: 043091 NSSS: WE TYPE: PWR

(NSIC 221922) OXYGEN CONCENTRATION WITHIN THE WASTE GAS HOLDUP SYSTEM WAS > 2% FOR > 48 HOURS. ON 3/29/91, TECH. SPEC. ACTION STATEMENT 3.11.2.5.A WAS ENTERED DUE TO AN OXYGEN CONCENTRATION OF 2.3% BY VOLUME IN THE NO. 13 WASTE GAS DECAY TANK (WGDT). THE TANK WAS PURGED WITH NITROGEN. TANK DISCHARGE WAS ATTEMPTED ON 3/30/91; HOWEVER, DUE TO A PRIOR FAILURE OF THE 1R41C PLANT VENT RADIATION MONITOR, THE WASTE GAS DISCHARGE VALVE (1WG41) COULD NOT BE REMOTELY OPENED. FAILURE OF THE 1R41C CHANNEL RESULTS IN A 1WG41 ISOLATION SIGNAL. THE NOS. 11 AND 14 WGDTS ALSO EXCEEDED 2% OXYGEN ON MARCH 31 AND MARCH 30, RESPECTIVELY. THE MAXIMUM OXYGEN CONCENTRATION OBSERVED IN ANY TANK WAS 2.7% . THE ROOT CAUSE, OF NOT REDUCING THE OXYGEN CONCENTRATION WITHIN 48 HOURS TO < 2%, IS ATTRIBUTED TO EQUIPMENT FAILURE OF THE 1R41C CHANNEL. A CONTRIBUTING FACTOR WAS THE MISUNDERSTANDING OF THE 1WG41 VALVE ISOLATION SIGNAL CIRCUIT DESIGN. THE SOURCE OF THE OXYGEN IS ATTRIBUTED TO THE CVCS HOLDUP TANKS' COVER GAS. OXYGEN CONCENTRATION, IN THE NO. 13 CVCS HOLDUP TANK, WAS 7.56% AS OF 3/19/91. OXYGEN, ENTRAINED IN REACTOR COOLANT DUE TO REFUELING OUTAGE ACTIVITIES, COMES OUT OF SOLUTION IN THE CVCS HOLDUP TANKS. THE WASTE GAS SYSTEM OXYGEN LEVEL WAS REDUCED TO BELOW THE 2% LIMIT. THIS EVENT HAS BEEN REVIEWED BY OPERATIONS DEPARTMENT MANAGEMENT. IT WILL BE DISCUSSED WITH APPLICABLE DEPARTMENT PERSONNEL.

[108] SAN ONOFRE 1 DOCKET 50-206 LER 90-016 REV 01
UPDATE ON SUSCEPTIBILITIES TO EMERGENCY CORE COOLING SYSTEM FAILURES.
EVENT DATE: 072790 REPORT DATE: 043091 NSSS: WE TYPE: PWR

(NSIC 221913) IN A LETTER TO NRC REGION V DATED 3/17/89, SCE COMMITTED TO A REANALYSIS OF THE 1976 SINGLE FAILURE ANALYSIS (SFA) PERFORMED ON THE SONGS UNIT 1 EMERGENCY CORE COOLING SYSTEM (ECCS) AND SUPPORTING SYSTEMS. AT 1330 ON 7/27/90, WITH UNIT 1 SHUT DOWN AND DEFUELED, EIGHT SINGLE FAILURE SUSCEPTIBILITIES WHICH COULD HAVE POTENTIALLY IMPACTED THE PERFORMANCE OF SOME ECCS FUNCTIONS WERE CONFIRMED IN THIS 1990 REANALYSIS EFFORT. DETAILS WERE PROVIDED TO THE NRC IN AN INTERIM REPORT DATED 7/31/90. THIS INTERIM REPORT ALSO IDENTIFIED OTHER SINGLE FAILURE ISSUES WHICH WERE UNDER REVIEW AT THAT TIME. PRIOR TO UNIT 1 STARTUP FROM THE CYCLE 11 OUTAGE, SCE PROVIDED A FOLLOW-UP REPORT TITLED, "SONGS 1 ECCS SINGLE FAILURE ANALYSIS FOLLOW-UP REPORT," DATED 2/27/91 . THIS REPORT DISCUSSED IN DETAIL THE SAFETY SIGNIFICANCE AND CORRECTIVE ACTIONS ASSOCIATED WITH NINE SINGLE FAILURE DEFICIENCIES WHICH REQUIRED MODIFICATIONS. THESE DEFICIENCIES WERE, IN PART, CAUSED BY INADEQUATE SFA GUIDANCE FOR NON-STANDARD CONFIGURATION PLANTS AND PROGRAMMATIC WEAKNESSES PREVIOUSLY IDENTIFIED TO THE NRC. THE 1990 ECCS SFA SCOPE, METHODOLOGY, CRITERIA UTILIZED, AND RESULTS WERE ALSO SUBMITTED UNDER A SEPARATE COVER LETTER DATED 2/22/91. THIS INFORMATION REPRESENTS THE CURRENT DESIGN BASIS, AND SUPERSEDED PREVIOUS INFORMATION PROVIDED REGARDING THE SINGLE FAILURE DESIGN BASIS FOR SONGS UNIT 1.

[109] SAN ONOFRE 1 DOCKET 50-206 LER 91-001 REV 03
UPDATE ON CONTAINMENT SPRAY FLOW LIMITER VALVE ACTUATOR INCORRECTLY COUPLED.
EVENT DATE: 010791 REPORT DATE: 051391 NSSS: WE TYPE: PWR

(NSIC 221977) ON 12/23/90, WITH UNIT 1 IN MODE 6, AN EVALUATION OF THE PERFORMANCE OF A REFUELING WATER PUMP (RWP) (WHICH PROVIDES CONTAINMENT SPRAY) FULL FLOW INSERVICE TEST (IST) REVEALED THAT POSITION INDICATION FOR FLOW RESTRICTING BALL VALVE CV-518 WAS REVERSED (I.E., THE VALVE INDICATED OPEN WHEN IN THE CLOSED POSITION AND VICE VERSA). THIS DUAL FUNCTION VALVE IS DESIGNED TO OPEN TO INCREASE FLOW TO THE CONTAINMENT SPRAY HEADER DURING LOSS OF COOLANT

(LOCA) AND MAIN STEAM LINE BREAK (MSLB) ACCIDENTS, AND CLOSE TO REDUCE FLOW DURING RECIRCULATION. ON 1/7/91, FOLLOWING INVESTIGATION AND ANALYSIS, IT WAS DETERMINED THAT THIS CONDITION EXISTED DURING PLANT OPERATION, AND WOULD HAVE AFFECTED THE RESPONSE OF THE CONTAINMENT SPRAY SYSTEM (CSS) AND THE CONTAINMENT RECIRCULATION SYSTEM (CRS) TO A LOCA OR MSLB INSIDE CONTAINMENT. DURING A FEBRUARY 1989 MAINTENANCE ACTIVITY IN WHICH THE ACTUATOR WAS REMOVED FROM THE VALVE, THE POSITION OF THE VALVE WAS IMPROPERLY CHANGED RESULTING IN IMPROPER ALIGNMENT BETWEEN THE VALVE AND ACTUATOR WHEN THE ACTUATOR WAS REINSTALLED. THE CAUSE OF THIS EVENT INCLUDED PROCEDURAL DEFICIENCIES, MAINTENANCE IMPLEMENTATION DEFICIENCIES, DEFICIENT DESIGN CHARACTERISTICS, AND A MISSED OPPORTUNITY TO IDENTIFY THE MISALIGNED VALVE. CORRECTIVE ACTIONS TO PREVENT RECURRENCE INCLUDE: PROCEDURAL REVIEWS AND CHANGES, TRAINING, AND DESIGN CHANGES.

[110] SAN ONOFRE 1 DOCKET 50-206 LER 91-006
FAILURE TO PERFORM RADIOACTIVE GASEOUS EFFLUENT TECHNICAL SPECIFICATION REQUIRED SAMPLE.
EVENT DATE: 021991 REPORT DATE: 041591 NSSS: WE TYPE: PWR

(NSIC 221909) ON 3/14/91, WITH UNIT 1 IN MODE 3, IT WAS DISCOVERED DURING THE CALCULATION OF GASEOUS DISCHARGES FOR THE FEBRUARY MONTHLY EFFLUENT REPORT THAT PARTICULATE AND IODINE DATA FOR A TWO DAY TIME PERIOD WAS NOT AVAILABLE. TECH SPEC (TS) 4.6.1, REQUIRES CONTINUOUS SAMPLING FOR IODINE AND PARTICULATE ACTIVITY FOR THE PLANT VENT STACK FLOWPATH. ON 2/21/91, WHILE IN MODE 5 AND WITH THE NORMAL PLANT VENT STACK MONITORS OUT OF SERVICE, AUX. SAMPLING EQUIPMENT WAS CONNECTED AS REQUIRED COMPENSATION FOR THE OUT OF SERVICE MONITORS. FOLLOWING A SHIFT CHANGE, THE PERMANENT MONITORS WERE RETURNED TO SERVICE, THE AUXILIARY EQUIPMENT WAS REMOVED AND THE FILTERS WHICH COLLECT THE PARTICULATE AND IODINE SAMPLES WERE REMOVED AND ANALYZED. FILTERS ON THE PERMANENT RADIATION MONITORS WERE ALSO REMOVED AND DISCARDED AT THIS TIME. THE CONTRIBUTION OF THE PARTICULATE AND IODINE ACTIVITY FROM THE PLANT VENT STACK FROM 2/19/91 TO 2/21/91 CAN NOT BE POSITIVELY DETERMINED. ROOT CAUSE OF THIS EVENT IS INADEQUATE COMMUNICATION DURING THE SHIFT CHANGE TURN-OVER PROCESS. THE TURN-OVER INSTRUCTIONS, DOCUMENTED IN THE TURN-OVER LOG BY SWING SHIFT CHEMISTRY TECHNICIANS, WERE INADEQUATE IN THAT THEY DIRECTED THE GRAVE YARD SHIFT CHEMISTRY TECHNICIANS TO REMOVE THE AUXILIARY SAMPLE EQUIPMENT AND REPLACE THE FILTERS IN THE PERMANENT MONITORS, BUT DID NOT INCLUDE SAVING PERMANENT MONITOR FILTERS.

[111] SAN ONOFRE 1 DOCKET 50-206 LER 91-005
TECHNICAL SPECIFICATION REQUIRED EFFLUENT SAMPLES DISCARDED DUE TO INADEQUATE LABELING.
EVENT DATE: 032191 REPORT DATE: 042291 NSSS: WE TYPE: PWR

(NSIC 221884) ON 3/21/91 DURING THE MIDNIGHT SHIFT, A CHEMISTRY TECHNICIAN INADVERTENTLY DISCARDED THE WEEKLY CONTINUOUS RELEASE PATHWAY LIQUID COMPOSITE SAMPLES FOR THE WEEK OF MARCH 11 - 17, 1991. THUS, THE MARCH 1991 MONTHLY GROSS ALPHA AND FIRST QUARTER 1991 FE-55/SR-89/SR-90 COMPOSITE SAMPLES WILL NOT ACCURATELY REFLECT UNIT RELEASES AS REQUIRED BY TECH SPEC (TS) 4.5.1, "RADIOACTIVE LIQUID EFFLUENTS - LIQUID EFFLUENT CONCENTRATION." THE ROOT CAUSE WAS ATTRIBUTED TO PERSONNEL ERROR SINCE A CHEMISTRY TECHNICIAN HAD DEVIATED FROM PROCEDURAL REQUIREMENTS GOVERNING CHEMISTRY SAMPLE LABELING. THE SAMPLE CONTAINERS WERE NOT LABELED SUCH THAT THE REQUIREMENT FOR RETENTION WAS APPARENT. THE CHEMISTRY TECHNICIAN RESPONSIBLE FOR THE LABELING DEFICIENCY RECEIVED APPROPRIATE DISCIPLINARY ACTION, AND THIS EVENT WAS REVIEWED WITH OTHER APPROPRIATE CHEMISTRY PERSONNEL. SINCE THERE IS TYPICALLY LITTLE DEVIATION IN THE SAMPLE'S ISOTOPIC CONCENTRATION FROM WEEK TO WEEK, A COMPOSITE CONSISTING OF SAMPLES OBTAINED IN THE FIRST, THIRD, AND FOURTH WEEKS IS CONSIDERED TO BE REPRESENTATIVE OF THE ENTIRE MONTH. AS SUCH, THIS WILL BE REPORTED IN THE NEXT SEMI-ANNUAL EFFLUENT REPORT. THERE WAS NO SAFETY SIGNIFICANCE TO THIS EVENT SINCE THE CONTRIBUTIONS OF GROSS ALPHA, SR-89, SR-90, AND FE-55 ISOTOPES TO THE ANNUAL PROJECTED DOSE ARE TYPICALLY INSIGNIFICANT.

[112] SAN ONOFRE 1 DOCKET 50-206 LER 91-007
UNDERESTIMATED REACTOR VESSEL REFILL VOLUME UTILIZED IN THE LARGE BREAK LOSS OF COOLANT ACCIDENT ANALYSIS.
EVENT DATE: 032891 REPORT DATE: 042991 NSSS: WE TYPE: PWR

(NSIC 221910) ON 3/28/91 WITH UNIT 1 OPERATING AT 20% REACTOR POWER, IT WAS DETERMINED THAT THE VALUE FOR THE REACTOR VESSEL REFILL VOLUME USED IN THE LARGE BREAK LOSS OF COOLANT ACCIDENT (LBLOCA) ANALYSIS PERFORMED BY WESTINGHOUSE WAS UNDERESTIMATED BY APPROXIMATELY 182 CUBIC FEET. CONSERVATIVELY EVALUATING THE EFFECT OF THIS DISCREPANCY SHOWS THAT THE ASSUMED TIME FOR REFILL OF THE AREA BELOW THE ACTIVE CORE REGION FOLLOWING A LBLOCA COULD BE EXTENDED BY AS MUCH AS 17.3 SECONDS. ASSUMING WORST CASE CONDITIONS (E.G., WORST POSSIBLE PEAKING FACTORS, ETC.), THE VOLUME DISCREPANCY WOULD HAVE RESULTED IN THE PREDICTED LBLOCA PEAK CLAD TEMPERATURE (PCT) EXCEEDING THE SONGS 1 ACCEPTANCE CRITERIA OF 2300 DEGREES F. UTILIZING ACTUAL PLANT OPERATING PARAMETERS, THE PREDICTED PCT WOULD HAVE REMAINED BELOW THE SONGS 1 ACCEPTANCE CRITERIA. A REVIEW BY WESTINGHOUSE OF THE AVAILABLE DOCUMENTATION WAS NOT SUCCESSFUL IN DETERMINING THE BASIS FOR THE ORIGINAL CALCULATION OF THE REFILL VOLUME IN THE LBLOCA ANALYSIS. OUR REVIEW OF THE ROOT CAUSE OF THIS DEFICIENCY IS CONTINUING AND WILL BE ADDRESSED IN A SUPPLEMENT TO THIS REPORT. CORRECTIVE ACTIONS TAKEN INCLUDED: 1) RESTRICTION OF REACTOR POWER LEVEL TO 75% WAS IMPLEMENTED (AT THIS POWER LEVEL THE SONGS 1 ACCEPTANCE CRITERIA FOLLOWING A LBLOCA WOULD HAVE BEEN SATISFIED), AND 2) FULL POWER OPERATION WAS SUBSEQUENTLY AUTHORIZED.

[113] SAN ONOFRE 2 DOCKET 50-361 LER 91-006
INADVERTENT TOXIC GAS ISOLATION SYSTEM ACTUATION DURING THE PERFORMANCE OF MAINTENANCE TROUBLESHOOTING.
EVENT DATE: 032591 REPORT DATE: 042491 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: SAN ONOFRE 3 (PWR)

(NSIC 221881) AT 1530 ON 3/25/91, WITH UNIT 2 AT 60% POWER AND UNIT 3 AT 100% POWER, TOXIC GAS ISOLATION SYSTEM (TGIS) TRAIN "A" ACTUATED ON HIGH AMMONIA GAS LEVEL. ALL TGIS TRAIN "A" COMPONENTS WERE VERIFIED TO HAVE ACTUATED AS REQUIRED. THE ACTUATION OCCURRED WHILE TROUBLESHOOTING WAS BEING PERFORMED ON THE TRAIN "A" AMMONIA CHANNEL, WHICH HAD FAILED ON 3/24/91. A MAINTENANCE TECHNICIAN INADVERTENTLY BUMPED THE JUMPER USED TO BYPASS THE TGIS ACTUATION CIRCUITRY, RESULTING IN THE JUMPER BEING MOMENTARILY DISLODGED. SINCE THE AMMONIA LEVEL HAD BEEN INCREASED ABOVE THE ACTUATION SETPOINT TO PERFORM THE TROUBLESHOOTING, AN ACTUATION ON HIGH AMMONIA OCCURRED. THE ROOT CAUSE OF THIS EVENT IS THAT THE LOCATION OF THE JUMPER USED FOR BYPASSING THE TGIS ACTUATION CIRCUITRY WAS ADJACENT TO THE AREA REQUIRING ACCESS DURING MAINTENANCE ACTIVITIES. THEREFORE, A POTENTIAL EXISTED FOR DISTURBANCE OF THE JUMPER DURING THESE ACTIVITIES. FOR CORRECTIVE ACTIONS: 1) APPROPRIATE DISCIPLINARY ACTION HAS BEEN ADMINISTERED TO THE TECHNICIAN INVOLVED IN THIS EVENT, 2) THIS EVENT HAS BEEN REVIEWED WITH APPROPRIATE MAINTENANCE PERSONNEL, AND 3) THE BYPASS JUMPER, WHEN INSTALLED IN THE FUTURE, WILL BE LOCATED TO AN AREA LESS LIKELY TO BE AFFECTED BY MAINTENANCE ACTIVITIES.

[114] SEABROOK 1 DOCKET 50-443 LER 91-002
MANUAL REACTOR TRIP DUE TO LOSS OF A VITAL BUS.
EVENT DATE: 033091 REPORT DATE: 042691 NSSS: WE TYPE: PWR
VENDOR: GOULD INC.

(NSIC 221966) ON MARCH 30, 1991 AT 11:48 A.M., A MANUAL REACTOR TRIP WAS INITIATED WHILE THE PLANT WAS AT APPROXIMATELY 50% POWER. THE REACTOR TRIP WAS INITIATED DUE TO A TURBINE RUNBACK COINCIDENT WITH A LOSS OF THE CONDENSER STEAM DUMP VALVES. WHILE THE PLANT WAS AT 100% POWER, AN ELECTRICAL FAULT OCCURRED IN THE TRANSFORMER SECTION OF 480 VOLT AC UNIT SUBSTATION EDE-US-52 RESULTING IN A LOSS OF ALL LOADS POWERED FROM THIS BUS. CONSEQUENTLY, A TURBINE RUNBACK WAS INITIATED BY A LOSS OF GENERATOR STATOR COOLING (GSC) SYSTEM CONTROL POWER. ADDITIONALLY, CONTROL POWER TO THE CONDENSER STEAM DUMP VALVES WAS LOST. IN RESPONSE TO THE LOSS OF THE CONDENSER STEAM DUMP VALVES, ALL FOUR ATMOSPHERIC STEAM DUMP VALVES AND SEVERAL STEAM GENERATOR SAFETY VALVES AUTOMATICALLY ACTUATED. FOLLOWING THE REACTOR TRIP AND TURBINE TRIP, A MAIN FEEDWATER

ISOLATION AND AN EMERGENCY FEEDWATER ACTUATION OCCURRED AS DESIGNED. ADDITIONALLY, UPON RESTORATION OF POWER, THE CONTROL ROOM EMERGENCY AIR CLEANUP AND FILTRATION SUBSYSTEM ACTUATED AS DESIGNED. THE ROOT CAUSE FOR THE ELECTRICAL FAULT HAS BEEN DETERMINED TO BE A TURN-TO-TURN FAULT ON ONE PHASE OF THE 4160/480 VOLT AC TRANSFORMER'S PRIMARY WINDING. THE TRANSFORMER FOR THE BUS WAS REPLACED. ADDITIONALLY, THIS EVENT WILL BE DISCUSSED WITH OPERATIONS PERSONNEL DURING REQUALIFICATION TRAINING.

[115] SEQUOYAH 1 DOCKET 50-327 LER 91-006
THE REACTOR BUILDING ANNULUS FIRE PROTECTION SYSTEM WAS NOT PROPERLY VERIFIED AS REQUIRED BY TS BECAUSE OF A FAILURE TO UPDATE THE ASSOCIATED PROCEDURE AS A RESULT OF A MODIFICATION.
EVENT DATE: 040391 REPORT DATE: 050391 NSSS: WE TYPE: PWR

(NSIC 221938) ON APRIL 3, 1991, AT 1430 EASTERN DAYLIGHT TIME WITH UNIT 1 AT 100 PERCENT POWER , SQN ENTERED LIMITING CONDITION FOR OPERATION (LCO) 3.7.11.2 FOR UNIT 1 WHEN IT WAS DISCOVERED THAT SEVEN SPRINKLER HEADS IN THE UNIT 1 REACTOR BUILDING ANNULUS HAD NOT BEEN INSPECTED, AS REQUIRED BY TS. IN JUNE 1988, WORK PLAN (WP) 418-01 ADDED SEVEN SPRINKLER HEADS TO THE EXISTING 45 IN THE ANNULUS AND RELOCATED 10 EXISTING SPRINKLER HEADS. THE SURVEILLANCE INSTRUCTION (SI) THAT IMPLEMENTS THE SURVEILLANCE REQUIREMENT WAS NOT UPDATED TO INDICATE THE ADDITIONAL SPRINKLER HEADS, AND THEREFORE, THESE SPRINKLER HEADS HAD NOT BEEN VERIFIED SINCE THE IMPLEMENTATION OF THE WP IN 1988. IMMEDIATE CORRECTIVE ACTION FOR THIS EVENT WAS TO ENTER THE LCO 3.7.11.2 AND ESTABLISH CONTINUOUS FIRE WATCHES IN THE UNIT 1 ANNULUS. THE SPRINKLER SYSTEM WAS WALKED DOWN, THE SI PERFORMED, AND A PRESSURE TEST WAS CONDUCTED AS A POST MAINTENANCE TEST. THE SI WAS REVISED TO REFLECT THE CURRENT NUMBER OF SPRINKLER HEADS. THE LCO WAS EXITED AT 2040 ON APRIL 4, 1991.

[116] SEQUOYAH 2 DOCKET 50-328 LER 91-002 REV 01
UPDATE ON FAILURE TO COMPLY WITH TECH SPEC ACTION STATEMENT AND ESTABLISH THE APPROPRIATE COMPENSATORY MEASURES.
EVENT DATE: 021191 REPORT DATE: 050291 NSSS: WE TYPE: PWR
VENDOR: PACIFIC PUMPS

(NSIC 221939) ON 2/19/91, WITH UNIT 2 IN MODE 1, IT WAS DETERMINED THAT ON 2/11/91, UNIT 2 HAD OPERATED IN A CONDITION PROHIBITED BY TECH SPEC 3.3.3.8 LIMITING CONDITION FOR OPERATION (LCO) ACTION STATEMENT (A). ACTION STATEMENT (A) REQUIRES A FIRE WATCH TO BE ESTABLISHED WITHIN AN HOUR UPON ENTERING THE LCO. ON 2/11/91, FIRE PROTECTION PANEL 0-L-630 WAS REMOVED FROM SERVICE FOR MAINTENANCE, AND OPERATIONS ENTERED LCO 3.3.3.8. BECAUSE OF THE EXPECTED SHORT DURATION, NO FIRE WATCH WAS ESTABLISHED. OPERATIONS ORDERED THE WORK STOPPED AND RETURN OF THE PANEL TO NORMAL WHEN ALL FOUR FIRE PUMPS STARTED UNEXPECTEDLY. OPERATIONS PREMATURELY EXITED THE LCO WHEN INFORMED THAT THE PANEL HAD BEEN RETURNED TO NORMAL. COMMUNICATIONS BETWEEN OPERATIONS PERSONNEL AND MAINTENANCE PERSONNEL WAS INADEQUATE CAUSING OPERATIONS TO CONSIDER THE PANEL TO BE OPERABLE. TROUBLESHOOTING LATER REVEALED THE PANEL WAS INOPERABLE AND OPERATIONS WAS NOTIFIED. LCO 3.3.3.8 WAS RE-ENTERED. UNIT 2 OPERATED APPROXIMATELY SIX HOURS WITHOUT ESTABLISHING A FIRE WATCH. MULTIPLE CAUSES AND CONTRIBUTING FACTORS HAVE BEEN IDENTIFIED, INCLUDING AN INADEQUATE PROCEDURE, POOR COMMUNICATION, INADEQUATE TRAINING, AND FAILURE TO FOLLOW PROCEDURES. CORRECTIVE ACTIONS ARE BEING TAKEN TO ADDRESS THESE ITEMS.

[117] SEQUOYAH 2 DOCKET 50-328 LER 91-004
COMPUTER POINT OUT OF SCAN ON THE P-250 COMPUTER AS A RESULT OF NOT MAINTAINING CONFIGURATION CONTROL.
EVENT DATE: 032991 REPORT DATE: 042991 NSSS: WE TYPE: PWR

(NSIC 221940) ON MARCH 29, 1991, AT 1142 EASTERN STANDARD TIME, WITH UNIT 2 IN MODE 1, TVA IDENTIFIED THE PLANT PROCESS COMPUTER P-250 COMPUTER POINT C0019A (CONTROL BANK C, GROUP 2, POSITION K-10) WAS OUT-OF-SCAN. THE COMPUTER POINT MONITORS A CONTROL ROD CLUSTER POSITION, WHICH PROVIDES INPUT TO THE "ROD DEVIATION AND SEQUENCE POWER RANGE TILTS" ALARM. SURVEILLANCE EQUIPMENT (SR)

4.1.3.2 REQUIRES THE COMPARISON OF THE DEMAND POSITION AND ROD POSITION INDICATION SYSTEMS AT LEAST ONCE EVERY FOUR HOURS IF THIS ALARM IS INOPERABLE. THE COMPUTER POINT WAS TAKEN OUT OF SCAN ON MARCH 21, 1991, WHILE TROUBLESHOOTING THE "ROD DEVIATION AND SEQUENCE POWER RANGE TILTS" ALARM, WHICH CAME ON DURING THE PERFORMANCE OF PERIODIC INSTRUCTION (PPI), "BIMONTHLY RESETTING OF CONTROL ROD FULLY WITHDRAWN POSITION." THE CAUSE OF THE EVENT IS FAILURE TO RETURN THE POINT TO CONFIGURATION CONTROL DURING TROUBLESHOOTING OF THE ALARM WAS NOT IMPLEMENTED. IMMEDIATE CORRECTIVE ACTION WAS TO PLACE THE COMPUTER POINT BACK INTO SCAN. ADDITIONAL CORRECTIVE ACTIONS INCLUDE DISCUSSIONS WITH PERSONNEL CONCERNING THE IMPORTANCE OF CONFIGURATION CONTROL AND IMPLEMENTATION OF A METHOD TO MAINTAIN CONFIGURATION CONTROL.

[118] SHEARON HARRIS 1 DOCKET 50-400 LER 90-019
POTENTIAL LOSS OF RESIDUAL HEAT REMOVAL SYSTEM DUE TO LOSS OF COMPONENT COOLING WATER CAUSED BY SYSTEM DESIGN.
EVENT DATE: 082490 REPORT DATE: 101590 NSSS: WE TYPE: PWR

(NSIC 221517) ON 8/24/90, THE PLANT WAS OPERATING AT 100% POWER. DURING THE INVESTIGATION INVOLVING COMPONENT COOLING WATER (CCW) RELIEF VALVE EVENT (LICENSEE EVENT REPORT (LER) 90-018-00), A SECOND SAFETY CONCERN WAS IDENTIFIED. THIS CONCERN INVOLVED SYSTEM LINEUPS WHICH COULD POSSIBLY CAUSE A CCW PUMP TO EXCEED ITS DESIGN FLOW CAPACITY AND RESULT IN PUMP RUNOUT. LOSS OF THE CCW SYSTEM COULD PREVENT THE RESIDUAL HEAT REMOVAL (RHR) SYSTEM FROM PERFORMING ITS DESIGN SAFETY FUNCTION. THE POTENTIAL CCW PUMP RUNOUT CONDITION IDENTIFIED IN PROCEDURE EMERGENCY OPERATING PROCEDURE (EOP) - END PATH PROCEDURE (EPP) EOP-EPP-010 "TRANSFER TO COLD LEG RECIRCULATION" (SEE FIGURE 1) WAS EVALUATED AS NOT BEING A VALID CONCERN BY PLANT CHANGE REQUEST (PCR)-5459. HOWEVER, FURTHER PROCEDURE REVIEWS REVEALED TWO ADDITIONAL SCENARIOS THAT INVOLVE SIMILAR SAFETY CONCERNS WHICH PCR-5459 COULD NOT DISCOUNT. BOTH SCENARIOS INVOLVE SYSTEM LINEUPS REQUIRING TWO CCW PUMPS SUPPLYING THE NONESSENTIAL HEADER AND TWO ESSENTIAL HEADER TRAINS. THE PROCEDURES INVOLVED: EOP-PATH-1, OPERATING PROCEDURE (OP)-111 (RHR SYSTEM) AND OP-145 (CCW SYSTEM) DO NOT SPECIFY THE SEPARATION OF BOTH CCW TRAINS DURING A SMALL BREAK LOSS OF COOLANT ACCIDENT (LOCA) WITH SAFETY INJECTION (SI) ACTUATION OR DURING NORMAL PLANT MODES 4-6 OPERATIONS.

[119] SHEARON HARRIS 1 DOCKET 50-400 LER 91-006
TECHNICAL SPECIFICATION VIOLATION DUE TO MISSED SHUTDOWN MARGIN VERIFICATION WHILE TESTING SOURCE RANGE NUCLEAR INSTRUMENTS.
EVENT DATE: 031991 REPORT DATE: 041791 NSSS: WE TYPE: PWR

(NSIC 221963) ON 3/19/91, DURING SURVEILLANCE TESTING ON THE SOURCE RANGE NUCLEAR INSTRUMENTS, CONTROL ROOM OPERATORS DID NOT VERIFY SHUTDOWN MARGIN REQUIREMENTS PER TECHNICAL SPECIFICATIONS (TS). UPON DISCOVERY, SHUTDOWN MARGIN WAS IMMEDIATELY VERIFIED TO BE ADEQUATE. THIS CONDITION WAS CAUSED BY A MISINTERPRETATION OF TS REQUIREMENTS. CORRECTIVE ACTIONS WILL INCLUDE A REVISION TO THE APPROPRIATE PROCEDURES AND DISSEMINATION OF THIS INFORMATION TO OPERATIONS PERSONNEL TO CLARIFY THE REQUIREMENTS OF THIS TS. THIS IS BEING REPORTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(I)(B) AS A TS VIOLATION.

[120] SHEARON HARRIS 1 DOCKET 50-400 LER 91-007
TECHNICAL SPECIFICATION VIOLATION DUE TO INAPPROPRIATELY REMOVING AN EMERGENCY SERVICE WATER PUMP FROM SERVICE.
EVENT DATE: 032591 REPORT DATE: 042591 NSSS: WE TYPE: PWR

(NSIC 221870) ON 3/25/91, AT APPROXIMATELY 1900 HOURS, WITH THE PLANT IN MODE 5 DURING A REFUELING OUTAGE, ONE EMERGENCY SERVICE WATER (ESW) PUMP WAS INAPPROPRIATELY REMOVED FROM SERVICE FOR OUTAGE WORK. THE EVENT WAS DISCOVERED AT 0055 HOURS ON 3/26/91, AND OPERABILITY WAS RESTORED BY 0245 HOURS. TECHNICAL SPECIFICATIONS REQUIRE TWO OPERABLE RESIDUAL HEAT REMOVAL (RHR) LOOPS WHEN IN MODE 5 WITH REACTOR COOLANT LOOPS NOT FILLED. SITE TECHNICAL SPECIFICATION INTERPRETATIONS SPECIFY THAT COMPONENT COOLING WATER (CCW) AND ESW PUMPS AND FLOWPATHS ARE REQUIRED TO SUPPORT RHR. THIS EVENT WAS ATTRIBUTED TO

PERSONNEL/MANAGEMENT ERROR AND WILL BE REVIEWED WITH OPERATIONS MANAGEMENT PERSONNEL. TECHNICAL SPECIFICATIONS FOR ESW AND CCW WILL BE ANNOTATED TO INDICATE APPLICABILITY OF TECHNICAL SPECIFICATION INTERPRETATION 90-001 .

[121] SHOREHAM DOCKET 50-322 LER 91-001
UNPLANNED ACTUATION OF ENGINEERED SAFETY FEATURE SYSTEMS.
EVENT DATE: 032491 REPORT DATE: 042291 NSSS: GE TYPE: BWR
VENDOR: AGASTAT RELAY CO.

(NSIC 221858) ON MARCH 24, 1991 AT 1221, THE REACTOR BUILDING NORMAL VENTILATION SYSTEM (RBNVS) OUTBOARD EXHAUST VALVE (1T46*AOV-037B) CLOSED FOR NO APPARENT REASON. WITH RBNVS OPERATING IN A RECIRCULATION MODE, THIS ISOLATION OF THE EXHAUST FLOW PATH CAUSED THE REACTOR BUILDING PRESSURE TO RISE ABOVE -.30" (WATER GAUGE) WHICH IN TURN CAUSED THE UNPLANNED INITIATION OF ENGINEERED SAFETY FEATURE SYSTEMS REACTOR BUILDING STANDBY VENTILATION SYSTEM "A" AND CONTROL ROOM AIR CONDITIONING "A". THESE TWO SYSTEMS WERE RESTORED TO THEIR NORMAL LINEUPS AT 1225. PLANT MANAGEMENT PERSONNEL WERE INFORMED OF THIS EVENT AND THE NRC WAS NOTIFIED AT 1426 PER 10CFR50.72(B)(2)(II). A DEFINITE CAUSE FOR THE OUTBOARD EXHAUST VALVE CLOSING COULD NOT BE DETERMINED. THE EXHAUST VALVE ITSELF, ITS CONTROL CIRCUIT AND THE COMPONENTS THAT VENT AIR OFF THE VALVE TO ASSIST IT IN CLOSING WERE ALL INSPECTED OR TESTED BUT NOTHING ABNORMAL WAS FOUND.

[122] SOUTH TEXAS 1 DOCKET 50-498 LER 90-025 REV 01
UPDATE ON REACTOR TRIP DUE TO A GENERATOR GROUND FAULT RELAY ACTUATION CAUSED BY A STATOR COIL END TURN FAILURE.
EVENT DATE: 112490 REPORT DATE: 050191 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 221971) ON 11/24/90, UNIT 1 WAS IN MODE 1 AT 100% POWER. AT 1448 HOURS, THE GENERATOR RUNNING GROUND FAULT RELAY ACTUATED DUE TO A STATOR COIL END TURN FAILURE WHICH INITIATED AN AUTOMATIC REACTOR TRIP. ALL SYSTEMS RESPONDED AS EXPECTED. FEEDWATER ISOLATION OCCURRED ON LOW REACTOR COOLANT SYSTEM AVERAGE TEMPERATURE, AND AUXILIARY FEEDWATER (AFW) ACTUATED ON LOW - LOW STEAM GENERATOR LEVEL AS EXPECTED. INITIAL INTERNAL INSPECTION OF THE GENERATOR REVEALED DAMAGE TO THE END TURN OF BOTTOM STATOR COIL #23 AT THE TURBINE END. THE STATOR COIL END TURN FAILURE WAS DUE TO VIBRATION-INDUCED FATIGUE CRACKING BROUGHT ON BY EXCESSIVE END TURN VIBRATION WHICH RESULTED FROM STRUCTURAL DEGRADATION OF GENERATOR BLOCKING AND LACING. THE MOST PROBABLE CAUSE OF THE STRUCTURAL DEGRADATION WAS LATENT DAMAGE THAT RESULTED FROM THE JANUARY 20, 1989 GENERATOR HYDROGEN COOLING TRANSIENT (REFER TO LER 89-005). EXTENSIVE REMEDIAL ACTIONS TO REPAIR THE DAMAGE HAVE BEEN IMPLEMENTED FOR THE UNIT 1 GENERATOR. CORRECTIVE ACTIONS INCLUDE INSTALLATION OF "WINDING IMPROVEMENT MODULES;" INSTALLATION OF AN END TURN VIBRATION MONITORING SYSTEM; AND, INSTALLATION OF A DUAL TOWER HYDROGEN GAS DRYER. THE LATTER TWO ACTIONS ARE ALSO SCHEDULED TO BE IMPLEMENTED IN UNIT 2. A TURBINE-GENERATOR TASK FORCE, CONSISTING OF HL&P AND WESTINGHOUSE PERSONNEL, HAS BEEN INITIATED.

[123] SOUTH TEXAS 1 DOCKET 50-498 LER 91-005
TECHNICAL SPECIFICATION VIOLATION DUE TO OPENING THE FUEL HANDLING BUILDING (FHB) TRUCK BAY DOORS DURING FUEL MOVEMENT.
EVENT DATE: 021891 REPORT DATE: 032091 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SOUTH TEXAS 2 (PWR)

(NSIC 221278) ON 2/18/81, 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 TECH SPEC 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.

[124] SOUTH TEXAS 1 DOCKET 50-498 LER 91-009
DIESEL GENERATOR #12 AND #13 VALID FAILURES DUE TO CRACKED FUEL INJECTOR NOZZLE TIPS.
EVENT DATE: 031191 REPORT DATE: 041791 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SOUTH TEXAS 2 (PWR)
VENDOR: COOPER-BESSEMER CO.

(NSIC 221973) ON MARCH 10, 1991 UNIT 1 WAS IN A REFUELING OUTAGE IN MODE 5 AND UNIT 2 WAS OPERATING AT FULL POWER. FOLLOWING A MAINTENANCE TEST RUN OF STANDBY DIESEL GENERATOR (SDG) #12 IN UNIT 1 A CRACKED FUEL INJECTOR NOZZLE TIP WAS DISCOVERED. SUBSEQUENTLY, A MARCH 13TH RUN OF SDG #13 LED TO DISCOVERY OF A LEAKING FUEL INJECTOR NOZZLE. ON MARCH 20TH, A NONDESTRUCTIVE TEST OF THE SDG #13 INJECTOR NOZZLE TIP IDENTIFIED ANOTHER CRACKED NOZZLE TIP. NONDESTRUCTIVE EXAMINATIONS OF 151 INJECTOR NOZZLE TIPS FROM THESE AND OTHER SDGS AS WELL AS FROM SPARES IDENTIFIED SEVERAL ADDITIONAL CRACKED NOZZLE TIPS. THE CRACKING FAILURE MODE APPEARS TO BE HIGH-CYCLE FATIGUE RELATED TO A SPECIFIC LOT OF NOZZLE TIPS, HOWEVER, ADDITIONAL DETAILED ANALYSIS IS ONGOING. COOPER-BESSEMER, THE MANUFACTURER OF THE SIX SOUTH TEXAS SDGS, HAS NOTIFIED THE NRC, AS WELL AS AFFECTED CUSTOMERS, OF THIS ISSUE PURSUANT TO 10CFR21. ACTION WAS TAKEN TO REPLACE NOZZLE TIPS FROM TWO LOTS WHICH WERE CONSIDERED SUSPECT FROM EACH OF THE UNIT 1 AND UNIT 2 SDGS. HL&P HAS SENT THE CRACKED NOZZLE TIP FROM SDG #12 TO AN OFFSITE LABORATORY FOR ANALYSIS. HL&P HAS ALSO REQUESTED THAT THE COOPER-BESSEMER OWNERS GROUP INITIATE A GENERIC EVALUATION OF THE CRACKING. ONCE THESE ACTIVITIES ARE COMPLETE A SUPPLEMENTAL REPORT WILL BE PROVIDED.

[125] SOUTH TEXAS 1 DOCKET 50-498 LER 91-008
PARTIAL LOSS OF OFFSITE POWER ON TRAINS A AND B CAUSED BY INADEQUATE PROCEDURE.
EVENT DATE: 031591 REPORT DATE: 041591 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 221972) 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 WILL BE 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.

[126] SOUTH TEXAS 1 DOCKET 50-498 LER 91-010
MANUAL ENGINEERED SAFETY FEATURES ACTUATION DUE TO A TOXIC GAS ALARM.
EVENT DATE: 040491 REPORT DATE: 050291 NSSS: WE TYPE: PWR

(NSIC 221974) 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 OPTICS DATA ACQUISITION CONTROLLER SUBSYSTEM PRINTED CIRCUIT BOARD. THE FAILED PRINTED CIRCUIT BOARD HAS BEEN REPLACED AS A RESULT OF THE EVENT.

[127] SOUTH TEXAS 2 DOCKET 50-499 LER 91-003
REACTOR TRIP CAUSED BY ACTUATION OF A GENERATOR PROTECTIVE RELAY.
EVENT DATE: 031491 REPORT DATE: 041591 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SOUTH TEXAS 1 (PWR)

(NSIC 221975) 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.

[128] SOUTH TEXAS 2 DOCKET 50-499 LER 91-004
REACTOR TRIP CAUSED BY ACTUATION OF A GENERATOR PROTECTIVE RELAY.
EVENT DATE: 033091 REPORT DATE: 042991 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SOUTH TEXAS 1 (PWR)

(NSIC 221976) 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.

[129] ST. 2 DOCKET 50-389 LER 91-003
2A SHUTDOWN COC , HEAT EXCHANGER OUT OF SERVICE DUE TO MISPOSITIONED COMPONENT COOLING WATER VALVE CAUSED BY PERSONNEL ERROR.
EVENT DATE: 042691 REPORT DATE: 043091 NSSS: CE TYPE: PWR
VENDOR: PRATT, HENRY COMPANY

(NSIC 221959) THIS IS AN INTERIM REPORT. A FOLLOWUP REPORT WILL BE SUBMITTED. AT 0110 ON APRIL 26, 1991, WITH UNIT 2 AT 100% POWER, OPERATIONS PERSONNEL BEGAN SEARCHING FOR A DC GROUND. AT 0400, PER PLANT PROCEDURE, OPERATIONS CYCLED HCV-14-3A, COMPONENT COOLING WATER (CCW) OUTLET FROM THE 2A SHUTDOWN COOLING (SDC) HEAT EXCHANGER (HX) TO DE-ENERGIZE ITS SOLENOID OPERATOR IN AN EFFORT TO LOCATE THE DC GROUND. AFTER THE VALVE OPENED, NO FLOW THROUGH THE HEAT EXCHANGER WAS INDICATED. UPON INVESTIGATION, THE 2A SDC HX CCW RETURN ISOLATION VALVE SB-14365 WAS FOUND TO BE LOCKED CLOSED. THIS VALVE IS REQUIRED TO BE LOCKED OPEN. IT HAD BEEN ENTERED INTO THE VALVE SWITCH DEVIATION LOG ON OCTOBER 23, 1990 AS LOCKED THROTTLED AND RESTORED NOVEMBER 29, 1990. AS THIS IS THE MOST RECENT DOCUMENTED MANIPULATION DATE, IT IS ASSUMED TO HAVE BEEN MISPOSITIONED AT THIS TIME. THE VALVE POSITION POINTER WAS BROKEN AND INDICATED OPEN. THE CAUSE OF THE MISPOSITIONING IS UNDER INVESTIGATION. CORRECTIVE ACTIONS INCLUDE: THE VALVE WAS CORRECTLY REALIGNED; THE REDUNDANT TRAIN'S VALVE WAS CHECKED; BOTH UNITS PERFORMED A FULL VALVE STATUS CHECK.

[130] SUMMER 1 DOCKET 50-395 LER 90-009 REV 01
UPDATE ON DESIGN DEFECT IN THE CHILLED WATER SYSTEM.
EVENT DATE: 102690 REPORT DATE: 042991 NSSS: WE TYPE: PWR

(NSIC 221960) ON OCTOBER 26, 1990, A 10CFR21 NOTIFICATION WAS MADE BY GILBERT/COMMONWEALTH, INC., REGARDING A DESIGN DEFECT IN THE VIRGIL C. SUMMER NUCLEAR STATION (VCSNS) CHILLED WATER SYSTEM. THIS DESIGN DEFECT WAS DISCOVERED WHILE EVALUATING THE NON-ESSENTIAL HEADER ISOLATION VALVE STROKE TIME REQUIREMENTS. THE DESIGN DEFECT INVOLVES THE INABILITY OF THE EXPANSION TANK INSTRUMENTATION TO DETECT A LOSS OF INVENTORY DUE TO A POSTULATED FAILURE OF THE NON-ESSENTIAL HEADER AND ACTUATE THE NON-ESSENTIAL HEADER ISOLATION VALVES. THIS COULD RESULT IN A LOSS OF COOLING TO BOTH TRAINS OF THE CHARGING/SAFETY INJECTION PUMPS AND THE COMPONENT COOLING PUMPS. UPON RECEIPT OF THE 10CFR21 NOTIFICATION, THE ISOLATION VALVES WERE IMMEDIATELY CLOSED AND PLACED UNDER THE CONTROL OF STATION ADMINISTRATION PROCEDURE 201, "DANGER TAGGING." A PERMANENT MODIFICATION IS BEING IMPLEMENTED TO CORRECT THE DESIGN DEFECT.

[131] SURRY 1 DOCKET 50-280 LER 91-002
TWO CHARGING PUMPS AND ONE CHARGING PUMP SERVICE WATER PUMP INOPERABLE SIMULTANEOUSLY DUE TO INSTRUMENT AIR LINE FAILURE CAUSED BY PERSONNEL ERROR.
EVENT DATE: 032691 REPORT DATE: 042391 NSSS: WE TYPE: PWR

(NSIC 221900) ON 3/26/91, WITH UNIT 1 OPERATING AT 100% POWER, CHARGING PUMP 1-CH-P-1B WAS DECLARED INOPERABLE DUE TO A FAILED INSTRUMENT AIR LINE TO ITS LUBE OIL TEMPERATURE CONTROL VALVE, 1-SW-TCV-108B. THIS RESULTED IN TWO OF THREE CHARGING PUMPS BEING CONSIDERED INOPERABLE, SINCE THE "A" AND "C" PUMPS WERE BOTH ALIGNED TO THE SAME EMERGENCY BUS. AT THE TIME THE "B" CHARGING PUMP WAS DECLARED INOPERABLE, CHARGING PUMP SERVICE WATER PUMP 1-SW-P-10A WAS OUT OF SERVICE FOR MAINTENANCE. WITH TWO CHARGING PUMPS AND ONE CHARGING PUMP SERVICE WATER PUMP INOPERABLE SIMULTANEOUSLY, A CONDITION NOT ALLOWED BY TECH SPECS EXISTED AND A SIX HOUR ACTION STATEMENT WAS ENTERED IN ACCORDANCE WITH TECH SPEC 3.0.1. THE FAILED INSTRUMENT AIR LINE WAS ISOLATED. THE "C" CHARGING PUMP WAS TRANSFERRED TO THE REDUNDANT BUS AT 1443 HOURS AND THE SIX HOUR ACTION STATEMENT WAS EXITED. THE "B" CHARGING PUMP COULD HAVE BEEN OPERATED IF SAFETY INJECTION HAD BEEN REQUIRED AND THERE WERE NO ACTUAL OR POTENTIAL CONSEQUENCES TO PUBLIC HEALTH AND SAFETY. EVENT WAS CAUSED BY PERSONNEL ERROR. A CONTRACTOR-EMPLOYED PAINTING FOREMAN INADVERTENTLY MADE CONTACT WITH AND CAUSED FAILURE OF THE INSTRUMENT AIR LINE DURING PAINTING OF THE CHARGING PUMP CUBICLE. THE EVENT IS BEING COMMUNICATED TO STATION AND CONSTRUCTION CRAFT PERSONNEL. THIS REPORT IS REQUIRED BY 10 CFR 50.73(A)(2)(I)(B).

[132] SURRY 2 DOCKET 50-281 LER 91-002
MAIN STEAM SAFETY VALVES OUT OF TOLERANCE DUE TO MINOR SETPOINT DRIFT.
EVENT DATE: 032691 REPORT DATE: 042391 NSSS: WE TYPE: PWR
VENDOR: CONSOLIDATED VALVE CORP.

(NSIC 221901) ON MARCH 26, 1991, WITH UNIT 2 AT 75% POWER DURING END OF FUEL CYCLE COAST DOWN, SETPOINT TESTING OF THE FIFTEEN MAIN STEAM CODE SAFETY VALVES REVEALED THAT THE LIFT SETTING OF 2-MS-SV-205B WAS OUTSIDE OF THE +/- 3% "AS FOUND" RANGE ALLOWED BY SURRY'S TECHNICAL SPECIFICATIONS. THE VALVE EXHIBITED A LIFT SETTING OF 3.3% BELOW ITS NOMINAL VALUE (1135 PSIG). THE SAFETY VALVE WAS READJUSTED TO BE WITHIN THE REQUIRED SETPOINT TOLERANCE. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(I)(B) FOR CONDITIONS PROHIBITED BY TECHNICAL SPECIFICATIONS 3.6.A, 3.6.B.3 AND 3.6.D.

[133] SUSQUEHANNA 1 DOCKET 50-387 LER 91-003 REV 01
UPDATE ON FIRE DAMPER NOT INSTALLED IN FIRE RATED BARRIER.
EVENT DATE: 021491 REPORT DATE: 050291 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: SUSQUEHANNA 2 (BWR)

(NSIC 221958) ON FEBRUARY 14, 1991 WITH UNIT 1 AND UNIT 2 OPERATING IN CONDITION 1 AT 100% AND 99% POWER RESPECTIVELY, IT WAS IDENTIFIED THAT A FIRE RATED DAMPER WAS NOT FOUND IN A FIRE RATED BARRIER WHICH IS REQUIRED TO MEET TECHNICAL

SPECIFICATION 3.7.7, AND SINCE IT CANNOT BE CONFIRMED THAT A CONTINUOUS FIREWATCH WAS IN PLACE TO MEET THE ACTION STATEMENT FOR THIS TECHNICAL SPECIFICATION DURING THE ENTIRE TIME THE DAMPER WAS MISSING, IT WAS DETERMINED THAT THE CONDITION CONSTITUTED AN OPERATION PROHIBITED BY THE TECHNICAL SPECIFICATIONS. A CONTINUOUS FIREWATCH WAS ESTABLISHED. THIS CONDITION IS BEING EVALUATED TO DETERMINE THE REQUIREMENT FOR THIS FIRE BARRIER. THE INITIAL EVALUATION APPEARED TO INDICATE THAT THIS FIRE BARRIER WAS NOT REQUIRED. FURTHER EVALUATION HAS VERIFIED THAT THIS FIRE BARRIER DOES NOT SEPARATE REDUNDANT SAFE SHUTDOWN EQUIPMENT OR CABLES, HAS NO EFFECT ON THE SAFE SHUTDOWN ANALYSIS, AND DOES NOT PROTECT SAFETY RELATED EQUIPMENT. BASED UPON THIS ANALYSIS, NO OPERATION PROHIBITED BY THE TECHNICAL SPECIFICATIONS EXISTED.

[134] SUSQUEHANNA 1 DOCKET 50-387 LER 91-004
ESF ACTUATIONS DUE TO RPS ELECTRICAL PROTECTION ASSEMBLY BREAKERS SPURIOUS TRIP.
EVENT DATE: 032191 REPORT DATE: 042291 NSSS: GE TYPE: BWR

(NSIC 221868) AT 1415 HOURS ON 3/21/91 WITH UNIT 1 OPERATING AT 100% POWER, THE PRIMARY POWER SUPPLY TO THE "A" REACTOR PROTECTION SYSTEM (RPS) POWER DISTRIBUTING PANEL WAS LOST WHEN ITS ELECTRICAL PROTECTION ASSEMBLY (EPA) BREAKERS TRIPPED. THIS INTERRUPTION OF POWER TO "A" RPS, PER DESIGN, CAUSED REACTOR BUILDING HVAC ZONES I AND III, THE REACTOR WATER CLEANUP SYSTEM, AND VARIOUS OTHER PRIMARY CONTAINMENT ISOLATION SYSTEM COMPONENTS TO ISOLATE AND THE STANDBY GAS TREATMENT SYSTEM AND CONTROL ROOM EMERGENCY OUTSIDE AIR SUPPLY SYSTEM TO AUTO INITIATE. ALL PLANT SYSTEMS AND COMPONENTS FUNCTIONED PROPERLY AND AS EXPECTED IN RESPONSE TO THE EVENT. NO REACTOR PARAMETERS WERE AFFECTED AND NO EMERGENCY CORE COOLING SYSTEMS WERE ACTUATED. THE EPA BREAKERS WERE RESET AT 1420 HOURS WITH NO INDICATION OF ABNORMALITIES AND ALL ISOLATION SIGNALS WERE RESET BY 1502 HOURS. FULL POWER OPERATION OF THE UNIT CONTINUED WITHOUT INTERRUPTION. THE CAUSE OF THE BREAKER TRIP WILL BE INVESTIGATED DURING THE NEXT UNIT OUTAGE. AN EPA STUDY PREPARED BY GE - NUCLEAR ENERGY FOR THE BWR OWNERS GROUP IS CURRENTLY UNDER PP&L REVIEW TO ASSESS ANY OPERATIONAL OR DESIGN CHANGES THAT MAY BE WARRANTED. LER 90-007 (DOCKET NO. 50-388/LICENSE NO. NPF-22) WILL BE UPDATED TO PROVIDE THE RESULTS OF THIS ASSESSMENT. ALL ESF SYSTEMS AND COMPONENTS FUNCTIONED PROPERLY AND PER DESIGN.

[135] THREE MILE ISLAND 2 DOCKET 50-320 LER 91-003
PROCESSED WATER DISPOSAL SYSTEM VALVE MISALIGNMENT.
EVENT DATE: 040391 REPORT DATE: 050391 NSSS: BW TYPE: PWR

(NSIC 221995) ON APRIL 3, 1991, THE TMI-2 PROCESSED WATER DISPOSAL SYSTEM (PWDS) WAS OPERATING IN THE "COUPLED MODE" (I.E., EVAPORATOR COUPLED TO THE VAPORIZER) WHEN A VAPORIZER BLOW-DOWN VALVE WAS FOUND CLOSED. THE PURPOSE OF THIS FLOW PATHWAY IS TO DRAW-OFF SOLIDS FROM THE VAPORIZER DISPOSAL. THE CLOSURE OF THE BLOW-DOWN VALVE HAS THE POTENTIAL TO ADVERSELY IMPACT THE SYSTEM DECONTAMINATION FACTOR (DF) OF THE PWDS; HOWEVER, NO IMPACT WAS OBSERVED. THE EVENT WAS A RESULT OF PERSONNEL ERROR IN THAT THE EVAPORATOR OPERATOR DID NOT FOLLOW THE OPERATING PROCEDURE. A CONTRIBUTING FACTOR WAS AN UNWIELDY PROCEDURE THAT DID NOT LEND ITSELF TO EASE OF USE. UPON DISCOVERY OF THE MISPOSITIONED VALVE, THE IMMEDIATE CORRECTIVE ACTION TAKEN WAS TO OPEN THE VALVE AND NOTIFY THE TMI-2 CONTROL ROOM. LONGER TERM CORRECTIVE ACTIONS INCLUDE ADDITIONAL TRAINING FOR THE EVAPORATOR OPERATORS AND A REVISION TO THE PWDS OPERATING PROCEDURE. TMI-2 TECH SPEC 3.9.13 STATES, "ACCIDENT GENERATED WATER SHALL BE DISPOSED OF IN ACCORDANCE WITH NRC-APPROVED PROCEDURES." PER THE NRC-APPROVED PWDS OPERATING PROCEDURE, THE BLOW-DOWN VALVE WAS REQUIRED TO BE OPEN DURING COUPLED MODE OPERATIONS. THEREFORE, PWDS OPERATION IN THIS MANNER, ALTHOUGH MODE OPERATIONS. THEREFORE, PWDS OPERATION IN THIS MANNER, ALTHOUGH INADVERTENT, WAS PROHIBITED BY THE PLANT'S TECH. SPECS. SIMILAR EVENT: LER 91-02.

[136] THREE MILE ISLAND 2 DOCKET 50-320 LER 91-004
PROCESSED WATER DISPOSAL SYSTEM SAMPLE VALVE MISALIGNMENT.
EVENT DATE: 041291 REPORT DATE: 051091 NSSS: BW TYPE: PWR

(NSIC 221996) ON 4/12/91, THE TMI-2 PROCESSED WATER DISPOSAL SYSTEM (PWDS) WAS

OPERATING IN THE "COUPLED MODE" (I.E., EVAPORATOR COUPLED TO THE VAPORIZER) WHEN A VAPORIZER EXHAUST SAMPLE VALVE WAS FOUND CLOSED. A SAMPLE OF THE VAPORIZER EXHAUST IS ROUTINELY ANALYZED TO CALCULATE THE SYSTEM DECONTAMINATION FACTOR (DF); IT IS NOT USED TO DETERMINE RADIO-NUCLIDE RELEASE. THIS EVENT WAS A RESULT OF PERSONNEL ERROR IN THAT THE EVAPORATOR OPERATOR DID NOT FOLLOW THE OPERATING PROCEDURE. CONTRIBUTING FACTORS WERE THE UNWIELDY NATURE OF THE RELEVANT PROCEDURE (I.E., IT DOES NOT LEND ITSELF TO EASE OF USE) AND THE DECISION TO HAVE THE TMI-2 CONTROL ROOM SIGN-OFF ON A SEPARATE COPY OF THE PROCEDURE. UPON DISCOVERY OF THE MISPOSITIONED VALVE, CORRECTIVE ACTIONS TAKEN WERE TO OPEN THE VALVE AND INITIATE AN INVESTIGATION INTO THE CAUSE OF THE MISPOSITIONING. THE PWDS WAS SUBSEQUENTLY SHUT DOWN. LONGER TERM CORRECTIVE ACTIONS INCLUDE ADDITIONAL TRAINING FOR THE EVAPORATOR OPERATORS AND A REVISION TO THE PWDS OPERATING PROCEDURE. TMI-2 TECH. SPEC. 3.9.13 STATES, "ACCIDENT GENERATED WATER SHALL BE DISPOSED OF IN ACCORDANCE WITH NRC-APPROVED PROCEDURES." PER THE NRC-APPROVED PWDS OPERATING PROCEDURE, THE VAPORIZER EXHAUST SAMPLE VALVE WAS REQUIRED TO BE OPEN DURING COUPLED MODE OPERATIONS. SIMILAR EVENTS: LERS 91-02 AND 91-03.

[137] TROJAN DOCKET 50-344 LER 90-006 REV 02
UPDATE ON BOTH TRAINS OF EMERGENCY CORE COOLING SYSTEM WERE INOPERABLE DURING MODE 3 SURVEILLANCE TESTING TO PROCEDURAL INADEQUACY.
EVENT DATE: 022190 REPORT DATE: 041791 NSSS: WE TYPE: PWR

(NSIC 221948) ON 2/21/90 THE PLANT WAS IN MODE 1, POWER OPERATION, WITH THE REACTOR COOLANT SYSTEM (RCS) AT 584F, AND 2239 PSIG. DURING A REVIEW OF CORRECTIVE ACTION FOR A NRC OPEN ITEM, IT WAS NOTED THAT A REVISION OF A PERIODIC OPERATING TEST (POT), CAUSED BOTH TRAINS OF EMERGENCY CORE COOLING SYSTEM (ECCS) TO BE INOPERABLE WHEN SECTION 7.6 OF POT 2-4 WAS PERFORMED IN MODE 3, HOT STANDBY, ON 7/24/89. THIS CONDITION VIOLATED TROJAN TECH SPEC (TTS) 3.5.2, ECCS SUBSYSTEM - TAVG 350F, AND CONSTITUTED AN UNINTENTIONAL ENTRY INTO TTS 3.0.3. THE INITIAL CONDITION FOR PERFORMANCE OF THE POT WAS RCS PRESSURE OF AT LEAST 1800 PSIG, WHICH CORRESPONDS TO MODE 3 OR MODE 4 ABOVE 260F. THE POT DID NOT DIFFERENTIATE BETWEEN THE TWO MODES WITH RESPECT TO TTS REQUIREMENTS FOR ECCS OPERABILITY. THE CAUSE OF THE OCCURRENCE WAS AN INADEQUATE PROCEDURE DUE TO INADEQUATE TECHNICAL AND SAFETY REVIEW OF THE REVISION. CORRECTIVE ACTIONS INCLUDE REVIEW AND REVISION OF THE POT TO PREVENT UNINTENTIONAL TTS 3.0.3 ENTRY. ALSO, A TASK FORCE WAS ASSEMBLED TO REVIEW OTHER POTS TO ENSURE ADDITIONAL SIMILAR CONDITIONS DID NOT EXIST WHICH WOULD ALLOW TESTING OF ECCS SYSTEMS TO CAUSE INADVERTENT ENTRIES INTO TTS 3.0.3. BOTH TRAINS OF ECCS WERE INOPERABLE DURING THE PERFORMANCE OF THE SURVEILLANCE TEST BUT COULD HAVE BEEN QUICKLY REALIGNED HAD IT BEEN NECESSARY.

[138] TROJAN DOCKET 50-344 LER 90-038 REV 01
UPDATE ON INCOMPLETE PERFORMANCE OF SURVEILLANCES DUE TO PERSONNEL ERRORS IN INTERPRETATION OF TECHNICAL SPECIFICATIONS AND IN DEVELOPING PROCEDURE REVISION.
EVENT DATE: 090790 REPORT DATE: 042191 NSSS: WE TYPE: PWR

(NSIC 221860) ON 9/7/90 THE PLANT WAS IN MODE 1 AT A NOMINAL GENERATOR LOAD OF 1140 MWE. WHILE PERFORMING A DESIGN REVIEW, A DESIGN ENGINEER DISCOVERED THAT A PORTION OF THE TURBINE TRIP ON REACTOR TRIP CIRCUIT WAS NOT TESTED BY EXISTING TROJAN TECH SPEC SURVEILLANCE PROCEDURES. PORTION OF THE CIRCUIT NOT TESTED CONTAINED WIRING AND WIRING CONNECTIONS, NOT ELECTRO-MECHANICAL COMPONENTS. EVALUATION DETERMINED THAT SURVEILLANCE TESTING OF THE FEEDWATER ISOLATION FUNCTION UPON A REACTOR TRIP WAS TESTED EVERY OTHER ANNUAL REFUELING, RATHER THAN ONCE PER 18 MONTHS AS REQUIRED BY TROJAN TECH SPEC 3/4.3.2, "ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION". PRIMARY CAUSES OF THESE EVENTS WERE AN INADEQUATE REVIEW OF PROCEDURES TO IDENTIFY NEEDED CHANGES TO IMPLEMENT A TROJAN TECH SPEC AMENDMENT AND AN INCORRECT INTERPRETATION OF THE REQUIRED SURVEILLANCE FREQUENCY FOR CHANNELS WHICH INITIATE FEEDWATER ISOLATION FUNCTION UPON A REACTOR TRIP. REQUIRED SURVEILLANCES WERE PERFORMED DURING THE 9/25/90 FORCED OUTAGE. CORRECTIVE ACTIONS WHICH HAVE BEEN OR WILL BE TAKEN INCLUDE STRENGTHENING ADMINISTRATIVE CONTROLS FOR IDENTIFICATION OF PROCEDURE CHANGES NEEDED DUE TO TROJAN TECH SPEC AMENDMENT AND REVIEWING PLANT PROCEDURES TO ENSURE

THAT OTHER INTERLOCK FUNCTIONS OF THE SOLID STATE PROTECTION SYSTEM ARE TESTED AS REQUIRED.

[139] TROJAN DOCKET 50-344 LER 91-003
DILUTION OF SAMPLE STREAM DURING VENTING OF CONTAINMENT BUILDING RESULTS IN IMPROPERLY MONITORED RELEASE.
EVENT DATE: 013091 REPORT DATE: 030191 NSSS: WE TYPE: PWR

(NSIC 221158) AT 2024 ON JANUARY 30, 1991, WITH THE PLANT IN MODE 1 (POWER OPERATION), VENTING OF AIR FROM THE CONTAINMENT BUILDING WAS BEGUN TO REDUCE PRESSURE IN THE BUILDING. ON JANUARY 31, 1991, A CHEMISTRY EFFLUENT ANALYST REVIEWING THE PREVIOUS DAY'S CONTAINMENT MONITOR SYSTEM DATA NOTICED THAT THE NOBLE GAS COUNT RATE WAS APPROXIMATELY 1,500 COUNTS PER MINUTE (CPM) BEFORE THE EVOLUTION AND APPROXIMATELY 400 CPM AFTER THE EVOLUTION WAS BEGUN. VENTING WAS STOPPED AT 0708. SUBSEQUENT INSPECTION OF THE FILTER ADSORBER CARTRIDGE HOLDER OF THE IODINE MONITOR REVEALED THAT THE O-RING WHICH PROVIDES A SEAL AGAINST AIR LEAKAGE INTO THE SYSTEM WAS OUT OF ITS GROOVE THROUGH APPROXIMATELY 90 DEGREES OF ARC. THE O-RING WAS RETURNED TO ITS GROOVE, AND THE HOLDER WAS REINSTALLED. EXPECTED COUNT RATES WERE THEN OBSERVED. ANALYSIS OF THE NOBLE GAS EFFLUENT DATA SHOWED THAT THERE WAS A SAMPLE DILUTION FACTOR OF APPROXIMATELY 5.9 INTRODUCED BY THE LEAKAGE INTO THE SAMPLE STREAM. THUS, THE DILUTION OF THE SAMPLE STREAM WOULD HAVE ALLOWED ACTUAL CONCENTRATIONS OF MONITORED EFFLUENTS TO BE GREATER THAN THE 2 TIMES BACKGROUND ALLOWED BY TROJAN TECHNICAL SPECIFICATIONS BEFORE AUTOMATIC ISOLATION OF THE CONTAINMENT BUILDING VENTILATION WOULD HAVE OCCURRED.

[140] TROJAN DOCKET 50-344 LER 91-009
DEFICIENT PROCEDURES CAUSE ENVIRONMENTAL QUALIFICATION OF ELECTRICAL COMPONENTS LOCATED IN UNENCLOSED SPACES TO BE INDETERMINANT FOR LOW TEMPERATURE SERVICE CONDITIONS.
EVENT DATE: 031591 REPORT DATE: 041591 NSSS: WE TYPE: PWR
VENDOR: AUTOMATIC SWITCH COMPANY (ASCO)
NAMCO CONTROLS

(NSIC 221949) ON MARCH 15, 1991 WHILE IN MODE 5 (COLD SHUTDOWN) DURING A REVIEW TO ADDRESS EQUIPMENT QUALIFICATION CONCERNS IDENTIFIED ON MARCH 7, 1991, PORTLAND GENERAL ELECTRIC COMPANY (PGE) DETERMINED THAT QUALIFICATION INFORMATION TO QUALIFY ASCO CATALOG NO. NP-1 SOLENOID VALVES FOR LOW-TEMPERATURE (-5F) SERVICE CONDITIONS WAS NOT AVAILABLE. THE CAUSE OF THESE DEFICIENCIES HAS NOT YET BEEN DETERMINED. HOWEVER, PRELIMINARY INDICATIONS ARE THAT PROCEDURES WERE DEFICIENT BECAUSE OF A FAILURE TO RECOGNIZE THE MORE STRINGENT NORMAL OPERATING TEMPERATURE REQUIREMENTS OF NUREG-0588 VERSUS THE DIVISION OF OPERATING REACTORS (DOR) GUIDELINES WHICH WERE USED FOR PREVIOUS EQUIPMENT EVALUATIONS. THE ASSESSMENT OF THESE DEFICIENCIES, INCLUDING SAFETY SIGNIFICANCE DURING PREVIOUS POWER OPERATION, AND THE DETERMINATION OF THE SCOPE OF THESE QUALIFICATION WEAKNESSES ARE ONGOING. CORRECTIVE ACTION TO SUPPORT PLANT OPERATION WILL BE COMPLETED PRIOR TO THE END OF THE CURRENT REFUELING OUTAGE. THE LOW-TEMPERATURE QUALIFICATION ISSUE HAS LIMITED SAFETY SIGNIFICANCE AT THE PRESENT TIME DUE TO THE PLANT BEING SHUT DOWN AND THE RELATIVELY MODERATE TEMPERATURE CURRENTLY BEING EXPERIENCED AT THE TROJAN SITE.

[141] VERMONT YANKEE DOCKET 50-271 LER 91-008
POTENTIAL ENVIRONMENTAL CONDITIONS NOT PREVIOUSLY EVALUATED AS A RESULT OF OMISSION FROM ORIGINAL LINE BREAK ANALYSES.
EVENT DATE: 032591 REPORT DATE: 050191 NSSS: GE TYPE: BWR

(NSIC 221921) ON 03/25/91, AT APPROXIMATELY 1805 HOURS WITH THE PLANT AT 100% POWER, IT WAS DETERMINED THAT FAILURE OF HOUSE HEATING STEAM LINES HAD NOT BEEN ADDRESSED IN PREVIOUS HIGH ENERGY LINE BREAK (HELB) ANALYSES. THIS COULD CREATE ENVIRONMENTAL CONDITIONS OUTSIDE THOSE PREVIOUSLY EVALUATED. THIS WAS IDENTIFIED AS A RESULT OF AN INVESTIGATION RELATIVE TO USNRC INFO. NOTICE 90-053 THAT NOTIFIED LICENSEES OF SIMILAR CONDITIONS AT OTHER FACILITIES. A JUSTIFICATION FOR CONTINUED OPERATION (JCO) WAS PREPARED AND APPROVED ON 04/01/91, ALLOWING CONTINUED OPERATION OF THE HEATING SYSTEM FOR THE REMAINDER OF THE HEATING

SEASON. THE HOUSE HEATING SYSTEM WAS ERRONEOUSLY OMITTED FROM HELB ANALYSES PERFORMED IN 1973 AND 1974. A CORRECTIVE ACTION PLAN WILL BE DEVELOPED, EVALUATED AND IMPLEMENTED; AND A REVIEW WILL BE PERFORMED OF OTHER SYSTEMS TO DETERMINE IF THERE WERE ANY OTHER SIMILAR OMISSIONS FROM THE HELB ANALYSES.

[142] WATERFORD 3 DOCKET 50-382 LER 89-007 REV 01
UPDATE ON INADEQUATE DESIGN OF AIR ACCUMULATORS DUE TO INCOMPLETE REVIEW OF POST-TMI ACTION PLAN.
EVENT DATE: 121884 REPORT DATE: 030891 NSSS: CE TYPE: PWR

(NSIC 221201) AT 1100 HOURS ON MARCH 31, 1989, WATERFORD STEAM ELECTRIC UNIT 3 WAS OPERATING AT 100% POWER WHEN THE ISSUE OF REPORTABILITY WAS RAISED ON THE SIZING OF THE INSTRUMENT AIR (IA) ACCUMULATORS WHICH SUPPLY THE SAFETY INJECTION (SI) RECIRCULATION SUMP OUTLET ISOLATION VALVES, SI-602A&B. DESIGN REQUIREMENTS DID NOT CONSIDER CERTAIN ACCIDENT SCENARIOS, WITH A POSTULATED LOSS OF IA WHERE OPERATION OF THE VALVES MAY BE REQUIRED. MANUAL OPERATION OF THE VALVES WAS NOT CONSIDERED AN ADEQUATE BACKUP DUE TO POTENTIAL RADIATION LEVELS AT THE VALVE LOCATION. THEREFORE, THE PLANT WAS OPERATED IN AN UNANALYZED CONDITION SINCE INITIAL STARTUP. ON FEBRUARY 6, 1991, A REVIEW OF SURVEILLANCE PROCEDURES REVEALED THAT THE PLANT WAS OPERATED WITH A NITROGEN ACCUMULATOR IV LEAKAGE RATE OF 57.6 PSI/HR VICE THE 55 PSI/HR REQUIRED BY DESIGN BASIS DOCUMENTATION (DBD). ACCUMULATOR IV SUPPLIES NITROGEN TO OPERATE THE SI PUMP SUCTION VALVES TO THE REFUELING WATER STORAGE POOL ON LOSS OF IA. THIS CONDITION EXISTED FROM NOVEMBER 23, 1990 THROUGH FEBRUARY 7, 1991. THE ROOT CAUSE OF THIS EVENT WAS AN INADEQUATE REVIEW OF DESIGN REQUIREMENTS IMPLEMENTED AS PART OF THE POST-TMI ACTION PLAN. PHASE ONE OF DC 3195 HAS BEEN IMPLEMENTED TO PROVIDE A NITROGEN SOURCE OF GAS.

[143] WATERFORD 3 DOCKET 50-382 LER 91-004
FAILURE TO COMPLETE TECHNICAL SPECIFICATION SURVEILLANCE DUE TO INADEQUATE ATTENTION TO DETAIL.
EVENT DATE: 040491 REPORT DATE: 050391 NSSS: CE TYPE: PWR

(NSIC 221957) ON APRIL 4, 1991, WATERFORD STEAM ELECTRIC STATION UNIT 3 WAS IN MODE 6, REFUEL 4, WHEN THE TECHNICAL SPECIFICATION SURVEILLANCE OP-903-099, SPENT FUEL HANDLING MACHINE OPERABILITY, WAS NOT PERFORMED SATISFACTORILY. THE ACCEPTANCE CRITERIA FOR THE SURVEILLANCE WAS NOT COMPLETED WITHIN 72 HOURS PRIOR TO THE START OF FUEL MOVEMENT. TECHNICAL SPECIFICATION 3.9.7, SURVEILLANCE REQUIREMENT 4.9.7.1 STATES THAT THE FUEL HANDLING MACHINE SHALL BE DEMONSTRATED OPERABLE WITHIN 72 HOURS PRIOR TO THE START OF FUEL ASSEMBLY MOVEMENT. FUEL MOVEMENT BEGAN BEFORE THE SUPERVISOR REVIEWED AND APPROVED THE SURVEILLANCE RESULTS. THEREFORE, THE PLANT OPERATED IN A CONDITION PROHIBITED BY TECHNICAL SPECIFICATIONS. THE ROOT CAUSE OF THIS EVENT IS INADEQUATE ATTENTION TO DETAILS. TECHNICAL SPECIFICATION REQUIREMENT 4.9.7.1, TO PERFORM THE SURVEILLANCE WITHIN 72 HOURS PRIOR TO THE START OF FUEL ASSEMBLY MOVEMENT, WAS NOT PROPERLY COMPLETED. TO PREVENT RECURRENCE, THE INDIVIDUAL WHO PERFORMED THE SURVEILLANCE WAS COUNSELED ON THE REQUIREMENTS OF PROCEDURE OP-903-099 AND TECHNICAL SPECIFICATION 3.9.7. BECAUSE THE FUEL HANDLING MACHINE WAS SUBSEQUENTLY TESTED SATISFACTORY AND DEMONSTRATED OPERABLE, THIS EVENT DID NOT THREATEN THE HEALTH AND SAFETY OF THE GENERAL PUBLIC OR PLANT PERSONNEL.

[144] WOLF CREEK 1 DOCKET 50-482 LER 91-003
FAILURE TO PERFORM ASME SECTION XI REQUIRED VISUAL INSPECTION OF ASME CODE CLASS 2 PIPING.
EVENT DATE: 020291 REPORT DATE: 030491 NSSS: WE TYPE: PWR

(NSIC 221508) ON FEBRUARY 1, 1991, IT WAS DISCOVERED THAT AN ASME CODE CLASS 2 PORTION OF THE FUEL POOL COOLING AND CLEANUP SYSTEM WAS NOT INCLUDED IN THE PROCEDURE FOR THE ASME SECTION XI REQUIRED VISUAL INSPECTION. THE FAILURE TO EXAMINE THESE LINES IS NOT IN ACCORDANCE WITH TECHNICAL SPECIFICATION 4.0.5. PERFORMANCE OF VALVE LEAK CHECKS AND LOCAL LEAK RATE TESTS HAVE INDICATED THAT THIS PIPING WOULD HAVE MET THE CRITERIA REQUIRED BY ASME SECTION XI FOR A VISUAL EXAMINATION OF ASME CODE CLASS 2 PIPING. THIS EVENT WAS THE RESULT OF A PERSONAL ERROR DURING PROCEDURAL DEVELOPMENT OF THE SURVEILLANCE PROCEDURE. A RANDOM

REVIEW OF SURVEILLANCE PROCEDURES COVERING A WIDE VARIETY OF PLANT SYSTEMS WAS CONDUCTED TO ENSURE THAT ALL ASME CODE CLASS 1, 2, AND 3 COMPONENTS WERE INCLUDED. NO SIMILAR SITUATIONS WERE FOUND DURING THE REVIEW. SURVEILLANCE PROCEDURE STS PE-048C, REFUELING POOL SKIMMER SYSTEM PRESSURE TEST", WILL BE REVISED TO INCORPORATE THE ASME CODE CLASS 2 PIPING BY MAY 1, 1991. ALSO, THE ASME SECTION XI VISUAL EXAMINATION ON THIS PIPING WILL BE PERFORMED DURING REFUEL V.

[145] WPPSS 2 DOC[illegible] LER 90-022 REV 01
UPDATE ON ESF ACTUATION OF CONTAINMENT INSTRUME[illegible] DEPLETING THE NITROGEN CRYOGENIC TANK.
EVENT DATE: 093090 REPORT DATE: 042291 NSS[illegible] TYPE: BWR

(NSIC 221869) AT 1600 HOURS ON 9/30/90, WHILE PLANT [illegible] INERTING THE PRIMARY CONTAINMENT DURING A REACTOR STARTUP, A PRESSURE DECREASE OCCURRED IN THE CONTAINMENT INSTRUMENT AIR (CIA) SYSTEM. THE PRESSURE LOSS OCCURRED WHEN THE NITROGEN CRYOGENIC TANK (CN-TK-1) (THE NORMAL SUPPLY USED FOR CONTAINMENT INERTING AND FOR CIA) WAS INADVERTENTLY DEPLETED. THIS PRESSURE DECREASE CAUSED THE SAFETY RELATED PART OF THE CIA SYSTEM TO BE ISOLATED AND AUTOMATICALLY PLACED THE BACKUP BOTTLED NITROGEN SOURCE INTO SERVICE. THIS ACTION IS CONSIDERED AN ENGINEERED SAFETY FEATURE ACTUATION. FURTHER EVALUATION SHOWED THAT THE PRESSURE MAINTAINED BY THE BOTTLED NITROGEN SOURCE IN DIVISION II DID NOT MEET DESIGN REQUIREMENTS BECAUSE OF A MISADJUSTED PRESSURE REGULATOR. THE ROOT CAUSE OF THE DEPLETED NITROGEN SUPPLY IN CN-TK-1 WAS LESS THAN ADEQUATE PROCEDURES THAT DID NOT CONTAIN PRECAUTIONS FOR CONTAINMENT INERTING WITH LOW TANK LEVELS. A SECOND ROOT CAUSE WAS LESS THAN ADEQUATE SUPERVISORY OVERSIGHT AND A CONTRIBUTING ROOT CAUSE WAS AN EQUIPMENT DESIGN DEFICIENCY ASSOCIATED WITH THE ALARM SETPOINT ON CN-TK-1. THE ROOT CAUSE OF THE LOW NITROGEN PRESSURE IN DIVISION II WAS AN EQUIPMENT DESIGN DEFICIENCY ASSOCIATED WITH THE PRESSURE CONTROL VALVE CIA-PCV-2B. THE ROOT CAUSE OF THE EVENT WHERE PLANT OPERATORS DID NOT RESPOND TO THE ABNORMAL CONDITION ALARM WAS A KNOWLEDGE BASED ERROR.

[146] WPPSS 2 DOCKET 50-397 LER 91-004
INADEQUATE FIRE PROTECTION (THERMOLAG) OF DIVISION II SAFE SHUTDOWN CABLES DUE TO INADEQUATE INSTALLATION AND INSPECTION.
EVENT DATE: 032891 REPORT DATE: 042891 NSSS: GE TYPE: BWR

(NSIC 221961) ON MARCH 28, 1991, AS PART OF A SCHEDULED ANNUAL INSPECTION, TWO DEFICIENCIES IN THE THERMOLAG APPLICATION ON A CRITICAL DIVISION II CABLE TRAY RUNNING THROUGH A DIVISION I AREA WERE DISCOVERED. (THERMOLAG IS A FIRE PROTECTIVE COATING.) AT 1615 HOURS ON MARCH 29, 1991, AFTER EVALUATION OF THE LOCATION AND NATURE OF THE DEFECTS, THE DEFICIENCIES WERE DETERMINED TO BE REPORTABLE AS CONDITIONS WHICH ARE OUTSIDE OF PLANT DESIGN BASIS. IT WAS DETERMINED THAT A DESIGN BASIS FIRE IN THE DIVISION I AREA COULD POTENTIALLY ALSO COMPROMISE CRITICAL DIVISION II EQUIPMENT DUE TO INSUFFICIENT FIRE PROTECTOR OF THE CABLE TRAY. THE NRC OPERATIONS CENTER WAS NOTIFIED AT 1640 HOURS ON MARCH 29, 1991, WITHIN THE REQUIRED ONE HOUR NOTIFICATION TIME. IMMEDIATELY AFTER DISCOVERY, THE DEFICIENCIES WERE ADDED TO THE CABLE SPREADING ROOM FIRE IMPAIRMENT LIST AND INCLUDED AS PART OF THE FIRE TOUR. THE DEFECTS, THOUGH ON THE SAME CABLE TRAY SECTION, ARE DISTINCT AND INDEPENDENT. THE ROOT CAUSES FOR THE DEFECTS IS INADEQUATE INSTALLATION OF THE THERMOLAG AND INADEQUATE WORK PRACTICES RELATING TO INSPECTION TECHNIQUES. THE DEFECTS WILL BE REPAIRED AND INSPECTION PERSONNEL WILL BE INSTRUCTED REGARDING THE IMPORTANCE OF CHECKING ALL CABLE TRAY SURFACES FOR DEGRADATION OF THERMOLAG PROTECTION. THIS PARTICULAR EVENT HAD NO SAFETY SIGNIFICANCE.

[147] WPPSS 2 DOCKET 50-397 LER 91-005
OXYGEN CONCENTRATION IN SUPPRESSION CHAMBER WAS NOT VERIFIED PER TECHNICAL SPECIFICATION REQUIREMENTS.
EVENT DATE: 040291 REPORT DATE: 050291 NSSS: GE TYPE: BWR

(NSIC 221962) ON APRIL 2, 1991 A PLANT OPERATIONS ENGINEER DETERMINED THE OXYGEN CONCENTRATION IN THE WETWELL WAS NOT BEING VERIFIED TO BE WITHIN LIMITS ONCE PER

SEVEN (7) DAYS AS REQUIRED BY THE WNP-2 PLANT TECHNICAL SPECIFICATIONS. THIS CONDITION WAS DETERMINED AS A RESULT OF AN EVALUATION OF A PREVIOUS EVENT IN WHICH THE TECHNICAL SPECIFICATION LIMIT FOR OXYGEN CONCENTRATION IN THE WETWELL WAS EXCEEDED. THE IMMEDIATE CORRECTIVE ACTION WAS TO IMPLEMENT A PROCEDURE DEVIATION TO INCLUDE THE TECHNICAL SPECIFICATION OXYGEN VERIFICATION REQUIREMENTS FOR THE WETWELL. THE ROOT CAUSES FOR FAILING TO ROUTINELY MONITOR THE OXYGEN CONCENTRATION IN THE WETWELL PER THE TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS INCLUDE: 1) THE PROCEDURES WERE LESS THAN ADEQUATE BECAUSE THEY DID NOT REQUIRE WETWELL OXYGEN CONCENTRATION BE VERIFIED TO BE WITHIN LIMITS, AND 2) MANAGEMENT DIRECTION WAS LESS THAN ADEQUATE TO ENSURE THAT ALL TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS ARE INCLUDED WITHIN THE PLANT PROCEDURES. A CONTRIBUTING CAUSE WAS A DESIGN DEFICIENCY IN THE CONTAINMENT MONITORING SYSTEM (CMS-CP-1301 AND CMS-CP-1401).

[148] ZION 2 DOCKET 50-304 LER 91-002
SYSTEM AUXILIARY TRANSFORMER DELUGE AND REACTOR TRIP.
EVENT DATE: [illegible] REPORT DATE: 042091 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: ZION 1 (PWR)
VENDOR: AUTOMATIC SPRINKLER CORPORATION

(NSIC 221474) AT 0930, DURING THE PERFORMANCE OF OPERATING PERIODIC TEST (PT)-211, WET PIPE SPRINKLER SYSTEM TEST, THE UNIT 2 UNIT AUXILIARY TRANSFORMER (UAT) AND THE MAIN POWER TRANSFORMER (MPT) WERE INADVERTENTLY DELUGED. AT 1309, ANOTHER DELUGE OCCURRED ON THE SYSTEM AUXILIARY TRANSFORMER (SAT) CAUSING THE SAT TO TRIP. THE BUSES FED FROM THE SAT AUTOMATICALLY TRANSFERRED TO THE UAT. THE 2A FEEDWATER PUMP TRIPPED WHEN THE SAT TRIPPED CAUSING A REACTOR TRIP ON LO-LO STEAM GENERATOR LEVEL. WHEN THE MAIN GENERATOR TRIPPED, ONE DIESEL GENERATOR (D/G) WAS OUT OF SERVICE (OOS) FOR MAINTENANCE SO AN ESSENTIAL BUS WAS NOT AUTOMATICALLY RE-ENERGIZED. A GENERATING STATION EMERGENCY PLAN (GSEP) UNUSUAL EVENT (EAL 3D) WAS DECLARED AT 1335 AND BOTH UNITS WERE STARTED TOWARD COLD SHUTDOWN. THE EVENT WAS CAUSED BY SPURIOUS ACTUATION OF THE UNIT 2 TRANSFORMER'S FIRE PROTECTION DELUGE SYSTEM AND THE IMPROPER POSITIONING OF THE FIRE PROTECTION DELUGE NOZZLES. DURING THIS EVENT ALL FAILURES AND ACTIONS TAKEN WERE WITHIN THE BOUNDS OF THE TECH SPEC LIMITING CONDITIONS FOR OPERATION. VARIOUS CORRECTIVE ACTIONS HAVE BEEN DEVELOPED TO ADDRESS THE CONCERNS THAT WERE RAISED AS A RESULT OF THIS EVENT.

COMPONENT INDEX

This index is based on component and component-related keywords assigned by the NSIC staff when the summaries of the LERs are prepared for computer entry.

COMPONENT INDEX

SYSTEM INDEX

This index is based on system and system-related keywords assigned by the NSIC staff when the summaries of the LERs are prepared for computer entry. Please note that the terms "/SSF" or "/TSF" appended to the system keyword of interest indicate sub-system fault or total system fault, respectively.

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SYSTEM INDEX

KEYWORD INDEX

This index is based on the keywords assigned by the NSIC staff when the summaries of the LERs are prepared for computer entry.

KEYWORD INDEX

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12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

This monthly report contains Licensee Event Report (LER) operational information that was processed into the LER data file of the Nuclear Safety Information Center (NSIC) during the one month period identified on the cover of the document. The LERs, from which this information is derived, are submitted to the Nuclear Regulatory Commission (NRC) by nuclear power plant licensees in accordance with federal regulations. Procedures for LER reporting for revisions to those events occurring prior to 1984 are described in NRC Regulatory Guide 1.16 and NUREG-0161, Instructions for Preparation of Data Entry Sheets for Licensee Event Reports. For those events occurring on and after January 1, 1984, LERs are being submitted in accordance with the revised rule contained in Title 10 Part 50.73 of the Code of Federal Regulations (10 CFR 50.73 - Licensee Event Report System) which was published in the Federal Register (Vol. 48, No. 144) on July 26, 1983. NUREG-1022, Licensee Event Report System - Description of Systems and Guidelines for Reporting, provides supporting guidance and information on the revised LER rule. The LER summaries in this report are arranged alphabetically by facility name and then chronologically by event date for each facility. Component, system, keyword, and component vendor indexes follow the summaries. Vendors are those identified by the utility when the LER form is initiated; the keywords for the component, system, and general keyword indexes are assigned by the computer using correlation tables from the Sequence Coding and Search System.

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