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Licensee Event Report (LER) Compilation

For month of January 1988

Oak Ridge National Laboratory

Prepared for
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Oak Ridge National Laboratory
Nuclear Safety Information 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-1061, 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 86-007 REV 01
UPDATE ON PRESSURIZER CODE SAFETY VALVES INOPERABLE DUE TO INCORRECT SET PRESSURE.
EVENT DATE: 122186 REPORT DATE: 103087 NSSS: BW TYPE: PWR
VENDOR: DRESSER INDUSTRIAL VALVE & INST DIV

(NSIC 206862) PRESSURIZER CODE SAFETY VALVE PSV-1001 WAS REPLACED DURING THE 1R7 REFUELING OUTAGE FOLLOWING SET PRESSURE ADJUSTMENT AT AN OFF-SITE VENDOR FACILITY. DUE TO PRIOR FINDINGS OF LOW SET PRESSURES ON VALVES TESTED OFF-SITE (REFERENCE LER 50-368/86-12), IN-SITU SET PRESSURE TESTING WAS PERFORMED ON PSV-1001 DURING UNIT HEATUP. PSV-1001 WAS FOUND TO HAVE A SET PRESSURE APPROXIMATELY 22 PSIG LOWER THAN ALLOWED (2500 PSIG +/- 1 PERCENT). NO ROOT CAUSE OF ON-SITE VERSUS OFF-SITE SET PRESSURE DEVIATIONS COULD BE DETERMINED. INDEPENDENT THIRD PARTY INVESTIGATIONS HAVE FOUND EITHER TESTING METHOD TO BE ACCURATE AND ACCEPTABLE. ALTHOUGH NO WORK WAS PERFORMED DURING 1R7 ON THE OTHER PRESSURIZER CODE SAFETY VALVE, PSV-1002, AN IN-SITU SET PRESSURE TEST WAS ALSO CONDUCTED ON THIS PRESSURIZER CODE SAFETY VALVE, PSV-1002 WAS FOUND TO HAVE A SET PRESSURE APPROXIMATELY 500 PSIG HIGHER THAN ALLOWED. DURING THE PREVIOUS OPERATING CYCLE, PSV-1002 HAD BEEN REFURBISHED AND SET PRESSURE ADJUSTED ON-SITE WITH NO PROBLEMS NOTED. HOWEVER, IT HAS BEEN CONCLUDED THAT PERSONNEL ERROR OR PROCEDURE INADEQUACY RESULTED IN THE HIGH SET PRESSURE DURING THOSE MAINTENANCE ACTIVITIES, MOST LIKELY DUE TO INCORRECT TESTING.

[2] ARNOLD DOCKET 50-331 LER 87-027
RCIC PRIMARY CONTAINMENT VALVE CLOSURE FROM LOOSE BREAKER CONNECTIONS.
EVENT DATE: 091087 REPORT DATE: 101287 NSSS: GE TYPE: BWR

(NSIC 206678) ON SEPTEMBER 10, 1987, WITH THE PLANT IN POWER OPERATION, A PRIMARY CONTAINMENT ISOLATION VALVE (PCIS) IN THE REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM CYCLED TO ITS FAIL-SAFE CLOSED POSITION DUE TO A LOSS OF POWER. THE POWER TO THE CONTROLLING SOLENOID VALVE OF AIR-OPERATED VALVE CV2411 WAS LOST DUE TO LOOSE TERMINAL CONNECTIONS AT THE POWER SUPPLY BREAKER. THE ROOT CAUSE OF THE LOOSE CONNECTIONS IS UNKNOWN. CV2411 IS DESIGNED TO ISOLATE THE RCIC STEAM LINE CONDENSATE DRAIN PATH UPON RCIC INITIATION. THE OTHER, NON-PCIS, VALVE POWERED FROM THIS BREAKER ALSO CYCLED TO ITS FAIL-SAFE POSITION. THE POSITION INDICATIONS FOR BOTH VALVES IN THE CONTROL ROOM WERE LOST, AS WOULD BE EXPECTED ON A LOSS OF VALVE CONTROL POWER. THE UNPLANNED ISOLATION ON SEPTEMBER 10, 1987 DID NOT RENDER RCIC INOPERABLE. POWER WAS RESTORED AFTER NINE MINUTES AT APPROXIMATELY THE TIME THE CLOSED-IN BREAKER SWITCH WAS TOUCHED DURING INVESTIGATION INTO THE CAUSE OF THE PCIS ISOLATION. THE BREAKER WAS LATER EXAMINED AND TERMINAL CONNECTIONS TIGHTENED. A PREVIOUS INTERMITTENT POWER EVENT FROM A BREAKER ON THE SAME PANEL OCCURRED IN MAY OF 1987. PREVENTIVE MAINTENANCE RECENTLY PERFORMED ON THESE BREAKERS MAY HAVE CONTRIBUTED TO THE LOOSE CONNECTIONS. PLANT TRAINING HAS BEEN INFORMED.

[3] BEAVER VALLEY 1 DOCKET 50-334 LER 87-016
FAILURE TO ANALYZE DIESEL FUEL OIL FOR WATER AND SEDIMENT.
EVENT DATE: 091487 REPORT DATE: 101387 NSSS: WE TYPE: PWR

(NSIC 206679) DURING THE PERIOD FROM 10/15/86 TO 11/26/86, SAMPLES OF DIESEL GENERATOR (DG) FUEL OIL WERE NOT ANALYZED FOR SEDIMENT IN ACCORDANCE WITH THE PROVISIONS OF TECH SPEC 4.8.1.1.2.A.3. ALTHOUGH SAMPLES WERE TAKEN FROM THE FUEL OIL DAY TANKS AT THE REQUIRED FREQUENCY, NO ANALYSIS WAS PERFORMED BECAUSE THE CENTRIFUGE WAS OUT OF SERVICE. ACCORDING TO THE AMERICAN SOCIETY OF TESTING AND MATERIALS (ASTM) MANUAL, WHICH IS REFERENCED BY TECH SPEC 4.8.1.1.2.A.3, A CENTRIFUGE IS THE ONLY APPROVED METHOD FOR ANALYSIS. THEREFORE, FAILURE TO PERFORM THIS ANALYSIS MEANT THE LIMITS OF TECH SPEC 4.0.2 WERE EXCEEDED, AND THIS REPORT IS BEING SUBMITTED UNDER 10 CFR 50.73.A.2.I.B. THE FAILURE TO MEET THE SURVEILLANCE REQUIREMENTS WAS ORIGINALLY NOTED IN AN INTERNAL AUDIT AND REPORTED TO PLANT MANAGEMENT ON 8/11/87, HOWEVER, IT WAS NOT RECOGNIZED AS A REPORTABLE

EVENT UNTIL 9/14/87. THE SAMPLES WERE ANALYZED ON 1/5/87 AND FOUND TO BE SATISFACTORY, SO THAT DG OPERATION WAS NOT IMPAIRED. SINCE THE DGS WERE CAPABLE OF PERFORMING THEIR SAFETY FUNCTION, NO SAFETY IMPLICATIONS RESULTED FROM THIS EVENT. THIS EVENT WAS CAUSED BY THE FAILURE OF COGNIZANT CHEMISTRY PERSONNEL TO REALIZE THAT ALTHOUGH THE FUEL OIL SAMPLES HAD BEEN TAKEN IN ACCORDANCE WITH THE SURVEILLANCE SCHEDULE NOT ALL REQUIRED ANALYSES HAD BEEN PERFORMED.

[4] BEAVER VALLEY 1 DOCKET 50-334 LER 87-017
INOPERABLE FIRE BARRIERS.
EVENT DATE: 100987 REPORT DATE: 110987 NSSS: WE TYPE: PWR
VENDOR: SCHNEIDER INC.

(NSIC 206926) ON 10/9/87, A SPECIAL FIRE DAMPER INSPECTION WAS BEING CONDUCTED TO OBTAIN DATA FOR BEAVER VALLEY'S ENGINEERING DEPARTMENT. DURING THIS INVESTIGATION, TWO DAMPER PENETRATIONS WERE FOUND TO BE UNSEALED. A ONCE PER HOUR FIRE TOUR OF THE AFFECTED AREAS WAS IMMEDIATELY ESTABLISHED. THE INOPERABLE PENETRATIONS WERE SEALED WITH AN APPROVED, FIRE RATED FOAM ON 10/23/87. ALTHOUGH THE DUCT WORK HAD APPARENTLY BEEN UNSEALED SINCE ITS ORIGINAL INSTALLATION (PRIOR TO 1976), THIS DEFICIENCY HAD NOT BEEN DISCOVERED EARLIER DUE TO THE EXTREMELY LIMITED ACCESS THAT WAS AVAILABLE TO THE DAMPERS' DUCT WORK (THE DAMPERS' DUCT WORK WAS ENCASED IN A CINDER BLOCK WALL ON ONE SIDE OF THE FIRE BARRIER AND INACCESSIBLE DUE TO CABLE TRAYS ON THE OTHER SIDE). ALL DAMPERS PENETRATING SAFETY-RELATED FIRE BARRIERS ARE CURRENTLY BEING INSPECTED TO VERIFY THAT THEY ARE CORRECTLY SEALED. THERE WERE NO SAFETY IMPLICATIONS DUE TO THIS EVENT. AS PER BEAVER VALLEY'S APPENDIX R REVIEW, EVEN WITH A FIRE ON BOTH SIDES OF THE BREACHED FIRE BARRIER, THE STATION WOULD HAVE STILL MAINTAINED SAFE SHUTDOWN CAPABILITY.

[5] BEAVER VALLEY 2 DOCKET 50-412 LER 87-020
REACTOR TRIP FOLLOWING MAIN STEAM ISOLATION VALVE CLOSURE.
EVENT DATE: 090987 REPORT DATE: 100787 NSSS: WE TYPE: PWR

(NSIC 206698) ON 9/9/87, WITH THE UNIT AT 30% REACTOR POWER, THE MAIN STEAM ISOLATION VALVES (MSIVS; 2MSS*AOV101A, B, C) WERE CLOSED BY INITIATING A STEAM LINE ISOLATION IN ACCORDANCE WITH THE PROVISIONS OF A STARTUP PROGRAM TEST. THIS TEST METHOD RENDERED THE CONDENSER UNAVAILABLE. THEREFORE, THE LEVEL IN ALL THREE STEAM GENERATORS (2RCS*SG21A, B, C) SHRUNK AS A RESULT OF THE RAPID SECONDARY SIDE PRESSURE RISE INDUCED BY THE SUDDEN MSIV CLOSURE. A REACTOR TRIP OCCURRED AT 0131 HOURS ON 9/9/87 DUE TO SG21A LOW-LOW LEVEL. THE REACTOR PROTECTION SYSTEM AND ENGINEERED SAFETY FEATURES (AUXILIARY FEEDWATER PUMPS) OPERATED PROPERLY, WHILE THE ATMOSPHERIC STEAM DUMPS (2SVS*PCV101A, B, C) OPENED TO REMOVE HEAT. THEREFORE, THE CONSEQUENCES OF THE ACCIDENT WERE MITIGATED IN ACCORDANCE WITH THE ANALYSIS PROVIDED IN SECTIONS 15.2.3 AND 15.2.5 OF THE UNIT II FINAL SAFETY ANALYSIS REPORT, AND NO SAFETY IMPLICATIONS RESULTED FROM THE EVENT. THE OPERATORS FOLLOWED THE UNIT EMERGENCY PROCEDURE FOR A REACTOR TRIP AND STABILIZED THE PLANT IN HOT STANDBY. THE NRC WAS NOTIFIED AT 0230 HOURS, WITHIN THE FOUR HOUR PROVISION OF 10 CFR 50.72.B.2.II. THE REACTOR WAS TAKEN CRITICAL AT 0855 HOURS ON 9/9/87. SINCE THE MSIV CLOSURE TEST WAS A UNIQUE TEST REQUIRED AS PART OF THE STARTUP PROGRAM, A SIMILAR TRIP SHOULD NOT OCCUR IN THE FUTURE.

[6] BEAVER VALLEY 2 DOCKET 50-412 LER 87-021
ZERO DRIFT OF RCS PRESSURIZER PROTECTION PRESSURE TRANSMITTERS.
EVENT DATE: 091487 REPORT DATE: 101487 NSSS: WE TYPE: PWR
VENDOR: BARTON INSTRUMENT CO., DIV OF ITT

(NSIC 206724) ON 7/22/87, OPERATIONS PERSONNEL OBSERVED A DIFFERENCE BETWEEN THE RCS PRESSURE PROTECTION CHANNELS AND THE RCS PRESSURE CONTROL CHANNELS. THE

INDICATION OF ALL PROTECTION CHANNELS AGREED AMONG THEMSELVES, AS DID THE CONTROL CHANNELS. AN INVESTIGATION WAS INITIATED (VERIFYING RACK CALIBRATIONS AND EXAMINING PREVIOUS INSTRUMENT CALIBRATION RECORDS FOR POTENTIAL COMMON MODE FAULTS) BUT COULD FIND NO PROBLEM OUTSIDE CONTAINMENT. ON 9/14/87, AFTER THE PLANT WAS SHUTDOWN FOR A MAINTENANCE OUTAGE, THE PRESSURE TRANSMITTERS WERE EXAMINED. THE THREE PROTECTION CHANNELS WERE FOUND TO HAVE DRIFTED OUT OF CALIBRATION. THE POTENTIAL FOR OUTPUT DRIFT OF THIS MODEL TRANSMITTER (BARTON MODEL 763) HAD BEEN PREVIOUSLY IDENTIFIED IN WESTINGHOUSE TECHNICAL BULLETIN NSID-TB-85-11. THE TRANSMITTERS WERE RECALIBRATED, USING A CALIBRATION TECHNIQUE DESIGNED TO ELIMINATE THE TYPE OF OUTPUT DRIFT DETAILED IN NSID-TB-85-11. SUBSEQUENT TO THE RECALIBRATION, NO OUTPUT DRIFT HAS OCCURRED. THERE WERE NO SAFETY IMPLICATIONS DUE TO THIS EVENT. ALTHOUGH THE OUTPUT DRIFT WOULD HAVE CAUSED THE HIGH RCS PRESSURE REACTOR TRIP TO OCCUR AT A PRESSURE GREATER THAN ALLOWED BY TECHNICAL SPECIFICATIONS, THE TRIP WOULD HAVE OCCURRED AT A LOWER PRESSURE THAN ASSUMED IN THE STATION'S SAFETY ANALYSIS (BVPS UNIT 2 FSAR SECTION 15.0, TABLE 15.0-4).

[7] BEAVER VALLEY 2 DOCKET 50-412 LER 87-022
AUTOMATIC START OF NO. 1 EMERGENCY DIESEL GENERATOR ON LOSS OF POWER TO 2AE 4KV EMER. BUS.
EVENT DATE: 092187 REPORT DATE: 102187 NSSS: WE TYPE: PWR
VENDOR: ITE/GOULD

(NSIC 206830) ON 9/21/87 AT 0504 HOURS, WITH THE PLANT IN COLD SHUTDOWN, OPERATIONS PERSONNEL WERE PREPARING TO START THE 21A CHARGING PUMP ON THE 2AE 4160 VAC EMERGENCY BUS. UPON PUMP START, ONE OF THE 2AE EMERGENCY BUS TIE SUPPLY BREAKERS TO THE 2A 4160 VAC BUS (ACB-2E7) TRIPPED ON ACTUATION OF THE "A" AND "C" PHASE OVERCURRENT RELAYS 51-VE207A,C (ITE TYPE 51E). THIS CAUSED THE NO. 1 EMERGENCY DIESEL GENERATOR TO AUTOMATICALLY START. THE DIESEL GENERATOR REACHED OPERATING SPEED AND VOLTAGE, BUT ITS OUTPUT BREAKER DID NOT CLOSE ONTO THE 2AE BUS DUE TO THE OVERCURRENT RELAY ACTUATION. THIS IS A PROTECTIVE DESIGN INTERLOCK. OPERATORS RESET THE OVERCURRENT RELAYS, RECLOSED THE 2AE BUS TIE BREAKERS, RESTORING POWER TO THE 2AE BUS. THE CAUSE FOR THIS INCIDENT WAS DUE TO A FAULTY CONNECTION BETWEEN THE TAP-TAP BLOCK COMBINED WITH AN INCORRECT PERSONNEL GROUND LOCATION. THE OVERCURRENT RELAY WAS RE-CALIBRATED, THE RELAY CASE WAS REPLACED, AND THE RELAY WAS RETURNED TO SERVICE. ADDITIONALLY, INCORRECT WIRING OF THE CURRENT TRANSFORMER LOOP GROUND LOCATION WAS ALSO CORRECTED. THERE WERE NO SAFETY IMPLICATIONS TO THE PUBLIC AS A RESULT OF THIS EVENT. THE OVERCURRENT INTERLOCK IS DESIGNED TO PREVENT THE DIESEL GENERATOR FROM LOADING ONTO A FAULTED BUS. THE LOSS OF ONE 4160 VAC EMERGENCY BUS WILL NOT IMPAIR A CONTROLLED SAFE SHUTDOWN (FSAR SECTION 8.3.1.1.2).

[8] BEAVER VALLEY 2 DOCKET 50-412 LER 87-023
REACTOR TRIP DUE TO STEAM FLOW/FEEDFLOW MISMATCH WITH LOW LEVEL.
EVENT DATE: 092887 REPORT DATE: 102887 NSSS: WE TYPE: PWR
VENDOR: BARTON INSTRUMENT CO., DIV OF ITT
WESTINGHOUSE ELECTRIC CORP.

(NSIC 206831) ON 9/28/87, UNIT 2 WAS OPERATING AT 73% POWER WITH ONLY ONE MAIN FEEDWATER PUMP (MFP) RUNNING, AS REQUIRED BY A STARTUP PROGRAM TEST. THE "B" AND "C" MAIN FEEDWATER REGULATING VALVES (MFRVS) WERE IN MANUAL CONTROL DUE TO ERRATIC AUTOMATIC OPERATION. UPON TEST COMPLETION, THE SECOND MFP WAS STARTED AND PLANT CONDITIONS WERE STABILIZED. THE "B" MFRV WAS THEN PLACED IN AUTOMATIC TO DETERMINE IF IT WOULD NOW CONTROL PROPERLY. IMMEDIATELY UPON TRANSFER TO AUTOMATIC, THE "B" MFRV FAILED OPEN. IT WAS RETURNED TO MANUAL CONTROL, AND SHUT, WHICH RESULTED IN INCREASED FLOW IN THE OTHER FEEDWATER LINES. THE "A" MFRV (IN AUTOMATIC) STARTED SHUTTING IN RESPONSE TO THE INCREASED FLOW, WHICH RESULTED IN A RAPID FLOW DECREASE AND IN A STEAM FLOW/FEED FLOW MISMATCH BISTABLE ACTUATION. SINCE A LEVEL CHANNEL FOR THE "A" STEAM GENERATOR (SG) HAD PREVIOUSLY

BEEN DECLARED INOPERABLE AND HAD ITS BISTABLE TRIPPED, THE MISMATCH RESULTED IN A 1/2 STEAMFLOW/FEEDFLOW MISMATCH COINCIDENT WITH A 1/2 LOW SG LEVEL REACTOR TRIP AT 0156 HOURS. THE PLANT WAS QUICKLY STABILIZED IN ACCORDANCE WITH EMERGENCY PROCEDURES. THERE WERE NO SAFETY IMPLICATIONS BECAUSE ALL TRIP AND SAFETY FEATURES ACTUATED AS REQUIRED TO MITIGATE THE ACCIDENT. THE CAUSE OF THE "B" MFRV FAILURE WAS A FAULTY CONTROLLER CARD IN THE AUTOMATIC CIRCUIT, WHICH WAS REPLACED. THE REACTOR WAS TAKEN CRITICAL AT 1047 HOURS ON 9/28/87.

[9] BEAVER VALLEY 2 DOCKET 50-412 LER 87-024
SAFETY INJECTION DUE TO LOW STEAMLINE PRESSURE SIGNAL CAUSED BY GOVERNOR VALVE OPENING.
EVENT DATE: 092987 REPORT DATE: 102887 NSSS: WE TYPE: PWR

(NSIC 206832) ON 9/29/87 AT APPROXIMATELY 0200 HOURS, WITH THE UNIT IN POWER OPERATION, A TURBINE STARTUP WAS IN PROGRESS IN ACCORDANCE WITH OPERATING MANUAL 2.52.4A. THE OPERATORS WERE PERFORMING THE OVERSPEED PROTECTION CONTROLLER (OPC) TRIP CHECK. THE OPC SWITCH WAS PLACED IN THE TEST POSITION CAUSING GOVERNOR AND INTERCEPTOR VALVE CLOSURE. THE OPC SWITCH WAS THEN PLACED IN THE IN-SERVICE POSITION. SINCE THE PROCEDURE DID NOT ADDRESS THE VALVE POSITION LIMITER IT WAS STILL IN THE 100% POSITION. THE GOVERNOR AND INTERCEPTOR VALVES IMMEDIATELY OPENED, CAUSING A SAFETY INJECTION AT 0219 HOURS, ON THE RATE SENSITIVE LOW STEAMLINE PRESSURE SIGNAL. A RESULTANT REACTOR TRIP IMMEDIATELY FOLLOWED. THE OPERATORS UTILIZED THE EMERGENCY OPERATING PROCEDURES TO STABILIZE THE PLANT IN HOT STANDBY. AS A RESULT OF THE SAFETY INJECTION, APPROXIMATELY 3500 GALLONS OF BORATED WATER WERE INJECTED INTO THE REACTOR COOLANT SYSTEM. THE CAUSE FOR THIS EVENT WAS A DEFICIENT PROCEDURE. THE PROCEDURE WAS RECENTLY REVISED AND DID NOT INCLUDE INSTRUCTIONS REGARDING THE VALVE POSITION LIMITER. THE PROCEDURE HAS BEEN REVISED TO INCLUDE INSTRUCTIONS REGARDING THE VALVE POSITION LIMITER. THERE WERE NO SAFETY IMPLICATIONS TO THE PUBLIC AS A RESULT OF THIS EVENT.

[10] BEAVER VALLEY 2 DOCKET 50-412 LER 87-025
AUXILIARY FEEDWATER ACTUATION.
EVENT DATE: 093087 REPORT DATE: 102887 NSSS: WE TYPE: PWR

(NSIC 206833) ON 9/30/87, A PLANT START-UP WAS IN PROGRESS. AT 1129 HOURS, THE STATION WAS AT 29% POWER. ONLY ONE MAIN FEEDWATER PUMP WAS IN SERVICE. AT THIS TIME, A LOW FEEDWATER PUMP SUCTION PRESSURE TRANSIENT OCCURRED. THIS CAUSED THE RUNNING MAIN FEEDWATER PUMP TO TRIP, RESULTING IN A LOSS OF FEEDWATER FLOW. BOTH MOTOR DRIVEN AUXILIARY FEED PUMPS AUTO-STARTED, AS PER DESIGN. OPERATORS, OBSERVING THAT MAIN FEEDWATER PUMP SUCTION PRESSURE HAD IMMEDIATELY RECOVERED, SUCCESSFULLY STARTED THE STANDBY MAIN FEEDWATER PUMP. WITH FEED FLOW AGAIN BEING SUPPLIED BY A MAIN FEEDWATER PUMP, THE AUXILIARY FEED PUMPS WERE SHUT DOWN. INVESTIGATION DETERMINED THAT THE EVENT HAD BEEN INITIATED WHEN A CONSTRUCTION WORKER INADVERTENTLY BROKE THE AIR LINE TO A CONDENSATE RECIRCULATION VALVE, FAILING THE VALVE OPEN, DECREASING NET FLOW/PRESSURE TO THE SUCTION OF THE MAIN FEEDWATER PUMPS. THE CONDENSATE POLISHING BYPASS VALVE AUTOMATICALLY OPENED IN RESPONSE TO THE LOW CONDENSATE PRESSURE, RESTORING PRESSURE TO NORMAL. THE AIRLINE TO THE RECIRCULATION VALVE WAS REPAIRED AND THE SYSTEM RETURNED TO NORMAL ARRANGEMENT. THERE WERE NO SAFETY IMPLICATIONS DUE TO THIS EVENT. BOTH MOTOR DRIVEN AUXILIARY FEED PUMPS STARTED ON THE LOSS OF MAIN FEED, AS PER DESIGN (FSAR SECTION 15.2.6).

[11] BEAVER VALLEY 2 DOCKET 50-412 LER 87-026
REACTOR TRIP DUE TO "A" STEAM GENERATOR LOW LEVEL COINCIDENT WITH STEAM FLOW/FEED FLOW MISMATCH.
EVENT DATE: 100887 REPORT DATE: 110987 NSSS: WE TYPE: PWR
VENDOR: BARTON INSTRUMENT CO., DIV OF ITT

(NSIC 207109) ON 10/8/87, WITH THE UNIT AT 55% REACTOR POWER, THE 'A' MAIN FEED PUMP WAS REMOVED FROM SERVICE TO PERMIT TESTING GROUP TO OBTAIN SINGLE PUMP OPERATION DATA ON THE 'B' MAIN FEED PUMP. APPROXIMATELY 5 SECONDS LATER, STEAM GENERATOR (SG) FEED PUMP RECIRCULATION VALVE (2FWR-FCV150B) OPENED FOLLOWED BY THE SG FEED PUMP RECIRCULATION VALVE (2FWR-FCV150A). THE 'A' MAIN FEED PUMP WAS RESTARTED AND SUBSEQUENTLY BOTH SG RECIRCULATION VALVES CLOSED. SG 'A' AND 'C' MAIN FEED REGULATING VALVES BEGAN TO OSCILLATE. SINCE THE NARROW RANGE SG 'A' WATER LEVEL CHANNEL (2FWS-L475) HAD BEEN DECLARED OUT OF SERVICE AND ITS BISTABLES TRIPPED, THE FEED FLOW SWING ON THE 'A' SG RESULTED IN A FEED FLOW/STEAM FLOW MISMATCH WITH SG 'A' LOW LEVEL REACTOR TRIP. MAINTENANCE DISCOVERED THE PIPING TO THE 'B' MAIN FEED PUMP SUCTION FLOW INDICATING SWITCHES WAS REVERSED. THIS CAUSED BOTH MAIN FEED PUMPS RECIRCULATION VALVES TO OPEN WHEN THE 'A' MAIN FEED PUMP WAS SECURED. THE MAIN FEED REGULATION VALVES OVER RESPONDED CAUSING THEM TO SWING FULL RANGE. THE PIPING ERROR WAS CORRECTED AND THE GAIN ON THE MAIN FEED REG. VALVES ADJUSTED. THERE WERE NO SAFETY IMPLICATIONS. THE REACTOR TRIPPED AS DESIGNED AND THE AUXILIARY FEED PUMPS STARTED.

[12]   BEAVER VALLEY 2   DOCKET 50-412   LER 87-027
MISSED MAINTENANCE SURVEILLANCE.
EVENT DATE: 101387   REPORT DATE: 111287   NSSS: WE   TYPE: PWR

(NSIC 206960) ON 10/13/87, DURING A REVIEW OF THE MAINTENANCE SURVEILLANCE PROCEDURE (MSP) COMPLETION LOG, IT WAS DISCOVERED THAT THE PRESSURIZER PRESSURE PROTECTION CHANNEL I CALIBRATION HAD NOT BEEN PERFORMED WITHIN ITS REQUIRED MONTHLY FREQUENCY. THE PROTECTION CHANNEL WAS IMMEDIATELY CALIBRATED. THE "AS-FOUND" DATA FROM THE CALIBRATION WAS SATISFACTORY AND THE CHANNEL REQUIRED NO ADJUSTMENTS. THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR ON THE PART OF THE ENGINEER SCHEDULING PROCEDURE PERFORMANCE. THIS ERROR MIGHT HAVE BEEN CAUGHT SOONER, EXCEPT FOR A POTENTIAL MULTI-DAY LAG BETWEEN WHEN AN MSP IS PERFORMED AND WHEN IT IS LOGGED "COMPLETE". THE PROCEDURE FOR UPDATING THE MSP COMPLETION LOG HAS BEEN REVISED TO INSURE THAT A PROCEDURE IS LOGGED "COMPLETE" ON THE DAY IT IS ACTUALLY PERFORMED. AS A PERMANENT CORRECTIVE ACTION, THE EXISTING MANUAL MSP TRACKING SCHEDULE IS BEING REPLACED BY A COMPUTERIZED SCHEDULE. THERE WERE NO SAFETY IMPLICATIONS DUE TO THIS EVENT. THE OTHER TWO PROTECTION CHANNELS WERE FULLY OPERABLE DURING THIS INTERVAL. ADDITIONALLY, AS SHOWN BY THE "AS-FOUND" DATA IN THE COMPLETED MSP ON 10/13/87, CHANNEL I STILL WOULD HAVE FUNCTIONED PROPERLY DURING THIS PERIOD.

[13]   BEAVER VALLEY 2   DOCKET 50-412   LER 87-028
TURBINE TRIP/REACTOR TRIP ON LOW EHC PRESSURE.
EVENT DATE: 101487   REPORT DATE: 111387   NSSS: WE   TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 207110) AT APPROXIMATELY 1000 HOURS ON 10/14/87, AN OPERATOR IN THE UNIT 2 CONTROL ROOM RECEIVED AN ALARM INDICATING A LOSS OF TURBINE ELECTROHYDRAULIC (EH) FLUID PRESSURE. A NON-LICENSED NUCLEAR OPERATOR (NO) WAS DIRECTED TO OPEN THE MANUAL ISOLATION VALVES FOR THE EH FLUID PRESSURE SWITCHES IN ORDER TO RESTORE PRESSURE. THESE SWITCHES HAD BEEN ISOLATED PREVIOUSLY DUE TO ERRATIC OPERATION THAT HAD LED TO A TURBINE TRIP. BEFORE THE SUO COULD REACH THE VALVES, HOWEVER, THE TURBINE TRIPPED ON A LOW EH FLUID PRESSURE SIGNAL. SINCE REACTOR POWER WAS 99%, WHICH IS GREATER THAN THE TURBINE TRIP INTERLOCK SETPOINT (P-9) OF 49%, THE TURBINE TRIP RESULTED IN A REACTOR TRIP AT 1010 HOURS. OPERATORS FOLLOWED EMERGENCY PROCEDURES TO STABILIZE THE PLANT BY 1025 HOURS. NO SAFETY IMPLICATIONS RESULTED BECAUSE THE REACTOR PROTECTION SYSTEM AND ALL APPROPRIATE ENGINEERED SAFETY FEATURES OPERATED PROPERLY TO MITIGATE THE CONSEQUENCES OF THE TRANSIENT. THE CAUSE OF THE EVENT WAS DETERMINED TO BE LEAKAGE PAST THE EH PRESSURE SWITCH ISOLATION VALVES. TO CORRECT THE PROBLEM, THE VALVES WERE OPENED AND THE EH LOW PRESSURE CIRCUITS BYPASSED. THE REACTOR WAS TAKEN CRITICAL AT

2115 HOURS, 10/14/87. WESTINGHOUSE WILL MODIFY THE CIRCUITS DURING THE FIRST REFUELING OUTAGE.

[14] BRAIDWOOD 1 DOCKET 50-456 LER 87-024 REV 01
UPDATE ON LOSS OF PRIMARY AND BACKUP POWER SUPPLY TO PRESSURE TRANSMITTER 1PT-505 TURBINE IMPULSE PRESSURE.
EVENT DATE: 052287 REPORT DATE: 101987 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 206893) ON MAY 22, 1987, AT 1959, AS OPERATING PERSONNEL WERE REMOVING THE BACKUP POWER SUPPLY TO PANEL 1PA01J, THE FUSE FOR THE PRIMARY POWER SUPPLY TO CARD FRAME TWO BLEW. A TURBINE TRIP SIGNAL WAS PRESENT DUE TO THE TURBINE STOP VALVES BEING CLOSED. THIS CONDITION, IN CONJUNCTION WITH THE BLOWN FUSE, RESULTED IN A REACTOR TRIP SIGNAL. THERE WERE NO COMPONENT OPERATIONS, ONLY ANNUNCIATION IN THE CONTROL ROOM. ON MAY 27, 1987, AT 2051, THE FUSE FOR THE PRIMARY POWER SUPPLY TO CARD FRAME TWO AGAIN BLEW, RESULTING IN THE SAME SEQUENCE OF EVENTS AS NOTED ABOVE. IN BOTH OF THESE CASES, THE BLOWN FUSE WAS REPLACED, THE INSTRUMENTATION RETURNED TO NORMAL, AND THE REACTOR TRIP CLEARED. THE CAUSE OF THE EVENT IS THE DESIGNED FUSE SIZE IN CARD FRAME TWO OF CABINET 1PA01J. A 25 AMPERE, SLOW BLOW FUSE WILL BE TEMPORARILY INSTALLED IN PLACE OF THE CURRENT FUSE UNTIL A MODIFICATION HAS BEEN IMPLEMENTED TO REDUCE LOADING.

[15] BRAIDWOOD 1 DOCKET 50-456 LER 87-049
EXCEEDED ANALYSIS FREQUENCY ON WASTE GAS OXYGEN ANALYSIS.
EVENT DATE: 090587 REPORT DATE: 100587 NSSS: WE TYPE: PWR

(NSIC 206742) A RADIATION CHEMISTRY TECHNICIAN WAS VERBALLY DIRECTED TO PERFORM THE REQUIRED ANALYSIS FOR THE OAIT-GW004, WASTE GAS OXYGEN ANALYZER ON SEPTEMBER 5, 1987. HE WAS ADDITIONALLY DIRECTED TO TAKE A SECOND SAMPLE FOR AN UNRELATED ROUTINE SURVEILLANCE. AT APPROXIMATELY 1100 ON SEPTEMBER 6, 1987, IT WAS DISCOVERED WHILE REVIEWING THE CHEMISTRY RESULTS FROM THE REQUIRED OXYGEN SAMPLE, THAT THE ANALYSIS PERFORMED ON SEPTEMBER 5, 1987 WAS AN ISOTOPIC RATHER THAN THE REQUIRED OXYGEN CONCENTRATION. THE RESULTS OF THE SEPTEMBER 6, 1987 OXYGEN ANALYSIS WERE SATISFACTORY. THE CAUSE OF THE EVENT IS A PROGRAMMATIC DEFICIENCY IN THAT THE NON-ROUTINE SAMPLES AND ANALYSIS WERE NOT FORMALLY TRACKED. THE SAMPLING PROGRAM HAS BEEN FORMALIZED AND TRAINING HAS BEEN COMPLETED. ADDITIONALLY, THIS EVENT HAS BEEN DISCUSSED WITH THE RADIATION CHEMISTRY TECHNICIANS AT A SAFETY MEETING. SEPARATE DISCUSSIONS WERE HELD WITH THE FOREMAN, THE LEAD CHEMIST AND HEALTH PHYSICIST. A TRAINING PROGRAM IS BEING DEVELOPED TO COVER THE REQUIREMENTS OF THE WASTE GAS MANAGEMENT SYSTEM FOR APPROPRIATE PERSONNEL. CORRECTIVE ACTION ASSOCIATED WITH A PREVIOUS SIMILAR EVENT ARE STILL IN PROGRESS. THIS ACTION IS THE DEVELOPMENT OF A MATRIX FOR IDENTIFYING ROUTINE AND NON-ROUTINE SAMPLING REQUIREMENTS AS THEY RELATE TO TECH SPECS.

[16] BRAIDWOOD 1 DOCKET 50-456 LER 87-045
AXIAL FLUX DIFFERENCE SURVEILLANCE NOT INITIATED WHEN PROCESS COMPUTER WAS REBOOTED DUE TO MISINTERPRETATION.
EVENT DATE: 090887 REPORT DATE: 100787 NSSS: WE TYPE: PWR

(NSIC 206730) ON SEPTEMBER 8, 1987, THE PROCESS COMPUTER WAS REBOOTED, CAUSING THE AXIAL FLUX DIFFERENCE (AFD) MONITOR ALARM TO BE INOPERABLE. THE SURVEILLANCE FOR TECH SPEC 4.2.1.1.B SHOULD HAVE BEEN INITIATED AT THE TIME, BUT WAS NOT PERFORMED UNTIL APPROXIMATELY SEVEN HOURS LATER. THE ROOT CAUSE OF THE EVENT WAS A MISINTERPRETATION BY THE STATION CONTROL ROOM ENGINEER (SCRE) FAILING TO REALIZE THAT REBOOTING RENDERS THE COMPUTER INOPRABLE. THE IMMEDIATE CORRECTIVE ACTION TAKEN WAS INITIATION OF THE SURVEILLANCE. THIS EVENT HAS BEEN REVIEWED WITH THE INDIVIDUAL INVOLVED IN THE EVENT. THERE HAS BEEN ONE PREVIOUS

OCCURRENCE. THE CORRECTIVE ACTION ASSOCIATED WITH THIS EVENT IS STILL IN PROGRESS.

[17] BRAIDWOOD 1 DOCKET 50-456 LER 87-046
INCREASE IN REACTOR COOLANT SYSTEM COOLDOWN RATE CAUSING LO LO STEAM GENERATOR LEVEL AND AUTO-START OF THE 1B DIESEL DRIVEN AUXILIARY FEEDWATER PUMP.
EVENT DATE: 091087 REPORT DATE: 100287 NSSS: WE TYPE: PWR

(NSIC 206731) DURING THE PERFORMANCE OF THE SHUTDOWN FROM OUTSIDE THE CONTROL ROOM STARTUP TEST, THE 1B AUXILIARY FEEDWATER PUMP AUTO STARTED DUE TO LO LO LEVEL IN THE 1D STEAM GENERATOR FOLLOWING AN INCREASE IN COOLDOWN RATE. 1D STEAM GENERATOR SECONDARY WATER LEVEL WAS RESTORED AND THE 1B AUXILIARY FEEDWATER PUMP WAS SECURED. THE CAUSE OF THIS EVENT WAS A PROCEDURAL DEFICIENCY IN THAT INCOMPLETE GUIDANCE WAS PROVIDED TO DIRECT CONTINUOUS MONITORING OF STEAM GENERATOR NARROW RANGE LEVEL WHEN COOLDOWN RATES ARE CHANGED. TO PREVENT RECURRENCE, AN OPEN ITEM WILL BE INITIATED AGAINST THE UNIT 2 STARTUP TEST REQUIRING THE NARROW RANGE STEAM GENERATOR LEVEL BE MONITORED DURING CHANGES IN COOLDOWN RATE. NO PREVIOUS OCCURRENCES.

[18] BRAIDWOOD 1 DOCKET 50-456 LER 87-050
REACTOR TRIP DUE TO INADVERTENT TRANSFORMER DELUGE ACTUATION DURING MAINTENANCE ACTIVITY.
EVENT DATE: 092387 REPORT DATE: 101487 NSSS: WE TYPE: PWR

(NSIC 206743) AT 2153 ON SEPTEMBER 23, 1987, MAINTENANCE PERSONNEL WERE REINSTALLING THE HANDLE ON TRANSFORMER 141-1 DELUGE ALARM TEST VALVE 1FP5098F WHEN THEY UNINTENTIONALLY TURNED THE VALVE STEM APPROXIMATELY FIVE DEGREES. THIS ACTUATED THE DELUGE SYSTEM ON THE UNIT AUXILIARY TRANSFORMER (UAT) AND CAUSED THE EIGHTY-SIX LOCKOUT RELAY TO ACTUATE CAUSING BOTH UATS TO BE ELECTRICALLY ISOLATED FROM THE 345 KV SWITCHYARD AND THE 6.9 AND 4 KV BUSES FROM THE UNIT; THIS LED TO A TURBINE TRIP FOLLOWED BY A REACTOR TRIP. THE 1FP5098F VALVE IS A BALL VALVE WITH NINETY DEGREES FROM FULLY CLOSED TO FULLY OPEN. THE HANDLE WAS BEING DRILLED TO ALLOW INSTALLATION OF TAMPER SEALS DUE TO THE PREVIOUS OCCURRENCE DISCUSSED BELOW. CAUSE OF THE UAT TRIP WAS MAINTENANCE PERSONNEL INADVERTENTLY TURNING THE STEM OF THE VALVE APPROXIMATELY FIVE DEGREES. CONTRIBUTING TO THIS WAS THE DECISION TO NOT REMOVE THE DELUGE SYSTEMS FROM SERVICE PRIOR TO ALLOWING THE WORK TO BE PERFORMED. STABLE PLANT CONDITIONS WERE ESTABLISHED AT 2206. THIS EVENT HAS BEEN REVIEWED WITH APPROPRIATE PERSONNEL STRESSING THE IMPORTANCE OF PERFORMING THIS TYPE OF ACTIVITY WITH THE SYSTEMS REMOVED FROM SERVICE. PREVIOUS OCCURRENCE INVOLVING A DELUGE SYSTEM ACTUATION AND TRANSFORMER TRIP, AND ALL THE CORRECTIVE ACTIONS ASSOCIATED WITH IT HAVE NOT BEEN COMPLETED.

[19] BRAIDWOOD 1 DOCKET 50-456 LER 87-051
CONTROL ROOM VENTILATION SWITCHOVER DUE TO SPURIOUS NOISE ON RADIATION MONITOR CHANNEL ORE-PRO33B.
EVENT DATE: 092387 REPORT DATE: 101487 NSSS: WE TYPE: PWR

(NSIC 206744) AT 1804 ON SEPTEMBER 23, 1987, THE CONTROL ROOM TRAIN B VENTILATION SYSTEM SHIFTED TO ITS EMERGENCY MAKEUP MODE DUE TO A SPURIOUS SPIKE ON THE CONTROL ROOM OUTSIDE AIR INTAKE GAS CHANNEL ORE-PRO33B. THE INSTRUMENT MAINTENANCE DEPARTMENT COULD FIND NO SPECIFIC PROBLEMS OR EQUIPMENT FAILURES AND THE RADIATION MONITOR WAS DECLARED OPERABLE AT 2000 ON SEPTEMBER 24, 1987. AT 2030 ON SEPTEMBER 24, 1987, WITH THE CONTROL ROOM TRAIN B VENTILATION SYSTEM STILL RUNNING IN THE EMERGENCY MAKEUP MODE, ANOTHER SPURIOUS SPIKE ON ORE-PRO33B WAS RECEIVED. EXTENSIVE TROUBLESHOOTING REVEALED NO PROBLEMS OR EQUIPMENT FAILURES. THERE WAS A SIGNIFICANT AMOUNT OF CONSTRUCTION ACTIVITY IN THE AREA DURING BOTH SPIKING OCCURRENCES. ALTHOUGH THIS THEORY CANNOT BE PROVEN, IT IS SUSPECTED THAT IT MAY HAVE CONTRIBUTED TO THE EVENT. THE ORE-PRO33B WAS DECLARED

OPERABLE AT APPROXIMATELY 0200 ON SEPTEMBER 25, 1987. THE SPIKING HAS NOT RECURRED. THERE HAVE BEEN TWO PREVIOUS OCCURRENCES OF CONTROL ROOM RADIATION MONITORS SPIKING CAUSING THE VENTILATION SYSTEM TO SHIFT TO ITS EMERGENCY MAKEUP MODE. HOWEVER, THE CAUSE WAS A FAILED POWER SUPPLY IN ONE CASE AND THE OTHER WAS DUE TO A RADIO BEING KEYED WITHIN THE DESIGNATED EXCLUSION AREA. NEITHER WERE THE CAUSE OF THIS EVENT.

[20] BRAIDWOOD 1 DOCKET 50-456 LER 87-052
REACTOR TRIP DUE TO MAIN POWER TRANSFORMER OVEREXCITATION RELAY ACTUATION FOR UNKNOWN REASON.
EVENT DATE: 092487 REPORT DATE: 101987 NSSS: WE TYPE: PWR
VENDOR: UNIBUS INC

(NSIC 206763) AT 2210 ON SEPTEMBER 24, 1987, BWGP 100-3, POWER ASCENSION 5% TO 100%, WAS IN PROGRESS. THE MAIN TRANSFORMER OVEREXCITATION RELAY ACTUATED DURING THE PROGRESS OF SYNCHRONIZING THE MAIN GENERATOR TO THE SYSTEM GRID. THIS ACTIVATED THE EIGHTY-SIX LOCKOUT RELAY WHICH CAUSED THE MAIN GENERATOR, MAIN POWER TRANSFORMER AND UNIT AUXILIARY TRANSFORMER TO BE ELECTRICALLY ISOLATED FROM THE GRID. THE ROOT CAUSE OF THE EVENT IS UNKNOWN. INVESTIGATIONS HAVE REVEALED A LOOSE CONNECTION ON A MOVABLE CONTACT BLOCK LOCATED ON C PHASE OF THE 25 KV REGULATING POTENTIAL TRANSFORMER. THIS DID CONTRIBUTE TO THE EVENT. HOWEVER, IT WAS NOT THE ROOT CAUSE. OPERATOR ERROR IS RULED OUT AS THE PROCEDURE WAS FOLLOWED AND THIS EVOLUTION DOES NOT REQUIRE ANY SPECIAL TRAINING OR PRACTICE. DURING THE SUBSEQUENT PLANT STARTUP AND POWER ASCENSION, THE A AND C PHASE POTENTIAL TRANSFORMERS WERE MONITORED. INCLUDED IN THIS COMPREHENSIVE MONITORING PROGRAM WERE THE OVEREXCITATION RELAY, WHICH ACTIVATED DURING THIS EVENT, AND THE VOLTAGE REGULATOR. THIS TESTING REVEALED THAT ALL READINGS WERE NORMAL. SHOULD ANY ADDITIONAL INFORMATION CONCERNING THE CAUSE OF THIS EVENT BECOME AVAILABLE, THIS REPORT WILL BE SUPPLEMENTED.

[21] BRAIDWOOD 1 DOCKET 50-456 LER 87-053
CONTAINMENT ISOLATION FROM LOSS OF POWER TO RADIATION MONITOR 1RT-AR011.
EVENT DATE: 092487 REPORT DATE: 101487 NSSS: WE TYPE: PWR

(NSIC 206745) AT 0928 ON SEPTEMBER 24, 1987, THE CONTAINMENT FUEL HANDLING INCIDENT AREA RADIATION MONITOR 1RT-AR011 EXPERIENCED A MOMENTARY LOSS OF POWER. THIS CAUSED A TRAIN A CONTAINMENT VENTILATION ISOLATION SIGNAL TO BE GENERATED. NO COMPONENTS OPERATED AT THE TIME AS THE CONTAINMENT PURGE ISOLATION VALVES WERE ALREADY IN THE CLOSED POSITION. THE EVENT WAS IMMEDIATELY EVALUATED AND CONSIDERED SPURIOUS. THE CONTAINMENT VENTILATION SIGNAL WAS MANUALLY RESET FOLLOWING THIS EVALUATION. THE EXACT CAUSE IS UNKNOWN AND IS BELIEVED TO BE RELATED TO CONSTRUCTION ACTIVITY IN THE AREA. THE POWER CABLE FASTENING SCREWS WERE FOUND LOOSE AND THE FLEXIBLE CONNECTIONS WERE COVERED IN PLASTIC IN PREPARATION FOR PAINTING THE AREA. IT IS SURMISED THAT SUFFICIENT FORCE WAS APPLIED TO THE FLEXIBLE CONNECTIONS WHICH RESULTED IN A MOMENTARY INTERRUPTION OF POWER. CORRECTIVE ACTIONS INCLUDED RESETTING OF THE CONTAINMENT ISOLATION SIGNAL, THE MONITORS HAD THE PLASTIC REARRANGED, TEMPORARY SIGNS WERE POSTED TO WARN AGAINST RADIO TRANSMISSION IN THE AREA, AND WORK WAS EXPEDITED TO COMPLETE THE PAINTING IN THE AREA. THE POWER CABLE CONNECTIONS WERE REMOVED, CLEANED AND PROPERLY REINSTALLED. THE FOUR PREVIOUS OCCURRENCES OF CONTAINMENT VENTILATION ISOLATION EVENTS WERE DUE TO A DIFFERENT CAUSE THAN THE PRESENT EVENT.

[22] BRAIDWOOD 1 DOCKET 50-456 LER 87-055
SPURIOUS CONTAINMENT ISOLATION SIGNALS CAUSED BY RADIATION DETECTOR 1RT-AR012 FAILURE.
EVENT DATE: 092787 REPORT DATE: 102087 NSSS: WE TYPE: PWR
VENDOR: GENERAL ATOMIC CO.

(NSIC 206850) A TRAIN B CONTAINMENT VENTILATION ISOLATION SIGNAL ANNUNCIATED IN THE MAIN CONTROL ROOM ON THREE SEPARATE OCCASIONS: AT 2301 ON SEPTEMBER 27, 1987, 1903 ON SEPTEMBER 28, 1987; AND 0230 ON OCTOBER 2, 1987. NO EQUIPMENT ACTUALLY OPERATED AS THE CONTAINMENT PURGE ISOLATION VALVES WERE ALREADY IN THE CLOSED POSITION. THE SIGNALS WERE ACKNOWLEDGED BY A CONTROL ROOM OPERATOR AND THE RADIATION MONITOR 1RT-AR012 RETURNED TO NORMAL OPERATING STATUS. THE CONTAINMENT ISOLATION SIGNALS WERE IMMEDIATELY MANUALLY RESET. CAUSE OF THE THIRD OCCURRENCE WAS A DEFECTIVE DETECTOR. CAUSE OF THE FIRST TWO OCCURRENCES IS UNKNOWN. THE MOST PROBABLE CAUSE OF THESE OCCURRENCES IS THE DETECTOR GOING BAD. OTHER POSSIBILITIES INCLUDE: THE AREA AND EQUIPMENT BEING WRAPPED IN PLASTIC AND PAPER WHICH COULD PREVENT PROPER HEAT DISSIPATION, WORK ACTIVITES IN THE AREA, OR RADIO CONSTRUCTION ACTIVITY IN THE AREA HAS BEEN COMPLETED AND THE AREA HAS BEEN RETURNED TO NORMAL. THE RADIO TRANSMISSION EXCLUSION AREA HAS BEEN ENLARGED. THERE HAS BEEN ONE PREVIOUS OCCURRENCE OF A CONTAINMENT OR CONTROL ROOM VENTILATION RADIATION MONITOR DETECTOR FAILURE. HOWEVER, THE CAUSE AND MODE OF FAILURE ARE DIFFERENT.

[23] BRAIDWOOD 1 DOCKET 50-456 LER 87-047
MISSED DIESEL GENERATOR SURVEILLANCE DUE TO A COGNITIVIE PERSONNEL ERROR.
EVENT DATE: 092987 REPORT DATE: 101687 NSSS: WE TYPE: PWR

(NSIC 206851) AT 2340 ON SEPTEMBER 28, 1987, IT WAS DISCOVERED THAT THE REQUIRED SURVEILLANCE FOR THE 1A DIESEL GENERATOR HAD NOT BEEN PERFORMED BY 2120 ON SEPTEMBER 28, 1987. THE DIESEL GENERATOR WAS IMMEDIATELY DECLARED INOPERABLE AND THE SURVEILLANCE WAS STARTED. ALSO, AS A CONSERVATIVE MEASURE, A GENERATING STATION EMERGENCY PLAN (GSEP) UNUSUAL EVENT WAS DECLARED AT 2349 AS A RESULT OF THE 2A DIESEL GENERATOR ALSO BEING INOPERABLE. AT 0008 ON SEPTEMBER 29, 1987, THE GSEP UNUSUAL EVENT WAS TERMINATED FOLLOWING FURTHER REVIEW. AT 0020 ON SEPTEMBER 29, 1987, THE SURVEILLANCE WAS COMPLETED AND THE 1A DIESEL DECLARED OPERABLE. CAUSE OF THE MISSED SURVEILLANCE WAS DUE TO A COGNITIVE PERSONNEL ERROR BY A LICENSED OPERATOR IN THAT A SEMI-ANNUAL COVERSHEET FOR THE MONTHLY SURVEILLANCE WAS USED FOR THE PRIOR SURVEILLANCE. THIS MISLED THE OPERATING DEPARTMENT SURVEILLANCE COORDINATOR IN THAT HE WAS UNABLE TO LOCATE THE EXECUTED COPY OF THE SURVEILLANCE. SUBSEQUENTLY, UPDATING OF THE SURVEILLANCE SCHEDULE WAS DELAYED AND THE OPERATING SHIFT PERSONNEL WERE NOT NOTIFIED OF THE REQIREMENT IN A TIMELY MANNER. CONTRIBUTING WAS A NEW PERSON FULFILLING THE FUNCTION OF THE SURVEILLANCE COORDINATOR. DISCUSSIONS WITH OPERATING PERSONNEL ABOUT THE IMPORTANCE OF VERIFYING THE CORRECT SURVEILLANCE COVERSHEET IS USED WILL BE PERFORMED.

[24] BRAIDWOOD 1 DOCKET 50-456 LER 87-056
CONTROL ROOM VENTILATION SHIFT TO EMERGENCY MAKEUP MODE DUE TO A FAILURE OF RADIATION MONITOR OPR33J PRESSURE SWITCH.
EVENT DATE: 100987 REPORT DATE: 103087 NSSS: WE TYPE: PWR
VENDOR: BARKSDALE COMPANY
GENERAL ATOMIC CO.

(NSIC 206892) AT 0946 ON OCTOBER 9, 1987, WITH CONTROL ROOM VENTILATION TRAIN A IN SERVICE AND TRAIN B IN STANDBY, MAIN CONTROL ROOM OUTSIDE AIR INTAKE RADIATION MONITOR'S GAS CHANNEL PROCESSED A SPURIOUS HIGH RADIATION SIGNAL TO THE ENGINEERED SAFETY FEATURES SYSTEM. THIS CAUSED THE TRAIN B OF CONTROL ROOM VENTILATION TO SHIFT TO ITS EMERGENCY MODE OF OPERATION. THE SIGNAL WAS DETERMINED TO BE A SPURIOUS SPIKE AND TRAIN B OF CONTROL ROOM VENTILATION WAS MANUALLY RESET TO ITS NORMAL MODE OF OPERATION. A NUCLEAR WORK REQUEST WAS INITIATED TO INVESTIGATE THE SOURCE OF NOISE ON THE RADIATION MONITOR. THE INVESTIGATION REVEALED THAT THE PRESSURE SWITCH WAS DEFECTIVE AND WAS REPLACED. NO ADDITIONAL CORRECTIVE ACTION IS CONSIDERED NECESSARY. THERE HAVE BEEN NO PREVIOUS OCCURRENCES OF PRESSURE SWITCH FAILURES IN THE AREA OR PROCESS RADIATION MONITORS.

[25] BRAIDWOOD 1 DOCKET 50-456 LER 87-057
TURBINE TRIP AND SUBSEQUENT REACTOR TRIP DURING MONTHLY TURBINE VALVE CYCLE SURVEILLANCE.
EVENT DATE: 100987 REPORT DATE: 110587 NSSS: WE TYPE: PWR

(NSIC 207151) AT 1700 ON OCTOBER 9, 1987, SURVEILLANCE 1BWOS 3.4.2-A-1, TURBINE THROTTLE, GOVERNOR, REHEAT AND INTERCEPT VALVE WAS IN PROGRESS. THE DIGITAL ELECTRO HYDRAULIC CONTROL SYSTEM (DEH) WAS IN THE AUTOMATIC MODE. MEGAWATT FEEDBACK LOOP WAS OUT, PER PROCEDURE, AND THE TURBINE THROTTLE VALVES WERE IN THE TEST MODE. AT APPROXIMATELY 1725, AS THROTTLE VALVE #2 BEING CYCLED, MEGAWATT OUTPUT BECAME ERRATIC, AND THE GOVERNOR VALVES WENT TO THEIR FULLY CLOSED POSITION. ATTEMPTS WERE MADE TO MANUALLY OPEN THE GOVERNOR VALVES, BUT WERE UNSUCCESSFUL. THE TURBINE TRIPPED ON ANTI-MOTORING WHICH RESULTED IN A REACTOR TRIP AT 1726. THE ROOT CAUSE OF THE EVENT HAS NOT YET BEEN DETERMINED. IMMEDIATE CORRECTIVE ACTION WAS TO ESTABLISH STABLE PLANT CONDITIONS. ADDITIONALLY, TEMPORARY INSTRUMENTATION HAS BEEN INSTALLED ON THE SENSING LINE OF TURBINE IMPULSE PRESSURE TRANSMITTER 1PI-MS002 TO MONITOR THE IMPULSE PRESSURE SIGNAL. ALSO, WESTINGHOUSE, CONTROL AND MONITORING SECTION, OF THE STEAM TURBINE GENERATOR DIVISION, IS CURRENTLY TESTING THE BRAIDWOOD TURBINE DEH SOFTWARE ON THEIR SIMULATOR. WHEN THE ROOT CAUSE OF THE TURBINE TRIP IS DETERMINED, A SUPPLEMENTAL REPORT WILL BE ISSUED. THERE HAVE BEEN NO PREVIOUS OCCURRENCES OF A TURBINE TRIP RESULTING FROM AN UNSTABLE IMPULSE PRESSURE DURING VALVE TESTING.

[26] BROWNS FERRY 1 DOCKET 50-259 LER 86-006 REV 02
UPDATE ON TORNADO MISSILE PROTECTION FOR VENT TOWERS.
EVENT DATE: 011686 REPORT DATE: 091887 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 206864) DURING A DESIGN EVALUATION OF CONTROL BAY VENTILATION MODIFICATIONS, DESIGN ENGINEERS IDENTIFIED AN UNANALYZED CONDITION INVOLVING TORNADO-MISSILE PROTECTION FOR EQUIPMENT LOCATED IN THE CONTROL BAY VENT TOWERS. THESE VENT TOWER BUILDINGS HOUSE COMPONENTS UTILIZED IN THE CONTROL BAY VENTILATION SYSTEM. THE GENERAL CONCERN RELATES TO THE SUSCEPTIBILITY OF THE EQUIPMENT IN THIS AREA TO TORNADO-GENERATED MISSILES. THE DESIGN BASIS FOR PROTECTING EXISTING EQUIPMENT WAS ESTABLISHED AND A DETERMINATION OF WHICH EQUIPMENT LOCATED IN THE VENT TOWER THAT COULD REQUIRE PROTECTION WAS PERFORMED. THE RESULTS OF THIS EVALUATION WERE USED TO PERFORM A SITE SPECIFIC RISK ASSESSMENT. THE ASSESSMENT RESULTS INDICATE THAT THE RISK TO THE EQUIPMENT IS EXTREMELY LOW. MODIFICATIONS TO THE VENT TOWERS ARE NOT REQUIRED.

[27] BROWNS FERRY 1 DOCKET 50-259 LER 87-006 REV 01
UPDATE ON IMPROPER FLOW TESTING OF CONTROL ROOM EMERGENCY VENTILATION FANS LEADS TO PROHIBITED CONDITION.
EVENT DATE: 030287 REPORT DATE: 100287 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 206956) ON 4/3/87, IT WAS DISCOVERED THAT A ROUTINE FLOW RATE SURVEILLANCE ON CONTROL ROOM EMERGENCY VENTILATION (CREV) TRAIN B HAD BEEN PERFORMED INCORRECTLY THE NIGHT BEFORE. THE SURVEILLANCE HAD MADE USE OF NEW FLOW TEST INSTRUMENTATION FOR THE FIRST TIME, AND THE TEST PERSONNEL HAD NOT PROPERLY ZEROED THE INSTRUMENT. ENGINEERS, REVIEWING THE TEST DATA, NOTED THAT A THROTTLE DAMPER ADJUSTMENT HAD BEEN MADE TO MEET THE FLOW ACCEPTANCE CRITERIA. RETESTING WITH A PROPERLY ZEROED INSTRUMENT THEY FOUND THE FLOW TO BE SOMEWHAT BELOW THE DESIGN BASIS REQUIREMENTS. THE THROTTLE DAMPER MAY NOT HAVE BEEN SET TO DELIVER THE DESIGN BASIS FLOW SINCE INITIAL SYSTEM INSTALLATION. OPERATIONS DECLARED CREV B INOPERABLE WHEN INFORMED AT 1620 HOURS. THE FLOW WAS IMMEDIATELY MEASURED AND CORRECTLY SET. HOWEVER, ON THE MORNING OF 3/2/87, WHEN CREV A WAS BELIEVED

OPERABLE, FOUR NEW FUEL BUNDLES WERE MOVED TO THE REFUEL FLOOR AND PLACED IN THE UNIT 1 SPENT FUEL POOL. THE UNIT 1 REACTOR VESSEL AND DRYWELL HEADS WERE IN PLACE, BUT, INTERPRETING THIS FUEL MOVEMENT AS A REFUELING OPERATION, TECH SPECS RELATIVE TO CREV OPERABILITY WERE NOT SATISFIED. CONFUSION REGARDING THE CORRECT ZEROING PROCEDURE FOR THE MICROMANOMETER HAS BEEN RESOLVED, AND NO PERSONNEL WILL PERFORM THE TEST WHO HAVE NOT BEEN INSTRUCTED ON THE CORRECT USE OF THE INSTRUMENT.

[28] BROWNS FERRY 1 DOCKET 50-259 LER 87-012 REV 01
UPDATE ON BREAKER FAILS TO CLOSE AND ENGINEERED SAFETY FEATURE ACTUATES DUE TO INADEQUATE MAINTENANCE PROCEDURE.
EVENT DATE: 050687 REPORT DATE: 092587 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)
VENDOR: GENERAL ELECTRIC CO.

(NSIC 206575) ON MAY 6, 1987, AT 0600 DURING TRANSFER OF A SHUTDOWN BUS TO ITS ALTERNATE POWER SUPPLY, THE ALTERNATE POWER SUPPLY BREAKER FAILED TO CLOSE. THIS CAUSED STANDBY GAS TREATMENT, CONTROL ROOM EMERGENCY VENTILATION, PRIMARY CONTAINMENT ISOLATIONS, AND REACTOR PROTECTION SYSTEM ACTUATIONS. AFTER INSPECTING THE ALTERNATE BREAKER PER PLANT PROCEDURES THE BREAKER WAS RETURNED TO SERVICE. AT 0225 ON MAY 7, 1987, THE ALTERNATE BREAKER AGAIN FAILED TO CLOSE DURING A TRANSFER AND ENGINEERED SAFETY FEATURES ACTUATED. A REINSPECTION OF THE BREAKER FOUND A GREASE AND DIRT BUILDUP ON THE CONTROL CELL LINKAGES WHICH PRVENTED PROPER BREAKER OPERATION. THE LINKAGES WERE CLEANED AND LUBRICATED. AFTER THE SHUTDOWN BUS WAS SUCCESSFULLY TRANSFERRED TO THE ALTERNATE POWER SUPPLY, THE NORMAL SUPPLY BREAKER WAS INSPECTED. DURING THIS WORK ANOTHER ENGINEERED SAFETY FEATURE ACTUATION OCCURRED BECAUSE THE BREAKER LOGIC WAS INADVERTENTLY COMPLETED, OPENING THE ALTERNATE BREAKER. THE CAUSE OF THE BREAKERS FAILURE TO CLOSE WAS A GREASE AND DIRT BUILDUP ON CELL LINKAGES. THE CONTROL CELL LINKAGES ON THE 4 KV PLANT BREAKERS WILL BE INSPECTED, AND CLEANED. MAINTENANCE INSTRUCTIONS WILL BE REVISED TO INSPECT AND CLEAN THE ALL SIMILAR LINKAGES PERIODICALLY.

[29] BROWNS FERRY 1 DOCKET 50-259 LER 87-025
TECH SPECS VIOLATION FOR FAILURE TO PERFORM REQUIRED SURVEILLANCE ON DIESEL GENERATOR DUE TO PROCEDURAL INADEQUACY.
EVENT DATE: 090987 REPORT DATE: 100987 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)

(NSIC 206668) ON SEPTEMBER 9, 1987, WITH UNITS 1, 2 AND 3 IN REFUELING OUTAGES AND COMPLETELY DEFUELED, IT WAS DISCOVERED THAT A TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENT FOR THE DIESEL GENERATORS (DG) HAD NOT BEEN INCORPORATED INTO PLANT INSTRUCTIONS. THE PARTICULAR SURVEILLANCE REQUIRED THAT THE DIESEL START FROM AMBIENT CONDITIONS AND ENERGIZE THE EMERGENCY BUSES WITH THE PERMANENTLY CONNECTED LOADS. CONTRARY TO THIS REQUIREMENT THE NORMALLY CONNECTED 480 V SHUTDOWN BOARDS WHICH SUPPLY LOADS REQUIRED FOR SAFE SHUTDOWN WERE NOT TESTED. THE FAILURE TO TEST THE 480 V SHUTDOWN BOARD LOADS IS ATTRIBUTED TO A PROGRAMMATIC PROBLEM WITH PROCEDURES WHICH HAS BEEN CORRECTED THROUGH A PROCEDURES UPGRADE PROCESS. THE COMPLEXITY OF THE LOAD SHAPING, LOAD SEQUENCING AND OPERATIONAL CONFIGURATION MAKE TESTING OF THE 480 V SHUTDOWN BOARD LOADS A SUBJECT REQUIRING FURTHER EVALUATION. A TECH SPEC CHANGE MAY BE REQUIRED TO ALLOW SIMULATING THESE LOADS.

[30] BROWNS FERRY 1 DOCKET 50-259 LER 87-026
FAILURE TO MEET AUTOMATIC INITIATION FOR CONTROL ROOM EMERGENCY VENTILATION TRAIN A DUE TO DESIGN ERROR.
EVENT DATE: 091587 REPORT DATE: 101387 NSSS: GE TYPE: BWR

OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 206669) DURING A SYSTEM REVIEW OF THE CONTROL ROOM EMERGENCY VENTILATION (CREV) SYSTEM, IT WAS DISCOVERED THAT TRAIN A WOULD NOT AUTOMATICALLY START IF OFFSITE POWER WAS LOST. THE POWER SUPPLY TO THE TRAIN IS LOAD SHED WHEN OFFSITE POWER IS LOST AND IS NOT AUTOMATICALLY REPOWERED. THE CONDITION WAS CAUSED BY DESIGN ERROR. OPERATION ACTIVITIES THAT COULD REQUIRE CREV OPERATION HAVE BEEN STOPPED UNTIL THE CONDITION IS CORRECTED. A POWER SUPPLY WILL BE CONNECTED TO CREV TRAIN A SUCH THAT THE TRAIN WILL AUTOMATICALLY START WHENEVER REQUIRED.

[31] BROWNS FERRY 1 DOCKET 50-259 LER 87-027
INOPERABLE HAND CONTROL VALVE BLOCKS HIGH PRESSURE COOLANT INJECTION TORUS WATER SUPPLY.
EVENT DATE: 091587 REPORT DATE: 101387 NSSS: GE TYPE: BWR
VENDOR: ASSOCIATED CONTROL EQUIPMENT

(NSIC 206670) ON SEPTEMBER 15, 1987, AT 1000 HOURS, WHILE PERFORMING LAYUP MAINTENANCE ON UNIT 1 IT WAS DISCOVERED THAT A 16-INCH HAND CONTROL VALVE IN THE HIGH PRESSURE COOLANT INJECTION SUCTION LINE FROM THE TORUS HAD THE VALVE STEM SEPARATED FROM THE VALVE DISC AND COULD NOT BE OPENED. SINCE THE ASSOCIATED CONTROL EQUIPMENT, INC, HAND CONTROL VALVE CANNOT BE DISSASSEMBLED WITHOUT CUTTING THE VALVE OUT OF THE LINE, THE EXACT CAUSE OF THE STEM FAILURE CANNOT BE DETERMINED. WHEN THE VALVE IS REMOVED FROM THE LINE, THE INTERNALS WILL BE INSPECTED TO DETERMINE THE FAILURE MECHANISM. THE INSPECTION RESULTS WILL BE USED TO DETERMINE IF CORRECTIVE ACTION IS REQUIRED ON SIMILAR VALVES.

[32] BROWNS FERRY 1 DOCKET 50-259 LER 87-024
UNPLANNED REACTOR WATER CLEANUP ISOLATION DUE TO DEGRADED ELECTRICAL CONNECTIONS ON TEMPERATURE SWITCH.
EVENT DATE: 093087 REPORT DATE: 103087 NSSS: GE TYPE: BWR

(NSIC 206780) BETWEEN 2320 HOURS, ON SEPTEMBER 30, 1987, AND 2035 HOURS, ON OCTOBER 1, 1987, BFN UNIT 1 RECEIVED THREE UNPLANNED REACTOR WATER CLEANUP (RWCU) ISOLATIONS. INITIAL INVESTIGATIONS OF RELAY POSITIONS AND FUSES DID NOT IDENTIFY THE CAUSE OF THE ISOLATIONS. SUBSEQUENT INVESTIGATIONS REVEALED THAT THE TEMPERATURE SWITCH MONITORING THE NONREGENERATIVE HEAT EXCHANGER OUTLET TEMPERATURE HAD THREE DEGRADED ELECTRICAL CONNECTIONS. NORMAL WORK ACTIVITIES IN THE AREA AROUND THE SWITCH MAY HAVE CAUSED ENOUGH VIBRATION TO ALLOW BREAKS IN CONTINUITY AND INITIATION OF THE ISOLATIONS. THESE WERE UNPLANNED ENGINEERED SAFETY FEATURE ACTUATIONS. THE ISOLATION WAS RESET TWICE AND RWCU WAS RETURNED TO SERVICE AFTER THE INITIAL INVESTIGATIONS DID NOT INDICATE AN ISOLATION SIGNAL PRESENT. AFTER THE THIRD ISOLATION THE SYSTEM WAS ALLOWED TO REMAIN ISOLATED UNTIL SUBSEQUENT INVESTIGATIONS DETERMINED THE SOURCE OF THE ISOLATIONS AND CORRECTED THE ELECTRICAL CONNECTION PROBLEMS. SIMILAR INVESTIGATIONS HAVE BEEN INITIATED ON THE OTHER TWO UNITS. ALL THREE UNITS WERE DEFUELED DURING THESE EVENTS.

[33] BROWNS FERRY 1 DOCKET 50-259 LER 87-028
UNPLANNED CONTAINMENT ISOLATIONS DUE TO INADEQUATE PROCEDURES.
EVENT DATE: 100887 REPORT DATE: 110687 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 206910) AT 1720 HOURS ON OCTOBER 8, 1987, WITH ALL THREE UNITS DEFUELED, BFN UNIT 3 RECEIVED A DOWNSCALE ALARM FROM A RADIATION MONITOR ON THE REFUEL FLOOR. THE OPERATOR RESET THE ALARM AND THE ALARM CLEARED. THEN THE OPERATOR PLACED THE INSTRUMENT'S OPERATE SWITCH IN THE ZERO POSITION. THE UNIT RECEIVED

ISOLATIONS OF PRIMARY CONTAINMENT (GROUP 6, VENTILATION AND PURGE), SECONDARY CONTAINMENT (REFUEL ZONE), AND THE MAIN CONTROL ROOMS. ALSO INITIATED WERE STANDBY GAS TREATMENT AND CONTROL ROOM EMERGENCY VENTILATION. THESE WERE UNPLANNED ENGINEERED SAFETY FEATURE ACTUATIONS. THE OPERATOR RETURNED THE SWITCH TO OPERATE, RESET THE ISOLATION SIGNALS, AND RETURNED THE AFFECTED PLANT SYSTEMS TO THEIR NORMAL CONFIGURATION. THESE ACTIONS WERE COMPLETED BY 1730 HOURS. THE OPERATOR DID NOT KNOW THAT MOVING THAT INSTRUMENT'S OPERATE SWITCH OUT OF THE OPERATE POSITION WOULD INITIATE THESE ACTIONS. NEITHER THE ALARM RESPONSE PROCEDURE NOR THE RADIATION MONITORING SYSTEM OPERATING INSTRUCTION DIRECTED THE OPERATOR TO TAKE THE SWITCH OUT OF OPERATE NOR DID THEY WARN THE OPERATOR OF THE PLANT RESPONSE TO THAT ACTION. THE RADIATION MONITORING SYSTEM OPERATING INSTRUCTION AND THE RESPECTIVE ALARM RESPONSE PROCEDURES WILL BE REVISED TO PROVIDE A CLEARER DESCRIPTION OF INSTRUMENT RESPONSE. A REVIEW OF THIS EVENT WILL BE PROVIDED TO CURRENT OPERATIONS PERSONNEL.

[34] BROWNS FERRY 2 DOCKET 50-260 LER 87-007 REV 01
UPDATE ON DRYWELL CONTROL AIR ISOLATION VALVES OUTSIDE DESIGN BASIS BECAUSE OF DESIGN AND INSTALLATION ERRORS DURING A MODIFICATION.
EVENT DATE: 083187 REPORT DATE: 102987 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 1 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 206624) ON AUGUST 31, 1987, WITH ALL THREE UNITS DEFUELED, A RESTART TEST PERFORMED ON UNIT 2 DETERMINED THE DRYWELL CONTROL AIR PRIMARY CONTAINMENT ISOLATION VALVE ACTUATORS DID NOT CLOSE THE VALVES ON LOSS OF MOTIVE AIR. THE CONDITION WAS THE RESULT OF PERSONNEL ERROR DURING THE DESIGN AND IMPLEMENTATION OF A MODIFICATION. THE SOLENOID VALVES ASSOCIATED WITH THE VALVE ACTUATORS WERE REPLACED TO MEET ENVIRONMENTAL QUALIFICATION REQUIREMENTS OF 10 CFR 50.49. THE VALVE ACTUATORS WILL BE MODIFIED TO MEET ALL DESIGN REQUIREMENTS.

[35] BROWNS FERRY 2 DOCKET 50-260 LER 87-009
UNPLANNED REACTOR WATER CLEANUP ISOLATION CAUSED BY ELECTRICAL SHORT CIRCUIT.
EVENT DATE: 091187 REPORT DATE: 100987 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 1 (BWR)

(NSIC 206709) ON SEPTEMBER 11, 1987, AT 2230, WITH UNIT 2 DEFUELED, A PRESSURE SWITCH, IN THE REACTOR WATER CLEANUP SYSTEM (RWCU) WAS BEING CALIBRATED. A LEAD FROM THE TEST EQUIPMENT SLIPPED OUT OF THE CONNECTION AND CREATED A SHORT CIRCUIT WHICH BLEW A FUSE. THIS CIRCUIT FEEDS VARIOUS RWCU INSTRUMENTATION ONE OF WHICH IS THE TEMPERATURE SWITCH DOWNSTREAM OF THE NONREGENERATIVE HEAT EXCHANGER. WHEN THE FUSE BLEW, IT DEENERGIZED THE TEMPERATURE SWITCH SIMULATING A HIGH TEMPERATURE CONDITION, AND ISOLATING RWCU. THIS WAS AN ACTUATION OF AN ENGINEERED SAFETY FEATURE. THIS EVENT UNCOVERED A DRAWING ERROR WHICH MISREPRESENTED THE PRESSURE SWITCH'S ACTUAL POWER SUPPLY AND ITS RELATION TO THE TEMPERATURE SWITCH. THE BLOWN FUSE WAS REPLACED AND THE ISOLATION WAS RESET. THE RWCU SYSTEM WAS RETURNED TO SERVICE 1 HOUR 10 MINUTES AFTER THE ISOLATION WAS INITIATED. THE DRAWING ERROR WILL BE CORRECTED FOR UNITS 1 AND 2. THE UNIT 3 DRAWING WAS CORRECT. THE CRAFT PERSONNEL INVOLVED HAVE BEEN COUNSELED ON THE NEED FOR INCREASED CAUTION WHEN WORKING WITH ENERGIZED EQUIPMENT.

[36] BROWNS FERRY 2 DOCKET 50-260 LER 87-010
FIRE WATCH FUNCTION WAS NOT FULFILLED WHILE GRINDING ACTIVITIES WERE IN PROGRESS DUE TO PERSONNEL ERROR.
EVENT DATE: 092487 REPORT DATE: 102387 NSSS: GE TYPE: BWR

(NSIC 206781) ON SEPTEMBER 24, 1987, AT 1415, WITH UNIT 2 DEFUELED AND GRINDING WORK IN PROGRESS A GENERAL HOUSEKEEPING INSPECTION TEAM DISCOVERED A FIRE WATCH ASLEEP IN THE UNIT 2 DRYWELL. FIRE WATCHES ARE REQUIRED BY TECHNICAL

SPECIFICATION 3.11.H FOR THIS TYPE OF WORK. THE FIRE WATCH WAS IMMEDIATELY AWAKENED. DISCIPLINARY ACTION HAS BEEN INITIATED AGAINST THE FIRE WATCH. A DESCRIPTION OF THIS EVENT WILL BE PROVIDED TO PERSONNEL WHO MAY BE INVOLVED IN WORK REQUIRING FIRE WATCHES.

[37] BROWNS FERRY 3 DOCKET 50-296 LER 87-002 REV 01
UPDATE ON UNPLANNED REACTOR WATER CLEANUP ISOLATION DURING TESTING DUE TO FUSE FAILURE AND PERSONNEL ERROR.
EVENT DATE: 090487 REPORT DATE: 100987 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 1 (BWR)
VENDOR: BUSSMANN MFG (DIV OF MCGRAW-EDISON)

(NSIC 206625) TWO SIMILAR ENGINEERED SAFETY FEATURE ACTUATIONS OCCURRED WHILE TESTING THE REACTOR WATER CLEANUP (RWCU) SYSTEMS. ON SEPTEMBER 4, 1987, AT 2025, WITH THE UNIT DEFUELED, A FUNCTIONAL TEST OF THE LOGIC FOR THE PRIMARY CONTAINMENT ISOLATION OF THE RWCU SYSTEM WAS BEING PERFORMED ON UNIT 3. THE TEST INDIVIDUALLY TRIPS PARALLEL REDUNDANT CHANNELS OF ISOLATION LOGIC AND IS NOT INTENDED TO INITATE AN ACTUAL ISOLATION. WHEN CHANNEL A1 WAS DEENERGIZED THERE WAS AN UNEXPECTED ISOLATION OF THE RWCU SYSTEM BECAUSE CHANNEL B2 WAS ALREADY DEENERGIZED, AND THIS COMPLETED THE MINIMUM ACTUATION LOGIC. CHANNEL B2 WAS DEENERGIZED BECAUSE OF A BLOWN FUSE. THE FUSE WAS REPLACED AND THE ISOLATION RESET. THE RWCU SYSTEM WAS RETURNED TO SERVICE 2-1/4 HOURS AFTER THE ISOLATION WAS INITIATED. THE TEST PROCEDURE WILL BE REVISED TO ADD STEPS WHICH VERIFY THAT THE TRIP RELAYS ARE ENERGIZED AT THE START OF THE TEST. ON SEPTEMBER 11, 1987, AT 1810, THE UNIT 1 RWCU SYSTEM ISOLATED DUE TO PERSONNEL ERROR DURING FUNCTIONAL TESTING OF THE PRIMARY CONTAINMENT ISOLATION LOGIC. THE TEST INDIVIDUALLY ACTUATES REDUNDANT TEMPERATURE ELEMENTS AND IS NOT INTENDED TO INITIATE AN ACTUAL ISOLATION. WHEN THE TEMPERATURE SWITCH IN CHANNEL A1 WAS ACTUATED THERE WAS AN UNEXPECTED ISOLATION BECAUSE CHANNEL B2 HAD NOT RESET FROM PREVIOUS TESTING.

[38] BROWNS FERRY 3 DOCKET 50-296 LER 87-003
INADVERTENT REACTOR WATER CLEANUP ISOLATION.
EVENT DATE: 091887 REPORT DATE: 101687 NSSS: GE TYPE: BWR

(NSIC 206673) ON SEPTEMBER 18, 1987, AT 1841 HOURS, THE UNIT 3 REACTOR WATER CLEANUP (RWCU) SYSTEM ISOLATED UNEXPECTEDLY DURING TROUBLESHOOTING MAINTENANCE ON A SECONDARY CONTAINMENT ISOLATION DAMPER. A SECOND ISOLATION OCCURRED AT 1850. INVESTIGATION INTO THE ISOLATIONS DID NOT IDENTIFY A CAUSE, AND THE ISOLATION COULD NOT BE REPEATED DESPITE EFFORTS TO DUPLICATE THE CONDITIONS. NO CORRECTIVE ACTION IS PLANNED AT THIS TIME.

[39] BYRON 1 DOCKET 50-454 LER 87-022
CONTAINMENT VENTILATION ISOLATION DUE TO SPIKE FROM CONTAINMENT BUILDING FUEL HANDLING INCIDENT AREA RAD MONITOR.
EVENT DATE: 102687 REPORT DATE: 111087 NSSS: WE TYPE: PWR
VENDOR: GENERAL ATOMIC CO.

(NSIC 206962) ON OCTOBER 26, 1987, AT 1301, WITH THE PLANT IN POWER OPERATION (MODE 1) AT 98% REACTOR POWER, AREA RADIATION MONITOR 1RT-AR012 (CONTAINMNET BUILDING FUEL HANDLING INCIDENT) SPIKED ABOVE ITS ALERT ALARM SETPOINT. THIS AUTOMATICALLY TRANSFERRED PRIMARY CONTAINMENT MINI PURGE ISOLATION VALVE 1VQ005B TO ITS ENGINEERED SAFETY FEATURES (ESF) POSITION. THE SPIKE IS BELIEVED TO HAVE BENN CAUSED BY A FAULTY DETECTOR. THE DETECTOR WAS REPLACED AND THE MONITOR WAS RETURNED TO SERVICE ON OCTOBER 29, 1987. NOISE SPIKES ON OTHER MONITORS HAVE CAUSED ESF ACTUATIONS IN THE PAST (LER 86-024).

[40] BYRON 2 DOCKET 50-455 LER 87-001 REV 01
UPDATE ON REACTOR TRIP DUE TO 2 OUT OF 4 LOGIC ON OVER TEMPERATURE CHANNEL.
EVENT DATE: 011587 REPORT DATE: 110387 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: BYRON 1 (PWR)
VENDOR: RDF CORP.

(NSIC 207149) AT 2240 ON JANUARY 15, 1987 A UNIT 2 REACTOR TRIP OCCURRED DUE TO A REACTOR COOLANT RESISTANCE TEMPERATURE DETECTOR (RTD) FAILURE, CONCURRENT WITH A PREVIOUSLY TRIPPED BISTABLE REQUIRED FOR A STARTUP TEST PROCEDURE. PRIOR TO THE TRIP, AN INITIAL CRITICALITY/LOW POWER PHYSICS START-UP TEST WAS IN PROGRESS WHICH REQUIRED THE REACTOR COOLANT LOOP D OVER TEMPERATURE DELTA TEMPERATURE OT DELTA T AND OVER TEMPERATURE DELTA TEMPERATURE TURBINE RUNBACK ROD WITHDRAWAL STOP BISTABLES TO BE TRIPPED. DURING PERFORMANCE OF THE TEST THE A LOOP C NARROW RANGE RTD FAILED LOW. THE COMBINATION OF THE PREVIOUSLY TRIPPED LOOP D DELTA T BISTABLE AND LOOP C OT DELTA T BISTALBE TRIP DUE TO THE RTD FAILURE, SATISFIED A 2/4 REACTOR TRIP LOGIC COINCIDENCE FOR OT DELTA T. FOLLOWING THE TRIP ALL SYSTEMS RESPONDED AS REQUIRED. THE EVENT DID NOT COMPROMISE PLANT AND/OR PUBLIC SAFETY AS A MORE CONSERVATIVE PLANT CONDITION RESULTED. THE FAILED RTD WAS BENCH TESTED AND DETERMINED TO BE OPERABLE. THE ACTUAL FAILURE WAS BELIEVED TO BE CAUSED BY A POOR SPLICE CONNECTION BETWEEN THE RTD AND THE FIELD CABLE. THE SPLICE CONNECTION METHODS HAVE BEEN REVISED TO PREVENT RECURRENCE.

[41] BYRON 2 DOCKET 50-455 LER 87-016 REV 01
UPDATE ON INADVERTENT TRAIN A SAFETY INJECTION DURING SURVEILLANCE TEST DUE TO PERSONNEL ERROR.
EVENT DATE: 083187 REPORT DATE: 100887 NSSS: WE TYPE: PWR

(NSIC 206852) ON AUGUST 31, 1987, TWO NUCLEAR STATION OPERATORS (NSO) WERE PERFORMING THE UNIT 2 TRAIN A SOLID STATE PROTECTION SYSTEM BI-MONTHLY SURVEILLANCE. DURING THE SURVEILLANCE ONE NSO NOTICED A LIGHT BULB WAS BURNED OUT. HE REPLACED THE BULB AND THE SURVEILLANCE WAS CONTINUED. THE SECOND NSO ASSUMED THAT THE PREVIOUS STEP IN THE SURVEILLANCE HAD BEEN COMPLETED PRIOR TO THE BULB REPLACEMENT. HOWEVER, STEP 8 IN THE SURVEILLANCE WAS NEVER COMPLETED. FURTHER IN THE SURVEILLANCE, THE MAIN CONTROL ROOM UNIT 2 OPERATOR INFORMED THE TWO NSO'S THAT A SAFETY INJECTION HAD JUST OCCURRED. THE NSO'S PERFORMED THE SYSTEM RESTORATION AND EXITED THE SURVEILLANCE. THE ROOT CAUSE OF THE EVENT WAS A COMMUNICATION BREAKDOWN BETWEEN THE TWO NSO'S PERFORMING THE SURVEILLANCE. ALL TRAIN A SAFETY INJECTION EQUIPMENT STARTED AS DESIGNED, EXCEPT THE 2A DIESEL GENERATOR, WHICH WAS OUT-OF-SERVICE. THE UNIT WAS PROPERY RECOVERED WITHOUT INCIDENT. CORRECTIVE ACTION INCLUDED DISCIPLINARY ACTION AGAINST THE TWO NSO'S PERFORMING THE SURVEILLANCE. THERE HAVE BEEN TWO PREVIOUS OCCURRENCES.

[42] BYRON 2 DOCKET 50-455 LER 87-019
REACTOR TRIP FROM HI-2 STEAM GENERATOR LEVEL AND SUBSEQUENT LOSS OF OFFSITE POWER AS A RESULT OF PERSONNEL ERROR.
EVENT DATE: 100287 REPORT DATE: 103087 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: BYRON 1 (PWR)

(NSIC 207150) ON OCTOBER 2, 1987, AT 0446, UNIT 2 WAS RETURNING TO SERVICE. WHEN UNIT 2 WAS SYNCHRONIZED TO THE GRID, THE STEAM GENERATOR (SG) LEVELS INCREASED AND CAUSED A HI-2 S/G LEVEL TRIP. THE HI-2 SG LEVEL WAS REACHED ON SG 2C DUE TO EXCESSIVE "LEAK BY" OF THE 2FW530 VALVE. THE HIGH S/G LEVEL CAUSED A TURBINE TRIP AND A SUBSEQUENT REACTOR TRIP BECAUSE REACTOR POWER WAS ABOVE 10%. AN EQUIPMENT OPERATOR (EO) WAS INSTRUCTED TO REALIGN THE SWITCHYARD RING BUS AFTER THE TRIP. THE EO OPENED THE SYSTEM AUX TRANSFORMER DISCONNECTS INSTEAD OF THE MAIN POWER TRANSFORMER DISCONNECTS. THE SAFETY RELATED 4KV BUSES WERE DEENERGIZED CAUSING THE EMERGENCY DIESEL GENERATORS TO START, REENERGIZE THE BUSES, AND SEQUENCE THE SAFE SHUTDOWN LOADS. THE ROOT CAUSE OF THE LOSS OF OFFSITE POWER WAS DUE TO PERSONNEL ERROR. THE EO OPENED THE WRONG DISCONNECT.

THE CORRECTIVE ACTIONS ARE AS FOLLOWS: DISCIPLINARY ACTION WAS TAKEN, NO SWITCHYARD OPERATION WILL BE PERFORMED WITHOUT A SECOND PERSON PRESENT, TEMPORARY LABELS, A WALK THROUGH DEMONSTRATION, SAT DISCONNECTS ARE LOCKED, A CHECKLIST HAS BEEN DEVELOPED, ANNUAL HIGH VOLTAGE SWITCHING REQUALIFICATIONS, MODIFICATIONS INSTALLED TO REDUCE VOLTAGE, AND MAIN FEEDWATER REGULATING VALVE REPAIRED.

[43] CALLAWAY 1 DOCKET 50-483 LER 87-017 REV 01
UPDATE ON INOPERABILITY OF SAFETY INJECTION TRAINS AND TECH SPEC LIMITS UNKNOWINGLY ENTERED DUE TO PERSONNEL ERRORS.
EVENT DATE: 062987 REPORT DATE: 111387 NSSS: WE TYPE: PWR

(NSIC 207188) ON 8/4/87 AT 1545 CDT, AN EVALUATION (INITIATED BY A SHIFT TECHNICAL ADVISOR ON 7/17/87) DETERMINED THAT THE PLANT UNKNOWINGLY ENTERED TECHNICAL SPECIFICATION (T/S) 3.0.3. THIS OCCURRED WHEN THE SAFETY INJECTION (SI) PUMPS TO THE REFUELING WATER STORAGE SYSTEM (RWST) HAND CONTROL VALVE BN-HV-8813 WAS MOMENTARILY SHUT ON 6/29/87 AND 7/15/87 FOLLOWING MAINTENANCE OF TWO RESIDUAL HEAT REMOVAL (RHR) VALVES. THIS ACTION ISOLATED THE COMMON SI PUMP MINIFLOW RECIRCULATION LINE TO THE RWST AND RENDERED BOTH SI TRAINS INOPERABLE. ADDITIONALLY, IT WAS DISCOVERED DURING THE REVIEW OF THESE EVENTS THAT QUARTERLY SI PUMP SURVEILLANCES HAVE REQUIRED CYCLING THIS VALVE SINCE RECEIPT OF THE OPERATING LICENSE (6/11/84). THE PLANT WAS IN MODE 1, POWER OPERATIONS AT 100% POWER. THESE EVENTS CAN BE ATTRIBUTED TO PERSONNEL ERRORS IN THE SCHEDULING OF WORK ON THE RHR VALVES. THE QUARTERLY SURVEILLANCE PROCEDURE ERROR WAS DUE TO PERSONNEL ERRORS IN THE PREPARATION AND REVIEW OF THE SURVEILLANCE PROCEDURES. CORRECTIVE ACTIONS INCLUDE ASSURING APPROPRIATE SCHEDULING OF MAINTENANCE ACTIVITIES, ENHANCEMENTS TO THE PROCEDURE REVISION PROCESS, EVALUATIONS OF WORK CONTROL DOCUMENTS AND REVISIONS TO APPLICABLE SURVEILLANCE PROCEDURES. AN ENGINEERING EVALUATION WAS PERFORMED AND IT WAS CONCLUDED THAT AT NO TIME DID THESE EVENTS ENDANGER PUBLIC HEALTH AND SAFETY.

[44] CALLAWAY 1 DOCKET 50-483 LER 87-029
CONTAINMENT PURGE AND CONTROL ROOM VENTILATION ISOLATIONS (ONCE) -- (CAUSE UNKNOWN).
EVENT DATE: 081087 REPORT DATE: 110287 NSSS: WE TYPE: PWR

(NSIC 206986) AT 0342 CDT ON 10/8/87, AN ENGINEERED SAFETY FEATURE (ESF) AUTOMATIC ACTUATION WAS RECEIVED FROM CONTAINMENT PURGE RADIATION MONITOR GT-RE-33. THIS RESULTED IN A CONTAINMENT PURGE ISOLATION (CPI) AND A CONTROL ROOM VENTILATION ISOLATION (CRVI). THE PLANT WAS WITHOUT FUEL IN THE REACTOR VESSEL. THE EVENT IS CONSIDERED A SPURIOUS ACTUATION. NO ABNORMAL AIRBORNE RADIOACTIVITY LEVELS WERE DETECTED. THE ESF SYSTEM FANS AND DAMPERS ACTUATED PROPERLY AND AT 0421 THE CONTAINMENT PURGE AND CONTROL ROOM EMERGENCY VENTILATION SYSTEMS WERE RESTORED TO NORMAL VENTILATION LINEUPS. THE EVENT WAS NOT RELATED TO CONTAINMENT ACTIVITY. ADDITIONALLY, THE ACTUATION PLACED THE ESF SYSTEM IN A SAFEGUARDS LINEUP. THEREFORE, THERE WAS NO THREAT TO THE PUBLIC HEALTH OR SAFETY.

[45] CALLAWAY 1 DOCKET 50-483 LER 87-022 REV 01
UPDATE ON TWO OCCURRENCES OF TECH SPEC 3.0.3 UNKNOWINGLY ENTERED WHEN ACTION STATEMENT FOR INOPERABLE AUX FEEDWATER PRESSURE CHANNEL NOT MET DUE TO PROCEDURAL INADEQUACIES.
EVENT DATE: 082887 REPORT DATE: 101287 NSSS: WE TYPE: PWR

(NSIC 206685) ON 8/28/87, AT 0306 CDT AN AUXILIARY FEEDWATER SUCTION PRESSURE CHANNEL WAS PLACED OUT OF SERVICE TO PERFORM A TECHNICAL SPECIFICATION (T/S) SURVEILLANCE. PER T/S 3.3.2.C.6-H, THIS CHANNEL SHOULD HAVE BEEN PLACED IN A TRIPPED CONDITION WITHIN ONE HOUR OF BEING DECLARED INOPERABLE. AT 0720, UTILITY LICENSED OPERATORS DISCOVERED THAT THE ACTION STATEMENT TIME LIMIT HAD BEEN EXCEEDED AND IMMEDIATELY ENTERED T/S 3.0.3. AT 0730, THE CHANNEL WAS PLACED IN A

TRIPPED CONDITION, THUS EXITING T/S 3.0.3. THE SURVEILLANCE PROCEDURE WAS REVISED. THE CHANNEL TESTED SATISFACTORILY AND WAS RESTORED TO OPERABLE STATUS AT 1809. A REVIEW OF PAST SURVEILLANCES REVEALED A PREVIOUS UNIDENTIFIED OCCURENCE OF THIS EVENT ON 5/5/86. THE PLANT WAS IN MODE 1, POWER OPERATIONS, AT 95% POWER. THE T/S REQUIREMENT TO PLACE THE CHANNEL IN A TRIPPED CONDITION WITHIN ONE HOUR OF INOPERABILITY WAS NOT IDENTIFIED IN THE SURVEILLANCE PROCEDURE AND THUS THE ACTION STATEMENT WAS NOT MET. AS A CONTRIBUTING FACTOR, THIS PARTICULAR CHANNEL, UNLIKE MOST OTHER CHANNELS, IS NOT READILY TESTABLE IN A TRIPPED CONDITION. THE APPROPRIATE SURVEILLANCE PROCEDURES HAVE BEEN REVISED TO PLACE THE CHANNEL IN A TRIPPED CONDITION PRIOR TO THE ONE HOUR TIME LIMIT BEING EXCEEDED.

[46] CALLAWAY 1 DOCKET 50-483 LER 87-023
FAILURE TO ESTABLISH TECH SPEC ALTERNATE CONTINUOUS SAMPLING DURING A SURVEILLANCE DUE TO PROCEDURAL INADEQUACIES.
EVENT DATE: 083187 REPORT DATE: 093087 NSSS: WE TYPE: PWR

(NSIC 206602) TECHNICAL SPECIFICATION (T/S) 3.3.3.10 ACTION 43 REQUIRES ALTERNATE CONTINUOUS SAMPLING WHEN THE PARTICULATE AND IODINE SAMPLER (GT-RE-21A) IS OUT OF SERVICE. ON 8/31/87 AT 0528 CDT, GT-RE-21A AND THE NOBLE GAS ACTIVITY MONITOR (GT-RE-21B) WERE PLACED INTO THE EQUIPMENT OUT OF SERVICE LOG (EOSL) IN PREPARATION FOR A SURVEILLANCE ON GT-RE-21B. AT 1707, THE GT-RE-21A PUMP WAS DE-ENERGIZED AND WAS RESTARTED AT 1749. AT 2057, DURING A REVIEW OF THE ALARM PRINTOUTS, A UTILITY LICENSED OPERATOR DETERMINED THAT AUXILIARY CONTINUOUS SAMPLING HAD NOT BEEN ESTABLISHED AS REQUIRED BY T/S. THE PLANT WAS IN MODE 1, POWER OPERATION, AT 91% POWER DURING THESE EVENTS. THE PRIMARY CAUSE OF THIS EVENT WAS FAILURE OF THE PROCEDURE TO ADEQUATELY COMMUNICATE TO THE COUNT ROOM, OPERATIONS, AND I&C PERSONNEL THE SPECIFIC REQUIREMENTS TO MEET THE T/S BY PERFORMING ALTERNATE CONTINUOUS SAMPLING. COUNT ROOM PERSONNEL COMMENCED ALTERNATE SAMPLING AT 2119 AND CONTINUED SAMPLING UNTIL GT-RE-21A AND GT-RE-21B WERE REMOVED FROM THE EOSL AT 1500 ON 9/1/87. TO PREVENT RECURRENCE, THIS PROCEDURE, AND OTHER PROCEDURES THAT COULD CAUSE A SIMILAR EVENT, WILL BE REVISED TO PRESCRIBE SPECIFIC REQUIREMENTS FOR PERFORMING ALTERNATE CONTINUOUS SAMPLING.

[47] CALLAWAY 1 DOCKET 50-483 LER 87-024
MISSED HOURLY TECH SPEC FIREWATCHES DUE TO RCA ACCESS RESTRICTION FOR PRESENCE OF NOBLE GAS AND DUE TO MISCOMMUNICATION.
EVENT DATE: 090387 REPORT DATE: 100587 NSSS: WE TYPE: PWR

(NSIC 206740) SINCE 7/7/87, HOURLY FIREWATCHES (FW) WERE REQUIRED BY TECHNICAL SPECIFICATION (T/S) 3.7.11 ACTION (A) FOR AN INOPERABLE FIRE-RATED ASSEMBLY LOCATED IN ROOM 3101 ON THE CONTROL BLDG.-1974' LEVEL ACCESSED FROM THE AUXILIARY BLDG-1974 LEVEL (AB-1974) IN THE RADIOLOGICAL CONTROLLED AREA (RCA). THE 0500-0800 CDT FW'S WERE MISSED ON 9/3/87 WHEN ACCESS WAS RESTRICTED TO THE RCA DUE TO THE PRESENCE OF AIRBORNE CONTAMINATION (NOBLE GAS) DETECTED BY RADIATION MONITORS AT 0502. THE PLANT WAS IN MODE 1 - POWER OPERATION AT 89% POWER. AT 0520 ACCESS WAS RESTRICTED TO AB-1974 UNTIL THE ORIGIN, COMPOSITION, AND MAGNITUDE OF THE AIRBORNE CONTAMINATION COULD BE DETERMINED AND A FULL ASSESSMENT OF PERSONNEL SAFETY WAS PERFORMED. THE 0500 TO 0600 FW WAS MISSED BECAUSE THE UTILITY LICENSED SHIFT SUPERVISOR BELIEVED THAT NO T/S FW'S WOULD BE AFFECTED. LACK OF ADEQUATE COMMUNICATION BETWEEN HEALTH PHYSICS (HP), OPERATIONS, AND FW PERSONNEL LED TO THE TWO ADDITIONAL MISSED FW PATROLS. ACCESS WAS RESTORED TO AB-1974 AT 0800. FW PATROL REQUIREMENTS AND DOCUMENTATION WILL BE ENHANCED TO MAKE T/S FW'S MORE READILY APPARENT. THE IMPORTANCE OF REGAINING ACCESS WAS EMPHASIZED. THIS EVENT WAS DISCUSSED WITH FW PERSONNEL. IT WILL BE REVIEWED WITH APPROPRIATE HP AND OPERATIONS PERSONNEL.

[48] CALLAWAY 1 DOCKET 50-483 LER 87-025
CONTAINMENT PURGE, FUEL BUILDING AND CONTROL ROOM VENTILATION ISOLATIONS OCCUR AS RESULT OF A FAILED POWER SUPPLY.
EVENT DATE: 090487 REPORT DATE: 100587 NSSS: WE TYPE: PWR
VENDOR: CONSOLIDATED CONTROLS CORP.

(NSIC 206741) ON 9/4/87, ENGINEERED SAFETY FEATURE (ESF) AUTOMATIC ACTUATIONS OCCURRED RESULTING IN A CONTAINMENT PURGE ISOLATION (CPI), A CONTROL ROOM VENTILATION ISOLATION (CRVI), AND A FUEL BUILDING VENTILATION ISOLATION (FBVI). AT THE TIME OF THE ACTUATIONS THE PLANT WAS IN MODE 1, POWER OPERATION, 88% REACTOR POWER AND MODE 1, 86% REACTOR POWER, RESPECTIVELY. THE ROOT CAUSE OF THE ESF AUTOMATIC ACTUATIONS IS ATTRIBUTABLE TO A FAILURE OF POWER SUPPLY, PS-4, WHICH FEEDS THE ENGINEERED SAFETY FEATURES ACTUATION SYSTEM CABINET SA-036D. FOLLOWING THE FIRST EVENT AT 0220 CDT, AN ASSESSMENT OF INDICATIONS WAS MADE AND THE POWER SUPPLY OUTPUT WAS MONITORED WITH THE SYSTEM RESTORED TO NORMAL FOLLOWING THE SECOND EVENT AT 0823, THE POWER SUPPLY WAS DETERMINED TO BE DEFECTIVE AND REPLACED. SINCE THIS IS AN ENCAPSULATED POWER SUPPLY, THE CAUSE OF FAILURE CANNOT BE DETERMINED. SINCE THE EVENT PLACED THE ESF SYSTEM IN A SAFEGUARDS LINEUP, THERE WAS NO THREAT TO THE PUBLIC HEALTH OR SAFETY.

[49] CALLAWAY 1 DOCKET 50-483 LER 87-026
THREE ENGINEERED SAFETY FEATURES ACTUATIONS ON HIGH RADIATION DURING REACTOR VESSEL HEAD VENT DUE TO IMPROPER TAGOUT OF CONTAINMENT VENTILATION DAMPER.
EVENT DATE: 091687 REPORT DATE: 101687 NSSS: WE TYPE: PWR

(NSIC 206682) ON 9/16/87, AT 1256 CDT, THE CONTAINMENT PURGE ISOLATION (CPI) AND CONTROL ROOM VENTILATION ISOLATION (CRVI) SYSTEMS OF THE ENGINEERED SAFETY FEATURES (ESF) WERE ACTUATED DURING THE VENTING OF THE REACTOR VESSEL HEAD FOLLOWING REACTOR COOLANT SYSTEM DRAINDOWN PRIOR TO REFUEL. TWO SUBSEQUENT CPI ACTUATIONS OCCURRED AT 1402 AND 1409 DURING RESTORATION FROM THE INITIAL CPI. THE PLANT WAS IN MODE 5, COLD SHUTDOWN FOR THESE EVENTS. A NON-LICENSED EQUIPMENT OPERATOR HAD INCORRECTLY PLACED WORKMAN'S PROTECTION ASSURANCE TAG #22 ON CIRCUIT #22 VICE CIRCUIT #38 OF THE 480VAC MOTOR CONTROL CENTER ON 9/13/87. THIS CAUSED THE CONTAINMENT SHUTDOWN PURGE EXHAUST FAN DISCHARGE DAMPER TO FAIL IN A CLOSED POSITION, THUS PREVENTING EXHAUST AIR- FLOW. WITH NO AIRFLOW THROUGH THE SHUTDOWN PURGE, GAS VENTED FROM THE REACTOR VESSEL HEAD CONCENTRATED IN THE EXHAUST DUCT AND EXCEEDED THE SETPOINTS OF THE RADIATION MONITORS. UPON RECEIPT OF THE CPI AND CRVI, LICENSED OPERATORS VERIFIED PROPER ACTUATION OF THE FANS AND DAMPERS AND SHUT THE REACTOR VESSEL HEAD VENT. THE HEAD VENT DISCHARGE WAS THEN RE-ROUTED TO THE CONTAINMENT ATMOSPHERE CONTROL SYSTEM FOR THE REMAINDER OF THE VENTING EVOLUTION. THE IMPROPERLY PLACED TAG WAS WAS REMOVED FROM CIRCUIT #22 AND WAS RESTORED TO NORMAL SERVICE.

[50] CALLAWAY 1 DOCKET 50-483 LER 87-027
INOPERABLE ESSENTIAL SERVICE WATER TRAIN WITHOUT MEETING TECH SPEC 3.7.4 ACTION STATEMENT WHEN VALVE NOT SURVEILLED DUE TO PROGRAMMATIC DEFICIENCIES.
EVENT DATE: 091687 REPORT DATE: 101687 NSSS: WE TYPE: PWR

(NSIC 206675) ON 9/16/87 DURING A UTILITY QUALITY ASSURANCE SURVEILLANCE, IT WAS DISCOVERED THAT AN ESSENTIAL SERVICE WATER (ESW) TRAIN 'A' VALVE, EF-HV-0087, WAS NOT TESTED PER TECHNICAL SPECIFICATION (T/S) 4.0.5 BY 5/4/86, THE T/S ALLOWABLE EXTENSION DATE. IT WAS SATISFACTORILY TESTED ON 7/10/86. ESW TRAIN 'A' WAS THEREFORE INOPERABLE FROM 5/5 TO 7/10. THE CONDITION WENT UNRECOGNIZED AND THE T/S 3.7.4 ACTION STATEMENT WAS NOT ENTERED. THE PLANT WAS IN MODE 1 - POWER OPERATION AT 100% POWER DURING THE EVENT. EF-HV-0087 WAS TAGGED SHUT ON 2/28/86 PER THE WORKMAN'S PROTECTION ASSURANCE (WPA) PROGRAM TO SUPPORT WORK ON A CORROSION CLOG DOWNSTREAM. THUS, IT WAS NOT STROKE TESTED ON THE T/S 4.0.5 DUE DATE WITH OTHER ESW VALVES. ON 4/21/86, THE VALVE WAS PLACED IN THE EQUIPMENT OUT OF SERVICE LOG (EOSL). DUE TO PROGRAMMATIC DEFICIENCIES, UTILITY OPERATIONS

PERSONNEL DID NOT REFER TO THE EOSL WHEN CLEARING THE WPA LATER ON 4/21/86. AS A RESULT, THE VALVE WAS LEFT IN ITS NORMAL POSITION (OPEN) WITHOUT PERFORMING THE TEST LISTED ON THE EOSL. TWO ACTIONS HAD BEEN TAKEN SUBSEQUENT TO THE EVENT BUT PRIOR TO ITS DISCOVERY. (1) WPA SHEETS ARE REQUIRED TO REFERENCE EOSL SHEETS. (2) THE SURVEILLANCE PROGRAM REQUIRES NEW TASK SHEETS TO BE ISSUED WHEN SURVEILLANCES ARE PARTIALLY COMPLETED.

[51] CALLAWAY 1 DOCKET 50-483 LER 87-028
FAILURE TO MAINTAIN CONTINUOUS FIREWATCH FOR BLOCKED OPEN FIRE DOOR WHEN PERSONNEL MISINTERPRETED A FIRE ALARM.
EVENT DATE: 100387 REPORT DATE: 102887 NSSS: WE TYPE: PWR

(NSIC 206843) A TECHNICAL SPECIFICATION CONTINUOUS FIRE WATCH (FW) WAS POSTED IN THE ENGINEERED SAFETY FEATURES SWITCHGEAR ROOMS (3301 AND 3302) BECAUSE THE FIRE DOORS WERE BLOCKED OPEN TO RUN AN ELECTRICAL CABLE BETWEEN THE ROOMS FOR TESTING SWITCHGEAR EQUIPMENT. AT 1027 CDT ON 10/3/87, THE CONTROL ROOM PANTRY IONIZATION TYPE FIRE DETECTOR ACTUATED WHEN AN OPERATOR OVERCOOKED HIS LUNCH CAUSING AN AUDIBLE FIRE ALARM. BETWEEN 1028 AND 1030, PERSONNEL IN ROOMS 3301 AND 3302, INCLUDING THE FW, LEFT THE AREA, MISTAKENLY ASSUMING THAT THE HALON FIRE-SUPPRESSION SYSTEM WAS PREPARING TO DUMP. THE FW WAS RE-ESTABLISHED AT 1040. THE PLANT WAS IN MODE 6, REFUELING. THE FW FAILED TO CONTACT THE CONTROL ROOM PRIOR TO EVACUATING THE AREA. THE FW DID NOT KNOW THE DIFFERENCE BETWEEN THE FIRE ALARM AND THE HALON ALARM. ADDITIONALLY, HE WAS INFLUENCED BY THE OTHER PERSONNEL LEAVING THE AREA. TO PREVENT RECURRENCE, THE TRAINING PROGRAM FOR FW PERSONNEL WILL BE REVISED. ADDITIONAL INSTRUCTIONS ON FIRE ALARMS AND APPROPRIATE ACTIONS TO BE TAKEN WILL BE PROVIDED FOR USE BY ROVING AND CONTINUOUS FW PERSONNEL. TO PREVENT DOOR OBSTRUCTIONS IN THIS AREA, ELECTRICAL RECEPTACLES ARE TO BE INSTALLED IN ROOMS 3301 AND 3302 FOR FUTURE TESTING EVOLUTIONS.

[52] CALVERT CLIFFS 1 DOCKET 50-317 LER 87-013
REACTOR TRIP CAUSED BY REACTOR PUMP SURGE CAPACITOR FAILURE.
EVENT DATE: 091187 REPORT DATE: 100687 NSSS: CE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 206616) AT 0515 ON SEPTEMBER 11, 1987, WHILE OPERATING IN MODE 1 AT 100% POWER, THE UNIT 1 REACTOR AUTOMATICALLY TRIPPED ON A LOW REACTOR COOLANT FLOW SIGNAL INITIATED BY REACTOR COOLANT PUMP 12A BREAKER TRIPPING OPEN. THE RCP BREAKER TRIP WAS DUE TO A FAILED SURGE CAPACITOR. THE SURGE CAPACITOR WAS REPLACED AND THE PUMP RETURNED TO SERVICE ON SEPTEMBER 13, 1987. DURING THE AUTOMATIC SHUTDOWN OF THE UNIT, #12 CHARGING PUMP FAILED TO START ON AN AUTOMATIC SIGNAL. THE PUMP WAS MANUALLY STARTED. THE CAUSE OF THE FAILURE WAS A LOOSE CONNECTION IN THE BREAKER WHICH CONTROLS THE PUMP STARTS. THE CORRECTIVE ACTION IS TO REPLACE THE RCP SURGE CAPACITORS WITH INDUCTORS LOCATED AT THE RCP BREAKER SWITCHGEAR. THIS IS SCHEDULED FOR THE NEXT REFUELING OUTAGE FOR EACH UNIT. THE LOOSE CONNECTION ON THE #12 CHARGING PUMP BREAKER HAS BEEN FIXED AND A PROGRAM IS UNDER WAY TO CHECK ALL SIMILAR BREAKERS FOR LOOSE CONNECTIONS.

[53] CALVERT CLIFFS 1 DOCKET 50-317 LER 87-014
CONTAINMENT PERSONNEL AIR LOCK DOOR GASKET SURVEILLANCE NOT PERFORMED.
EVENT DATE: 101687 REPORT DATE: 110687 NSSS: CE TYPE: PWR

(NSIC 206958) A ROUTINE ENTRY WAS MADE INTO THE UNIT 1 REACTOR CONTAINMENT BUILDING ON 10-09-87 AT 0930 TO SAMPLE SAFETY INJECTION TANKS FOR BORON CONCENTRATION. DUE TO AN ADMINISTRATIVE OVERSIGHT, A CONTAINMENT AIR LOCK DOOR SEAL FUNCTIONAL TEST WAS NOT PERFORMED AS REQUIRED WITHIN 72 HOURS AFTER USE. ON 10-16-87 AT 0650, THIS OVERSIGHT WAS REALIZED AND THE FUNCTIONAL TEST WAS PERFORMED SATISFACTORILY AT 0807. TO PREVENT RECURRENCE, THE APPROPRIATE EMPLOYEES WILL BE COUNSELED AND THE INSTRUCTION OF CONTAINMENT ACCESS

REQUIREMENTS WILL BE ROUTED TO REAPPRISE RESPONSIBLE INDIVIDUALS OF THEIR OBLIGATIONS.

[54] CALVERT CLIFFS 2 DOCKET 50-318 LER 87-006
EHC TURBINE RUPTURE FORCES MANUAL TRIP OF REACTOR.
EVENT DATE: 090787 REPORT DATE: 100187 NSSS: CE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 206617) AT 1513 ON SEPTEMBER 7, 1987 UNIT TWO'S REACTOR WAS MANUALLY TRIPPED IN RESPONSE TO A RUPTURE IN THE MAIN TURBINE'S ELECTRO-HYDRAULIC CONTROL (EHC) SYSTEM HYDRAULIC FLUID TUBING. AT THE TIME OF THE INCIDENT THE UNIT WAS OPERATING AT 100% POWER. THE UNIT WAS MANUALLY TRIPPED AFTER OPERATORS RECEIVED LOW PRESSURE AND LOW LEVEL ALARMS IN THE EHC SYSTEM AND WHEN THE CLOSURE OF THE TURBINE MAIN STEAM VALVES (AND SUBSEQUENT REACTOR TRIP) WAS IMMINENT. THE UNIT'S SAFETY SYSTEMS PERFORMED AS EXPECTED AND THE PLANT WAS RETURNED TO A STABLE CONDITION WITHOUT INCIDENT. SUBSEQUENT INVESTIGATION PINPOINTED THE LOCATION OF THE FLUID RUPTURE TO BE IN A 1/2" STAINLESS STEEL TUBING LEADING TO A PRESSURE TRANSMITTER OFF OF #21 EHC PUMP DISCHARGE LINE. CORRECTIVE ACTION CONSISTED OF REPLACING THE RUPTURED TUBING AND OF INSTALLING ADDITIONAL TUBING CLAMPS TO BETTER SUPPORT THIS TUBING. FRACTURE MECHANIC ANALYSIS DONE ON THE AFFECTED SECTION OF TUBING REVEALED THE CAUSE OF FAILURE TO BE FROM VIBRATION INDUCED METAL FATIGUE. UNIT TWO WAS RESTARTED AND PARALLELED TO THE ELECTRIC DISTRIBUTION GRID AT 0222 ON SEPTEMBER 8, 1987.

[55] CALVERT CLIFFS 2 DOCKET 50-318 LER 87-007
UNPLANNED ACTUATION OF AUXILIARY FEEDWATER DURING A MAINTENANCE EVOLUTION.
EVENT DATE: 101287 REPORT DATE: 111187 NSSS: CE TYPE: PWR
VENDOR: VITRO ENGINEERING DIVISION

(NSIC 206924) AT 1357 HOURS ON OCTOBER 12, 1987, CALVERT CLIFFS UNIT 2 WAS OPERATING IN MODE 1 AT 100% POWER WHEN AN UNPLANNED ACTUATION OF THE AUXILIARY FEEDWATER ACTUATION SYSTEM (AFAS) WAS EXPERIENCED. THE AUTOMATIC ACTUATION OF THE AFAS OCCURRED DURING A MAINTENANCE EVOLUTION, WHICH REQUIRED ONE OF FOUR SENSOR CHANNELS TO BE DE-ENERGIZED. TWO OF FOUR SENSOR CHANNEL COINCIDENCE IS REQUIRED. "AFAS START" AND "AFAS BLOCK" LOGIC SIGNALS WERE SIMULTANEOUSLY GENERATED WHEN AFAS ACTUATED. TO ENSURE AFW OPERABILITY WITH ALL SENSOR CHANNELS POWERED SURVEILLANCE TEST PROCEDURE 0.9.2, AFAS MONTHLY LOGIC CHECK, WAS IMMEDIATELY CONDUCTED WITH SATISFACTORY RESULTS. ENGINEERING TEST PROCEDURE NO. 87-26, UNIT 2 AFAS CABINET TROUBLESHOOTING, WAS CONDUCTED ON NOVEMBER 10, 1987 WITH THE FOLLOWING RESULTS: 1) NO UNUSUAL OCCURRENCES COULD BE REPEATED DURING SUBSEQUENT CABINET DE-ENERGIZATIONS, AND 2) ALL INDICATED SYMPTOMS OCCURRED ONLY WHEN TWO SENSOR CABINETS WERE SECURED SIMULTANEOUSLY. NO PREVIOUS SIMILAR EVENTS HAVE OCCURRED WITH AFAS.

[56] CATAWBA 1 DOCKET 50-413 LER 87-027 REV 01
UPDATE ON AUXILIARY FEEDWATER PUMP INOPERABLE DUE TO INSTRUMENTATION BEING UNKNOWINGLY ISOLATED.
EVENT DATE: 070687 REPORT DATE: 102987 NSSS: WE TYPE: PWR

(NSIC 207112) ON JULY 6, 1987, AT 1516:52 HOURS, FOLLOWING A MANUAL REACTOR TRIP (SEE LER 413/87-26), THE AUXILIARY FEEDWATER (CA) PUMP 1B DISCHARGE TO STEAM GENERATOR (S/G) 1C ISOLATION VALVE CLOSED AUTOMATICALLY DUE TO A FLOW OPTIMIZATION ALIGNMENT SIGNAL. CONTROL ROOM PERSONNEL NOTED THIS WHEN THEY OBSERVED ZERO FLOW TO S/G 1C AFTER SECURING TURBINE DRIVEN CA PUMP 1. THE VALVE WAS OPENED MANUALLY UPON DISCOVERY, REESTABLISHING FLOW TO S/G 1C FROM MOTOR DRIVEN CA PUMP 1B. THE UNIT WAS IN MODE 3, HOT STANDBY, ON JULY 7 WHEN THE CAUSE FOR THE VALVE CLOSURE WAS DISCOVERED. THIS INCIDENT IS CLASSIFIED AS EVENT CAUSE CODE X, OTHER. THE AUTOMATIC CLOSURE OF THE VALVE WAS DUE TO AN ISOLATED

PRESSURE SWITCH AT THE DISCHARGE OF CA PUMP 1A. THE RESPONSIBLE ACTIVITY WHICH RESULTED IN ISOLATION OF THIS PRESSURE SWITCH COULD NOT BE DETERMINED. FLOW TO S/G 1C WAS ASSURED BY RESTARTING TURBINE DRIVEN CA PUMP 1 AND ANNUALLY OPENING THE AFFECTED VALVE. THE PRESSURE SWITCH WAS CALIBRATED AND VALVED IN SHORTLY AFTER DISCOVERY. THE HEALTH AND SAFETY OF THE PUBLIC WERE UNAFFECTED BY THIS INCIDENT.

[57] CATAWBA 1 DOCKET 50-413 LER 87-035
POTENTIAL CONTROL ROOM AREA VENTILATION AND CHILLED WATER SYSTEM INOPERABILITY DURING DIESEL GENERATOR LOAD SEQUENCER TESTING DUE TO A PROCEDURAL DEFICIENCY.
EVENT DATE: 081087 REPORT DATE: 092587 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: CATAWBA 2 (PWR)

(NSIC 206502) ON AUGUST 10, 1987, AT 1000 HOURS, IT WAS DISCOVERED THAT PLACING A DIESEL GENERATOR (D/G) LOAD SEQUENCER IN TEST ON ONE UNIT WOULD PREVENT ACTUATION OF THE CONTROL ROOM AREA VENTILATION (VC) CHILLED WATER (YC) SYSTEM ON THAT TRAIN IF A LOSS OF COOLANT ACCIDENT (LOCA) SIGNAL WAS RECEIVED ON THE OTHER UNIT. THE DISCOVERY WAS MADE WHILE PREPARING A TEST PROCEDURE FOR RETYPE. BOTH UNITS HAD OPERATED IN ALL MODES PRIOR TO DISCOVERY OF THE EVENT. ALTHOUGH THE EVENT IS NOT REPORTABLE, ON AUGUST 26, 1987, DUKE POWER DECIDED TO SUBMIT THIS REPORT AS A VOLUNTARY LER FOR INFORMATION PURPOSES. THIS INCIDENT IS ATTRIBUTED TO A DEFECTIVE PROCEDURE. VARIOUS STATION PROCEDURES THAT PLACED THE D/G LOAD SEQUENCERS IN TEST DID NOT ENSURE THAT THE TRAIN OF VC/YC ASSOCIATED WITH THE D/G LOAD SEQUENCER UNDER TEST WAS DECLARED INOPERABLE. APPROPRIATE PROCEDURES WILL BE REVISED TO ENSURE THAT THE APPLICABLE TRAIN OF VC/YC IS DECLARED INOPERABLE WHENEVER THAT TRAIN'S D/G LOAD SEQUENCER FOR EITHER UNIT IS PLACED IN TEST. REVIEW OF PREVIOUSLY COMPLETED PROCEDURES DID NOT REVEAL ANY INSTANCE WHEN TECHNICAL SPECIFICATIONS WERE VIOLATED AS A RESULT OF THIS PROBLEM. DISCUSSION WITH DUKE POWER STATION PERSONNEL INDICATED THAT THE POSSIBILITY EXISTS WHERE ONE TRAIN'S D/G LOAD SEQUENCER WAS IN TEST WITH THE OTHER TRAIN OF VC/YC SIMULTANEOUSLY INOPERABLE.

[58] CATAWBA 1 DOCKET 50-413 LER 87-034
REACTOR TRIP BREAKERS OPEN WHILE IN HOT STANDBY DUE TO FAILURE OF SOURCE RANGE NEUTRON DETECTOR DURING A UNIT SHUTDOWN REQUIRED BY TECHNICAL SPECIFICATIONS.
EVENT DATE: 082387 REPORT DATE: 092287 NSSS: WE TYPE: PWR
VENDOR: ELECTROMAX INSTRUMENTS, INC.
WESTINGHOUSE ELECTRIC CORP.

(NSIC 206501) ON AUGUST 23, 1987, UNIT 1 WAS IN THE PROCESS OF SHUTTING DOWN DUE TO AN INABILITY TO SATISFY TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS FOR REACTOR COOLANT (NC) SYSTEM LEAKAGE. NC PUMP LB AND LC SEAL LEAKOFF RATES WERE LOW AND ERRATIC, CAUSING UNSTABLE CONDITIONS FOR CALCULATING NC LEAKAGE. WITH THE INTERMEDIATE RANGE (I/R) POWER LEVEL AT 10 E(-10) AMPERES AND DECREASING, THE SOURCE RANGE (S/R) NEUTRON DETECTORS ENERGIZED AUTOMATICALLY. S/R DETECTOR N31 IMMEDIATE FAILED HIGH AND GENERATED A REACTOR TRIP SIGNAL. BOTH REACTOR TRIP BREAKERS OPENED, AND ALL UNINSERTED CONTROL RODS FELL INTO THE CORE. A MAIN FEEDWATER (CF) ISOLATION OCCURRED DUE TO THE BREAKERS OPENING WITH NC SYSTEM AVERAGE TEMPERATURE BELOW 564 DEGREES F. THE CF ISOLATION TRIPPED THE OPERATING CF PUMP, WHICH GENERATED AN AUXILIARY FEEDWATER (CA) PUMP AUTO-START. THE UNIT WAS IN MODE 3, HOT STANDBY, WHEN THE TRIP SIGNAL OCCURRED. THIS INCIDENT IS CLASSIFIED AS EVENT CAUSE CODE X, OTHER, DUE TO THE FAILURE OF THE S/R NEUTRON DETECTOR. THE FAILURE WAS MOST LIKELY CAUSED BY WATER IN THE DETECTOR CANISTER AND POSSIBLY INSIDE THE DETECTOR ASSEMBLY ITSELF. THE DETECTOR ASSEMBLY WAS REPLACED WITH A NEWER DESIGN THAT INCLUDES PROVISIONS FOR WATER DRAINAGE OUT OF THE ASSEMBLY. DRAIN HOLES IN THE CANISTERS WILL BE MODIFIED TO TWICE THEIR CURRENT DIAMETER.

[59] CATAWBA 1 DOCKET 50-413 LER 87-036
TECH SPECS VIOLATION REGARDING INOPERABILITY OF THE NUCLEAR SERVICE WATER SYSTEM DUE TO INCORRECT DESIGN RECOMMENDATION.
EVENT DATE: 083087 REPORT DATE: 103087 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 206853) ON OCTOBER 12, 1987, DUKE POWER PERSONNEL DETERMINED THAT A VIOLATION OF TECH SPECS OCCURRED FROM 2030 HOURS ON AUGUST 30, 1986 TO 0830 HOURS ON SEPTEMBER 4, 1987. THE DISCOVERY WAS MADE DURING THE PREPARATION OF THIS LER, WHICH AT THE TIME OF DISCOVERY WAS TO BE SUBMITTED AS A VOLUNTARY LER. SUBMITTAL OF THIS REPORT WAS ORIGINALLY SCHEDULED FOR SEPTEMBER 30, 1987, BUT IT WAS DELAYED TO OCTOBER 30, 1987 PER A SEPTEMBER 30, 1987 LETTER FROM H. B. TUCKER TO THE NRC DOCUMENT CONTROL DESK. ON AUGUST 22, 1986, UNIT 1 WAS IN MODE 5, COLD SHUTDOWN, PREPARING FOR REFUELING AND DIESEL GENERATOR (D/G) 1A WAS REMOVED FROM SERVICE FOR MAINTENANCE. UNIT 2 WAS IN MODE 1, POWER OPERATION. DESIGN ENGINEERING ISSUED THROTTLED NUCLEAR SERVICE WATER (RN) FLOWS AND RV CROSSOVER SUPPLY VALVE ALIGNMENTS THOUGHT TO ACCOMMODATE THE SITUATION BUT WHICH WERE INCORRECT. ON AUGUST 30, 1986, AT APPROXIMATELY 2030 HOURS, THE INCORRECT ALIGNMENT WAS PERFORMED RENDERING RN TRAIN A UNKNOWINGLY INOPERABLE WITH RESPECT TO UNIT 2 POWER OPERATION, AND TECH SPECS REQUIRED THAT UNIT 2 BE PLACED IN MODE 5, COLD SHUTDOWN, BY 0830 HOURS, ON SEPTEMBER 4, 1986. UNIT 2 ENTERED MODE 5 ON SEPTEMBER 8 AT 0440 HOURS FOR MAIN TURBINE GENERATOR MAINTENANCE.

[60] CATAWBA 2 DOCKET 50-414 LER 87-023 REV 01
UPDATE ON UNUSUAL EVENT DECLARED BECAUSE OF UNISOLABLE CONTAINMENT VALVE DUE TO A MANAGEMENT DEFICIENCY.
EVENT DATE: 080787 REPORT DATE: 102087 NSSS: WE TYPE: PWR
VENDOR: BORG-WARNER CORP.

(NSIC 206834) ON AUGUST 7, 1987, AT 0340 HOURS, WITH UNIT 2 AT 85% POWER, A UNIT SHUTDOWN WAS COMMENCED PER TECHNICAL SPECIFICATIONS WHEN ONE OF THE STEAM GENERATOR (S/G) MAIN FEEDWATER BYPASS TO AUXILIARY FEEDWATER (CA) NOZZLE VALVES WAS DISCOVERED PARTIALLY OPEN AND UNISOLABLE WHILE PERFORMING THE RETEST FOR THE TURBINE DRIVEN CA PUMP DISCHARGE CHECK VALVE TO S/ G 2B. DUKE POWER STATION PERSONNEL DECLARED THE VALVE INOPERABLE, COMMENCED UNIT SHUTDOWN, AND DECLARED AN UNUSUAL EVENT AS REQUIRED BY TECHNICAL SPECIFICATIONS. THE UNIT ENTERED MODE 3, HOT STANDBY, AT 0735 HOURS. THE VALVE WAS REPAIRED AND SUCCESSFULLY RETESTED AT 1854 HOURS. THE STATION WAS SECURED FROM THE UNUSUAL EVENT AT 1915 HOURS. DUKE POWER PERSONNEL HAD ORIGINATED A WORK REQUEST ON SEPTEMBER 9, 1986, AT 1500 HOURS TO INVESTIGATE/REPAIR 2CA150 PASSING APPROXIMATELY 300 GPM WHILE INDICATING CLOSE. THE UNIT WAS IN MODE 5, COLD SHUTDOWN, AT THE TIME AND THE VALVE WAS NOT REQUIRED TO BE OPERABLE. THIS WORK REQUEST WAS SCHEDULED FOR COMPLETION DURING THE UNIT'S FIRST REFUELING OUTAGE. THIS INCIDENT IS ATTRIBUTED TO A MANAGEMENT DEFICIENCY. THE FAILURE OF THE VALVE IS DUE TO THE IMPROPER SETTING OF THE AIR ACTUATOR FOLLOWING MAINTENANCE ACTIVITY.

[61] CATAWBA 2 DOCKET 50-414 LER 87-025
REACTOR TRIP DUE TO A STEAM GENERATOR OVERFILL BECAUSE OF PERSONNEL ERRORS AND A MANAGEMENT DEFICIENCY.
EVENT DATE: 090387 REPORT DATE: 100287 NSSS: WE TYPE: PWR
VENDOR: FISHER CONTROLS CO.

(NSIC 206746) ON SEPTEMBER 3, 1987, AT APPROXIMATELY 0617 HOURS, AN AUTOMATIC MAIN FEEDWATER (CF) ISOLATION AND MAIN TURBINE GENERATOR TRIP OCCURRED DUE TO A HIGH HIGH STEAM GENERATOR (S/G) WATER LEVEL. THE CF ISOLATION CAUSED THE CF PUMP TURBINE (CFPT) TO TRIP AND THE MOTOR DRIVEN AUXILIARY FEEDWATER (CA) PUMPS STARTED AUTOMATICALLY. REACTOR POWER WAS BEING MANUALLY RUN BACK FROM 22% POWER WHEN AT APPROXIMATELY 0619 HOURS, AN AUTOMATIC REACTOR TRIP OCCURRED FOLLOWING A S/G LOW LOW LEVEL REACTOR TRIP SIGNAL WHILE TRANSFERRING CF FLOW TO THE LOWER S/G

FEEDWATER NOZZLE. THE TURBINE DRIVEN CA PUMP STARTED AUTOMATICALLY UPON RECEIPT OF A SECOND S/G LOW LOW LEVEL ALARM. FOLLOWING THE REACTOR TRIP, S/G LEVELS WERE RESTORED TO NORMAL, AND THE UNIT WAS STABILIZED IN MODE 3, HOT STANDBY. THE UNIT ENTERED MODE 1 AT 1108 HOURS ON SEPTEMBER 4, 1987. THIS INCIDENT IS ATTRIBUTED TO A PERSONNEL ERROR BECAUSE THE CONTROL ROOM OPERATORS (CROS) DECIDED TO CONTINUE THE NOZZLE TRANSFER WITHOUT FIRST SEEKING ADVICE FROM THE UNIT SUPERVISOR WHEN THEY RECOGNIZED THAT THE CF CONTROL VALVE WAS IN AN ABNORMAL CONDITION. THIS INCIDENT IS ALSO ATTRIBUTED TO A MANAGEMENT DEFICIENCY. THE UNIT SUPERVISOR IN THE CONTROL ROOM DID NOT EXERCISE APPROPRIATE SUPERVISORY CONTROL OVER THE TWO CROS. THE INCIDENT WILL BE REVIEWED WITH THE APPROPRIATE PERSONNEL.

[62] CATAWBA 2 DOCKET 50-414 LER 87-026
SHUTDOWN DUE TO BEARING FAILURE ON TURBINE DRIVEN AUXILIARY FEEDWATER PUMP.
EVENT DATE: 091287 REPORT DATE: 101287 NSSS: WE TYPE: PWR

(NSIC 206761) ON SEPTEMBER 12, 1987, AT 0800 HOURS, WITH THE UNIT AT 100% POWER, A UNIT SHUTDOWN WAS COMMENCED DUE TO THE INOPERABILITY OF TURBINE DRIVEN AUXILIARY FEEDWATER (CA) PUMP NO. 2. THE PUMP WAS DECLARED INOPERABLE ON SEPTEMBER 9, 1987, AT 1055 HOURS IN ORDER TO TEST ITS PERFORMANCE TO SATISFY ITS TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS. THE CA PUMP EXPERIENCED A HIGH PUMP OUTBOARD BEARING TEMPERATURE LATER THAT DAY WHILE BEING RUN FOR THE TESTING, REQUIRING THE BEARING TO BE REPLACED. PROBLEMS ENCOUNTERED DURING THE BEARING REPLACEMENT RESULTED IN THE PUMP BEING INOPERABLE LONGER THAN WAS ALLOWED BY TECHNICAL SPECIFICATIONS, REQUIRING THAT THE UNIT BE SHUT DOWN TO MODE 3, HOT STANDBY. THE UNIT ENTERED MODE 3 ON SEPTEMBER 12, 1987, AT 1315 HOURS. THE BEARING WORK WAS COMPLETED AND THE PUMP WAS DECLARED OPERABLE ON SEPTEMBER 14, 1987, AT 0715 HOURS. THIS INCIDENT IS ASSIGNED CAUSE CODE X, OTHER. THE CA PUMP BEARING FAILURE RESULTED IN THE UNIT SHUTDOWN. MAINTENANCE PERSONNEL REPLACED THE PUMP OUTBOARD BEARING. THE FUNCTIONAL TEST WAS SATISFACTORILY COMPLETED AND THE PUMP WAS RETURNED TO OPERABLE STATUS. A FAILURE ANALYSIS HAS BEEN ORDERED FOR THE BEARINGS REPLACED DURING THIS INCIDENT. THIS INCIDENT IS REPORTABLE PURSUANT TO 10CFR50.73, SECTION (A)(2)(I)(A) AND 10CFR50.72, SECTION (B)(1)(I)(A).

[63] CATAWBA 2 DOCKET 50-414 LER 87-027
MANUAL REACTOR TRIP DUE TO FEEDWATER CONTROL VALVE CIRCUIT CARD FAILURE.
EVENT DATE: 091587 REPORT DATE: 101587 NSSS: WE TYPE: PWR
VENDOR: BORG-WARNER CORP.
FISHER FLOW CONTROL DIV (ROCKWELL INT)
ROTORK INC.
WESTINGHOUSE ELECTRIC CORP.

(NSIC 206762) ON SEPTEMBER 15, 1987, AT 0253:35 HOURS, UNIT 2 WAS MANUALLY TRIPPED DUE TO IMPENDING LOW STEAM GENERATOR (S/G) 2B LEVEL. THE TRANSIENT WAS INITIATED WHEN THE S/G 2B MAIN FEEDWATER (CF) CONTROL VALVE FAILED CLOSED CAUSING S/G 2B LEVEL TO DECREASE RAPIDLY. CONTROL ROOM (CR) PERSONNEL ATTEMPTED TO MANUALLY REOPEN THIS VALVE BUT IT WOULD NOT RESPOND. CR PERSONNEL MANUALLY OPENED THE S/G 2B FEEDWATER BYPASS CONTROL VALVE IN AN ATTEMPT TO RECOVER S/G FEED FLOW. HOWEVER, S/G 2B LEVEL DECREASED BELOW 20% AND THE REACTOR WAS MANUALLY TRIPPED. FOLLOWING THE REACTOR TRIP, AN AUXILIARY FEEDWATER (CA) AUTOSTART OCCURRED DUE TO 2 OUT OF 4 LOW LOW S/G LEVELS. ALSO, A SECOND CA AUTOSTART SIGNAL OCCURRED ON LOSS OF BOTH CF PUMPS WHEN CF PUMP 2A TRIPPED WHILE CR PERSONNEL WERE ATTEMPTING TO RESTART IT. THE UNIT WAS AT 48% POWER AT THE TIME OF THE REACTOR TRIP AND WAS IN MODE 3, HOT STANDBY, AT THE TIME OF THE SECOND CA AUTOSTART. THIS INCIDENT IS ASSIGNED CAUSE CODE X, OTHER. THE CAUSE OF THE CONTROL VALVE CLOSING WAS A FAILURE OF THE VALVE'S CONTROLLER/DRIVER CARD IN THE 7300 PROCESS CABINET. SEVERAL COMPONENTS IN THE CARD'S POWER SUPPLY CIRCUIT OVER HEATED RESULTING IN A ZERO VALVE DEMAND SIGNAL. THE SECOND CA AUTOSTART SIGNAL IS CLASSIFIED AS CAUSE CODE X, OTHER.

[64] CATAWBA 2 DOCKET 50-414 LER 87-028
CONTAINMENT PRESSURE CHANNEL INOPERABLE DUE TO PRESSURE TRANSMITTER ISOLATION VALVE CLOSED FOR UNKNOWN REASON.
EVENT DATE: 091587 REPORT DATE: 101587 NSSS: WE TYPE: PWR
VENDOR: DRAGON VALVE, INC.
ELECTROMAX INSTRUMENTS, INC.

(NSIC 206835) ON SEPTEMBER 15, 1987, WITH THE UNIT IN MODE 2, STARTUP, PERSONNEL DISCOVERED THE TRAIN B WIDE RANGE (W/R) CONTAINMENT PRESSURE ISOLATION VALVE CLOSED. SUBSEQUENT INVESTIGATION OF THE PRESSURE TRANSMITTER'S HISTORY REVEALED THAT THE ISOLATION VALVE HAD BEEN FOUND CLOSED DURING A LEAK RATE TEST IN JUNE, 1986. THE LEAK RATE TEST PROCEDURE IDENTIFIED VALVE POSITIONS REQUIRED DURING THE TEST BUT NOT NORMAL POSITIONS IN WHICH VALVES SHOULD BE FOUND, THEREFORE THE ISOLATION VALVE WAS RETURNED TO ITS AS FOUND POSITION. PRIOR TO JUNE, 1986, ALL ASSOCIATED COMPLETED PROCEDURES INDICATED THAT THE ISOLATION VALVE WAS INDEPENDENTLY VERIFIED AS LEFT IN THE OPEN POSITION. THIS INCIDENT HAS BEEN ASSIGNED CAUSE CODE X, OTHER. THE EXACT TIME THE ISOLATION VALVE WAS FIRST LEFT CLOSED COULD NOT BE IDENTIFIED DUE TO NORMAL CONTAINMENT PRESSURE VARIATIONS BEING TOO SMALL TO BE SEEN ON THE WIDE RANGE CHART RECORDER. SINCE THE VALVE HANDLE WAS SECURE ON THE VALVE STEM AND NO OTHER ABNORMALITIES WERE NOTED, THE ROOT CAUSE OF THIS INCIDENT REMAINS UNKNOWN. THE VALVE WAS RETURNED TO THE OPEN POSITION AFTER THE CHANNEL CALIBRATION. A CHECK OF ALL SIMILAR ISOLATION VALVES WAS COMPLETED AND NO OTHER VALVES WERE FOUND OUT OF PROPER ALIGNMENT. THE CONTAINMENT ISOLATION VALVE LEAK RATE TEST PROCEDURE WILL BE MODIFIED. THIS INCIDENT IS REPORTABLE PURSUANT TO 10CFR 50.73, SECTION (A)(2)(I)(B).

[65] CLINTON 1 DOCKET 50-461 LER 87-054
VIOLATION OF THE PLANT'S TECHNICAL SPECIFICATIONS RESULTING FROM FAILURE TO ADEQUATELY TRACK AND PERFORM A CHEMISTRY SURVEILLANCE.
EVENT DATE: 091687 REPORT DATE: 100687 NSSS: GE TYPE: BWR

(NSIC 206631) ON SEPTEMBER 16, 1987 AT 1230 HOURS, WITH THE PLANT IN MODE 1 (POWER OPERATION) AT APPROXIMATELY 95% REACTOR POWER, REVIEW OF CHEMISTRY SURVEILLANCE RESULTS BY THE CHEMIST-NUCLEAR IDENTIFIED THAT THE PARTICULATE FILTER AND IODINE CARTRIDGE FOR HEATING VENTILATING AND AIR CONDITIONING (HVAC) STACK PROCESS RADIATION MONITOR (PRM) ORIX-PROOL WAS NOT CHANGED OUT WITHIN THE SEVEN DAY SAMPLING PERIOD REQUIRED BY THE TECHNICAL SPECIFICATIONS. THE SURVEILLANCE WAS PERFORMED APPROXIMATELY FOUR DAYS LATE. THE CAUSE OF THE EVENT IS ATTRIBUTED TO UTILITY PERSONNEL ERROR. THE CHEMIST-NUCLEAR FAILED TO ADEQUATELY TRACK THE NEED TO PERFORM THE SURVEILLANCE WITHIN THE SEVEN DAY REQUIREMENT. THE CHEMIST-NUCLEAR WAS COUNSELLED FOR THE FAILURE TO ADEQUATELY TRACK THE SURVEILLANCE REQUIREMENT. THE SAFETY SIGNIFICANCE OF THE EVENT WAS ASSESSED AS INSIGNIFICANT SINCE CONTINUOUS MONITORING OF THE HVAC EXHAUST WAS AVAILABLE DURING THE PERIOD THAT THE PRM EXCEEDED THE SURVEILLANCE REQUIREMENT; AND, ANALYSIS OF THE PARTICULATE FILTER AND IODINE CARTRIDGE FOUND NO UNUSUAL LEVELS OF ACTIVITY. THIS EVENT IS REPORTABLE UNDER THE PROVISIONS OF LOCFR50.73(A)(2)(I)(B) DUE TO AN OPERATION PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATIONS.

[66] CLINTON 1 DOCKET 50-461 LER 87-055
REACTOR TRIP ON HIGH WATER LEVEL DUE TO FAULTY FUNCTION GENERATOR CARD IN FEEDWATER SYSTEM.
EVENT DATE: 092187 REPORT DATE: 101687 NSSS: GE TYPE: BWR
VENDOR: BAILEY METER COMPANY

(NSIC 206734) ON SEPTEMBER 21, 1987 AT 1245 HOURS, WITH THE PLANT IN MODE 1 (POWER OPERATION) AT APPROXIMATELY 98% REACTOR POWER, THE REACTOR PROTECTION SYSTEM (RPS) AUTOMATICALLY ACTUATED. THE ACTUATION OCCURRED DURING REACTOR

FEEDWATER CONTROL SYSTEM RESPONSE TESTING FOR THE POWER ASCENSION TEST PROGRAM. THE TEST PROCEDURE CALLED FOR A -20% STEP CHANGE TO BE INSERTED INTO THE "A" REACTOR FEED PUMP (RFP), FOLLOWED BY A +20% STEP CHANGE. WHEN THE +20% STEP CHANGE WAS INSERTED, THE STANDBY CONDENSATE/CONDENSATE BOOSTER PUMPS BOTH AUTOMATICALLY STARTED CAUSING AN INCREASED FLOW OF WATER INTO THE REACTOR. REACTOR WATER LEVEL BEGAN INCREASING. AT 48 INCHES, THE OPERATOR ATTEMPTED TO BRING THE FLOW UNDER CONTROL BY SHIFTING THE "A" RFP INTO MANUAL CONTROL AND BY REDUCING SPAED BY TWO POTENTIOMETER TURNS. THE REACTOR TRIPPED DUE TO HIGH WATER LEVEL AT 52 INCHES. THE CAUSE WAS DETERMINED TO BE A FAULTY "B" RFP NET POSITIVE SUCTION HEAD FUNCTION GENERATOR CARD. THE FUNCTION GENERATOR CARD HAS BEEN REPLACED. THE EVENT WAS NOT SAFETY SIGNIFICANT FOR EXISTING PLANT CONDITIONS OR OTHER PLANT MODES. THE EFFECTS OF A SIMILAR INCREASE IN REACTOR WATER LEVEL HAVE BEEN ANALYZED FOR A MORE SEVERE TRANSIENT IN CHAPTER 15 OF THE FINAL SAFETY ANALYSIS REPORT AND HAVE BEEN FOUND ACCEPTABLE. THIS EVENT IS REPORTABLE UNDER THE PROVISIONS OF 10CFR50.73(A)(2)(IV) DUE TO AN AUTOMATIC ACTUATION OF THE RPS.

[67] CLINTON 1 DOCKET 50-461 LER 87-056
TECH SPEC VIOLATION DUE TO UTILITY LICENSED OPERATOR FAILURE TO RECOGNIZE THE REQUIREMENT TO ENTER AN ACTION STATEMENT.
EVENT DATE: 092387 REPORT DATE: 100887 NSSS: GE TYPE: BWR

(NSIC 206735) ON SEPTEMBER 23, 1987 AT APPROXIMATELY 2030 HOURS, WITH THE PLANT IN MODE 1 (POWER OPERATION), AT APPROXIMATELY 23% REACTOR POWER AND NORMAL OPERATING TEMPERATURE AND PRESSURE, OPERATORS IDENTIFIED THAT TECHNICAL SPECIFICATIONS HAD BEEN VIOLATED. THE TECHNICAL SPECIFICATION ACTION STATEMENT REQUIRES THAT GRAB SAMPLES BE COLLECTED AT LEAST ONCE EVERY FOUR HOURS WHEN NO MAIN CONDENSER OFF-GAS TREATMENT SYSTEM HYDROGEN MONITORS ARE OPERABLE. A FOUR HOUR GRAB SAMPLE WAS REQUIRED TO BE TAKEN BY 1943 HOURS ON SEPTEMBER 23. AT 2030 HOURS, OPERATORS REALIZED THE SAMPLE WAS MISSED. OPERATORS IMMEDIATELY CALLED FOR A GRAB SAMPLE TO BE TAKEN; HOWEVER, ONE OFF-GAS HYDROGEN MONITOR WAS RESTORED TO OPERABILITY BEFORE THE SAMPLE WAS TAKEN. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO UTILITY LICENSED OPERATOR ERROR. THE LINE ASSISTANT SHIFT SUPERVISOR (LASS) FAILED TO RECOGNIZE THE REQUIREMENT TO ENTER THE TECHNICAL SPECIFICATION ACTION STATEMENT. THE LASS HAS BEEN COUNSELLED ON THE FAILURE TO RECOGNIZE THIS REQUIREMENT. THIS EVENT WAS NOT SAFETY SIGNIFICANT SINCE THE LATE SAMPLE PROVIDED ASSURANCE THAT THE CONCENTRATION OF HYDROGEN REMAINED BELOW THE TECHNICAL SPECIFICATION LIMIT OF FOUR PERCENT BY VOLUME. THIS EVENT IS REPORTABLE UNDER THE PROVISIONS OF 10CFR50.73(A)(2)(I)(B) DUE TO AN OPERATION PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATIONS.

[68] CLINTON 1 DOCKET 50-461 LER 87-057
MISSED 8-HOUR VERIFICATION OF OFFSITE POWER BREAKER LINEUPS DURING DIESEL GENERATOR INOPERABILITY DUE TO LINE ASSISTANT SHIFT SUPERVISOR OVERSIGHT.
EVENT DATE: 092987 REPORT DATE: 101987 NSSS: GE TYPE: BWR

(NSIC 206736) ON SEPTEMBER 29, 1987 AT APPROXIMATELY 2100 HOURS WITH THE PLANT IN MODE 1 (POWER OPERATION) AT APPROXIMATELY 96% REACTOR POWER, THE SHIFT SUPERVISOR DETERMINED THAT A TECHNICAL SPECIFICATION ACTION STATEMENT WHICH REQUIRES VERIFICATION OF OFFSITE POWER BREAKER LINEUPS EVERY EIGHT HOURS HAD BEEN VIOLATED. THE TECHNICAL SPECIFICATION ACTION STATEMENT WAS ENTERED AT 0200 HOURS ON SEPTEMBER 29 WHEN THE DIVISION 2 DIESEL GENERATOR WAS REMOVED FROM SERVICE FOR SCHEDULED MAINTENANCE. THE THIRD EXECUTION OF THE TECHNICAL SPECIFICATION ACTION STATEMENT WAS NOT PERFORMED BY 1752 HOURS ON SEPTEMBER 29 AS REQUIRED. THE CAUSE OF THE EVENT HAS BEEN ATTRIBUTED TO UTILITY LICENSED OPERATOR ERROR. THE LINE ASSISTANT SHIFT SUPERVISOR (LASS) BECAME PREOCCUPIED WITH SUPERVISING A POWER ASCENSION TEST PROGRAM TEST AND OVERLOOKED THE ACTION STATEMENT REQUIREMENT. THE ACTION STATEMENT WAS SATISFACTORILY IMPLEMENTED AT 2110 HOURS ON SEPTEMBER 29. THE LASS HAS BEEN COUNSELED ON THE FAILURE TO IMPLEMENT THE ACTION STATEMENT AND ALL APPROPRIATE OPERATIONS SUPERVISORS HAVE BEEN MADE AWARE OF THIS EVENT. THE

EVENT WAS ASSESSED AS NOT SAFETY SIGNIFICANT SINCE THE ACTION STATEMENT WAS SATISFACTORILY IMPLEMENTED AT 2110 HOURS. THIS EVENT IS REPORTABLE UNDER THE PROVISIONS OF 10CFR50.73(A)(2)(I)(B) DUE TO AN OPERATION PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATIONS.

[69] CLINTON 1 DOCKET 50-461 LER 87-058
MISSED CONTAINMENT AIR LOCK DOOR LEAK TEST DUE TO LINE ASSISTANT SHIFT SUPERVISOR MISINTERPRETATION OF THE TECH SPEC.
EVENT DATE: 093087 REPORT DATE: 101987 NSSS: GE TYPE: BWR

(NSIC 206737) ON SEPTEMBER 30, 1987 AT 2245 HOURS, WITH THE PLANT IN MODE 1 (POWER OPERATION) AT APPROXIMATELY 85% REACTOR POWER, THE SHIFT SUPERVISOR IDENTIFIED THAT A PRIMARY CONTAINMENT AIR LOCK DOOR SEAL LEAK TEST HAD NOT BEEN PERFORMED. THE TEST WAS REQUIRED TO BE PERFORMED BY 1413 HOURS ON SEPTEMBER 28. THE OUTER AIR LOCK DOOR BECAME INOPERABLE AT 0700 HOURS ON SEPTEMBER 28, AND THE INNER AIR LOCK DOOR WAS SUBSEQUENTLY LOCKED CLOSED AS REQUIRED BY THE TECHNICAL SPECIFICATION ACTION STATEMENT. THE ASSISTANT SHIFT SUPERVISOR CHECKED THE TECHNICAL SPECIFICATIONS AND BECAME CONVINCED THAT THE ACTION STATEMENT WAS MET BY LOCKING THE INNER AIR LOCK DOOR CLOSED AND THAT THE DOOR SEAL LEAKAGE TEST WAS NOT REQUIRED UNTIL THE OUTER DOOR WAS REPAIRED AND RESTORED TO OPERABILITY. THE CAUSE OF THE EVENT IS ATTRIBUTED TO UTILITY LICENSED OPERATOR ERROR DUE TO MISINTERPRETING THE TECHNICAL SPECIFICATIONS. OPERATIONS HAS BEEN PROVIDED WITH A FORMAL INTERPRETATION OF THE TECHNICAL SPECIFICATIONS FOR LEAK TEST REQUIREMENTS AND ASSOCIATED ACTIONS TO BE TAKEN FOR AN INOPERABLE AIR LOCK DOOR. THE EVENT WAS ASSESSED AS NOT SAFETY SIGNIFICANT SINCE THE SUBSEQUENT LEAK TEST WAS SATISFACTORILY COMPLETED AND CONTAINMENT INTEGRITY VERIFIED. THIS EVENT IS REPORTABLE UNDER THE PROVISIONS OF 10CFR50.73(A)(2)(I)(B) DUE TO AN OPERATION PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATIONS.

[70] CLINTON 1 DOCKET 50-461 LER 87-059
REACTOR CORE ISOLATION COOLING ISOLATION RESULTING FROM CONTROL AND INSTRUMENTATION TECHNICIAN MISWIRING OF TEMPORARY JUMPER CABLE.
EVENT DATE: 100287 REPORT DATE: 102787 NSSS: GE TYPE: BWR

(NSIC 206841) ON OCTOBER 2, 1987 AT APPROXIMATELY 1059 HOURS, WITH THE PLANT IN MODE 1 (POWER OPERATION), AT APPROXIMATELY 99% REACTOR POWER, THE REACTOR CORE ISOLATION COOLING (RCIC) STEAM SUPPLY INBOARD CONTAINMENT ISOLATION VALVE ACTUATED. THE ACTUATION OCCURRED DURING INSTALLATION OF A TEMPORARY MODIFICATION BY A UTILITY CONTROL AND INSTRUMENTATION (C&I) TECHNICIAN. THE TECHNICIAN UNKNOWINGLY INSTALLED A MULTI-WIRE JUMPER CABLE WITH A MISWIRED MULTI-PIN CONNECTOR FROM RESIDUAL HEAT REMOVAL/RCIC CIRCUITRY TO TRANSIENT ANALYSIS RECORDING SYSTEM POINTS TO PERMIT MONITORING OF RCIC STEAM FLOW SIGNALS. WHEN THE JUMPER WAS CONNECTED, THE RCIC STEAM FLOW TRANSMITTER PEGGED HIGH AND THE ISOLATION VALVE ACTUATED. THE JUMPER WAS REMOVED AND THE ISOLATION VALVE WAS RETURNED TO THE NORMAL CONFIGURATION. THE CAUSE OF THE EVENT WAS ATTRIBUTED TO THE FAILURE OF THE C&I TECHNICIAN TO VERIFY THE JUMPER CONNECTOR CONFIGURATION PRIOR TO USE. THE TECHNICIAN HAS BEEN COUNSELLED FOR THE FAILURE TO VERIFY THE CONNECTOR CONFIGURATION. APPROPRIATE C&I PERSONNEL HAVE BEEN INSTRUCTED TO PERFORM VERIFICATION OF JUMPER CONFIGURATION PRIOR TO USE AND PROCEDURES WILL BE REVISED TO REQUIRE THIS VERIFICATION. THE EVENT WAS ASSESSED AS NOT SAFETY SIGNIFICANT.

[71] CLINTON 1 DOCKET 50-461 LER 87-060
MISOPERATION OF NON-CLASS 1E 125 VOLTS DIRECT CURRENT BREAKER BY UTILITY NON-LICENSED OPERATOR RESULTING IN AUTOMATIC REACTOR TRIP.
EVENT DATE: 100287 REPORT DATE: 102587 NSSS: GE TYPE: BWR

(NSIC 206842) ON OCTOBER 2, 1987 AT 1238 HOURS, WITH THE PLANT IN MODE 1 (POWER

OPERATION) AT 90% REACTOR POWER, A NON-CLASS 1E 125 VOLTS DIRECT CURRENT (VDC) BREAKER MISOPERATION BY A UTILITY NON-LICENSED OPERATOR INITIATED A TRANSIENT RESULTING IN AN AUTOMATIC REACTOR TRIP. THE OPERATOR WAS ALIGNING BREAKERS TO SUPPORT MAINTENANCE OF THE NON-CLASS 1E BATTERIES. THE OPERATOR INCORRECTLY OPENED A CROSS TIE BREAKER BETWEEN THE TWO NON-CLASS 1E 125 VDC DISTRIBUTION PANELS, DEENERGIZING ONE OF THE PANELS RESULTING IN A REACTOR HIGH WATER LEVEL TRIP. THE CAUSE OF THE EVENT HAS BEEN ATTRIBUTED TO UTILITY NON-LICENSED OPERATOR FAILURE TO CORRECTLY ALIGN THE CIRCUIT BREAKERS. THE OPERATOR WAS NOT CERTAIN THAT HE HAD FOUND THE CORRECT BREAKER PRIOR TO THE MISOPERATION. THE OPERATOR HAS BEEN COUNSELLED FOR THE FAILURE TO RESOLVE THE BREAKER QUESTION WITH HIS SUPERVISOR. THE NON-CLASS 1E 125 VDC BUS POWER FEEDS ARE BEING REVIEWED AND WILL BE RELABELED AS NECESSARY. AS AN INTERIM MEASURE, A ONE-LINE DIAGRAM HAS BEEN INSTALLED ON THE NON-CLASS 1E DIRECT CURRENT BREAKER PANELS IDENTIFYING THE LOCATION OF BREAKERS. THE EVENT WAS ASSESSED AS NOT SAFETY SIGNIFICANT; THE TRANSIENT IS BOUNDED BY CHAPTER 15 OF THE FINAL SAFETY ANALYSIS REPORT.

[72] CONNECTICUT YANKEE DOCKET 50-213 LER 87-016
RCS SAFETY VALVE AS-FOUND LIFT PRESSURE HIGH DUE TO SETPOINT DRIFT.
EVENT DATE: 100987 REPORT DATE: 110687 NSSS: WE TYPE: PWR

(NSIC 206870) TWO OCCURRENCES OF PRESSURIZER CODE SAFETY VALVES FAILING TO MEET LIFT SETPOINT TOLERANCES WERE NOT REPORTED VIA IOCFR50.73 DUE TO A MISINTERPRETATION OF THE TECHNICAL SPECIFICATION. DURING THE 1986 AND 1987 REFUELING OUTAGES, ONE OF THE THREE PRESSURIZER CODE SAFETY VALVES FAILED TO MEET THE 1% DESIGN SETPOINT TOLERANCE. SINCE THE TECHNICAL SPECIFICATIONS ONLY REFERENCED SECTION VIII OF THE ASME BOILER AND PRESSURE VESSEL CODE, AN ALLOWABLE SETPOINT TOLERANCE OF PLUS OR MINUS 3% (FOR BOILER AND SECTION VIII VESSELS) WAS USED AS THE REPORTABILITY CRITERIA. ON OCTOBER 17, 1987, WITH THE PLANT SHUTDOWN IN MODE 6, IT WAS DETERMINED THAT SECTION I OF THE ASME CODE WAS THE APPLICABLE CODE FOR SETPOINT TOLERANCE. THIS WOULD IMPLY A REPORTABILITY CRITERIA OF PLUS OR MINUS 1% OF SETPOINT. THE CORRECTIVE ACTION IS TO ENSURE THAT THE PROPOSED STANDARD TECHNICAL SPECIFICATIONS INCLUDES EXPLICIT LIMITING CONDITIONS FOR OPERATIONS (LCOS) FOR SETPOINT TOLERANCE ON THESE VALVES. THIS EVENT IS REPORTABLE PER 10CFR50.73 (A)(2)(I) SINCE IT INVOLVED A CONDITION PROHIBITED BY TECHNICAL SPECIFICATIONS.

[73] COOK 1 DOCKET 50-315 LER 86-020 REV 01
UPDATE ON MAIN STEAM SAFETY VALVES OUT OF SPECIFICATION DUE TO SETPOINT DRIFT.
EVENT DATE: 110285 REPORT DATE: 102987 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: COOK 2 (PWR)
VENDOR: DRESSER INDUSTRIAL VALVE & INST DIV

(NSIC 206774) THIS IS A SUPPLEMENTAL REPORT TO A PREVIOUSLY SUBMITTED LER, 315/86-20- 00. ON NOVEMBER 2, 1985, WITH THE REACTOR IN HOT STAND-BY, EIGHT OF TWENTY UNIT 1 MAIN STEAM SAFETY VALVES (MSSV) LIFT SETPOINTS WERE FOUND OUT OF SPECIFICATION DURING SURVEILLANCE TESTING. FOUR OF TWENTY UNIT 2 MSSV LIFT SETPOINTS WERE DISCOVERED OUT OF SPECIFICATION DURING SURVEILLANCE TESTING ON JUNE 23, 1986. ALSO, SIMILAR SURVEILLANCE TEST FAILURES ON OCTOBER 16, 1983 AND JULY 2, 1984 WERE NOT PROPERLY REPORTED. THESE EVENTS WERE DETERMINED TO BE REPORTABLE ON AUGUST 25, 1986 AFTER A REVIEW OF DOCUMENTATION ON MSSV SETPOINT VERIFICATION. A LACK OF PROCEDURAL INSTRUCTIONS CONTRIBUTED TO THE FAILURE TO REPORT THE EVENTS WITHIN 30 DAYS AS REQUIRED BY 10CFR50.73. IN EACH CASE THE MSSVS' LIFT SETPOINTS WERE CORRECTED AND LEFT OPERABLE PRIOR TO COMPLETION OF THE SURVEILLANCE TEST PROCEDURE (STP). THE APPARENT MSSV SETPOINT DRIFT COULD HAVE BEEN ATTRIBUTABLE TO TWO FACTORS, 1) TESTING METHOD, AND; 2) SETPOINT DRIFT DUE TO VALVE DESIGN/APPLICATION. THE INVESTIGATION CONCLUDED THAT THE OLD TESTING METHOD HAD A HIGH PROBABILITY OF CONTRIBUTING TO THE APPARENT MSSV SETPOINT DRIFT. TO PREVENT RECURRENCE, MSSV SETPOINTS HAVE BEEN TESTED WITH AN IMPROVED

TESTING METHOD. ALSO, THE APPLICABLE STP HAS BEEN MODIFIED TO ENSURE THAT FUTURE MSSV FAILURES ARE PROMPTLY REPORTED.

[74] COOK 1 DOCKET 50-315 LER 87-017 REV 01
UPDATE ON FAILURE TO INCORPORATE CHANGES TO PRESSURIZER LEVEL PROTECTION SET VALUES INTO PROCEDURES.
EVENT DATE: 081587 REPORT DATE: 120187 NSSS: WE TYPE: PWR

(NSIC 207186) THIS REVISION TO LER 50-315/87-017-00 IS BEING SUBMITTED TO CORRECT TYPOGRAPHICAL ERRORS AND NOTE METHOD OF DISCOVERY (SEE MARGINAL MARKINGS). ON AUGUST 7, 1987 IT WAS DISCOVERED DURING ROUTINE PERIODIC PROCEDURE REVIEWS, THAT THE VALUES FOR THE TRANSMITTER SPAN FOR THE PRESSURIZER LEVEL WERE INCORRECT IN CALIBRATION PROCEDURES 1 THP 6030 IMP.108, 109 AND 100 "PRESSURIZER LEVEL PROTECTION SET FOR CHANNELS I, II AND III," RESPECTIVELY. THE CORRECT VALUES WERE CONTAINED IN ENGINEERING CONTROL PROCEDURE (ECP) 12-NI-01 APPROVED AND ISSUED ON JANUARY 28, 1977. THESE VALUES HAD NEVER BEEN INCORPORATED INTO UNIT ONE PROCEDURES. IT WAS DETERMINED ON AUGUST 15, 1987 THAT THE UNIT ONE HIGH PRESSURIZER LEVEL REACTOR TRIP SETPOINT OF 91 PERCENT WAS THE EQUIVALENT OF 93.27 PERCENT OF INDICATED SPAN AS DETERMINED BY THE ECP VALUES. THIS IS OUTSIDE THE TECH SPEC ALLOWABLE VALUE OF LESS OR EQUIVALENT TO 93 PERCENT. THE REASON FOR THE ERROR WAS DEFECTIVE PROCEDURES ALTHOUGH THE ACTUAL CAUSE FOR THE CONDITION IS UNKNOWN DUE TO THE TIME FRAME INVOLVED. THE UNIT ONE PROCEDURES WERE CHANGED TO REFLECT THE ECP VALUES AND THE PRESSURIZER LEVEL TRANSMITTERS WERE RECALIBRATED WITH IMP.109 AND 110 BEING COMPLETED ON AUGUST 10, 1987 AND IMP.108 BEING COMPLETED ON AUGUST 20, 1987.

[75] COOK 1 DOCKET 50-315 LER 87-020
LACK OF ISOLATION BETWEEN BALANCE OF PLANT AND ESSENTIAL SAFETY SYSTEM LOADS DUE TO POTENTIAL DESIGN DEFICIENCY.
EVENT DATE: 091787 REPORT DATE: 101987 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: COOK 2 (PWR)
VENDOR: BUSSMANN MFG (DIV OF MCGRAW-EDISON)
HEINEMANN ELECTRIC CO.

(NSIC 206791) ON SEPTEMBER 17, 1987, DURING A REVIEW OF SAFETY SYSTEM FUNCTIONAL INSPECTION ON OUR AUXILIARY FEEDWATER SYSTEM, IT WAS DETERMINED THAT IN THE EVENT OF A FAULT IN CERTAIN BALANCE OF PLANT (BOP) CABLES, WHICH WOULD INVOLVE DISTRIBUTION PANELS FROM BOTH INDEPENDENT TRAINS, A LOSS OF CONTROL POWER ON BOTH INDEPENDENT TRAINS OF RELATED (ESS) PANELS COULD OCCUR. THE ESS LOADS THAT MAY HAVE BEEN AFFECTED ARE CERTAIN CONTAINMENT ISOLATION VALVES, REACTOR HEAD VENT VALVES, POST-ACCIDENT SAMPLING VALVES, AND STEAM GENERATOR STOP VALVE DUMP VALVES. THE CAUSE OF THIS EVENT WAS A POTENTIALLY DEFICIENT DESIGN WHICH COULD HAVE CAUSED INSUFFICIENT BREAKER INTERRUPTING CAPABILITY, IN CONJUNCTION WITH A LACK OF ISOLATION BETWEEN BOP AND ESS LOADS. FOR BOTH UNITS 1 AND 2, DESIGN CHANGES WERE IMPLEMENTED TO ISOLATE THE BOP LOADS FROM THE ESS LOADS. PROCEDURAL CHANGES, WHICH HAD BEEN IMPLEMENTED PRIOR TO THE IDENTIFICATION BUT AFTER THE OCCURRENCE OF THIS PROBLEM, WOULD NOW NECESSITATE THE REVIEW OF QUALIFICATION DOCUMENTATION AND TECHNICAL STUDIES OF THE BREAKER CAPABILITIES.

[76] COOK 1 DOCKET 50-315 LER 87-021
ESF ACTUATION (REACTOR TRIP) DUE TO FEEDWATER FLOW/STEAM FLOW MISMATCH COINCIDENT WITH LOW STEAM GENERATOR LEVEL RESULTING FROM COMPONENT FAILURE.
EVENT DATE: 101387 REPORT DATE: 111287 NSSS: WE TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 206920) ON OCTOBER 13, 1987, AT 0818 HOURS, AN ENGINEERED SAFETY FEATURES ACTUATION (REACTOR TRIP) OCCURRED DUE TO A FEEDWATER FLOW/STEAM FLOW MISMATCH COINCIDENT WITH LOW LEVEL ON STEAM GENERATOR NUMBER 11. AT APPROXIMATELY 0817

HOURS, A LICENSED OPERATOR REMOVED THE EAST MAIN FEEDWATER PUMP ALTERNATING CURRENT AUXILIARY OIL PUMP FROM SERVICE AS DIRECTED BY PROCEDURE. THE SHAFT DRIVEN OIL PUMP FAILED TO MAINTAIN CONTROL OIL PRESSURE AS EXPECTED. THIS RESULTED IN A MAIN FEEDWATER SYSTEM TRANSIENT WHICH CULMINATED IN THE REACTOR TRIP. POST-EVENT EVALUATION IDENTIFIED NO COMPONENT FAILURE, OTHER THAN THE SHAFT- DRIVEN OIL PUMP, WHICH SIGNIFICANTLY AFFECTED THE EVENT. THE EAST MAIN FEEDWATER PUMP SHAFT DRIVEN OIL PUMP WAS FOUND TO BE DAMAGED. IT WAS REPAIRED AND THE FEEDWATER PUMP WAS RETURNED TO SERVICE.

[77]   COOK 2   DOCKET 50-316   LER 87-011
ESF ACTUATION CAUSED BY PERSONNEL ERROR - BLOCKS NOT REINSTATED WHEN RETURNING SOLID STATE PROTECTION SYSTEM TO SERVICE.
EVENT DATE: 100287   REPORT DATE: 110287   NSSS: WE   TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 206921) ON OCTOBER 2, 1987 AT 1657 HOURS WITH THE UNIT IN COLD SHUTDOWN, A TRAIN A SAFETY INJECTION SIGNAL (SI) OCCURRED WHEN INSTRUMENT AND CONTROL (I AND C) TECHNICIANS WERE RETURNING THE TRAIN A SOLID STATE PROTECTION SYSTEM (SSPS) TO SERVICE AFTER REPAIR. PRIOR TO THE EVENT, BOTH TRAINS OF SSPS HAD BEEN DISABLED FOR OUTAGE WORK. AT OUTAGE COMPLETION, SSPS WAS RETURNED TO SERVICE, BUT NOT YET TESTED FOR OPERABILITY. ONE OF THE OPERABILITY TESTS WAS BEING PERFORMED ON SOURCE RANGE NUCLEAR INSTRUMENTATION (NI) WHEN A TRAIN A REACTOR TRIP SIGNAL WAS NOT RECEIVED AS REQUIRED. I AND C THEN STARTED TROUBLESHOOTING AND REPAIR WITHOUT THE BENEFIT OF EVALUATION OF THE CHANGE IN WORK SCOPE. CONSEQUENTLY, THEY DID NOT UTILIZE THE PROCESS DESIGNED TO CONTROL REPAIR EVOLUTIONS. THE SOURCE OF THE NI PROBLEM WAS FOUND TO BE THE UNDERVOLTAGE OUTPUT BOARD OF THE SSPS. WHEN RETURNING THE SSPS TO SERVICE, THE TECHNICIANS FAILED TO PROPERLY SEQUENCE REINSTATING OF SI BLOCKS WITH SSPS SWITCH POSITIONS. IMMEDIATE CORRECTIVE ACTION INVOLVED ASSESSMENT AND TERMINATION OF THE SI SIGNAL (NO INJECTION OCCURRED). TRAIN A SSPS WAS FUNCTIONALLY TESTED AFTER REPAIR AND PROVED OPERABLE. TRAIN B WAS ALSO TESTED WITH NO PROBLEMS IDENTIFIED.

[78]   COOK 2   DOCKET 50-316   LER 87-012
INADVERTENT OPENING OF REACTOR TRIP BREAKERS CAUSED BY PERSONNEL ERROR - WRONG POWER RANGE NUCLEAR INSTRUMENTATION CHANNEL TESTED.
EVENT DATE: 100887   REPORT DATE: 110587   NSSS: WE   TYPE: PWR

(NSIC 206922) ON OCTOBER 8, 1987 AT 2328 HOURS, AN ENGINEERED SAFETY FEATURES ACTUATION (REACTOR TRIP BREAKERS OPENING) OCCURRED DUE TO PERSONNEL ERROR. THE UNIT WAS IN HOT STANDBY AND SURVEILLANCE TESTING WAS BEING PERFORMED PRIOR TO STARTUP. INSTRUMENT AND CONTROL (I AND C) TECHNICIANS PLACED POWER RANGE NUCLEAR INSTRUMENTATION CHANNEL IV (N-44) INTO TRIP AND MISTAKENLY BEGAN TESTING ON CHANNEL III (N-43) WHICH SATISFIED THE TWO OF FOUR LOGIC REQUIRED FOR A REACTOR TRIP SIGNAL. FEEDWATER ISOLATION FOLLOWED DUE TO COINCIDENT LOW TAVG. MAIN TURBINE AND MAIN FEEDPUMP TURBINE TRIPS WERE ALSO RECEIVED DUE TO REACTOR TRIP. NO ABNORMAL REACTOR TRIP SEQUENCE RESPONSES WERE NOTED. IMMEDIATE CORRECTIVE ACTIONS INCLUDED RETURNING N-43 TO NORMAL AND EVALUATING PLANT RESPONSES. N-44 SURVEILLANCE TESTING WAS THEN COMPLETED. THE EVENT WAS DISCUSSED WITH THE INDIVIDUAL INVOLVED WHO WAS COUNSELLED TO BE MORE CAREFUL WHEN WORKING ON SAFETY RELATED EQUIPMENT, ESPECIALLY DURING TIMES SUCH AS UNIT STARTUP WHEN NUMEROUS I AND C ACTIVITIES ARE TAKING PLACE. ADDITIONALLY, A MEETING WAS HELD WITH I AND C PERSONNEL TO DISCUSS THE EVENT AND TO REMIND THEM OF THE RESPONSIBILITY EACH PERSON IN A WORK GROUP HAS, TO FOLLOW EACH STEP OF THE JOB TO ENSURE ITS CORRECT EVOLUTION.

[79] COOK 2 DOCKET 50-316 LER 87-013
REACTOR TRIP CAUSED BY TURBINE TRIP WITH SETPOINT FOR PERMISSIVE P-13 BEING TOO CONSERVATIVE.
EVENT DATE: 101087 REPORT DATE: 110587 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: COOK 1 (PWR)

(NSIC 206923) ON OCTOBER 10, 1987 AT 1213 HOURS THE UNIT TWO REACTOR TRIPPED FROM A TURBINE TRIP. THE TURBINE WAS BEING ROLLED IN PREPARATION FOR THE OVERSPEED TESTING AND POWER WAS BEING MAINTAINED BELOW THE PERMISSIVE P-13 INTERLOCK SETPOINT IN ORDER TO AVOID A REACTOR TRIP WHEN THE TURBINE WAS TRIPPED ON OVERSPEED. AT APPROXIMATELY 8 PERCENT POWER THE TURBINE TRIPPED AND A REACTOR TRIP OCCURRED BECAUSE THE P-13 BISTABLE SETPOINT WAS TOO CONSERVATIVE, CAUSED BY A SETPOINT CALCULATION ERROR. THE TECHNICAL SPECIFICATION REQUIREMENT IS FOR P-13 TO BE ACTIVE AT GREATER THAN OR EQUAL TO 51 PSIG TURBINE IMPULSE PRESSURE. THE BISTABLE SETPOINT WAS TO BE AT 45 PSIG FOR CONSERVATISM, HOWEVER, AN ERROR WAS MADE IN CALCULATING THE SETPOINT AND AS A RESULT, THE SETPOINT WAS ACTUALLY EQUIVALENT TO 31.132 PSIG. NO ABNORMAL REACTOR TRIP SEQUENCE RESPONSES WERE NOTED. THE NRC WAS NOTIFIED VIA THE ENS AT 1300 HOURS ON OCTOBER 10, 1987. UPON DISCOVERY OF THE ERROR, THE SETPOINT WAS RECALCULATED AND THE BISTABLE WAS CALIBRATED TO THE CORRECT VALUE.

[80] COOPER DOCKET 50-298 LER 87-023
SETPOINT DRIFT OF BARKSDALE BOURDON TUBE PRESSURE SWITCHES, MODEL B2T CAUSED BY SEASONAL ENVIRONMENTAL VARIATIONS.
EVENT DATE: 041487 REPORT DATE: 102687 NSSS: GE TYPE: BWR
VENDOR: BALKSDALE VALVE COMPANY

(NSIC 206789) DUE TO SETPOINT DISCREPANCIES BETWEEN AS-LEFT AND SUBSEQUENT AS-FOUND CONDITIONS DETERMINED DURING SURVEILLANCE TESTING FOR BARKSDALE MODEL B2T PRESSURE SWITCHES, AN EVALUATION WAS PERFORMED IN AN EFFORT TO DETERMINE THE CAUSE. THE EVALUATION HAS REVEALED THAT THE SETPOINT DRIFT EXPERIENCED BY THESE SWITCHES MAY BE RELATED TO SEASONAL ENVIRONMENTAL VARIATIONS. THE SETPOINTS APPEAR TO DRIFT UPWARD IN THE SUMMER AND DOWNWARD IN THE WINTER, TYPICALLY PEAKING HIGH IN JULY AND LOW IN DECEMBER. THE SEASONAL DRIFT, COMBINED WITH A RIGOROUS CALIBRATION METHOD, COULD RESULT IN AN EFFECTIVE SETPOINT VARIANCE EXCEEDING THE MANUFACTURER'S SPECIFICATION (PLUS OR EQUAL TO 1 PERCENT OF SPAN). THE SEASONAL DRIFT DOES NOT REPRESENT A SAFETY PROBLEM, PROVIDED THAT ADEQUATE MARGIN EXISTS BETWEEN THE ACTUAL SETPOINT AND THE TECHNICAL SPECIFICATION SETPOINT LIMIT. BASED UPON THE REVIEW CONDUCTED, NO INSTANCE WAS DISCOVERED WHERE THE AS-FOUND TRIP POINT FOR MORE THAN ONE SWITCH PER CHANNEL WAS DETERMINED TO VIOLATE THE REQUIRED SETPOINT AS PRESCRIBED IN TECHNICAL SPECIFICATIONS. THIS REPORT IS BEING SUBMITTED AS AN ITEM OF GENERIC INTEREST TO AID OTHER UTILITIES IN THE EVALUATION OF ANY ONGOING BARKSDALE SWITCH SETPOINT PROBLEMS.

[81] COOPER DOCKET 50-298 LER 87-022
UNPLANNED CLOSURE OF REACTOR WATER CLEANUP SYSTEM ISOLATION VALVE DUE TO PERSONNEL ERROR DURING SURVEILLANCE TESTING.
EVENT DATE: 051287 REPORT DATE: 101587 NSSS: GE TYPE: BWR

(NSIC 206757) DURING SURVEILLANCE TESTING OF THE REACTOR WATER CLEANUP (RWCU) SYSTEM ON MAY 12, 1987, INLET ISOLATION VALVE RWCU-MO-15 UNEXPECTEDLY CLOSED AT 1:15 PM, ISOLATING THE RWCU SYSTEM. THE UNEXPECTED CLOSURE OCCURRED SUBSEQUENT TO SATISFACTORY PERFORMANCE OF SURVEILLANCE TESTING OF ONE OF THE TWO RWCU SYSTEM HIGH FLOW SENSING INSTRUMENTS WHILE THE VALVE/BREAKER LINEUP WAS BEING RETURNED TO NORMAL. AT THE TIME OF THE EVENT, THE PLANT WAS IN OPERATION AT APPROXIMATELY 85 PERCENT POWER. PLANT OPERATION WAS UNAFFECTED. AT 1:26 PM, THE RWCU SYSTEM WAS PLACED BACK IN SERVICE. AN EVALUATION OF THE EVENT WAS PERFORMED AND IT WAS DETERMINED TO HAVE BEEN CAUSED BY A FAILURE TO ENSURE THAT THE ISOLATION CONDITION FOR RWCU-MO-15 WAS RESET BEFORE RE- ENERGIZING THE VALVE OPERATOR. THE

IMPORTANCE OF LOCALLY VERIFYING THAT THE RELAY IN THE VALVE CLOSURE LOGIC, WHICH IS INSTALLED IN A RELAY CABINET IN THE AUXILIARY RELAY ROOM, IS ENERGIZED AND SEALS IN PRIOR TO RE-ENERGIZING THE VALVE OPERATOR WAS DISCUSSED WITH THE PERSONNEL INVOLVED. A PROCEDURE WAS INITIATED TO ENSURE IMPROVED AWARENESS OF THE POTENTIAL CONSEQUENCES IF THE RELAY IS NOT ENERGIZED AND SEALED IN. ADDITIONALLY, AN ENGINEERING INVESTIGATION WILL BE CONDUCTED TO DETERMINE IF IMPROVED STATUS INFORMATION ASSOCIATED WITH THIS ISOLATION CIRCUITRY CAN BE PROVIDED.

[82] COOPER DOCKET 50-298 LER 87-021
UNPLANNED AUTOMATIC ACTUATIONS OF THE DIESEL GENERATORS STARTING LOGIC IN 1984 AND 1985 WHICH WERE NOT PREVIOUSLY REPORTED.
EVENT DATE: 090487 REPORT DATE: 100287 NSSS: GE TYPE: BWR

(NSIC 206674) IN RESPONSE TO AN NRC INFORMATION REQUEST REGARDING THE FREQUENCY OF LICENSEE EVENT REPORTS (LERS) ASSOCIATED WITH DIESEL GENERATOR AUTOMATIC ACTUATIONS, A DETAILED REVIEW OF THE CONTROL ROOM LOG FROM 1/1/84, THROUGH 6/10/86, WAS CONDUCTED. IT WAS DETERMINED THAT SEVEN (7) UNPLANNED AUTOMATIC DIESEL GENERATOR ACTUATIONS HAD OCCURRED. FIVE (5) EVENTS, OCCURRING IN 1984 AND 1985, HAD NOT BEEN REPORTED. TWO (2) OF THE EVENTS WERE CAUSED BY LIGHTNING INDUCED VOLTAGE FLUCTUATIONS ON THE 69 KV TRANSMISSION LINE, THE POWER SOURCE FOR THE EMERGENCY TRANSFORMER. TWO (2) OTHER EVENTS OCCURRED WHEN 125V DC DISTRIBUTION PANEL POWER WAS RESTORED CAUSING ACTUATION OF 4160V BREAKER RELAYING LOGIC. ONE (1) EVENT OCCURRED DURING THE COURSE OF ACCEPTANCE TESTING FOLLOWING IMPLEMENTATION OF A 4160V BREAKER RELAY LOGIC DESIGN CHANGE. ONE (1) OF THE FIVE (5) EVENTS (A LIGHTNING STRIKE) OCCURRED WHILE THE PLANT WAS IN OPERATION. THE OTHER FOUR (4) OCCURRED WHILE THE PLANT WAS IN A COLD SHUTDOWN CONDITION. THE CAUSE OF THESE EVENTS, OTHER THAN THE TWO WHICH HAVE BEEN ATTRIBUTED TO LIGHTNING, HAS BEEN DETERMINED TO BE AS A RESULT OF PROCEDURAL INADEQUACIES. THE REPORTING CRITERIA WAS NOT BELIEVED TO BE APPLICABLE SINCE THE DIESEL GENERATORS STARTED AND, THUS, DID NOT COMPLETE THEIR ESF FUNCTION.

[83] CRYSTAL RIVER 3 DOCKET 50-302 LER 87-007 REV 01
UPDATE ON DESIGN OVERSIGHT RESULTS IN POTENTIAL FOR EXCEEDING EMERGENCY DIESEL SEQUENTIAL LOADING LIMITS.
EVENT DATE: 042087 REPORT DATE: 102987 NSSS: BW TYPE: PWR

(NSIC 206718) THE ENGINEERED SAFEGUARDS (ES) 480 VOLT BUSES ARE NOT DIRECTLY ADDRESSED IN THE PRESENT DESIGN OF THE DEGRADED VOLTAGE PROTECTION RELAYING. CONSEQUENTLY, IF A DEGRADED VOLTAGE CONDITION WAS COINCIDENT WITH AN ES ACTUATION, THEN THE DECAY HEAT CLOSED CYCLE COOLING PUMP WILL BE IMMEDIATELY LOADED ON TO THE EMERGENCY DIESEL GENERATOR. THIS IS CONTRARY TO THE NORMAL SEQUENCE IN WHICH ITS LOADING IS DELAYED FOR 15 SECONDS (BLOCK 4). THE NET EFFECT IS THAT IT CAUSES THE FIRST BLOCK OF STARTING LOADS (BLOCK 1) TO BE SLIGHTLY MORE THAN DESIGNED; THUS DUE TO THE ADDITIONAL BLOCK ONE LOAD THE RESULTANT VOLTAGE DIP IS MORE THAN THE PREVIOUSLY STATED VALUES. THE CAUSE WAS A DESIGN OVERSIGHT WHICH IS BEING CORRECTED DURING REFUEL VI OUTAGE.

[84] CRYSTAL RIVER 3 DOCKET 50-302 LER 87-018
COGNITIVE ERROR LEADS ISI ENGINEER TO RECOMMEND USE OF A CURRENT ASME SECTION XI CODE EDITION FOR ISI HYDROS RESULTING IN TECH SPEC VIOLATION.
EVENT DATE: 090287 REPORT DATE: 101887 NSSS: BW TYPE: PWR

(NSIC 206719) ON SEPTEMBER 2, 1987, AT 1700, CRYSTAL RIVER UNIT 3 WAS OPERATING AT 64% RATED POWER AND WAS GENERATING 533 MWE WITH REACTOR POWER LIMITED BY HAVING ONE REACTOR COOLANT PUMP INOPERABLE. DURING A REVIEW OF THE ISI PROGRAM IT WAS DETERMINED THAT CERTAIN HYDROSTATIC TESTS (HYDROS) PERFORMED IN 1980, AND 1983 WERE NOT PERFORMED IN ACCORDANCE WITH THE TECH SPEC REQUIRED 1974 ASME

SECTION XI CODE. THIS RESULTED FROM A MISUNDERSTANDING ON THE PART OF THE LEAD ISI ENGINEER REGARDING THE USE OF CURRENT EDITIONS OF THE ASME SECTION XI, VERSES THE OLDER 1974 EDITION OF ASME SECTION XI REQUIRED BY TECH SPECS. THE DIFFERENCE BETWEEN THESE EDITIONS INVOLVED TEMPERATURE AND HOLD TIME REQUIREMENTS. SYSTEM ISI HYDROS PERFORMED AT CR-3 HAVE BEEN REVIEWED AND IT HAS BEEN DETERMINED THAT REPERFORMANCE OF THESE HYDROS IS NOT NECESSARY. THE CODE EDITION OF ASME SECTION XI WHICH IS AUTHORIZED FOR CURRENT USE AT CR-3 HAS THE SAME TEMPERATURE AND HOLD TIME REQUIREMENTS AS THE CODE USED IN 1980 AND 1983.

[85] CRYSTAL RIVER 3 DOCKET 50-302 LER 87-020
PERSONNEL ERROR DURING ORIGINAL PLANT DESIGN SPECIFICATION DEVELOPMENT LEADS TO ULTIMATE HEAT SINK TEMPERATURE EXCEEDING LIMIT AND TO OPERATION OUTSIDE DESIGN BASIS.
EVENT DATE: 090387 REPORT DATE: 100287 NSSS: BW TYPE: PWR

(NSIC 206769) ON SEPTEMBER 3, 1987, CRYSTAL RIVER UNIT 3 WAS OPERATING AT APPROXIMATELY 63% RATED THERMAL POWER. DURING AN NRC AUDIT OF PLANT COOLING WATER SYSTEMS IT WAS DETERMINED THAT THE ULTIMATE HEAT SINK (HS) TEMPERATURE WAS IN EXCESS OF THE MAXIMUM VALUE ASSUMED IN THE PLANT DESIGN BASIS. ALSO, THE PLANT TECH SPEC LIMIT FOR RHS TEMPERATURE WAS FOUND TO BE IN ERROR AND IN EXCESS OF THE DESIGN BASIS. THIS EVENT WAS THE RESULT OF AN INADEQUATE PLANT DESIGN SPECIFICATION. THE MAXIMUM SEAWATER TEMPERATURE SPECIFIED FOR PLANT DESIGN WAS 85 DEGREES F, WHILE ACTUAL TEMPERATURES EXCEED THIS VALUE DURING THE SUMMER MONTHS. PRELIMINARY ANALYSIS PERFORMED FOLLOWING THE DISCOVERY OF THIS CONDITION INDICATE THAT THE NUCLEAR SERVICES CLOSED CYCLE COOLING SYSTEM 105 DEGREES F TEMPERATURE LIMIT CAN BE MET WITH SEAWATER TEMPERATURES AS HIGH AS 92.3 DEGREES F. THE TECH SPEC ERROR APPEARS TO HAVE BEEN CAUSED BY INADVERTENTLY SELECTING A TEMPERATURE LIMIT FROM A CLOSED CYCLE COOLING LOOP RATHER THAN THE UHS DESIGN SPECIFICATION. A NEW UHS TEMPERATURE LIMIT, BASED ON THE RESULTS OF ANALYSES CURRENTLY IN PROGESS, WILL BE ESTABLISHED. THIS LIMIT WILL BE REFLECTED IN REVISIONS TO APPROPRIATE PLANT PROCEDURES AND LICENSING BASIS DOCUMENTS.

[86] CRYSTAL RIVER 3 DOCKET 50-302 LER 87-019
EXCEEDING EMERGENCY DIESEL GENERATOR DESIGN RATING DURING SURVEILLANCE TESTING.
EVENT DATE: 091487 REPORT DATE: 100887 NSSS: BW TYPE: PWR
VENDOR: COLT INDUSTRIES, INC.

(NSIC 206768) ON SEPTEMBER 14, 1987, COLT INDUSTRIES CONFIRMED IN WRITING THAT THE 30 MINUTE DESIGN RATING HAD BEEN EXCEEDED DURING PREVIOUS SURVEILLANCE TESTING. WHEN OPERATED AT BETWEEN 3001 AND 3300 KW THE EMERGENCY DIESEL GENERATORS ARE LIMITED TO A MAXIMUM CUMULATIVE OPERATING TIME OF 30 MINUTES, (30 MINUTE DESIGN RATING). TECH SPECS 4.8.1.1.2(D)(4) REQUIRES THE OPERABILITY OF THE EMERGENCY DIESEL GENERATORS TO BE DEMONSTRATED BY TEST OPERATING THEM AT GREATER THAN OR EQUAL TO 3000 KW FOR GREATER THAN OR EQUAL TO SIXTY MINUTES. THIS LED TO TESTING ABOVE DESIGN LIMITS. THE APPARENT CAUSE WAS MISINTERPRETATION OF THE EMERGENCY DIESEL RATINGS EARLY IN PLANT LIFE. PRIOR TO THE END OF THE CURRENT REFUELING OUTAGE FLORIDA POWER CORPORATION (FPC) WILL COMPLETE A SPECIAL INSPECTION ON BOTH EMERGENCY DIESEL GENERATORS. APPLICABLE CHANGES TO THE TECH SPEC ARE BEING CONSIDERED. ENHANCEMENTS FOR UPGRADING THE EMERGENCY DIESEL RATING ARE BEING CONSIDERED AS A LONG TERM RESOLUTION.

[87] CRYSTAL RIVER 3 DOCKET 50-302 LER 87-023
DEFECTIVE PROCEDURES LEAD TO TECH SPEC VIOLATION OF NOT PROVIDING AUDIBLE NEUTRON FLUX INDICATIONS PRIOR TO ENTERING MODE 6.
EVENT DATE: 093087 REPORT DATE: 110487 NSSS: BW TYPE: PWR

(NSIC 207145) ON SEPTEMBER 26, 1987, CRYSTAL RIVER UNIT 3 WAS SHUT DOWN (MODE 5 COLD SHUTDOWN) PREPARING FOR REFUELING. THE NEUTRON FLUX MONITOR AUDIBLE

INDICATION IN THE CONTROL ROOM AND CONTAINMENT WAS TESTED SATISFACTORILY AND DE-ENERGIZED, AWAITING CORE ALTERATIONS. OPERATIONAL MODE 6 (REFUELING) WAS ENTERED AT 2355 ON SEPTEMBER 30, 1987, WHEN DETENSIONING OF THE REACTOR VESSEL CLOSURE HEAD STUDS BEGAN. ENTRY INTO MODE 6 WAS ACCOMPLISHED BEFORE NEUTRON FLUX MONITOR AUDIBLE INDICATION IN THE CONTROL ROOM AND CONTAINMENT WAS RE-ENERGIZED. ENTRY INTO MODE 6 WITHOUT AUDIBLE NEUTRON FLUX INDICATION IN OPERATION IS PROHIBITED BY THE CRYSTAL RIVER UNIT 3 TECHNICAL SPECIFICATIONS. THIS EVENT WAS THE RESULT OF DEFECTIVE PROCEDURES. THE REFUELING OPERATIONS DAILY DATA REQUIREMENTS SURVEILLANCE PROCEDURE DID NOT SPECIFY THAT AUDIBLE INDICATION BE OPERATING PRIOR TO ENTERING MODE 6. THE UNIT SHUTDOWN SURVEILLANCE PLAN, WHICH IS USED TO SCHEDULE SURVEILLANCES DURING PLANT SHUTDOWN OPERATIONS, IS ALSO DEFECTIVE IN THIS REGARD. THE AUDIBLE NEUTRON FLUX INDICATION WAS RE-ENERGIZED AT 0730 ON OCTOBER 1, 1987. THE REFUELING OPERATIONS DAILY DATA REQUIREMENTS SURVEILLANCE PROCEDURE WAS REVISED ON OCTOBER 7, 1987. IT NOW REQUIRES THAT AUDIBLE INDICATION BE OPERATING PRIOR TO ENTERING MODE 6. THE UNIT SHUTDOWN SURVEILLANCE PLAN WILL BE REVISED SIMILARLY.

[88] CRYSTAL RIVER 3 DOCKET 50-302 LER 87-024
UNPLANNED EXPOSURE RESULTS FROM UNAUTHORIZED SHIELDING REMOVAL IN EFFORT TO BETTER DETERMINE REACTOR CAVITY SEAL PLATE WATER LEAKAGE.
EVENT DATE: 100887 REPORT DATE: 110987 NSSS: BW TYPE: PWR

(NSIC 207146) ON OCTOBER 8, 1987, CRYSTAL RIVER UNIT 3 WAS SHUT DOWN IN MODE 6 FOR REFUELING. AT APPROXIMATELY 2300 THE OPERATION TO FILL THE FUEL TRANSFER CANAL WAS STARTED. AN UNLICENSED NUCLEAR AUXILIARY OPERATOR WAS AT THE BASEMENT LEVEL OF THE REACTOR BUILDING FOR THE PURPOSE OF CHECKING WATER LEAKAGE PAST THE REACTOR CAVITY SEAL PLATE. THE OPERATOR REMOVED SOME LEAD BRICKS FROM A PERMANENT SHIELDING BARRIER IN THE ACCESS OPENING TO THE REACTOR CAVITY TO LOOK DIRECTLY INTO THE CAVITY. HE WAS EXPOSED TO A RADITION FIELD OF 55 REM/HR DURING THE TIME HE SPENT DIRECTLY IN FRONT OF THE OPENING. AN UPPER LIMIT TO HIS TOTAL DOSE WAS CONSERVATIVELY ESTABLISHED AS 1.8 REM BY A HEALTH PHYSICS SUPERVISOR. THIS EXPOSURE WAS AN UNPLANNED, UNCONTROLLED EXPOSURE THAT VIOLATED THE PLANT ADMINISTRATIVE LIMITS. THE ROOT CAUSE WAS THE LACK OF A GOOD METHOD FOR EARLY DETECTION OF WATER LEAKAGE PAST THE REACTOR CAVITY SEAL PLATE. CORRECTIVE ACTIONS INCLUDE PLANT MODIFICATIONS, TRAINING REVISION, SUPERVISORY EMPHASIS, AND PROCEDURE REVISION.

[89] DAVIS-BESSE 1 DOCKET 50-346 LER 87-011
REACTOR TRIP FROM FULL POWER CAUSED BY IMPROPER CONTROL ROD MOVEMENT.
EVENT DATE: 090687 REPORT DATE: 100687 NSSS: BW TYPE: PWR
VENDOR: BAILEY METER COMPANY
DRESSER INDUSTRIAL VALVE & INST DIV
FISHER CONTROLS CO.
VELAN VALVE CORP.
WESTINGHOUSE ELECTRIC CORP.

(NSIC 206641) WITH THE UNIT AT FULL POWER, A FEEDWATER FLOW TRANSMITTER FAILED. THE INTEGRATED CONTROL SYSTEM (ICS) THEN INCREASED FEEDWATER FLOW TO STEAM GENERATOR NO. 2. DUE TO AN ABNORMAL CONTROL ROD GROUP SELECTION, THE CONTROL ROOM OPERATOR INADVERTENTLY DROVE AXIAL POWER SHAPING RODS INSTEAD OF CONTROL RODS CAUSING AN UNDESIRED POWER INCREASE. THE RECTOR PROTECTION SYSTEM (RPS) TRIPPED THE REACTOR ON HIGH FLUX. FOLLOWING THE TRIP SEVERAL OTHER ANOMALIES OCCURRED. BREAKER HX01A FAILED TO AUTOMATICALLY TRANSFER DUE TO A FAILED POSITION SWITCH AND INADEQUATE ADJUSTMENT OF THE FLOOR TRIPPER, ONE MAIN STEAM SAFETY VALVE FAILED TO FULLY RESEAT DUE TO A FAILURE OF A DISC COLLAR; AND A SERVICE WATER PUMP FAILED TO AUTOMATICALLY START WHEN THE EMERGENCY DIESEL GENERATOR STARTED DUE TO A MISSING WIRE IN A CONTROL CIRCUIT. AS CORRECTIVE ACTION, BREAKER HX01A WAS ADJUSTED AND THE POSITION SWITCH REPLACED, THE MAIN STEAM SAFETY VALVE WAS REMOVED AND BLANKED; AND THE MISSING WIRE WAS INSTALLED

INTO THE SERVICE WATER PUMP CONTROL CIRCUIT. AFTER THE TRIP, A TURBINE BYPASS VALVE FAILED OPEN DUE TO AN IMPROPER VALVE TRAVEL STOP ADJUSTMENT CAUSING A STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM (SFRCS) ACTUATION. AS CORRECTIVE ACTION, THE TRAVEL STOP ADJUSTMENTS WERE CORRECTED.

[90] DAVIS-BESSE 1 DOCKET 50-346 LER 87-012
INADVERTENT, INCONSEQUENTIAL TRIP OF CONTROL ROD DRIVE BREAKERS IN HOT SHUTDOWN.
EVENT DATE: 091087 REPORT DATE: 101287 NSSS: BW TYPE: PWR

(NSIC 206642) ON SEPTEMBER 10, 1987 AT 1650 HOURS, DURING THE PERFORMANCE OF THE REACTOR PROTECTION SYSTEM (RPS) FUNCTIONAL TEST IN SHUTDOWN BYPASS, ST5030.16, THE UNIT EXPERIENCED AN INADVERTENT ANTICIPATORY REACTOR TRIP SYSTEM (ARTS) TRIP OF THE CONTROL ROD DRIVE (CRD) BREAKERS. THE UNIT HAD BEEN IN MODE 4 SINCE SEPTEMBER 7, 1987. SINCE ALL CONTROL AND SAFETY RODS HAD PREVIOUSLY BEEN INSERTED, THERE WAS NO EFFECT ON PLANT SYSTEMS WHEN THE CRD BREAKERS OPENED. THE CAUSE WAS AN INADEQUATE PROCEDURE WHICH DID NOT ANTICIPATE THE PRE-TEST CONDITIONS ENCOUNTERED. THE PROCEDURE THEREFORE DID NOT INCLUDE THE ADDITIONAL STEPS TO BE TAKEN TO PREVENT THE ARTS TRIP WITH A STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM (SFRCS) TRIP (1/5) EXISTING IN THE ARTS CHANNEL. ST5030.16 IS BEING CHANGED UNDER PROCEDURE CHANGE NUMBER C-680 TO PERMIT TESTING WITH SIMILAR PRE-TEST CONDITIONS WITHOUT CAUSING THE ARTS TRIP. ON SEPTEMBER 11, 1987 AT 1330 HOURS, THE 1/5 TRIP CONDITION WAS CLEARED AND THE TEST SUCCESSFULLY PERFORMED (ST5030.16).

[91] DAVIS-BESSE 1 DOCKET 50-346 LER 87-013
LOSS OF Y2 ESSENTIAL 120 VAC BUS DUE TO PERSONNEL ERROR DURING TROUBLESHOOTING.
EVENT DATE: 101687 REPORT DATE: 111687 NSSS: BW TYPE: PWR

(NSIC 207125) ON OCTOBER 16, 1987, AT 1236 HOURS THE UNIT LOST ESSENTIAL 120 VAC BUS Y2 WHICH DE-ENERGIZED CHANNEL 2 (CH 2) OF THE REACTOR PROTECTION SYSTEM (RPS), THE STEAM AND FEEDWATER RUPTURE CONTROL SYSTEM (SFRCS), THE ANTICIPATORY REACTOR TRIP SYSTEM (ARTS), AS WELL AS THE SAFETY FEATURES ACTUATION SYSTEM (SFAS), AND PLACED THE UNIT IN THE ACTION STATEMENTS OF THE APPLICABLE TECH SPEC. AT 1320 HOURS, THE SHIFT SUPERVISOR REALIZED THAT BOTH TRAINS OF CREVS WERE INOPERABLE WHILE RE-ENERGIZING THE BUS. SINCE THIS WAS OUTSIDE THE ACTION STATEMENT OF CREVS TECH SPEC 3.7.6.1, TECH SPEC 3.0.3 WAS APPLICABLE. AT 1341 HOURS, Y2 WAS RE-ENERGIZED. AT 1416 HOURS, CREVS TRAIN 2 WAS DECLARED OPERABLE, AND TECH SPEC 3.0.3 WAS NO LONGER APPLICABLE. BY 2013 HOURS THE LAST OF THE SYSTEMS HAD BEEN RE-ENERGIZED, TESTED, AND DECLARED OPERABLE. THE EVENT WAS INITIATED BY PERSONNEL ERROR. AN INAPPROPRIATE PROBE TERMINATOR WAS SELECTED FOR THE OSCILLOSCOPE USED TO TROUBLESHOOT FSAR CH. 2. THE CENTER CONDUCTOR OF THE PROBE TERMINATOR SHORTED TO GROUND, RESULTING IN A LOAD FUSE OPENING, ISOLATING POWER TO SFAS CH. 2. DUE TO A PREVIOUSLY IDENTIFIED CHARACTERISTIC WITH THE PROTECTION OF THE INVERTER WHICH POWERS THE Y2 BUS, THE INVERTER INPUT FUSE OPENED AND DE-ENERGIZED THE ENTIRE BUS. THE TRAINING PROGRAM FOR I&C TECHNICIANS WILL BE CHANGED TO ADD EMPHASIS TO SELECTION OF TEST EQUIPMENT.

[92] DIABLO CANYON 1 DOCKET 50-275 LER 87-015
FAILURE TO MEET REQUIREMENTS FOR HIGH RADIATION AREA WHEN AREA WAS LEFT UNLOCKED.
EVENT DATE: 092487 REPORT DATE: 102687 NSSS: WE TYPE: PWR

(NSIC 206783) AT 1130 PDT, SEPTEMBER 24, 1987, WITH THE UNIT IN MODE 1 (POWER OPERATION) AT 100 PERCENT POWER, THE REQUIREMENTS FOR TECHNICAL SPECIFICATION (TS) 6.12.2 WERE NOT MET. TS 6.12.2 WAS NOT MET WHEN A VERY HIGH RADIATION AREA (VHRA) DOOR WAS LEFT UNLOCKED AND THE VHRA WAS NOT POSTED WITH A FLASHING WARNING LIGHT. THE DOOR WAS RELOCKED ON SEPTEMBER 28, 1987, AT 1130 PDT. THIS EVENT RESULTED FROM A COMBINATION OF TWO EVENTS: A CHEMISTRY AND RADIATION PROTECTION (C&RP) FOREMAN FAILING TO INVOKE VHRA KEY ISSUANCE PROCEDURE RCP G-130, "CONTROL

OF ACCESS FOR RADIATION PROTECTION PURPOSES," AND A C&RP TECHNICIAN FORGETTING THE REQUIREMENT TO POST WARNING LIGHTS BEFORE LEAVING THE AREA UNATTENDED. THERE WAS NO SIGNIFICANT RADIATION EXPOSURE ASSOCIATED WITH THIS EVENT AS THE VHRAS WITHIN THE FACILITY WERE POSTED (WITH THE EXCEPTION OF THE FLASHING WARNING LIGHTS) AND ARE NORMALLY INACCESSIBLE, DUE TO THE CONFIGURATION OF THE FACILITY. THEREFORE, NO ADVERSE SAFETY CONSEQUENCES OR IMPLICATIONS RESULTED FROM THIS EVENT. CORRECTIVE ACTIONS TO PREVENT RECURRENCE INCLUDE: REVISING PROCEDURES ADDRESSING THE QUALIFICATIONS REQUIRED FOR CHECKING OUT A VHRA KEY; REVIEWING THE EVENT WITH ALL C&RP TECHNICIANS; ISSUING A MEMO TO ALL C&RP FOREMEN, AND USING CAUTION WHEN ISSUING VHRA KEYS.

[93] DIABLO CANYON 1 DOCKET 50-275 LER 87-016
ENTRY INTO TECHNICAL SPECIFICATION 3.0.3 DUE TO FOUR SHUTDOWN BANK CONTROL RODS NOT BEING FULLY WITHDRAWN DUE TO A FUSE FAILURE FROM UNKNOWN CAUSE.
EVENT DATE: 100687 REPORT DATE: 110587 NSSS: WE TYPE: PWR

(NSIC 206911) ON 10/6/87, AT 0405 PDT, WITH THE UNIT IN MODE 1 (POWER OPERATION AT 100% POWER A PLANT SHUTDOWN WAS INITIATED IN ACCORDANCE WITH TECHNICAL SPECIFICATION (TS) 3.0.3 AND AN UNUSUAL EVENT WAS DECLARED WHEN FOUR SHUTDOWN CONTROL RODS (AA)(ROD), INSERTED ONE STEP, WERE NOT FULLY WITHDRAWN WITHIN ONE HOUR AS REQUIRED BY TS 3.1.3.5. WHEN THE SHUTDOWN BANK A GROUP 2 CONTROL RODS WERE INSERTED ONE STEP THE INDICATOR FUSES FOR BOTH SHUTDOWN AND CONTROL BANK A GROUP 2 CONTROL RODS FAILED. NOTIFICATIONS REQUIRED BY 10 CFR 50.72 WERE COMPLETED BY 0434 PDT. ON 10/6, AT 0530 PDT AFTER REPLACING THE BLOWN FUSES, THE FOUR SHUTDOWN RODS WERE FULLY WITHDRAWN, THEREBY EXITING TS 3.0.3. THE UNUSUAL EVENT WAS TERMINATED AT 0543 PDT. SINCE THIS EVENT HAD NO EFFECT ON THE ABILITY OF THE RODS TO DROP INTO THE CORE IN RESPONSE TO A REACTOR TRIP SIGNAL AND SINCE CONTROL BANK D IS OFTEN INSERTED MUCH FURTHER INTO THE CORE WITHOUT CAUSING A SIGNIFICANT FLUX PERTURBATION, THIS EVENT, AT 100 PERCENT POWER OR AT ANY OTHER POWER LEVEL, WOULD NOT AFFECT THE HEALTH AND SAFETY OF THE PUBLIC. THE CAUSE OF THE FUSE FAILURE IS UNKNOWN. THE FAILED FUSES WERE SENT OFFSITE FOR ANALYSIS. PG&E WILL SUBMIT A REVISION TO THIS LER WHEN THE ROOT CAUSE AND APPROPRIATE ACTIONS HAVE BEEN DETERMINED. UNTIL THEN, THE CONTROL ROD DRIVE FUSES WILL BE INSPECTED MONTHLY.

[94] DIABLO CANYON 2 DOCKET 50-323 LER 87-005 REV 01
UPDATE ON INTERRUPTION OF RHR FLOW DURING RCS MIDLOOP OPERATION.
EVENT DATE: 041087 REPORT DATE: 102987 NSSS: WE TYPE: PWR
VENDOR: GRINNELL CORP.

(NSIC 206792) ON APRIL 10, 1987, AT 2123 PDT, WITH THE UNIT IN MODE 5 (COLD SHUTDOWN) DURING A REFUELING OUTAGE, RESIDUAL HEAT REMOVAL (RHR) FLOW WAS INTERRUPTED WHEN BOTH RHR TRAINS BECAME INOPERABLE DUE TO AIRBOUND RHR PUMPS. THE 4-HOUR NONEMERGENCY EVENT REPORT REQUIRED BY 10 CFR 50.72 WAS MADE AT 2230 PDT, APRIL 10, 1987. THE REACTOR COOLANT SYSTEM (RCS) HAD BEEN DRAINED TO MIDLOOP LEVEL TO FACILITATE REMOVAL OF STEAM GENERATOR (SG) PRIMARY MANWAYS FOR NOZZLE DAM INSTALLATION. IN ADDITION, PREPARATIONS WERE IN PROGRESS FOR LOCAL LEAK RATE TESTING OF A SEAL WATER RETURN LINE (INCLUDING DRAINING OF THE PENETRATION). DUE TO A LEAKING VALVE USED AS A CLEARANCE POINT IN THE PIPING TO THE PENETRATION BEING DRAINED, RCS INVENTORY WAS LOST TO THE REACTOR COOLANT DRAIN TANK. THIS LOSS OF INVENTORY CAUSED A DECREASE IN RCS WATER LEVEL, VORTEXING IN THE PUMPS' SUCTION LINE, AND AIR ENTRAINMENT IN THE RHR PUMPS. AT 2251 PDT, AFTER VERIFICATION THAT THE SG MANWAYS WERE STILL INSTALLED AND AFTER VENTING OF THE RHR PUMPS, THE RCS WAS FLOODED FROM THE REFUELING WATER STORAGE TANK AND AN RHR PUMP STARTED. RHR FLOW WAS INTERRUPTED FOR APPROXIMATELY 1 HOUR AND 28 MINUTES. THIS RESULTED IN SOME LOCALIZED BOILING BUT NO DAMAGE TO THE CORE OR SIGNIFICANT RADIOLOGICAL RELEASE. THE UNIT WAS STABLE AT 0130 PDT, APRIL 11, 1987, AND WAS RETURNED TO NORMAL MODE 5 MIDLOOP OPERATION.

[95] DIABLO CANYON 2 DOCKET 50-323 LER 87-015 REV 01
UPDATE ON ESF ACTUATION CONSISTING OF A MAIN TURBINE TRIP AND FEEDWATER ISOLATION DUE TO INABILITY TO CONTROL STEAM GENERATOR LEVEL DURING STARTUP.
EVENT DATE: 071487 REPORT DATE: 102787 NSSS: WE TYPE: PWR

(NSIC 206793) AT 0441 PDT, JULY 14, 1987, WITH THE UNIT IN MODE 1 DURING STARTUP, A MAIN TURBINE TRIP, MAIN FEEDWATER PUMP TRIP, AND FEEDWATER ISOLATION VALVE CLOSURE OCCURRED DUE TO STEAM GENERATOR 2-2 REACHING ITS HIGH LEVEL SETPOINT (P-14). THE HIGH STEAM GENERATOR LEVEL WAS CAUSED BY FEEDWATER CONTROL DIFFICULTIES DURING STARTUP RESULTING FROM MECHANICAL PROBLEMS WITH 2 STEAM DUMP VALVES AND OPERATING WITH A POSITIVE MODERATOR TEMPERATURE COEFFICIENT (MTC). OPERATORS PROMPTLY REDUCED REACTOR POWER AND STABILIZED THE PLANT IN MODE 2. A 4-HOUR NONEMERGENCY REPORT WAS MADE AT 0531 PDT, JULY 14, 1987, SINCE A MAIN TURBINE TRIP AND FEEDWATER ISOLATION IS CONSIDERED AN ESF. THE IMMEDIATE CORRECTIVE ACTIONS WERE TO REPAIR AND TEST THE CONDENSER STEAM DUMP SYSTEM AND TO CONTACT OTHER FACILITIES WITH POSITIVE MTC EXPERIENCE FOR INFORMATION ON OPERATING STRATEGIES. THESE STRATEGIES WERE REVIEWED WITH ALL OPERATORS. THE LONG-TERM CORRECTIVE ACTIONS WILL BE TO REVISE STARTUP PROCEDURES TO: PROVIDE MORE DETAIL FOR FEEDWATER CONTROL ON STARTUP, IDENTIFY SYSTEMS REQUIRED TO BE FULLY OPERABLE AND PROVIDE METHODS FOR STARTING UP WITH A POSITIVE MTC. PG&E WIIL REVISE THE SIMULATOR PROGRAM TO MODEL THE CHARACTERISTICS OF THE CORE AFTER A RELOAD IF THERE IS A SIGNIFICANT DIFFERENCE BETWEEN THE REACTOR CORE AND THE SIMULATOR CORE MODEL.

[96] DIABLO CANYON 2 DOCKET 50-323 LER 87-021
FAILURE TO MEET TECHNICAL SPECIFICATION REQUIREMENT FOR INOPERABLE ROD POSITION DEVIATION MONITOR DUE TO PERSONNEL ERROR.
EVENT DATE: 082987 REPORT DATE: 092887 NSSS: WE TYPE: PWR

(NSIC 206589) AT 1425 PDT, AUGUST 29, 1987, WITH THE UNIT IN MODE 1 POWER OPERATION AT 100 PERCENT POWER, THE TIME INTERVAL REQUIREMENT SPECIFIED BY TECHNICAL SPECIFICATION (TS) 4.1.3.1.1, INCLUDING THE ALLOWED EXTENSION OF TS 4.0.2, WAS EXCEEDED. TS 4.1.3.1.1 REQUIRES VERIFICATION, EVERY 4 HOURS WHEN THE ROD POSITION DEVIATION MONITOR (RPDM) IS INOPERABLE, THAT EACH FULL-LENGTH ROD IS WITHIN ITS GROUP DEMAND LIMIT. THE PLANT PROCESS COMPUTER (P-250) WAS REBOOTED TO CORRECT A LOG TYPEWRITER PROBLEM WITHOUT UPDATING THE GROUP ROD POSITIONS AT 0442 PDT, AUGUST 29, 1987, CAUSING INOPERABILITY OF THE RPDM PROGRAM. AT 0454 PDT, AUGUST 30, 1987, DURING PERFORMANCE OF SURVEILLANCE TEST PROCEDURE STP 1-42, IT WAS DETERMINED THAT THE RPDM WAS INOPERABLE. THE RPDM WAS RETURNED TO SERVICE AT 0545 PDT, AUGUST 30, 1987, AND THE CORRECT VERIFICATION FREQUENCY REQUIRED BY TS 4.1.3.1.1 WAS RESUMED. THIS EVENT WAS CAUSED BY A PERSONNEL ERROR IN THAT THE P-250 COMPUTER WAS IMPROPERLY REBOOTED BY PLANT PERSONNEL. A CONTRIBUTING FACTOR WAS A PROCEDURAL DEFICIENCY IN THAT THE PROCEDURE FOR REBOOTING THE P-250 COMPUTER DID NOT CONTAIN ADEQUATE GUIDANCE AS TO WHEN GROUP ROD POSITIONS REQUIRED UPDATING. AN ON-THE-SPOT PROCEDURE REVISION WAS MADE TO THE P-250 REBOOT PROCEDURE, CLARIFYING WHEN ROD BANK POSITION IS REQUIRED TO BE UPDATED WHEN REBOOTING THE COMPUTER.

[97] DIABLO CANYON 2 DOCKET 50-323 LER 87-022
FUEL HANDLING BUILDING VENTILATION SYSTEM SHIFTED TO IODINE REMOVAL MODE.
EVENT DATE: 101287 REPORT DATE: 111087 NSSS: WE TYPE: PWR

(NSIC 207132) ON OCTOBER 12, 1987, AT 0147 PDT, WITH THE UNIT IN MODE 1 (POWER OPERATION) AT 97% POWER, THE FUEL HANDLING BUILDING (FHB)(ND) VENTILATION SYSTEM (VG) SHIFTED INTO THE IODINE REMOVAL MODE. THIS MODE CHANGE CONSTITUTES AN ACTUATION OF AN ENGINEERED SAFETY FEATURE. THE FHB VENTILATION SYSTEM WAS SHIFTED BACK TO THE NORMAL MODE OF OPERATION AT 0220 PDT. THE 4-HOUR NONEMERGENCY REPORT REQUIRED BY 10 CFR 50.72 WAS COMPLETED BY 0406 PDT. THE CAUSE FOR THE FHB VENTILATION SYSTEM SHIFTING INTO IDONE REMOVAL MODE HAS NOT YET

BEEN DETERMINED. THERE WERE NO PRINTOUTS ON RHW ANNUNCIATOR TYPEWRITER (ID) INDICATING A FAN FAILURE OR A RADIATION MONITOR SIGNAL. INVESTIGATION INCLUDED CHECKS OF THE DIGITAL CIRCUITRY LOGIC DIAGRAMS AND LOGIC STATES TO DETERMINE POSSIBLE CAUSES. NO ABNORMALITIES WERE IDENTIFIED. TO AID IN DETERMINING THE CAUSE, A MULTI POINT DATA RECORDER WILL BE INSTALLED TO GETHER DATA IF THIS EVENT RECURS. PG&E WILL SUBMIT A SUPPLEMENTAL LER WITH ACTIONS TO PREVENT RECURRENCE WHEN THE CAUSE HAS BEEN DETERMINED. SINCE THERE WAS NO INDICATION OF AN ACTUAL HIGH RADIATION CONDITION, THIS FHB VENTILATION SYSTEM MODE CHANGE FROM NORMAL TO IODINE REMOVAL MODE REPRESENTS A CONSERVATIVE RESPONSE. THUS, THERE WERE NO ADVERSE SAFETY CONSEQUENCES OR IMPLICATIONS RESULTING FROM THIS EVENT.

[98] DRESDEN 2 DOCKET 50-237 LER 87-001 REV 01
UPDATE ON UT INDICATIONS FOUND ON PRIMARY SYSTEM PIPING DUE TO INTERGRANULAR STRESS CORROSION CRACKING.
EVENT DATE: 010287 REPORT DATE: 011187 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 206955) ON 1/2/87 AT 1420 HOURS, WITH UNIT 2 IN THE REFUEL MODE, ULTRASONIC TESTING (UT) OF A 12 INCH RISER IN THE REACTOR RECIRCULATION SYSTEM SHOWED WELD PD5-D20 WITH A 2.5 INCH, 15% THROUGH WALL INDICATION ON THE ELBOW SIDE OF THE WELD AND .625 INCH, 20% THROUGH WALL INDICATION ON THE PIPE SIDE OF THE WELD. ON 1/21/87, AT 1615 HOURS, A CIRCUMFERENTIAL INDICATION, 2.0 INCHES IN LENGTH AND 36% THROUGH WALL IN DEPTH WAS DISCOVERED WHILE PERFORMING A UT ON REACTOR WATER CLEANUP (RWCU) SYSTEM 8 INCH PIPE WELD 8-12. THE ROOT CAUSE HAS BEEN ATTRIBUTED TO INTERGRANULAR STRESS CORROSION CRACKING. AS A CORRECTIVE ACTION AN ENGINEERING EVALUATION ON THE INDICATIONS WAS PERFORMED. THE EVALUATION DETERMINED THAT THE TWO INDICATIONS FOUND ON THE REACTOR RECIRCULATION SYSTEM WELD WOULD NOT REQUIRE A WELD OVERLAY OR REPLACEMENT FOR AT LEAST ONE MORE OPERATING CYCLE. HOWEVER, A WELD OVERLAY WAS REQUIRED ON THE CIRCUMFERENTIAL INDICATION FOUND ON THE RWCU SYSTEM. IN ADDITION TO THE ENGINEERING EVALUATION AND WELD OVERLY REPAIR, THE NUMBER OF WELD INSPECTIONS WAS INCREASED. A TOTAL OF 63 WELDS WERE INITIALLY SCHEDULED FOR INSPECTION DURING THE UNIT 2 REFUELING OUTAGE AS PART OF THE IN-SERVICE INSPECTION PROGRAM. IN COMPLIANCE WITH SECTION 3.6.F OF THE TECH SPECS, 21 ADDITIONAL WELDS WERE INSPECTED AS A RESULT OF THE INDICATIONS FOUND.

[99] DRESDEN 2 DOCKET 50-237 LER 87-025
FAILURE TO OBTAIN GRAB SAMPLE OF UNIT 2/3 CHIMNEY EFFLUENT DUE TO PERSONNEL ERROR.
EVENT DATE: 090387 REPORT DATE: 100187 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: DRESDEN 3 (BWR)

(NSIC 206607) WITH UNIT 2 AT 95% POWER AND UNIT 3 SHUT DOWN FOR FEEDWATER SYSTEM MODIFICATIONS, THE LOW, MEDIUM, AND HIGH RANGE SEPARATE, PARTICULATE, IODINE, AND NOBLE GAS (SPING) 2/3 CHIMNEY MONITOR WAS OUT OF SERVICE FOR CALIBRATION/REPAIR. THIS REQUIRED THAT 2/3 CHIMNEY NOBLE GAS GRAB SAMPLES BE TAKEN ONCE PER SHIFT IN ACCORDANCE WITH TECH SPEC TABLE 3.2.5. HOWEVER, THE AFTERNOON SHIFT GRAB SAMPLE WAS NOT TAKEN ON SEPTEMBER 3, 1987, DUE TO PERSONNEL ERROR ON THE PART OF THE RADIATION-CHEMISTRY FOREMAN ON DUTY IN THAT A RADIATION-CHEMISTRY TECHNICIAN WAS NOT A ADEQUATELY INFORMED OF THE NEED TO PERFORM THIS TASK. THE FOREMAN INVOLVED WAS SPECIFICALLY COUNSELLED ON PERFORMANCE EXPECTATIONS IN THE AREA OF JOB ASSIGNMENTS AND PROPER FOLLOW-UP. THE SAFETY SIGNIFICANCE OF THIS EVENT WAS MINIMAL SINCE THE GENERAL ELECTRIC CHIMNEY MONITORING SYSTEM WAS IN SERVICE DURING THIS PERIOD, ALLOWING PARTICULATE, IODINE, AND LOW RANGE NOBLE GAS MONITORING AT ALL TIMES. A PREVIOUS EVENT INVOLVING FAILURE TO COMPLETE THE GRAB SAMPLE REQUIREMENT IS REPORTED BY LER NO. 86-25 ON DOCKET 50-237.

[100] DRESDEN 2 DOCKET 50-237 LER 87-026
LOW REACTOR WATER LEVEL SCRAM SWITCH FOUND BELOW SETPOINT LIMITS DUE TO LOGIC CARD INSTRUMENT DRIFT.
EVENT DATE: 091587 REPORT DATE: 100887 NSSS: GE TYPE: BWR
VENDOR: ROSEMOUNT, INC.

(NSIC 206702) AT 2300 HOURS ON SEPTEMBER 15, 1987, WITH DRESDEN UNIT 2 IN THE RUN MODE AT 96% POWER, WHILE PERFORMING DRESDEN INSTRUMENT SURVEILLANCE (DIS) 500-2, THE CHANNEL B REACTOR VESSEL LOW WATER LEVEL SCRAM SETPOINT WAS FOUND AT 13 INCHES BELOW INSTRUMENT ZERO. THIS INSTRUMENTATION ALSO INITIATES AUTOMATIC PRIMARY CONTAINMENT GROUP II AND III ISOLATIONS. THE TECH SPEC LIMIT, WHEN COMPENSATED FOR 100% STEAM FLOW, IS 8 INCHES ABOVE INSTRUMENT ZERO. THE PROXIMATE CAUSE WAS DETERMINED TO BE INSTRUMENT DRIFT OF THE MASTER TRIP UNIT 2-263-140B LOGIC CARD. THE SAFETY SIGNIFICANCE OF THE EVENT WAS DEEMED MINIMAL BECAUSE THE REDUNDANT CHANNEL B SWITCH IN COMBINATION WITH THE A SWITCHES WERE AVAILABLE TO PROVIDE THE AUTOMATIC SCRAM AND PRIMARY CONTAINMENT ISOLATIONS. THE SETPOINT OF MASTER TRIP UNIT 2-263-140B WAS ADJUSTED WITHIN THE LIMITS OF DIS 500-2 WITHIN APPROXIMATELY ONE HOUR. THE LOGIC CARD WAS SUBSEQUENTLY REPLACED AND A WORK REQUEST GENERATED TO REPAIR THE OLD CARD AND DETERMINE THE ROOT CAUSE OF THE TRIP UNIT'S DRIFT. DIS 500-2 WILL CONTINUE TO BE PERFORMED ON A MONTHLY BASIS. THIS WAS THE FIRST OCCURRENCE OF INSTRUMENT DRIFT OF THE ROSEMOUNT MASTER TRIP UNIT AT DRESDEN STATION.

[101] DRESDEN 2 DOCKET 50-237 LER 87-028
FAILURE OF SECONDARY CONTAINMENT LEAK TEST DUE TO EXCESSIVE IN-LEAKAGE.
EVENT DATE: 091887 REPORT DATE: 101587 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: DRESDEN 3 (BWR)

(NSIC 206703) AT 0230 HOURS ON SEPTEMBER 18, 1987, AN ORDERLY UNIT SHUTDOWN OF UNITS 2 AND 3 WAS INITIATED DUE TO INABILITY TO MAINTAIN 0.25 INCHES OF WATER NEGATIVE DIFFERENTIAL PRESSURE IN THE REACTOR BUILDINGS WITH RESPECT TO THE OUTSIDE ATMOSPHERE DURING PERFORMANCE OF A SECONDARY CONTAINMENT LEAK RATE TEST (SCLRT). UNIT 2 AND UNIT 3 WERE OPERATING AT 93% AND 59% RATED THERMAL POWER, RESPECTIVELY. ALTHOUGH THE SCLRT IS NORMALLY ONLY PERFORMED AT EACH REFUELING OUTAGE IN ACCORDANCE WITH TECH SPEC 4.7.C.1.C, THIS TEST WAS PERFORMED DUE TO AN OPERATOR OBSERVATION THAT WHILE RUNNING THE STANDBY GAS TREATMENT SYSTEM ON SEPTEMBER 16, 1987, THE CONTROL ROOM DIFFERENTIAL PRESSURE INDICATION DID NOT SHOW 0.25 INCHES OF WATER NEGATIVE. A THOROUGH WALKDOWN OF THE SECONDARY CONTAINMENT WAS INITIATED TO IDENTIFY SOURCES OF AIR IN-LEAKAGE. THE SCLRT WAS THEN PERFORMED IN ORDER TO DETERMINE SECONDARY CONTAINMENT OPERABILITY IN THE AS-FOUND CONDITION. DURING PERFORMANCE OF THE SCLRT, A DIFFERENTIAL PRESSURE OF APPROXIMATELY 0.2 INCHES OF WATER WAS OBSERVED, WHICH DID NOT MEET THE TECH SPEC 4.7.C.1.C OPERABILITY CRITERIA OF 0.25 INCHES OF WATER. ALTHOUGH THE AMBIENT WIND CONDITIONS WERE IN EXCESS OF THE CALM (<5 MPH) CONDITIONS SPECIFIED IN TECH SPEC 4.7.C.1.C, THE ORDERLY SHUTDOWN WAS PERFORMED.

[102] DRESDEN 2 DOCKET 50-237 LER 87-029
HIGH PRESSURE COOLANT INJECTION SYSTEM INOPERABLE DUE TO STEAM LEAK.
EVENT DATE: 100187 REPORT DATE: 102787 NSSS: GE TYPE: BWR

(NSIC 206896) ON OCTOBER 1, 1987 AT 1450 HOURS, WITH UNIT 2 AT 97% RATED THERMAL POWER WHILE PERFORMING DRESDEN OPERATING SURVEILLANCE (DOS) 2300-3 HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM PUMP TEST, AND DOS 2300-6, MONTHLY HPCI SYSTEM PUMP TEST FOR IN-SERVICE TESTING (IST) PROGRAM, THE HPCI SYSTEM WAS CONSERVATIVELY DECLARED INOPERABLE AS A RESULT OF A STEAM LEAK IN THE VICINITY OF THE HPCI TURBINE SHAFT SEAL. THE EXACT CAUSE OF THIS LEAKAGE IS UNKNOWN; INVESTIGATION BY OPERATIONS AND MAINTENANCE PERSONNEL VERIFIED PROPER OPERATION OF THE GLAND SEAL LEAKOFF SYSTEM. CORRECTIVE ACTION INCLUDED OBSERVATIONS OF THE HPCI SYSTEM ON THREE INDIVIDUAL STARTUPS IN WHICH NO LEAKAGE WAS OBSERVED IN THE

SEAL AREA. THE SAFETY SIGNIFICANCE WAS MINIMAL BECAUSE HPCI WAS CAPABLE OF INITIATION IN THIS CONDITION AND DUE TO THE AVAILABILITY OF THE AUTOMATIC DEPRESSURIZATION, ISOLATION CONDENSER, LOW PRESSURE INJECTION, AND CORE SPRAY SYSTEMS TO PROVIDE REDUNDANT MEANS OF REACTOR INVENTORY AND PRESSURE CONTROL DURING ANY DESIGN BASIS ACCIDENT. A PREVIOUS EVENT INVOLVING AN INOPERABLE HPCI SYSTEM IS REPORTED BY LER #87-17 ON DOCKET #050249.

[103] DRESDEN 2 DOCKET 50-237 LER 87-031
REACTOR BUILDING VENTILATION ISOLATION START OF SBGT SYSTEM DUE TO IRRADIATED METAL ON FUEL CASK.
EVENT DATE: 100387 REPORT DATE: 102687 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: DRESDEN 3 (BWR)

(NSIC 206895) WITH DRESDEN UNIT 2 OPERATING AT 93% RATED THERMAL POWER ON OCTOBER 3, 1987, FUEL HANDLERS WERE REMOVING A SHIPPING CASK FROM THE UNIT 2 FUEL POOL. IRRADIATED METAL HAD BECOME ATTACHED TO THE BOTTOM OF THE CASK STABILIZER. RADIATION LEVELS EXCEEDED THE 90 MR/HR SETPOINT OF THE REFUEL FLOOR AREA RADIATION MONITORS. THIS CAUSED AN UNPLANNED UNIT 2 REACTOR BUILDING VENTILATION ISOLATION AND STANDBY GAS TREATMENT (SBGT) SYSTEM AUTO START. THE UNIT 3 REACTOR BUILDING VENTILATION WAS MANUALLY TRIPPED AND ISOLATED. IN ORDER TO PREVENT RECURRENCE, THE CASK PAD AND SURROUNDING AREAS IN BOTH DRESDEN REFUEL POOLS WILL BE INSPECTED TO IDENTIFY AND REMOVE ANY DEBRIS. FURTHERMORE, CASK HANDLING PROCEDURES WILL BE MODIFIED TO INCLUDE AN INSPECTION OF THE CASK PAD PRIOR TO USE. LASTLY, A CHECKLIST FOR SHIFT SUPERVISOR NOTIFICATION WILL BE ADDED TO CASK HANDLING PROCEDURES. THIS EVENT WAS OF MINIMAL SAFETY SIGNIFICANCE SINCE THE SBGT FUNCTIONED AS REQUIRED AND A RADIATION CHEMISTRY TECHNICIAN WAS IN CONTINUOUS ATTENDANCE AT THE REFUELING FLOOR, ENSURING THAT WORKER EXPOSURE WAS KEPT WITHIN PROPER LIMITS. THE LAST EVENT OF THIS TYPE WAS REPORTED BY LER #86-001 ON DOCKET #050249.

[104] DRESDEN 2 DOCKET 50-237 LER 87-034
NON-CONSERVATIVE CORE THERMAL POWER (CTP) CALCULATION DUE TO INADEQUATE CALIBRATION PROCEDURE.
EVENT DATE: 101687 REPORT DATE: 110487 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: DRESDEN 3 (BWR)
VENDOR: ROSEMOUNT ENGINEERING COMPANY

(NSIC 206964) DURING REVIEW OF A GENERAL ELECTRIC PLANT PERFORMANCE MONITORING PROGRAM, IT WAS DISCOVERED THAT THE ROSEMOUNT FEEDWATER FLOW TRANSMITTERS HAD NOT BEEN CALIBRATED TO ACCOUNT FOR THE EFFECTS OF STATIC PRESSURE SPAN COMPRESSION. CURRENTLY, THREE ROSEMOUNT 1151 DP TRANSMITTERS ARE IN USE IN UNIT 2, AND ONE ON UNIT 3. ALTHOUGH THE ERROR IS SMALL (0.44% AT NOMINAL OPERATING CONDITIONS) IT IS NON-CONSERVATIVE, WHICH LEADS TO A NON-CONSERVATIVE CORE THERMAL POWER CALCULATION. BECAUSE OF THE ERRANT TRANSMITTER CALIBRATION, IT IS ESTIMATED THAT UNIT 2 AND UNIT 3 OPERATED 46 HOURS AND 100 HOURS, RESPECTIVELY, ABOVE THE CORE THERMAL POWER LIMIT OF 2527 THERMAL MEGAWATTS SINCE THE INSTALLATION OF THESE TRANSMITTERS (UNIT 3 - SEPTEMBER, 1985, UNIT 2 - JANUARY 1985). THE EVENT CAN BE ATTRIBUTED TO AN INADEQUATE PROCEDURE FOR CALIBRATING THE ROSEMOUNT 1151 DP TRANSMITTERS. UPON DISCOVERY OF THE PROBLEM, APPROPRIATE LOAD REDUCTIONS WERE ADMINISTERED, AND AFFECTED FEEDWATER FLOW TRANSMITTERS WERE RECALIBRATED. TO PREVENT RECURRENCE, CALIBRATION PROCEDURES WILL BE REVISED ACCORDINGLY. A SIMILAR EVENT WAS REPORTED IN LER #86-04 ON DOCKET #050237.

[105] DRESDEN 3 DOCKET 50-249 LER 87-014
PLANT SHUTDOWN DUE TO INOPERABLE HIGH PRESSURE COOLANT INJECTION AND ISOLATION CONDENSER SYSTEMS.
EVENT DATE: 090587 REPORT DATE: 100287 NSSS: GE TYPE: BWR
VENDOR: CRANE VALVE CO.

DIETERICH STANDARD CORP.

(NSIC 206765) ON SEPTEMBER 5, 1987, WITH UNIT 3 AT 6% RATED THERMAL POWER, A SPECIAL TEST OF THE ISOLATION CONDENSER WAS BEING PERFORMED IN ORDER TO PINPOINT THE CAUSE OF A PREVIOUS SPURIOUS ISOLATION CONDENSER ISOLATION AS REPORTED BY LER #87-13 ON DOCKET #050249. WHILE PERFORMING THIS TESTING AT 0720 HOURS, THE ISOLATION CONDENSER SPURIOUSLY ISOLATED DUE TO DIFFERENTIAL PRESSURE SIGNALS GENERATED BY AN ANNUBAR FLOW SENSING INSTRUMENT LOCATED ON THE ISOLATION CONDENSER CONDENSATE RETURN LINE. THE ISOLATION CONDENSER WAS DECLARED INOPERABLE AND THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM WAS IMMEDIATELY TESTED SATISFACTORILY AS REQUIRED BY TECH SPEC 4.5.E.2. HOWEVER, WHEN HPCI WAS SUBSEQUENTLY TESTED ON SEPTEMBER 6, 1987 AT 17% RATED THERMAL POWER IT WAS DECLARED INOPERABLE DUE TO INABILITY TO OBTAIN THE REQUIRED DISCHARGE PRESSURE IN THE SURVEILLANCE MODE AND TRIPPING OF THE GLAND SEAL LEAKOFF BLOWER. AN ORDERLY UNIT SHUTDOWN WAS THEN COMPLETED, AND MODIFICATIONS TO THE ISOLATION CONDENSER ISOLATION CIRCUITRY AND REPAIRS TO A HPCI TEST RETURN LINE VALVE AND GLAND SEAL LEAKOFF BLOWER WERE PERFORMED. SAFETY SIGNIFICANCE WAS MITIATED BY THE LOW POWER LEVELS DURING THIS TESTING PROGRAM AND THE AVAILABILITY OF THE AUTOMATIC DEPRESSURIZATION AND LOW PRESSURE EMERGENCY CORE COOLING SYSTEMS.

[106] DRESDEN 3 DOCKET 50-249 LER 87-017
HPCI SYSTEM INOPERABLE DUE TO TRIPPING OF THE GLAND SEAL LEAKOFF BLOWER CAUSED BY CONDENSER OVERFLOW.
EVENT DATE: 091287 REPORT DATE: 100987 NSSS: GE TYPE: BWR
VENDOR: MASONEILAN INTERNATIONAL, INC.

(NSIC 206767) ON SEPTEMBER 12, 1987, WITH UNIT 3 AT 12% RATED THERMAL POWER AND 340 PSIG REACTOR PRESSURE, SURVEILLANCE TESTING OF THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM WAS BEING PERFORMED TO VERIFY HPCI OPERABILITY FOLLOWING REPLACEMENT OF THE HPCI GLAND SEAL LEAKOFF (GSLO) DRAIN PUMP. THE GSLO DRAIN PUMP WAS REPLACED AS A RESULT OF AN EVENT ON SEPTEMBER 6, 1987 (REFER TO LER #87-14 ON DOCKET #050249), WHEN HPCI WAS DECLARED INOPERABLE DUE TO TRIPPING OF THE GSLO BLOWER CAUSED BY OVERFLOW OF THE HPCI GSLO CONDENSER. THE ISOLATION CONDENSER ISOLATION CIRCUITRY HAD BEEN COMPLETED AND TESTED SATISFACTORILY. WHILE PERFORMING THE HPCI TESTING AT 2200 HOURS ON SEPTEMBER 12, 1987, THE HPCI GSLO BLOWER AGAIN TRIPPED DUE TO HIGH WATER LEVEL IN THE GSLO CONDENSER. IN ORDER TO PREVENT RECURRENCE OF GSLO CONDENSER OVERFLOW, A GSLO CONDENSER COOLING WATER INLET PRESSURE CONTROL VALVE WAS CLEANED, LUBRICATED AND RESET. SAFETY SIGNIFICANCE WAS MITIGATED BY THE LOW POWER LEVELS DURING THIS TESTING PROGRAM AND THE AVAILABILITY OF THE AUTOMATIC DEPRESSURIZATION AND LOW PRESSURE EMERGENCY CORE COOLING SYSTEMS.

[107] DRESDEN 3 DOCKET 50-249 LER 87-015
HPCI SYSTEM INOPERABLE DUE TO FAILURE OF MINIMUM FLOW VALVE.
EVENT DATE: 091587 REPORT DATE: 101487 NSSS: GE TYPE: BWR
VENDOR: LIMITORQUE CORP.

(NSIC 206766) ON SEPTEMBER 15, 1987 AT 0450 HOURS, WITH UNIT 3 AT 10% RATED THERMAL POWER WHILE PERFORMING DRESDEN OPERATING SURVEILLANCE (DOS) 2300-3, HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM PUMP TEST, HPCI MINIMUM FLOW VALVE M03-2301-14 TRIPPED THERMALLY WHILE TRAVELING IN THE CLOSED DIRECTION. SUBSEQUENT INVESTIGATION REVEALED THAT THE VALVE GEARBOX WAS DEGRADED AND AT 0950 HOURS ON SEPTEMBER 6, 1987 THE HPCI SYSTEM WAS DECLARED INOPERABLE. THE ROOT CAUSE OF THE M03-2301-14 VALVE FAILURE WAS DETERMINED TO BE WATER INTRUSION INTO THE LIMITORQUE MOTOR-OPERATOR GEAR CASE. THIS RESULTED IN INTERNAL CORROSION OF THE GEARBOX. CORRECTIVE ACTION INCLUDED REBUILDING OF THE MOTOR OPERATOR AND PERFORMING A MOTOR CURRENT SIGNATURE TO ENSURE PROPER OPERATION. THE SAFETY SIGNIFICANCE OF THIS EVENT WAS MINIMAL DUE TO AVAILABILITY OF THE AUTOMATIC DEPRESSURIZATION, ISOLATION CONDENSER, LOW PRESSURE COOLANT INJECTION, AND CORE

SPRAY SYSTEMS TO PROVIDE REDUNDANT MEANS OF REACTOR INVENTORY AND PRESSURE CONTROL DURING ANY DESIGN BASIS LOSS OF COOLANT ACCIDENT. A PREVIOUS EVENT INVOLVING AN INOPERABLE HPCI SYSTEM IS REPORTED BY LER #87/17 ON DOCKET #050249.

[108] DRESDEN 3 DOCKET 50-249 LER 87-016
PRIMARY CONTAINMENT GROUP I ISOLATION AND REACTOR SCRAM DUE TO APPARENT PERSONNEL ERROR.
EVENT DATE: 092887 REPORT DATE: 102687 NSSS: GE TYPE: BWR
VENDOR: ANDERSON, GREENWOOD & CO.

(NSIC 206860) ON SEPTEMBER 28, 1987 AT 0231 HOURS WITH THE UNIT 3 REACTOR OPERATING AT 78% RATED POWER, A PRIMARY CONTAINMENT GROUP I ISOLATION AND SUBSEQUENT REACTOR SCRAM OCCURRED WHILE AN INSTRUMENT MECHANIC (IM) WAS PERFORMING A CALIBRATION PROCEDURE FOR THE MAIN STEAM LINE (MSL) HIGH FLOW SWITCHES. THE ROOT CAUSE OF THE GROUP I ISOLATION HAS BEEN ATTRIBUTED TO APPARENT PERSONNEL ERROR BY THE IM IN NOT PROPERLY ISOLATING MSL HIGH FLOW SWITCH DPIS 3-261-2N DURING THE SURVEILLANCE. PRIOR TO LOADING THE PRESSURE SWITCH FOR CALIBRATION, THE IM VENTED THE HIGH AND LOW SIDE LEGS OF THE PRESSURE SWITCH. DURING THE VENTING PROCESS, AN UNEXPECTED INCREASE IN DIFFERENTIAL PRESSURE OCCURRED. THE IM IMMEDIATELY CLOSED THE EQUALIZING VALVE IN AN ATTEMPT TO STOP THE PRESSURE INCREASE. THIS ACTION CAUSED A PRESSURE SPIKE IN THE COMMON HEADER, IN WHICH THREE OTHER PRESSURE SWITCHES ARE CONNECTED, AND RESULTED IN THE GROUP I ISOLATION AND SUBSEQUENT REACTOR SCRAM ON MAIN STEAM ISOLATION VALVE (MSIV) CLOSURE. CORRECTIVE ACTION INCLUDED DISCUSSION OF THIS EVENT WITH THE IM, AND TESTING/INSPECTION OF THE INSTRUMENT VALVE MANIFOLD. THE INSTRUMENT VALVE MANIFOLD, WHICH CONTAINS THE HIGH AND LOW ISOLATION VALVES AND EQUALIZING VALVE, WAS REPLACED AND PRESSURE SWITCH DPIS 3-261-2N WAS CALIBRATED.

[109] DRESDEN 3 DOCKET 50-249 LER 87-018
FIRE STOP 18 MONTH SURVEILLANCE INTERVAL EXCEEDED DUE TO PROCEDURAL DEFICIENCY.
EVENT DATE: 093087 REPORT DATE: 102687 NSSS: GE TYPE: BWR

(NSIC 206859) AT 1600 HOURS ON SEPTEMBER 30, 1987, WITH DRESDEN UNIT 3 AT 37% POWER, IT WAS FOUND THAT DRESDEN FIRE PROTECTION PROCEDURE (DFPP) 4175-3, SHUTDOWN FIRE STOP/BREAK SURVEILLANCE, WAS INCORRECTLY CLASSIFIED IN THE SURVEILLANCE PROGRAM AS DUE EACH REFUELING OUTAGE. PERFORMANCE OF THIS SURVEILLANCE IS REQUIRED ON AN 18 MONTH INTERVAL BY TECH SPEC 4.12.F.1. A REVIEW OF THE PAST SURVEILLANCE HISTORY IDENTIFIED THAT A PREVIOUS SURVEILLANCE INTERVAL WAS EXCEEDED IN NOVEMBER 1985 DUE TO THIS PROCEDURAL DEFICIENCY. CURRENT SURVEILLANCES WERE PERFORMED WITHIN THE REQUIRED INTERVAL. SINCE THE REACTOR WAS IN AN EXTENDED REFUELING OUTAGE AT THE TIME OF THE EXCEEDED INTERVAL, THE EVENT WAS OF MINIMAL SAFETY SIGNIFICANCE. CORRECTIVE ACTIONS INCLUDED A REVIEW OF ALL SURVEILLANCES AGAINST THE APPROPRIATE TECH SPEC SURVEILLANCE INTERVAL. A CHANGE WAS ALSO MADE TO THE SURVEILLANCE PROGRAM PROCEDURE SUCH THAT DFPP 4175-3 IS IDENTIFIED AS REQUIRING PERFORMANCE ON AN 18 MONTH INTERVAL. A PREVIOUS OCCURRENCE INVOLVING EXCEEDING A SURVEILLANCE INTERVAL IS REPORTED BY LER #87-27 ON DOCKET #050237.

[110] FARLEY 1 DOCKET 50-348 LER 87-015
FIRE DAMPERS INOPERABLE DUE TO FAILURE TO CLOSE WITH AIR FLOW.
EVENT DATE: 090787 REPORT DATE: 100287 NSSS: WE TYPE: PWR

(NSIC 206597) ON 7-13-87, FNP SUBMITTED A SPECIAL REPORT (LER 87-011-00) CONCERNING INOPERABLE CONTROL ROOM FIRE DAMPERS. AS A RESULT, A FIRE DAMPER MAINTENANCE AND TESTING PROGRAM IS IN PROGRESS. OF THE 63 TESTED FIRE DAMPERS, THE FOLLOWING WOULD NOT CLOSE WITH AIR FLOW (THE DATE OF THE TEST IS SHOWN IN PARENTHESES): 1-113-05 (9-14-87), 1-113-06 (9-14-87), 1-115-15 (9-22-87), AND 1-119-18 (8-31-87). THESE EVENTS WERE CAUSED BY DESIGN DEFICIENCY IN THAT THE

FIRE DAMPERS WILL NOT CLOSE FULLY AND LATCH WITH AIR FLOW. DESIGN CHANGES WILL BE INITIATED TO EVALUATE THE OPTIONS AVAILABLE AND PROVIDE THE APPROPRIATE DESIGN TO ENSURE THE PROPER OPERATION OF THE FIRE DAMPERS. THESE DESIGN CHANGES ARE EXPECTED TO BE IMPLEMENTED WITHIN THE NEXT SIX MONTHS. TECHNICAL SPECIFICATION 3.7.12 REQUIRES THESE FIRE DAMPERS TO BE RETURNED TO OPERABLE STATUS WITHIN SEVEN DAYS OR A SPECIAL REPORT MUST BE SUBMITTED WITHIN THE FOLLOWING 30 DAYS. THEREFORE THIS SPECIAL REPORT IS BEING SUBMITTED. ALL TECHNICAL SPECIFICATION ACTION STATEMENT REQUIREMENTS FOR THE FIRE DAMPERS ARE BEING MET.

[111] FARLEY 1 DOCKET 50-348 LER 87-016
FIRE DAMPERS INOPERABLE DUE TO FAILURE TO CLOSE WITH AIR FLOW.
EVENT DATE: 092487 REPORT DATE: 102687 NSSS: WE TYPE: PWR

(NSIC 206805) ON 7-13-87, FNP SUBMITTED A SPECIAL REPORT (LER 87-011-00) CONCERNING INOPERABLE CONTROL ROOM FIRE DAMPERS. AS A RESULT, A FIRE DAMPER MAINTENANCE AND TESTING PROGRAM IS IN PROGRESS. AS PART OF THIS PROGRAM, ON 9-17-87, FIRE DAMPERS 1-115-05 AND 1-115-08 WERE TESTED AND WOULD NOT CLOSE WITH AIR FLOW. DAMPER 1-115-05 WAS LUBRICATED AND SUCCESSFULLY CLOSED UNDER NO FLOW CONDITIONS. THIS DAMPER HAD BEEN INSTALLED IMPROPERLY. IT WAS REMOVED FROM THE SYSTEM AND RE-INSTALLED CORRECTLY. FOLLOWING THIS ADJUSTMENT, THE DAMPER SUCCESSFULLY CLOSED WITH AIR FLOW AND WAS DECLARED OPERABLE AT 1109 ON 10/18/87. DAMPER 1-115-08 INITIALLY CLOSED UNDER NO FLOW CONDITIONS AND AFTER FURTHER LUBRICATION AND ADJUSTMENT IT CLOSED UNDER AIR FLOW CONDITIONS. IT WAS DECLARED OPERABLE AT 0800 ON 10/16/87. CORRECTIVE ACTION TO PREVENT RECURRENCE OF THIS EVENT WAS STATED IN LER 87-011-00. TECHNICAL SPECIFICATION 3.7.12 REQUIRES THESE FIRE DAMPERS TO BE RETURNED TO OPERABLE STATUS WITHIN SEVEN DAYS OR A SPECIAL REPORT MUST BE SUBMITTED WITHIN THE FOLLOWING 30 DAYS. THEREFORE THIS SPECIAL REPORT IS BEING SUBMITTED. ALL TECHNICAL SPECIFICATION ACTION STATEMENT REQUIREMENTS FOR THE FIRE DAMPERS HAVE BEEN MET.

[112] FARLEY 1 DOCKET 50-348 LER 87-017
FIRE DOOR 496 INOPERABLE FOR MORE THAN SEVEN DAYS.
EVENT DATE: 100887 REPORT DATE: 110387 NSSS: WE TYPE: PWR

(NSIC 206931) AT 2130 ON 10-1-87, FIRE DOOR 496 WAS DECLARED INOPERABLE AND COULD NOT BE REPAIRED WITHIN SEVEN DAYS. TECHNICAL SPECIFICATION 3.7.12 REQUIRES THE FIRE DOOR TO BE RETURNED TO OPERABLE STATUS WITHIN SEVEN DAYS OR A SPECIAL REPORT MUST BE SUBMITTED WITHIN THE FOLLOWING 30 DAYS. THEREFORE THIS SPECIAL REPORT IS BEING SUBMITTED. THE FIRE DOOR WAS DECLARED INOPERABLE BECAUSE THE DOOR WAS BENT AND THE LATCHING MECHANISM WAS STUCK SO THAT THE DOOR COULD NOT BE LATCHED. THE DOOR WAS ADJUSTED, BUT COULD NOT BE REPAIRED AND MUST BE REPLACED. A NEW DOOR HAS BEEN ORDERED AND IS EXPECTED TO BE RECEIVED ON DECEMBER 9, 1987. REPAIRS WILL BE COMPLETED AS SOON AS PRACTICABLE AFTER THE DOOR HAS BEEN RECEIVED. FIRE WATCH REQUIREMENTS CONTINUE TO BE MET AND A FIRE WATCH WILL BE MAINTAINED UNTIL THE DOOR IS REPLACED AND DETERMINED FUNCTIONAL. ALL TECHNICAL SPECIFICATION ACTION STATEMENT REQUIREMENTS FOR THE FIRE DOOR ARE BEING MET.

[113] FARLEY 1 DOCKET 50-348 LER 87-018
DRAWING ERROR LEADS TO BOTH CONTAINMENT HYDROGEN RECOMBINERS BEING INOPERABLE.
EVENT DATE: 101387 REPORT DATE: 110987 NSSS: WE TYPE: PWR

(NSIC 206932) AT 1615 ON 10-13-87, IT WAS RECOGNIZED THAT A DRAWING ERROR EXISTED WHICH SHOWED THE LOCATIONS OF THE 1A AND THE 1B CONTAINMENT HYDROGEN RECOMBINERS TO BE REVERSED. EARLIER, THE BREAKER FOR THE 1A CONTAINMENT HYDROGEN RECOMBINER HAD BEEN TAGGED OPEN TO ALLOW FOR MAINTENANCE ON THE RECOMBINER. AT APPROXIMATELY 1030 ON 10-13-87, THE CABLES PROVIDING POWER TO WHAT WAS THOUGHT TO BE THE 1A RECOMBINER WERE DETERMINATED. ON INVESTIGATION OF THE DRAWING ERROR, IT WAS FOUND THAT THE WIRING HAD IN FACT BEEN DETERMINATED FROM THE 1B RECOMBINER

AND NOT FROM THE 1A RECOMBINER. THUS, BOTH RECOMBINERS WERE INOPERABLE. THEREFORE, TECHNICAL SPECIFICATION 3.0.3 HAD BEEN APPLICABLE SINCE 1020. UPON RECOGNITION THAT BOTH RECOMBINERS WERE INOPERABLE, EFFORTS WERE BEGUN IMMEDIATELY TO RETURN THE 1A RECOMBINER TO SERVICE BY CLEARING THE TAGGING ORDER AND CLOSING THE BREAKER. THE 1A RECOMBINER WAS RETURNED TO SERVICE AT 1629. THEREFORE, TECHNICAL SPECIFICATION 3.0.3 WAS NO LONGER IN EFFECT. THIS EVENT WAS CAUSED BY AN INCORRECT DRAWING. THE DRAWING WILL BE CORRECTED TO SHOW THE CORRECT LOCATION OF THE CONTAINMENT HYDROGEN RECOMBINERS.

[114] FARLEY 1 DOCKET 50-348 LER 87-019
VISITOR FOUND UNESCORTED IN A VITAL AREA.
EVENT DATE: 101387 REPORT DATE: 110387 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: FARLEY 2 (PWR)

(NSIC 206933) AT 0655 ON 10-13-87, A SECURITY FORCE MEMBER ENCOUNTERED AN UNESCORTED VISITOR IN A VITAL AREA. THEREFORE, THIS REPORT IS BEING SUBMITTED IN ACCORDANCE WITH 10CFR73.71(B). THE VISITOR AND HIS ESCORT HAD BEEN IN THE DOSIMETRY LABORATORY. UPON EXITING THE DOSIMETRY LABORATORY, THE VISITOR AND THE ESCORT BECAME SEPARATED. THE ESCORT PROCEEDED DOWN THE HALL WHILE THE VISITOR REALIZED HE HAD LEFT HIS HARD HAT AND RETURNED TO PICK IT UP. A SECURITY FORCE MEMBER ENCOUNTERED THE VISITOR AT 0655 AND ESCORTED HIM OUT OF THE AUXILIARY BUILDING. THE ESCORT REALIZED THAT HE HAD LOST CONTACT WITH THE VISITOR AND BEGAN A SEARCH. AT 0708, THE ESCORT INFORMED SECURITY THAT HE WAS UNABLE TO LOCATE THE VISITOR. NO SAFETY SYSTEMS WERE AFFECTED OR THREATENED. IT IS ESTIMATED THAT THE VISITOR HAD BEEN AWAY FROM HIS ESCORT FOR LESS THAN ONE MINUTE WHEN HE WAS ENCOUNTERED BY THE SECURITY FORCE MEMBER. THE ESCORT HAS BEEN RETRAINED IN SECURITY REQUIREMENTS. IN ADDITION, MOVES HAVE BEEN TAKEN TO MINIMIZE THE NUMBER OF VISITORS REQUIRING ACCESS.

[115] FARLEY 1 DOCKET 50-348 LER 87-020
UNAUTHORIZED ACCESS GAINED TO THE MAIN CONTROL ROOM.
EVENT DATE: 101687 REPORT DATE: 110387 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: FARLEY 2 (PWR)

(NSIC 206934) AT 0742 ON 10-15-87, A CONTRACTOR EMPLOYEE GAINED UNAUTHORIZED ACCESS TO THE MAIN CONTROL ROOM. SINCE THE CONTROL ROOM IS A VITAL AREA, THIS REPORT IS BEING SUBMITTED IN ACCORDANCE WITH 10CFR73.71(B). THE CONTRACTOR AND THREE FARLEY NUCLEAR PLANT MECHANICS WERE PROCEEDING TO THE UNIT 2 AUXILIARY FEEDWATER PUMP AREA WHICH THEY WERE ALL AUTHORIZED TO ENTER. THE MECHANICS NEEDED TO ENTER THE CONTROL ROOM TO OBTAIN A KEY TO THE PUMP AREA. THE MECHANICS USED THEIR KEY-CARDS TO ENTER THE CONTROL ROOM. THE CONTRACTOR ALSO USED HIS KEY-CARD, BUT HE WAS NOT AUTHORIZED TO ENTER THE CONTROL ROOM. ACCESS TO THE CONTROL ROOM FOR THE CONTRACTOR WAS DENIED WHICH CAUSED AN ALARM IN THE SECURITY CENTRAL ALARM STATION. HOWEVER, DESPITE THE FACT THAT HE WAS NOT AUTHORIZED, THE CONTRACTOR FOLLOWED THE MECHANICS INTO THE CONTROL ROOM. HE REMAINED WITH THE MECHANICS WHILE IN THE CONTROL ROOM AREA. SECURITY PERSONNEL RESPONDED TO THE ALARM. A SEARCH REVEALED NO UNAUTHORIZED PERSONNEL IN THE CONTROL ROOM. THE CONTRACTOR WAS LOCATED IN THE AUXILIARY FEEDWATER PUMP AREA AND ESCORTED OUT OF THE AUXILIARY BUILDING. NO SAFETY SYSTEMS WERE AFFECTED OR THREATENED. THE APPROPRIATE PERSONNEL HAVE BEEN COUNSELED CONCERNING THIS EVENT.

[116] FERMI 2 DOCKET 50-341 LER 87-041 REV 01
UPDATE ON ACCIDENT RANGE MONITOR IS INOPERABLE BECAUSE OF DELETED SYSTEM PARAMETERS.
EVENT DATE: 032987 REPORT DATE: 103087 NSSS: GE TYPE: BWR
VENDOR: EBERLINE INSTRUMENT CORP.

(NSIC 206877) A REVIEW OF THE STANDBY GAS TREATMENT SYSTEM (SGTS) DIVISION II

ACCIDENT RANGE MONITOR (AXM) CHANNEL PARAMETERS WAS CONDUCTED. IT WAS REVEALED THAT THE CHANNELS PARAMETERS WERE SET AT DEFAULT VALUES FOR CALIBRATION CONSTANTS AND THE SET POINTS. THIS RESULTED IN THE MONITOR BEING INOPERABLE WHICH IS BEING REPORTED AS A VOLUNTARY LICENSEE EVENT REPORT. THE CAUSE OF THIS EVENT WAS INADEQUATE SYSTEM TURN OVER FOLLOWING PREOPERATIONAL TESTING. THIS RESULTED IN THE IMPLEMENTATION OF INADEQUATE SURVEILLANCE AND TEST/PREVENTATIVE MAINTENANCE PROCEDURES. A DAILY CHANNEL CHECK IS BEING PERFORMED FOR THE SYSTEM PARTICULATE, IODINE AND NOBLE GAS (SPING) AND AXM UNITS. THE PRACTICE WILL BE CONTINUED UNTIL APPROPRIATE CHANGES ARE MADE TO THE PROCEDURES. THE SPING AND AXM PROCEDURES WILL BE CHANGED TO CONTROL AND VERIFY MONITOR PARAMETER EDITING. A TECHNICAL SPECIFICATION INTERPRETATION WAS MADE TO CLARIFY SGTS OPERABILITY VERSUS AXM RADIATION MONITOR AVAILABILITY.

[117] FERMI 2 DOCKET 50-341 LER 87-019 REV 02
UPDATE ON MISSED SURVEILLANCE OF STANDBY GAS TREATMENT SYSTEM DUE TO INCORRECT SCHEDULING.
EVENT DATE: 052387 REPORT DATE: 102987 NSSS: GE TYPE: BWR

(NSIC 206798) IT WAS DISCOVERED THAT SURVEILLANCE PROCEDURE 24.404.06 "STANDBY GAS TREATMENT SYSTEM MANUAL ACTUATION PUMP TEST" WAS INCORRECTLY SCHEDULED FOR ONCE EVERY 24 MONTHS INSTEAD OF AT LEAST ONCE EVERY 18 MONTHS. DUE TO THIS ERROR, THE TEST INTERVAL, INCLUDING THE 25% MAXIMUM ALLOWABLE EXTENSION, WAS EXCEEDED BY APPROXIMATELY 41 DAYS. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR. THE TEST FREQUENCY FOR THIS SURVEILLANCE WAS INCORRECTLY INPUT INTO THE SURVEILLANCE SCHEDULING PROGRAM. THE PROGRAM HAS BEEN REVISED TO REFLECT THE CORRECT TEST FREQUENCY. CORRECTIVE ACTION INCLUDED A REVIEW OF TECH SPECS AGAINST THE APPROPRIATE PROCEDURES AND THE SURVEILLANCE SCHEDULING PROGRAM. IN ADDITION, A TECH SPEC - PROCEDURE CROSS REFERENCE REPORT WAS DEVELOPED. A DETAILED TECHNICAL REVIEW OF SURVEILLANCE PROCEDURES WILL BE MADE. THE INOPERABILITY OF THE SYSTEM RESULTED IN OPERATIONS WHICH WERE NOT IN ACCORDANCE WITH TECH SPECS. HOWEVER, SUCCESSFUL COMPLETION OF THE SURVEILLANCE PROCEDURE VERIFIED THE CONTINUED AVAILABILITY OF THE SYSTEM.

[118] FERMI 2 DOCKET 50-341 LER 87-031 REV 01
UPDATE ON REACTOR SCRAM DUE TO FALSE HIGH TURBINE BEARING VIBRATION.
EVENT DATE: 072087 REPORT DATE: 102287 NSSS: GE TYPE: BWR
VENDOR: WEIDMULLER TERMINATIONS INC.

(NSIC 206799) ON JULY 20, 1987 AT 1813 HOURS, A HIGH VIBRATION SIGNAL FOR THE MAIN TURBINE GENERATOR NUMBER ONE BEARING WAS RECEIVED. DUE TO THE SIGNAL, A TURBINE CONTROL VALVE FAST CLOSURE OCCURRED AND A REACTOR SCRAM RESULTED. DURING THE REACTOR SCRAM, ALL SAFETY SYSTEMS FUNCTIONED NORMALLY. THIS ACTUATION OF THE REACTOR PROTECTION SYSTEM WAS THE RESULT OF AN ERRONEOUS HIGH VIBRATION SIGNAL FROM A FAULTY SHAFT RIDER SENSOR CONNECTION. AS AN INTERIM MEASURE TO PREVENT SPONTANEOUS TRIPS OF THE MAIN TURBINE GENERATOR DUE TO FAULTY VIBRATION DETECTION EQUIPMENT, THE OPERATORS WERE DIRECTED TO TRIP THE MAIN TURBINE GENERATOR MANUALLY ON HIGH VIBRATION INDICATION. THE ALARM RESPONSE PROCEDURE FOR HIGH VIBRATION SIGNALS WAS REVISED TO PROVIDE CLEAR INSTRUCTION TO CONTROL ROOM PERSONNEL. A DESIGN CHANGE HAS BEEN IMPLEMENTED TO REQUIRE TWO HIGH VIBRATION SIGNALS TO TRIP THE UNIT. THIS CHANGE WAS IMPLEMENTED PRIOR TO PLANT RESTART ON OCTOBER 9, 1987.

[119] FERMI 2 DOCKET 50-341 LER 87-034 REV 01
UPDATE ON INADEQUATE SCHEDULING OF FIRE BRIGADE PRACTICE DUE TO MISINTERPRETATION OF THE REQUIREMENT.
EVENT DATE: 073087 REPORT DATE: 100987 NSSS: GE TYPE: BWR

(NSIC 206636) ON JULY 30, 1987, IT WAS DISCOVERED THAT FIRE BRIGADE PRACTICE

SESSIONS WERE NOT BEING HELD AT INTERVALS OF ONE YEAR AS REQUIRED BY THE FIRE BRIGADE PROCEDURE. INSTEAD FIRE BRIGADE PRACTICE SESSIONS WERE BEING HELD ON A ONCE PER CALENDAR YEAR BASIS FROM APRIL TO OCTOBER. THE CAUSE OF THIS EVENT WAS A MISINTERPRETATION OF THE REQUIREMENTS FOR FIRE BRIGADE TRAINING SCHEDULE. TO PREVENT RECURRENCE, THE FIRE PROTECTION PROGRAM WILL BE REVISED TO INCLUDE A TWENTY FIVE PERCENT EXTENSION INTERVAL. THIS REVISION HAS BEEN SUBMITTED TO THE NUCLEAR REGULATORY COMMISSION FOR REVIEW AND APPROVAL.

[120] FERMI 2 DOCKET 50-341 LER 87-042
CONTROL CENTER HEATING VENTILATION AND AIR CONDITIONING SYSTEM SHIFTS TO THE CHLORINE MODE WITH NO CHLORINE PRESENT.
EVENT DATE: 090287 REPORT DATE: 100287 NSSS: GE TYPE: BWR

(NSIC 206626) ON SEPTEMBER 2, 1987 AT 0557 HOURS, THE CONTROL CENTER HEATING, VENTILATION AND AIR CONDITIONING SYSTEM SHIFTED FROM NORMAL OPERATION TO THE CHLORINE MODE. THE ACTUATION OF THIS ENGINEERED SAFETY FEATURE IS BELIEVED TO HAVE BEEN CAUSED BY THE USE OF A TWO-WAY RADIO ON THE FOURTH FLOOR OF THE AUXILIARY BUILDING. IT IS BELIEVED THE TWO-WAY RADIO AFFECTED THE CHLORINE DETECTOR ON THE FIFTH FLOOR OF THE AUXILIARY BUILDING CAUSING IT TO ACTUATE. HOWEVER FURTHER INVESTIGATION IS NEEDED BEFORE AN EXACT CAUSE CAN BE DETERMINED. A TEST WAS CONDUCTED WITH THE SUSPECT TWO-WAY RADIO TO DETERMINE IF IT COULD ACTUATE THE CHLORINE DETECTOR FROM THE FOURTH FLOOR OF THE AUXILIARY BUILDING. HOWEVER, THIS EVENT WAS NOT ABLE TO BE DUPLICATED. FURTHER TESTING WILL BE COMPLETED IN TWO WEEKS.

[121] FERMI 2 DOCKET 50-341 LER 87-044
TECHNICAL SPECIFICATIONS FOR PRIMARY CONTAINMENT INTEGRITY NOT FULLY IMPLEMENTED BECAUSE OF PROCEDURAL INADEQUACIES.
EVENT DATE: 090387 REPORT DATE: 100387 NSSS: GE TYPE: BWR

(NSIC 206637) ON SEPTEMBER 3, 1987 AT 1729 HOURS, IT WAS CONCLUDED THAT A DEFICIENCY IN THE SURVEILLANCE OF TEST, VENT AND DRAIN (TVD) CONNECTIONS EXISTED. TECHNICAL SPECIFICATION 4.6.1.1.B REQUIRES ALL PRIMARY CONTAINMENT PENETRATIONS NOT CAPABLE OF BEING CLOSED BY OPERABLE AUTOMATIC ISOLATION VALVES BE VERIFIED CLOSED AT LEAST ONCE PER 31 DAYS. CONTRARY TO THIS, 82 TVDS AND 5 BONNET TAPS WERE FOUND NOT TO BE VERIFIED WITHIN EXISTING SURVEILLANCE PROCEDURES. NO DEGRADATION OF THE CONTAINMENT INTEGRITY OCCURRED AS A RESULT OF THIS DEFICIENCY. THE APPROPRIATE SURVEILLANCE PROCEDURES HAVE BEEN REVISED TO INCLUDE THE TVDS AND THE BONNET TAPS.

[122] FERMI 2 DOCKET 50-341 LER 87-045
LOW PRESSURE COOLANT INJECTION SWING BUS DESIGN FLAW IDENTIFIED BY PERSONNEL ERROR.
EVENT DATE: 090887 REPORT DATE: 100887 NSSS: GE TYPE: BWR

(NSIC 206638) ON SEPTEMBER 8, 1987, AN OPERATOR, REMOVED A FUSE WHICH DEENERGIZED THE DC CONTROL POWER TO THE BUS 72C WHILE ATTEMPTING TO DEENERGIZE A COMPONENT FOR MAINTENANCE ACTIVITIES. THIS BUS IS THE NORMAL FEED TO THE LOW PRESSURE COOLANT INJECTION (LPCI) SWING BUS. THE LOSS OF DC CONTROL POWER RESULTED IN THE LOSS OF THE POWER SUPPLY TO THE SWING BUS AND THUS TO THE LPCI LOOP SELECTION VALVES. AN OPERATOR PREPARED A MAINTENANCE PROTECTION TAG OUT PACKAGE THAT IDENTIFIED THE WRONG FUSE TO REMOVE. WHEN AN OPERATOR IN THE FIELD REMOVED THE FUSE, DC CONTROL POWER WAS LOST TO BUS 72C POSITION 3C AND THAT DEENERGIZED THE LPCI SWING BUS. SUBSEQUENTLY, THE DESIGN WAS REVIEWED BY NUCLEAR ENGINEERING AND IT WAS DETERMINED THAT THE DC CONTROL CIRCUITRY FOR BUS 72C EQUIPMENT WAS INADEQUATE. THE LPCI OPERATION COULD BE PREVENTED BY EITHER OF TWO INDEPENDENT FAILURES EXTERNAL TO THE SWING BUS. THE DC CONTROL CIRCUITRY FOR THE BUS 72C HAS BEEN REDESIGNED TO MEET THE PLANTS DESIGN BASE.

[123] FERMI 2 DOCKET 50-341 LER 87-047
SURVEILLANCE NOT COMPLETED AS REQUIRED FOR CONTROL CENTER HEATING VENTILATION AND AIR CONDITIONING OPERABILITY.
EVENT DATE: 091787 REPORT DATE: 101787 NSSS: GE TYPE: BWR

(NSIC 206639) ON SEPTEMBER 18, 1987 AT 0200 HOURS, IT WAS DISCOVERED THE DIVISION I OF CONTROL CENTER HEATING VENTILATING AND AIR CONDITIONING CHILLER PUMP AND VALVE OPERABILITY SURVEILLANCE, 24.413.01 HAD NOT BEEN COMPLETED BY THE CRITICAL COMPLETION DATE. THE CAUSE OF THIS EVENT WAS INADEQUATE PRACTICES AND TRAINING OF THE OPERATORS IN TRACKING TECHNICAL SPECIFICATION SURVEILLANCES. THE OPERATORS WILL BE REQUIRED TO ENTER EXPIRING TECHNICAL SPECIFICATION SURVEILLANCES FOR OUT OF SERVICE EQUIPMENT IN THE OUT OF SPECIFICATION LOG. THIS WILL ENSURE THE TECHNICAL SPECIFICATION SURVEILLANCES ARE COMPLETED BEFORE EQUIPMENT IS RETURNED TO SERVICE. DISCUSSIONS WILL BE HELD BETWEEN OPERATIONS PERSONNEL AND SURVEILLANCE COORDINATION PERSONNEL TO RESOLVE PROBLEMS IN TRACKING TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS.

[124] FERMI 2 DOCKET 50-341 LER 87-049 REV 01
UPDATE ON DEFICIENCIES IN THE FIRE PROTECTION PROGRAM DISCOVERED DURING AN INSPECTION.
EVENT DATE: 092587 REPORT DATE: 110687 NSSS: GE TYPE: BWR

(NSIC 206959) ON SEPTEMBER 25, 1987 THREE DEFICIENCIES IN THE FIRE PROTECTION PROGRAM WERE IDENTIFIED. FIRST, FIRE BRIGADE REQUALIFICATION TRAINING WAS NOT BEING PERFORMED ON THE SCHEDULE COMMITTED TO IN THE UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR). A FIRE PROTECTION VALVE WAS RESTORED TO ITS CORRECT POSITION BUT NOT LOCKED AFTER A MAINTENANCE INSPECTION. FINALLY, A CONTROL ROOM OPERATOR INDICATED TO A NUCLEAR REGULATORY COMMISSION INSPECTOR THAT HE WOULD TAKE ACTIONS CONTRARY TO APPROVED PROCEDURES ON ACTIVATING THE FIRE BRIGADE UPON RECEIPT OF A FIRE ALARM. THE FIRE BRIGADE TRAINING PROGRAM WAS REVISED TO INCLUDE THE REQUALIFICATION TRAINING ON THE SCHEDULE COMMITTED TO IN THE UFSAR. THE VALVE WAS LOCKED AND PROCEDURE COVERING THE RESTORATION OF EQUIPMENT WILL BE REVISED. DISCUSSIONS WILL BE HELD IN TRAINING CLASSES ON THE APPROPRIATE ACTIONS TO TAKE UPON RECEIPT OF A FIRE ALARM.

[125] FERMI 2 DOCKET 50-341 LER 87-050
UNIDENTIFIED FEEDWATER PIPING DESIGN TEMPERATURE LIMIT EXCEEDED.
EVENT DATE: 092687 REPORT DATE: 102687 NSSS: GE TYPE: BWR

(NSIC 206760) ON AUGUST 21, 1987 IT WAS DISCOVERED THE REACTOR WATER CLEANUP (RWCU) SYSTEM OPERATING PROCEDURE ALLOWED THE RWCU HEAT EXCHANGER TO BE PLACED IN BYPASS WHEN THE REACTOR COOLANT TEMPERATURE COULD REACH 544 DEGREES FAHRENHEIT. THE RETURN FLOW PASSES THROUGH LOOP B OF THE FEEDWATER SYSTEM BEFORE RETURNING TO THE REACTOR VESSEL. THE FEEDWATER SYSTEM IS AMERICAN SOCIETY OF MECHANICAL ENGINEERS (ASME) CLASS 1 PIPING RATED FOR A MAXIMUM TEMPERATURE OF 450 DEGREES FAHRENHEIT. A REVIEW OF THE OPERATIONS LOGS WAS MADE. IT WAS CONCLUDED THAT THERE WERE 18 HOURS AND 5 MINUTES WHEN THE FEEDWATER PIPING WAS OR COULD HAVE BEEN SUBJECTED TO TEMPERATURES GREATER THEN ITS DESIGN LIMIT. THE RWCU SYSTEM OPERATING PROCEDURE HAS BEEN REVISED TO PRECLUDE VIOLATION OF THIS LIMITATION. OTHER ASME SYSTEMS TEMPERATURE DESIGN LIMITS WERE REVIEWED. THE DESIGN LIMITATIONS FOR TEMPERATURE HAD NOT BEEN EXCEEDED.

[126] FERMI 2 DOCKET 50-341 LER 87-048
CHANNEL CHECK NOT PERFORMED FOR DRYWELL INSTRUMENTS BECAUSE OF AN INADEQUATE PROCEDURE.
EVENT DATE: 100887 REPORT DATE: 110787 NSSS: GE TYPE: BWR

(NSIC 206800) ON OCTOBER 8, 1987 AT 1400 HOURS, IT WAS DISCOVERED THAT THE

TECHNICAL SPECIFICATION REQUIREMENT TO PERFORM A CHANNEL CHECK AT LEAST ONCE PER 12 HOURS FOR REACTOR PROTECTION SYSTEM (RPS) DRYWELL HIGH PRESSURE INSTRUMENTATION WAS NOT INCLUDED IN ANY SURVEILLANCE PROCEDURE. THIS EVENT WAS CAUSED BY AN INCOMPLETE SURVEILLANCE PROCEDURE. THE "SHIFTLY, DAILY, WEEKLY AND SITUATION REQUIRED SURVEILLANCES" PROCEDURE DID NOT INCLUDE THE CHANNEL CHECK FOR RPS DRYWELL HIGH PRESSURE INSTRUMENTS. THE CORRECTIVE ACTION WAS TO REVISE THE SURVEILLANCE PROCEDURE TO INCLUDE THIS REQUIREMENT. ALL TECHNICAL SPECIFICATION SURVEILLANCE PROCEDURES ARE SCHEDULED TO BE REVIEWED BY MARCH 31, 1988 UNDER THE SURVEILLANCE PROCEDURE ENHANCEMENT PROGRAM.

[127] FITZPATRICK DOCKET 50-333 LER 87-013
REACTOR CORE ISOLATION COOLING SYSTEM, STEAM SUPPLY ISOLATION DUE TO SPURIOUS ANALOG TRANSMITTER TRIP UNIT GROSS FAILURE ALARM/TRIP.
EVENT DATE: 090587 REPORT DATE: 100587 NSSS: GE TYPE: BWR
VENDOR: ROSEMOUNT, INC.

(NSIC 206618) AT APPROXIMATELY 16:55 ON 9/5/87 AND WHILE AT 98.2% REACTOR THERMAL POWER, A SPURIOUS TRIP UNIT GROSS FAILURE ALARM AND AUTO ISOLATION OF THE REACTOR CORE ISOLATION COOLING (RCIC) (EIIS CODE BN) SYSTEM OCCURRED. THE INITIATING SIGNAL FOR THIS EVENT CAME FROM THE RCIC HIGH STEAM LINE FLOW (HSLF) ANALOG TRANSMITTER TRIP SYSTEM (ATTS) MASTER TRIP UNIT (MTU). THE CAUSE OF THE SPURIOUS GROSS FAILURE ON THE MTU COULD NOT BE DETERMINED. AFTER COMPLETION OF APPLICABLE FUNCTIONAL AND CALIBRATION TESTS, THE SYSTEM WAS RETURNED TO NORMAL CONFIGURATION. AT APPROXIMATELY 21:00 HOUR ON 9/5/87 THE RCIC SYSTEM AGAIN ISOLATED DUE TO A SPURIOUS GROSS FAILURE TRIP FROM THE SAME MTU. TROUBLESHOOTING BY THE INSTRUMENT AND CONTROLS' TECHNICIAN YIELDED NO CAUSE. AT THIS TIME IT WAS DETERMINED TO DECLARE RCIC ADMINISTRATIVELY INOPERABLE, CONNECT A STRIP CHART RECORDER TO THE TRANSMITTER OUTPUT AND MONITOR FOR 24 HOURS. NO GROSS FAILURE OCCURRED AFTER MONITORING WAS INITIATED. AT 13:30 HOURS ON 9/7/87 RCIC WAS DECLARED OPERABLE AND RETURNED TO SERVICE. ON 9/10/87 DURING AN UNSCHEDULED PLANT SHUTDOWN (LER-87-012) THE ELECTRONIC TRANSMITTER (13-DPT-83) AND MASTER TRIP UNIT (13-MTU-283) WERE REPLACED. THEY ARE BEING SENT TO THE MANUFACTURER FOR ANALYSIS. THE LER NUMBER OF A SIMILAR PREVIOUS EVENT IS LER 85-028.

[128] FITZPATRICK DOCKET 50-333 LER 87-014
ACTUATION OF 4 KV EMERGENCY BUS DEGRADED VOLTAGE PROTECTION DURING LOAD TRANSFER.
EVENT DATE: 091287 REPORT DATE: 100987 NSSS: GE TYPE: BWR

(NSIC 206619) DURING A NORMAL PLANT START-UP, WITH THE REACTOR POWER AT 28 PERCENT, AN AUTOMATIC INITIATION OF THE EMERGENCY BUS DEGRADED VOLTAGE PROTECTION SYSTEM (JE) OCCURRED. THE EVENT TOOK PLACE DURING THE TRANSFER OF STATION ELECTRICAL LOADS FROM THE RESERVE STATION TRANSFORMERS (OFF-SITE SOURCE) TO THE NORMAL STATION TRANSFORMER (MAIN GENERATOR SOURCE) (TB, EL, FK). THE ADDITION OF THE FOURTH AND FINAL ELECTRICAL BUS TO THE NORMAL STATION TRANSFORMER CAUSED VOLTAGE ON THE 4160 VOLT EMERGENCY BUSES (EB) TO DROP BELOW THE DEGRADED VOLTAGE SETPOINT (3780 VOLTS WHICH IS 90% OF NOMINAL BUS VOLTAGE. AFTER A NINE SECOND TIME DELAY THE EMERGENCY DIESEL GENERATORS (EK) AUTOMATICALLY STARTED. THE CONTROL ROOM OPERATOR MANUALLY RESTORED THE BUSES TO NORMAL VOLTAGE USING THE LOAD TAP CHANGER ON THE NORMAL STATION TRANSFORMER. THIS ACTION RESET THE DEGRADED VOLTAGE LOGIC AND PREVENTED THE INTERRUPTION OF POWER TO THE EMERGENCY BUSES. THE EMERGENCY DIESEL GENERATORS WERE RETURNED TO THE STANDBY CONDITION PER NORMAL OPERATING PROCEDURE. THERE WERE NO SYSTEM OR COMPONENT FAILURES DURING THE EVENT AND ALL EQUIPMENT PERFORMED NORMALLY. NO OTHER ENGINEERED SAFETY FEATURE SYSTEMS INITIATED. AN ENGINEERING RE- EVALUATION OF THE SYSTEM SETPOINTS SHALL BE PERFORMED TO DETERMINE IF IT IS POSSIBLE TO ACHIEVE GREATER MARGIN DURING ANTICI PATED ELECTRICAL SYSTEM TRANSIENTS. SIMILAR EVENT: 333/87-009.

[129] FITZPATRICK DOCKET 50-333 LER 87-015
HIGH PRESSURE COOLANT INJECTION INOPERABLE DUE TO UNSTABLE SUPPRESSION CHAMBER LEVEL SWITCH.
EVENT DATE: 091687 REPORT DATE: 101687 NSSS: GE TYPE: BWR
VENDOR: ROBERTSHAW CONTROLS COMPANY

(NSIC 206620) ON 9/16/87, WITH REACTOR POWER AT 99%.AND WHILE PERFORMING AN INCREASED FREQUENCY TEST OF THE SUPPRESSION CHAMBER WATER LEVEL SWITCH, 23-LS-91B WAS FOUND TO HAVE EXCEEDED THE OPERATING TECHNICAL SPECIFICATION (TS) TABLE 3.2-2 LIMIT OF < 6.0 INCHES ABOVE NORMAL WATER LEVEL BY 0.47 INCHES. REDUNDANT INSTRUMENT TRIP CHANNEL (23-LS-91A) WAS OPERABLE AND TESTED SATISFACTORILY. TO FACILITATE REMOVAL OF 23-LS-91B FROM THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM PUMP SUCTION TRANSFER LOGIC, HPCI WAS DECLARED INOPERABLE AND TS REQUIRED SYSTEMS TESTED FOR OPERABILITY AND FOUND SATISFACTORILY. CAUSE OF SWITCH INSTABILITY WAS A SLIGHT AMOUNT OF MATERIAL DEPOSITED ON SWITCH INTERNAL FLOAT MECHANISM. CORRECTIVE ACTION WAS TO CLEAN MECHANISM AND BENCH TEST SWITCH FOR PROPER OPERATION. AFTER RETURN TO SERVICE, 3 CONSECUTIVE WEEKLY TESTS HAVE FOUND SWITCH OPERATION TO BE SATISFACTORY. LER'S OF A SIMILAR EVENT AT FITZPATRICK - NONE

[130] FITZPATRICK DOCKET 50-333 LER 87-017
REACTOR LOW LEVEL SCRAM FOLLOWING FEED PUMP TURBINE TRIP ON HIGH VIBRATION.
EVENT DATE: 092487 REPORT DATE: 102387 NSSS: GE TYPE: BWR

(NSIC 206796) ON 9/24/87 AT 0738, WITH THE PLANT AT 100% POWER, REACTOR SCRAMMED ON LOW WATER LEVEL FOLLOWING A REACTOR FEED PUMP (RFP) A TRIP ON HIGH VIBRATION. ALL SYSTEMS WERE IN A NORMAL LINE UP EXCEPT THE RECIRCULATION SYSTEM SCOOP TUBE POSITIONERS WERE LOCKED UP DUE TO CONTROL PROBLEMS. THE CONTROL ROOM OPERATOR RESET SCOOP TUBE POSITIONERS ALLOWING RECIRCULATION SYSTEM RUNBACK TO 44% SPEED WHICH REDUCED STEAM FLOW TO WITHIN THE CAPACITY OF ONE RFP. OPERATION WITH SCOOP TUBE POSITIONERS LOCKED MAY HAVE INCREASED THE MAGNITUDE OF THE LEVEL TRANSIENT. A DEFECTIVE VIBRATION DETECTOR SHAFT RIDER ON RFP A TURBINE INBOARD BEARING WAS DISCOVERED. PRIOR TO RETURNING RFP A TO SERVICE THE OIL SYSTEM, TRIP FUNCTIONS AND VIBRATIONS WERE THOROUGHLY EVALUATED. RFP A TURBINE INBOARD BEARING AND SHAFT RIDER WERE REPLACED. LONG TERM ACTION IS TO MODIFY THE RECIRCULATION SPEED CONTROL SYSTEM TO ALLOW OPERATION WITH SCOOP TUBES UNLOCKED AND RFP TURBINE BEARINGS VIBRATION DETECTORS WILL BE ON A PREVENTIVE MAINTENANCE PROGRAM. LER 84-009, 84-010, 84-023 AND 87-008 ARE SIMILAR.

[131] FT. CALHOUN 1 DOCKET 50-285 LER 87-025
DG-2 SHUTDOWN ON HIGH COOLANT TEMPERATURE DUE TO DAMPER MALFUNCTION.
EVENT DATE: 092387 REPORT DATE: 102387 NSSS: CE TYPE: PWR

(NSIC 206786) ON SEPTEMBER 23, 1987, AT 0906 CDT, FOLLOWING REPAIR OF THE EXHAUST PIPE, DIESEL GENERATOR NO. 2 (DG-2) WAS STARTED AND LOADED PER OPERATING INSTRUCTION OI-DG-2 AS REQUIRED BY SURVEILLANCE TEST ST-ESF-6. APPROXIMATELY 14 MINUTES INTO THE TEST, DG-2 AUTOMATICALLY SHUTDOWN DUE TO HIGH COOLANT TEMPERATURE. INVESTIGATIONS REVEALED THAT THE AIR OPERATED EXHAUST DAMPER FOR THE DIESEL GENERATOR RADIATOR MAY NOT HAVE FULLY OPENED AUTOMATICALLY AS DESIGNED WHEN THE DIESEL WAS RUNNING, THUS RESTRICTING THE REQUIRED AIR FLOW THROUGH THE RADIATOR. THE CAUSE OF THE DAMPER MALFUNCTION WAS POSTULATED TO BE THE PRESENCE OF RESIDUE CAUSING THE PILOT VALVE THAT DIRECTS AIR FLOW TO SOMETIMES STICK. ON JULY 6, WATER WAS INTRODUCED INTO THE INSTRUMENT AIR SYSTEM DURING THE PERFORMANCE OF A SURVEILLANCE TEST ON THE FIRE PROTECTION SYSTEM DRY PIPE VALVE FOR THE DIESEL GENERATOR ROOMS. THE WATER INTRUSION WAS LIMITED TO THE AUXILIARY BUILDING AT OR BELOW ELEVATION 1025. AN EXTENSIVE PROGRAM WAS UNDERTAKEN (IN JULY) AND WAS REPEATED AS NECESSARY DURING THE MONTHS OF AUGUST AND SEPTEMBER TO BLOWDOWN AIR OPERATED DEVICES INCLUDING AIR OPERATED VALVES AND TO CYCLE THOSE VALVES AS ALLOWED DURING POWER OPERATION. AFTER THE TRIP OF DG-2, THE PILOT

VALVE WAS INSPECTED AND CLEANED AND THE ACCUMULATOR DRAINED. SIMILAR ACTIONS WERE TAKEN FOR DG-1. TO PREVENT A POSSIBLE RECURRENCE, AN EXTENSIVE CORRECTIVE ACTION PROGRAM IS IN PROGRESS.

[132] FT. CALHOUN 1 DOCKET 50-285 LER 87-026
FAILURE TO CONTROL A VERY HIGH RADIATION AREA.
EVENT DATE: 101487 REPORT DATE: 102887 NSSS: CE TYPE: PWR

(NSIC 206787) FORT CALHOUN STATION TECHNICAL SPECIFICATION 5.11.2 REQUIRES THAT DOORS TO VERY HIGH RADIATION AREAS BE LOCKED. ON OCTOBER 14, 1987 AT 1430 HOURS CDT, THE DOOR TO ROOM 11 (A VERY HIGH RADIATION AREA) IN THE AUXILIARY BUILDING WAS FOUND UNLOCKED BY THE LICENSEE. ON SEPTEMBER 9, 1987, THE DOOR TO ROOM 5 (ALSO A VERY HIGH RADIATION AREA) WAS FOUND UNLOCKED BY THE NRC. CORRECTIVE ACTION HAS INCLUDED REPLACEMENT AND/OR MODIFICATION TO DOOR HARDWARE; CHANGES TO EXISTING PROCEDURES; AND INCREASING OPPD PERSONNEL ATTENTION TO THE PROCEDURES FOR CONTROLLING VERY HIGH RADIATION AREAS.

[133] GRAND GULF 1 DOCKET 50-416 LER 87-015
REACTOR WATER CLEANUP (RWCU) SYSTEM ISOLATION.
EVENT DATE: 091787 REPORT DATE: 101687 NSSS: GE TYPE: BWR

(NSIC 206630) ON SEPTEMBER 17, 1987 AT APPROXIMATELY 0212, THE REACTOR WATER CLEANUP (RWCU) SYSTEM ISOLATED ON AN EXPECTED HIGH DIFFERENTIAL FLOW SIGNAL RECEIVED WHILE PERFORMING STARTUP ACTIVITIES ON RWCU "B". THE SYSTEM WAS IMMEDIATELY RESTORED TO OPERATION. SYSTEM OPERATING INSTRUCTION (SOI) 04-1-01-G33-1-TEMP-6 WILL BE CHANGED TO REQUIRE THE USE OF AN EXISTING BYPASS SWITCH WHICH WILL BYPASS THE RWCU ISOLATION LOGIC DURING STARTUP IN THE PREPUMP MODE OF OPERATION. THE PREPUMP MODE IS BEING USED UNTIL THE POWER SUPPLY TO CERTAIN CONTAINMENT ISOLATION VALVES ARE MODIFIED AS DESCRIBED IN LER 87-011. THIS MODIFICATION IS SCHEDULED FOR THE SECOND REFUELING OUTAGE.

[134] GRAND GULF 1 DOCKET 50-416 LER 87-016
AREA RADIATION SURVEY EXCEEDED LIMITING CONDITION FOR OPERATION TIME LIMIT DUE TO PERSONNEL ERROR.
EVENT DATE: 100187 REPORT DATE: 102887 NSSS: GE TYPE: BWR

(NSIC 206836) ON OCTOBER 1, 1987 AT APPROXIMATELY 1600, OPERATIONS PERSONNEL DISCOVERED THAT AN AREA RADIATION SURVEY HAD EXCEEDED THE TECHNICAL SPECIFICATION LIMITING CONDITION FOR OPERATION (LCO) REQUIRED TWENTY- FOUR HOUR FREQUENCY. THE INITIAL SURVEY WAS PERFORMED ON SEPTEMBER 30, 1987 AT 1530. THE SECOND SURVEY WAS REQUIRED TO BE PERFORMED WITHIN TWENTY-FOUR HOURS; HOWEVER, IT WAS NOT PERFORMED UNTIL 1630 ON OCTOBER 1, 1987. THE INCIDENT OCCURRED DUE TO THE FAILURE OF THE OPERATIONS SHIFT SUPERVISOR TO CONTACT THE HEALTH PHYSICS SUPERVISOR AND VERIFY COMPLETION OF THE LCO ACTION IN A TIMELY MANNER. ANOTHER CONTRIBUTING FACTOR WAS THE HEALTH PHYSICS SUPERVISOR MISUNDERSTANDING THAT THE SURVEY WAS REQUIRED TO MEET AN LCO ACTION STATEMENT. THE INCIDENT WAS INVESTIGATED BY THE MANAGEMENT REVIEW COMMITTEE. OPERATIONS AND HEALTH PHYSICS PERSONNEL RECEIVED COUNSELLING EMPHASIZING SECTION INTERFACE. AN LCO TRACKING BOARD HAS BEEN PROVIDED FOR USE IN THE HEALTH PHYSICS LABORATORY. HEALTH PHYSICS SUPERVISORS ARE NOW REQUIRED TO REVIEW AND SIGN ACKNOWLEDGEMENT OF LCO STATUS AT THE BEGINNING OF EACH SHIFT TO ENSURE APPROPRIATE ACTIONS ARE PERFORMED.

[135] GRAND GULF 1 DOCKET 50-416 LER 87-017
DRYWELL INTEGRITY INSPECTION NOT COMPLETED DUE TO PROCEDURAL INADEQUACIES.
EVENT DATE: 100587 REPORT DATE: 110487 NSSS: GE TYPE: BWR

(NSIC 207114) ON OCTOBER 5, 1987 A SERI QUALITY PROGRAMS REVIEW REVEALED THAT THE

VISUAL INSPECTION OF THE DRYWELL INTERIOR AND EXTERIOR SURFACES REQUIRED BY TECHNICAL SPECIFICATION 4.6.2.4.1 MAY NOT HAVE BEEN SUFFICIENTLY PERFORMED DURING THE SHUTDOWN FOR THE TYPE A CONTAINMENT INTEGRATED LEAKAGE RATE TEST (ILRT) WHICH OCCURRED IN NOVEMBER 1985. THE PROCEDURE USED FOR THE INSPECTION INCLUDED ONLY AREAS BELOW THE SUPPRESSION POOL LEVEL. NO PROCEDURE WAS SPECIFICALLY WRITTEN TO PERFORM A STRUCTURAL INTEGRITY INSPECTION OF SURFACE AREAS ABOVE THE SUPPRESSION POOL LEVEL. HOWEVER, THE "FIRE RATED ASSEMBLY VISUAL INSPECTION" SURVEILLANCE IS PERFORMED AT 18 MONTH INTERVALS AND INSPECTS THESE AREAS. THE EXTENT OF THIS INSPECTION MEETS THE EXTENT OF INSPECTION REQUIRED FOR THE STRUCTURAL INTEGRITY INSPECTION. A DRYWELL BYPASS LEAKAGE TEST WAS CONDUCTED SUBSEQUENT TO THE ILRT AND CONFIRMED THE STRUCTURAL INTEGRITY OF THE DRYWELL. THEREFORE, NO IMMEDIATE COMPENSATORY ACTIONS WERE NECESSARY. ALL CONTAINMENT TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS WERE REVIEWED WHICH CONFIRMED THAT THE SCOPE OF EACH REQUIREMENT HAD BEEN PROPERLY IMPLEMENTED EXCEPT FOR THAT REQUIRED BY TECHNICAL SPECIFICATION 4.6.2.4.1. THE PROCEDURAL DEFICIENCIES CONTRIBUTING TO THIS SITUATION PROMPTED A REVIEW OF THE SURVEILLANCE PROCEDURE PREPARATION AND REVIEW PROCESS. IMPROVED GUIDANCE WILL BE IMPLEMENTED BY FEBRUARY 1, 1988.

[136]   HATCH 1   DOCKET 50-321   LER 86-039 REV 02
UPDATE ON BLOWN FUSES MAKE CONTROL ROOM ENVIRONMENTAL SYSTEM INOPERABLE FOR AUTOMATIC FUNCTIONS.
EVENT DATE: 102386   REPORT DATE: 102087   NSSS: GE   TYPE: BWR
OTHER UNITS INVOLVED: HATCH 2 (BWR)
VENDOR: BUSSMANN MFG (DIV OF MCGRAW-EDISON)

(NSIC 206775) ON 10/23/86 AT 0400 CST AND AT 0500 CST, UNIT 1 WAS IN THE RUN MODE AT AN APPROXIMATE POWER OF 2428 MWT (99 PERCENT OF RATED THERMAL POWER). AT THESE TIMES, PLANT PERSONNEL WERE PERFORMING A SURVEILLANCE AND DETERMINED THAT THE MAIN CONTROL ROOM ENVIRONMENTAL CONTROL (MCREC) SYSTEM WOULD NOT SWITCH AUTOMATICALLY TO THE PRESSURIZATION OR ISOLATION MODES OF OPERATION UPON RECEIPT OF A HIGH CHLORINE SIGNAL. THE EVENT WAS CAUSED BY TWO BLOWN POWER SUPPLY FUSES (ONE IN EITHER MCREC SYSTEM TRIP CHANNEL). THE BLOWN FUSES PREVENTED THE REQUIRED AUTOMATIC ISOLATIONS. THE CAUSE OF THE BLOWN FUSES IS NOT KNOWN AT THIS TIME. SUBSEQUENT TO THIS EVENT, IT WAS DETERMINED AND REPORTED BY ANOTHER LER, THAT THE MCREC SYSTEM HAD A DESIGN DEFICIENCY. CORRECTIVE ACTIONS FOR THE EVENT IN THIS LER AND FOR THE DESIGN DEFICIENCY WERE: 1) REPLACING THE FUSES AND VERIFYING OPERABILITY, 2) PERFORMING AN ENGINEERING STUDY INVESTIGATING THE CAUSE OF THE BLOWN FUSES AND EVALUATING THE SAFETY CONSEQUENCES OF THE EVENT, 3) DEVELOPING A JUSTIFICATION FOR CONTINUED OPERATION, 4) ISSUING A STANDING ORDER, AND 5) DEVELOPING A DESIGN CHANGE.

[137]   HATCH 1   DOCKET 50-321   LER 86-043 REV 01
UPDATE ON THRUST BEARINGS WEARS OUT CAUSING SCOOP TUBE POSITIONER FAILURE RESULTING IN REACTOR SCRAM.
EVENT DATE: 112286   REPORT DATE: 110487   NSSS: GE   TYPE: BWR
VENDOR: BAILEY METER COMPANY

(NSIC 206867) ON 11/22/86 AT APPROXIMATELY 2342 CST, UNIT 1 WAS AT AN APPROXIMATE POWER LEVEL OF 1887 MWT (77 PERCENT OF RATED THERMAL POWER). THE REACTOR MODE SWITCH WAS IN THE RUN POSITION AND THE REACTOR WAS RAMPING TO RATED POWER. THE SCOOP TUBE FOR THE "B" RECIRCULATION PUMP (EIIS CODE AD) MOTOR GENERATOR (MG) SET WAS LOCKED DUE TO PREVIOUS ERRATIC OPERATION. AT THAT TIME, OPERATIONS PERSONNEL WERE PREPARING TO INCREASE REACTOR POWER BY INCREASING RECIRCULATION FLOW. AN ASSISTANT PLANT OPERATOR (APO) UNLOCKED THE SCOOP TUBE PER PROCEDURE. HOWEVER, THE PUMP SPEED AND LOOP FLOW INCREASED AND CAUSED THE REACTOR SCRAM ON A REACTOR POWER SPIKE ON THE AVERAGE POWER RANGE MONITOR (APRM EIIS CODE IG). THE ROOT CAUSE OF THE EVENT IS EQUIPMENT FAILURE: THE THRUST BEARING OF SCOOP TUBE POSITIONER FAILED DUE TO NORMAL WEAR OUT. THE CORRECTIVE ACTIONS FOR THIS EVENT INCLUDED: 1) CHECKING THE RECIRCULATION SPEED CONTROLS, 2) PLACING AN OPERATOR

AID ON THE RECIRCULATION SCOOP TUBE CONTROL SWITCH, 3) GENERATING A PROCEDURE TO INVESTIGATE THE FAILURE, 4) REPLACING THE BEARING AND CALIBRATING EQUIPMENT, 5) SUBMITTING PROCEDURE REVISION REQUESTS, AND 6) SCHEDULING INSPECTIONS OF THE OTHER SCOOP TUBE POSITIONERS.

[138] HATCH 1 DOCKET 50-321 LER 87-017
PERSONNEL ERRORS IN CLEARANCE AND WORK STEPS RESULT IN WET CARBON FILTER IN SBGT.
EVENT DATE: 100887 REPORT DATE: 110987 NSSS: GE TYPE: BWR
VENDOR: BETTIS CORPORATION
FARR CO.
KENNEDY VALVE MFG CO.

(NSIC 206925) ON 10/8/87 AT APPROXIMATELY 1235 CDT, PLANT OPERATIONS PERSONNEL ON ROUNDS NOTICED WATER DRIPPING FROM THE B SGTS FILTER TRAIN DRAIN. MAINTENANCE WAS IN PROGRESS TO REMOVE AND RELOCATE SOME FIRE PROTECTION VALVES. PLANT PERSONNEL DETERMINED THE LEAKAGE WAS FROM A CLEARANCE BOUNDARY VALVE, WHICH WAS TORQUED TO DECREASE THE LEAKAGE. THE A AND B FILTER TRAINS WERE INSPECTED AND THE CARBON FILTERS WERE NOT WET. SUBSEQUENTLY, ON 10/9/87 AT APPROXIMATELY 0845 CDT, OPERATIONS PERSONNEL FOUND THE A SGTS CARBON FILTER TRAIN WAS WET. THE ROOT CAUSE OF THIS EVENT IS DUE TO PERSONNEL ERRORS ON THE PART OF ENGINEERING AND OPERATIONS PERSONNEL. ENGINEERING PERSONNEL DEVELOPED WORK SEQUENCE STEPS THAT CONTRIBUTED TO THE WETTING OF THE FILTER TRAIN. OPERATIONS PERSONNEL DID NOT REQUIRE A DRAIN VALVE TO BE OPEN. WHEN THE CLEARANCE BOUNDARY VALVE LEAKED, WATER WETTED THE A SGTS CARBON FILTER. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDED: 1) REPLACING THE CARBON FILTER MEDIUM, 2) RETURNING THE SGTS TO OPERABLE STATUS, 3) PERFORMING TESTS ON THE SGTS, 4) COUNSELING INVOLVED PERSONNEL, AND 5) VERIFYING UNIT 2 DID NOT HAVE A SIMILAR PROBLEM.

[139] HATCH 2 DOCKET 50-366 LER 85-017 REV 01
UPDATE ON MOLDED CASE BREAKER SETPOINTS OUTSIDE OF TECH SPEC LIMITS.
EVENT DATE: 043085 REPORT DATE: 102087 NSSS: GE TYPE: BWR

(NSIC 206773) ON 04/30/85 AND 05/01/85, DURING PERFORMANCE OF THE "MOLDED CASE BREAKERS PROTECTING THE PRIMARY CONTAINMENT PENETRATION CONDUCTORS SURVEILLANCE" PROCEDURE (HNP-2-3850), PLANT PERSONNEL NOTED THAT MOLDED CASE CIRCUIT BREAKERS (MCC) HAD TRIP SETPOINTS AND/OR BREAKER LOCATIONS THAT WERE CONTRARY TO THE SPECIFIED SETPOINTS AND/OR SPECIFIED LOCATIONS OF TECHNICAL SPECIFICATIONS TABLE 3.8.2.6-1. THE AS-FOUND SETPOINTS OF THE BREAKERS WERE STILL SUFFICIENTLY LOW SO THAT THE CONDUCTORS WOULD HAVE ADEQUATE MARGINS OF PROTECTION. THE CAUSES WERE INADEQUATE INSTRUCTIONS FOR IMPLEMENTING A DESIGN CHANGE REQUEST (DCR) AND PERFORMING MAINTENANCE ON THE BREAKERS (I.E., ADJUSTING THE TRIP SETPOINT WHEN THE BREAKER WAS RETURNED TO SERVICE). CORRECTIVE ACTIONS FOR THIS EVENT INCLUDED: 1) AMENDING THE TECHNICAL SPECIFICATIONS, 2) RESETTING THE BREAKERS, 3) INSTALLING SIGNS, 4) RELOCATING AND TESTING BREAKERS, 5) ISSUING TRAINING DIRECTIVES, 6) IMPLEMENTING CORRECTIVE ACTIONS FOR THE ARCHITECT ENGINEERS, AND 7) REVISING PROCEDURES.

[140] HATCH 2 DOCKET 50-366 LER 87-012
CALCIUM DEPOSITS FOUL CHILLER CAUSING HIGH ROOM TEMPERATURE AND ESF VALVE ISOLATION.
EVENT DATE: 091687 REPORT DATE: 101687 NSSS: GE TYPE: BWR
VENDOR: CARRIER CORP.

(NSIC 206653) ON 9/16/87 AT APPROXIMATELY 0730 CDT, UNIT 2 WAS IN THE RUN MODE AT AN APPROXIMATE POWER LEVEL OF 2209 MWT (APPROXIMATELY 90 PERCENT OF RATED THERMAL POWER). AT THAT TIME, PLANT OPERATIONS PERSONNEL NOTED THAT THE OUTBOARD REACTOR WATER CLEANUP (RWCU EIIS CODE CE) SUCTION VALVE HAD CLOSED. THIS VALVE IS ALSO A PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS EIIS CODE JM) VALVE. THE VALVE CLOSED

AS A RESULT OF A HIGH AMBIENT AIR TEMPERATURE CONDITION IN THE RWCU HEAT EXCHANGER ROOM. THE CLOSURE OF THE VALVE WAS AN UNANTICIPATED ACTUATION OF AN ENGINEERED SAFETY FEATURE (ESF). THE ROOT CAUSE OF THIS EVENT IS DUE TO EQUIPMENT DEGRADATION. SPECIFICALLY A WATER CHILLER (EIIS CODE KM) WAS FOULED BY CALCIUM DEPOSITS WHICH TRIPPED THE CHILLER MOTOR AS A RESULT OF A HIGH CONDENSER PRESSURE CONDITION. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDED: 1) REMOVING THE CHILLER FROM SERVICE AND REPLACING THE COOLING FUNCTION WITH THE "B" CHILLER, 2) REMOVING THE CALCIUM DEPOSITS FROM THE "A" CHILLER AND LATER RETURNING THE "A" CHILLER TO A STANDBY CONFIGURATION, AND 3) INSTALLING A CHEMICAL TREATMENT SYSTEM ON THE CHILLERS' COOLING TOWER TO PREVENT FUTURE CALCIUM ACCUMULATIONS.

[141] HOPE CREEK 1 DOCKET 50-354 LER 87-040
REACTOR WATER CLEANUP (RWCU) SYSTEM ISOLATIONS (2) FOLLOWING RWCU PUMP MAINTENANCE DUE TO NOT ADHERING TO PROCEDURES.
EVENT DATE: 091087 REPORT DATE: 101287 NSSS: GE TYPE: BWR

(NSIC 206647) ON SEPTEMBER 10, 1987 TWO ISOLATIONS OF THE REACTOR WATER CLEANUP SYSTEM OCCURRED DURING ATTEMPTS TO RESTORE "A" RWCU PUMP TO SERVICE FOLLOWING MAINTENANCE. AT 1046, AN RWCU ISOLATION WAS INITIATED BY AN RWCU HIGH DIFFERENTIAL FLOW SIGNAL. INVESTIGATION DETERMINED THAT TWO DRAIN VALVES ON THE DISCHARGE SIDE OF THE "A" RWCU PUMP HAD BEEN LEFT OPEN, RESULTING IN A HIGH DIFFERENTIAL FLOW CONDITION WHEN WARMING UP THE PUMP. AT 1339, ANOTHER RWCU ISOLATION OCCURRED WHEN THE "A" RWCU PUMP MECHANICAL SEAL FAILED DUE TO LACK OF COOLING WATER TO THE SEAL CAVITY COOLER. THIS RESULTED IN LOCALIZED HIGH TEMPERATURES IN THE RWCU PUMP ROOM, AND A STEAM LEAK DETECTION SYSTEM HIGH ROOM TEMPERATURE SIGNAL INITIATED THE ISOLATION. THE CAUSE OF THIS OCCURRENCE HAS BEEN ATTRIBUTED TO A COMBINATION OF FACTORS, THE MOST PREDOMINATE BEING PERSONNEL ERRORS COMMITTED WHILE PLACING THE PUMP BACK IN SERVICE. CORRECTIVE ACTIONS WERE PRIMARILY ADMINISTRTATIVE IN NATURE, AND THE EQUIPMENT OPERATOR INVOLVED IN THE PERSONNEL ERROR WAS COUNSELED WITH RESPECT TO THE ERRORS MADE.

[142] HOPE CREEK 1 DOCKET 50-354 LER 87-041
OVERDUE CHANNEL FUNCTIONAL TEST DUE TO PERSONNEL BEING UNFAMILIAR WITH NEW COMPUTERIZED SCHEDULING SYSTEM.
EVENT DATE: 091087 REPORT DATE: 101287 NSSS: GE TYPE: BWR

(NSIC 206627) TECHNICAL SPECIFICATIONS REQUIRE THAT AT LEAST ONCE PER 31 DAYS, A CHANNEL FUNCTIONAL TEST BE PERFORMED ON THE PRIMARY CONTAINMENT INSTRUMENT GAS SYSTEM (PCIG) LOW-LOW PRESSURE ALARM SYSTEM TO SATISFY OPERABILITY REQUIREMENTS OF THE AUTOMATIC DEPRESSURIZATION SYSTEM (ADS). ON SEPTEMBER 10, 1987 IT WAS DETERMINED THAT THE 31 DAY CHANNEL FUNCTIONAL TEST ON THE PCIG LOW-LOW PRESSURE ALARM SYSTEM WAS 4 DAYS OVERDUE. UPON DISCOVERY, THE OVERDUE FUNCTIONAL TEST WAS PERFORMED. THE PRIMARY CAUSE OF THIS EVENT WAS PERSONNEL BEING UNFAMILIAR WITH A NEW COMPUTERIZED SCHEDULING SYSTEM WHICH HAD BEEN RECENTLY IMPLEMENTED. CORRECTIVE ACTIONS CONSISTED PRIMARILY OF ENHANCED TRAINING FOR THE PLANNING DEPARTMENT AND MAINTENANCE DEPARTMENT REGARDING THE PROPER USE OF THE NEW SCHEDULING SYSTEM.

[143] HOPE CREEK 1 DOCKET 50-354 LER 87-042
INVALID LOSS OF COOLANT ACCIDENT (LOCA) SIGNAL ISOLATION WHEN PERFORMING TEST DUE TO LEAKING INSTRUMENT VALVE.
EVENT DATE: 092087 REPORT DATE: 102087 NSSS: GE TYPE: BWR

(NSIC 206722) A CHANNEL "A" LOSS OF COOLANT ACCIDENT (LOCA) SIGNAL RESULTED IN THE ISOLATION OF VARIOUS CHANNEL "A" COMPONENTS. THE INVALID LOCA SIGNAL OCCURRED DURING AN I&C DEPARTMENT SENSOR CALIBRATION OF A REACTOR VESSEL REFERENCE LEG PRESSURE TRANSMITTER. WHILE PRESSURIZING THE ISOLATED TRANSMITTER FROM A TEST SOURCE, THE ASSOCIATED INSTRUMENT ISOLATION VALVE LEAKED, RESULTING

IN A PRESSURE SPIKE ON THE REFERENCE LEG. THE PRESSURE SPIKE TRIPPED THE LEVEL TRANSMITTER WHICH PROVIDES THE CHANNEL "A" LOCA SIGNAL. SUBSEQUENT INVESTIGATION DETERMINED THAT THE LEAKING INSTRUMENT VALVE WAS THE ROOT CAUSE OF THIS INCIDENT. CORRECTIVE ACTIONS CONSISTED OF REPAIRING THE LEAKING VALVE AND RE- PERFORMING THE SUBJECT SENSOR CALIBRATION.

[144] HOPE CREEK 1 DOCKET 50-354 LER 87-043
RESIDUAL HEAT REMOVAL SYSTEM (RHR) ISOLATION WHILE PERFORMING SURVEILLANCE TEST DUE TO INSUFFICIENT WORK SPACE - DESIGN DEFICIENCY.
EVENT DATE: 092287 REPORT DATE: 102187 NSSS: GE TYPE: BWR

(NSIC 206811) AN NSSSS CHANNEL "D" ISOLATION OF THE RESIDUAL HEAT REMOVAL SYST M OCCURRED WHEN A FUSE WAS BLOWN ON A PORTION OF THE CHANNEL "D" ISOLATION LOGIC. THE FUSE BLEW DURING PERFORMANCE OF AN I&C SURVEILLANCE PROCEDURE WHEN TEST EQUIPMENT LEADS INADVERTENTLY CAME IN CONTACT WITH A GROUND BUS INSIDE THE STEAM LEAK DETECTION PANEL IN WHICH THE TEST WAS BEING CONDUCTED. THE ISOLATION CAUSED THE RHR "A" AND "B" LOOP COMMON SUCTION VALVE TO CLOSE, AND RESULTED IN A LOSS OF THE RHR SYSTEM. FOLLOWING REPLACEMENT OF THE BLOWN FUSE, THE ISOLATION LOGIC WAS RESET AND RHR "B" LOOP WAS RETURNED TO SERVICE. INCIDENTS OF THIS NATURE HAVE BEEN IDENTIFIED IN PREVIOUS LER'S, THE ROOT CAUSE HAVING BEEN DETERMINED TO BE INSUFFICIENT WORK SPACE INSIDE THE STEAM LEAK DETECTION PANELS. A DESIGN CHANGE TO ENABLE BETTER ACCESS TO MONITORING POINTS INSIDE THE PANEL WILL BE IMPLEMENTED PRIOR TO COMPLETION OF THE STATIONS FIRST REFUELING OUTAGE.

[145] HOPE CREEK 1 DOCKET 50-354 LER 87-044
RESIDUAL HEAT REMOVAL (RHR) SYSTEM ISOLATION WHILE PERFORMING SURVEILLANCE TEST - CAUSE UNKNOWN.
EVENT DATE: 092587 REPORT DATE: 102687 NSSS: GE TYPE: BWR

(NSIC 206812) AN ISOLATION OF THE RESIDUAL HEAT REMOVAL SYSTEM OCCURRED DURING PERFORMANCE OF AN I&C SURVEILLANCE TEST ON A REACTOR VESSEL PRESSURE TRANSMITTER. THE ISOLATION CAUSED THE RHR "A" AND "B" LOOP INBOARD COMMON SUCTION VALVE TO CLOSE, AND RESULTED IN A LOSS OF THE RHR SYSTEM. AFTER DETERMINING THE ISOLATION WAS INITIATED BY AN INVALID SPURIOUS SIGNAL, THE ISOLATION WAS RESET AND RHR "B" LOOP WAS RETURNED TO SERVICE. INVESTIGATION SUBSEQUENT TO THE INCIDENT COULD NOT DETERMINE THE ACTUAL CAUSE OF THE EVENT, HOWEVER, THE MOST LIKELY CAUSE IS A LOOSE CONNECTION DISTURBED DURING THE SURVEILLANCE TEST. A LOOSE CONNECTION ON A CALIBRATION CARD OR WIRE COULD HAVE CAUSED THE SPURIOUS ISOLATION SIGNAL. I&C DEPARTMENT WILL ENSURE ALL PERSONNEL RESPONSIBLE FOR PERFORMANCE OF I&C SURVEILLANCES ARE AWARE OF THIS SITUATION. ADDITIONALLY, I&C DEPARTMENT WILL CHECK ALL CARDS FOR MIS-SEATING OR LOOSE WIRES DURING THE COURSE OF NORMAL STEAM LEAK DETECTION CABINET MONTHLY SURVEILLANCES.

[146] HOPE CREEK 1 DOCKET 50-354 LER 87-045
PRIMARY CONTAINMENT ISOLATION SYSTEM INITIATION WHEN RESTORING POWER TO LOGIC CABINET DUE TO SPURIOUS LOGIC MODULE INPUTS.
EVENT DATE: 100787 REPORT DATE: 110687 NSSS: GE TYPE: BWR

(NSIC 206937) AN INITIATION OF PRIMARY CONTAINMENT ISOLATION SYSTEM CHANNEL "C" OCCURRED DURING THE IMPLEMENTATION OF A DESIGN CHANGE AFFECTING THE OVERHEAD ANNUNCIATORS IN THE CONTROL ROOM. THE DESIGN CHANGE CALLED FOR, IN PART, INSTALLING A LOGIC MODULE IN A BAILEY CONTROL CABINET (1CC652). WHEN THE LOGIC MODULE HAD BEEN INSTALLED AND WIRED UP, THE TECHNICIANS PERFORMING THE WORK RE-ENERGIZED POWER TO THE CONTROL CABINET IN PREPARATION FOR TESTING OF THE MODULE. SHORTLY AFTER THE TECHNICIANS BEGAN POWERING UP THE INDIVIDUAL LOGIC MODULES IN THE CABINET, A PCIS CHANNEL "C" SIGNAL WAS RECEIVED. SUBSEQUENT INVESTIGATION REVEALED THAT THE PCIS CHANNEL "C" LOGIC MODULE HAD RECEIVED

SPURIOUS INPUT SIGNALS DURING THE POWER-UP OF THE CONTROL CABINET AND INITIATED THE SUBJECT ISOLATION.

[147] HOPE CREEK 1 DOCKET 50-354 LER 87-047
SAFETY/RELIEF VALVE (SRV) FAILURE TO CLOSE DUE TO SAND BLASTING GRIT IN SOLENOID.
EVENT DATE: 101087 REPORT DATE: 110987 NSSS: GE TYPE: BWR
VENDOR: TARGET ROCK CORP.

(NSIC 206813) ON OCTOBER 10, 1987 AT 1950 HOURS, THE PLANT WAS IN OPERATIONAL CONDITION 2 (STARTUP/HOT STANDBY) AT 10% POWER GENERATING 0 MWE WHEN THE "J" SRV FAILED TO CLOSE ON SIGNAL. THE SRV HAD BEEN OPENED DURING A TEST WHICH WAS BEING PERFORMED TO COLLECT BASELINE DATA FOR THE ACOUSTIC MONITORS. THE REACTOR WAS SCRAMMED MANUALLY AT 1952 HOURS IN ACCORDANCE WITH THE HCGS ABNORMAL OPERATING PROCEDURES. PROCEDURES TO PLACE THE UNIT IN COLD SHUTDOWN WERE EXECUTED. THE ROOT CAUSE OF THIS OCCURRENCE WAS DETERMINED TO BE SAND BLASTING GRIT IN THE SOLENOID VALVE. CORRECTIVE ACTIONS INCLUDED REPLACEMENT OF THE MALFUNCTIONING TARGET ROCK VALVE AND INSPECTION OF A SAMPLING OF THE SOLENOIDS OF OTHER SRVS.

[148] HOPE CREEK 1 DOCKET 50-354 LER 87-046
PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) INITIATION WHEN SWAPPING REACTOR PROTECTION SYSTEM BUS POWER DUE TO LACK OF INDICATION - DESIGN DEFICIENCY.
EVENT DATE: 101687 REPORT DATE: 111687 NSSS: GE TYPE: BWR

(NSIC 207126) AN INITIATION OF PRIMARY CONTAINMENT ISOLATION SYSTEM CHANNEL "B" OCCURRED WHILE SWAPPING "B" REACTOR PROTECTION SYSTEM (RPS) FROM NORMAL BUS POWER TO ALTERNATE BUS POWER. BECAUSE AN EARLIER, UNDETECTED PCIS CHANNEL "B" TRIP INPUT HAD BEEN RECEIVED AND NOT RESET, WHEN THE "B" RPS BUS WAS MOMENTARILY DE-ENERGIZED DURING THE POWER SOURCE SWAP, THE PCIS CHANNEL "B" LOGIC WAS SATISIFED AND THE ISOLATION OCCURRED. THE PRIMARY CAUSE OF THIS OCCURRENCE WAS THE LACK OF INDICATION IN THE CONTROL ROOM TO ALERT THE OPERATORS THAT THE FIRST TRIP INPUT HAD BEEN RECEIVED. HAD THIS INDICATION BEEN AVAILABLE, THE FIRST TRIP SIGNAL WOULD HAVE BEEN RESET PRIOR TO THE BUS SWAP, AND THE ISOLATION WOULD NOT HAVE OCCURRED. CORRECTIVE ACTIONS INCLUDED EXPEDITING A PREVIOUSLY IDENTIFIED DESIGN CHANGE AND REVISING THE PROCEDURE DEALING WITH RPS BUS SWPAS TO INCLUDE A STEP TO ENSURE ALL ISOLATION SIGNALS ARE RESET PRIOR TO SWAPPING THE BUS.

[149] INDIAN POINT 2 DOCKET 50-247 LER 87-010
HIGH RADIOACTIVITY IN CONTAINMENT CAUSES ISOLATION OF CONTAINMENT VENTILATION SYSTEM.
EVENT DATE: 100887 REPORT DATE: 110787 NSSS: WE TYPE: PWR

(NSIC 207153) ON OCTOBER 8, 1987, WHILE THE PLANT WAS AT COLD SHUTDOWN, THE CONTAINMENT NOBLE GAS RADIOACTIVITY LEVEL WAS MOMENTARILY ELEVATED. CONSEQUENTLY, THE NOBLE GAS MONITOR (MON) INSIDE CONTAINMENT ACTIVATED A CONTAINMENT EVACUATION ALARM. CONTAINMENT VENTILATION WAS PROMPTLY ISOLATED. IN ACCORDANCE WITH PLANT DESIGN, ISOLATION OF CONTAINMENT VENTILATION INITIATED PARTIAL OPERATION OF THE WELD CHANNEL PENETRATION PRESSURIZATION SYSTEM (WCPPS). THE WCPPS IS CLASSIFIED AS AN ENGINEERED SAFETY FEATURE (ESF). THE LOGIC REQUIREMENTS OF THE ESF ACTUATION SYSTEM (ESFAS) WERE NOT REQUIRED NOR FULFILLED. THE CAUSE OF THE RADIOACTIVITY SPIKE WAS A MOMENTARY RELEASE OF PRESSURE FROM THE REACTOR VESSEL (VSL) DURING MAINTENANCE OF A CONOSEAL CONNECTOR (CON). A RELEASE OF RADIOACTIVITY TO THE CONTAINMENT OCCURRED WHICH EXCEEDED THE SETPOINT OF THE NOBLE GAS MONITOR (MON).

[150] INDIAN POINT 2 DOCKET 50-247 LER 87-011
FAILURE OF SERVICE WATER PUMPS DURING SURVEILLANCE TEST.
EVENT DATE: 100987 REPORT DATE: 110987 NSSS: WE TYPE: PWR
VENDOR: AURORA PUMP

(NSIC 206905) ON OCTOBER 8, 1987 WHILE THE PLANT WAS AT COLD SHUTDOWN FOR A REFUELING OUTAGE, TWO SERVICE WATER PUMPS (P) FAILED AN ASME SECTION XI SURVEILLANCE TEST. THE DISCHARGE HEAD DEVELOPED BY THE PUMPS (P) WAS BELOW THE MINIMUM SPECIFIED IN ASME SECTION XI. THE CAUSE OF THE TEST FAILURE IS ATTRIBUTABLE IN ONE INSTANCE TO PUMP (P) DAMAGE CAUSED BY VORTEXING. THE PERFORMANCE OF THE SECOND PUMP (P) IS UNDER EVALUATION. ITS SLIGHT DECLINE IN PERFORMANCE IS NOT BELIEVED TO BE SIGNIFICANT. THERE ARE A TOTAL OF SIX SERVICE WATER PUMPS (P) SUPPLYING TWO HEADERS (3 PUMPS PER HEADER). THE TWO FAILED PUMPS (P) WERE ON ONE HEADER. A THIRD PUMP (P) SUPPLYING THE SECOND HEADER COULD NOT BE TESTED DUE TO MALFUNCTION OF A VALVE OPERATOR. THE DELAYED TEST OF THE THIRD PUMP (P) IS A CONDITION PERMITTED BY THE TECHNICAL SPECIFICATION WHEN THE PLANT IS IN COLD SHUTDOWN. AT NO TIME WAS THERE ANY IMPACT UPON PLANT SAFETY. THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED.

[151] INDIAN POINT 3 DOCKET 50-286 LER 87-009
FAILURE OF 480 VOLT CIRCUIT BREAKER CELL SWITCHES.
EVENT DATE: 090387 REPORT DATE: 100287 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 206594) ON MAY 15, 1987, WITH THE UNIT AT COLD SHUTDOWN FOR A SCHEDULED REFUELING OUTAGE, THE OUTPUT BREAKER FOR EMERGENCY DIESEL GENERATOR (DG) NO. 31 DID NOT CLOSE AS DESIGNED WHEN 480 VOLT BUS NO. 2A WAS INADVERTENTLY DE-ENERGIZED. INVESTIGATION REVEALED THAT CONTACTS IN THE CELL SWITCH FOR BUS 2A'S NORMAL SUPPLY BREAKER DID NOT PROVIDE THE PERMISSIVE SIGNAL TO CLOSE THE DG OUTPUT BREAKER. THE CELL SWITCH WAS FOUND TO HAVE A DEFORMED SPRING RETAINER WHICH PREVENTED THE CELL SWITCH FROM ACTUATING PROPERLY WHEN ITS ASSOCIATED BREAKER WAS RACKED-OUT. AN EXTENSIVE INSPECTION FOUND THAT THIRTY-FIVE OF THIRTY- SEVEN SIMILAR SAFETY-RELATED CELL SWITCHES IN THE 480 VOLT SYSTEM EXHIBITED SOME SIGNS OF RETAINER DEFORMATION. ON SEPTEMBER 3, 1987 IT WAS SUBSEQUENTLY DETERMINED, IN TWO SEPARATE SCENARIOS, THAT IF A 480 VOLT TIE BREAKER WAS RACKED OUT AND THE CELL SWITCH IN THIS BREAKER FAILED TO OPERATE PROPERLY AND A LOSS OF POWER ON AN ASSOCIATED BUS OCCURRED, THE ASSOCIATED DIESEL GENERATOR(S) MIGHT NOT RECEIVE THE PERMISSIVE SIGNAL ALLOWING SYNCHRONIZATION TO THE BUS. ALL SIMILAR SAFETY-RELATED CELL SWITCHES WERE REPLACED. THE SPRING RETAINERS BECAME DEFORMED BECAUSE OF THE CONTINUOUS FORCE OF THE SPRING ON THE RETAINER WHEN THE BREAKER IS RACKED IN.

[152] INDIAN POINT 3 DOCKET 50-286 LER 87-010
INADVERTENT SUBCRITICAL SAFETY INJECTION ACTUATION WHILE TROUBLESHOOTING INVERTER.
EVENT DATE: 090387 REPORT DATE: 100587 NSSS: WE TYPE: PWR

(NSIC 206756) AT 2336 HOURS ON SEPTEMBER 3, 1987, WITH THE REACTOR SUBCRITICAL, THE REACTOR OPERATOR WAS IN THE PROCESS OF MANUALLY INSERTING CONTROL RODS IN ORDER TO SHUTDOWN FOR TURBINE MAINTENANCE. DURING THE COURSE OF THIS SHUTDOWN A REACTOR TRIP AND SAFETY INJECTION (SI) ACTUATION OCCURRED AUTOMATICALLY AND ALL EQUIPMENT FUNCTIONED PROPERLY. NO WATER WAS INJECTED INTO THE REACTOR BECAUSE THE REACTOR COOLANT SYSTEM (RCS) WAS AT NORMAL OPERATING PRESSURE. INVESTIGATION REVEALED THAT, WHILE TROUBLE SHOOTING NO. 32 STATIC INVERTER, AN OPERATOR INADVERTENTLY INTERRUPTED THE POWER SUPPLY TO INSTRUMENT BUS 32 (PROTECTION CHANNEL I), CAUSING ITS VOLTAGE TO DROP TO ZERO. THE LOSS OF POWER TO INSTRUMENT BUS 32 (PROTECTION CHANNEL I) CAUSED ASSOCIATED REACTOR PROTECTION RELAYS TO DE-ENERGIZE AND INITIATED A REACTOR TRIP VIA THE NUCLEAR INSTRUMENTATION SYSTEM (NIS) INTERMEDIATE RANGE 35 HIGH FLUX SIGNAL. DUE TO THE DE-ENERGIZATION OF PROTECTION CHANNEL I, ALL SI RELAYS ASSOCIATED WITH PROTECTION CHANNEL I STEAM

FLOW TRANSMITTERS AND LOW TAVERAGE FOR RCS LOOP 1 ALSO DE-ENERGIZED. THE REMAINING PORTION OF THE SI LOGIC WAS MADE UP WHEN THE ACTUAL RCS LOOP 2 TAVERAGE DECREASED BELOW THE LOW SET-POINT (542F) FOR SI ACTUATION. IN ORDER TO PRECLUDE RECURRENCE, A DISCUSSION OF THIS EVENT WILL BE INCORPORATED INTO THE LICENSED OPERATOR REQUALIFICATION PROGRAM.

[153] KEWAUNEE DOCKET 50-305 LER 87-010
VIOLATION OF TECHNICAL SPECIFICATIONS ON CONTAINMENT INTEGRITY DUE TO OPERATOR ERROR.
EVENT DATE: 090487 REPORT DATE: 100487 NSSS: WE TYPE: PWR

(NSIC 206758) ON SEPTEMBER 4, 1987, AT 0226 CDT WITH THE PLANT AT 100% POWER, TECHNICAL SPECIFICATIONS PERTAINING TO CONTAINMENT INTEGRITY PROVISIONS WERE VIOLATED. THE REDUNDANT CONTAINMENT ISOLATION (CI) SUMP A DISCHARGE CONTROL VALVES (MD(R)-134 AND MD(R)-135) WERE OPENED WHILE VALVE MD(R)-134 WAS CONSIDERED INOPERABLE. THE REACTOR OPERATOR OPENED BOTH VALVES IN RESPONSE TO A HIGH CONTAINMENT SUMP LEVEL ALARM PER OPERATING PROCEDURE A-MDS-30. VALVE MD(R)- 134 WAS ADMINISTRATIVELY INOPERABLE BECAUSE IT HAD NOT BEEN COMPLETELY RETESTED FOLLOWING REPLACEMENT OF ITS ASSOCIATED SOLENOID VALVE. THE ROOT CAUSE OF THE EVENT WAS THE REACTOR OPERATOR FAILING TO RECOGNIZE THAT VALVE MD(R)-134 WAS ADMINISTRATIVELY INOPERABLE. IN ADDITION, DESIGN CHANGE PROCEDURE 1544-15 FAILED TO ADEQUATELY IDENTIFY VALVES MD(R)-134 AND MD(R)-135 AS REDUNDANT CONTAINMENT ISOLATION VALVES. CONTAINMENT ISOLATION VALVE, MD(R)-134, SATISFACTORILY COMPLETED ITS RETEST REQUIREMENTS WITHOUT ANY FURTHER ADJUSTMENTS ON THE SUBSEQUENT WORK SHIFT THE NEXT DAY. THE VALVE WAS DECLARED OPERABLE AT 1210 ON SEPTEMBER 4. IMMEDIATE CORRECTIVE ACTIONS INCLUDED AN INFORMAL REVIEW WITH THE PERSONNEL INVOLVED. OTHER CORRECTIVE ACTIONS WILL INCLUDE REVIEWING THE CI SYSTEM WITH OPERATIONS PERSONNEL DURING REQUALIFICATION TRAINING.

[154] LA SALLE 1 DOCKET 50-373 LER 87-002 REV 01
UPDATE ON FAILURE OF "O" DIESEL GENERATOR OUTPUT BREAKER TO CLOSE ONTO BUS 241Y.
EVENT DATE: 011587 REPORT DATE: 102387 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LA SALLE 2 (BWR)

(NSIC 206856) ON JANUARY 15, 1987, AT APPROXIMATELY 1045 HOURS, LASALLE OPERATING SURVEILLANCE LOS-DG-M1 ("O" DIESEL GENERATOR OPERABILITY TEST) WAS BEING PERFORMED. UNIT 1 WAS IN RUN AT APPROXIMATELY 89% POWER AND UNIT 2 WAS IN REFUEL. THE UNIT 1 OPERATOR (LICENSED REACTOR OPERATOR) HAD JUST COMPLETED A PORTION OF LOS-DG-M1. THE OPERATOR AT THIS TIME COMMENCED TO SYNCHRONIZE THE "O" DIESEL GENERATOR (DG) OUTPUT WITH BUS 241Y AND CLOSE THE "O" DG OUTPUT BREAKER TO BUS 241Y BUT THE BREAKER WOULD NOT CLOSE EVEN AFTER TWO ATTEMPTS. A SECOND OPERATOR ALSO TRIED TO SYNCHRONIZE THE "O" DG OUTPUT WITH BUS 241Y AND CLOSE THE BREAKER BUT WITHOUT SUCCESS. THE OPERATING DEPARTMENT WITH ELECTRICAL MAINTENANCE IN ATTENDANCE RACKED THE BREAKER FROM "CONNECT" TO "TEST" AND FOUND THAT THE BREAKER WOULD CLOSE WHILE IN "TEST". AFTER THE BREAKER WAS RACKED TO "CONNECT" IT WAS ABLE TO BE CLOSED AND PASSED ITS SURVEILLANCE SATISFACTORY. TROUBLESHOOTING EFFORTS ON THE BREAKER CLOSE CIRCUITRY REVEALED NO DISCREPANCIES. ALL BREAKER COMPONENTS, INCLUDING ASSOCIATED CLOSURE PERMISSIVE CONTACTS, WERE VERIFIED TO OPERATE AS DESIGNED FOLLOWING THE EVENT. SINCE THIS EVENT COULD NOT BE DUPLICATED UNDER TEST CONDITIONS, THE CAUSE OF THE OUTPUT BREAKER FAILING TO CLOSE IS UNKNOWN. THIS EVENT IS REQUIRED TO BE REPORTED AS A SPECIAL REPORT PER TECH SPECS 4.8.1.1.3.

[155] LA SALLE 1 DOCKET 50-373 LER 87-031
LOSS OF 1B REACTOR PROTECTION SYSTEM MOTOR GENERATOR DUE TO RELAY FAILURES.
EVENT DATE: 090587 REPORT DATE: 100587 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LA SALLE 2 (BWR)
VENDOR: GENERAL ELECTRIC CO.

(NSIC 206660) AT 0900 HOURS ON SEPTEMBER 5, 1987, WITH UNIT 1 IN OPERATIONAL CONDITION 4 (COLD SHUTDOWN) AND UNIT 2 IN OPERATIONAL CONDITION 1 (RUN) AT 100% POWER, THE "B" REACTOR PROTECTION SYSTEM (RPS) MOTOR GENERATOR (MG) SET TRIPPED DUE TO A CONTROL RELAY FAULT, WHICH IN TURN, DEENERGIZED THE "B" RPS BUS. LOSS OF THE "B" RPS BUS CAUSED A HALF SCRAM AND PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) GROUPS II AND VII ISOLATIONS (INBOARD) ON UNIT 1, AND A PCIS GROUP IV ISOLATION ON UNIT 2. POWER WAS RESTORED TO THE "B" RPS BUS THROUGH THE ALTERNATE FEED AND ALL ISOLATIONS WERE RESET BY 0920 HOURS ON SEPTEBMER 5, 1987. THE CAUSE OF THE CONTROL RELAY FAILURE IS UNKNOWN. AN INVESTIGATION OF THE CONTROL CIRCUITRY REVEALED NO CONDITIONS (GROUNDS, SHORTS) WHICH COULD HAVE CONTRIBUTED TO THE RELAY'S FAILURE. THE SAFETY CONSEQUENCES OF THIS EVENT WERE MINIMAL. ALL ISOLATIONS AND ACTUATIONS OCCURRED AS DESIGNED FOR THIS EVENT. THIS EVENT IS REPORTABLE PURSUANT TO THE REQUIREMENTS OF 10CFR50.73(A)(2)(IV) DUE TO THE AUTOMATIC ACTUATION OF AN ENGINEERED SAFETY FEATURE.

[156] LA SALLE 1 DOCKET 50-373 LER 87-032
REACTOR SCRAM ON LOW REACTOR WATER LEVEL DUE TO DIFFICULTY CONTROLLING REACTOR WATER LEVEL WITH THE FEEDWATER REGULATING VALVE IN LOW FLOW, LOW POWER CONDITIONS.
EVENT DATE: 091787 REPORT DATE: 101687 NSSS: GE TYPE: BWR

(NSIC 206661) AT 0322 HOURS ON SEPTEMBER 17, 1987, WITH UNIT 1 IN OPERATIONAL CONDITION 2 (STARTUP) AT 5% POWER, THE UNIT SCRAMMED ON A LOW REACTOR WATER LEVEL SIGNAL DURING A CONTROLLED SHUTDOWN. REACTOR WATER LEVEL INITIALLY INCREASED TO A LEVEL WHICH TRIPPED THE MOTOR DRIVEN REACTOR FEED PUMP. THIS CAUSED A LEVEL DECREASE TO OCCUR WHICH RESULTED IN A UNIT SCRAM ON A LOW REACTOR WATER LEVEL SIGNAL. THE ROOT CAUSE OF THIS EVENT IS THE DIFFICULTY OF CONTROLLING REACTOR WATER LEVEL WITH THE FEEDWATER REGULATING VALVE (FRV) AT LOW FLOW, LOW POWER CONDITIONS DUE TO SYSTEM DESIGN. THE SAFETY CONSEQUENCES OF THIS EVENT WERE MINIMAL. ALL ACTUATIONS OCCURRED AS DESIGNED DURING THIS EVENT. DURING THE NEXT REFUELING OUTAGE FOR EACH UNIT, A LOW LOAD REGULATING VALVE WILL BE ADDED TO THE FEEDWATER SYSTEM, WHICH SHOULD PROVIDE IMPROVED FEEDWATER AND REACTOR WATER LEVEL CONTROL AT LOW FLOW, LOW POWER CONDITIONS. THIS EVENT IS REPORTABLE PURSUANT TO THE REQUIREMENTS OF 10CFR50.73(A)(2)(IV) DUE TO THE AUTOMATIC ACTUATION OF THE REACTOR PROTECTION SYSTEM.

[157] LA SALLE 1 DOCKET 50-373 LER 87-033
FAILURE OF DIESEL GENERATOR 1A TO CLOSE ONTO 142Y DURING SURVEILLANCE TESTING.
EVENT DATE: 091787 REPORT DATE: 101687 NSSS: GE TYPE: BWR

(NSIC 206723) ON SEPTEMBER 17, 1987 AT APPROXIMATELY 1115 HOURS, WITH UNIT 1 IN HOT SHUTDOWN, LASALLE OPERATING SURVEILLANCE LOS-DG-M2 ("1A" DIESEL GENERATOR OPERABILITY TEST) WAS BEING PERFORMED. THE UNIT 1 OPERATOR (LICENSED REACTOR OPERATOR) HAD STARTED THE "1A" DIESEL GENERATOR (DG) AND ATTEMPTED TO CLOSE THE OUTPUT BREAKER AND SYNCHRONIZE TO BUS 142Y. HOWEVER, THE OUTPUT BREAKER WOULD NOT CLOSE. SEVERAL ATTEMPTS WERE MADE AND ALL WERE UNSUCCESSFUL. THE "1A" DG WAS THEN SHUT DOWN AND DECLARED INOPERABLE. TROUBLESHOOTING EFFORTS ON THE OUTPUT BREAKER CLOSING CIRCUITRY REVEALED NO DISCREPANCIES. ALL BREAKER COMPONENTS, INCLUDING ASSOCIATED CLOSURE PERMISSIVE CONTACTS, WERE VERIFIED TO OPERATE AS DESIGNED FOLLOWING THE EVENT. LOS-DG-M2 WAS THEN PERFORMED SUCCESSFULLY AND THE "1A" DG WAS DECLARED OPERABLE AT 1645 HOURS ON SEPTEMBER 17, 1987. SINCE THIS EVENT COULD NOT BE REPEATED UNDER TEST CONDITIONS, THE CAUSE IS UNKNOWN. THEREFORE, ELECTRICAL MAINTENANCE AND ENGINEERING PERSONNEL HAVE BEEN PRESENT FOR ALL DG OPERABILITY SURVEILLANCES SINCE THIS EVENT. IN ADDITION, LASALLE TECHNICAL STAFF PROCEDURE LTP-500-1 (DIESEL GENERATOR OUTPUT BREAKER TROUBLESHOOTING) WAS WRITTEN FOR THE PURPOSES OF TROUBLESHOOTING THE OUTPUT BREAKER CIRCUITRY IF AN EVENT OF THIS TYPE SHOULD OCCUR DURING FUTURE DG OPERABILITY SURVEILLANCES.

[158] LA SALLE 2 DOCKET 50-374 LER 85-011 REV 01
UPDATE ON FAILURE OF TYPE C LEAK RATE TEST.
EVENT DATE: 022685 REPORT DATE: 110487 NSSS: GE TYPE: BWR
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 206963) ON FEBRUARY 26, 1985, AT 1155 HOURS WITH THE UNIT AT 99% POWER, THE 2FC086 AND 2FC115 REACTOR WELL DRAIN VALVES FAILED THEIR LOCAL LEAK RATE TEST (EXCEEDED 0.6 LA LEAKAGE RATE FOR WORST VALVE IN LINE). THE ACTIONS OF TECH SPEC 3.6.3 WERE TAKEN AS APPROPRIATE. INITIAL INSPECTION SHOWED THAT FOREIGN MATTER ON SEAT AND SEAT IRREGULARITIES CAUSED THE VALVES TO FAIL THE LOCAL LEAK RATE TEST. THE VALVES WERE DISASSEMBLED AND THE SEAT SURFACES LAPPED. A TABLE IS INCLUDED TO SUMMARIZE ALL OTHER LOCAL LEAK RATE TEST FAILURES DURING THE UNIT 2 OUTAGE.

[159] LA SALLE 2 DOCKET 50-374 LER 87-002 REV 01
UPDATE ON CONTAINMENT LEAKAGE LIMIT OF 0.6 LA EXCEEDED ON LOCAL LEAK RATE TESTING OF FEEDWATER CHECK VALVES.
EVENT DATE: 011787 REPORT DATE: 102987 NSSS: GE TYPE: BWR
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 206855) ON JANUARY 17, 1987, WHILE UNIT 2 WAS SHUT DOWN FOR A SCHEDULED REFUEL OUTAGE, LOCAL LEAK RATE TESTS WERE PERFORMED ON THE INBOARD 2B21-F010A/B, OUTBOARD 2B21-F032A/B FEEDWATER CHECK VALVES, 2E12-F053A RESIDUAL HEAT REMOVAL MOTOR OPERATED VALVE AND THE 2IN031 TIP PURGE ASSEMBLY SUPPLY STOP VALVE. A SUMMATION OF LEAKAGE OF 510.28 SCFH WAS OBSERVED WHICH IS IN EXCESS OF 0.6 LA LIMITS. TABLE 1 IS INCLUDED IN THE TEXT OF THE REPORT TO SUMMARIZE THE CAUSES AND CORRECTIVE ACTIONS OF SPECIFIC LEAK RATE TEST FAILURES AND TOTAL PRIMARY CONTAINMENT TYPE A TEST "AS FOUND" FAILURE THAT OCCURRED DURING THE FIRST REFUEL OUTAGE. THIS REPORT IS BEING SUBMITTED TO DISCUSS THE MODE OF FAILURE OF THE SUBJECT FEEDWATER CHECK VALVES AND RESIDUAL HEAT REMOVAL MOTOR OPERATED VALVE. IN ADDITION, TABLE 1 IS PROVIDED TO SUMMARIZE OTHER LLRT FAILURES WHICH OCCURRED DURING THE OUTAGE.

[160] LA SALLE 2 DOCKET 50-374 LER 87-019
FAILURE OF STATIC-O-RING DIFFERENTIAL PRESSURE SWITCH DUE TO LEAKAGE ACROSS DIAPHRAGM.
EVENT DATE: 102187 REPORT DATE: 112087 NSSS: GE TYPE: BWR
VENDOR: STATIC-O-RING

(NSIC 207190) AT 1400 HOURS ON OCTOBER 21, 1987, REACTOR CORE ISOLATION COOLING (RCIC) STEAM LINE HIGH FLOW ISOLATION SWITCH PDS-2E31-N013AA WAS FOUND TO HAVE LEAKAGE ACROSS THE DIAPHRAGM. THIS WAS DISCOVERED DURING THE PERFORMANCE OF LASALLE INSTRUMENT SURVEILLANCE LIS-RI-201, "UNIT 2 STEAM LINE HIGH FLOW RCIC ISOLATION CALIBRATION." AT THE TIME, UNIT 2 WAS INOPERATIONAL CONDITION 1 (RUN) AT 99% THERMAL POWER. A NEW DIFFERENTIAL PRESSURE SWITCH, IDENTICAL TO THE FAILED ONE, WAS CERTIFIED FOR SERVICE, INSTALLED, AND CALIBRATED ON OCTOBER 22, 1987. THE ROOT CAUSE OF THIS FAILURE HAS NOT BEEN DETERMINED. THE DEFECTIVE SWITCH HAS BEEN SENT TO ITS MANUFACTURER, STATIC-O-RING INC., FOR DISASSEMBLY AND INSPECTION. THE CONDITION OF FLOW SWITCH PDS-2E31-N013AA COMPROMISED THE OUTBOARD ISOLATION FUNCTION OF THE RCIC STEAM LINE IN THE EVENT OF A HIGH FLOW CONDITION. HOWEVER, REDUNDANT INSTRUMENTATION WAS AVAILABLE TO PROVIDE THE INBOARD ISOLATION OF THE RCIC STEAM LINE HAD AN ACTUAL HIGH FLOW CONDITION EXISTED. THIS EQUIPMENT FAILURE IS REPORTED VOLUNTARILY TO THE NUCLEAR REUGLATORY COMMISSION IN ACCORDANCE WITH THE REQUIREMENTS OF INSPECTION ENFORCEMENT BULLETIN 86-02, "STATIC-O-RING DIFFERENTIAL PRESSURE SWITCHES."

[161] LACROSSE DOCKET 50-409 LER 87-006
CONTAINMENT VENTILATION ISOLATION DUE TO CB DELAYED PARTICULATE MONITOR.
EVENT DATE: 092387 REPORT DATE: 102187 NSSS: AC TYPE: BWR
VENDOR: TRACER LAB

(NSIC 206826) CONTAINMENT VENTILATION VALVES CLOSED AUTOMATICALLY DUE TO A HIGH ACTIVITY ALARM ON THE CONTAINMENT BUILDING DELAYED PARTICULATE MONITOR. WHEN THE OPERATOR CHECKED ALL ACTIVITY MONITORS, INDICATIONS WERE NORMAL, INCLUDING THE ALARMED MONITOR. THE MONITOR WAS CHECKED OUT. NO PROBLEMS WERE FOUND THAT COULD HAVE CAUSED A SPIKE. NO ADDITIONAL SPIKING OCCURRED.

[162] LIMERICK 1 DOCKET 50-352 LER 87-015 REV 01
UPDATE ON HIGH PRESSURE COOLANT INJECTION SYSTEM INOPERABLE DUE TO LOOSE WIRE CONNECTION.
EVENT DATE: 051487 REPORT DATE: 093087 NSSS: GE TYPE: BWR
VENDOR: SCHUTTE AND KOERING COMPANY

(NSIC 206643) ON MAY 14, 1987 AT 2209 HOURS, THE HIGH PRESSURE COOLANT INJECTION (HPCI) TURBINE SHUTDOWN AFTER RUNNING FOR 25 MINUTES DURING THE PERFORMANCE OF AN HPCI SYSTEM SURVEILLANCE TEST. THE UNPLANNED HPCI TURBINE SHUTDOWN IS BELIEVED TO HAVE RESULTED FROM A LOOSE LEAD ON THE OUTPUT OF THE FLOW CONTROLLER. ALSO THE VALVE STEM OF HPCI TURBINE STOP VALVE FV-56-112 WAS DISCOVERED SEPARATED FROM THE SPLIT COUPLING WHICH CONNECTS THE VALVE STEM TO THE ACTUATOR STEM. THE THREADS INSIDE THE SPLIT COUPLING HAD BEEN STRIPPED. THE SEPARATION OF THE VALVE STEM FROM THE SPLIT COUPLING IS BELIEVED TO HAVE OCCURRED WHEN THE VALVE OPENED AND THE DISC OVERTRAVELLED DUE TO INCREASED MOMENTUM CREATED BY A DRIFT IN THE BALANCE CHAMBER ADJUSTMENT. THE BALANCE CHAMBER HAD BEEN ADJUSTED IN ACCORDANCE WITH GENERAL ELECTRIC SERVICE INFORMATION LETTER #352 IN APRIL, 1985. THE LOOSE LEAD ON THE FLOW CONTROLLER WAS TIGHTENED AND THE TIGHTNESS OF THE OTHER WIRE CONNECTIONS WERE CHECKED AND VERIFIED. THE TURBINE STOP VALVE HAS BEEN REPAIRED. THE BALANCE CHAMBER ADJUSTMENT WILL BE CHECKED QUARTERLY FOR AT LEAST THREE QUARTERS TO PREVENT RECURRENCE OF THE VALVE FAILURE. THE CONSEQUENCES OF THIS EVENT WERE MINIMAL BECAUSE THE OTHER EMERGENCY CORE COOLING SYSTEMS WERE OPERABLE AND COULD HAVE RESPONDED TO LOW REACTOR WATER LEVEL OR HIGH DRYWELL PRESSURE CONDITIONS.

[163] LIMERICK 1 DOCKET 50-352 LER 87-045
STANDBY GAS TREATMENT SYSTEM ACTUATION DUE TO FAILURE OF UNIT 2 TRANSFORMER.
EVENT DATE: 083087 REPORT DATE: 100287 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LIMERICK 2 (BWR)
VENDOR: BROWN BOVERI

(NSIC 206644) ON AUGUST 30, 1987 AT 0817 HOURS THE STANDBY GAS TREATMENT SYSTEM (SGTS) AN ENGINEERED SAFETY FEATURE (ESF) INITIATED DUE TO LOW DIFFERENTIAL PRESSURE BETWEEN THE INTERIOR AND THE EXTERIOR OF THE REACTOR ENCLOSURE. THE CAUSE OF THIS EVENT WAS A POLYPHASE FAULT TO GROUND ON A UNIT 2 LOAD CENTER TRANSFORMER WHICH PRODUCED A MOMENTARY LOW VOLTAGE CONDITION ON A UNIT 1 INSTRUMENT PANEL. THIS CAUSED A TRIP OF THE REACTOR ENCLOSURE AIR SUPPLY AND EXHAUST FANS WHICH RESULTED IN A LOSS OF DIFFERENTIAL PRESSURE BETWEEN THE REACTOR ENCLOSURE AND THE OUTSIDE ATMOSPHERE. THE STANDBY GAS TREATMENT SYSTEM THEN STARTED AS DESIGNED AND RETURNED THE DIFFERENTIAL PRESSURE TO NORMAL. THERE WERE NO ADVERSE CONSEQUENCES TO THIS EVENT. THERE WAS NO RELEASE OF RADIOACTIVE MATERIAL AS A RESULT OF THIS EVENT. IMMEDIATELY FOLLOWING THE EVENT THE FAILED TRANSFORMER WAS TRIPPED FROM THE BUS. THE ISOLATIONS WERE VERIFIED AND RESET IN ACCORDANCE WITH PROCEDURE. THE TRANSFORMER WAS RETURNED TO THE MANUFACTURER AND UNWOUND TO DETERMINE THE CAUSE OF THE FAILURE. THE DAMAGE WAS SO EXTENSIVE THAT THE CAUSE OF THE FAILURE COULD NOT BE DETERMINED. NO ACTIONS TO PREVENT RECURRENCE COULD BE IDENTIFIED FROM THIS EVENT.

[164] LIMERICK 1 DOCKET 50-352 LER 87-047
ENGINEERED SAFETY FEATURE ACTUATION DUE TO CHLORINE CONCENTRATION ABOVE ALLOWABLE LEVEL.
EVENT DATE: 091487 REPORT DATE: 101487 NSSS: GE TYPE: BWR

(NSIC 206646) ON SEPTEMBER 14, 1987 AT 1338 HOURS AN ISOLATION OF THE MAIN CONTROL ROOM HVAC OCCURRED DUE TO ELEVATED CHLORINE LEVELS DETECTED IN THE CONTROL ENCLOSURE INTAKE PLENUM. THIS ISOLATION CAUSED AN ACTUATION OF THE CONTROL ROOM EMERGENCY FRESH AIR SUPPLY (CREFAS) (AN ENGINEERED SAFETY FEATURE). THE SOURCE OF THE CHLORINE IS UNKNOWN. THERE WERE NO ADVERSE CONSEQUENCES TO THIS EVENT. THERE WAS NO RELEASE OF RADIOACTIVE MATERIAL AS A RESULT OF THIS EVENT. FOLLOWING THE EVENT ALL ELECTRONIC INDICATIONS WERE CHECKED AND VERIFIED TO BE OPERATING PROPERLY. AN AIR SAMPLE OF THE INTAKE PLENUM WAS TAKEN AT APPROXIMATELY 1430 HOURS AND INDICATED THAT NO DETECTABLE CHLORINE WAS PRESENT. THE CHLORINE DETECTORS WERE VERIFIED AS BEING AT NORMAL LEVELS AND AT 1458 HOURS THE HVAC WAS RESTORED TO NORMAL. NO ACTIONS TO PREVENT RECURRENCE HAVE BEEN IDENTIFIED FOR THIS EVENT.

[165] LIMERICK 1 DOCKET 50-352 LER 87-050
REACTOR ENCLOSURE SECONDARY CONTAINMENT ISOLATION ON LOW PRESSURE DUE TO INSTRUMENT AIR LINE LEAK.
EVENT DATE: 091787 REPORT DATE: 101987 NSSS: GE TYPE: BWR
VENDOR: MCC POWERS CORP.

(NSIC 206865) ON SEPTEMBER 17, 1987 AT 1016 HOURS, THE REACTOR ENCLOSURE SECONDARY CONTAINMENT ISOLATED AND THE STANDBY GAS TREATMENT SYSTEM (SGTS) AND THE REACTOR ENCLOSURE RECIRCULATION SYSTEM (RERS), ENGINEERED SAFETY FEATURES, INITIATED AS DESIGNED. THE ISOLATION OCCURRED WHEN THE DIFFERENTIAL PRESSURE BETWEEN THE REACTOR ENCLOSURE (RE) AND THE OUTSIDE DECREASED BLOW THE SETPOINT OF NEGATIVE 0.1 INCHES WATER GAUGE. THE SGTS MAINTAINED SECONDARY CONTAINMENT DURING THE EVENT AS DESIGNED. THE CAUSE OF THIS EVENT WAS DETERMINED TO BE A LEAK IN THE INSTRUMENT AIR LINE (NON-Q, NON-SEISMIC) TO A RE EXHAUST FAN BLADE PITCH CONTROL DEVICE. THE HOLE WAS CREATED BY COPPER TUBING VIBRATING AGAINST A TUBING CLAMP MOUNTING BOLT. THE LEAK WAS REPAIRED AND NORMAL RE VENTILATION OPERATION WAS RESTORED BY 1835 HOURS. SIMILAR INSTRUMENT AIR LINE COPPER TUBING AND CLAMPING DEVICES WILL BE INSPECTED AND REPAIRED OR REPLACED AS REQUIRED. THERE WERE NO ADVERSE CONSEQUENCS OF THIS EVENT AND THE REQUIRED SYSTEMS RESPONDED AS DESIGNED.

[166] LIMERICK 1 DOCKET 50-352 LER 87-048
REACTOR SCRAM FOLLOWING MAIN TURBINE TRIP DUE TO EHC PIPE WELD RUPTURE.
EVENT DATE: 091987 REPORT DATE: 101987 NSSS: GE TYPE: BWR

(NSIC 206806) ON SEPTEMBER 19, 1987 AT 0910 HOURS, THE REACTOR PROTECTION SYSTEM INITIATED A REACTOR SCRAM FROM 90% POWER AND A RECIRCULATION PUMP TRIP, FOLLOWING A MAIN TURBINE TRIP ON LOW ELECTRO-HYDRAULIC CONTROL (EHC) SYSTEM OIL PRESSURE. FOLLOWING THE TURBINE TRIP, OPERATION OF THE TURBINE BYPASS VALVES WAS MAINTAINED UNTIL THEIR EHC ACCUMULATOR PRESSURE BLED DOWN. REACTOR PRESSURE REACHED A PEAK VALUE OF 1093 PSIG AND REACTOR VESSEL WATER LEVEL REACHED A MINIMUM LEVEL OF MINUS 2 INCHES DURING THE EVENT. THERE WERE NO ADVERSE CONSEQUENCES AND THERE WAS NO RELEASE OF RADIOACTIVE MATERIAL RESULTING FROM THIS EVENT. THE ROOT CAUSE OF THE EVENT WAS THE FAILURE OF A TUBING SOCKET WELD IN THE EHC FLUID ACTUATING SUPPLY (FAS) LINE TO THE #3 MAIN TURBINE CONTROL VALVE (MTCV). THE SECTION OF EHC PIPE CONTAINING THE FAILED WELD WAS REMOVED FOR INSPECTION AND A NEW SECTION OF PIPE WAS WELDED INTO THE LINE. ANALYSIS OF THE MEASURED PIPING MOVEMENT AND VIBRATION LEVELS INDICATES THAT A PROPERLY BONDED WELD WOULD NOT HAVE FAILED AS A RESULT OF THE VIBRATIONS PRESENT. AS SUCH, THIS EVENT IS CONSIDERED AN ISOLATED INCIDENT.

[167] LIMERICK 1 DOCKET 50-352 LER 87-051
CONTROL ROOM HVAC ISOLATION RESULTING FROM FALSE HIGH CHLORINE CONCENTRATION SIGNAL.
EVENT DATE: 092287 REPORT DATE: 102287 NSSS: GE TYPE: BWR

(NSIC 206808) ON SEPTEMBER 22, 1987 AT 1618 HOURS, THE MAIN CONTROL ROOM VENTILATION SYSTEM ISOLATED DUE TO A FALSE 'D' CHANNEL HIGH CHLORINE CONCENTRATION SIGNAL. THE 'B' TRAIN OF THE CONTROL ROOM EMERGENCY FRESH AIR SYSTEM (AN ENGINEERED SAFETY FEATURE) INITIATED AS DESIGNED. THE EVENT OCCURRED DURING A RAIN SHOWER AND THE FALSE HIGH CHLORINE CONCENTRATION INDICATION IS BELIEVED TO HAVE BEEN CAUSED BY RAIN WATER COMING IN CONTACT WITH THE CHLORINE ANALYZER PROBE, WHICH RESULTS IN A CHEMICAL IMBALANCE IN THE PROBE'S ELECTROLYTE. THE ANALYZER PROBES ARE LOCATED CLOSE TO THE OUTSIDE AIR INTAKE LOUVERS OF THE CONTROL ENCLOSURE AIR INTAKE PLENUM. PLANT CHEMISTRY CONDUCTED SAMPLING AND DETERMINED THAT NO CHLORINE WAS PRESENT IN THE AIR INTAKE PLENUM. AFTER THE CHLORINE INDICATOR SPIKED, THE CONTROL ROOM OPERATORS VERIFIED THAT THE CHLORINE CONCENTRATION INDICATION WAS BELOW THE ALARM SETPOINT AND THE HVAC ISOLATION WAS RESET AT 2047 HOURS. THE DURATION OF THE CONTROL ROOM ISOLATION WAS 4 HOURS 29 MINUTES. THERE WERE NO ADVERSE CONSEQUENCES AND NO RELEASE OF RADIOACTIVE MATERIAL AS A RESULT OF THIS EVENT. A MODIFICATION TO CHANGE THE CHLORINE DETECTOR SYSTEM LOGIC AND LOCATION OF THE PROBES TO PREVENT FALSE CHLORINE INDICATIONS IS SCHEDULED TO BE COMPLETED BY THE END OF THE YEAR.

[168] LIMERICK 1 DOCKET 50-352 LER 87-052
REACTOR WATER CLEANUP SYSTEM ISOLATION DUE TO A DEFECTIVE DIFFERENTIAL TEMPERATURE SWITCH.
EVENT DATE: 092287 REPORT DATE: 102287 NSSS: GE TYPE: BWR
VENDOR: RILEY COMPANY, THE

(NSIC 206809) ON SEPTEMBER 22, 1987 AT 1630 HOURS AN UNPLANNED ISOLATION OF THE REACTOR WATER CLEANUP (RWCU) SYSTEM, AN ENGINEERED SAFETY FEATURE, OCCURRED DUE TO A SIGNAL FROM THE STEAM LEAK DETECTION SYSTEM. THE EVENT OCCURRED DURING THE PERFORMANCE OF A DAILY SURVEILLANCE LOG WHEN A PLANT OPERATOR PLACED THE DIFFERENTIAL TEMPERATURE SWITCH FOR THE RWCU NON-REGENERATIVE HEAT EXCHANGER ROOM IN THE 'READ' POSITION. THE RWCU NON-REGENERATIVE HEAT EXCHANGER ROOM TEMPERATURE WAS VERIFIED TO BE NORMAL AND THE RWCU SYSTEM WAS RETURNED TO SERVICE AT 1650 HOURS. THE CAUSE OF THIS EVENT WAS A DEFECTIVE DIFFERENTIAL TEMPERATURE SWITCH, MODEL 86VTFF MANUFACTURED BY THE RILEY COMPANY. WHEN THE SWITCH WAS PROPERLY PLACED IN THE READ POSITION, THE FAILSAFE ISOLATION LOGIC WAS SPURIOUSLY DE-ENERGIZED THEREBY CAUSING THE RWCU SYSTEM ISOLATION. THE DIFFERENTIAL TEMPERATURE SWITCH HAS BEEN REPLACED. THERE WERE NO ADVERSE CONSEQUENCES OF THIS EVENT OF SHORT DURATION. PREVIOUS EVENTS RESULTING FROM DEFECTIVE DIFFERENTIAL TEMPERATURE SWITCHES HAVE BEEN REPORTED.

[169] LIMERICK 1 DOCKET 50-352 LER 87-053
FAILURE TO SATISFACTORILY COMPLETE SURVEILLANCE TEST OF THE ROD WORTH MINIMIZER WITHIN THE SPECIFIED TIME.
EVENT DATE: 092687 REPORT DATE: 102687 NSSS: GE TYPE: BWR

(NSIC 206810) ON SEPTEMBER 26, 1987 AT 1912 HOURS OPERATION OF THE ROD WORTH MINIMIZER (RWM) WAS NOT IN COMPLIANCE WITH THE REQUIREMENTS OF TECHNICAL SPECIFICATION 4.1.4.A, SINCE SURVEILLANCE TEST (ST) ST-6-073-320-1 (ROD WORTH MINIMIZER OPERABILITY VERIFICATION) WAS NOT SATISFACTORILY COMPLETED WITHIN ONE HOUR AFTER RWM AUTOMATIC INITIATION. THE ST WAS NOT SATISFACTORILY COMPLETED BECAUSE A PAGE OF THE TEST WAS MISSING FROM THE COPY GIVEN TO THE OPERATORS. THE MISSING PAGE WAS DISCOVERED BY THE TECHNICAL ASSISTANT ON THE NEXT SHIFT DURING HIS REVIEW OF THE PARTIALLY COMPLETED ST. THERE WERE NO ADVERSE CONSEQUENCES ASSOCIATED WITH THIS EVENT. THERE WAS NO RELEASE OF RADIOACTIVE MATERIAL AS A RESULT OF THIS EVENT. THE MISSING SECTION OF THE ST WAS SATISFACTORILY PERFORMED

AT 0330 HOURS ON SEPTEMBER 27, 1987 TO VERIFY OPERABILITY OF THE RWM 'SELECT ERROR' LIGHT, COMPLETING THE FIRST ST. THE NON-COMPLIANCE EXISTED FOR 8 HOURS AND 18 MINUTES. A MEMO WAS DISTRIBUTED TO ALL SHIFT PERSONNEL EMPHASIZING THE IMPORTANCE OF CHECKING COPIES OF DOCUMENTS FOR CORRECTIONS AND COMPLETENESS. IN ADDITION, A REVISION WAS MADE TO THE CONTROL OF THE PREPARATION OF CONTROL ROOM COPIES TO REGULATE THE QUALITY OF COPIES, EFFECTIVE OCTOBER 19, 1987.

[170] LIMERICK 1 DOCKET 50-352 LER 87-054
FAILURE TO PERFORM SURVEILLANCE TEST WITHIN THE ALLOTTED TIME AS REQUIRED BY TECHNICAL SPECIFICATIONS DUE TO A PERSONNEL ERROR.
EVENT DATE: 092787 REPORT DATE: 102887 NSSS: GE TYPE: BWR

(NSIC 206878) ON SEPTEMBER 28, 1987 A REVIEW OF THE SURVEILLANCE TEST (ST) SCHEDULE DETERMINED THAT A SURVEILLANCE TEST HAD NOT BEEN PERFORMED WITHIN THE REQUIRED TIME PERIOD. THE INSTRUMENT AND CONTROL ST COORDINATOR HAD ASSIGNED AN INCORRECT DATE FOR THE PERFORMANCE OF THE ST. THEREFORE, THE MONTHLY FUNCTIONAL TEST OF THE 'E' CHANNEL FOR THE CONTROL ROD BLOCK FOR THE SCRAM DISCHARGE VOLUME WAS NOT COMPLETED WITHIN THE REQUIRED TIME PERIOD. THE ST (ST-2-047-614-1 - CRD BLOCK - SCRAM DISCHARGE VOLUME WATER LEVEL HIGH; CHANNEL 'E' CALIBRATION/FUNCTIONAL TEST - LSH- 47-1N013E) WAS 17 HOURS OVERDUE WHEN IT WAS SATISFACTORILY COMPLETED ON SEPTEMBER 28, 1987 AT 1455 HOURS. THERE WAS NO RELEASE OF RADIOACTIVE MATERIAL OR ADVERSE CONSEQUENCES ASSOCIATED WITH THIS EVENT. THE WEEKLY ST PERFORMANCE SCHEDULE LIST WILL BE MORE CLOSELY MONITORED TO ASSURE THAT THE STS ARE PERFORMED ON SCHEDULE.

[171] LIMERICK 1 DOCKET 50-352 LER 87-056
REACTOR ENCLOSURE VENTILATION ISOLATION DUE TO LOW DIFFERENTIAL PRESSURE.
EVENT DATE: 101487 REPORT DATE: 111687 NSSS: GE TYPE: BWR

(NSIC 206935) ON OCTOBER 14, 1987, AT 0410 HOURS, A REACTOR ENCLOSURE VENTILATION ISOLATION (AN ENGINEERED SAFETY FEATURE) OCCURRED RESULTING FROM LOW DIFFERENTIAL PRESSURE, WITH THE STANDBY GAS TREATMENT SYSTEM (SGTS) AND THE REACTOR ENCLOSURE RECIRCULATION SYSTEM (RERS) INITIATING AS DESIGNED. THE LOW DIFFERENTIAL PRESSURE CONDITION WAS CREATED BY THE LOSS OF AUXILIARY STATION HEATING STEAM IN THE INLET PLENUM, ALLOWING COOL OUTSIDE AIR TO ENTER, WARM, AND RAPIDLY EXPAND IN THE REACTOR ENCLOSURE. THIS OVERWHELMED THE EXHAUST FAN'S ABILITY TO CONTROL THE REACTOR ENCLOSURE AIR PRESSURE. THE SUBSEQUENT DECREASE IN REACTOR ENCLOSURE TO ATMOSPHERE DIFFERENTIAL PRESSURE RESULTED IN THE ISOLATION. THE AUXILIARY STEAM WAS RESTORED, THE LOW DIFFERENTIAL PRESSURE ISOLATION WAS RESET, AND NORMAL REACTOR ENCLOSURE VENTILATION WAS RESTORED AT 0436 HOURS. THE DURATION OF THE REACTOR ENCLOSURE ISOLATION WAS 26 MINUTES. THERE WERE NO ADVERSE CONSEQUENCES AND NO RELEASE OF RADIOACTIVE MATERIAL TO THE ENVIRONMENT AS A RESULT OF THIS EVENT. THE ENGINEERING DIVISION IS EVALUATING WAYS OF IMPROVING THE RELIABILITY OF THE AUXILIARY BOILERS.

[172] LIMERICK 1 DOCKET 50-352 LER 87-057
FALSE CONTROL ROOM HIGH RADIATION SIGNAL DURING TROUBLESHOOTING OF FAILED CHECK SOURCE ALARM ON RADIATION MONITOR.
EVENT DATE: 101787 REPORT DATE: 111687 NSSS: GE TYPE: BWR
VENDOR: GENERAL ATOMIC CO.

(NSIC 206936) ON OCTOBER 17, 1987 AT 0840 HOURS A FALSE CHANNEL 'A' CONTROL ROOM HIGH RADIATION SIGNAL INITIATED AFTER INSTRUMENT AND CONTROL (I & C) TECHNICIANS CLOSED THE BYPASS SWITCH ON GENERAL ATOMIC MODEL 3801 CONTROL ROOM VENTILATION RADIATION MONITOR RIX-26-0074-1 AND CYCLED THE POWER TO IT, DURING TROUBLESHOOTING OF A FAILED CHECK SOURCE. VENTILATION DAMPERS REPOSITIONED AS DESIGNED IN RESPONSE TO THE SIGNAL, CAUSING A PARTIAL ISOLATION OF THE CONTROL ROOM VENTILATION. THE HIGH RADIATION SIGNAL WAS VERIFIED TO BE FALSE, AND WAS

RESET, PER GENERAL PROCEDURE GP8. THE AFFECTED DAMPERS WERE RETURNED TO THEIR NORMAL POSITIONS. AT 1635 HOURS, OPERABILITY OF CONTROL ROOM VENTILATION RADIATION MONITOR RIX-26-0074-1 WAS VERIFIED. THE MONITOR WAS OUT OF SERVICE FOR 10 HOURS AND 56 MINUTES. THE ROOT CAUSE OF THE EVENT WAS INADEQUATE INSTRUCTION OF THE I&C TECHNICIANS CONCERNING THE CONTROL ROOM VENTILATION RADIATION MONITOR BYPASS SWITCH. THE TECHNICIANS ASSUMED THAT PLACING THE BYPASS SWITCH ON THE MONITOR IN THE "BYPASS POSITION" WOULD INHIBIT THE TRIP FUNCTIONS, AND IT DID NOT. THERE WERE NO ADVERSE CONSEQUENCES AND THERE WAS NO RELEASE OF RADIOACTIVE MATERIAL AS A RESULT OF THIS EVENT.

[173] MCGUIRE 1 DOCKET 50-369 LER 87-017
A SAFETY INJECTION/REACTOR TRIP OCCURRED DUE TO A DESIGN DEFICIENCY OF THE MAIN TURBINE CONTROLS FOLLOWED BY SEVERAL EQUIPMENT MALFUNCTIONS.
EVENT DATE: 081687 REPORT DATE: 092587 NSSS: WE TYPE: PWR
VENDOR: CROSBY VALVE & GAGE CO.
FISHER CONTROLS CO.
ROTORK INC.
WESTINGHOUSE ELECTRIC CORP.

(NSIC 206512) ON AUGUST 16, 1987, WHILE PUTTING THE UNIT 1 TURBINE GENERATOR ON LINE, A MALFUNCTION OF THE TURBINE CONTROLS TRIGGERED A SAFETY INJECTION/REACTOR TRIP. AN UNUSUAL EVENT WAS DECLARED. ALSO VALVE 1NV-142 (VOLUME CONTROL TANK ISOLATION) ACTUATOR BURNED UP AFTER PERFORMING ITS SAFETY FUNCTION. A REACTOR COOLANT LEAK (APPROXIMATELY 40 GPM) ALSO OCCURRED DUE TO A CRACK IN THE LETDOWN LINE AT A DRAIN LINE. THE LEAK WAS STOPPED. THE UNIT REMAINED DOWN FOR REPAIRS. THE ACTUATOR FOR VALVE 1NV-142 WAS REPLACED. THE LETDOWN LEAK WAS REPAIRED. SIMILAR UNIT 1 WELDS ON VENTS AND DRAINS WERE INSPECTED - ONE CRACK FOUND AND REPAIRED. UNIT 2 SIMILAR WELDS WERE INSPECTED - NO PROBLEMS FOUND. SIXTEEN PIPE SUPPORTS WERE ALSO INSPECTED - NO DAMAGE FOUND. THE TURBINE CONTROLS ON UNIT 1 ARE SCHEDULED TO BE REPLACED; UNIT 2 TURBINE CONTROLS HAVE ALREADY BEEN REPLACED. THIS EVENT HAS BEEN ATTRIBUTED TO DESIGN DEFICIENCY BECAUSE THE TURBINE CONTROLS PERFORMED IN AN [illegible] MANNER. THE DESIGN OF THE CONTROLS HAD PREVIOUSLY BEEN DETERMINED TO MERIT BEING REPLACED. THE UNIT WAS PLACED BACK ON LINE AUGUST 21, 1987.

[174] MCGUIRE 1 DOCKET 50-369 LER 87-018
UNIT 1 NUCLEAR SERVICE WATER FLOW BALANCE TESTS INVALIDATED - INCORRECT CALIBRATION OF UNIT 2 RN FLOW INSTRUMENTATION - CALIBRATION PROCEDURE OVERLOOKED DURING REVIEW.
EVENT DATE: 082587 REPORT DATE: 092487 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)
VENDOR: BINGHAM-WILLAMETTE CO.
ROSEMOUNT, INC.

(NSIC 206462) ON AUGUST 24, 1987 PERFORMANCE (PRF) NOTICED THAT THE NUCLEAR SERVICE WATER PUMP 2A PERFORMANCE HAD INCREASED USING THE OPERATOR AID COMPUTER (OAC) FLOW DATA. UPON INVESTIGATING THE FLOW INSTRUMENT CALIBRATIONS, PRF DETERMINED THE FLOW INSTRUMENT CALIBRATIONS FOR RN PUMPS 2A AND 2B WERE INCORRECT. THIS EVENT WAS DUE TO INADEQUATE POST MODIFICATION REVIEW OF PROCEDURES AND THE LACK OF ADEQUATE CONTROLS OVER PROCEDURES BECAUSE THE CALIBRATIONS PERFORMED ON THE FLOW INSTRUMENTS FOR RN PUMPS 2A AND 2B ON FEBRUARY 25, 1987 WERE PERFORMED UNDER THE RN PUMP FLOW INSTRUMENT CALIBRATION PROCEDURE, WHICH CONTAINED ERRONEOUS INSTRUMENT DATA. THE DATA WAS NOT UPDATED TO REFLECT A COMPLETED NUCLEAR STATION MODIFICATION WHICH CHANGED THE UNIT 2 RN PUMP FLOW ELEMENTS AND ASSOCIATED CALIBRATION DATA. THESE ERRORS IN CALIBRATION RESULTED IN DECREASED FLOWS FROM THE OAC INDICATED FLOW BY APPROXIMATELY 550 GPM AT A DISCHARGE FLOW RATE OF 6000 GPM. THE LOWER FLOW THROUGH THE UNIT 2 RN PUMPS INVALIDATED THE UNIT 1 RN FLOW BALANCE TESTS (FEBRUARY 25, 1987) DUE TO THE REQUIREMENT FOR 6000 GPM MINIMUM FLOW TO BE SUPPLIED FROM THE CORRESPONDING UNIT

2 TRAIN TO UNIT 2 LOADS. ON AUGUST 25, PRF PERFORMED FLOW BALANCE RETESTS FOR RN TRAINS 1A AND 1B.

[175] MCGUIRE 1 DOCKET 50-369 LER 87-019
AUXILIARY FEEDWATER AUTOSTART OCCURRED DUE TO A LACK OF SUFFICIENT PROCEDURAL INSTRUCTIONS TO PREVENT LOSS OF BOTH MAIN FEEDWATER PUMPS DURING SHUTDOWN.
EVENT DATE: 090487 REPORT DATE: 100587 NSSS: WE TYPE: PWR
VENDOR: WORTHINGTON PUMP CORP.

(NSIC 206655) ON SEPTEMBER 4, 1987 AT 0152, OPERATIONS (OPS) TRIPPED MAIN FEEDWATER (CF) PUMP 1A AT 42% REACTOR POWER. UNIT 1 ENTERED MODE 3, COLD SHUTDOWN. AT 0509, WHILE OPS ATTEMPTED TO OPEN 1HM95 (AUXILIARY STEAM SUPPLY TO CF PUMP TURBINES), CF PUMP 1B TRIPPED. BOTH AUXILIARY FEEDWATER (CA) MOTOR DRIVEN PUMPS STARTED, CA PUMP DISCHARGE VALVES OPENED FULLY, NUCLEAR SERVICE WATER (RN) PUMP 1A STARTED, AND STEAM GENERATOR BLOWDOWN ISOLATED. OPS RESET CA DISCHARGE BUT WERE UNABLE TO RESET CF PUMPS 1A AND 1B. AT 0605, OPS RESET CF PUMP 1B AND STOPPED THE CA PUMPS. AT 0625, OPS STOPPED RN PUMP 1A AND REESTABLISHED BLOWDOWN. THE CAUSE OF THE EVENT WAS ATTRIBUTED TO DEFECTIVE PROCEDURE (CONTROLLING PROCEDURE FOR UNIT SHUTDOWN AND CONTROLLING PROCEDURE FOR UNIT OPERATION). CF PUMP 1B WAS SHUTDOWN DURING POWER REDUCTION AND WAS LEFT IN THE "TRIPPED" CONDITION PER PROCEDURE. PRECAUTIONS TO BE TAKEN WHEN OPENING 1HM95 WERE NOT INCLUDED IN THE PROCEDURE. BOTH PROCEDURES WILL BE REVISED TO PREVENT RECURRENCE OF THE SAME TYPE EVENT. THE REASON CF PUMP 1A WOULD NOT RESET WILL BE INVESTIGATED AND APPROPRIATE REPAIRS MADE.

[176] MCGUIRE 1 DOCKET 50-369 LER 87-020
RESIDUAL HEAT REMOVAL PUMP 1B INOPERABLE BECAUSE THE FLOW INSTRUMENT CONTROLLING THE PUMP RECIRC VALVE WAS LEFT ISOLATED DUE TO PERSONNEL ERROR.
EVENT DATE: 090587 REPORT DATE: 102787 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 206817) ON 08/19/87 A WORK REQUEST (WR) WAS ISSUED TO CALIBRATE FLOW INSTRUMENTS 1MNDPG5050 AND 1MNDPG5051. ON 08/28/87 OPERATIONS (OPS) EVALUATED THE WR AND AUTHORIZED THE WORK TO BEGIN. THAT EVENING INSTRUMENT AND ELECTRICAL (IAE) COMPLETED THE CALIBRATION WORK BUT FAILED TO UNISOLATE THE FLOW INSTRUMENT 1MNDPG5050 (USED TO OPERATE THE RESIDUAL HEAT REMOVAL (ND) PUMP 1B RECIRC VALVE). AT 0005 ON 09/05/87, OPS STARTED ND PUMP 1B AND THE PUMP RECIRC VALVE FAILED TO OPEN. OPS DECLARED ND TRAIN 1B INOPERABLE AND INITIATED A WORK REQUEST. IAE DISCOVERED THE UNISOLATED FLOW INSTRUMENT AT 0146 AND UNISOLATED THE VALVE. OPS FUNCTIONALLY VERIFIED ND TRAIN 1B OPERABLE AND DECLARED IT OPERABLE AT 0230. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR WITH CONTRIBUTORY MANAGEMENT DEFICIENCY. THE STANDING WORK REQUESTS FOR THE FLOW INSTRUMENTS WERE REVISED WITH PRECAUTIONS THAT THE ITEM IS RELATED TO TECH SPECS. THE EVENT WILL BE COVERED DURING REQUAL TRAINING WITH ALL OPS SHIFT PERSONNEL. THIS EVENT WILL BE EVALUATED TO DETERMINE IF THERE ARE GENERIC IMPLICATIONS. STATION DIRECTIVE 4.2.1 WILL BE REVISED REGARDING USE OF PROCEDURES.

[177] MCGUIRE 1 DOCKET 50-369 LER 87-021
UNIT 1 BLACKOUT WHILE TESTING LOCKOUT RELAYS AND UNIT 2 REACTOR TRIP DUE TO LOSS OF INSTRUMENT AIR - ROOT CAUSE WAS PERSONNEL ERROR.
EVENT DATE: 091687 REPORT DATE: 102987 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 206879) ON SEPTEMBER 16, 1987, INSTRUMENT AND ELECTRICAL (IAE) AND RELAY PERSONNEL WERE PERFORMING A TEST TO VERIFY OPERATION OF LOCKOUT RELAYS AND LIGHTS ON WHICH WIRING MODIFICATIONS HAD BEEN PERFORMED. AT 1016, WHILE A LOCKOUT RELAY AND ITS ASSOCIATED LIGHTS WERE BEING TESTED, 2 ASSOCIATED RELAYS ENERGIZED CAUSING THE 1B BUSLINE BREAKERS TO TRIP RESULTING IN A UNIT 1 BLACKOUT. THE

BLACKOUT CAUSED A LOSS OF POWER TO INSTRUMENT AIR (VI) COMPRESSORS WHICH RESULTED IN THE CLOSURE OF THE UNIT 2 MAIN FEEDWATER REGULATOR VALVES. UNIT 2 SUBSEQUENTLY TRIPPED ON LOW-LOW STEAM GENERATOR LEVEL AT 1021. OPERATIONS RESTORED NORMAL POWER TO UNIT 1 BY ENERGIZING THE 1B BUSLINE AND STABILIZED UNIT 2 FROM THE REACTOR TRIP BY 1045. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR. PLANNED CORRECTIVE ACTIONS INVOLVING EVALUATION OF VI COMPRESSOR ALIGNMENTS, COMMUNICATING IMPORTANCE OF USING AND NOT EXCEEDING PROCEDURES, AND A DESIGN REVIEW OF THE POWER SUPPLY ALIGNMENTS FOR VI COMPRESSORS HAVE BEEN INITIATED.

[178] MCGUIRE 1 DOCKET 50-369 LER 87-022
A FIRE BARRIER BLANKET WAS BREACHED WITHOUT COMPENSATORY ACTION DUE TO PERSONNEL NOT RECOGNIZING THE BLANKET AS A FIRE BARRIER.
EVENT DATE: 092187 REPORT DATE: 102687 NSSS: WE TYPE: PWR

(NSIC 206818) ON SEPTEMBER 21, 1987 DURING AN ANNUAL FIRE PROTECTION AUDIT, A BREACHED FIRE BLANKET WAS DISCOVERED. THE FIRE BLANKET COVERED AN AUXILIARY FEEDWATER (CA) PRESSURE SWITCH. THE BLANKET HAD BEEN CUT TO ALLOW PENETRATION TO FACILITATE MAINTENANCE ON THE PRESSURE SWITCH INSTRUMENT. IT WAS DETERMINED THAT THE BREACH OCCURRED ON SEPTEMBER 12, 1987 WHEN INSTRUMENT AND ELECTRICAL PERSONNEL, WHO WERE UNAWARE THE BLANKET WAS A FIRE BARRIER, CUT THE BLANKET TO PERFORM PREVENTATIVE MAINTENANCE. THE CAUSE OF THE EVENT WAS ATTRIBUTED TO MANAGEMENT DEFICIENCY BECAUSE NO TRAINING HAD BEEN GIVEN TO THE TECHNICIANS TO ENABLE THEM TO RECOGNIZE THE COVERING (BLANKET) AS A FIRE BARRIER. PLACARDS WERE ATTACHED TO THE FIRE BLANKETS OVER THE INSTRUMENTATION IN BOTH UNIT 1 AND 2 CA ROOMS IDENTIFYING THEM AS FIRE BARRIERS. TRAINING ALREADY IN PLACE AS A RESULT OF A PREVIOUS EVENT WILL BE MODIFIED TO EMPHASIZE RECOGNIZING FIRE BARRIERS. INSTRUMENTS WHICH HAVE FIRE BLANKET COVERINGS WILL BE IDENTIFIED ON THEIR PREVENTATIVE MAINTENANCE/PERIODIC TESTS WORK REQUESTS AS SUCH TO AID IN IDENTIFYING THIS TYPE OF FIRE BARRIER.

[179] MCGUIRE 1 DOCKET 50-369 LER 87-023
CONTROL ROOM VENTILATION/CHILLED WATER SYSTEM TRAIN A AND B INOPERABLE - TRAIN A DUE TO BLOWN FUSE: TRAIN B DUE TO FAILED ACTUATOR MICROSWITCH.
EVENT DATE: 100187 REPORT DATE: 110287 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)
VENDOR: ITT GENERAL CONTROLS

(NSIC 206880) AT 1730 ON 10/01/87 THE CONTROL ROOM VENTILATION/CHILLED WATER (VC/YC) TRAIN A CHILLER STOPPED DUE TO A LOSS OF ESSENTIAL CONTROL POWER AND WAS DECLARED INOPERABLE. VC/YC TRAIN B WAS STARTED BUT AT 1745 OPERATIONS (OPS) NOTED THE CONTROL ROOM AIR HANDLING UNIT SUCTION DAMPER WAS NOT OPEN. TRAIN B WAS THEN DECLARED INOPERABLE. UNIT 1 AND 2 ENTERED TECH SPEC 3.0.3 AT THAT TIME. AT 1830, OPS STARTED REDUCING UNIT 2 POWER. THE REASON TRAIN A CHILLER TRIPPED (BLOWN FUSE) WAS DISCOVERED AND REPAIRED USING A TEMPORARY MODIFICATION TO INSTALL A HIGHER RATED FUSE. TRAIN A WAS THEN DECLARED OPERABLE, STARTED, AND UNIT 1 AND 2 EXITED TECH SPEC 3.0.3. THE ROOT CAUSE OF THE FAILURE WAS AN ACTUATOR MICROSWITCH THAT DISABLED TRAIN B. THE TRAIN A FUSE THAT BLEW WAS ATTRIBUTED TO DESIGN DEFICIENCY BECAUSE THE FUSE RATING WAS INSUFFICIENT. THE TEMPORARY MODIFICATION WAS REMOVED AND A PERMANENT MODIFICATION WAS MADE.

[180] MCGUIRE 2 DOCKET 50-370 LER 87-014
BOTH TRAINS OF CONTAINMENT SPRAY WERE DECLARED INOPERABLE WHEN NUCLEAR SERVICE WATER PUMP WAS TRIPPED BY MALFUNCTION OF OVERCURRENT RELAY.
EVENT DATE: 090187 REPORT DATE: 100187 NSSS: WE TYPE: PWR
VENDOR: I-T-E CIRCUIT BREAKER

(NSIC 206656) AT 1350 ON SEPTEMBER 1, 1987, UNIT 2 ENTERED TECHNICAL SPECIFICATION (TS) 3.0.3 WHEN BOTH TRAINS OF THE CONTAINMENT SPRAY SYSTEM (NS)

BECAME INOPERABLE. NS PUMP 2B HAD BEEN DECLARED INOPERABLE TO PERFORM ROUTINE MAINTENANCE ON ITS ROOM AIR HANDLING UNIT. NUCLEAR SERVICE WATER (RN) PUMP 2A WAS TRIPPED BY AN INSTANTANEOUS OVERCURRENT PROTECTIVE RELAY, FOR NO APPARENT REASON, THEREBY RENDERING NS PUMP 2A INOPERABLE. NS PUMP 2B WAS PHYSICALLY AVAILABLE AT THAT TIME AND WAS DECLARED OPERABLE AT 1430, AND UNIT 2 EXITED TECHNICAL SPECIFICATION 3.0.3. THE RELAY WAS BENCH TESTED AND THE MOTOR WINDINGS, CABLES, AND RELATED ELECTRICAL EQUIPMENT FOR RN PUMP 2A WERE CHECKED WITHOUT FINDING ANY DISCREPANCY OR PROBABLE CAUSE FOR THE OVERCURRENT TRIP. OPERATIONS STARTED THE PUMP AGAIN AT 1520 WITH RELAY MAINTENANCE PERSONNEL IN ATTENDANCE, AND IT FUNCTIONED NORMALLY. HOWEVER, THE PUMP TRIPPED AGAIN ON INSTANTANEOUS OVERCURRENT AT 1616. THE INSTANTANEOUS OVERCURRENT PROTECTIVE RELAY WAS CHANGED OUT AND THE PUMP WAS RETURNED TO OPERATION AT 2224. A CLASSIFICATION OF OTHER HAS BEEN ASSIGNED TO THIS EVENT BECAUSE OF THE APPARENT MALFUNCTION OF THE INSTANTANEOUS OVERCURRENT PROTECTIVE RELAY.

[181] MCGUIRE 2 DOCKET 50-370 LER 87-015
UNIT OPERATED ABOVE RATED THERMAL POWER DUE TO CALCULATED POWER BEING INCORRECT CAUSED BY PERSONNEL ERROR THUS MAIN FEEDWATER FLOW COEFFICIENT NOT REVISED.
EVENT DATE: 090287 REPORT DATE: 100287 NSSS: WE TYPE: PWR

(NSIC 206657) ON SEPTEMBER 1, 1987, UNIT 2 EXPERIENCED A TURBINE RUNBACK. THE UNIT RETURNED TO FULL INDICATED POWER AT 0500 THE NEXT DAY. BY 0815, PERFORMANCE (PRF) HAD ENTERED THE BASELINE FLOW COEFFICIENTS FOR THE VENTURI FLOW ELEMENTS (FOULING FACTOR USED TO MEASURE MAIN FEEDWATER FLOW FOR THERMAL POWER CALCULATIONS). ON SEPTEMBER 3, PRF COMPLETED THE VENTURI FOULING TEST WHICH INDICATED THAT 0500 - 0815 ON SEPTEMBER 2, THE UNIT WAS AT AN INDICATED 100% REACTOR THERMAL POWER (RTP), WHEN THE POWER LEVEL WAS ACTUALLY 100.6%. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR BECAUSE THE RESPONSIBLE ENGINEER DID NOT ENTER THE BASELINE FLOW COEFFICIENTS INTO THE OPERATOR AID COMPUTER (OAC) PRIOR TO RETURN TO 100% RTP. PRF REENTERED THE BASELINE VENTURI FLOW COEFFICIENT IN THE OAC AT 0815 ON SEPTEMBER 2. PRF REVISED THE TRANSIENT/REACTOR TRIP INVESTIGATION PROCEDURE TO INCLUDE A STEP TO RESET THE VENTURI FLOW COEFFICIENT TO BASELINE VALUES UNTIL ANOTHER VENTURI FOULING TEST IS COMPLETED TO PROVIDE ADDITIONAL PROTECTION AGAINST RECURRENCE OF THIS EVENT BY NOT REQUIRING A CASE BY CASE DETERMINATION.

[182] MCGUIRE 2 DOCKET 50-370 LER 87-016
REACTOR TRIP DUE TO OVERCURRENT FAULTS IN AN INSTRUMENT AIR COMPRESSOR MOTOR - CAUSED LOSS OF POWER TO A MAIN TURBINE CONTROL SYSTEM RELAY.
EVENT DATE: 090687 REPORT DATE: 100687 NSSS: WE TYPE: PWR
VENDOR: RELIANCE ELECTRIC COMPANY

(NSIC 206658) AT 100% POWER ON SEPTEMBER 6, 1987 AT 1035, A UNIT 2 REACTOR/TURBINE TRIP OCCURRED DUE TO HIGH PRESSURIZER PRESSURE WHEN MAIN TURBINE GOVERNOR AND INTERCEPT VALVES CLOSED AS DIRECTED BY THE DIGITAL ELECTRO-HYDRAULIC (DEH) TURBINE CONTROL SYSTEM. THE GOVERNOR AND INTERCEPT VALVE CLOSE SIGNAL WAS GENERATED BY LOSS OF POWER TO A DEH TURBINE CONTROL SYSTEM RELAY WHEN POWER WAS LOST TO KXB. POWER WAS LOST ON AUXILIARY POWER PANELBOARD KXB DUE TO AN OVERCURRENT FAULT BREAKER TRIP CAUSED BY A GROUNDED MOTOR LEAD CONNECTOR (INSULATING TAPE HAD WORN ALLOWING CONNECTING LUG TO GROUND TO MOTOR FRAME) ON INSTRUMENT AIR (VI) COMPRESSOR A. OPERATIONS IMPLEMENTED THE REACTOR TRIP PROCEDURE. POWER WAS RESTORED TO AUXILIARY POWER PANELBOARD KXB FROM STATIC INVERTER KXB. UNIT 2 RETURNED TO MODE 1, POWER OPERATION, ON SEPTEMBER 7, AT 2110. THE CONNECTING LUG WAS REINSULATED IN THE CONNECTION BOX AND THE COMPRESSOR WAS RETURNED TO SERVICE. VI COMPRESSOR MOTORS B&C WILL BE INSPECTED FOR SIMILAR CONDITION. SIMILAR MOTORS IN OTHER APPLICATIONS WILL BE INSPECTED AND RETAPED AS NECESSARY.

[183] MCGUIRE 2 DOCKET 50-370 LER 87-017
UNIT ENTERED TECH SPEC 3.0.3 TO PERFORM SOLID STATE PROTECTION SYSTEM TRAIN 2B TESTING WHILE NUCLEAR SERVICE WATER TRAIN 2A WAS INOPERABLE.
EVENT DATE: 090687 REPORT DATE: 100687 NSSS: WE TYPE: PWR

(NSIC 206659) ON SEPTEMBER 4, 1987, PERFORMANCE (PRF) CONDUCTED A FLOW BALANCE TEST ON NUCLEAR SERVICE WATER (RN) TRAIN 2A. THE TEST FAILED DUE TO INADEQUATE RN FLOWS THROUGH COMPONENT COOLING (KC) HEAT EXCHANGER (HX) 2A, CONTAINMENT SPRAY (NS) HX 2A, AND THE RESIDUAL HEAT REMOVAL (ND) PUMP 2A AIR HANDLING UNIT. PRF INFORMED OPERATIONS (OPS) AND OPS DECLARED RN TRAIN 2A INOPERABLE AT 1950. ON SEPTEMBER 6, UNIT 2 HAD A REACTOR TRIP AND SOLID STATE PROTECTION SYSTEM (SSPS) TRAIN 2B TESTING WAS REQUIRED PRIOR TO RETURNING THE UNIT TO POWER. IF SSPS TRAIN 2B WAS DECLARED INOPERABLE, RN SYSTEM 2B WOULD ALSO BE RENDERED INOPERABLE. WITH RN TRAIN 2A STILL INOPERABLE AND RN TRAIN 2B INOPERABLE, THE REQUIREMENTS OF TECH SPEC (T.S.) 3.7.4 COULD NOT BE MET. OPS SUPERVISION REVIEWED THE SITUATION AND PLACED UNIT 2 IN T.S. 3.0.3 WHEN OPS DECLARED SSPS TRAIN 2B INOPERABLE AT 1810. UNIT 2 WAS ALREADY IN MODE 3, HOT STANDBY, WHICH ALLOWED 7 HOURS BEFORE ANY FURTHER ACTION WOULD BE REQUIRED. INSTRUMENTATION AND ELECTRICAL COMPLETED TESTING OF SSPS TRAIN 2B AT 1915 AND UNIT 2 EXITED T.S. 3.0.3. KC HX 2A WAS MECHANICALLY CLEANED BY MECHANICAL MAINTENANCE. PRF CONDUCTED A SUCCESSFUL FLOW BALANCE RETEST OF RN TRAIN 2A. THE UNIT ENTERED MODE 2, STARTUP AT 1827 AND MODE 1, POWER OPERATION, AT 2110 ON SEPTEMBER 7, 1987. THIS EVENT HAS BEEN CLASSIFIED AS OTHER.

[184] MCGUIRE 2 DOCKET 50-370 LER 87-018
INOPERABLE FIRE BARRIER DUE TO A WALL SECTION BEING CONSTRUCTED WITHOUT PROPER END CONNECTION TREATMENT.
EVENT DATE: 092187 REPORT DATE: 102687 NSSS: WE TYPE: PWR

(NSIC 206819) ON SEPTEMBER 21, 1987 A QUALITY ASSURANCE (QA) INSPECTOR DISCOVERED SMALL GAPS BETWEEN CONCRETE COLUMNS AND THE ADJOINING WALL SECTION BETWEEN TWO ROOMS ON THE 733' ELEVATION OF THE AUXILIARY BUILDING. THE FIRE BARRIER WAS DECLARED INOPERABLE AT 1430 AND A FIRE WATCH WAS INITIATED. ON SEPTEMBER 23, 1987 THE FIRE BARRIER WAS REPAIRED IN ACCORDANCE WITH FIELD SEALANT SPECIFICATION 7005 AND WAS DECLARED OPERABLE AT 1230. THE CAUSE OF THE EVENT WAS CONSTRUCTION/INSTALLATION DEFICIENCY BECAUSE THE FIRE BARRIER SECTION WAS CONSTRUCTED WITHOUT PROPER END CONNECTION TREATMENT. A CONTRIBUTORY CAUSE OF QA DEFICIENCY HAS BEEN ASSIGNED BECAUSE OF QA FAILURE TO VERIFY THAT THE WALL SECTION WAS CONSTRUCTED ACCORDING TO APPLICABLE DRAWINGS. THE QA INSPECTION PROCEDURE USED DURING THE CONSTRUCTION (PRE-STARTUP) HAS BEEN INACTIVE SINCE APRIL 1987. THE FIRE BARRIER INSPECTION PROCEDURE WILL BE REVISED TO ENSURE THAT THE ENTIRE FIRE BARRIER IS INSPECTED (NOT JUST PENETRATIONS). THE QA CONDITION 3 FIRE WALL REPAIR PROCEDURE WILL BE REVISED TO INCLUDE SPECIFIC PROVISIONS FOR REPAIR OF GAPS BETWEEN GYPSUM DRYWALL AND CONCRETE. A COMPLETE INSPECTION OF GYPSUM DRYWALL FIRE BARRIERS WILL BE CONDUCTED TO VERIFY THEIR INTEGRITY AND/OR ANY REMAINING CONSTRUCTION DEFECTS. MAINTENANCE PROCEDURES WILL ALSO BE INCLUDED IN ANY FUTURE QA INSPECTIONS OF FIRE BARRIER WALLS FOR PROPER ACCEPTANCE CRITERIA.

[185] MILLSTONE 1 DOCKET 50-245 LER 86-006 REV 02
UPDATE ON FAILURE OF CONTAINMENT ISOLATION VALVES TO CLOSE.
EVENT DATE: 020686 REPORT DATE: 102987 NSSS: GE TYPE: BWR
VENDOR: NAMCO CONTROLS
NUMATICS

(NSIC 206889) ON FEBRUARY 6, 1986, AT 0240 HOURS WHILE AT 0% POWER (508F 721 PSI), VALVE 1-MS-1D PRODUCED DUAL POSITION INDICATION WHEN ITS CONTROL SWITCH WAS PLACED IN THE 'CLOSE' POSITION. THE BACKUP CONTAINMENT ISOLATION VALVE, 1-MS-2D, WAS CLOSED ONE MINUTE LATER, REQUIRING THREE ACTUATIONS OF ITS CONTROL SWITCH TO CLOSE. VALVE 1-MS-1D 'VALVE OPEN' LIMIT SWITCH HAD FAILED TO RESET FOLLOWING

VALVE CLOSURE. THE LIMIT SWITCH WAS DISASSEMBLED AND THE SLIDE PLATE AND CONTACT BLOCK WERE REPLACED. VALVE 1-MS-2D AIR (SLIDE) VALVE ASSEMBLY WAS DISASSEMBLED AND A SMALL AMOUNT OF DIRT WAS FOUND INSIDE THE LOWER COVER. THE SLIDE VALVE AND AIR FILTER WERE REPLACED. THERE WERE NO CONSEQUENCES. ON FEBRUARY 2, 1987, LICENSEE EVENT REPORT 86-006-01 REPORTED THE RESULTS OF A "SUBSTANTIAL SAFETY HAZARD EVALUATION". THESE RESULTS SHOWED THAT NO SUBSTANTIAL SAFETY HAZARD EXISTED AND THE FAILURE WAS RANDOM IN NATURE. THE REPORT FURTHER IDENTIFIED THAT DURING THE 1987 REFUELING OUTAGE SWITCH INTERNALS WOULD BE REPLACED ON ALL MAIN STEAM ISOLATION VALVE (MSIV) LIMIT SWITCHES AND STICK-ON TEMPERATURE INDICATORS THAT WERE PLACED ON THE MSIV LIMIT SWITCHES WOULD BE READ.

[186] MILLSTONE 1 DOCKET 50-245 LER 87-037
MISSED SURVEILLANCE FOR MANUAL REACTOR SCRAM FUNCTION.
EVENT DATE: 090887 REPORT DATE: 100787 NSSS: GE TYPE: BWR

(NSIC 206704) ON SEPTEMBER 8, 1987 WHILE OPERATING AT 100% POWER (530 DEGREES F, 1031 PSIG), SURVEILLANCE SP 609.1, MANUAL SCRAM FUNCTION, WAS NOT PERFORMED WITHIN THE REQUIRED FREQUENCY AS CALLED FOR IN THE TECHNICAL SPECIFICATIONS. THE CAUSE OF THE EVENT IS ATTRIBUTED TO OPERATOR ERROR. SINCE THE SURVEILLANCE WAS SUCCESSFULLY COMPLETED APPROXIMATELY ONE WEEK AFTER THE REQUIRED DATE THEREBY DEMONSTRATING THE OPERABILITY OF THE MANUAL SCRAM FUNCTION, THERE WERE NO SAFETY CONSEQUENCES.

[187] MILLSTONE 1 DOCKET 50-245 LER 87-040
FAILURE TO MEET ACCEPTANCE CRITERIA OF CONDENSER LOW VACUUM SCRAM FUNCTIONAL AND CALIBRATION SURVEILLANCE.
EVENT DATE: 091587 REPORT DATE: 101487 NSSS: GE TYPE: BWR
VENDOR: BARKSDALE CONTROLS DIV

(NSIC 206705) ON SEPTEMBER 15, 1987, AT 1300 HOURS, WHILE THE UNIT WAS AT 100% POWER (1030 PSIG, 530 DEGREES FAHRENHEIT), A ROUTINE SURVEILLANCE (SP 408J - CONDENSER LOW VACUUM SCRAM FUNCTIONAL AND CALIBRATION) WAS PERFORMED ON THE CONDENSER LOW VACUUM SCRAM PRESSURE SWITCHES; ALL FOUR PRESSURE SWITCHES (MANUFACTURER: BARKSDALE, MODEL: DIT-H18SS, RANGE: 0.8 TO 29.2 INCHES HG) FAILED TO MEET TECHNICAL SPECIFICATION SETPOINT REQUIREMENTS. THE PRESSURE SWITCHES HAD BEEN REPLACED AND CALIBRATED DURING THE 1987 REFUELING OUTAGE. THE SETPOINT DRIFT WAS ATTRIBUTED TO THE NEW PRESSURE SWITCH'S INITIAL OPERATING PERIOD, DURING WHICH MINOR SETPOINT DRIFT CAN BE EXPECTED. ALL PRESSURE SWITCHES WERE CALIBRATED TO WITHIN REQUIRED SETPOINTS AND SATISFACTORILY TESTED. NO SAFETY CONSEQUENCES RESULTED FROM THIS EVENT.

[188] MILLSTONE 1 DOCKET 50-245 LER 87-039
APR/LP CORE COOLING PUMP INTERLOCK SURVEILLANCE OVERDUE.
EVENT DATE: 092187 REPORT DATE: 102087 NSSS: GE TYPE: BWR

(NSIC 206777) ON SEPTEMBER 21, 1987 AT 0915 HOURS, WITH THE PLANT AT 100% POWER, UNIT 1 OPERATIONS WAS REVIEWING THE SURVEILLANCE DATA SHEET FOR THE AUTOMATIC PRESSURE RELIEF AND LOW PRESSURE CORE COOLING PUMP (ARP/LP) INTERLOCK (SYSTEM CODE BO AND BS) PRESSURE SENSORS WHEN IT WAS DETERMINED THE SURVEILLANCE TEST DUE DATE WAS BEYOND THE REQUIRED TIME INTERVAL, AS DEPICTED IN THE PLANT'S TECHNICAL SPECIFICATIONS. THE SURVEILLANCE TEST WAS SCHEDULED TO BE PERFORMED DURING THE WEEK ENDING SEPTEMBER 18, 1987. SURVEILLANCE TEST WAS NOT PERFORMED UNTIL SEPTEMBER 21, 1987. THE SURVEILLANCE TEST WAS IMMEDIATELY PERFORMED. THE RESULTS OF THE TEST WERE SATISFACTORY. THE CAUSE OF THE DELINQUENCY WAS DUE TO PERSONNEL ERROR ON THE PART OF THE I&C TECHNICIAN ASSIGNED TO PERFORM THE SURVEILLANCE TEST. THE TECHNICIAN THOUGHT THE SURVEILLANCE COULD HAVE BEEN PERFORMED THE FOLLOWING WEEK WITHOUT EXCEEDING THE REQUIRED TIME INTERVAL. TO REDUCE THE POSSIBILITY OF SUCH AN INCIDENT FROM OCCURRING IN THE FUTURE, THE I&C DEPARTMENT

HAS SINCE DEVELOPED A NEW POLICY WHICH REQUIRES ITS PLANNING DEPARTMENT TO PROVIDE AN INDEPENDENT REVIEW OF SURVEILLANCES PERFORMED, AND THEIR REQUIRED COMPLETION DATES, ON A WEEKLY BASIS.

[189] MILLSTONE 1 DOCKET 50-245 LER 87-041
DESIGN CHANGE PROHIBITED BY TECHNICAL SPECIFICATION.
EVENT DATE: 101687 REPORT DATE: 111387 NSSS: GE TYPE: BWR

(NSIC 206904) ON OCTOBER 16, 1987 AT 1630 HOURS, WHILE OPERATING AT 100 PERCENT POWER (528 DEGREES F AND 1032 PSIG), IT WAS DISCOVERED BY NORTHEAST NUCLEAR ENERGY COMPANY (NNECO) THAT A DESIGN CHANGE IMPLEMENTED DURING THE RECENTLY COMPLETED REFUELING OUTAGE, REQUIRING CHANGES TO TECHNICAL SPECIFICATIONS, HAD NOT RECEIVED PRIOR APPROVAL FROM THE NUCLEAR REGULATORY COMMISSION (NRC) AS REQUIRED PER 10CFR50.59(A)(1). THIS DISCREPANCY WAS IMMEDIATELY BROUGHT TO THE ATTENTION OF THE NRC STAFF AND A LICENSE AMENDMENT REQUEST WAS SUBMITTED ON OCTOBER 20, 1987.

[190] MILLSTONE 2 DOCKET 50-336 LER 87-009
REACTOR TRIP ON LOW #1 STEAM GENERATOR LEVEL.
EVENT DATE: 090287 REPORT DATE: 093087 NSSS: CE TYPE: PWR
VENDOR: COPES-VULCAN, INC.

(NSIC 206590) WHILE OPERATING AT 91% POWER ON SEPTEMBER 2, 1987 AT 2015 THE UNIT EXPERIENCED AN AUTOMATIC REACTOR TRIP DUE TO LOW STEAM GENERATOR LEVEL IN THE #1 STEAM GENERATOR. THE OPERATION'S STAFF WAS PERFORMING A ROUTINE REDUCTION IN POWER LEVEL FROM 100% TO 90%. DURING THIS EVOLUTION THE SECONDARY PLANT OPERATOR OBSERVED THE FEED FLOW TO THE #1 STEAM GENERATOR TO BE LESS THAN THE STEAM FLOW. THE OPERATOR TOOK MANUAL CONTROL OF THE FEEDWATER REGULATING VALVE, 2- FW-51A, AND THE FEEDWATER REGULATING BYPASS VALVE, 2-FW-41A, IN AN ATTEMPT TO RESTORE LEVEL IN THE #1 STEAM GENERATOR. LEVEL CONTINUED TO DECREASE IN THE #1 STEAM GENERATOR AND THE UNIT TRIPPED. OPERATIONS RESPONDED TO THE TRIP BY PERFORMING EOP 2525, "STANDARD POST TRIP ACTIONS" AND EOP 2526 "REACTOR TRIP RECOVERY". NO OTHER SYSTEMS WERE AFFECTED AND THE UNIT WAS PLACED IN A STABLE CONDITION. THE #1 FEEDWATER REGULATING VALVE, 2-FW-51A, WAS DISASSEMBLED AND REPAIRED. DURING THE DISASSEMBLY IT WAS DISCOVERED THAT THE STEM HAD SEPARATED FROM THE PLUG. THIS EVENT IS BEING REPORTED PURSUANT TO THE REQUIREMENTS OF PARAGRAPH 50.73(A)(2)(IV) DUE TO THE AUTOMATIC REACTOR TRIP ON LOW STEAM GENERATOR LEVEL. SIMILAR LER'S: 87-02.

[191] MILLSTONE 3 DOCKET 50-423 LER 86-058 REV 02
UPDATE ON INADEQUATE RADIATION MONITOR SURVEILLANCES DUE TO INADEQUATE TECHNICAL SPECIFICATION REVIEW.
EVENT DATE: 121786 REPORT DATE: 093087 NSSS: WE TYPE: PWR

(NSIC 206587) AT 1300 ON 12/17/86 WHILE OPERATING AT 100% POWER IN MODE 1, IT WAS NOTED THAT SURVEILLANCE PROCEDURES FOR EFFLUENT RADIATION MONITORS WERE INADEQUATE TO MEET REQUIREMENTS OF PLANT TECHNICAL SPECIFICATIONS. EFFLUENT RADIATION MONITORS WERE DECLARED INOPERABLE, THE LIMITED CONDITION FOR OPERATION AND CORRESPONDING ACTION STATEMENTS WERE ENTERED. PERIODIC GRAB SAMPLES WERE OBTAINED. SURVEILLANCE PROCEDURES WERE REVISED AND PROPER OPERATION OF EFFLUENT MONITORS WAS VERIFIED. REVIEW OF LICENSEE EVENT REPORTS SHARING INADEQUATE SURVEILLANCES AS ROOT CAUSE INDICATED A POSSIBLE TREND. REVIEW OF TWENTY SURVEILLANCE PROCEDURES, AS OUTLINED IN LER 86-058-01, REVEALED NO PROGRAMMATIC PROBLEMS, HOWEVER TWO AREAS WILL BE INVESTIGATED FURTHER. THIS INVESTIGATION WILL BE COMPLETED BY 12/31/87.

[192] MILLSTONE 3 DOCKET 50-423 LER 87-034
REACTOR TRIP DUE TO LOW STEAM GENERATOR LEVEL CAUSED BY FAILED SOLENOID VALVE.
EVENT DATE: 092387 REPORT DATE: 101687 NSSS: WE TYPE: PWR
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 206681) ON SEPTEMBER 23, 1987, AT 1351 HOURS, WHILE OPERATING AT 100% POWER IN MODE 1, A REACTOR TRIP OCCURRED DUE TO LOW LOW STEAM GENERATOR LEVEL IN THE A STEAM GENERATOR. THE OPERATORS VERIFIED THAT ALL CONTROL RODS WERE FULLY INSERTED, AND THAT ALL REACTOR TRIP BREAKERS HAD OPENED. IMMEDIATELY PRIOR TO THE TRIP, THE OPERATORS OBSERVED THAT THERE WAS A FEED FLOW/STEAM FLOW MISMATCH IN ALL THE STEAM GENERATORS, AND THAT THE MAIN FEEDWATER ISOLATION VALVE TO THE A STEAM GENERATOR, 3FWS*CTV41A, HAD CLOSED. THE PLANT TRIPPED BEFORE ANY CORRECTIVE MEASURES COULD BE TAKEN. THE ROOT CAUSE OF THE EVENT WAS EQUIPMENT FAILURE. AN INSPECTION OF THE ISOLATION VALVE CIRCUIT INDICATED THAT THE FAILURE WAS DUE TO THE FAILURE OF ONE OF THE SOLENOID VALVES WHICH CONTROLS HYDRAULIC OIL FLOW TO THE ISOLATION VALVE. THE FAILURE OF THE SOLENOID VALVE WAS INTERNAL TO THE SOLENOID COIL. THE SOLENOID WAS REPLACED. THERE HAVE BEEN TWO SIMILAR EVENTS OF REACTOR TRIPS DUE TO THE INADVERTENT CLOSURE OF A MAIN FEEDWATER ISOLATION VALVE.

[193] MILLSTONE 3 DOCKET 50-423 LER 87-035
SURVEILLANCE TEST METHOD NOT IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS.
EVENT DATE: 101687 REPORT DATE: 111687 NSSS: WE TYPE: PWR

(NSIC 207115) ON 10/16/87 WHILE OPERATING AT 98% POWER (MODE 1) OPERATIONS DEPARTMENT PERSONNEL PERFORMING A BIANNUAL PROCEDURE REVIEW DISCOVERED A DISCREPANCY BETWEEN THE PLANT TECHNICAL SPECIFICATION SURVEILLANCE METHOD SPECIFIED AND THE ACTUAL METHOD UTILIZED. IT WAS FOUND THAT THE SURVEILLANCE TEST METHOD WAS NOT IN COMPLIANCE WITH THE TEST METHOD SPECIFIED IN TECHNICAL SPECIFICATIONS. IMMEDIATE ACTION WAS TO ENTER A LIMITED CONDITION FOR OPERATION AND PERFORM AN OVERALL AIR LOCK LEAKAGE SURVEILLANCE TEST, VERIFYING THE OVERALL AIR LEAKAGE WITHIN ITS LIMITS. THE CAUSE OF THE ERROR WAS ADMINISTRATIVE. THE SURVEILLANCE PROCEDURE HAD NOT BEEN COMPLETELY UPDATED TO COMPLY WITH THE FINAL ISSUE OF TECHNICAL SPECIFICATIONS. AS ACTION TO PREVENT RECURRENCE THE TECHNICAL SPECIFICATION SURVEILLANCE HAS BEEN UPDATED TO COMPLY WITH THE EXISTING, SPECIFIED METHOD.

[194] NINE MILE POINT 1 DOCKET 50-220 LER 87-014
ELECTRICAL PRESSURE REGULATOR SERVO-VALVE MALFUNCTION RESULTS IN REACTOR SCRAM, HIGH PRESSURE COOLANT INJECTION MODE OF FEEDWATER AND MAIN STEAM ISOLATION VALVE CLOSURE.
EVENT DATE: 101687 REPORT DATE: 111687 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.
MOOG INC.

(NSIC 206902) DURING NORMAL OPERATION ON OCTOBER 16, 1987, A MALFUNCTION OF THE TURBINE CONTROL SYSTEM CAUSED STEAM PRESSURE OSCILLATIONS, RESULTING IN A REACTOR SCRAM FROM A POWER EXCURSION TO THE AVERAGE POWER RANGE MONITOR HIGH NEUTRON FLUX SCRAM SETPOINT. THE HIGH PRESSURE COOLANT INJECTION MODE OF FEEDWATER INITIATED DUE TO THE MOMENTARY LOW REACTOR WATER LEVEL, WHICH WAS THE RESULT OF THE REACTOR SCRAM. THE TURBINE CONTROL SYSTEM MALFUNCTION ALSO CAUSED REACTOR PRESSURE TO DECREASE SUCH THAT CLOSURE OF THE MAIN STEAM ISOLATION VALVES OCCURRED AS A PROTECTIVE ACTION. THE TURBINE CONTROL SYSTEM FAILURE HAS BEEN ATTRIBUTED TO A STUCK SERVO-VALVE IN THE ELECTRICAL PRESSURE REGULATOR HYDRAULIC ACTUATOR. AN IMMEDIATE CORRECTIVE ACTION INCLUDED REPLACEMENT OF THE SERVO-VALVE. INCREASED PREVENTIVE MAINTENANCE OF THE ELECTRICAL PRESSURE REGULATOR WILL ALSO BE SCHEDULED. ALTERNATIVE DESIGNS WILL BE EVALUATED. A VIOLATION OF TECHNICAL SPECIFICATIONS RESULTED DUE TO PERSONNEL ERROR REGARDING THE TARDINESS OF THE 10

CFR 50.72 FOUR HOUR NOTIFICATION FOR THE MAIN STEAM ISOLATION VALVE CLOSURE. ADDITIONAL TRAINING REGARDING THE REPORTAB. .ITY OF OCCURRENCES WILL BE DEVELOPED.

[195] NINE MILE POINT 1 DOCKET 50-220 LER 87-015
REACTOR SCRAM, TURBINE TRIP, AND HIGH PRESSURE COOLANT INJECTION MODE OF FEEDWATER SIGNALS DUE TO SPURIOUS TRIP OF NEUTRON MONITOR CAUSED BY NOISE.
EVENT DATE: 101787 REPORT DATE: 111687 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 206903) ON OCTOBER 17, 1987 NINE MILE POINT UNIT 1 WAS IN A COLD SHUTDOWN CONDITION. THE MODE SWITCH WAS IN THE "REFUEL" POSITION AND ALL CONTROL RODS WERE FULLY INSERTED. AT 1525 HOURS, THE UNIT EXPERIENCED A FULL REACTOR SCRAM, A TURBINE TRIP SIGNAL ACTUATION, AND ACTUATION OF HIGH PRESSURE COOLANT INJECTION (HPCI) MODE OF FEEDWATER INITIATION LOGIC. THE SCRAM WAS A RESULT OF AN INTERMEDIATE RANGE NEUTRON MONITOR (IRM) SPIKE ON ONE REACTOR PROTECTION SYSTEM (RPS) CHANNEL WHILE A ONE-HALF SCRAM ALREADY EXISTED ON THE OTHER CHANNEL DUE TO MAINTENANCE. THE HPCI MODE OF FEEDWATER INITIATION LOGIC ACTUATION WAS AN EXPECTED RESULT DUE TO THE SCRAM AND SUBSEQUENT TURBINE TRIP SIGNAL. THE ROOT CAUSE OF THE SCRAM WAS SPURIOUS ACTUATION OF AN IRM ON RPS TRIP CHANNEL 11 DUE TO NOISE. NO ADVERSE SAFETY CONSEQUENCES RESULTED FROM THIS EVENT. THERE WAS NO FEEDWATER INJECTION SINCE THE HPCI/FEEDWATER PUMPS WERE LOCKED OUT. REACTOR WATER LEVEL REMAINED CONSTANT AND STABLE THROUGHOUT THE EVENT. CORRECTIVE ACTIONS INCLUDED RESETTING THE SCRAM, TURBINE TRIP, AND HPCI MODE OF FEEDWATER LOGIC, AND PERFORMING AN EVALUATION OF THE SCRAM IN ACCORDANCE WITH EXISTING PROCEDURES. ALSO, SINCE NOISE HAS CAUSED PREVIOUS EVENTS AT NMP1, A PROBLEM REPORT WAS GENERATED TO INITIATE A MORE THOROUGH INVESTIGATION.

[196] NINE MILE POINT 2 DOCKET 50-410 LER 87-031 REV 01
UPDATE ON SCRAM DUE TO COLD WATER EXCURSION WHEN FEEDWATER LEVEL CONTROL VALVE FAILED OPEN.
EVENT DATE: 061287 REPORT DATE: 101387 NSSS: GE TYPE: BWR
VENDOR: VALTEK INC.

(NSIC 206629) ON JUNE 12, 1987 WITH NINE MILE POINT UNIT 2 IN "STARTUP" (MODE 2) AND AT LESS THAN THREE PERCENT POWER, OPERATIONS PERSONNEL OBSERVED REACTOR WATER LEVEL RISING AND NEUTRON FLUX LEVELS INCREASING RAPIDLY. FEEDWATER LEVEL CONTROL VALVE 2FWS-LV55A WAS OBSERVED FULL OPEN. OPERATIONS PERSONNEL PLACED THE CONTROLLER IN MANUAL AND ATTEMPTED TO CLOSE THE VALVE. 2FWS-LV55A FAILED TO RESPOND. THE RESULTING COLD WATER INJECTION CAUSED INTERMEDIATE RANGE MONITOR (IRM) CHANNELS B, C, AND D TO EXCEED THEIR HIGH FLUX SCRAM POINTS DUE TO POSITIVE REACTIVITY ADDITION. THE SUBSEQUENT SCRAM OCCURRED ON JUNE 12, 1987 AT 20:56 WITH THE REACTOR AT 842 POUNDS PER SQUARE INCH GAUGE (PSIG) AND 523F. THE CAUSE OF THE EVENT HAS BEEN ATTRIBUTED TO THE FEEDBACK ARM FROM THE VALVE STEM TO THE VALVE POSITIONER UNBOLTING ALLOWING THE VALVE TO RAMP OPEN. FWS-LV55A HAS BEEN REPAIRED AND THE SUBJECT BOLT LOCKWIRED.

[197] NINE MILE POINT 2 DOCKET 50-410 LER 87-052
MISSED SURVEILLANCE FOR STANDBY GAS TREATMENT SYSTEM TRAIN A WITH DIVISION II DIESEL GENERATOR INOPERABLE RESULTS IN INITIATION OF REQUIRED PLANT SHUTDOWN.
EVENT DATE: 090287 REPORT DATE: 100287 NSSS: GE TYPE: BWR

(NSIC 206694) ON SEPTEMBER 2, 1987 TWO RELATED TECHNICAL SPECIFICATION (TS) VIOLATIONS OCCURRED AT NINE MILE POINT UNIT 2. THE FIRST WAS A RESULT OF EXCEEDING A SURVEILLANCE REQUIREMENT OF A STANDBY GAS TREATMENT SYSTEM (GTS) CHARCOAL ADSORBER (TS 4.6.5.3.C). THE SECOND WAS A RESULT OF EXCEEDING A LIMITING CONDITION OF OPERATION (LCO) FOR THE AC POWER SOURCES (TS 3.8.1.1 ACTION E). AT THE TIME OF THE EVENT DISCOVERY THE REACTOR WAS AT APPROXIMATELY 40% RATED THERMAL POWER. AFTER THE EVENT HAD BEEN ANALYZED A THIRD TS VIOLATION (TS

6.8.1.D) WAS ALSO DISCOVERED. THE CAUSE OF THE EVENT WAS DETERMINED TO BE INADEQUATE PROCEDURAL CONTROL OF THE CUMULATIVE OPERATING HOURS STATUS OF GTS CHARCOAL ADSORBERS. CONTRIBUTING TO THE EVENT WAS THE UNTIMELY COMPLETION OF A REVIEW TO DETERMINE THE CUMULATIVE GTS OPERATING HOURS AND THE FAILURE TO NOTIFY THE STATION SHIFT SUPERVISOR THAT THE STATUS OF A TS REQUIRED SURVEILLANCE WAS IN QUESTION. CORRECTIVE ACTIONS INCLUDE THE FOLLOWING: 1. AN INADEQUATE PROCEDURE HAS BEEN REVISED AND A NEW PROCEDURE HAS BEEN GENERATED TO TRACK GTS TIME ON A DAILY BASIS. 2. A MODIFICATION TO ADD ELAPSED TIME METERS TO ALL SPECIAL FILTER TRAINS HAS BEEN COMPLETED. 3. ADMINISTRATIVE CONTROLS SHALL BE REVIEWED AND CHANGED TO PRECLUDE RECURRENCE OF SIMILAR EVENT SITUATIONS.

[198] NINE MILE POINT 2 DOCKET 50-410 LER 87-053
REACTOR WATER CLEANUP SYSTEM ISOLATION ON A FLOW DIFFERENTIAL SIGNAL DUE TO CONSTRUCTION AND DESIGN DEFICIENCIES.
EVENT DATE: 090387 REPORT DATE: 100287 NSSS: GE TYPE: BWR

(NSIC 206695) ON SEPTEMBER 3, 1987 AT 0849 HOURS, NINE MILE POINT UNIT 2 (NMP2) EXPERIENCED THE ACTUATION OF AN ENGINEERED SAFETY FEATURE (ESF), SPECIFICALLY, ISOLATION OF THE REACTOR WATER CLEANUP (RWCU) SYSTEM. AT THE TIME OF THE EVENT, THE PLANT WAS IN THE STARTUP MODE WITH REACTOR POWER LESS THAN 1% AND THE REACTOR MODE SWITCH IN THE "STARTUP/HOT STANDBY" POSITION. REACTOR PRESSURE AND TEMPERATURE WERE APPROXIMATELY 700 POUNDS PER SQUARE INCH GAUGE (PSIG) AND 505F, RESPECTIVELY. THE ROOT CAUSE FOR THE EVENT IS CONSTRUCTION AND DESIGN DEFICIENCIES. THE CORRECTIVE ACTIONS FOR THE EVENT ARE: 1. THE FLEX HOSES INSIDE PRIMARY CONTAINMENT HAVE BEEN REPLACED WITH RIGID TUBING AND THE BLOCKING GLOBE VALVES RE-ORIENTED. 2. A SPECIAL TASK FORCE HAS BEEN ASSIGNED TO EVALUATE AND TROUBLESHOOT RWCU FLOW TRANSMITTER PROBLEMS. 3. THE REJECT FLOW TRANSMITTER ORIFICE ELEMENT HAS BEEN RELOCATED.

[199] NINE MILE POINT 2 DOCKET 50-410 LER 87-054
MAIN STEAM VALVE ISOLATION SIGNAL DUE TO TURBINE STOP VALVE SURVEILLANCE TESTING.
EVENT DATE: 090987 REPORT DATE: 100987 NSSS: GE TYPE: BWR

(NSIC 206696) ON SEPTEMBER 9, 1987 AT 0106 HOURS, NINE MILE POINT UNIT 2 EXPERIENCED ACTUATION OF AN ENGINEERED SAFETY FEATURE (ESF), SPECIFICALLY, A MAIN STEAM ISOLATION VALVE (MSIV) ISOLATION SIGNAL. AT THE TIME OF THE EVENT, THE PLANT WAS IN COLD SHUTDOWN (OPERATIONAL CONDITION 4) WITH THE REACTOR MODE SWITCH IN "SHUTDOWN", AT AMBIENT PRESSURE, A TEMPERATURE OF 165F, AND THE MSIVS CLOSED. NO VALVES CHANGED POSITION DURING THIS ACTUATION OF THE MSIV LOGIC. THE ROOT CAUSE OF THE EVENT WAS A PROCEDURAL DEFICIENCY. CORRECTIVE ACTIONS FOR THIS EVENT ARE: 1. THE OPERATORS IMMEDIATELY INVESTIGATED THE EVENT, DETERMINED ITS CAUSE, AND RESET THE ISOLATION SEAL-IN SIGNAL. 2. CHANGES HAVE BEEN INCORPORATED INTO THE SURVEILLANCE PROCEDURE N2-ISP-RPS- R202 TO INCLUDE PLANT IMPACT ON NS4 (NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM)/MSIV ISOLATION SIGNALS. 3. THE INSTRUMENT AND CONTROL DEPARTMENT IS PRESENTLY REVIEWING THEIR SURVEILLANCE PROCEDURES TO DETERMINE IF ADDITIONAL COMMENTS NEED TO BE INCLUDED REGARDING THE NS4 ISOLATION LOGIC. 4. A PROBLEM REPORT HAS BEEN SUBMITTED TO ENGINEERING TO ADDRESS THE LACK OF ANNUNCIATION IN THE CONTROL ROOM REGARDING THE NS ISOLATION SIGNALS.

[200] NINE MILE POINT 2 DOCKET 50-410 LER 87-055
PARTIAL PRIMARY CONTAINMENT ISOLATION DUE TO INSTRUMENT LEADS BEING DISCONNECTED DURING A SURVEILLANCE/PERSONNEL ERROR.
EVENT DATE: 091687 REPORT DATE: 101487 NSSS: GE TYPE: BWR

(NSIC 206697) ON SEPTEMBER 16, 1987 AT 2218 WITH THE REACTOR IN COLD SHUTDOWN (OPERATIONAL CONDITION 4), NINE MILE POINT UNIT 2 EXPERIENCED AN ISOLATION OF THE DIVISION 2 RESIDUAL HEAT REMOVAL SYSTEM (RHR) SHUTDOWN COOLING (SDC) ISOLATION VALVES AND AN ISOLATION SIGNAL TO THE DIVISION 2 REACTOR CORE ISOLATION COOLING

SYSTEM (RCIC) ISOLATION VALVES. NO VALVE MOVEMENT OF THE RCIC VALVES OCCURRED SINCE THESE VALVES WERE ALREADY CLOSED. THIS ISOLATION, INITIATED BY A REACTOR BUILDING GENERAL AREA HIGH TEMPERATURE SIGNAL, OCCURRED WHEN LEADS WERE DISCONNECTED FROM A TEMPERATURE SENSING DEVICE WHILE PERFORMING A SURVEILLANCE TEST. THE LEADS WERE IMMEDIATELY RECONNECTED, THE ISOLATION WAS RESET, AND THE DIVISION 2 SDC MODE OF RHR WAS RESTORED TO NORMAL BY 2225 THAT SAME DAY. THE IMMEDIATE CAUSE FOR THIS EVENT IS PERSONNEL ERROR DUE TO INATTENTION TO DETAIL. THE CORRECTIVE ACTIONS FOR THIS EVENT ARE: 1. THE TECHNICIAN INVOLVED HAS BEEN COUNSELED. 2. THE EVENT WILL BE DISCUSSED IN THE I&C DEPARTMENT SAFETY MEETINGS. 3. TRAINING MODIFICATION RECOMMENDATIONS HAVE BEEN SUBMITTED REQUESTING TECHNICIAN AND OPERATOR TRAINING ON THIS EVENT. 4. SURVEILLANCE PROCEDURE N2-ISP-LDS-M006 WILL BE REVISED. 5. A REQUEST FOR AN ENGINEERING ANALYSIS HAS BEEN INITIATED, TO FIND ALTERNATIVES TO HAVING TO DISCONNECT INSTRUMENT LEADS WHEN PERFORMING SURVEILLANCES ON THESE DEVICES.

[201] NINE MILE POINT 2 DOCKET 50-410 LER 87-057
SHUTDOWN COOLING SYSTEM ISOLATION DUE TO PERSONNEL ERROR CAUSED BY FAILURE TO FOLLOW PROCEDURE.
EVENT DATE: 092187 REPORT DATE: 102087 NSSS: GE TYPE: BWR

(NSIC 206828) ON SEPTEMBER 21, 1987 AT 1103 HOURS, NINE MILE POINT UNIT 2 (NMP2) EXPERIENCED AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION, SPECIFICALLY, ISOLATION OF THE RESIDUAL HEAT REMOVAL (RHS) SHUTDOWN COOLING (SDC) SYSTEM. AT THE TIME OF THE EVENT, THE PLANT WAS IN THE COLD SHUTDOWN CONDITION WITH THE REACTOR MODE SWITCH IN THE "SHUTDOWN" POSITION. REACTOR PRESSURE WAS ATMOSPHERIC WITH A REACTOR COOLANT TEMPERATURE OF APPROXIMATELY 116 F. THE ROOT CAUSE OF THIS EVENT WAS COGNITIVE PERSONNEL ERROR; FAILURE TO FOLLOW PROCEDURE. THE FAILURE TO FOLLOW PROCEDURE WAS CAUSED BY AN INATTENTION TO DETAIL. INITIAL CORRECTIVE ACTIONS WERE FOR THE OPERATORS TO IDENTIFY THE CAUSE OF THE SDC ISOLATION, VERIFY THE PLANT STATUS AS NORMAL, AND RESTORE THE SDC SYSTEM TO SERVICE. ADDITIONAL CORRECTIVE ACTIONS FOR THIS EVENT ARE: 1. THE TECHNICIAN INVOLVED HAS BEEN COUNSELED. 2. A TRAINING MODIFICATION RECOMMENDATION HAS BEEN SUBMITTED REQUESTING INSTRUMENT AND CONTROL (I&C) TECHNICIAN TRAINING ON THIS EVENT. 3. THE EVENT WILL BE DISCUSSED IN THE I&C DEPARTMENT SAFETY MEETINGS.

[202] NINE MILE POINT 2 DOCKET 50-410 LER 87-056
BOTH GASEOUS EFFLUENT MONITORING SYSTEMS DECLARED INOPERABLE DUE TO A DESIGN DEFICIENCY.
EVENT DATE: 092587 REPORT DATE: 102387 NSSS: GE TYPE: BWR

(NSIC 206827) ON SEPTEMBER 25, 1987 AT 1315 WITH THE REACTOR IN COLD SHUTDOWN, (OPERATIONAL CONDITION 4) THE MAIN STACK AND THE RADWASTE/REACTOR BUILDING GASEOUS EFFLUENT MONITORING SYSTEMS (GEMS) WERE DECLARED INOPERABLE AT NINE MILE POINT UNIT 2 (NMP2) AS REQUIRED BY NMP2 TECHNICAL SPECIFICATION (TS) SECTION 3.3.7.10. THE GEMS SYSTEMS WERE CONSIDERED TO BE INOPERABLE WHEN IT WAS DETERMINED BY THE CONTROL ROOM SUPERVISOR (UPON REVIEWING A PROBLEM REPORT CONCERNING THE GEMS ANNUNCIATORS) THAT THE GEMS CONTROL ROOM ANNUNCIATION DID NOT MEET THE OPERABILITY REQUIREMENTS OF TS TABLE 4.3.7.10-1 NOTE C. IT IS ANTICIPATED THAT THE GEMS SYSTEM WILL BE RESTORED TO OPERATION BY DECEMBER 7, 1987. THE ROOT CAUSE FOR THIS EVENT IS A DESIGN DEFICIENCY. THE CORRECTIVE ACTIONS FOR THIS EVENT ARE: 1. AN ALTERNATIVE MEANS FOR MONITORING GASEOUS EFFLUENT WAS IMMEDIATELY IMPLEMENTED. 2. NIAGARA MOHAWK ENGINEERING HAS EVALUATED THE OPERABILITY REQUIREMENTS FOR THE GEMS SYSTEM. 3. THE GEMS CONTROL ROOM ANNUNCIATORS WILL BE MODIFIED. 4. THE GEMS SURVEILLANCE PROCEDURE WILL BE REVISED.

[203] NINE MILE POINT 2 DOCKET 50-410 LER 87-059
LACK OF NINE CHANNEL CHECKS IN SURVEILLANCE PROCEDURE RESULTS IN FOUR MISSED TECH SPECS SURVEILLANCE REQUIREMENTS.
EVENT DATE: 093087 REPORT DATE: 102887 NSSS: GE TYPE: BWR

(NSIC 206888) ON SEPTEMBER 30, 1987 FOUR TECHNICAL SPECIFICATION (TS) SURVEILLANCE REQUIREMENTS WERE FOUND TO HAVE BEEN EXCEEDED AT NINE MILE POINT UNIT 2. THE SURVEILLANCE REQUIREMENTS WERE EXCEEDED AS A RESULT OF THE FAILURE TO PERFORM NINE TS REQUIRED 12 HOUR INSTRUMENT CHANNEL CHECKS. AT THE TIME OF THE DISCOVERY THE PLANT WAS IN THE STARTUP CONDITION WITH THE REACTOR AT APPROXIMATELY 100F AND AMBIENT PRESSURE. THE CAUSE OF THE TS VIOLATIONS WAS A PROCEDURAL DEFICIENCY. OPERATIONS SURVEILLANCE PROCEDURE, N2-OSP-LOG-S001, "SHIFTLY CHECKS" DID NOT INCLUDE NINE TS REQUIRED CHANNEL CHECKS AS REQUIRED. THE FIRST TWO TS VIOLATIONS WERE DISCOVERED WHEN A NIAGARA MOHAWK OPERATOR QUESTIONED THE ABSENCE OF SIX TS REQUIRED 12 HOUR INSTRUMENT CHANNEL CHECKS FROM N2- OSP-LOG-S001. THE OTHER TWO TS VIOLATIONS WERE DISCOVERED AS A RESULT OF A PROCEDURAL REVIEW WHICH WAS INITIATED TO ASSURE THAT ALL OTHER TS REQUIRED CHANNEL CHECKS WERE INCLUDED IN APPLICABLE SURVEILLANCE PROCEDURES. IMMEDIATE CORRECTIVE ACTION WAS TO PERFORM THE MISSED TS SURVEILLANCES AND TO ADD THEM TO N2-OSP-LOG-S001. A REVIEW OF ALL TS ITEMS WHICH REQUIRED CHANNEL CHECKS AT REGULAR FREQUENCIES WAS COMPLETED BY THE END OF THE DAY ON SEPTEMBER 30, 1987. A TASK FORCE HAS BEEN ESTABLISHED TO ASSURE THAT EACH SURVEILLANCE ACTIVITY IS ADDRESSED BY PROCEDURE AND EACH PROCEDURE ADEQUATELY PERFORMS THE REQUIRED SURVEILLANCE.

[204] NINE MILE POINT 2 DOCKET 50-410 LER 87-058
REACTOR SCRAM ON HIGH NEUTRON FLUX DUE TO PERSONNEL ERROR.
EVENT DATE: 100187 REPORT DATE: 102387 NSSS: GE TYPE: BWR
VENDOR: BAILEY INSTRUMENT CO., INC.

(NSIC 206829) ON 10/1/87 AT 1213 HOURS, NINE MILE POINT UNIT 2 EXPERIENCED ACTUATION OF AN ENGINEERED SAFETY FEATURE (ESF), SPECIFICALLY, A HIGH NEUTRON FLUX SCRAM ON THE INTERMEDIAT E RANGE MONITORS (IRM). AT THE TIME OF THE EVENT, THE PLANT WAS IN THE STARTUP MODE WITH REACTOR POWER AT APPROXIMATELY 2.5% AND THE REACTOR MODE SWITCH IN THE "STARTUP/HOT STANDBY" POSITION. REACTOR PRESSURE AND TEMPERATURE WERE APPROXIMATELY 505 PSIG AND 477F, RESPECTIVELY. THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR. CORRECTIVE ACTIONS FOR THIS EVENT ARE: 1. THE INDIVIDUALS INVOLVED WITH THIS EVENT WERE IMMEDIATELY INSTRUCTED ON THE SYSTEM CHARACTERISTICS AND PARTICIPATED IN THE TRIP INVESTIGATION AND FOLLOW UP REPORT. THE EVENT WAS REVIEWED BY THE OPERATORS DURING SHIFT TURNOVERS. 2. A TRAINING MODIFICATION RECOMMENDATION HAS BEEN INITIATED TO ADDRESS THIS ISSUE. 3. A WORK REQUEST (WR# 125604) HAS BEEN WRITTEN TO INVESTIGATE/REPAIR THE CONDENSATE BOOSTER PUMP MINIMUM FLOW VALVE AND POSITION TRANSMITTER. 4. A TEMPORARY CHANGE NOTICE HAS REVISED THE OPERATING PROCEDURE N2-OP-101A GIVING INSTRUCTIONS TO MAINTAIN STEAM LOADS WITHIN THE CAPABILITY OF THE CONDENSATE LEVEL CONTROL VALVE LV-137. 5. A PROBLEM REPORT (PR# 7387) HAS BEEN SUBMITTED TO ENGINEERING TO EVALUATE LEVEL CONTROL VALVE LV-137. ENGINEERING WILL MAKE A DETERMINATION AS TO WHETHER THE VALVE IS UNDERSIZED. 6. A PROBLEM REPORT HAS BEEN INITIATED.

[205] NINE MILE POINT 2 DOCKET 50-410 LER 87-061
MISSED SURVEILLANCE OF CONTAINMENT ATMOSPHERE DUE TO PROCEDURAL DEFICIENCY.
EVENT DATE: 100187 REPORT DATE: 102887 NSSS: GE TYPE: BWR

(NSIC 206881) ON 10/1/87 IT WAS DETERMINED THAT NINE MILE POINT UNIT 2 HAD VIOLATED A TECHNICAL SPECIFICATION (TS) FOR REACTOR COOLANT SYSTEM (RCS) LEAKAGE. WITH THE PRIMARY CONTAINMENT AIRBORNE PARTICULATE MONITORING SYSTEM INOPERABLE, CONTAINMENT ATMOSPHERE SAMPLES SHOULD HAVE BEEN OBTAINED AND ANALYZED EVERY 12 HRS. BY THE APPROVED PROCEDURE, SAMPLES WERE OBTAINED EVERY 24 HRS. AT THE TIME OF THE DISCOVERY, THE REACTOR WAS AT APPROXIMATELY 1% OF RATED THERMAL POWER WITH

THE REACTOR MODE SWITCH IN THE "STARTUP/HOT STANDBY" POSITION. REACTOR PRESSURE AND COOLANT TEMPERATURE WERE APPROXIMATELY 40 POUNDS PER SQUARE INCH GAUGE AND 286F, RESPECTIVELY. THE ROOT CAUSE OF THESE EVENTS WAS A PROCEDURAL DEFICIENCY. A SIGNIFICANT CONTRIBUTING FACTOR IS THE INCONSISTENCY IN THE TWO TS REGARDING RCS LEAKAGE. CORRECTIVE ACTION WAS TO REVISE THE APPLICABLE PROCEDURE. PRIMARY CONTAINMENT ATMOSPHERE SAMPLES ARE NOW REQUIRED TO BE OBTAINED AND ANALYZED EVERY 12 HOURS WHEN EITHER THE AIRBORNE GASEOUS OR PARTICULATE MONITORING SYSTEM IS INOPERABLE. ADDITIONAL CORRECTIVE ACTION HAS BEEN INITIATED TO ESTABLISH A SET OF TS INTERPRETATIONS, WHICH WILL BE USED TO VERIFY CURRENT PLANT PROCEDURES COMPLY WITH TS REQUIREMENTS. ALSO, A REVIEW OF THE TWO APPLICABLE TS WILL BE CONDUCTED TO DETERMINE IF AN APPLICATION FOR AMENDMENT TO THE TS SHOULD BE SUBMITTED TO CLARIFY THE NECESSARY REQUIREMENTS FOR RCS LEAKAGE.

[206] NINE MILE POINT 2 DOCKET 50-410 LER 87-065
QUARTER INCH VACUUM IN SECONDARY CONTAINMENT NOT MAINTAINED DUE TO DESIGN DEFICIENCY.
EVENT DATE: 101587 REPORT DATE: 111087 NSSS: GE TYPE: BWR

(NSIC 207148) ON OCTOBER 15, 1987 FOLLOWING A NIAGARA MOHAWK ENGINEERING DEPARTMENT EVALUATION, IT WAS DETERMINED THAT THE SECONDARY CONTAINMENT (REACTOR BUILDING) FOR NINE MILE POINT UNIT 2 (NMP2) HAD NOT BEEN MAINTAINED AT A UNIFORM SUBATMOSPHERIC PRESSURE OF 0.25 INCH OF VACUUM WATER GAUGE. AT THE TIME OF THIS DETERMINATION, THE PLANT WAS IN A COLD SHUTDOWN CONDITION WITH THE REACTOR MODE SWITCH IN THE "SHUTDOWN" POSITION. REACTOR PRESSURE AND COOLANT TEMPERATURE WERE APPROXIMATELY ATMOSPHERIC AND 132F, RESPECTIVELY. THE ROOT CAUSE OF THIS EVENT WAS A DESIGN DEFICIENCY. THE SYSTEM TO MEASURE DIFFERENTIAL PRESSURE DID NOT ACCOUNT FOR THE DIFFERENCE IN THE PRESSURE GRADIENTS INSIDE AND OUTSIDE THE REACTOR BUILDING. THESE DIFFERING PRESSURE GRADIENTS RESULTED FROM THE DIFFERENCE IN TEMPERATURE BETWEEN INSIDE AND OUTSIDE THE REACTOR BUILDING. CORRECTIVE ACTIONS HAVE BEEN TO TEMPORARILY REVISE THE DIFFERENTIAL PRESSURE TRANSMITTERS' SETPOINTS TO ACCOUNT FOR THE DIFFERENCES IN THE PRESSURE GRADIENTS. A MODIFICATION HAS BEEN INITIATED TO RELOCATE THE DIFFERENTIAL PRESSURE ELEMENTS TO THE ROOF OF THE REACTOR BUILDING, THUS REMOVING THE EFFECT OF THE DIFFERING PRESSURE GRADIENTS ON THE SENSED DIFFERENTIAL PRESSURE.

[207] NORTH ANNA 1 DOCKET 50-338 LER 85-003 REV 01
UPDATE ON FLOODING POTENTIAL NOT PREVIOUSLY EVALUATED.
EVENT DATE: 031985 REPORT DATE: 101987 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: NORTH ANNA 2 (PWR)

(NSIC 206772) A RECENT REVIEW OF THE NORTH ANNA MAIN DAM AND OTHER HYDRO STRUCTURES REVEALED THAT PREVIOUS STUDIES HAD FAILED TO IDENTIFY THE UNIT 3 & 4 CONSTRUCTION AREA AS A POTENTIAL FLOOD PATH TO THE UNIT 1 & 2 TURBINE BUILDING. PARTIAL FLOODING OF THE TURBINE BUILDING IS ADDRESSED BY THE UFSAR. CURRENT PROCEDURES PROVIDE GUIDANCE FOR ACTION TO BE TAKEN IN RESPONSE TO RISING LAKE LEVEL. AS A PERMANENT CORRECTIVE ACTION A CONCRETE WALL WILL BE CONSTRUCTED ADJACENT TO THE WEST WALL OF THE TURBINE BUILDING. THE HEIGHT OF THIS CONCRETE WALL WILL BE SUFFICIENT TO PREVENT FLOODING OF THE TURBINE BUILDING ABOVE THE STILL-WATER PROBABLE MAXIMUM FLOOD LEVEL.

[208] NORTH ANNA 1 DOCKET 50-338 LER 87-015 REV 01
UPDATE ON REACTOR TRIP DUE TO 5A FEEDWATER HEATER HIGH-HIGH LEVEL.
EVENT DATE: 062987 REPORT DATE: 110387 NSSS: WE TYPE: PWR

(NSIC 206876) ON JUNE 29, 1987, AT 2248 HOURS, UNIT 1 TRIPPED FROM 18 PERCENT POWER. UNIT 2 WAS STABLE IN MODE 1 AT 92 PERCENT POWER. THE INITIATING SIGNAL FOR THIS REACTOR TRIP WAS A TURBINE SOLENOID TRIP WHICH RESULTED FROM A 5A FEEDWATER HEATER HIGH-HIGH LEVEL SIGNAL. THE HIGH-HIGH LEVEL IN THE 5A FEEDWATER

HEATER WAS CAUSED BY AN IMPROPER VALVE LINE-UP FOLLOWING A REFUELING OUTAGE. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(IV). THE ROOT CAUSE FOR THE IMPROPER VALVE LINE-UP IS FAILURE TO FOLLOW ADMINISTRATIVE CONTROLS FOR HANGING AND REMOVING DANGER TAGS AND RETURNING VALVES TO SERVICE.

[209] NORTH ANNA 1 DOCKET 50-338 LER 87-021
LOSS OF ENVIRONMENTAL QUALIFICATION OF SI ACCUMULATOR TANK PRESSURE TRANSMITTERS.
EVENT DATE: 091187 REPORT DATE: 100887 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: NORTH ANNA 2 (PWR)
VENDOR: ROSEMOUNT, INC.

(NSIC 206622) ON SEPTEMBER 11, 1987 AT 0900 HOURS WITH UNIT 1 IN MODE 5 (COLD SHUTDOWN), IT WAS DISCOVERED THAT THE ENVIRONMENTAL SEALS ON THE SENSOR NECKS OF THE SAFETY INJECTION (SI) ACCUMULATOR TANK PRESSURE TRANSMITTERS HAD BEEN BROKEN DURING INSTALLATION IN MAY, 1987. PURSUANT TO GENERIC LETTER 85-15, THIS EVENT IS REPORTABLE UNDER 10 CFR 50.73 (A) (2) (I) (B). DURING THE INSTALLATION OF ONE OF THE SIX EQ PRESSURE TRANSMITTERS ON THE UNIT 2 SI ACCUMULATOR TANKS ON SEPTEMBER 11, 1987, THE QUALITY CONTROL (QC) INSPECTOR WITNESSED THE ELECTRONICS HOUSING BEING ALLOWED TO TURN ON THE SENSOR NECK. IT WAS THEN DISCOVERED THAT THIS ACTION VOIDS THE ENVIRONMENTAL SEAL WHICH IS BAKED ON THE SENSOR NECK. THE CONSTRUCTION PERSONNEL THEN TOLD THE QC INSPECTOR THAT THE SAME METHOD OF INSTALLATION HAD BEEN USED IN MAY, 1987, DURING THE UNIT 1 REFUELING OUTAGE, CAUSING THE UNIT 1 TRANSMITTERS TO BE ENVIRONMENTALLY UNQUALIFIED. AS A CORRECTIVE ACTION, THE UNIT 1 TRANSMITTERS WERE REMOVED AND THE SENSOR NECKS WERE RESEALED. THE CONSTRUCTION PERSONNEL WERE TRAINED TO BE CAUTIOUS OF THE NECK SEAL AND THE PROPER INSTALLATION WAS COMPLETED ON SEPTEMBER 24, 1987. QC PERSONNEL HAVE BEEN TRAINED TO BE AWARE OF POSSIBLE DAMAGE TO THE NECK SEAL DURING INSTALLATION.

[210] NORTH ANNA 1 DOCKET 50-338 LER 87-022
INADVERTENT OPENING OF A PRESSURIZER POWER OPERATED RELIEF VALVE.
EVENT DATE: 101087 REPORT DATE: 110587 NSSS: WE TYPE: PWR

(NSIC 206928) ON OCTOBER 10, 1987, AT 2117 HOURS, WITH UNIT 1 IN MODE 5 (195 DEGREES F AND PRIMARY SYSTEM PRESSURE OF 375 PSIG), A PRESSURIZER POWER OPERATED RELIEF VALVE OPENED DUE TO ACTUATION OF THE OVERPRESSURE PROTECTION SYSTEM. THE EVENT WAS CAUSED BY A BRIEF REDUCTION IN PRIMARY SYSTEM TEMPERATURE DURING THE PERFORMANCE OF AN INSERVICE INSPECTION TEST OF THE RESIDUAL HEAT REMOVAL SYSTEM VALVES. THIS TEMPERATURE REDUCTION AUTOMATICALLY REDUCED THE SETPOINT OF THE OVERPRESSURE PROTECTION SYSTEM. THE REDUCED SETPOINT WAS BELOW THE REACTOR COOLANT SYSTEM PRESSURE AND THE OVERPRESSURE PROTECTION WAS INITIATED WHICH AUTOMATICALLY OPENED AT LEAST ONE POWER OPERATED RELIEF VALVE. THIS EVENT HAD NO IMPACT ON THE HEALTH AND SAFETY OF THE PUBLIC. THIS EVENT IS REPORTABLE PURSUANT TO TECHNICAL SPECIFICATION 3.4.9.3, ACTION C, AND SUBMITTED IN ACCORDANCE WITH TECHNICAL SPECIFICATION 6.9.2.

[211] NORTH ANNA 2 DOCKET 50-339 LER 87-012
THREE VALVES FAILED TYPE C LOCAL LEAK RATE TESTING.
EVENT DATE: 083187 REPORT DATE: 101487 NSSS: WE TYPE: PWR
VENDOR: FISHER CONTROLS CO.
MISSION VALVE AND PUMP COMPANY

(NSIC 206635) ON AUGUST 31, 1987, WITH UNIT 2 IN MODE 6 (REFUELING), TYPE C LOCAL LEAK RATE TESTING (LLRT) REVEALED THAT THREE CONTAINMENT ISOLATION VALVES, ON ONE CONTAINMENT PENETRATION, HAD UNACCEPTABLE "AS FOUND" LEAK RATES. THESE VALVES PROVIDE ISOLATION FOR THE CONDENSER AIR EJECTOR DISCHARGE WHEN IT IS DIVERTED TO THE CONTAINMENT. THE "AS FOUND" LEAK RATES OF THESE VALVES EXCEEDED THE TOTAL ALLOWABLE LLRT LEAKAGE OF 0.60 LA ALLOWED BY 10CFR50 APPENDIX J AND TECHNICAL

SPECIFICATION 3.6.1.2. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(II). THE FAILURE OF THE THREE VALVES, IN THE AIR EJECTOR DIVERT FLOWPATH, TO PASS THE LLRT COULD RESULT IN A DIRECT PATH FOR LEAKAGE FROM THE CONTAINMENT ATMOSPHERE TO THE ENVIRONMENT; HOWEVER, THIS IS A MONITORED RELEASE PATH. IN ADDITION, ANOTHER VALVE (NOT REQUIRED TO BE TYPE C TESTED) EXISTS IN THIS FLOWPATH AND AUTOMATICALLY CLOSES FOLLOWING A CONDENSER AIR EJECTOR HI/HI RADIOACTIVITY SIGNAL. NO ACTUAL RELEASE OCCURRED DURING THE PREVIOUS OPERATING CYCLE BECAUSE THE CONTAINMENT IS MAINTAINED SUBATMOSPHERIC DURING POWER OPERATION IN ACCORDANCE WITH TECHNICAL SPECIFICATION 3.6.1.5. THE CAUSE OF VALVE LEAKAGE WAS DUE TO SEATING SURFACE DEGRADATION CAUSED BY DEBRIS IN THE LINES.

[212] NORTH ANNA 2 DOCKET 50-339 LER 87-008
PRESSURIZER SAFETY VALVES SET PRESSURES NOT WITHIN TECHNICAL SPECIFICATIONS.
EVENT DATE: 091387 REPORT DATE: 100687 NSSS: WE TYPE: PWR
VENDOR: DRESSER INDUSTRIAL VALVE & INST DIV

(NSIC 206623) AT 0800 HOURS ON SEPTEMBER 13, 1987, AND 1030 HOURS ON SEPTEMBER 15, 1987, WITH UNIT 2 IN MODE 6 (REFUELING) ALL THREE PRESSURIZER CODE SAFETY VALVES "AS FOUND" SET PRESSURES WERE FOUND TO BE OUT OF TOLERANCE. THESE "AS FOUND" SET PRESSURES WERE NOT WITHIN THE SET PRESSURE OF 2485 PSIG +/- 1 PERCENT ALLOWED BY TECHNICAL SPECIFICATION (T.S.) 3.4.3. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(I)(B). ON SEPTEMBER 8, 1987, ALL THREE PRESSURIZER CODE SAFETY VALVES WERE SENT TO WYLE LABS FOR THE PERFORMANCE OF THE "PRESSURIZER CODE SAFETY VALVE SETPOINT VERIFICATION" PERIODIC TEST (1-PT-50). EACH VALVE WAS FUNCTIONALLY TESTED FOR THE "AS FOUND'' SET PRESSURE AND LEAK TIGHTNESS. THE "AS FOUND" SET PRESSURES FOR THE "A" AND "B" SAFETY VALVES WERE FOUND TO BE HIGHER THAN THE MAXIMUM SET PRESSURE ALLOWED BY T.S. 3.4.3 AND THE "AS FOUND" SET PRESSURE FOR THE "C" SAFETY VALVE WAS FOUND TO BE BELOW THE MINIMUM SET PRESSURE ALLOWED BY T.S. 3.4.3. THE "A" AND "C" SAFETY VALVES LEAKED FOLLOWING THE "AS FOUND" TESTING. AS AN CORRECTIVE ACTION THE SAFETY VALVES WERE READJUSTED AND REPAIRED AT WYLE LABS TO WITHIN THE CORRECT SETPOINT TOLERANCE ALLOWED BY (T.S.) 3.4.3. FOLLOWING REPAIR AND READJUSTMENT NONE OF THE SAFETY VALVES EXHIBITED LEAKAGE.

[213] NORTH ANNA 2 DOCKET 50-339 LER 87-009
MAIN STEAM SAFETY VALVES SET PRESSURES NOT WITHIN TECHNICAL SPECIFICATION LIMIT.
EVENT DATE: 091587 REPORT DATE: 100687 NSSS: WE TYPE: PWR
VENDOR: CROSBY VALVE & GAGE CO.

(NSIC 206633) AT 1015 HOURS ON 9/15/87, WITH UNIT 2 IN MODE 6 (REFUELING) 11 OF THE 15 MAIN STEAM LINE CODE SAFETY VALVES (MSSVS) "AS FOUND" SET PRESSURES WERE FOUND TO BE HIGHER THAN THE MAXIMUM LIFT SET PRESSURES ALLOWED BY TECHNICAL SPECIFICATION 3.7.1.1. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(I)(B). ON SEPTEMBER 9, 1987, ALL 15 MSSVS WERE SENT TO WYLE LABS TO DETERMINE THE "AS FOUND" SET PRESSURE, THE AMOUNT OF DISC TO SEAT LEAKAGE UNDER LIMITED FLOW CONDITIONS AND THE SET PRESSURE, BLOWDOWN AND THE AMOUNT OF DISC TO SEAT LEAKAGE UNDER FULL-FLOW CONDITIONS. ELEVEN OF THE MSSVS "AS FOUND" SET PRESSURES EXCEEDED THE TECHNICAL SPECIFICATION LIMITS AND FOURTEEN EXHIBITED DISC TO SEAT LEAKAGE FOLLOWING THE "AS FOUND" TESTING. AS A CORRECTIVE ACTION, THE MSSVS ARE BEING REFURBISHED AND RETESTED BY WYLE LABS UNTIL THE SET PRESSURES ARE WITHIN THE ALLOWABLE LIMITS OF TECHNICAL SPECIFICATION 3.7.1.1 AND NONE OF THE VALVES EXHIBIT DISC TO SEAT LEAKAGE. UPON COMPLETION OF ALL TESTING A SUPPLEMENTAL REPORT WILL BE SUBMITTED CONTAINING THESE RESULTS. THE IMPACT OF THE HIGH "AS FOUND" MSSVS SET PRESSURES HAS BEEN REVIEWED. THE EXPECTED PEAK MAIN STEAM PRESSURE RESULTING FROM THE "AS FOUND" PRESSURIZER AND MAIN STEAM SAFETY VALVES SET PRESSURES WAS FOUND TO BE LESS THAN THE MAIN STEAM DESIGN BASIS PRESSURE.

[214] NORTH ANNA 2 DOCKET 50-339 LER 87-011
BEARING TEMPERATURE SURVEILLANCE NOT PERFORMED YEARLY AS REQUIRED BY TECHNICAL SPECIFICATIONS.
EVENT DATE: 091687 REPORT DATE: 100987 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: NORTH ANNA 1 (PWR)

(NSIC 206634) AT 1400 HOURS ON SEPTEMBER 16, 1987, WITH UNIT 2 IN MODE 6 (REFUELING), IT WAS DISCOVERED THAT THE SURVEILLANCE TESTS FOR OBTAINING BEARING TEMPERATURES ON THE UNIT 2 "A" AND "B" QUENCH SPRAY (QS) PUMPS, REQUIRED BY TECHNICAL SPECIFICATION 4.0.5, HAD NOT BEEN PERFORMED SINCE JULY 1, 1985 AND MAY 15, 1984, RESPECTIVELY. DISCOVERY OF THIS EVENT PROMPTED A REVIEW OF BOTH UNITS QS PUMPS BEARING TEMPERATURE MEASUREMENTS. WHILE REVIEWING QS PUMP HISTORY, IT WAS DISCOVERED THAT THE SURVEILLANCE TEST FOR OBTAINING BEARING TEMPERATURES ON THE UNIT 1 "B" QS PUMPS HAD NOT BEEN PERFORMED BETWEEN MAY 16, 1984 AND SEPTEMBER 15, 1986. THIS TIME PERIOD EXCEEDS THE REQUIREMENTS OF TECHNICAL SPECIFICATION 4.0.5 WHICH, IN ACCORDANCE WITH ASME SECTION XI, REQUIRES THE BEARING TEMPERATURES TO BE MEASURED DURING AT LEAST ONE IN- SERVICE TEST EACH YEAR. THIS LER HAS BEEN WRITTEN TO ADDRESS THESE EVENTS, WHICH ARE REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(I)(B). THE CAUSE FOR THE MISSED PERIODIC TESTS WAS INADEQUATE SCHEDULING CONTROLS. AS AN IMMEDIATE CORRECTIVE ACTION, THE TESTS WERE PLACED ON THE PERIODIC TEST SCHEDULING SYSTEM AND SCHEDULED TO BE PERFORMED PRIOR TO THE END OF THE PRESENT REFUELING OUTAGE. NO SIGNIFICANT SAFETY CONSEQUENCES EXISTED DURING THIS EVENT.

[215] NORTH ANNA 2 DOCKET 50-339 LER 87-010
CONTROL ROD LATCHING WITHOUT CONTAINMENT ISOLATION.
EVENT DATE: 092187 REPORT DATE: 101987 NSSS: WE TYPE: PWR

(NSIC 206797) ON SEPTEMBER 21, 1987 AT 0130 HOURS, WITH UNIT 2 IN MODE 6 (REFUELING), BOTH CONTAINMENT AIR LOCK DOORS WERE DISCOVERED TO BE OPEN DURING CONTROL ROD LATCHING (CORE ALTERATIONS). THIS IS A VIOLATION OF TECHNICAL SPECIFICATION 3.9.4 AND IS REPORTABLE PURSUANT TO 10 CFR 50.73(A)(2)(I)(B). AT 2240 HOURS, SEPTEMBER 20, 1987, AFTER COMPLETION OF SETTING THE UPPER INTERNALS IN THE REACTOR VESSEL THE CONTAINMENT AIR LOCK DOORS WERE LEFT OPEN, SINCE CORE ALTERATIONS HAD BEEN TEMPORARILY SUSPENDED. CONTROL ROD LATCHING WAS COMMENCED AT 0041 HOURS, ON SEPTEMBER 21, 1987, WITH BOTH AIR LOCK DOORS OPEN. AT 0130 HOURS ON SEPTEMBER 21, 1987, DURING HIS REVIEW OF 2-PT-91 WHICH VERIFIES CONTAINMENT ISOLATION WHILE IN MODE 6 DURING THE CORE ALTERATIONS, THE UNIT 2 SRO CONTACTED THE REFUELING SRO, INSIDE CONTAINMENT, TO VERIFY THAT AT LEAST ONE AIR LOCK DOOR WAS CLOSED AND DISCOVERED THAT BOTH DOORS WERE OPEN. UPON INQUIRY BY THE UNIT 2 SRO AS TO THE CONDITION OF THE AIR LOCK DOORS, CONTROL ROD LATCHING (I.E. CORE ALTERATIONS) WAS PHYSICALLY DISCONTINUED. AFTER DISCOVERY OF THE OPEN DOORS, CONTROL ROD LATCHING WAS OFFICIALLY SECURED AT 0140 HOURS ON SEPTEMBER 21, 1987, AND BOTH AIR LOCK DOORS WERE CLOSED. THE CAUSE OF THIS EVENT WAS PROCEDURE INADEQUACY AND PERSONNEL ERROR. PROCEDURES WILL BE REVISED TO ENSURE VERIFICATION OF CONTAINMENT ISOLATION DURING CORE ALTERATIONS.

[216] NORTH ANNA 2 DOCKET 50-339 LER 87-013
NORMAL CHARGING FLOW INTO THE REACTOR COOLANT SYSTEM VIA THE HIGH HEAD SAFETY INJECTION FLOWPATH.
EVENT DATE: 102687 REPORT DATE: 111387 NSSS: WE TYPE: PWR

(NSIC 206929) AT 1440 HOURS ON OCTOBER 26, 1987, WITH UNIT 2 IN MODE 5 (REACTOR COOLANT SYSTEM PRIMARY TEMPERATURE AT 175 DEGREES F AND PRIMARY PRESSURE AT 335 PSIG, AND THE PRIMARY SYSTEM IN SOLID WATER OPERATION), AN INADVERTENT ENGINEERED SAFETY FEATURES ACTUATION OCCURRED DURING THE PERFORMANCE OF A VALVE INSERVICE INSPECTION WHEN THE WRONG VALVE WAS OPENED. DUE TO THE VALVE LINEUP CONDITIONS AT THE TIME OF THIS EVENT, OPENING THIS VALVE CAUSED FLOW THROUGH A SAFETY INJECTION LINE INTO THE REACTOR COOLANT SYSTEM. THIS FLOW INCREASED PRIMARY

SYSTEM PRESSURE AND RESULTED IN BOTH PRESSURIZER POWER OPERATED RELIEF VALVES OPENING. THIS EVENT IS REPORTABLE PURSUANT TO THE REQUIREMENTS OF 10CFR50.73(A)(2)(IV) AND TECHNICAL SPECIFICATION 3.4.9.3., ACTION C. THE CAUSE OF THIS EVENT IS FAILURE TO FOLLOW PROCEDURE AND PERSONNEL ERROR. AS AN IMMEDIATE CORRECTIVE ACTION, THE OPERATOR CLOSED THE VALVE. NO SIGNIFICANT SAFETY CONSEQUENCES EXISTED DURING THIS EVENT BECAUSE THE PRESSURIZER POWER OPERATED RELIEF VALVES RESPONDED AS EXPECTED TO PROVIDE OVERPRESSURIZATION PROTECTION AT LOW RCS TEMPERATURES.

[217] OYSTER CREEK DOCKET 50-219 LER 87-030 REV 01
UPDATE ON LIGHTNING ARRESTOR INSULATOR FAILURE INDUCED VOLTAGE TRANSIENT CAUSED CONTAINMENT ISOLATION & SBGTS INITIATION DUE TO AUTOMATIC BUS TRANSFER TIME EXCEEDING RPS RELAY DROPOUT.
EVENT DATE: 042286 REPORT DATE: 110587 NSSS: GE TYPE: BWR

(NSIC 206871) ON APRIL 22, 1986 AT APPROXIMATELY 0700 HOURS, PRIMARY AND SECONDARY CONTAINMENTS ISOLATED AND THE STANDBY GAS TREATMENT SYSTEM (SBGTS) INITIATED AS A RESULT OF A VOLTAGE TRANSIENT CAUSED BY A LIGHTNING ARRESTOR INSULATOR FAILURE. THE VOLTAGE TRANSIENT CAUSED VITAL AC POWER PANEL 1 (VACP-1) TO TRANSFER TO ITS ALTERNATE POWER SUPPLY. THE POWER SUPPLY TRANSFER CAUSED SEVERAL REACTOR PROTECTION SYSTEM (RPS) RELAYS TO DEENERGIZE, CAUSING THE CONTAINMENT ISOLATIONS AND SBGTS INITIATION. AT THE TIME OF THIS EVENT, THE REACTOR MODE SWITCH WAS LOCKED IN THE SHUTDOWN POSITION, THE VESSEL HEAD HAD BEEN REMOVED AND THE VESSEL AND REACTOR CAVITY HAD BEEN FLOODED IN PREPARATION FOR REFUELING. THE SHORTEST TRANSFER TIME ACHIEVABLE WITH THE AUTOMATIC TRANSFER SWITCH EXCEEDS THE DROPOUT TIME FOR THE RPS RELAYS. THE ISOLATION SIGNAL WAS RESET AND SBGTS WAS SECURED FOLLOWING THE EVENT. THE SAFETY SIGNIFICANCE OF THIS EVENT IS MINIMAL AS ONLY A CHALLENGE TO CONTAINMENT ISOLATION AND SBGTS INITIATION LOGIC CIRCUITS OCCURRED. TO PREVENT RECURRENCE OF THIS EVENT, ENGINEERING HAS PROPOSED A CONTINUOUS POWER SUPPLY BE CONNECTED TO THE CIRCUIT WHICH CONTAINS THE RELAYS THAT CAUSE CONTAINMENT ISOLATIONS AND SBGTS INITIATIONS WHEN CERTAIN POWER INTERRUPTIONS OCCUR.

[218] OYSTER CREEK DOCKET 50-219 LER 87-033
SAFETY LIMIT VIOLATION CAUSED BY PERSONNEL ERROR WHILE REMOVING REACTOR RECIRCULATION PUMPS FROM SERVICE.
EVENT DATE: 091187 REPORT DATE: 100987 NSSS: GE TYPE: BWR

(NSIC 206699) ON SEPTEMBER 11, 1987, AT 0217 HOURS A TECHNICAL SPECIFICATION SAFETY LIMIT REQUIRING AT LEAST TWO RECIRCULATION LOOPS TO BE FULLY OPEN WAS VIOLATED. THE SAFETY LIMIT WAS VIOLATED WHILE REMOVING REACTOR RECIRCULATION PUMPS FROM SERVICE IN RESPONSE TO A LEAK FROM THE REACTOR BUILDING CLOSED COOLING WATER SYSTEM WHICH OCCURRED DURING MAINTENANCE ACTIVITIES. THE PLANT WAS SHUT DOWN WITH REACTOR COOLANT TEMPERATURE APPROXIMATELY 140F. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR. UPON VIOLATION OF THE SAFETY LIMIT, IMMEDIATE CORRECTIVE ACTIONS WERE TAKEN TO OPEN OTHER RECIRCULATION LOOPS TO RESTORE THE PLANT TO AN ALLOWABLE CONFIGURATION. ANALYSIS OF THE DATA RELATED TO THE EVENT INDICATES THAT THERE WAS MINIMAL REDUCTION IN RECIRCULATION LOOP FLOW AREA DUE TO IMMEDIATE OPERATOR RESPONSES, THUS PROVIDING ADEQUATE COMMUNICATION BETWEEN THE CORE AREA AND VESSEL ANNULUS FOR VESSEL LEVEL INSTRUMENTATION. THEREFORE THE EVENT POSED NO DANGER TO THE HEALTH AND SAFETY OF THE PUBLIC, AND DID NOT CREATE THE POTENTIAL FOR CORE DAMAGE TO OCCUR. CORRECTIVE ACTIONS INCLUDE PROCEDURE REVISIONS, ADDITIONAL TRAINING FOR OPERATORS, AND EVALUATION OF A POSSIBLE TECHNICAL SPECIFICATION CHANGE REQUEST.

[219] OYSTER CREEK DOCKET 50-219 LER 87-034
INADEQUATE EMERGENCY LIGHTING DUE TO DESIGN DEFICIENCY.
EVENT DATE: 091687 REPORT DATE: 101687 NSSS: GE TYPE: BWR

(NSIC 206700) ON SEPTEMBER 16, 1987, DURING A DESIGN REVIEW OF THE EMERGENCY LIGHTING SYSTEM, IT WAS DISCOVERED THAT EMERGENCY LIGHTING IN THE A/B BATTERY ROOM AND A PORTION OF THE ACCESS ROUTE THERETO WAS INADEQUATE DUE TO A DESIGN DEFICIENCY. THE EXISTING LIGHTING IS INADEQUATE TO PERFORM MANUAL ACTIONS WHICH ARE REQUIRED IN THE EVENT OF A FIRE IN THE "A" 480 VOLT SWITCHGEAR ROOM (FIRE ZONE OB-FZ-6A). THE SAFETY SIGNIFICANCE OF THIS CONDITION IS MINIMAL SINCE A MINIMUM OF 3 HOURS IS AVAILABLE TO PERFORM THIS ACTION UTILIZING READILY AVAILABLE PORTABLE LIGHTING UNITS. MODIFICATIONS TO PROVIDE THE REQUIRED LIGHTING CAPABILITY HAVE BEEN COMPLETED.

[220] OYSTER CREEK DOCKET 50-219 LER 87-035
REACTOR SCRAM WITH REACTOR SHUTDOWN DUE TO DEGRADED ELECTRICAL CABLE.
EVENT DATE: 092487 REPORT DATE: 101987 NSSS: GE TYPE: BWR

(NSIC 206701) ON SEPTEMBER 24, 1987 AT 1:10 AM WITH THE REACTOR SHUTDOWN A REACTOR FULL SCRAM SIGNAL OCCURRED. AT THE TIME AN INSTRUMENT SURVEILLANCE WAS IN PROGRESS IN THE CONTROL ROOM THAT CAUSES HALF SCRAMS DURING THE COURSE OF ITS PERFORMANCE. COINCIDENT WITH A HALF SCRAM FROM THE SURVEILLANCE, AN INSTRUMENT TECHNICIAN OPENED AN INSTRUMENT DRAWER IN THE CONTROL ROOM ON THE NUCLEAR INSTRUMENTATION (UNRELATED TO THE SURVEILLANCE) TO REPLACE A CALIBRATION STICKER. A FULL SCRAM OCCURRED. THE HALF SCRAM FROM THE NUCLEAR INSTRUMENTATION IS ATTRIBUTED TO A POOR ELECTRICAL GROUND CAUSED BY A DEGRADED CONNECTION AT A REMOTE CABLE CONNECTOR JOINT. THE DEGRADED GROUND CAUSED A SPURIOUS SPIKE IN AN INTERMEDIATE RANGE MONITOR (IRM) WHEN ITS DRAWER WAS MOVED. ALL EQUIPMENT RESPONSES AND INDICATIONS WERE NORMAL DURING AND FOLLOWING THE SCRAM SIGNAL. THE SAFETY SIGNIFICANCE OF THIS EVENT IS CONSIDERED MINIMAL. THE CAUSE OF THE DEGRADED CONNECTION APPEARS TO BE DUE TO NORMAL WEAR AND TEAR FROM ROUTINE TESTING AGGRAVATED BY THE SHORT LENGTH OF THE CABLE ATTACHED TO THE CONNECTOR. THE CONNECTION WILL BE REPAIRED AND THE CABLE LENGTHENED TO REDUCE STRESS AT THE CONNECTION. OTHER IRM CABLES WILL BE INSPECTED FOR SIMILAR CONDITIONS AND REPAIRED AS NECESSARY.

[221] OYSTER CREEK DOCKET 50-219 LER 87-037
TECHNICAL SPECIFICATION VIOLATION CAUSED BY FAILURE TO PERFORM TWO MONTHLY DOSE CALCULATIONS DUE TO PERSONNEL ERROR.
EVENT DATE: 100987 REPORT DATE: 110687 NSSS: GE TYPE: BWR

(NSIC 206900) ON OCTOBER 9, 1987 IT WAS DISCOVERED THAT THE MONTHLY CALCULATIONS OF DOSE DUE TO RADIOIODINE AND PARTICULATES IN GASEOUS EFFLUENTS WERE NOT PERFORMED FOR JULY AND AUGUST 1987. THESE CALCULATIONS ARE REQUIRED BY THE PLANT'S RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS. AT THE TIME OF DISCOVERY THE PLANT WAS SHUT DOWN FOR MAINTENANCE, BUT THE PLANT HAD BEEN OPERATING AT FULL POWER FOR MOST OF JULY AND AUGUST. THE OCCURRENCE IS ATTRIBUTED TO PERSONNEL ERROR DUE TO INADEQUATE TRAINING. THE SAFETY SIGNIFICANCE OF THIS EVENT IS MINIMAL SINCE DATA WAS REGULARLY COLLECTED, NO EXCURSIONS WERE NOTED, AND RESULTS OF THE SUBSEQUENT CALCULATIONS SHOWED RELEASES WERE WELL BELOW THE ANNUAL LIMIT. IMMEDIATE CORRECTIVE ACTION WAS TAKEN TO COMPLETE THE CALCULATIONS UPON DISCOVERY OF THE CONDITION. FUTURE CORRECTIVE ACTION IS TRAINING OF MORE PERSONNEL ON PERFORMANCE OF DOSE CALCULATIONS.

[222] PALISADES DOCKET 50-255 LER 87-031
DIESEL GENERATOR INOPERABILITY DUE TO SLOW VOLTAGE REGULATOR RESPONSE.
EVENT DATE: 090387 REPORT DATE: 100587 NSSS: CE TYPE: PWR
VENDOR: BASLER ELECTRIC COMPANY

(NSIC 206747) ON SEPTEMBER 3, 1987 AT 1100, 1-1 DIESEL GENERATOR (DG) (EK;DG) WAS DECLARED ADMINISTRATIVELY INOPERABLE DUE TO POTENTIAL DEFICIENCIES IN THE EXCITER VOLTAGE REGULATOR (EK;EXC) RESPONSE TIME AND THE TERMINAL VOLTAGE SETTING. THIS POTENTIAL DEFICIENCY WAS IDENTIFIED BY CONSUMERS POWER ENGINEERING PERSONNEL THROUGH TECHNICAL SPECIFICATION (TS) SURVEILLANCE PROCEDURE DATA REVIEWS AND SUBSEQUENT COMPUTER SIMULATIONS TO MODEL ACCIDENT CONDITIONS. THE REACTOR WAS CRITICAL WITH THE PLANT OPERATING AT 92 PERCENT OF RATED POWER WHEN THE CONDITION WAS IDENTIFIED. WHILE DEVELOPING AND UPDATING THE CONTINUOUS SYSTEM MODELING PROGRAM (CSMP) FOR THE DG COMPUTER MODEL WHICH SIMULATES DG PERFORMANCE USING DATA OBTAINED FROM TS SURVEILLANCE PROCEDURE RO-8, "ENGINEERED SAFEGUARDS SYSTEMS", ENGINEERING PERSONNEL IDENTIFIED THAT 1-1 DG MAY NOT BE ABLE TO SUFFICIENTLY MAINTAIN OUTPUT VOLTAGE TO THE REQUIRED EQUIPMENT FOR SHORT PERIODS OF TIME UNDER DESIGN BASIS ACCIDENT (DBA) CONDITIONS. THE VOLTAGE TRANSIENT DUE TO STARTING AUXILIARY FEEDWATER PUMP (SJ;P) P8A WITH THE "AS FOUND" SLOW VOLTAGE TIME RESPONSE COMBINED WITH A LOW REFERENCE VOLTAGE SETTING, CREATED THE POTENTIAL FOR A LOSS OF THE 1-1 DG DUE TO LOW OUTPUT VOLTAGE.

[223] PALISADES DOCKET 50-255 LER 87-033
DETECTOR FAILURE AND INOPERABILITY GREATER THAN 7 DAY'S RESULTS IN TECH SPEC SPECIAL REPORT.
EVENT DATE: 090387 REPORT DATE: 101287 NSSS: CE TYPE: PWR
VENDOR: VICTOREEN INSTRUMENT DIVISION

(NSIC 2066[illegible]) ON SEPTEMBER 3, 1987 AT 1230, HIGH RANGE NOBLE GAS STACK MONITOR, RIA-2327 [[illegible]] WAS DECLARED INOPERABLE WHEN DETECTOR OUTPUT VALUES FAILED TO FALL WITHIN THE ACCEPTANCE CRITERIA AS DEFINED IN TECHNICAL SPECIFICATION (TS) SURVEILLANCE PROCEDURE RR-84C, "HIGH RANGE NOBLE GAS EFFLUENT MONITOR RE-2327 CALIBRATION". IN ACCORDANCE WITH ACTION STATEMENT 38 OF TS 3.24.2, THE PREPLANNED ALTERNATE MONITORING INSTRUMENTATION WAS INSTALLED AND DECLARED OPERABLE AT 1505. ON SEPTEMBER 10, 1987, AFTER EXTENSIVE TROUBLESHOOTING BY PLANT INSTRUMENT AND CONTROL PERSONNEL, THE DETECTOR WAS SENT TO THE MANUFACTURER FOR REPAIR OR REPLACEMENT. THIS REPORT IS BEING SUBMITTED IN ACCORDANCE WITH ACTION STATEMENT 38, WHICH REQUIRES A SPECIAL REPORT BE SUBMITTED IF THE INOPERABLE EQUIPMENT IS NOT RETURNED TO OPERABLE STATUS WITHIN SEVEN DAYS. PRELIMINARY INVESTIGATIONS BY THE MANUFACTURER INDICATE THE DETECTOR ANOMALY WAS CAUSED BY A FAILURE OF THE IONIZATION CHAMBER ANODE INSULATOR. A FULL FAILURE ANALYSIS WILL BE PROVIDED BY THE MANUFACTURER. CURRENT PLANS ARE TO REFURBISH THE FAILED DETECTOR AND PLACE IT BACK IN SERVICE BY NOVEMBER 15, 1987.

[224] PALISADES DOCKET 50-255 LER 87-032
PERSONNEL ERROR DURING PREVENTIVE MAINTENANCE ACTIVITY RESULTS IN INADVERTENT ENGINEERED SAFETY FEATURE ACTUATION.
EVENT DATE: 091087 REPORT DATE: 101287 NSSS: CE TYPE: PWR

(NSIC 206666) ON SEPTEMBER 10, 1987 AT 0845, LOW PRESSURE SAFETY INJECTION PUMP (LPSI), P-67B (BP;P), WAS INADVERTENTLY ACTUATED DURING THE PERFORMANCE OF A PREVENTIVE MAINTENANCE ACTIVITY TO CLEAN THE CONTACTS OF THE DESIGN BASIS ACCIDENT (DBA) SEQUENCER (JE;10). THE REACTOR WAS CRITICAL WITH THE PLANT OPERATING AT 93 PERCENT OF RATED POWER WHEN THE EVENT OCCURRED. AS PART OF PREVENTIVE MAINTENANCE ACTIVITY, ESS-036, THE TIGHTNESS OF THE SEQUENCER CAM LOCKING SCREW IS PHYSICALLY VERIFIED. WHILE VERIFYING TIGHTNESS, THE SCREWDRIVER BEING USED SLIPPED OFF THE HEAD OF THE LOCKING SCREW, CAUSING THE SEQUENCER CONTACTS WHICH ACTUATE P-67B TO MOMENTARILY CLOSE. THE LPSI PUMP WAS SECURED AND MAINTENANCE ACTIVITY SUSPENDED PENDING EVALUATION. THE LOCKING SCREW TIGHTNESS VERIFICATION HAS BEEN REMOVED FROM PREVENTIVE MAINTENANCE ACTIVITY ESS-036 AND PLACED IN ESS-100. THIS LATER ACTIVITY WILL ALLOW TIGHTNESS VERIFICATION WITH SEQUENCERS CONTACTS ISOLATED. CURRENT PLANS ARE TO REPLACE THE EXISTING ROTATING CAM, MECHANICAL SEQUENCERS WITH SOLID STATE PROGRAMMABLE SEQUENCERS.

[225] PALISADES DOCKET 50-255 LER 87-034
INADEQUATE PROCEDURE RESULTS IN RADIOACTIVE EFFLUENT TECHNICAL SPECIFICATION NONCOMPLIANCE.
EVENT DATE: 100287 REPORT DATE: 110287 NSSS: CE TYPE: PWR
VENDOR: METAL BELLOWS
VICTOREEN INSTRUMENT DIVISION

(NSIC 206873) ON OCTOBER 2, 1987 AT APPROXIMATELY 0603, A PLANT RADIATION PROTECTION TECHNICIAN (RPT) IDENTIFIED THAT NORMAL RANGE NOBLE GAS STACK EFFLUENT MONITOR, RIA-2326 (IL;RE) AND NORMAL RANGE PARTICULATE STACK EFFLUENT MONITOR, RIA-2325 (IL;RE) WERE BOTH INOPERABLE DUE TO A LOW FLOW CONDITION. THE RPT IDENTIFIED (ILLUMINATED LOCAL CONTROL ROOM PANEL INDICATOR) THE CONDITION DURING THE PERFORMANCE OF TECHNICAL SPECIFICATION (TS) SURVEILLANCE PROCEDURE DWR-10, "STACK EFFLUENT SAMPLING, CALCULATION AND RECORDS". UPON REVIEWING STRIP CHART RECORDER RESULTS, THE CONDITION WAS DETERMINED TO HAVE EXISTED FOR APPROXIMATELY 13.3 HOURS. FURTHER REVIEWS ALSO INDICATED NO ALTERNATIVE SAMPLING WAS PERFORMED. THIS CONDITION IS CONTRARY TO ACTION STATEMENT 37 OF TS 3.24.2 WHICH REQUIRES A GRAB SAMPLE BE TAKEN AT LEAST ONCE PER 12 HOURS WHEN THE MINIMUM CHANNELS OPERABLE REQUIREMENT IS NOT MET. THE PLANT WAS IN HOT SHUTDOWN CONDITION (PRIMARY COOLANT SYSTEM 529 DEGREES F) WHEN THE CONDITION WAS IDENTIFIED. OPERATORS DID NOT IDENTIFY THE FAILURE DUE TO AN INADEQUATE TS SURVEILLANCE PROCEDURE WHICH DEMONSTRATES OPERABILITY OF THE MONITORS AND THE LACK OF REFLASH CAPABILITY WITHIN THE ALARM WINDOW. THE ALARM WINDOW WILL BE MODIFIED TO INCLUDE REFLASH CAPABILITIES AND THE TS SURVEILLANCE PROCEDURE WILL BE REVISED TO DIRECT OPERATORS TO REVIEW THE STATUS OF LOCAL STACK FLOW INDICATOR LIGHTS.

[226] PALISADES DOCKET 50-255 LER 87-035
PERSONNEL ERROR RESULTS IN TEMPORARY LOSS OF SHUTDOWN COOLING.
EVENT DATE: 101587 REPORT DATE: 111687 NSSS: CE TYPE: PWR

(NSIC 206909) ON OCTOBER 15, 1987, AT 1837, LOW PRESSURE SAFETY INJECTION (LPSI) PUMP, P-67A (BP;P) WAS MANUALLY SECURED FROM OPERATION DUE TO ERRATIC DISCHARGE PRESSURE AND FLOW. THE REACTOR WAS IN COLD SHUTDOWN CONDITION WITH THE PRIMARY COOLANT SYSTEM (PCS) (AB) DRAINED TO THE 617 FOOT EIGHT INCH LEVEL (CENTERLINE OF THE HOT AND COLD LEGS IS AT 618 FEET TWO INCHES) AT THE TIME OF THE EVENT. THE PCS WAS DRAINED TO THE CENTERLINE IN ORDER TO SUPPORT STEAM GENERATOR (AB;SG) NOZZLE DAM MODIFICATIONS. AT THE TIME OF THE EVENT, LPSI PUMP P-67A WAS TAKING SUCTION FROM THE PCS AT THE HOT LEG, DISCHARGING THROUGH SHUTDOWN COOLING HEAT EXCHANGERS E-60A AND E-60B (BP;HX) AND RETURNING FLOW TO THE PCS AT THE COLD LEGS. THE ERRATIC DISCHARGE PRESSURE AND FLOW WERE THE RESULT OF AN IMPROPERLY PLACED JUMPER WHICH CAUSED A LPSI DISCHARGE VALVE TO CYCLE OPEN/CLOSED. THE FAILURE TO PROPERLY PLACE THE JUMPER WAS THE RESULT OF A DATA TRANSPOSITION ERROR DURING THE PLANNING PHASE FOR "AS-LEFT" VALVE TESTING. SHUTDOWN COOLING FLOW WAS ISOLATED FOR 29 MINUTES, WITH PCS TEMPERATURE INCREASING FROM 92 TO 129 DEGREES F. SHUTDOWN COOLING FLOW WAS RESTORED AFTER THE ERRANTLY PLACED JUMPER WAS REMOVED. ALL SIMILAR VALVE TESTING WAS IMMEDIATELY STOPPED AND ALL JUMPERS INSTALLED TO SUPPORT TESTING REMOVED.

[227] PALO VERDE 1 DOCKET 50-528 LER 87-022
FUEL HANDLING CRANE USED PRIOR TO PERFORMANCE OF SURVEILLANCE TEST.
EVENT DATE: 100187 REPORT DATE: 102887 NSSS: CE TYPE: PWR

(NSIC 206845) AT 1355 MST ON OCTOBER 1, 1987, PALO VERDE UNIT 1 WAS IN MODE 1 (POWER OPERATION) WHEN IT WAS IDENTIFIED THAT THE 10 TON NEW FUEL HANDLING CRANE WAS NOT SURVEILLANCE TESTED PRIOR TO USE ON SEPTEMBER 23, 1987 AND OCTOBER 1, 1987. THIS WAS CONTRARY TO SURVEILLANCE REQUIREMENTS 4.9.7. THE CAUSE OF THE EVENT WAS THAT THE GUIDANCE PROVIDED TO THE PERSONNEL WHO OPERATE THE CRANE DID NOT CONTAIN THE REQUIREMENT TO COMPLETE THE SURVEILLANCE TEST PRIOR TO USE. IMMEDIATE CORRECTIVE ACTION WAS TO PERFORM THE SURVEILLANCE TEST. THE PERSONNEL

INVOLVED WERE INSTRUCTED BY THEIR FOREMAN ON THE PERFORMANCE OF THE SURVEILLANCE TESTS PRIOR TO USE OF THE CRANE. ADDITIONALLY, A YELLOW CAUTION TAG WAS PLACED ON THE POWER SUPPLY DISCONNECT SWITCH TO CAUTION USERS TO VERIFY THAT THE SURVEILLANCE TEST IS COMPLETE PRIOR TO USING CRANE. AS CORRECTIVE ACTION TO PREVENT RECURRENCE, A TAG MADE OF 3 PLY LAMINATED ENGRAVING STOCK WILL BE ATTACHED TO THE PENDANT OF THE 10 TON NEW FUEL HANDLING CRANE. A TAG WILL ALSO BE ATTACHED TO THE KEY FOR THE CRANE. THESE TAGS WILL ALERT PERSONNEL TO VERIFY THAT THE SURVEILLANCE TEST HAS BEEN SATISFACTORILY COMPLETED BEFORE USING THE CRANE. THERE WERE NO PREVIOUS SIMILAR EVENTS REPORTED.

[228] PALO VERDE 2 DOCKET 50-529 LER 86-044 REV 01
UPDATE ON BOTH TRAINS OF CONTROL ROOM ESSENTIAL FILTRATION SYSTEM INOPERABLE DURING MAINTENANCE DUE TO INADEQUATE DESIGN.
EVENT DATE: 042386 REPORT DATE: 101587 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: PALO VERDE 1 (PWR)
PALO VERDE 3 (PWR)

(NSIC 206776) ON MAY 15, 1987, IT WAS DISCOVERED THAT BEGINNING ON APRIL 23, 1986 AT 1500 MST THROUGH APRIL 25, 1986 AT 1004, WITH PALO VERDE UNIT 2 IN MODE 3 (HOT STANDBY), BOTH INDEPENDENT TRAINS OF THE CONTROL ROOM ESSENTIAL FILTRATION SYSTEM (CREFS) WERE INOPERABLE AS A RESULT OF A MAINTENANCE ACTIVITY BEING CONDUCTED ON ONE TRAIN. WITH BOTH TRAINS OF CREFS INOPERABLE, THE ACTION STATEMENT FOR TECHNICAL SPECIFICATION 3.7.7 WAS EXCEEDED AND LIMITING CONDITION FOR OPERATION 3.0.3 SHOULD HAVE BEEN ENTERED. AS A RESULT OF AN ENGINEERING EVALUATION, IT WAS DISCOVERED THAT THE DESIGN OF THE CREFS IS SUCH THAT THE OPENING OF A SINGLE DUCT OR HOUSING ACCESS DOOR IN EITHER TRAIN OF CREFS CAN PREVENT THE UNAFFECTED TRAIN FROM MEETING THE CONTROL ROOM PRESSURIZATION REQUIREMENT OF TECHNICAL SPECIFICATION 4.7.7.D.3. A SUBSEQUENT REVIEW OF THE WORK CONDUCTED ON THIS SYSTEM LED TO THE DISCOVERY OF THE CREFS' APRIL 23, 1986 INOPERABILITY. INTERIM CONTROLS HAVE BEEN ESTABLISHED TO PRECLUDE UNAUTHORIZED OPENING OF THE DOORS/INSPECTION PANELS AFFECTING THE CONTROL ROOM PRESSURE BOUNDARY. AS LONG TERM CORRECTIVE ACTION TO PREVENT RECURRENCE, A PLANT CHANGE REQUEST HAS BEEN INITIATED TO INSTALL A PAIR OF ISOLATION DAMPERS IN EACH TRAIN OF CREFS FOR UNITS 1, 2, AND 3.

[229] PALO VERDE 2 DOCKET 50-529 LER 87-018
ESF ACTUATION CAUSED BY SPURIOUS SIGNAL FROM A RADIATION MONITOR.
EVENT DATE: 100487 REPORT DATE: 102987 NSSS: CE TYPE: PWR
VENDOR: CARRIER CORP.

(NSIC 206891) ON OCTOBER 4, 1987 AT 0750 MST, WITH PALO VERDE UNIT 2 IN MODE 1 (POWER OPERATION) OPERATING AT APPROXIMATELY 100 PERCENT POWER, A CONTROL ROOM ESSENTIAL FILTRATION ACTUATION SIGNAL (CREFAS) WAS AUTOMATICALLY INITIATED ON TRAIN "B" AND CROSS-TRIPPED TRAIN "A" AS DESIGNED. THIS ENGINEERED SAFETY FEATURE ACTUATION RESULTED FROM A SPURIOUS ALARM/TRIP SIGNAL ON THE "B" CONTROL ROOM VENTILATION INTAKE NOBLE GAS MONITOR (RU-30). ALL ASSOCIATED EQUIPMENT ACTUATED SATISFACTORILY WITH THE EXCEPTION OF THE "B" ESSENTIAL CHILLER WHICH TRIPPED ON HIGH BEARING TEMPERATURE AFTER STARTING. THE ROOT CAUSE OF THE SPURIOUS SIGNAL IS BELIEVED TO BE A VOLTAGE SPIKE WHICH WAS CAUSED BY ELECTRONIC CIRCUIT NOISE IN RU-30. THE ROOT CAUSE OF THE ESSENTIAL CHILLER TRIP WAS DUE TO AN EXCESSIVE OIL LEVEL IN THE SUMP. FOLLOWING THE CREFAS, THE "A" CONTROL ROOM VENTILATION INTAKE NOBLE GAS MONITOR (RU-29) AND THE PLANT VENT LOW RANGE MONITOR (RU-143) WERE VERIFIED TO BE READING NORMAL, THUS ENSURING THAT THERE WERE NO ACTUAL RADIATION LEVEL INCREASES. THE ACTUATED EQUIPMENT WAS SUBSEQUENTLY RESET AND RU-30 AND ESSENTIAL CHILLER "B" WERE DECLARED INOPERABLE PENDING INVESTIGATION OF THE EVENT. TO PREVENT RECURRENCE, A PLANT CHANGE PACKAGE (PCP) HAD PREVIOUSLY BEEN APPROVED FOR PALO VERDE UNITS 1, 2, AND 3.

[230] PALO VERDE 2 DOCKET 50-529 LER 87-020
ESF ACTUATION CAUSED BY SPURIOUS SIGNAL FROM A RADIATION MONITOR.
EVENT DATE: 101387 REPORT DATE: 110987 NSSS: CE TYPE: PWR

(NSIC 207122) ON OCTOBER 13, 1987 AT 2326 MST, WITH PALO VERDE UNIT 2 IN MODE 1 (POWER OPERATION) OPERATING AT APPROXIMATELY 100 PERCENT POWER, A CONTROL ROOM ESSENTIAL FILTRATION ACTUATION SIGNAL (CREFAS) WAS AUTOMATICALLY INITIATED ON TRAIN "B" AND CROSS-TRIPPED TRAIN "A" AS DESIGNED. THIS ENGINEERED SAFETY FEATURE ACTUATION RESULTED FROM A SPURIOUS ALARM/TRIP SIGNAL ON THE "B" CONTROL ROOM VENTILATION INTAKE NOBLE GAS MONITOR (RU-30). ALL ASSOCIATED EQUIPMENT ACTUATED SATISFACTORILY. FOLLOWING THE CREFAS, THE "A" CONTROL ROOM VENTILATION INTAKE NOBLE GAS MONITOR (RU-29) AND THE PLANT VENT LOW RANGE MONITOR (RU-143) WERE VERIFIED TO BE READING NORMAL, THUS ENSURING THAT THERE WERE NO ACTUAL RADIATION LEVEL INCREASES. THE ACTUATED EQUIPMENT WAS SUBSEQUENTLY RESET AND AN APPROVED WORK ORDER WAS INITIATED TO INVESTIGATE THE CAUSE OF THE EVENT. THE ROOT CAUSE OF THE SPURIOUS SIGNAL IS BELIEVED TO BE A VOLTAGE SPIKE WHICH WAS CAUSED BY ELECTRONIC CIRCUIT NOISE IN RU-30. TO PREVENT RECURRENCE, A PLANT CHANGE PACKAGE HAD PREVIOUSLY BEEN APPROVED FOR PALO VERDE UNITS 1, 2 AND 3 TO INSTALL AN ISOLATED GROUNDING SYSTEM FOR THE RADIATION MONITORING SYSTEM.

[231] PEACH BOTTOM 2 DOCKET 50-277 LER 87-016
CONTAINMENT ISOLATIONS ON LOSS OF A STARTUP SOURCE DUE TO CRANE OPERATION.
EVENT DATE: 082087 REPORT DATE: 092187 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: PEACH BOTTOM 3 (BWR)

(NSIC 206489) ON AUGUST 20, 1987 AT 1305 HOURS CONTAINMENT ISOLATIONS OCCURRED ON BOTH UNITS WHEN THE UNIT 2 STARTUP SOURCE TRIPPED. MOBILE CRANE MOVEMENT ON-SITE DREW AN ARC FROM THE STARTUP SOURCE LINE TO THE CRANE CAUSING THE TRIP. THE AFFECTED EMERGENCY BUSES AUTOMATICALLY SWITCHED TO THE UNIT 3 STARTUP SOURCE. THE DE ENERGIZATION OF RELAYS DURING THE SWITCHING CAUSED ISOLATIONS AND PUMP TRIPS ON BOTH UNITS INVOLVING VARIOUS SYSTEMS. THE UNIT 2 'A' REACTOR PROTECTION SYSTEM M-G SET OUTPUT BREAKERS TRIPPED ON UNDERVOLTAGE FOLLOWING THE EMERGENCY BUS SWITCHING, CAUSING A HALF-SCRAM SIGNAL. NO CONTROL RODS MOVED. NO ONE WAS INJURED, HOWEVER THE POTENTIAL EXISTED FOR SEVERE PERSONNEL INJURY. THERE WERE NO ADVERSE SAFETY CONSEQUENCES TO THE PLANT. DECAY HEAT REMOVAL WAS RESTORED WITHIN 30 MINUTES TO BOTH UNITS. THE DECAY HEAT LOAD ON BOTH UNITS WAS VERY LOW BECAUSE THE REACTORS HAD BEEN SHUTDOWN FOR APPROXIMATELY FIVE MONTHS. THE ROOT CAUSE OF THIS EVENT WAS INADEQUATE CONCERN FOR A POTENTIAL ELECTRICAL HAZARD BY MAINTENANCE CRAFT SUPERVISION. THE SUPERVISORS INVOLVED WERE COUNSELED, ALL MAINTENANCE PERSONNEL AND STATION GROUP SUPERVISORS WERE NOTIFIED OF THE EVENT, ADDITIONAL TRAINING IS PLANNED FOR MAINTENANCE FOREMEN TO DISCUSS THE EVENT, AND ALL MAINTENANCE RIGGERS WILL ATTEND A ONE HOUR TRAINING CLASS.

[232] PEACH BOTTOM 2 DOCKET 50-277 LER 87-019
SHUTDOWN COOLING ISOLATION VIA PRIMARY CONTAINMENT ISOLATION SYSTEM.
EVENT DATE: 090487 REPORT DATE: 100587 NSSS: GE TYPE: BWR

(NSIC 206714) ON SEPTEMBER 4, 1987, AT 0305 HOURS WITH THE UNIT IN COLD SHUTDOWN, A SHUTDOWN COOLING ISOLATION AND RESIDUAL HEAT REMOVAL (RHR) PUMP TRIP OCCURRED WHEN THE "A" RHR LOGIC WAS TEMPORARILY DE-ENERGIZED. THE CAUSE OF THE RHR LOGIC BEING DE-ENERGIZED WAS A LOOSE CONNECTION AT THE FUSE TERMINAL. THE CAUSE OF THE LOOSE CONNECTION IS NOT CERTAIN. HOWEVER, PREVIOUS HANDLING OF THE ASSOCIATED WIRING DURING WORK ACTIVITIES AROUND THE FUSE TERMINAL AND THE DESIGN OF THE CONNECTION MAY BE CONTRIBUTORS. PRIOR TO SUBMITTING A BLOCKING PROCEDURE, THE SHIFT TECHNICAL ADVISOR (STA) NEEDED TO VERIFY THAT LOGIC FUSES IN THE ELECTRICAL PANEL WERE PROPERLY LABELED. UNAWARE OF THE LOOSE CONNECTION, HE GENTLY MOVED THE WIRE IN AN ATTEMPT TO READ ITS LABEL. A LOOSE CONNECTION AT THE FUSE TERMINAL CAUSED THE "A" RHR LOGIC TO DEENERGIZE WHEN THE WIRE WAS SHIFTED. DE-ENERGIZING THE RHR RELAY RESULTED IN THE ISOLATION OF SHUTDOWN COOLING VIA THE PRIMARY

CONTAINMENT ISOLATION SYSTEM (PCIS). AT 0314, THE LEAD AT THE FUSE WAS TIGHTENED, THE ISOLATION RESET, AND SHUTDOWN COOLING RESTORED. NO ADDITIONAL CORRECTIVE ACTIONS ARE PLANNED. THERE WAS NO NOTICEABLE INCREASE IN REACTOR COOLANT TEMPERATURE. THERE WERE NO ADVERSE CONSEQUENCES AS A RESULT OF THIS EVENT. THIS EVENT IS REPORTABLE DUE TO THE ACTUATION OF THE PCIS, AN ENGINEERED SAFETY FEATURE.

[233] PEACH BOTTOM 2 DOCKET 50-277 LER 87-020
INOPERABILITY OF THE HPCI SYSTEM DUE TO THE OPENING OF A RESISTOR IN THE VOLTAGE DROPPING ASSEMBLY.
EVENT DATE: 090487 REPORT DATE: 100587 NSSS: GE TYPE: BWR
VENDOR: WOODWARD GOVERNOR COMPANY

(NSIC 206715) ON SEPTEMBER 4, 1987 AT 1430 HOURS, A SURVEILLANCE TEST (ST 2.15.23, "HPCI TURBINE GOVERNOR (EGM, RGSC) CALIBRATION") OF THE HPCI TURBINE GOVERNOR CONTROL SYSTEM DETERMINED THAT A 200 OHM, 70 WATT RESISTOR LOCATED IN THE VOLTAGE DROPPING RESISTOR ASSEMBLY OPENED THUS RENDERING THE HPCI SYSTEM INOPERABLE. THE VOLTAGE DROPPING RESISTOR ASSEMBLY IS SUPPLIED BY THE WOODWARD GOVERNOR COMPANY. THE PURPOSE OF THE VOLTAGE DROPPING RESISTOR ASSEMBLY IS TO REDUCE A 125 VDC INPUT VOLTAGE TO A 48 VDC POWER SUPPLY INPUT TO THE EGM CONTROL BOX WHICH ULTIMATELY CONTROLS THE POSITION OF THE CONTROL VALVE OF THE HPCI TURBINE. THE EXACT CAUSE FOR THE OPENING OF THE RESISTOR IS UNKNOWN. THERE WERE NO ADVERSE CONSEQUENCES AS A RESULT OF THIS EVENT WHICH WOULD HAVE AFFECTED PLANT SAFETY SINCE OTHER SYSTEMS ARE AVAILABLE TO MAINTAIN CORE COOLING CAPABILITY. AS CORRECTIVE ACTION, THE RESISTOR WAS REPLACED ON SEPTEMBER 14, 1987. AS AN ACTION TO PREVENT RECURRENCE, AN ENGINEERING EVALUATION WILL BE PERFORMED TO INVESTIGATE THIS DEFICIENCY AND EVALUATE POSSIBLE ALTERNATIVES FOR PROVIDING A MORE RELIABLE VOLTAGE SUPPLY TO THE EGM CONTROL BOX.

[234] PEACH BOTTOM 2 DOCKET 50-277 LER 87-017
PARTIAL VENTILATION ISOLATION DURING THE TEMPORARY CLEARANCE OF A SAFETY BLOCK.
EVENT DATE: 091687 REPORT DATE: 101687 NSSS: GE TYPE: BWR

(NSIC 206784) ON SEPTEMBER 16, 1987 AT 0600 HOURS, THE TEMPORARY CLEARING OF BLOCKED SYSTEM LOGIC INITIATED AN AUTO-START OF THE STANDBY GAS TREATMENT SYSTEM (SBGTS) AND AN OUTBOARD ISOLATION OF THE UNIT 2 REACTOR BUILDING VENTILATION SYSTEM (RBVS). THE ISOLATION WAS RESET, AND THE SBGTS FAN WAS TRIPPED PROMPTLY AFTER THE EVENT TO RE-ESTABLISH THE NORMAL VENTILATION CONFIGURATION. THE UNEXPECTED ACTUATION OF AN ENGINEERED SAFETY FEATURE (ESF) MAKES THIS EVENT REPORTABLE. THE CAUSE OF THIS EVENT IS THE INADEQUATE REVIEW OF BLOCKING PERMITS AND HOW THEIR APPLICATION OR REMOVAL AFFECTS ESF LOGIC SYSTEMS. THIS PERMIT WILL BE REVISED BY OCTOBER 30, 1987 TO INCLUDE INSTRUCTIONS TO RESET THE ISOLATION LOGIC OR NOTIFY THE OPERATOR THAT SBGTS WILL AUTO-START AND AN RBVS ISOLATION WILL RESULT WHEN THE BLOCK IS REMOVED. TO PREVENT SIMILAR EVENTS IN THE FUTURE, THE PLANT STAFF IS CURRENTLY DEVELOPING A SYSTEM WHICH REQUIRES OPERATIONS STAFF APPROVAL OF BLOCKING PERMITS WHICH INVOLVE ELECTRICAL BLOCKING OR CLEARANCE OF ELECTRICAL BLOCKING OF ESFS. AS A RESULT OF THIS EVENT, NO PERSONNEL OR EQUIPMENT WERE ENDANGERED, AND NO EQUIPMENT OR SYSTEMS WERE RENDERED INOPERABLE.

[235] PEACH BOTTOM 2 DOCKET 50-277 LER 87-021
PRIMARY CONTAINMENT ISOLATION CAUSED BY UNKNOWN POWER INTERRUPTION.
EVENT DATE: 091687 REPORT DATE: 101687 NSSS: GE TYPE: BWR

(NSIC 206716) ON SEPTEMBER 16, 1987 AT 0815 HOURS WITH THE UNIT IN COLD SHUTDOWN, A POWER INTERRUPTION TO A DISTRIBUTION PANEL CAUSED A PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) ACTUATION. AMONG THE ACTIONS RESULTING FROM THE PCIS ACTUATION WERE ISOLATION OF SHUTDOWN COOLING AND ISOLATION OF THE REACTOR WATER CLEANUP (RWCU) SYSTEM. IN ADDITION TO THE PCIS ACTUATION WERE A LOSS OF

INDICATION FOR STANDBY LIQUID CONTROL (SBLC) LEVEL AND A LOSS OF INDICATION FOR HALF OF THE REACTOR BUILDING VENTILATION DAMPERS. THE CAUSE OF THE POWER INTERRUPTION IS NOT CERTAIN, BUT IS BELIEVED TO BE AN INADVERTENT ACTUATION OF A DISCONNECT SWITCH. THE DISTRIBUTION PANEL WAS BEING POWERED FROM AN ALTERNATE SOURCE, AS REQUIRED BY AN ONGOING MODIFICATION. THE TEMPORARY FEED BETWEEN THE PANEL AND THE ALTERNATE SOURCE INCLUDED THREE DISCONNECT SWITCHES LOCATED IN MODERATELY TRAVELED AREAS OF THE PLANT. ONE OF THE DISCONNECT SWITCHES WAS INCORRECTLY LABELED. AT 0825, ISOLATIONS WERE RESET AND INDICATIONS RESTORED. THERE WAS NO NOTICEABLE INCREASE IN REACTOR COOLANT TEMPERATURE OR ADVERSE CONSEQUENCES. THE SHUTDOWN COOLING PATH WAS ISOLATED FOR APPROXIMATELY 10 MINUTES. THE INCORRECT LABEL WAS REMOVED. NO FURTHER CORRECTIVE ACTIONS ARE PLANNED. THIS EVENT IS REPORTABLE DUE TO ACTUATION OF THE PCIS, AND ENGINEERED SAFETY FEATURE.

[236] PEACH BOTTOM 2 DOCKET 50-277 LER 87-022
LOSS OF HIGH PRESSURE COOLANT INJECTION DUE TO UNKNOWN CAUSE.
EVENT DATE: 092987 REPORT DATE: 103087 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: PEACH BOTTOM 3 (BWR)
VENDOR: RUNDEL ELECTRIC

(NSIC 206890) ON SEPTEMBER 29, 1987 AT 0020 HOURS WITH THE UNIT IN COLD SHUTDOWN, THE HIGH PRESSURE COOLANT INJECTION SYSTEM (HPCI) WAS DISABLED DUE TO A BLOWN LOGIC FUSE. THE 2D BATTERY CHARGER WAS LATER DECLARED INOPERABLE DUE TO A FAILURE OF THE UNDERVOLTAGE RELAY. THE CAUSE OF THESE ACTIONS IS NOT CERTAIN. IT APPEARS THAT AN UNDERRATED SOCKET FOR THE UNDERVOLTAGE RELAY ON THE 2D BATTERY CHARGER DISABLED THE BATTERY CHARGER. THE FAULT ON THE 2D CHANNEL APPEARS TO HAVE BEEN TRANSMITTED TO THE 2B CHANNEL THROUGH DC BUS INTERCONNECTIONS, THEREBY AFFECTING THE HPCI FUSE. WEEKLY SURVEILLANCE TESTING OF A DIESEL GENERATOR WAS IN PROGRESS AT THE TIME OF THE EVENT. IT APPEARS THAT THIS TESTING WAS COINCIDENTAL TO THE EVENT. THERE WERE NO ADVERSE CONSEQUENCES AS A RESULT OF THIS EVENT. IMMEDIATE CORRECTIVE ACTIONS WERE TO REPLACE THE FUSE WITHIN 30 MINUTES, AND TO MONITOR BATTERY TERMINAL VOLTAGE EVERY FOUR HOURS UNTIL THE UTILITY BATTERY CHARGER WAS PLACED IN SERVICE. REPLACEMENT SOCKETS FOR THE UNDERVOLTAGE RELAYS ARE BEING PROCURED. AN INVESTIGATION OF THE EVENT IS ONGOING. THE RESULTS OF THE INVESTIGATION, INCLUDING ANY ADDITIONAL CORRECTIVE ACTIONS, WILL BE REPORTED AS A SUPPLEMENT TO THIS REPORT. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR 50.73(A)(2)(V).

[237] PEACH BOTTOM 2 DOCKET 50-277 LER 87-023
BLOWN LOGIC FUSE OF UNKNOWN CAUSE RESULTING IN HIGH PRESSURE COOLANT INJECTION INOPERABILITY.
EVENT DATE: 101487 REPORT DATE: 112487 NSSS: GE TYPE: BWR
VENDOR: BUSSMANN MFG (DIV OF MCGRAW-EDISON)

(NSIC 207181) ON OCTOBER 14, 1987 AT 1000 HOURS WITH THE UNIT IN COLD SHUTDOWN, THE HIGH PRESSURE COOLANT INJECTION SYSTEM (HPCI) WAS DISABLED DUE TO A BLOWN LOGIC FUSE. AN EXHAUSTIVE INVESTIGATION OF THE EVENT WAS UNABLE TO IDENTIFY THE EXACT CAUSE. THERE WERE NO ADVERSE CONSEQUENCES AS A RESULT OF THIS EVENT. AT THE TIME OF THE EVENT, THE HPCI SYSTEM WAS NOT REQUIRED TO BE OPERABLE BECAUSE REACTOR PRESSURE WAS BELOW 105 PSIG. IMMEDIATE CORRECTIVE ACTION WAS TO REPLACE THE FUSE WITHIN APPROXIMATELY FIVE MINUTES. AN INVESTIGATION OF THE EVENT WAS CONDUCTED WHICH INCLUDED INTERVIEWING PERSONNEL POTENTIALLY INVOLVED WITH THE EVENT AND RESEARCHING THE WORK HISTORIES OF THE ELECTRICAL PANELS IN THE CONTROL ROOM AND CABLE SPREADING ROOM. THE EVENT IS BELIEVED TO BE AN ISOLATED OCCURRENCE. NO ADDITIONAL CORRECTIVE ACTIONS ARE PLANNED. THIS EVENT IS REPORTABLE PURSUANT TO 10 CFR 50.73(A)(2)(V).

[238] PEACH BOTTOM 3 DOCKET 50-278 LER 87-007
INOPERABILITY OF THE HPCI SYSTEM DUE TO THE FAILURE OF A RELAY COIL.
EVENT DATE: 082987 REPORT DATE: 092887 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 206671) ON AUGUST 29, 1987 AT APPROXIMATELY 0645 HOURS, THE HIGH PRESSURE COOLANT INJECTION SYSTEM (HPCI) WAS RENDERED INOPERABLE DUE TO A LOSS OF INSTRUMENT POWER TO THE HPCI PRESSURE TRANSMITTERS AND FLOW CONTROL CIRCUIT. THE LOSS OF POWER WAS THE RESULT OF THE FAILURE OF RELAY COIL 23A-K50, LOCATED IN CONTROL ROOM PANEL 30C04B, WHICH MONITORS THE AC OUTPUT OF THE HPCI SYSTEM DC INVERTER. THE EVENT WAS DETECTED BY THE ANNUNCIATION OF THE "HPCI INVERTER POWER FAILURE" ALARM IN THE MAIN CONTROL ROOM. THERE WERE NO ADVERSE CONSEQUENCES OF THIS EVENT WHICH WOULD HAVE AFFECTED PLANT SAFETY. AT THE TIME OF THE EVENT, THE HPCI SYSTEM WAS NOT IN USE NOR WAS IT REQUIRED TO BE OPERABLE BECAUSE THE PRIMARY SYSTEM WAS DEPRESSURIZED. THE CAUSE OF THE RELAY COIL FAILURE WAS ATTRIBUTED TO DETERIORATION DUE TO NORMAL WEAR. THE RELAY WAS REPLACED ON SEPTEMBER 18, 1987. THE SYSTEM REMAINED OUT OF SERVICE DUE TO OTHER WORK ACTIVITIES BEING PERFORMED ON THIS SYSTEM. AS AN ACTION TO PREVENT RECURRENCE, THIS RELAY COIL WILL BE ADDED TO THE PREVENTIVE MAINTENANCE PROGRAM AND REPLACED ON A SIX YEAR INTERVAL.

[239] PEACH BOTTOM 3 DOCKET 50-278 LER 87-008
CONTAINMENT ISOLATIONS DUE TO LOAD CENTER BREAKER TRIP AS A RESULT OF PERSONNEL ERROR DURING TROUBLE-SHOOTING.
EVENT DATE: 100587 REPORT DATE: 110487 NSSS: GE TYPE: BWR

(NSIC 206912) ON OCTOBER 5, 1987 AT 2026 HOURS, SEVERAL GROUP II PRIMARY CONTAINMENT ISOLATION VALVES AUTOMATICALLY CLOSED WHILE THE REACTOR WAS IN THE COLD CONDITION. THE SHUTDOWN COOLING SUCTION VALVES CLOSED CAUSING THE "C" RESIDUAL HEAT REMOVAL (RHR) PUMP TO TRIP. NO OTHER RHR PUMP WAS IN SERVICE. THE FOLLOWING OTHER GROUP II VALVES CLOSED: LOW PRESSURE COOLANT INJECTION "A" LOOP INJECTION VALVE, TWO DRYWELL DRAIN SUMP VALVES AND THE OXYGEN ANALYZER SUPPLY VALVES. THERE WERE NO ADVERSE SAFETY CONSEQUENCES. SHUTDOWN COOLING WAS ISOLATED FOR ONLY ONE MINUTE. EVEN IF THIS EVENT HAD OCCURRED WITH A HIGHER DECAY HEAT LOAD OR WHILE AT POWER THERE WOULD HAVE BEEN NO ADVERSE SAFETY CONSEQUENCES. THIS EVENT WAS INITIATED WHEN A TEST ENGINEER OPENED AN EMERGENCY BREAKER COMPARTMENT DOOR. THE ENGINEER WAS NOT AWARE THAT OPENING THE DOOR WOULD INTERRUPT BREAKER CONTROL CIRCUITRY, WHICH INITIATED AN EMERGENCY LOAD CENTER BREAKER TRIP. THE RESULTING DE-ENERGIZATION OF RELAYS CAUSED THE ISOLATIONS. WARNING SIGNS HAVE BEEN ORDERED AND WILL BE INSTALLED ON THIS AND OTHER DOORS TO PREVENT RECURRENCE OF A SIMILAR EVENT. THIS EVENT HAS BEEN DISCUSSED WITH THE TEST ENGINEER INVOLVED AND THE NEED TO EXERCISE MORE CAUTION WAS STRESSED. THIS EVENT ALSO WAS REVIEWED WITH OTHER TEST ENGINEERS.

[240] PEACH BOTTOM 3 DOCKET 50-278 LER 87-009
SHUTDOWN COOLING ISOLATION DURING TESTING DUE TO AN INCORRECT PROCEDURE.
EVENT DATE: 101287 REPORT DATE: 111287 NSSS: GE TYPE: BWR

(NSIC 206913) AT 0000 HOURS ON OCTOBER 12, 1987, DURING THE TESTING OF PRIMARY CONTAINMENT ISOLATION SYSTEM LOGIC SYSTEM, THE UNIT 3 RESIDUAL HEAT REMOVAL (RHR) SHUTDOWN COOLING AND LOW PRESSURE COOLANT INJECTION ISOLATION VALVE (MO-3-10-25B) CLOSED UNEXPECTEDLY. SHUTDOWN COOLING WAS RETURNED TO SERVICE WITHIN THREE MINUTES OF THE ACTUATION. THE TEST WAS PERFORMED TO SATISFY PREREQUISITES FOR FUEL OFFLOAD. THE TEST PROCEDURE HAD BEEN TEMPORARILY CHANGED WITH THE INTENT TO ALLOW TESTING OF SECONDARY CONTAINMENT FUNCTIONS WITHOUT IMPACTING THE OPERATION OF THE SHUTDOWN COOLING MODE OF RHR OR REACTOR WATER CLEAN-UP SYSTEMS. THE TEMPORARY PROCEDURE CHANGE (TPC) ADDRESSED THE SHUTDOWN COOLING SUCTION VALVES, BUT OMITTED THE INJECTION VALVE WHICH CLOSED, AS DESIGNED, DURING THE TEST. THE CONSEQUENCES OF THE EVENT ARE MINIMAL, DUE TO THE SHORT OUT-OF-SERVICE TIME AND LOW RESIDUAL HEAT LOAD. THE TPC WAS REVISED, AND THE TEST WAS COMPLETED WITHOUT

INCIDENT. TO PREVENT RECURRENCE, THIS PROCEDURE FOR BOTH UNITS (ST 1.3-2, 1.3-3) WILL BE REVISED TO INCLUDE COMPLETE LISTS OF EXPECTED ACTUATIONS. THE UNEXPECTED ACTUATION OF AN ENGINEERED SAFETY FEATURE MAKES THIS EVENT REPORTABLE.

[241] PERRY 1 DOCKET 50-440 LER 87-051 REV 01
UPDATE ON FAILED LOCAL LEAK RATE TESTS RESULT IN EXCEEDING ALLOWABLE PRIMARY CONTAINMENT LEAKAGE RATES FOR MAIN STEAM LINES A, B, AND D.
EVENT DATE: 070587 REPORT DATE: 092987 NSSS: GE TYPE: BWR
VENDOR: ATWOOD & MORRILL CO., INC.
ROCKWELL MANUFACTURING COMPANY

(NSIC 206503) ON JULY 5, 1987 AT APPROXIMATELY 0300, IT WAS IDENTIFIED THAT THE PRIMARY CONTAINMENT LEAKAGE RATE THROUGH THE ISOLATION VALVES FOR MAIN STEAM LINES (MSL) A, B, AND D AS DEFINED BY TECHNICAL SPECIFICATION 3.6.1.2 HAD BEEN EXCEEDED. PRIMARY CONTAINMENT LEAKAGE PATHS IN EXCESS OF THE TECHNICAL SPECIFICATION LIMIT COULD EXIST THROUGH THE INBOARD AND OUTBOARD MAIN STEAM ISOLATION VALVES (MSIV) FOR MSLS A AND B, AND THROUGH THE INBOARD MSIV AND MSIV LEAKAGE CONTROL SYSTEM (LCS) STEAM TUNNEL ISOLATION VALVE FOR MSL D. THE CAUSE OF THE MSIV LEAKAGE IS WORN SEATS DUE TO COMPONENT WEAR BASED ON SERVICE SEEN DURING THE POWER ASCENSION PROGRAM. THE CAUSE OF THE MSIV LCS VALVES LEAKING HAS BEEN ATTRIBUTED TO THE DIFFICULTY IN OBTAINING ADEQUATE MATING AND SEALING OF THE BONNET PRESSURE SEAL RING WHEN ASSEMBLING THE VALVE BODY TO BONNET. AS A RESULT OF THIS EVENT, SEAT LAPPING WAS PERFORMED ON THE AFFECTED MSIVS TO REDUCE THEIR LEAKAGE. THE MSIV LCS VALVES HAVE BEEN REWORKED INCLUDING SEAT LAPPING AND REPLACEMENT OF THE BONNET PRESSURE SEAL RINGS. LLRTS HAVE BEEN SATISFACTORLY COMPLETED FOR ALL MAIN STEAM LINES FOLLOWING THE REPAIRS TO THE MSIVS AND MSIV LCS VALVES. APPROXIMATE FINAL LEAKAGE RATES FOR THE MSLS ARE AS FOLLOWS; 4.6 SCFH FOR MSL A, 2.2 SCFH FOR MSL B, 12.9 SCFH FOR MSL C, AND 5.9 SCFH FOR MSL D.

[242] PERRY 1 DOCKET 50-440 LER 87-063
REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION UPON RETURNING LEAK DETECTION BYPASS SWITCH TO NORMAL DUE TO INDETERMINATE CAUSE.
EVENT DATE: 090687 REPORT DATE: 100687 NSSS: GE TYPE: BWR

(NSIC 206727) ON SEPTEMBER 6, 1987 AT 0430, AN UNEXPECTED DIVISION II REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM ISOLATION OCCURRED DURING PERFORMANCE OF LEAK DETECTION (LD) SYSTEM TROUBLESHOOTING MAINTENANCE. THE RCIC SYSTEM ISOLATED WHEN CONTROL ROOM OPERATORS RESTORED THE RCIC LD ISOLATION BYPASS SWITCH TO THE NORMAL POSITION. AT 0530 THE RCIC ISOLATION SIGNAL WAS RESET AND THE RCIC SYSTEM WAS RESTORED TO STANDBY READINESS. THE CAUSE OF THE RCIC SYSTEM ISOLATION HAS NOT BEEN DETERMINED. EXTENSIVE TROUBLESHOOTING, PERSONNEL INTERVIEWS, AND PROCEDURE REVIEWS HAVE IDENTIFIED NO PROBABLE CAUSE. HOWEVER, IN AN ATTEMPT TO PREVENT RECURRENCE AND IMPROVE RCIC AVAILABILITY, AN ENGINEERING DESIGN CHANGE HAS BEEN INITIATED TO PROVIDE INDICATION OF THE RCIC TRIP RELAY STATUS WHEN TAKING THE RCIC LD ISOLATION CHANNELS FROM BYPASS TO NORMAL. ADDITIONALLY, ALL APPLICABLE PROCEDURES WILL BE REVISED ACCORDINGLY AND ALL APPLICABLE PERSONNEL WILL BE TRAINED.

[243] PERRY 1 DOCKET 50-440 LER 87-064
FEEDWATER PUMP OVERSPEED CAUSED BY A DEFICIENT PERIODIC TEST INSTRUCTION RESULTED IN A REACTOR SCRAM, HIGH PRESSURE CORE SPRAY INJECTION AND AN UNUSUAL EVENT.
EVENT DATE: 090987 REPORT DATE: 100987 NSSS: GE TYPE: BWR
VENDOR: AMERACE CORP.
GENERAL ELECTRIC CO.
NAMCO CONTROLS

(NSIC 206728) ON SEPTEMBER 9, 1987 AT 2033 A REACTOR SCRAM OCCURRED DUE TO HIGH REACTOR WATER LEVEL (LEVEL 8, + 219.5 INCHES ABOVE TOP OF ACTIVE FUEL) CAUSED BY

A SUDDEN CHANGE IN THE FEEDWATER FLOW CONTROL SIGNAL AND THE TURBINE DRIVEN FEEDWATER PUMPS (TDFP) RAMPING TO FULL SPEED. THE HIGH REACTOR WATER LEVEL ALSO CAUSED THE MAIN TURBINE AND THE FEEDWATER PUMPS TO TRIP. REACTOR WATER LEVEL THEN DROPPED TO LEVEL 2 (+ 129.8 INCHES ABOVE TOP OF ACTIVE FUEL) CAUSING THE HIGH PRESSURE CORE SPRAY (HPCS) AND REACTOR CORE ISOLATION COOLING SYSTEMS TO START AND INJECT TO THE REACTOR VESSEL. AN UNUSUAL EVENT WAS DECLARED DUE TO THE AUTOMATIC ACTUATION OF HPCS. THE CAUSE OF THE EVENT WAS A PROCEDURAL DEFICIENCY. A PERIODIC TEST INSTRUCTION (PTI), WHICH STROKES THE LOW PRESSURE STOP VALVES FOR THE TDFP, DID NOT ISOLATE A TURBINE TRIP SIGNAL FROM THE FEEDWATER CONTROL SYSTEM. ADDITIONALLY, THE LIMIT SWITCH USED FOR THE TRIP SIGNAL PROVIDED A FALSE TURBINE TRIP SIGNAL. THE PTI HAS BEEN REVISED TO ENSURE A TURBINE TRIP SIGNAL WILL NOT CAUSE A FLOW CONTROL SIGNAL CHANGE DURING THE TEST. A DESIGN CHANGE IS BEING DEVELOPED TO PREVENT A FALSE TURBINE TRIP SIGNAL. ADDITIONALLY, ALL PTI'S WHICH HAVE NOT BEEN PREVIOUSLY PERFORMED WILL BE REVIEWED PRIOR TO PERFORMANCE.

[244] PERRY 1 DOCKET 50-440 LER 87-065
BAD CONNECTION IN ISOLATION LOGIC RESULTS IN UNEXPECTED ISOLATION OF A MAIN STEAM DRAIN LINE INBOARD ISOLATION VALVE DURING SURVEILLANCE TESTING.
EVENT DATE: 091487 REPORT DATE: 101387 NSSS: GE TYPE: BWR
VENDOR: AMERACE CORP.

(NSIC 206729) ON SEPTEMBER 14, 1987 AT APPROXIMATELY 1355, A MAIN STEAM DRAIN LINE INBOARD ISOLATION VALVE CLOSED UNEXPECTEDLY DUE TO A NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NSSSS) ISOLATION SIGNAL DURING THE PERFORMANCE OF A LEAK DETECTION (LD) SYSTEM SURVEILLANCE INSTRUCTION (SVI). PLANT OPERATORS VERIFIED THAT NO VALID NSSSS ISOLATION SIGNAL EXISTED AND OPENED THE ISOLATION VALVE AT 1510. THE CAUSE FOR THE CLOSURE OF THE MAIN STEAM DRAIN LINE INBOARD ISOLATION VALVE IS SUSPECTED TO BE DUE TO A BAD CONNECTION IN THE CHANNEL C NSSSS ISOLATION LOGIC. THEREFORE, WHEN THE B NSSSS ISOLATION LOGIC WAS TESTED AS REQUIRED BY THE SVI, THE ISOLATION RELAY WAS DE-ENERGIZED CAUSING CLOSURE OF THE VALVE. SUBSEQUENT TROUBLESHOOTING IDENTIFIED A FAULTY CONTACT IN A RELAY IN THE C NSSSS LOGIC, HOWEVER FAILURE OF THIS CONTACT WOULD NOT HAVE RESULTED IN THE ISOLATION. THE AFFECTED RELAY HAS BEEN REPLACED AND TESTED SATISFACTORILY. IN ADDITION, A PHYSICAL VERIFICATION OF THE WIRING IN THE AFFECTED PORTION OF THE NSSSS C LOGIC WAS PERFORMED TO ENSURE NO LOOSE OR DAMAGED CONNECTIONS EXIST. NO ADDITIONAL ISOLATIONS IN THE NSSSS SYSTEM HAVE BEEN OBSERVED SINCE THIS EVENT. HOWEVER, THE ROUTINE SYSTEM PERFORMANCE MONITORING WILL CONTINUE AS REQUIRED.

[245] PERRY 1 DOCKET 50-440 LER 87-066
INADEQUATE SURVEILLANCE INSTRUCTIONS RESULT IN TECH SPEC VIOLATION FOR DIESEL GENERATOR OPERABILITY.
EVENT DATE: 092687 REPORT DATE: 102387 NSSS: GE TYPE: BWR

(NSIC 206838) ON SEPTEMBER 26, 1987 AT 0800, IT WAS DISCOVERED THAT THE SURVEILLANCE INSTRUCTIONS (SVI) FOR THE DIVISION 1 AND DIVISION 2 STANDBY DIESEL GENERATORS (DG) DID NOT ADEQUATELY TEST ALL OF THE DG LOCKOUT FEATURES AS REQUIRED BY TECHNICAL SPECIFICATION 4.8.1.1.2.E.14.A. AT THE TIME OF THE DISCOVERY, THE TECHNICAL SPECIFICATION ACTION STATEMENT FOR OPERATIONAL CONDITION 4 (COLD SHUTDOWN) WAS BEING COMPLIED WITH. THE CAUSE OF THIS EVENT WAS INADEQUATE PROCEDURES. SURVEILLANCE INSTRUCTIONS TESTED THE PULL-TO-LOCK SWITCH BY PLACING THE SWITCH IN PULL-TO-LOC, THEN ATTEMPTING TO TURN THE SWITCH TO THE START POSITION. AS A RESULT, ONLY THE LOCKING FEATURE OF THE SWITCH WAS TESTED. IN ADDITION, TESTING OF THE BARRING DEVICE ENGAGED WAS PERFORMED WHILE THE INOP/NORMAL SWITCH WAS IN THE INOP POSITION. SINCE THE INOP/NORMAL SWITCH IN THE INOP POSITION WILL PREVENT THE DG FROM STARTING, THE BARRING DEVICE FEATURE WAS NOT INDEPENDENTLY TESTED. AS A RESULT OF THIS EVENT, SVI-R43-T1327 & T1328 HAVE BEEN REVISED AND THE LOCKOUT FEATURES FOR THE DIVISION 1 AND 2 DGS WERE SUCCESSFULLY TESTED ON SEPTEMBER 28 AND 30 RESPECTIVELY. IN ADDITION, AN ONGOING

CURRENCY REVIEW OF SVIS IS CONTINUING TO ENSURE INCORPORATION OF ALL TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS.

[246] PERRY 1 DOCKET 50-440 LER 87-067
IMPROPER MAINTENANCE RESULTS IN THE B MAIN STEAM LINE PENETRATION EXCEEDING TECHNICAL SPECIFICATION LEAKAGE LIMIT.
EVENT DATE: 092687 REPORT DATE: 102387 NSSS: GE TYPE: BWR
VENDOR: ATWOOD & MORRILL CO., INC.
ROCKWELL MANUFACTURING COMPANY

(NSIC 206952) ON SEPTEMBER 26, 1987 IT WAS IDENTIFIED THAT THE PRIMARY CONTAINMENT LEAKAGE RATE THROUGH THE ISOLATION VALVES FOR MAIN STEAM LINE (MSL) B AS DEFINED BY TECH SPEC 3.6.1.2 HAD BEEN EXCEEDED, HOWEVER, THE LEAKAGE PATH HAD NOT BEEN DETERMINED. THE INBOARD MAIN STEAM ISOLATION VALVE (MSIV) STEM AND ACTUATOR HAD PREVIOUSLY SEPARATED ON SEPTEMBER 22 CAUSING IT TO BE INOPERABLE. ON SEPTEMBER 29 THE OUTBOARD MSIV WAS VERIFIED TO BE THE LEAKAGE PATH HAVING EXCESSIVE SEAT LEAKAGE. THE CAUSE OF THE STEM AND ACTUATOR SEPARATION ON THE INBOARD MSIV WAS THE STEM PLATE SET SCREW NOT BEING PROPERLY INSTALLED DURING PREVIOUS MAINTENANCE ALLOWING THE STEM TO ROTATE AND BECOME DETACHED FROM THE ACTUATOR STEM PLATE. LEAKAGE THROUGH THE OUTBOARD MSIV WAS DUE TO DEFORMATION IN THE LOWER PART OF THE VALVE SEATING SURFACE WHICH WAS NOT REMOVED BY LAPPING DURING PREVIOUS MAINTENANCE. AS THE VALVE WAS STROKED THE SEAT CONTACT POINT MOVED FURTHER DOWN ON THE SEAT SURFACE INTO THE DEFORMATION ALLOWING EXCESSIVE LEAKAGE. BOTH MSIVS WERE DISASSEMBLED AND INSPECTED. THERE WAS NO POPPET OR SEAT DAMAGE TO THE INBOARD MSIV. THE OUTBOARD MSIV SEAT WAS RELAPPED TO REMOVE ALL INDICATIONS OF THE DEFORMATION. A REVIEW OF MSIV WORK PACKAGES WILL BE CONDUCTED TO IDENTIFY IMPROVEMENTS FOR FUTURE MSIV MAINTENANCE AND A GENERIC MAINTENANCE INSTRUCTION WILL BE DEVELOPED.

[247] PERRY 1 DOCKET 50-440 LER 87-068
RESIDUAL HEAT REMOVAL SHUTDOWN COOLING ISOLATION DURING SURVEILLANCE TESTING DUE TO A DRIFTING PRESSURE REGULATOR.
EVENT DATE: 092987 REPORT DATE: 102987 NSSS: GE TYPE: BWR

(NSIC 206839) ON SEPTEMBER 29, 1987 AT 1507, A RESIDUAL HEAT REMOVAL (RHR) SHUTDOWN INBOARD ISOLATION OCCURRED DURING PERFORMANCE OF A SURVEILLANCE INSTRUCTION (SVI). THE ISOLATION OCCURRED WHEN THE PRESSURE REGULATOR WAS ADJUSTED FOR AN OUTPUT OF APPROXIMATELY 5 PSIG TO FACILITATE PRESSURE TRANSMITTER VENTING AND THE REGULATOR DRIFTED ABOVE THE 135 PSIG SHUTDOWN COOLING ISOLATION SETPOINT. OPERATORS RESTORED THE RHR SHUTDOWN COOLING AT 1537. THE CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO SEVERAL CONTRIBUTING FACTORS. THE SVI DID NOT REQUIRE A BYPASS JUMPER DURING THE VENTING PORTION, NOR DID IT PROVIDE CAUTIONS OR NOTES CONCERNING THE POTENTIAL FOR CAUSING A SHUTDOWN COOLING ISOLATION DURING THE VENTING. THE PERSONNEL INVOLVED DID NOT PAY SUFFICIENT ATTENTION TO DETAIL TO IDENTIFY ANY DRIFTING IN THE PRESSURE REGULATOR DURING THE PERFORMANCE OF THE TEST. TO PREVENT RECURRENCE, THE APPLICABLE SVIS WILL BE REVISED TO REQUIRE INSTALLATION OF A BYPASS JUMPER TO PREVENT ANY ISOLATIONS PRIOR TO THE VENTING ACTIVITY. A REVIEW OF OTHER ACTIVITIES UTILIZING THIS PRESSURE REGULATING EQUIPMENT WAS PERFORMED WHICH IDENTIFIED NO ADDITIONAL POTENTIAL PROBLEMS.

[248] PERRY 1 DOCKET 50-440 LER 87-069
UNEXPECTED ANNULUS EXHAUST GAS TREATMENT SYSTEM TRAIN B AUTO START FOLLOWING TRANSFER OF OPERATING TRAINS, DUE TO INDETERMINATE CAUSE.
EVENT DATE: 093087 REPORT DATE: 102987 NSSS: GE TYPE: BWR

(NSIC 206953) ON SEPTEMBER 30, 1987 AT 0253, AFTER SWITCHING THE INSERVICE ANNULUS EXHAUST GAS TREATMENT SYSTEM (AEGTS) TRAIN "B" TO THE STANDBY TRAIN "A", THE B FAN AUTOMATICALLY RESTARTED DUE TO A HIGH DIFFERENTIAL PRESSURE (D/P)

SIGNAL ACROSS THE A FAN. THE A FAN WAS IMMEDIATELY SECURED, AND AN OPERATOR WAS SENT TO INVESTIGATE WHY THE RECIRCULATION DAMPER FOR THE A TRAIN DID NOT MODULATE OPEN PRIOR TO SHUTDOWN OF THE B TRAIN. NO APPARENT CAUSE WAS IDENTIFIED, AND THE A FAN WAS RESTARTED AT 0313. THE ROOT CAUSE OF THIS EVENT IS INDETERMINATE. DURING TROUBLESHOOTING TESTS, FLUCTUATIONS OF ANNULUS PRESSURE WERE NOTED DURING THE FAN SWITCH DUE TO THE STARTUP/SHUTDOWN OF THE TWO TRAINS OF AEGTS. THIS EVENT MAY HAVE OCCURRED MERELY BECAUSE THE OPERATOR SHUTDOWN THE B TRAIN BEFORE THE A TRAIN WAS ABLE TO RESPOND TO ELIMINATE THE FLUCTUATIONS. ANOTHER POSSIBLE CAUSE IS THAT THE DAMPER, UPON FIRST DEMAND, DID NOT IMMEDIATELY RESPOND DUE TO A DIRTY RELAY CONTACT OR STICKING DAMPER, EITHER OF WHICH WOULD HAVE CORRECTED THEMSELVES UPON SUBSEQUENT DEMANDS. IN ORDER TO PREVENT RECURRENCE, A CHANGE HAS BEEN APPROVED TO THE SECTION OF THE AEGTS SYSTEM OPERATING INSTRUCTION WHICH CONTROLS THE SWITCHING OF THE AEGTS TRAINS.

[249] PILGRIM 1 DOCKET 50-293 LER 87-007
AUTOMATIC ACTUATION OF THE REACTOR PROTECTION SYSTEM DUE TO UTILITY PERSONNEL ERROR.
EVENT DATE: 101887 REPORT DATE: 111387 NSSS: GE TYPE: BWR

(NSIC 207189) ON OCTOBER 18, 1987, AT 0513 HOURS, AN AUTOMATIC ACTUATION OF THE REACTOR PROTECTION SYSTEM RPS OCCURRED DURING PERFORMANCE OF SURVEILLANCE PROCEDURE B.M.1-3.1, "AVERAGE POWER RANGE MONITOR APRM SETDOWN FUNCTIONAL" TESTS. THE PLANT WAS IN A COLD CONDITION WITH THE REACTOR MODE SWITCH IN REFUEL AND ALL FUEL ASSEMBLIES LOADED INTO REACTOR VESSEL. NO CONTROL RODS WERE WITHDRAWN AND THE REACTOR VESSEL HEAD WAS REMOVED. DURING PERFORMANCE OF THE SURVEILLANCE TEST ON APRM "A", A TECHNICIAN INADVERTENTLY MOVED THE WRONG APRM MODE SWITCH OUT OF THE OPERATE POSITION CAUSING A FULL RPS SCRAM TRIP SIGNAL. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO UTILITY INSTRUMENTATION AND CONTROL (I&C) TECHNICIAN ERROR. THE MODE SWITCHES FOR THE APRM'S "A" AND "D" WERE IMMEDIATELY RETURNED TO THE OPERATE POSITION AND FURTHER APRM TESTING WAS SUSPENDED. THE TECHNICIAN RESPONSIBLE FOR THIS EVENT WAS CAUTIONED REGARDING THE OPERATION OF THESE SWITCHES AND TWO PROCEDURES WERE REVISED TO REQUIRE TAGS TO BE PLACED ON THE SWITCHES NOT BEING TESTED. DURING THIS EVENT, THE RPS FUNCTIONED AS EXPECTED. ALTHOUGH THIS EVENT CHALLENGED THE RPS, IT DID NOT RESULT IN A CONDITION ADVERSE TO THE SAFE OPERATION OF THE PILGRIM NUCLEAR POWER STATION.

[250] POINT BEACH 2 DOCKET 50-301 LER 87-004
DEGRADATION OF STEAM GENERATOR TUBES DUE TO INTERNAL OBSTRUCTION.
EVENT DATE: 102787 REPORT DATE: 111087 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 206918) UNIT 2 WAS SHUT DOWN FOR REFUELING 13 ON OCTOBER 2, 1987. EDDY CURRENT EXAMINATION OF THE STEAM GENERATOR TUBES WAS CONDUCTED FROM OCTOBER 8 TO OCTOBER 22, 1987, USING A DIGITAL MULTI-FREQUENCY EDDY CURRENT SYSTEM. EDDY CURRENT INSPECTION OF THE "A" STEAM GENERATOR DETERMINED 10 TUBES DEGRADED EQUAL TO OR GREATER THAN 40%, 19 TUBES WITH AN UNDEFINED SIGNAL, AND 1 TUBE COULD NOT BE INSPECTED DUE TO INTERNAL OBSTRUCTION. ONE TUBE WAS PULLED FOR FURTHER ANALYSIS. 30 TUBES AND THE TUBESHEET HOLE WERE PLUGGED. EDDY CURRENT INSPECTION OF THE "B" STEAM GENERATOR DETERMINED 5 TUBES DEGRADED EQUAL TO OR GREATER THAN 40%, AND 9 TUBES WITH UNDEFINED SIGNALS. TWO TUBES SLEEVED DURING THE CURRENT SLEEVING PROGRAM WERE PLUGGED DUE TO UNACCEPTABLE UPPER JOINTS AND 1 TUBE WAS PULLED FOR FURTHER ANALYSIS. 16 TUBES AND THE TUBESHEET HOLE WERE PLUGGED. IN ADDITION, 87 TUBES WERE SLEEVED IN THE "B" STEAM GENERATOR COLD LEG. THE 800 PSID LEAK TEST REVEALED 1 EXPLOSIVELY PLUGGED TUBE LEAKING IN THE "B" COLD LEG, TWO EXPLOSIVELY PLUGGED TUBES AND FOUR SLEEVED TUBES LEAKING IN THE "A" HOT LEG. ONE TUBE WAS WELD REPAIRED AND THE OTHERS WERE LEFT AS FOUND BECAUSE OF THE MINIMAL LEAKAGE OBSERVED AND THE RADIOLOGICAL EXPOSURE INVOLVED IN THE REPAIR.

[251] PRAIRIE ISLAND 1 DOCKET 50-282 LER 86-006 REV 01
UPDATE ON SIMULTANEOUS INOPERABILITY OF ONE DIESEL GENERATOR AND ONE DIESEL COOLING WATER PUMP RESULTING FROM COMPONENT FAILURES DURING TESTING.
EVENT DATE: 090886 REPORT DATE: 111287 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: PRAIRIE ISLAND 2 (PWR)
VENDOR: FAIRBANKS MORSE

(NSIC 207158) ON SEPTEMBER 7, 1986, BOTH UNITS WERE AT STEADY-STATE POWER, UNIT 1 AT 100% AND UNIT 2 AT 88% POWER. AT 2355, DURING AN OPERABILITY TEST, D1 DIESEL GENERATOR FAILED TO START AFTER CRANKING FOR TEN SECONDS. D1 HAS DECLARED INOPERABLE. AT 0330, WHILE RUNNING NO. 22 DIESEL COOLING WATER PUMP TO PROVE OPERABILITY PER TECHNICAL SPECIFICATION 3.7.B.2, A SMALL OIL LINE BURST; THE PUMP WAS SHUT DOWN AND DECLARED INOPERABLE. WITH THESE TWO COMPONENTS INOPERABLE, A POWER DECREASE WAS BEGUN ON BOTH UNITS. REPAIR OF THE OIL LEAK ON NO. 22 DIESEL COOLING WATER PUMP WAS COMPLETED AND AT 0406 THE PUMP WAS RESTARTED AND PROVEN OPERABLE. THE POWER DECREASE WAS HALTED AND BOTH UNITS WERE RETURNED TO NORMAL OPERATION. CAUSE OF THE INOPERABILITY OF NO. 22 DIESEL COOLING WATER PUMP WAS THE RUPTURE OF BRASS TUBING, WHICH WAS REPLACED WITH STAINLESS STEEL. CAUSE OF THE INOPERABILITY OF D1 DIESEL GENERATOR WAS NOT IMMEDIATELY APPARENT BUT FURTHER INVESTIGATION AND TESTING REVEALED THAT LEAKAGE THROUGH THE FUEL HEADER PRESSURE RETURN ORIFICE CHECK VALVE ALLOWED THE FUEL OIL HEADER TO DRAIN DURING IDLE PERIODS. THE CHECK VALVE WAS REPLACE. THIS EVENT IS REPORTABLE UNDER 50.73(A)(2)(I)(B). THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED SINCE OFFSITE POWER SOURCES WERE INTACT THROUGHOUT THE EVENT.

[252] PRAIRIE ISLAND 1 DOCKET 50-282 LER 86-011 REV 01
UPDATE ON HOSE FAILURE DURING TESTING CAUSED INOPERABILITY OF SECOND DIESEL COOLING WATER PUMP WITH ONE PUMP ALREADY OUT OF SERVICE.
EVENT DATE: 122786 REPORT DATE: 112587 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: PRAIRIE ISLAND 2 (PWR)
VENDOR: AEROQUIP CORP.

(NSIC 207185) ON 12/27/86, BOTH UNITS WERE AT 100% POWER. NO. 22 DIESEL COOLING WATER PUMP (DCLP)(P) WAS OUT OF SERVICE FOR PLANNED MAINTENANCE. DURING THE DAILY OPERABILITY RUN OF NO. 12 DCLP (REQUIRED BY TECHNICAL SPECIFICATIONS WHEN ONE DIESEL COOLING WATER PUMP IS OUT OF SERVICE), A JACKET WATER HOSE RUPTURED. THE NO. 12 DCLP WAS IMMEDIATELY SHUTDOWN AND DECLARED INOPERABLE AT 0848. AT 0947, A LOAD DECREASE WAS BEGUN ON BOTH UNITS SINCE BOTH DIESEL COOLING WATER PUMPS WERE INOPERABLE. A NOTIFICATION OF UNUSUAL EVENT (NUE) WAS DECLARED. REPLACEMENT OF THE JACKET WATER HOSE ON NO. 12 DCLP WAS ACCOMPLISHED QUICKLY; AT 1008 THE PUMP WAS DECLARED OPERABLE AND THE LOAD DECREASE AS STOPPED. THE NUE WAS TERMINATED AND A LOAD INCREASE TO FULL POWER WAS BEGUN. CAUSE OF THE EVENT WAS FAILURE OF A FLEXIBLE HOSE MADE BY AEROQUIP CORP. PRELIMINARY INVESTIGATION INDICATE THAT EXPECTED SERVICE LIFE OF THIS HOSE IS ABOUT 5 YEARS; THE HOSE HAD BEEN INSTALLED IN EARLY 1980. THIS EVENT IS REPORTABLE UNDER 10 CFR 50.73(A)(2)(I)(B). THE HEALTH AND SAFETY OF THE PUBLIC WERE UNAFFECTED SINCE OFFSITE POWER SOURCES WERE AVAILABLE THROUGHOUT THE EVENT. ALL THE FLEXIBLE HOSES ON NO. 12 DCLP WERE REPLACED WITH HOSES OF A HIGHER TEMPERATURE RATING. HOSES ON NO. 22 DCLP WERE REPLACED WITH HOSES OF HIGHER RATING LAST YEAR.

[253] QUAD CITIES 1 DOCKET 50-254 LER 87-016
LEAK RATE FROM ALL VALVES AND PENETRATIONS IN EXCESS OF TECH SPEC LIMIT.
EVENT DATE: 091287 REPORT DATE: 092487 NSSS: GE TYPE: BWR

(NSIC 206708) ON SEPTEMBER 12. 1987 QUAD CITIES UNIT ONE WAS SHUTDOWN FOR THE END OF CYCLE REFUELING AND MAINTENANCE OUTAGE. AT 1600 HOURS. IT WAS DETERMINED THAT THE MEASURED COMBINED LEAKAGE RATE FROM ALL PENETRATIONS AND VALVES, EXCLUDING THE MAIN STEAM ISOLATION VALVES, EXCEEDED THE TECHNICAL SPECIFICATION (3.7.A 2) LIMIT OF 293.75 SCFH 0.60 LA). THIS WAS IDENTIFIED WHILE LOCAL LEAK RATE TESTING

THE MAIN STEAM LINE N VALVES. MO 1-220-1 AND 2. THE FAILURE MODE OF THE PENETRATIONS AND VALVES IS NOT KNOWN AT THIS TIME SINCE THE TESTING AND REPAIR OF THESE ITEMS IS NOT COMPLETE. A SUPPLEMENTAL REPORT WILL ADDRESS RATE TESTING AND REPAIRS WHEN THIS IS COMPLETED. THIS REPORT IS SUBMITTED TO COMPLY WITH THE REQUIREMENTS OF 10CFR50.73(A)(2)(II).

[254] QUAD CITIES 1 DOCKET 50-254 LER 87-019
FAILURE OF AS-FOUND INTEGRATED LEAK RATE TEST - CAUSE TO BE DETERMINED.
EVENT DATE: 091487 REPORT DATE: 092987 NSSS: GE TYPE: BWR

(NSIC 206665) ON SEPTEMBER 12, 1987. QUAD CITIES UNIT ONE WAS SHUTDOWN FOR THE START OF A REFUEL AND MAINTENANCE OUTAGE. ON SEPTEMBER 13, 1987, CONTAINMENT PRESSURIZATION WAS BEGUN PER QTS 150-1, INTEGRATED PRIMARY CONTAINMENT LEAK RATE TEST (IPCLRT). ON SEPTEMBER 14, AT 0800 HOURS, IT WAS DETERMINED THAT THE "AS FOUND" CONTAINMENT LEAKAGE ALTHOUGH NOT YET QUANTIFIED WAS PROBABLY IN EXCESS OF THE TECHNICAL SPECIFICATION 3.7.A.2.B LIMIT OF WT %/DAY. NRC NOTIFICATION OF THIS EVENT WAS COMPLETED AT 1045 HOURS TO SATISFY REQUIREMENTS OF 10CFR50.72. THE EXACT CAUSE FOR THIS FAILURE HAS NOT BEEN DETERMINED AND CORRECTIVE ACTION WILL BE BASED ON WHAT IS DISCOVERED DURING SUBSEQUENT INVESTIGATION AND TESTING. THIS INFORMATION WILL BE PROVIDED IN A SUPPLEMENT TO THIS REPORT. THIS REPORT IS PROVIDED TO COMPLY WITH THE REQUIREMENTS OF 10CFR50.73 (A)(2)(II).

[255] QUAD CITIES 2 DOCKET 50-265 LER 87-011
REACTOR SCRAM FROM LOW LEVEL SIGNAL DUE TO PERSONNEL ERROR WHILE PERFORMING SURVEILLANCE TEST.
EVENT DATE: 091787 REPORT DATE: 100587 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: QUAD CITIES 1 (BWR)

(NSIC 206711) ON SEPTEMBER 17, 1987, QUAD CITIES UNIT TWO WAS IN THE RUN MODE AT 93 PERCENT OF CORE THERMAL POWER. WHILE INSTRUMENT MAINTENANCE (IM) PERSONNEL WERE PERFORMING QIS 11-2 (LOW LOW REACTOR WATER LEVEL FUNCTIONAL TEST), A REACTOR SCRAM AND OTHER RELATED ENGINEERED SAFETY FEATURE (ESF) ACTUATIONS OCCURRED DUE TO AN INDICATED LOW REACTOR WATER LEVEL. THIS WAS A RESULT OF PERSONNEL ERROR BECAUSE THE IM FAILED TO PREPRESSURIZE LEVEL INSTRUMENT LIS 2-263-72C BEFORE REOPENING ITS ISOLATION VALVE. THIS CAUSED A PRESSURE TRANSIENT THAT ACTUATED LEVEL SWITCHES LIS 2-263-57A AND B WHICH CAUSED THE REACTOR SCRAM AND ASSOCIATED ESF ACTUATIONS. NRC NOTIFICATION PER 10CFR50.72 WAS COMPLETED AT 0717 HOURS. DISCIPLINARY ACTION WAS ADMINISTERED TO THE IM INVOLVED. ADDITIONALLY. THIS EVENT HAS BEEN DISCUSSED WITH ALL IM PERSONNEL. QIS 11-2 IS TO BE REVISED TO INCLUDE NOTES AND CAUTION STATEMENTS AND THE ASSOCIATED CHECKLIST WILL BE REVISED TO ADD A SIGNOFF STEP FOR PREPRESSURIZATION. ALL SIMILAR PROCEDURES WILL BE REVIEWED AND REVISED AS NECESSARY IF THEY LACK ADEQUATE CAUTIONS AND SIGNOFFS. THIS REPORT IS PROVIDED TO COMPLY WITH 10CFR50.73(A)(2)(IV).

[256] RANCHO SECO DOCKET 50-312 LER 87-006 REV 03
UPDATE ON ASSESSMENT OF MOTOR OPERATED VALVES.
EVENT DATE: 011587 REPORT DATE: 102387 NSSS: BW TYPE: PWR
VENDOR: ALLIS CHALMERS
ALOYCO, INC.
AMERICAN WARMING & VENTILATING INC.
ANCHOR VALVE CO.
ANCHOR/DARLING VALVE CO.

(NSIC 206790) DURING PRESENT COLD SHUTDOWN CONDITIONS AND AS A RESULT OF THE PROGRAM THAT WAS DEVELOPED IN RESPONSE TO IE BULLETIN NO. 85-03, "MOTOR OPERATED VALVE COMMON MODE FAILURES DURING PLANT TRANSIENTS DUE TO IMPROPER SWITCH SETTINGS", THE FOLLOWING PROBLEMS WITH THE MOVS HAVE BEEN IDENTIFIED. 1. OVER THRUST CONDITIONS. 2. BRAKE APPLICATION. 3. UNDERSIZE POWER CABLES TO THE

OPERATORS. 4. LACK OF STAKING OF STEM NUTS. 5. VALVE INTERNALS DAMAGE. 6. UNQUALIFIED OPERATOR GREASE. 7. PICKUP/DROP VOLTAGE OUT OF SPECIFICATION. THESE CONDITIONS HAVE EXISTED IN PART SINCE INITIAL PLANT OPERATION. THE ORIGINAL MOV DESIGN WAS ADEQUATE AT THAT TIME BUT NEW STATE-OF-THE- ART TESTING HAS RESULTED IN IDENTIFICATION OF FUNDAMENTAL DESIGN PROBLEMS. THE ASSESSMENT OF THE SAFETY CONSEQUENCES OF MOV PROBLEMS IS IN PROCESS AND WILL NOT BE COMPLETED UNTIL ALL VALVE TESTING IS COMPLETE. IT SHOULD BE NOTED THAT THE DISTRICT'S COMMITMENT TO CORRECTING THESE PAST FAILINGS IS EXEMPLIFIED BY HAVING INCLUDED ALL THE MOTOR OPERATED VALVES, BOTH SAFETY AND NON-SAFETY RELATED, IN THE PLANT, IN THE PRE AND POST RESTART REFURBISHMENT PROGRAM, AND NOT SOLELY THE 30 MOV'S REQUIRED BY THE NRC IE BULLETIN 85-03.

[257] RANCHO SECO DOCKET 50-312 LER 87-042
AMPHENOL BLUE RIBBON CONNECTOR FAILURES DUE TO NON-CONDUCTIVE COATING.
EVENT DATE: 091287 REPORT DATE: 100987 NSSS: BW TYPE: PWR
VENDOR: AMPHENOL

(NSIC 206615) DURING THE PERFORMANCE OF ROUTINE TEST PROCEDURES IN APRIL AND MAY 1987, I&C TECHNICIANS DETECTED ELECTRICAL CONTACT PROBLEMS WITH AMPHENOL BLUE RIBBON CONNECTORS. FURTHER INVESTIGATION BY I&C SUPERVISION REVEALED A HIGHER THAN NORMAL FREQUENCY OF SIMILAR PROBLEMS WITH AMPHENOL BLUE RIBBON CONNECTORS. CONNECTOR SAMPLES AND CONNECTOR CLEANING SOLUTION SAMPLES WERE SENT TO A DISTRICT-CONTRACTED LABORATORY FOR ANALYSIS. PRELIMINARY FINDINGS OUTLINED IN SURFACE SCIENCE LABORATORIES REPORT 6018-0887 FOUND THE PROBLEM CONNECTORS HAD A NON-CONDUCTIVE COATING ON THE CONNECTION POINTS OF CONTACT. IT IS SUSPECTED THE COATING DEVELOPED AS THE RESULT OF A CLEANING PROCESS USED BY THE DISTRICT IN 1986 AND 1987. DUE TO THE POTENTIAL IMPACT OF THIS CONDITION, THE DISTRICT JUDGED THIS DEVELOPMENT TO BE REPORTABLE IN ACCORDANCE WITH 10 CFR 50.73(A)(2)(V)(A). THE PLANT HAS BEEN IN COLD SHUTDOWN FROM THE TIME OF THE CLEANING TO THIS DISCOVERY.

[258] RANCHO SECO DOCKET 50-312 LER 87-043
PERSONNEL ERROR RESULTED IN IMPROPER MONITORING OF A NEW EFFLUENT RELEASE POINT.
EVENT DATE: 092487 REPORT DATE: 102387 NSSS: BW TYPE: PWR

(NSIC 206721) FOR THE PAST TWO YEARS THE DISTRICT HAS BEEN IN THE PROCESS OF INSTALLING A NEW GASEOUS EFFLUENT RELEASE POINT FROM THE AUXILIARY BUILDING. THE TECHNICAL SPECIFICATIONS STATE THAT THE MONITORING OF THIS RELEASE POINT WILL NOT BECOME EFFECTIVE UNTIL THE VENTILATION SYSTEM HAS BEEN DECLARED OPERABLE. THE TESTING OF THE SYSTEM AND ITS RELATED EFFLUENT MONITOR HAS RESULTED IN SEVERAL PROBLEMS INCLUDING: 1) UNMONITORED RELEASES AND 2) BEING UNABLE TO MEET THE TECHNICAL SPECIFICATION REQUIRED LOWER LIMITS OF DETECTION (LLD). THESE PROBLEMS ARE BEING REPORTED IN ACCORDANCE WITH 10 CFR 50.73.A.2.I(B). A REVIEW OF PREVIOUS EVENTS DID NOT DISCLOSE ANY THAT WERE SIMILAR TO THESE. IT APPEARS THAT THESE PROBLEMS ARE UNIQUE IN NATURE DUE TO THE SPECIAL SITUATION INVOLVING THE TECHNICAL SPECIFICATIONS. AS A CORRECTIVE MEASURE TO PREVENT RECURRENCE OF THESE OR SIMILAR PROBLEMS, THE OPERATION'S GROUP IS NOW REQUIRED TO CONTACT RADIATION PROTECTION PRIOR TO STARTING OR STOPPING THE SYSTEM AND THE EFFLUENT RELEASE POINT IS NOW BEING TREATED AS IF THE TECHNICAL SPECIFICATIONS WERE NOW IN EFFECT.

[259] RIVERBEND 1 DOCKET 50-458 LER 86-021 REV 01
UPDATE ON TRANSFORMER FAULT RESULTS IN HIGH PRESSURE CORE SPRAY INJECTION.
EVENT DATE: 030186 REPORT DATE: 101487 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 206683) AT 0648 ON 3/1/86, THE UNIT AUTOMATICALLY SCRAMMED ON LOW REACTOR WATER LEVEL 3 FROM 39 PERCENT POWER. THE SCRAM RESULTED FROM A TRANSFORMER FAULT AND VOLTAGE TRANSIENT ON THE 13.8 KV SWITCHGEAR. THE VOLTAGE TRANSIENT

APPARENTLY CAUSED THE LOSS OF THE MAIN FEEDWATER OIL PUMP, AS WELL AS VARIOUS OTHER LOADS, WHICH CAUSED A LOW LUBE OIL PRESSURE FEED PUMP TRIP. REACTOR CORE ISOLATION COOLING (RCIC) WAS MANUALLY STARTED AT 0650 AS REACTOR WATER LEVEL DROPPED TO THE LOW LEVEL 2 SETPOINT. AT THIS TIME HIGH PRESSURE CORE SPRAY (HPCS) AUTOMATICALLY INITIATED. AT 0653 LEVEL HAD BEEN RESTORED AND HPCS AND RCIC TRIPPED ON HIGH LEVEL 8. THERE WAS NO ADVERSE AFFECT ON THE HEALTH AND SAFETY OF THE PUBLIC AS ALL EMERGENCY SYSTEMS OPERATED AS DESIGNED.

[260] RIVERBEND 1 DOCKET 50-458 LER 87-019
FAILURE TO REPORT A CHANGE TO STARTUP PROCEDURE IN ACCORDANCE WITH OPERATING LICENSE.
EVENT DATE: 032086 REPORT DATE: 103087 NSSS: GE TYPE: BWR

(NSIC 206883) ON 9/30/87 DURING A REVIEW OF CHANGES MADE TO THE RIVER BEND STATION (RBS) FINAL SAFETY ANALYSIS REPORT (FSAR) CHAPTER 14.2, POST-FUEL-LOADING INITIAL TEST PROGRAM, IT WAS DISCOVERED THAT A CHANGE WAS MADE TO THE INITIAL TEST PROGRAM IN ACCORDANCE WITH 10CFR50.59 ON 3/20/86 BUT NOT REPORTED TO THE NRC IN ACCORDANCE WITH RBS OPERATING LICENSE ITEM 2.C.(12). THIS STARTUP TEST (ST) PROCEDURE CHANGE WAS MADE TO REFLECT THE REQUIREMENTS OF TECHNICAL SPECIFICATION 3.4.2.2. A REVIEW OF THE AS-CONDUCTED STS HAS CONCLUDED THAT ALL STS HAVE BEEN PERFORMED UTILIZING ACCEPTANCE CRITERIA WITHIN THOSE STATED IN THE USAR. IN ADDITION, A REVIEW OF THE REQUESTS TO CHANGE THE FSAR AND USAR ALSO SHOW THIS INSTANCE TO BE AN ISOLATED CASE. SINCE ALL STS DESCRIBED IN FSAR SECTION 14.2 HAVE BEEN PERFORMED, THERE WILL BE NO FURTHER CHANGES MADE TO THE STARTUP PROGRAM. THEREFORE, THERE IS NO APPLICABLE CORRECTIVE ACTION TO PREVENT RECURRENCE. SINCE THIS CHANGE WAS MADE TO LEVEL 2 ACCEPTANCE CRITERIA IN ACCORDANCE WITH 10CFR50.59 MERELY TO REFLECT THE REQUIREMENTS OF TECHNICAL SPECIFICATION 3.4.2.2, AND THE REPORTABLE CONDITION IS A FAILURE TO REPORT THIS CHANGE IN THE APPROPRIATE TIME FRAME, THIS EVENT HAD NO IMPACT ON THE HEALTH AND SAFETY OF THE PUBLIC.

[261] RIVERBEND 1 DOCKET 50-458 LER 86-068 REV 02
UPDATE ON RCIC ISOLATION DUE TO A FAILED PRESSURE TRANSMITTER.
EVENT DATE: 122386 REPORT DATE: 103087 NSSS: GE TYPE: BWR
VENDOR: ROSEMOUNT, INC.

(NSIC 206868) ON 12/23/86 AT 212, WITH THE UNIT AT 100 PERCENT POWER, A REACTOR CORE ISOLATION COOLING (RCIC) HIGH STEAM SUPPLY LINE FLOW DETECTION TRANSMITTER FAILED. A DIVISION I ISOLATION OF THE STEAM SUPPLY LINE OCCURRED AS A RESULT OF THE TRANSMITTER FAILURE. THE TRANSMITTER WAS REPLACED AND THE SYSTEM DECLARED OPERABLE AT 0329 ON 12/27/86. THE FAILED TRANSMITTER WAS RETURNED TO THE MANUFACTURER FOR FAILURE ANALYSIS. FURTHER CONTACT AND INVESTIGATION WITH THE MANUFACTURER DETERMINED THE ROOT CAUSE OF THE FAILURE TO BE MOISTURE ACCUMULATING IN THE ELECTRONICS HEAD. THERE WAS NO IMPACT ON THE HEALTH AND SAFETY OF THE PUBLIC AS A RESULT OF THIS EVENT SINCE RCIC ISOLATED AS DESIGNED UPON FAILURE OF THE TRANSMITTER. ADDITIONALLY, OTHER EMERGENCY CORE COOLING SYSTEMS REMAINED AVAILABLE HAD EMERGENCY REACTOR VESSEL INVENTORY MAKEUP BEEN REQUIRED.

[262] RIVERBEND 1 DOCKET 50-458 LER 87-008 REV 01
UPDATE ON CONTROL ROOM CHARCOAL FILTRATION START DUE TO RADIATION MONITOR SPIKE.
EVENT DATE: 052587 REPORT DATE: 103087 NSSS: GE TYPE: BWR

(NSIC 206882) AT 1550 ON 5/25/87 WITH THE UNIT AT APPROXIMATELY 70 PERCENT POWER, AN AUTOMATIC INITATION OF THE "B" TRAIN OF THE MAIN CONTROL ROOM FILTRATION SYSTEM OCCURRED. THE INITIATION WAS CAUSED BY SPURIOUS SIGNALS FROM MAIN CONTROL ROOM LOCAL INTAKE RADIATION MONITOR 1RMS*RE13B. THE OPERATORS DETERMINED THAT NO ACTUAL HIGH RADIATION CONDITION EXISTED AND RETURNED THE SYSTEM TO ITS NORMAL CONFIGURATION. RADIATION MONITOR 1RMS*RE13B AND OTHER RADIATION MONITORS HAVE

PREVIOUSLY SHOWN SUSCEPTABILITY TO ELECTRICAL NOISE. THESE OCCURRENCES WERE PREVIOUSLY REPORTED IN LERS 86-020 (1RMS*RELLA), 86-040 (1RMS*RE13B AND 1RMS*RE14B), 86-052 (1RMS*RE13B) AND 86-062 (1RMS*RELLA). FURTHER INVESTIGATION IS BEING CONDUCTED WITH THE RADIATION MONITOR VENDOR (SORRENTO ELECTRONICS) TO PROVIDE ADDITIONAL REDUCTION IN SENSITIVITY TO ELECTRICAL NOISE. THE SAFE OPERATION OF THE PLANT AND HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED AS A RESULT OF THIS EVENT. THE CONTINUED OPERATION OF THE PLANT WILL HAVE NO IMPACT ON SAFETY SINCE SYSTEMS WHICH ACTUATE PLACE THE PLANT IN A MORE CONSERVATIVE CONFIGURATION BY FILTERING THE AIR PRIOR TO RELEASING IT.

[263] RIVERBEND 1 DOCKET 50-458 LER 87-020
MISSED POST MAINTENANCE SURVEILLANCE ON ISI VALVES.
EVENT DATE: 060687 REPORT DATE: 102387 NSSS: GE TYPE: BWR

(NSIC 206840) DURING AN INTERNAL AUDIT CONDUCTED IN SEPTEMBER 1987 ON AL ASME WORK PACKAGES, ON SEPTEMBER 23, 1987, IT WAS DISCOVERED THAT AFTER PERFORMING MAJOR REPAIR WORK ON ASME VALVE 1WCS*MOV172, IT WAS DECLARED OPERABLE AND PLACED BACK INTO SERVICE ON 12/10/86 WITHOUT PERFORMING REQUIRED ASME XI VALVE EXERCISE TESTS. ALSO ON SEPTEMBER 25 AND SEPTEMBER 26, 1987 A SIMILAR DISCOVERY WAS MADE FOR ASME VALVE 1B21*AOVF032B WHICH WAS PLACED BACK INTO SERVICE ON NOVEMBER 29, 1986 AND ASME VALVE 1E21*AOVF006 WHICH WAS PLACED BACK INTO SERVICE ON 6/6/86. THE OMISSION OF THESE TESTS ON MAINTENANCE WORK DOCUMENTS WAS DUE TO PROGRAM INADEQUACIES IN ENGINEERING CONTROL AND MAINTENANCE PLANNING. CORRECTIVE ACTION INCLUDED ENGINEERING REORGANIZATION IN JANUARY, 1987, AND RETRAINING OF THE ENGINEERING, OPERATIONS, AND MAINTENANCE DEPARTMENTS ON THE CIRCUMSTANCES OF THIS REPORTABLE CONDITION. ALSO, REVISIONS OF PLANT AND ENGINEERING PROCEDURES WHICH COVER ASME XI REPAIR & REPLACEMENT AND THE MAINTENANCE WORK ORDER WERE REVISED TO PRECLUDE THESE OCCURRENCES. SINCE ALL VALVES IN QUESTION WERE VERIFIED OPERABLE AND COULD HAVE PERFORMED THEIR SAFETY FUNCTIONS, THERE WAS NO ADVERSE IMPACT ON THE HEALTH AND SAFETY OF THE PUBLIC, AS A RESULT OF THIS CONDITION.

[264] RIVERBEND 1 DOCKET 50-458 LER 87-021
FIRE SEAL NOT INSTALLED IN SPENT FUEL POOL COOLING PUMP CUBICLE WALL.
EVENT DATE: 101587 REPORT DATE: 111287 NSSS: GE TYPE: BWR

(NSIC 207120) AT 1000 HOURS ON 10/15/87 WITH THE UNIT IN COLD SHUTDOWN, GULF STATES UTILITIES COMPANY (GSU ENGINEERING IDENTIFIED TWO UNSEALED PENETRATIONS INTO THE CUBICLE FOR THE "B" SPENT FUEL POOL COOLING PUMP 1SFC*P1B. AT THE TIME OF DISCOVERY, LIMITING CONDITION FOR OPERATION (LCO) 87-079 WAS IN EFFECT AND A ROVING FIRE WATCH WAS POSTED FOR THIS ELEVATION. THEREFORE, THE REQUIRED TECHNICAL SPECIFICATION ACTIONS WERE IN PLACE AT THE TIME OF DISCOVERY. MAINTENANCE WORK ORDER REQUEST (MWOR) 111141 WAS SUBSEQUENTLY WRITTEN TO SEAL THESE OPENINGS WITH THREE HOUR RATED FIRE SEALS. FURTHER INVESTIGATION DETERMINED THAT THESE OPENINGS WERE CUT INTO THE CONCRETE BLOCK MAKING UP THE LEFT SIDE OF THE DOOR OPENING TO THE "B" SPENT FUEL POOL COOLING PUMP CUBICLE TO ACCOMMODATE ENTRY OF CONDUIT AND INSTRUMENTATION LINES. THESE PENETRATIONS WERE NOT PLACED IN A "TYPICAL" WALL AND THEREFORE, THE PENETRATION SEAL CONTRACTOR WAS UNAWARE OF THESE OPENINGS AND THEREFORE, DID NOT SEAL THEM. THIS RESULTED IN THESE PENETRATIONS BEING UNSEALED SINCE THE CONCLUSION OF CONSTRUCTION ACTIVITIES. FIELD CHANGE NOTICE 3 TO MODIFICATION REQUEST 86-0846 HAS BEEN INITIATED TO ADD THESE PENETRATIONS TO THE APPLICABLE DESIGN DRAWINGS. ADDITIONALLY, A REVIEW OF OTHER NON- TYPICAL WALL INSTALLATIONS HAS CONCLUDED THAT THIS IS AN ISOLATED CASE.

[265] RIVERBEND 1 DOCKET 50-458 LER 87-022
RESIDUAL HEAT REMOVAL SYSTEM ISOLATION DUE TO PROCEDURAL ERROR.
EVENT DATE: 101687 REPORT DATE: 111387 NSSS: GE TYPE: BWR

(NSIC 207121) ON 10/16/87 AT 1118 HOURS WITH THE UNIT IN MODE 5 (REFUELING), AN ISOLATION OF THE RESIDUAL HEAT REMOVAL (RHR) SYSTEM SHUTDOWN COOLING SUPPLY VALVE (1E12*MOVF008) OCCURRED DURING THE PERFORMANCE OF SURVEILLANCE TEST PROCEDURE (STP)-051-4209, "RPS/RHR - REACTOR VESSEL STEAM DOME PRESSURE - HIGH, MONTHLY CHFUNCT, 18 MONTH CHCAL, AND 18 MONTH LSFT (B21-N078A, B21-N678A, B21-N679A)". THE CAUSE OF THIS ISOLATION WAS DETERMINED TO BE IMPROPERLY SEQUENCED STEPS IN STP-051-4209 WHICH CAUSED ALTERNATE POWER TO BE DISCONNECTED FROM THE ISOLATION RELAY COIL (K129A) PRIOR TO RESTORING NORMAL POWER. STP-051-4209 WAS CORRECTED. STP-051-4210 (CHANNEL B) AND 051-4211 (CHANNEL C) WERE DISCOVERED TO CONTAIN THE SAME ERROR AND WERE CORRECTED. THE TIME THAT RHR WAS ISOLATED DURING THIS EVENT WAS MINIMAL (TWO MINUTES), AND ALTERNATE MEANS OF DECAY HEAT REMOVAL WERE AVAILABLE IF RHR SHUTDOWN COOLING COULD NOT HAVE BEEN RESTORED. THERE WAS NO IMPACT ON THE SAFE OPERATION OF THE PLANT OR TO THE HEALTH AND SAFETY OF THE PUBLIC AS A RESULT OF THIS EVENT.

[266] ROBINSON 2 DOCKET 50-261 LER 87-023
SHORT TERM LOSS OF AVAILABLE ONSITE EMERGENCY.
EVENT DATE: 082687 REPORT DATE: 093087 NSSS: WE TYPE: PWR
VENDOR: FAIRBANKS CO, THE
WOODWARD GOVERNOR COMPANY

(NSIC 206608) ON AUGUST 26, AT 0145 HOURS, AND AT 2236 HOURS ON SEPTEMBER 8, 1987, WITH UNIT 2 OPERATING AT 100 PERCENT POWER AND DIESEL GENERATOR "B" (DG-B) OUT OF SERVICE FOR MAINTENANCE, DIESEL GENERATOR "A" (DG-A) AUTOMATICALLY TRIPPED ON OVERSPEED UPON STARTING. A SECOND START ATTEMPT ON BOTH DATES RESULTED IN A SECOND AUTOMATIC OVERSPEED TRIP. ON BOTH OCCASIONS, DAILY OPERABILITY TESTING OF DG-A WAS BEING CONDUCTED AS REQUIRED BY THE OCCASIONS, DG-A WAS DECLARED INOPERABLE. THE LICENSEE, ON EACH OCCASION, NOTIFIED THE NRC EMERGENCY NOTIFICATION SYSTEM PURSUANT TO 10 CFR 50.72(B)(2)(II). FOLLOWING THE INCIDENT ON AUGUST 26, DG-A WAS DECLARED OPERATIONAL AND RETURNED TO SERVICE WITHIN 2.25 HOURS. FOLLOWING THE INCIDENT ON SEPTEMBER 8, DG-B WAS RETURNED TO SERVICE WITHIN 6.5 HOURS WHILE DG-A UNDERWENT TROUBLESHOOTING TO DETERMINE CAUSE.

[267] ROBINSON 2 DOCKET 50-261 LER 87-024
DISCOVERY OF REACTOR PROTECTION AND CONTROL ANALOG INSTRUMENTATION RACK INADEQUATE ANCHORAGE DUE TO INSTALLATION DISCREPANCY.
EVENT DATE: 091887 REPORT DATE: 101387 NSSS: WE TYPE: PWR
VENDOR: HAGAN CONTROLS
WESTINGHOUSE ELECTRIC SUPPLY COMPANY

(NSIC 206710) ON SEPTEMBER 4, 1987, DURING A WALKDOWN OF ELECTRICAL EQUIPMENT IN THE REACTOR PROTECTION AND CONTROL ANALOG INSTRUMENTATION RACKS (HAGAN RACKS), ANCHORAGE TO MAINTAIN SEISMIC CLASS I REQUIREMENTS WAS QUESTIONED. ON SEPTEMBER 11, 1987, UNIT 2 WAS REDUCED TO HOT SHUTDOWN TO REPAIR LEAKS IN THE MAIN UNIT GENERATOR HYDROGEN COOLERS. SINCE PRELIMINARY CALCULATIONS INDICATED THAT THE HAGAN RACKS DID HAVE INSUFFICIENT ANCHORAGE, A PLANT MODIFICATION WAS IMPLEMENTED DURING THE OUTAGE TO UPGRADE THE ANCHORAGE AS A PRECAUTIONARY MEASURE WHILE MORE DETAILED ANALYSES WERE PERFORMED. ON SEPTEMBER 18, 1987, RESULTS OF THE MORE DETAILED ANALYSES REVEALED THAT THE ANCHORAGE AS FOUND INDEED WAS INADEQUATE. THE NRC WAS NOTIFIED BY THE LICENSEE VIA THE EMERGENCY NOTIFICATION SYSTEM PURSUANT TO 10CFR50.72(B)(2)(I). THE CAUSE OF THIS EVENT IS ATTRIBUTED TO IMPROPER INSTALLATION DURING ORIGINAL CONSTRUCTION, ALTHOUGH INFORMATION AVAILABLE FROM THE CONSTRUCTION TIME PERIOD WAS INADEQUATE TO ENABLE PLANT STAFF TO ESTABLISH THE ORIGINAL INSTALLATION REQUIREMENTS OF HE HAGAN RACKS. THE MODIFICATION TO UPGRADE THE HAGAN RACK ANCHORAGE WAS COMPLETED ON SEPTEMBER 15, 1987, PRIOR TO RETURN TO POWER OPERATION.

[268] ROBINSON 2 DOCKET 50-261 LER 87-025
UNPLANNED REACTOR TRIP DUE TO COGNITIVE PERSONNEL ERROR DURING REACTOR TRIP BREAKER TESTING.
EVENT DATE: 092887 REPORT DATE: 102787 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 206782) ON SEPTEMBER 28, 1987, UNIT 2 WAS AT 100 PERCENT POWER WHEN THE REACTOR TRIPPED AT 1445 HOURS DUE TO A COGNITIVE PERSONNEL ERROR. A PLANT MAINTENANCE INSTRUMENTATION & CONTROL TECHNICIAN PERFORMING A SCHEDULED TEST OF THE REACTOR PROTECTION LOGIC TRAIN "B" INADVERTENTLY TRIPPED THE TRAIN "A" REACTOR TRIP BREAKER RESULTING IN A ONE-OUT-OF-TWO REACTOR TRIP LOGIC. THE NRC SENIOR RESIDENT INSPECTOR FOR THE PLANT WAS NOTIFIED AT 1510 HOURS. THE UTILITY-LICENSED SHIFT FOREMAN NOTIFIED THE NRC AT 1618 VIA THE EMERGENCY NOTIFICATION SYSTEM PURSUANT TO 10CFR50.72(B)(2)(I). THE REACTOR AUTOMATICALLY WENT TO ZERO PERCENT POWER AND INTO A SAFE, STABLE HOT SHUTDOWN CONDITION. THE TRIP WAS CLEARED, THE SURVEILLANCE TEST PROPERLY COMPLETED, AND THE REACTOR MADE CRITICAL AT 2333 HOURS, RETURNING ON-LINE AT 0047 HOURS, SEPTEMBER 29, 1987. THE LICENSEE HAS DISCIPLINED THE TECHNICIAN RESPONSIBLE AND HAS REVIEWED THE EVENT TO DETERMINE WHETHER ADDITIONAL ACTIONS ARE NECESSARY TO PRECLUDE RECURRENCE. THERE WERE NO COMPONENT FAILURES DURING THE EVENT. THERE ARE NO KNOWN PRIOR EVENTS SIMILAR TO THIS ISOLATED INCIDENT. THE EVENT IS REPORTED PURSUANT TO 10CFR50.73.(A)(2)(I)(A).

[269] SALEM 1 DOCKET 50-272 LER 87-011
POTENTIALLY INADEQUATE BREAKER COORDINATION.
EVENT DATE: 091787 REPORT DATE: 101687 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 2 (PWR)

(NSIC 206749) ON SEPTEMBER 17, 1987 AT 2000 HOURS, IT WAS DETERMINED THAT BREAKER COORDINATION COULD NOT BE SHOWN TO BE DOCUMENTED FOR SEVERAL VOLTAGE LEVELS IN EITHER SALEM UNIT 1 OR UNIT 2. THIS CONCLUSION IS BASED ON A DRAFT REPORT OF AN EVALUATION OF PROTECTIVE RELAYING OF VITAL BUSSES WITH RESPECT TO THE REQUIREMENTS OF 10CFR50 APPENDIX R. THERE IS A POSSIBILITY THAT CIRCUITS ASSOCIATED WITH NON-SHUTDOWN LOADS DAMAGED BY A POSTULATED FIRE COULD CAUSE THE LOSS OF POWER TO SHUTDOWN EQUIPMENT FED FROM SEPARATE VITAL BUSSES. THE ROOT CAUSE HAS NOT BEEN DETERMINED. INVESTIGATIONS ARE CONTINUING TO IDENTIFY THE HISTORICAL BREAKER COORDINATION BASIS AND TO ESTABLISH THE ADEQUACY OF THE CURRENT CONFIGURATION. ROOT CAUSE WILL BE IDENTIFIED BASED UPON THE RESULTS OF THESE INVESTIGATIONS. RESULTS ARE EXPECTED BY THE END OF DECEMBER 1987. ACTIONS TAKEN UNTIL COMPLETION OF THE BREAKER COORDINATION STUDY INCLUDES ESTABLISHMENT OF FIRE WATCHES WHERE POSTULATED FIRES COULD POTENTIALLY DAMAGE REDUNDANT VITAL CABLING.

[270] SALEM 1 DOCKET 50-272 LER 87-012
INOPERABLE UNDERVOLTAGE TRIP CHANNEL WAS NOT PLACED IN THE TRIPPED CONDITION WITHIN ONE HOUR.
EVENT DATE: 093087 REPORT DATE: 102987 NSSS: WE TYPE: PWR

(NSIC 207127) ON SEPTEMBER 30, 1987 AT 1309 HOURS, PREVENTIVE MAINTENANCE WAS INITIATED ON THE 1G GROUP BUS REACTOR COOLANT PUMP UNDERVOLTAGE TRIP RELAY (27X). TECH SPEC 3.3.1.1 ACTION 6 WAS ENTERED AT THIS TIME. AT 1409 HOURS, TECH SPEC 3.0.3 WAS ENTERED SINCE THE INOPERABLE UNDERVOLTAGE TRIP CHANNEL WAS NOT PLACED IN THE TRIPPED CONDITION WITHIN ONE HOUR. TECH SPEC 3.0.3 WAS EXITED 15 MINUTES AFTER ENTRY WHEN THE RCP UNDERVOLTAGE TRIP SOLID STATE PROTECTION BI-STABLE WAS PLACED IN THE TRIPPED CONDITION. THE ROOT CAUSE OF THIS EVENT WAS INADEQUATE RELAY DEPARTMENT PROCEDURES. THE RELAY DEPARTMENT MANUAL WILL BE REVISED TO INCLUDE A DIRECTIVE IDENTIFYING THOSE RELAYS ASSOCIATED WITH TECH SPEC RELATED SAFETY FUNCTIONS. THE DIRECTIVE WILL SPECIFY A REQUIREMENT TO PLACE THE ASSOCIATED SSP CIRCUIT IN A TRIPPED/BYPASSED CONDITION, IN ACCORDANCE WITH TECH

SPECS, WHEN DISABLING THE RELAY. ADDITIONALLY, RELAY DEPARTMENT PROCEDURES ARE BEING DEVELOPED, IN LIEU OF THE RELAY MANUAL, WHICH WILL PRECLUDE RECURRENCE OF THIS EVENT. ADDITIONALLY, WORK ORDERS ASSOCIATED WITH RELAYS WHICH HAVE TECH SPEC RELATED FUNCTIONS, WILL BE MODIFIED,

[271] SALEM 1 DOCKET 50-272 LER 87-013
REACTOR TRIP FROM 0% POWER DUE TO INADEQUATE DESIGN OF DETECTOR HOUSING.
EVENT DATE: 100287 REPORT DATE: 103087 NSSS: WE TYPE: PWR

(NSIC 207129) ON OCTOBER 2, 1987 A CONTROLLED SHUTDOWN WAS IN PROGRESS IN PREPARATION FOR THE INITIATION OF THE UNIT'S SEVENTH REFUELING OUTAGE. AT 2216 HOURS, WITH THE UNIT AT 0% POWER AND SUBCRITICAL, A REACTOR TRIP OCCURRED ON SOURCE RANGE HIGH NEUTRON FLUX, CHANNEL N31 (IG). THE ROOT CAUSE OF THE TRIP WAS INADEQUATE DESIGN. SOURCE RANGE DETECTOR N31 WAS FOUND TO CONTAIN WATER RESULTING IN AN INCREASE OF LEAKAGE CURRENT BETWEEN THE DETECTOR TUBE WALL AND THE DETECTOR HOUSING WALL. THE INCREASED LEAKAGE CURRENT CAUSED THE SOURCE RANGE METER TO PEG HIGH. THE SOURCE OF WATER WAS DETERMINED TO BE CONDENSATION. THE DETECTOR HOUSING WAS NOT DESIGNED TO ALLOW EASY REMOVAL OF WATER COLLECTED FROM CONDENSATION. THE DETECTOR AND ITS HOUSING ASSOCIATED WITH SOURCE RANGE CHANNEL N31 WAS REPLACED. THE NEW HOUSING HAS CONDENSATION DRAINAGE HOLES.

[272] SALEM 2 DOCKET 50-311 LER 86-008
EQUIPMENT HATCH OPENED AND NO FIRE WATCH ESTABLISHED DUE TO PERSONNEL ERROR.
EVENT DATE: 091786 REPORT DATE: 101686 NSSS: WE TYPE: PWR

(NSIC 206564) ON SEPTEMBER 17, 1986 AT 2225 HOURS, THE 9' BY 16' EQUIPMENT HATCH, BETWEEN ELEVATIONS 122' AND 100' IN THE AUXILIARY BUILDING, WAS DISCOVERED OPEN BY A FIRE DEPARTMENT ROVING PATROL. NO FIRE WATCH WAS POSTED AS REQUIRED BY TECH SPEC ACTION STATEMENT 3.7.11.A. THE ROOT CAUSE OF THIS EVENT IS PERSONNEL ERROR DUE TO INADEQUATE COMMUNICATION OF THE REQUIREMENT TO TREAT THE SUBJECT HATCH AS A FIRE BARRIER WHICH REQUIRES AN IMPAIRMENT PERMIT PRIOR TO OPEN IT. UPON DISCOVERY OF THE OPEN HATCH A ROVING FIRE WATCH WAS ESTABLISHED TO MAINTAIN SURVEILLANCE OF THE IMPAIRMENT UNTIL THE HATCH WAS CLOSED. TO PREVENT FUTURE RECURRENCE OF THIS EVENT A LETTER TO ALL APPLICABLE PERSONNEL HAS BEEN ISSUED SPECIFYING FLOOR HATCHES IN THE AUXILIARY BUILDING (EXCEPT THOSE WITH SPECIFIC EXEMPTION) ARE CONSIDERED FIRE BARRIERS AND OPENING THEM IS AN IMPAIRMENT REQUIRING COMPLIANCE WITH ADMINISTRATIVE PROCEDURE AP-25, "FIRE PROTECTION PROGRAM". ALSO, THE HATCHES WHICH ARE FIRE BARRIERS, WILL BE UNIQUELY IDENTIFIED.

[273] SALEM 2 DOCKET 50-311 LER 87-009 REV 04
UPDATE ON APPENDIX R CRITERIA NON-CONFORMANCE.
EVENT DATE: 091087 REPORT DATE: 100687 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 1 (PWR)

(NSIC 206764) THE FOLLOWING SYSTEM/COMPONENT CONDITIONS WERE IDENTIFIED BY A PSE&G TASK FORCE REVIEWING/EVALUATING SALEM STATION'S COMPLIANCE WITH THE REQUIREMENTS OF 10CFR 50 APPENDIX R. LER 87-009-00 ADDRESSED A SW SYSTEM CABLING APPENDIX R SEPARATION CRITERIA INADEQUACY. THE ROOT CAUSE WAS INADEQUATE DESIGN REVIEW. THE CURRENT DESIGN MEETS THE ORIGINAL ELECTRICAL SEPARATION CRITERIA, BUT NOT THE APPENDIX R CRITERIA . PSE&G IS REVIEWING DESIGN CHANGE OPTIONS. LER 87-009-01 ADDRESSED NON-SEISMICALLY QUALIFIED MARINITE WALLS LOCATED IN SALEM UNITS 1 & 2 460V SWITCHGEAR ROOM. THE WALLS HAVE BEEN REINFORCED TO SEISMIC CRITERIA. A SAMPLE OF DESIGN CHANGES INSTALLED BEFORE IMPLEMENTATION OF CURRENT DESIGN CONTROL PROCEDURES IS BEING CONDUCTED. LER 87-009-02 ADDRESSES RHR ROOM COOLERS (VF) CABLING APPENDIX R INADEQUACIES AND CONTROL CABLING APPENDIX R INADEQUACIES FOR RHR ROOM COOLERS. CHARGING PUMP ROOM COOLERS (VF) AND DIESEL GENERATOR FUEL OIL TRANSFER PUMPS (DC). THE ROOT CAUSE OF THESE APPENDIX R INADEQUACIES IS INADEQUATE DESIGN REVIEW. IN BOTH CASES, AN HOURLY ROVING FIRE

WATCH PATROL WAS ESTABLISHED FOR THE RESPECTIVE AREAS. A DESIGN CHANGE CORRECTING THESE DEFICIENCIES WILL BE MADE. LER 87-009-03 ADDRESSED A D/G POWER CABLING APPENDIX R SEPARATION CRITERIA DEFICIENCY. THE ROOT CAUSE WAS INADEQUATE DESIGN REVIEW.

[274] SAN ONOFRE 1 DOCKET 50-206 LER 87-013
FAILURE TO TEST REDUNDANT HYDRAZINE PUMP PRIOR TO PERFORMING MAINTENANCE.
EVENT DATE: 090287 REPORT DATE: 100287 NSSS: WE TYPE: PWR

(NSIC 206606) AT 0919, ON 9/2/87, WITH UNIT 1 AT 92% POWER, CALIBRATION OF A HYDRAZINE STORAGE TANK LEVEL TRANSMITTER LIS-500A COMMENCED, WHICH INVOLVED THE DISABLING OF THE AUTO-START CAPABILITY OF HYDRAZINE PUMP G-200A. TECH SPEC 3.3.1.C REQUIRES REDUNDANT PUMP G-200B BE TESTED TO DEMONSTRATE ITS AVAILABILITY PRIOR TO INITIATING MAINTENANCE. AT 0921, CONTROL ROOM OPERATORS DISCOVERED THAT THE REDUNDANT PUMP G-200B HAD NOT BEEN TESTED PRIOR TO INITIATING THIS MAINTENANCE, CONTRARY TO TECH SPEC 3.3.1.C. THE AUTO-START CAPABILITY OF G-200A WAS RESTORED IMMEDIATELY. FOLLOWING SATISFACTORY TESTING OF G-200B, CALIBRATION OF LIS-500A WAS RESUMED. EQUIPMENT CONTROL PERSONNEL, UPON REVIEW OF THE WORK PLAN AND ASSOCIATED TECH SPEC, FAILED TO RECOGNIZE AND DOCUMENT IN THE WORK AUTHORIZATION RECORD (WAR) THE NEED FOR REDUNDANT COMPONENT TESTING. ALSO, LICENSED CONTROL ROOM OPERATORS INVOLVED WITH THE REVIEW AND APPROVAL OF THE WAR FAILED TO RECOGNIZE THAT THE "REDUNDANT TEST REQUIREMENT" EVALUATION WAS INCORRECT. AS A RESULT, THE OPERATORS UTILIZING THE WAR TO DIRECT WORK ACTIVITY DID NOT PERFORM THE REDUNDANT COMPONENT TEST. PERSONNEL INVOLVED HAVE BEEN COUNSELED. THIS EVENT WILL BE REVIEWED WITH APPROPRIATE PERSONNEL IN ORDER TO IMPROVE THE REVIEW AND IMPLEMENTATION OF WARS. AN ENHANCEMENT TO THE WAR EVALUATION PROCESS IS UNDER DEVELOPMENT.

[275] SAN ONOFRE 1 DOCKET 50-206 LER 87-015
ENGINEERED SAFETY SYSTEMS DESIGN FAILS TO MEET SINGLE FAILURE CRITERIA.
EVENT DATE: 100787 REPORT DATE: 110687 NSSS: WE TYPE: PWR

(NSIC 206869) ON OCTOBER 7, 1987, WITH UNIT 1 AT 92% POWER, AN ONGOING ENGINEERED SAFETY FEATURES (ESF) ANALYSIS DETERMINED SEVERAL SCENARIOS WHERE A SINGLE FAILURE COULD PREVENT CERTAIN ESF SYSTEMS FROM PERFORMING THEIR FUNCTIONS AS REQUIRED FOR DESIGN BASIS TRANSIENTS AND ACCIDENTS. THE CAUSE OF THE EVENT WAS DETAILED IN LETTERS FROM M. O. MEDFORD (SCE) TO DOCUMENT CONTROL DESK (NRC), SUBJECT: ESF SINGLE FAILURE ANALYSIS, DOCKET NO. 50-206, WHICH WERE SUBMITTED ON OCTOBER 16, 1987, AND NOVEMBER 6, 1987. IMMEDIATE CORRECTIVE ACTION TO PRECLUDE OCCURRENCE OR MITIGATE THE CONSEQUENCES WAS TAKEN BY ENHANCING ADMINISTRATIVE CONTROLS, OPERATOR TRAINING AND COMPLETION OF BEST ESTIMATE ANALYSIS. PLANT MODIFICATIONS TO CORRECT SINGLE FAILURE DISCREPANCIES UNDER CONSIDERATION ARE TO BE COMPLETED DURING THE NEXT REFUELING OUTAGE. THE HEALTH AND SAFETY OF PLANT PERSONNEL AND THE PUBLIC WERE NOT AFFECTED BY THIS EVENT.

[276] SAN ONOFRE 2 DOCKET 50-361 LER 86-022 REV 02
UPDATE ON REACTOR TRIP FOLLOWING MAIN STEAM ISOLATION SIGNAL.
EVENT DATE: 081286 REPORT DATE: 110687 NSSS: CE TYPE: PWR

(NSIC 206899) ON AUGUST 12, 1986 AT 1330 WITH UNIT 2 AT 100% POWER, A REACTOR TRIP OCCURRED WHEN REACTOR COOLANT SYSTEM (RCS) PRESSURE REACHED THE CORE PROTECTION CALCULATOR (CPC) AUXILIARY TRIP SET POINT OF 2375 PSIA. THE RCS PRESSURE TRANSIENT RESULTED FROM MAIN STEAM ISOLATION VALVE (MSIV) CLOSURE WHEN THE MAIN STEAM ISOLATION SYSTEM (MSIS) WAS ACTUATED DURING SURVEILLANCE TESTING OF THE MSIS AUTOMATIC ACTUATION LOGIC. THE TRIP RECOVERY PROCEEDED NORMALLY AND THERE WERE NO SAFETY CONSEQUENCES ASSOCIATED WITH THIS EVENT. THE MSIS GROUP ACTUATION RELAYS ARE MAINTAINED ENERGIZED BY CURRENT THROUGH TWO PARALLEL CIRCUITS FROM THE ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS) TRIP

INITIATION SOLID STATE RELAYS (SSR). DURING THE TECHNICAL SPECIFICATION REQUIRED SURVEILLANCE TEST, ONE SIDE OF THE PARALLEL CIRCUIT IS DE-ENERGIZED FOR BOTH TRAINS "A" AND "B" SIMULTANEOUSLY WHILE THE OTHER SIDE REMAINS ENERGIZED. DURING THIS TESTING, DE-ENERGIZATION OF THE SSR IN THE REMAINING PARALLEL CIRCUITRY RESULTED IN ACTUATION OF THE MSIS. FOLLOWING THE OCCURRENCE, EXTENSIVE INSPECTION AND TESTING OF ESFAS CIRCUITRY WAS CONDUCTED, HOWEVER, THE CONDITION COULD NOT BE DUPLICATED AND A FAILED COMPONENT COULD NOT BE IDENTIFIED. THROUGH SUBSEQUENT INVESTIGATION, HOWEVER, THE CAUSE OF THE OCCURRENCE WAS DETERMINED TO BE AN INTERMITTENT FAILURE OF A DC POWER SUPPLY FILTER CAPACITOR WHICH HAD REACHED THE END OF ITS SERVICE LIFE.

[277] SAN ONOFRE 2 DOCKET 50-361 LER 87-014 REV 01
UPDATE ON SHUTDOWN COOLING ISOLATION VALVE PACKING GLAND LEAKAGE.
EVENT DATE: 083187 REPORT DATE: 102687 NSSS: CE TYPE: PWR
VENDOR: LIMITORQUE CORP.
WKM VALVE DIVISION

(NSIC 206814) ON 8/31/87, AT APPROX. 1900, WITH UNIT 2 IN MODE 5 AND THE REACTOR COOLANT SYSTEM (RCS) AT APPROX. 350 PSIA AND 127 DEGREES F, FAILURE OF ALLOY STEEL PACKING GLAND FOLLOWER STUDS DURING MANUAL OPERATION OF MOTOR OPERATED SHUTDOWN COOLING SYSTEM (SDCS) SUCTION ISOLATION VALVE 2HV-9378 RESULTED IN LEAKAGE ESTIMATED AT 100 GPM THROUGH THE PACKING GLAND. OPERATION OF THE SDCS CONTINUED VIA A REDUNDANT FLOW PATH. RCS INVENTORY WAS MAINTAINED BY ISOLATING LETDOWN FLOW AND USING CHARGING PUMPS AS DEPRESSURIZATION AND VENTING OF THE RCS PROCEEDED. CONTAINMENT CLOSURE WAS PROMPTLY RESTORED AND THERE WAS NO EFFLUENT RELEASE FROM CONTAINMENT ABOVE REGULATORY LIMITS. AT 1100 ON SEPTEMBER 1, A TEMPORARY REPAIR WAS COMPLETED WHICH REDUCED THE LEAK RATE TO APPROXIMATELY 1/4 GPM, EFFECTIVELY TERMINATING THE EVENT. THE CAUSE OF STUD FAILURE HAS BEEN ATTRIBUTED TO (1) PACKING LEAKAGE RESULTING IN WASTAGE DUE TO BORIC ACID CORROSION, (2) DECREASE IN LUBRICATING CHARACTERISTICS AND HARDENING OF PACKING, AND (3) THE INITIAL THRUST REQUIRED TO OPEN THE VALVE. CORRECTIVE ACTIONS INCLUDE REDUCTION OF THE MAXIMUM THRUST NECESSARY TO OPEN THE VALVE BY INSTALLATION OF A MODIFIED PACKING GLAND ASSEMBLY LESS SUSCEPTIBLE TO LEAKAGE AND HARDENING OF PACKING, AND REPLACEMENT OF PACKING GLAND STUDS WITH CORROSION RESISTENT MATERIAL. THE HEALTH AND SAFETY OF PLANT PERSONNEL AND THE PUBLIC WERE NOT AFFECTED BY EVENT.

[278] SAN ONOFRE 2 DOCKET 50-361 LER 87-013
FUEL HANDLING ISOLATION SYSTEM (FHIS) ACTUATIONS DURING PRESSURIZER MANWAY REMOVAL.
EVENT DATE: 090487 REPORT DATE: 100587 NSSS: CE TYPE: PWR

(NSIC 206648) ON 9/4/87, AT 2329, AND 2354, WITH UNIT 2 IN MODE 5, FUEL HANDLING ISOLATION SYSTEM (FHIS) TRAIN "A" AND "B", RESPECTIVELY, ACTUATED DUE TO HIGH AIRBORNE ACTIVITY LEVELS IN THE FUEL HANDLING BUILDING (FHB). THE HIGH LEVELS RESULTED FROM RADIOACTIVE GASES DISCHARGED INTO CONTAINMENT (AND SUBSEQUENTLY INTO THE FHB) FROM THE PRESSURIZER. ALL FHIS COMPONENTS WERE VERIFIED TO HAVE ACTUATED IN ACCORDANCE WITH DESIGN. AT 0305, ON 9/5/87, AIRBORNE ACTIVITY LEVELS WERE VERIFIED TO BE BELOW THE ACTUATION SETPOINT, AND FHIS WAS RESET. THE CAUSE OF THE EVENT WAS THE ERRONEOUS ATTACHMENT OF A VENTILATION SUCTION HOSE TO A TEMPORARY COVER INSTALLED ON THE PRESSURIZER MANWAY. THE MAINTENANCE FOREMAN INCORRECTLY DIRECTED THIS ACTION BASED ON HIS PRIOR EXPERIENCE WITH THE REMOVAL OF STEAM GENERATOR MANWAYS. THE MAINTENANCE FOREMAN HAS BEEN COUNSELED REGARDING THE PERFORMANCE OF WORK ACTIVITIES WITHOUT PROPER AUTHORIZATION. ADDITIONALLY, THIS EVENT WILL BE REVIEWED WITH ALL MAINTENANCE PERSONNEL. THERE IS NO SAFETY SIGNIFICANCE TO THIS EVENT SINCE ALL FHIS COMPONENTS FUNCTIONED AS DESIGNED.

[279] SAN ONOFRE 2 DOCKET 50-361 LER 87-015
CONTAINMENT PURGE ISOLATION SYSTEM (CPIS) SPURIOUS ACTUATIONS.
EVENT DATE: 090587 REPORT DATE: 100587 NSSS: CE TYPE: PWR
VENDOR: NUCLEAR MEASUREMENTS CORP.

(NSIC 206649) BETWEEN 9/5/87 AND 9/21/87, WITH UNIT 2 SHUTDOWN FOR REFUELING, SPURIOUS ACTUATIONS OF THE SAN ONOFRE UNIT 2 TRAIN "A" CONTAINMENT PURGE ISOLATION SYSTEM (CPIS) WERE INITIATED BY CONTAINMENT AIRBORNE RADIATION MONITOR 2RT- 7856 ON 20 OCCASIONS. THE SPURIOUS ACTUATIONS REPORTED HERE, AND IN PREVIOUS LERS, ARE MOST FREQUENTLY CAUSED BY OPERATION OF TRAIN "A" HIGH PRESSURE SAFETY INJECTION (HPSI) SYSTEM COMPONENTS. CONTAINMENT PURGE ISOLATION VALVES WERE IN THE OPEN POSITION FOR 19 OF THE 20 ACTUATIONS, AND IN EACH INSTANCE THEY CLOSED IN ACCORDANCE WITH DESIGN REQUIREMENTS. IN ALL CASES, CONTAINMENT RADIATION LEVELS WERE BELOW THE CPIS ACTUATION SET POINT. INVESTIGATION INTO THE CAUSE OF THE FREQUENT 2RT-7856 INITIATED CPIS ACTUATIONS HAS IDENTIFIED A NUMBER OF GROUNDING NON-CONFORMANCES WHICH INCREASE THE SENSITIVITY OF 2RT-7856 TO ELECTRONIC NOISE. CORRECTIVE ACTIONS IMPLEMENTED INCLUDE MODIFYING 2RT-7856 TO CONFORM WITH THE DESIGN REQUIREMENTS AND INSTALLATION OF A TIME DELAY IN THE MONITOR LOGIC CIRCUIT. MONITOR 2RT- 7856 WILL BE TESTED TO ASSURE SUBSTANTIALLY REDUCED SENSITIVITY TO NOISE PRIOR TO BEING RETURNED TO SERVICE. THE HEALTH AND SAFETY OF THE PUBLIC AND PLANT PERSONNEL WERE NOT EFFECTED BY THESE OCCURRENCES AS CPIS AND MONITOR 2RT-7856 REMAINED OPERABLE AT ALL TIMES.

[280] SAN ONOFRE 2 DOCKET 50-361 LER 87-016
FUEL HANDLING ISOLATION SYSTEM (FHIS) ACTUATIONS DUE TO SAMPLE CARTRIDGE ACTIVITY BUILDUP.
EVENT DATE: 091587 REPORT DATE: 101387 NSSS: CE TYPE: PWR

(NSIC 206650) ON 09/15/87, AT 1415, AND AGAIN ON 09/28/87, AT 1404, WITH UNIT 2 IN MODE 6 DURING REFUELING OPERATIONS, AN ACTUATION OF THE FUEL HANDLING ISOLATION SYSTEM (FHIS) OCCURRED WHEN AIRBORNE MONITORS 2RT-7822 AND 2RT-7823, RESPECTIVELY, REACHED THE PARTICULATE/IODINE CHANNEL HIGH LEVEL ACTUATION SETPOINT. THE CAUSE OF THE FHIS ACTUATIONS WAS A BUILDUP OF IODINE IN THE CHARCOAL ABSORBER SAMPLE CARTRIDGES OF THE FHIS MONITORS. THE SAMPLE CARTRIDGES ON 2RT-7822 AND 2RT-7823 WERE REPLACED. FOR BOTH ACTUATIONS, AFTER VERIFYING FUEL HANDLING BUILDING (FHB) RADIATION LEVELS WERE BELOW THE ACTUATION SETPOINT, THE FHIS WAS RESET/SECURED. ALL FHIS COMPONENTS FUNCTIONED AS DESIGNED. THE FREQUENCY FOR SAMPLE CARTRIDGE REPLACEMENT WAS INCREASED FROM WEEKLY TO DAILY FOLLOWING THE 9/15 ACTUATION, AND WAS FURTHER INCREASED TO SHIFTLY FOLLOWING THE 9/28 ACTUATION. ADDITIONALLY, THE SHIFTLY SURVEILLANCE PROCEDURE FOR THESE MONITORS WILL BE REVISED TO ENSURE CARTRIDGE REPLACEMENT IS INITIATED WHEN REQUIRED. THERE IS NO SAFETY SIGNIFICANCE TO THIS EVENT SINCE ALL FHIS COMPONENTS FUNCTIONED AS DESIGNED.

[281] SAN ONOFRE 2 DOCKET 50-361 LER 87-017
FUEL HANDLING ISOLATION SYSTEM (FHIS) SPURIOUS ACTUATIONS.
EVENT DATE: 091987 REPORT DATE: 101987 NSSS: CE TYPE: PWR
VENDOR: NUCLEAR MEASUREMENTS CORP.

(NSIC 206815) ON 09/19/87, AT 1505 AND 1544, AND ON 09/20/87, AT 0705, WITH THE REACTOR CORE OFFLOADED TO THE FUEL HANDLING BUILDING (FHB), THE FUEL HANDLING ISOLATION SYSTEM (FHIS) WAS ACTUATED BY FHIS MONITORS 2RT-7822 AND 2RT-7823. AFTER EACH ACTUATION, FHB RADIATION LEVELS WERE VERIFIED TO BE BELOW THE ACTUATION SETPOINT. INVESTIGATION INTO THE CAUSE OF THE FHIS ACTUATIONS REVEALED THAT CONDENSATION OF MOISTURE IN THE MONITOR SAMPLE LINES LED TO THE ACCUMULATION OF WATER WITHIN THE DETECTORS AND THE SPURIOUS ACTUATIONS. LACK OF HEAT TRACING OR DRAINAGE COLLECTION ON THE SAMPLE LINES ALLOWED WATER TO ENTER THE DETECTOR ASSEMBLY. BOTH DETECTORS WERE REPAIRED AND, FOLLOWING COMPLETION OF FUNCTIONAL TESTING, 2RT-7822 AND 2RT-7823 WERE DECLARED OPERABLE ON 09/21/87, AT 1830, AND

09/22/87, AT 1630, RESPECTIVELY. A DESIGN CHANGE TO THE FHIS MONITOR SAMPLE LINES WILL BE IMPLEMENTED TO ELIMINATE WATER ENTRY INTO THE DETECTOR ASSEMBLY. THERE IS NO SAFETY SIGNIFICANCE TO THIS EVENT SINCE ALL FHIS COMPONENTS FUNCTIONED AS DESIGNED.

[282] SAN ONOFRE 2 DOCKET 50-361 LER 87-018
CONTAINMENT PURGE ISOLATION SYSTEM (CPIS) SPURIOUS ACTUATIONS DUE TO INDUCED NOISE.
EVENT DATE: 100787 REPORT DATE: 110687 NSSS: CE TYPE: PWR

(NSIC 206938) ON 10/7/87, AT 0109, 0504, 0800 AND 0905, AND ON 10/17/87 AT 1853, WITH UNIT 2 SHUTDOWN FOR REFUELING AND WITH CONTAINMENT PURGE IN PROGRESS, SPURIOUS ACTUATIONS OF THE TRAIN "B" CONTAINMENT PURGE ISOLATION SYSTEM (CPIS) WERE INITIATED FROM CONTAINMENT AREA RADIATION MONITOR 2RT-7857. ON EACH OCCASION, CPIS COMPONENTS WERE VERIFIED TO HAVE OPERATED PROPERLY AND CONTAINMENT RADIATION LEVELS WERE VERIFIED TO BE BELOW THE ACTUATION SETPOINT. THE CAUSE OF THE ACTUATIONS HAS BEEN ATTRIBUTED TO INDUCED NOISE FROM WELDING, COMBINED WITH THE INCREASED SUSCEPTIBILITY OF THE INSTRUMENT DUE TO THE LOWERED SETPOINT REQUIRED DURING REFUELING OPERATIONS. WELDING GROUND LEADS HAD BEEN TERMINATED ON A BUILDING GROUND NEAR THE WELDING MACHINE RATHER THAN AT THE WELD PIECE. THIS CAN CAUSE VOLTAGE VARIANCES ON THE EQUIPMENT GROUND BUS, AND IN TURN, CAN AFFECT CPIS MONITOR CIRCUITRY WHICH IS GROUNDED TO THE EQUIPMENT GROUND BUS. SITE WELDERS ARE BEING INSTRUCTED TO ROUTE BOTH THE POSITIVE AND GROUND WELDING LEADS TO THE WELD LOCATION. CPIS INSTRUMENTS WILL BE MONITORED WITH WELDING ACTIVITIES IN PROGRESS TO CONFIRM THAT, BY CONNECTING WELDING MACHINE GROUND LEADS TO THE WELD LOCATIONS, NOISE INTERFERENCE IS REDUCED FROM THE LEVEL PREVIOUSLY OBSERVED. IF THIS ACTION IS EFFECTIVE, APPROPRIATE LONG TERM ACTIONS WILL BE TAKEN TO ENSURE WELDING GROUND LEADS ARE INVARIABLY CONNECTED AT WELD LOCATIONS.

[283] SAN ONOFRE 2 DOCKET 50-361 LER 87-019
CONTAINMENT AREA RADIATION MONITORS INOPERABLE DURING PURGE.
EVENT DATE: 100887 REPORT DATE: 110687 NSSS: CE TYPE: PWR

(NSIC 207099) ON OCTOBER 8, 1987, FROM 0315 TO 0359, WITH UNIT 2 IN MODE 6, CONTAINMENT WAS PURGED WITH NEITHER OF THE CONTAINMENT PURGE ISOLATION SYSTEM (CPIS) AREA RADIATION MONITORS OPERABLE, CONTRARY TO ACTION 17B OF TECHNICAL SPECIFICATION SECTIONS 3.3.2. AND 3.3.3.1. DISCUSSIONS WITH REPRESENTATIVES OF REGION VINDICATED THAT SUCH A PURGE WAS ACCEPTABLE AND SHOULD BE CONTROLLED AS A VOLUNTARY ENTRY INTO TECHNICAL SPECIFICATION 3.0.3 IN ACCORDANCE WITH PROCEDURES. AT 0915 ON OCTOBER 7, 1987, CPIS MONITOR 2RT-7857 WAS DECLARED INOPERABLE IN ORDER TO INVESTIGATE SPURIOUS ACTUATIONS ASSOCIATED WITH THE MONITOR. SINCE THE REDUNDANT CPIS MONITOR 2RT-7856 WAS ALREADY INOPERABLE DUE TO TROUBLE- SHOOTING OF THE MONITOR AND PERFORMANCE OF ITS 18 MONTH CALIBRATION, ISOLATION OF THE CONTAINMENT ATMOSPHERE, AS REQUIRED BY THE TECHNICAL SPECIFICATIONS, WAS ACCOMPLISHED. WITH THE CONTAINMENT ATMOSPHERE ISOLATED, INTERNAL PRESSURE SLOWLY INCREASED AS A RESULT OF MINOR LEAKAGE OF PRESSURIZED GAS SOURCES AND TEMPERATURE INCREASE INSIDE CONTAINMENT. THIS INCREASE IN PRESSURE CAUSED 1) THE WATER LEVEL IN THE SPENT FUEL POOL (SFP), WHICH WAS CONNECTED VIA THE OPEN FUEL TRANSFER TUBE, TO ALMOST REACH AN OVERFLOW CONDITION AND 2) THE REFUELING CAVITY WATER LEVEL TO DECREASE TO A POINT BELOW WHICH FUEL MOVEMENT IS PROHIBITED BY PROCEDURES.

[284] SAN ONOFRE 3 DOCKET 50-362 LER 87-014 REV 01
UPDATE ON SAFETY INJECTION TANK LEVEL INSTRUMENTATION.
EVENT DATE: 072587 REPORT DATE: 102887 NSSS: CE TYPE: PWR
VENDOR: FOXBORO CO., THE

(NSIC 206816) AT 1405 ON 7/25/87, WITH UNIT 3 AT 100% POWER, BOTH NARROW RANGE

CHANNELS OF SAFETY INJECTION TANK (SIT) T-009 LEVEL INDICATION WERE RETURNED TO SERVICE FOLLOWING COMPLETION OF REPAIR AND CALIBRATION CHECKS TO DETERMINE THE CAUSE OF INCONSISTENCIES BETWEEN THE LEVEL INDICATED BY EACH OF THE NARROW AND THE WIDE RANGE CHANNELS. BEFORE THEIR REMOVAL FROM SERVICE, THE NARROW RANGE CHANNELS INDICATED LEVELS EQUIVALENT TO 33 CUBIC FEET AND 7 CUBIC FEET ABOVE THE MINIMUM VOLUME REQUIRED BY TECHNICAL SPECIFICATION (TS) LIMITING CONDITION FOR OPERATION (LCO) 3.5.1; UPON RETURN TO SERVICE, THE INSTRUMENTS INDICATED LEVELS EQUIVALENT TO 17 CUBIC FEET AND 9 CUBIC FEET BELOW THE MINIMUM VOLUME REQUIRED. THE WIDE RANGE LEVEL INDICATION CHANNEL, HOWEVER, CONTINUED TO INDICATE LEVELS EQUIVALENT TO 26 CUBIC FEET ABOVE THE TS MINIMUM VOLUME. BASED ON THESE INDICATIONS, SIT T-009 WAS DECLARED INOPERABLE AND ACTION STATEMENT (A) OF LCO 3.5.1 WAS ENTERED, THE CONTAINED VOLUME WAS RESTORED TO THE REQUIRED LIMITS BY ALL THREE LEVEL INDICATIONS AND THE TANK WAS DECLARED OPERABLE AT 1501 ON 7/25/87. BASED ON THE NARROW RANGE LEVEL READINGS FOLLOWING CALIBRATION, THE MINIMUM VOLUME REQUIREMENT OF THE LCO MAY NOT HAVE BEEN MET FOR LONGER THAN THE HOUR PERMITTED BY THE ACTION STATEMENT, AND IS THEREFORE BEING REPORTED PURSUANT TO 10 CFR 50.73(A)(2)(I)(B).

[285] SAN ONOFRE 3 DOCKET 50-362 LER 87-016
MISSED CONTROL ELEMENT ASSEMBLY POSITION VERIFICATION DUE TO PERSONNEL ERROR.
EVENT DATE: 091787 REPORT DATE: 101687 NSSS: CE TYPE: PWR

(NSIC 206651) ON 9/17/87 AT 1036, WITH UNIT 3 AT 100% POWER, CORE PROTECTION CALCULATOR (CPC) CHANNEL "A" WAS REMOVED FROM SERVICE FOR THE INVESTIGATION AND REPAIR OF AN INTERMITTENT SENSOR FAILURE. IN ACCORDANCE WITH PROCEDURE, CONTROL ELEMENT ASSEMBLY CALCULATOR (CEAC) #1, ASSOCIATED WITH CPC CHANNEL "A", WAS ALSO REMOVED FROM SERVICE. WITH ONE CEAC INOPERABLE, THE POSITION OF EACH CONTROL ELEMENT ASSEMBLY (CEA) IS REQUIRED BY TECHNICAL SPECIFICATION TABLE 3.3-1 ACTION 6, TO BE VERIFIED WITHIN 7 INCHES OF ALL OTHERS IN ITS GROUP AT LEAST EVERY 4 HOURS. WHEN A PROCEDURAL STEP WAS MISINTERPRETED BY THE CONTROL OPERATOR, THIS ACTION REQUIREMENT WAS NOT INITIATED. THE CEAC WAS RETURNED TO SERVICE AT 1530. AT THAT TIME OPERATORS RECOGNIZED THAT THE CEAC HAD BEEN OUT OF SERVICE FOR MORE THAN 4 HOURS WITHOUT PERFORMING THE REQUIRED CEA POSITION VERIFICATIONS. THIS EVENT WAS CAUSED BY PROCEDURAL INADEQUACY AND PERSONNEL ERROR. THE CPC/CEAC OPERATING PROCEDURE DID NOT ADEQUATELY ALERT THE CONTROL OPERATOR TO PERFORM THE REQUIRED CEA VERIFICATIONS. THE CPC/CEAC PROCEDURE WILL BE REVISED TO CLARIFY REQUIRED ACTIONS WHEN REMOVING A CPC AND/OR CEAC FROM SERVICE. PERSONNEL ASSOCIATED WITH THIS EVENT HAVE BEEN COUNSELED AND THIS EVENT WILL BE DISCUSSED IN SHIFT BRIEFINGS WITH OPERATIONS PERSONNEL.

[286] SEABROOK 1 DOCKET 50-443 LER 87-017
TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENT NOT SATISFIED DUE TO PROCEDURE INADEQUACY.
EVENT DATE: 100787 REPORT DATE: 110687 NSSS: WE TYPE: PWR

(NSIC 206961) ON OCTOBER 7, 1987, DURING A REVIEW OF SURVEILLANCE PROCEDURES OX1456.24, IT WAS IDENTIFIED THAT THE PROCEDURE DID NOT SATISFY TECH SPEC SURVEILLANCE REQUIREMENTS. OPERABILITY OF THE SAFETY INJECTION ACCUMULATOR ISOLATION VALVE ENGINEERED SAFETY FEATURES SLAVE RELAY WAS THEREFORE NOT DEMONSTRATED PRIOR TO ENTERING MODE 4 ON FEBRUARY 10, 1987. FURTHER REVIEW INDICATED THAT A SIMILAR SITUATION EXISTED FOR PROCEDURES OX1456.23, OX1456.45, AND OX1456.56. THE AFFECTED VALVES ARE REQUIRED BY TECH SPECS TO HAVE THEIR POWER REMOVED WHILE THE UNIT IS IN MODES 1 THROUGH 5, THEREFORE, THESE VALVES ARE NOT STROKED AS PART OF THE SURVEILLANCE. REVIEW OF THIS SITUATION INDICATED THAT THE SURVEILLANCE SHOULD HAVE INCLUDED A CONTINUITY CHECK OF THE SLAVE RELAY CONTACTS AT THE ACTUATION DEVICE. THERE WERE NO ADVERSE CONSEQUENCES AS A RESULT OF THIS EVENT. THE VALVES ACTUATED BY THESE RELAYS ARE REQUIRED BY TECH SPECS TO HAVE THEIR POWER REMOVED WHILE IN THE MODES FOR WHICH THE ACTUATING DEVICES ARE REQUIRED TO BE OPERABLE. THEREFORE, THE INOPERABILITY OF THE ACTUATING DEVICES

REPRESENTS NO ACTUAL DEGRADATION OF ANY ENGINEERED SAFETY FEATURES. THE AFFECTED PROCEDURES ARE BEING REVISED TO INCLUDE A CONTINUITY CHECK AS PART OF THE SURVEILLANCE A REVIEW OF ALL ESF SLAVE RELAY TESTS.

[287] SEQUOYAH 1 DOCKET 50-327 LER 87-048
FAILURE OF SILICONE RUBBER INSULATED CABLES DURING TESTING.
EVENT DATE: 042287 REPORT DATE: 102687 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)
WATTS BAR 1 (PWR)
WATTS BAR 2 (PWR)
VENDOR: ANACONDA WIRE AND CABLE CO.
ROCKBESTOS COMPANY

(NSIC 206794) THIS REPORT IS BEING SUBMITTED AS A "VOLUNTARY LER" TO IDENTIFY A POTENTIAL PROBLEM WITH SILICONE RUBBER INSULATED CABLES AND TO KEEP NRC INFORMED OF ONGOING ACTIVITIES AT SEQUOYAH NUCLEAR PLANT. AN EMPLOYEE CONCERN WHICH ORIGINATED AT WATTS BAR NUCLEAR PLANT SUGGESTED THAT THE CABLE PULLING PRACTICES COULD HAVE DAMAGED THE CABLES AND COULD RESULT IN FAILURES. GENERAL CONSTRUCTION SPECIFICATION G-38, "INSTALLING INSULATED CABLES RATED UP TO 15,000 VOLTS, "RATED UP TO 15,000 VOLTS, "PREPARED THE MATERIALS AND PROCEDURES FOR INSTALLING, TERMINATING, SPLICING, AND MARKING FIELD-INSTALLED CABLES AT BOTH WATTS BAR NUCLEAR PLANT AND SEQUOYAH NUCLEAR PLANT. SUBSEQUENTLY, HIGH-POTENTIAL CABLE TESTS WERE CONDUCTED TO ASSESS DAMAGE ASSOCIATED WITH PULL-BYS, JAMMING, AND VERTICAL CABLE SUPPORTED BY 90 DEGREE CONDULETS. FAVORABLE RESULTS WERE OBTAINED FROM THE TESTS WITH THE EXCEPTION OF THE VERTICAL CABLES SUPPORTED BY 90 DEGREE CONDULETS. ON APRIL 22, 1987, WITH BOTH UNITS IN MODE 5 (COLD SHUTDOWN), 3 OUT OF 16 CONDUCTORS WHICH WERE CONSIDERED WORST-CASE VERTICAL DROP CABLES FAILED THE HIGH-POTENTIAL TEST. IT WAS DETERMINED AFTER LABORATORY TESTING THAT THE CABLES HAD NOT FAILED AT THE 90 DEGREE CONDULETS THAT SUPPORTED THE VERTICAL CABLE (HIGH-STRESS POINT); HOWEVER, IT WAS DISCOVERED THAT SILICONE RUBBER CABLES WERE MORE SUSCEPTIBLE TO IMPACT DAMAGE THAN EXPECTED.

[288] SEQUOYAH 1 DOCKET 50-327 LER 87-044 REV 01
UPDATE ON INADEQUATE COMMUNICATION BETWEEN DESIGN ORGANIZATIONS RESULTS IN POTENTIAL PROBLEM WITH 1E ELECTRICAL EQUIPMENT DUE TO UNANALYZED FLOODING EFFECTS.
EVENT DATE: 072487 REPORT DATE: 111987 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 207187) THIS LER IS BEING REVISED TO PROVIDE AN UPDATE TO THE CORRECTIVE ACTION SECTION. THIS LER IS BEING PROVIDED AS A VOLUNTARY REPORT TO INFORM NRC OF A POTENTIALLY GENERIC ISSUE CONCERNING POSTULATED BREAKS IN MODERATE ENERGY LINES. FLOODING EFFECTS OF MODERATE ENERGY LINE BREAKS (MELBS) WERE NOT ADEQUATELY ANALYZED DUE TO INADEQUATE COMMUNICATION BETWEEN THE PIPE BREAK ANALYST AND THE SYSTEM ENGINEERS RESPONSIBLE FOR PERFORMING THE ANALYSIS. AS A RESULT, TVA CONTRACTED WITH SARGENT AND LUNDY TO PERFORM A STUDY OF MELBS. SARGENT AND LUNDY HAS DETERMINED THAT FLOODING ASSOCIATED WITH MODERATE ENERGY LINE BREAKS (MELBS) COULD POTENTIALLY SUBMERGE AND IMPAIR COMPONENTS REQUIRED FOR SAFE SHUTDOWN AND THREATEN INTEGRITY OF CERTAIN STRUCTURES DUE TO INCREASED LOADING. ALL IDENTIFIED CONCERNS RESULTING FROM THE MELB FLOODING STUDY HAVE BEEN ADDRESSED BY TVA AND PREVIOUSLY REPORTED OR EVALUATED BY A REALISTIC ANALYSIS WHICH INDICATES NO FAILURES ARE LIKELY TO OCCUR. TO COMPLY WITH TVA INTERNAL DESIGN STANDARDS, CERTAIN ACTIONS (SUCH AS SEALING OF BUILDING WALLS) ARE BEING TAKEN TO MINIMIZE THE IMPACT OF POTENTIAL MELBS BEFORE RESTART OF UNIT 2. OTHER ACTIONS (SUCH AS PROTECTING CABLES FROM WATER) ARE BEING IMPLEMENTED BEFORE RESTART FROM FUEL CYCLE 5 FOR EITHER UNIT.

[289] SEQUOYAH 1 DOCKET 50-327 LER 87-060 REV 01
UPDATE ON INADVERTENT DIESEL GENERATOR START CAUSED BY INADEQUATE PROCEDURE DURING SHUTDOWN BOARD 1B-B MAINTENANCE.
EVENT DATE: 082787 REPORT DATE: 100987 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)
VENDOR: GENERAL ELECTRIC CO.
MORRISON-KNUDSON COMPANY, INC.
WOODWARD GOVERNOR COMPANY

(NSIC 206752) THIS REVISION PROVIDES ADDITIONAL INFORMATION CONCERNING THE FAILURE OF THE HYDRAULIC ACTUATOR ON DIESEL GENERATOR (D/G) 1A-A AND ALSO FULFILLS THE SPECIAL REPORTING REQUIREMENT FOR THE INOPERABLE FIRE PUMPS. ON AUGUST 27, 1987, AT 1140 EST WITH UNITS 1 AND 2 IN MODE 5 (COLD SHUTDOWN), AN INADVERTENT START OF THREE OF THE FOUR D/G UNITS (1A-A, 2A-A, AND 2B-B) OCCURRED DURING THE PERFORMANCE OF A SPECIAL MAINTENANCE INSTRUCTION (SMI). IN PERFORMING THE INSTRUCTION, AN ELECTRICIAN'S KNIFE BLADE INADVERTENTLY CAME INTO CONTACT WITH TWO BARE ENERGIZED STUDS AND PICKED UP THE TRIP COIL OF THE NORMAL FEEDER BREAKER FOR THE 1B-B SHUTDOWN BOARD. THE RESULTING TRIP OF THE SHUTDOWN BOARD SUPPLY BREAKER CAUSED A BLACKOUT CONDITION WHICH IN TURN STARTED ALL OF THE OPERABLE D/GS. THE 1B-B D/G DID NOT START BECAUSE IT WAS OUT OF SERVICE FOR MAINTENANCE. THE 1B-B SHUTDOWN BOARD WAS REENERGIZED AT 1149 EST, AND ALL OPERATING D/GS WERE SHUT DOWN AT 1153 EST. DURING SHUTDOWN OF THE 1A-A D/G, A MALFUNCTION OF THE HYDRAULIC ACTUATOR CAUSED THE DIESEL TO TRIP ON OVERSPEED. THE ROOT CAUSE OF THE D/G START WAS AN INADEQUATE PROCEDURE WHICH PLACED THE WORKERS IN A POSITION SUSCEPTIBLE TO ERROR. ALSO, THE INSTRUCTION DID NOT REQUIRE ISOLATING THE SHUTDOWN BOARD FROM THE DIESEL START CIRCUITRY.

[290] SEQUOYAH 1 DOCKET 50-327 LER 87-059
ALPHA CONTAMINATION CHECKS FOR TWO SEALED SOURCES NOT PERFORMED AS REQUIRED BY TECH SPEC DUE TO LESS THAN ADEQUATE ACCOUNTABILITY FOR A SURVEILLANCE REQUIREMENT.
EVENT DATE: 083187 REPORT DATE: 093087 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 206751) ON 8/31/87, AT 1300 EST WITH UNITS 1 AND 2 IN MODE 5, AN EXTENSIVE REVIEW OF PAST RECORDS, PLANT EQUIPMENT, AND STORAGE LOCATIONS, PERFORMED AS A RESULT OF SEALED SOURCE INVENTORY DEFICIENCIES IDENTIFIED AT TVA'S BROWNS FERRY NUCLEAR PLANT, REVEALED THAT SEALED SOURCE NO. 283N HAD BEEN USED BY THE INSTRUMENT MAINTENANCE SECTION FOR A SOURCE CHECK. THIS SOURCE, WHILE NORMALLY RESIDING IN THE UNIT 1 BORONOMETER, HAD NOT BEEN TESTED FOR ALPHA CONTAMINATION BEFORE USE AS REQUIRED BY TECH SPEC SURVEILLANCE REQUIREMENT (SR) 4.7.10.2.B. ON 9/9/87, IT WAS ALSO DISCOVERED THAT SEALED SOURCE NO. 662N HAD NOT BEEN TESTED FOR ALPHA CONTAMINATION ON THE PREVIOUS PERFORMANCE OF SURVEILLANCE INSTRUCTION (SI)-56, "BYPRODUCT MATERIAL INVENTORY AND SEALED SOURCE LEAK TEST UNITS 1 AND 2," AS REQUIRED BY TECH SPEC SR 4.7.10.2.A. FURTHER REVIEWS INDICATED THAT OTHER SEALED SOURCES HAD NOT BEEN PROPERLY INVENTORIED AND CONTROLLED BY SI-56. SI-56 IS CONTROLLED AND DOCUMENTED BY THE CHEMISTRY SECTION. HOWEVER, SOME BYPRODUCT MATERIAL SEALED SOURCES ARE ACCESSIBLE TO OTHER PLANT PERSONNEL. IN ADDITION, RADIOLOGICAL CONTROL PERFORMS THE ACTUAL CHECKS FOR LEAKAGE. THUS, WHILE PERSONNEL ERROR CONTRIBUTED TO THESE DEFICIENCIES, THE GOVERNING ROOT CAUSE WAS A LESS THAN ADEQUATE ACCOUNTABILITY FOR THE TS SR.

[291] SEQUOYAH 1 DOCKET 50-327 LER 87-061
APPENDIX R - ASSOCIATED CIRCUITS THAT SHARE A COMMON POWER SUPPLY WITH A APPENDIX R CIRCUIT LACKING SELECTIVE COORDINATION DUE TO IMPROPER CABLE LENGTHS USED IN IMPENDANCE CALCULATIONS.
EVENT DATE: 083187 REPORT DATE: 093087 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 206753) ON AUGUST 31, 1987, AT 0730 EST WITH UNITS 1 AND 2 IN MODE 5 (0

PERCENT POWER, 3 PSIG, 130 DEGREES F AND 0 PERCENT POWER, 330 PSIG, 130 DEGREES F, RESPECTIVELY), IT WAS DETERMINED DURING A CALCULATION REVIEW THAT, IN SEVERAL CASES, A FAULT ON APPENDIX R-ASSOCIATED CIRCUITS COULD CAUSE A REQUIRED CIRCUIT TO BE INTERRUPTED ON THE 125-VOLT DC VITAL BATTERY BOARDS AND 480-VOLT SHUTDOWN BOARDS. THIS WAS DUE TO IMPROPER FUSE/BREAKER SELECTIVE COORDINATION OF CIRCUITS WITH A COMMON POWER SUPPLY. THE DESIGN CABLE LENGTH VALVES WERE USED WHEN SIZING FUSES AND BREAKERS. THE ACTUAL CONSTRUCTED CABLE LENGTHS WERE NOT EVALUATED TO DETERMINE THE EFFECTS ON FUSE/BREAKER COORDINATION USED TO PROTECT REQUIRED CIRCUITS IN THE EVENT OF A FAULT. THE ACTUAL CONSTRUCTED CABLE LENGTHS WERE FOUND TO BE NONCONSERVATIVE. THE CIRCUITS IDENTIFIED HAVE BEEN EVALUATED, AND CORRECTIVE ACTIONS TO OBTAIN PROPER SELECTIVE COORDINATION WILL BE COMPLETED BEFORE RESTART. TO PREVENT RECURRENCE, PROCEDURE METHOD (PM) 87-26 (EEB), "CABLE LENGTH VALUES TO BE USED IN ELECTRICAL CALCULATIONS," HAS BEEN WRITTEN TO ENSURE DESIGN LENGTHS ARE NOT IMPROPERLY USED, AND SEQUOYAH ENGINEERING PROCEDURE (SQEP)-34, "PROCEDURE FOR IMPLEMENTATION OF THE ELECTRICAL FUSE TABULATION," HAS BEEN WRITTEN TO PROVIDE FUSE COORDINATION.

[292] SEQUOYAH 1 DOCKET 50-327 LER 87-062
UNPLANNED LOSS OF MANHOLE 7B MISSILE PROTECTION FOR B TRAIN DIESEL GENERATOR CABLES DUE TO A PROGRAMMATIC DEFICIENCY IN THE DESIGN CHANGE PROCESS.
EVENT DATE: 091187 REPORT DATE: 100987 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 206754) ON SEPTEMBER 11, 1987, AT 1345 EST, WITH UNITS 1 AND 2 IN MODE 5, AN UNPLANNED LOSS OF TORNADO GENERATED MISSILE PROTECTION FOR CABLES ASSOCIATED WITH DIESEL GENERATORS (D/GS) 1B-B AND 2B-B WAS IDENTIFIED. THIS CONDITION RESULTED FROM THE EXCAVATION OF EARTH, AND THE CUTTING OF A HOLE IN THE SIDE OF MANHOLE (MH) 7B FOR THE INSTALLATION OF ADDITIONAL CONDUITS AS PART OF AN APPROVED MODIFICATION. IT WAS ALSO NOTED THAT THE TIME THAT COVERS WERE INSTALLED ON MH 7B WHICH DID NOT MEET MISSILE PROTECTION REQUIREMENTS. UPON DETERMINING THAT THE MISSILE PROTECTION FOR MH 7B HAD BEEN COMPROMISED, THE UNIT 1 AND UNIT 2 B TRAIN D/GS WERE DECLARED INOPERABLE, AND THE WORKPLAN FOR THE INVOLVED MODIFICATION WAS CHANGED TO REQUIRE A STEEL PLATE TO BE PLACED OVER THE HOLE IN MH 7B. FOLLOWING THE INSTALLATION OF THE PLATE OVER THE HOLE IN MH 7B, AND THE INSTALLATION OF THE PERMANENT MANHOLE COVERS, THE UNIT 1 AND UNIT 2 B TRAIN D/GS WERE DECLARED OPERABLE. ON OCTOBER 2, 1987, IT WAS DETERMINED THAT THE EXCAVATED PORTION OF THE CONCRETE WALL FOR MH 7B DID NOT PROVIDE ADEQUATE MISSILE PROTECTION WITHOUT AN EARTH COVER, AND THAT THE TEMPORARY PLATE COVERING THE HOLE WAS INADEQUATE BECAUSE OF AN UNACCEPTABLE MOUNTING.

[293] SEQUOYAH 1 DOCKET 50-327 LER 87-063
CONTROL ROOM ISOLATION CAUSED BY SIMULATENOUS CONTACT OF TWO TERMINALS WITH AN OPEN ALLIGATOR CLIP DEFEATING BLOCK FUNCTION DURING RADIATION MONITOR TESTING.
EVENT DATE: 092487 REPORT DATE: 102287 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 206857) ON SEPTEMBER 24, 1987, AT 1057 EST WITH UNITS 1 AND 2 IN MODE 5 (0 PERCENT POWER, 3 PSIG, 120 DEGREES F AND 0 PERCENT POWER, 65 PSIG, 119 DEGREES, F, RESPECTIVELY), AN INADVERTENT CONTROL ROOM VENTILATION ISOLATION WAS INITIATED. THIS WAS CAUSED BY AN ALLIGATOR CLIP ACCIDENTALLY CONTACTING TWO TERMINAL POINTS SIMULTANEOUSLY DURING PERFORMANCE OF A RADIATION MONITOR FUNCTIONAL TEST. THIS DEFEATED THE BLOCK FUNCTION ON THE MAIN CONTROL ROOM INTAKE MONITOR 0-RM-90-126 AND INITIATED A HIGH RADIATION SIGNAL. THE HIGH RADIATION SIGNAL THEN GENERATED A MAIN CONTROL ROOM ISOLATION. UPON DETERMINATION THAT THE SIGNAL WAS INVALID, THE VENTILATION SYSTEM WAS RESET AND AN INVESTIGATION TO DETERMINE THE CAUSE WAS PERFORMED. DURING TROUBLESHOOTING TO DETERMINE THE CAUSE OF THE ISOLATION, A PREPLANNED ISOLATION OCCURRED INDICATING THE HIGH RADIATION SIGNAL WAS GENERATED BY AN OPEN ALLIGATOR CLIP MAKING CONTACT WITH TWO TERMINALS. SURVEILLANCE INSTRUCTION (SI)-82, "FUNCTIONAL TESTS FOR THE

RADIATION MONITORING SYSTEM," WILL BE CHANGED TO ENSURE "ALLIGATOR CLIPS ARE NOT USED FOR THIS APPLICATION AND LEADS ARE NOT ATTACHED TO TERMINALS. SINCE OTHER RADIATION MONITORS HAVE SIMILAR ARRANGEMENTS, THE SI-82 REVISION WILL ENSURE ALL RADIATION MONITORS ARE ADDRESSED.

[294] SEQUOYAH 1 DOCKET 50-327 LER 87-058
INADVERTENT DIESEL GENERATOR START DUE TO NOT RENDERING DIESEL PHYSICALLY INOPERABLE BEFORE PULLING FUSES.
EVENT DATE: 092687 REPORT DATE: 102387 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 206795) ON SEPTEMBER 26, 1987, AT 0455 EST WITH UNITS 1 AND 2 IN MODE 5 (0 PERCENT POWER, 3 PSIG, 121 DEGREES F AND 0 PERCENT POWER, 58 PSIG, 118 DEGREES F, RESPECTIVELY), A CONTROL ROOM OPERATOR NOTED A START OF 2B-B DIESEL GENERATOR (D/G). THIS WAS CAUSED BY PERSONNEL FAILING TO RENDER THE DIESEL PHYSICALLY INOPERABLE BEFORE PULLING THE FUSES IN CIRCUIT C-3 ON VITAL BATTERY BOARD IV WHICH GOES TO 2B-B SHUTDOWN BOARD LOGIC PANEL. WHEN THE C-3 FUSES WERE PULLED, THIS DROPPED OUT THE ES2BY RELAY WHICH STARTED THE 2B-B D/G. THE ROOT CAUSE OF THIS EVENT WAS INADEQUATE PREPLANNING OF THE WORK TO BE PERFORMED. THE CLEARANCE REQUEST DID PREVENT STARTING OF THE OTHER THREE D/GS; HOWEVER, IT DID NOT ADEQUATELY PREVENT START OF THE 2B-B D/G. UPON DISCOVERY, THE C-3 FUSES WERE THEN REPLACED TO ALLOW A NORMAL STOP OF THE D/G. AFTER THE D/G WAS SHUT DOWN, IT WAS PHYSICALLY PLACED INOPERABLE AND TAGGED OUT. THE CIRCUIT C-3 FUSES WERE THEN REPULLED AND TAGGED WHICH AGAIN DROPPED OUT THE ES2BY RELAY BUT DID NOT START THE D/G. A MEMORANDUM WILL BE ISSUED TO OPERATIONS PERSONNEL INFORMING THEM OF THE EVENT AND EMPHASIZING THE IMPORTANCE OF ADEQUATE PREPLANNING FOR TAGGING AND REMOVING EQUIPMENT FROM SERVICE.

[295] SEQUOYAH 2 DOCKET 50-328 LER 86-011 REV 02
UPDATE ON CONTAINMENT VENTILATION ISOLATION FROM ELECTROMAGNETIC INTERFERENCE ON A RADIATION MONITOR DUE TO A PERSONNEL ERROR.
EVENT DATE: 120786 REPORT DATE: 102187 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 1 (PWR)
VENDOR: GENERAL ATOMIC CO.

(NSIC 206863) THIS LER IS BEING REVISED TO CHANGE THE SCOPE OF THE CORRECTIVE ACTION AND THE CORRECTIVE ACTION DUE DATE. ON DECEMBER 7, 1986, AT 0820 CST WITH UNIT 2 IN MODE 5 (0 PERCENT POWER, 330 PSIG, 122 DEGREES F), A CONTAINMENT VENTILATION ISOLATION (CVI) OCCURRED. OPERATIONS PERSONNEL WERE RESPONDING TO AN INSTRUMENT MALFUNCTION ALARM FOR LOW SAMPLE LINE FLOW THROUGH THE CONTAINMENT BUILDING LOWER COMPARTMENT AIR MONITOR 2-RM-90-106 AND WERE ADJUSTING THE SAMPLE LINE THROTTLE VALVE ON THE RADIATION MONITOR (RM) WHEN THE CVI OCCURRED. ADJUSTING THE SAMPLE LINE THROTTLE VALVE CAUSED THE SAMPLE LINE FLOW SWITCH TO CHATTER GENERATING ELECTROMAGNETIC INTERFERENCE (EMI) WHICH INITIATED A HIGH RADIATION SIGNAL. FURTHER INVESTIGATION REVEALED THAT THE LOW FLOW ALARM HAD BEEN CAUSED BY A PARTIALLY CLOGGED CHARCOAL FILTER AND A PARTICULATE ROLL FILTER PAPER THAT HAD RUN OUT. CHEMICAL LABORATORY PERSONNEL USE TECHNICAL INSTRUCTION (TI)-16, "SAMPLING METHODS," FOR REPLACING THE FILTERS; A CONTRIBUTING CAUSE TO THIS EVENT COULD BE THE LACK OF ARC SUPPRESSION EQUIPMENT ON THE RM. A DESIGN CHANGE REQUEST HAS BEEN APPROVED TO INSTALL ARC SUPPRESSION EQUIPMENT ON THE RM WHICH SHOULD REDUCE CVIS CAUSED BY EMI. CHEMICAL LABORATORY PERSONNEL HAVE BEEN COUNSELED ON THE IMPORTANCE OF REPLACING FILTERS IN A TIMELY MANNER.

[296] SHEARON HARRIS 1 DOCKET 50-400 LER 87-053
PLANT OPERATED IN AN UNANALYZED CONDITION DUE TO A FAILED OPEN BLOWDOWN ISOLATION VALVE.
EVENT DATE: 081387 REPORT DATE: 110687 NSSS: WE TYPE: PWR
VENDOR: ITT HAMMEL DAHL CONOFLOW

(NSIC 207124) THE PLANT WAS OPERATING IN MODE 1, POWER OPERATION, AT 100 PERCENT REACTOR POWER ON AUGUST 13, 1987. STEAM GENERATOR 1A BLOWDOWN CONTAINMENT ISOLATION VALVE, 1BD-11 WAS DECLARED INOPERABLE AT 1030 HOURS AFTER FAILING AN INSERVICE INSPECTION TEST. THE VALVE WAS OPEN AND COULD NOT BE CLOSED EITHER BY THE CONTROL SWITCH OR LOCALLY. PLANT OPERATION CONTINUED UNDER THE TECHNICAL SPECIFICATIONS AFTER THIS VALVE FAILED OPEN. CONTINUED OPERATION WAS JUSTIFIED AT THE TIME THROUGH THE INTERPRETATION OF A FOOTNOTE INCLUDED IN THE LOW POWER TECHNICAL SPECIICATIONS AND NOW INCLUDED IN A PLANT PROCEDURE. FURTHER EVALUATION BY PLANT STAFF IDENTIFIED AN UNANALYZED CONDITION HAD BEEN INTRODUCED IN REGARDS TO ENSURING ADEQUATE STEAM GENERATOR WATER LEVEL UNDER ACCIDENT SCENARIOS WITH CERTAIN SINGLE FAILURES AND THIS VALVE OPEN. ON SEPTEMBER 11, 1987 AT 1720 HOURS, IT WAS DETERMINED TO SHUT DOWN THE PLANT TO MODE 3, HOT STANDBY, AND REPAIR THE VALVE. THE REPAIR WAS COMPLETED AT 1600 HOURS ON SEPTEMBER 13 AND THE PLANT RETURNED TO SERVICE AT 1712 HOURS ON SEPTEMBER 14. THE CAUSE OF THE VALVE FAILURE WAS BINDING OF THE CARBON SEAL RINGS BETWEEN THE INSIDE DIAMETER OF THE VALVE CAGE AND THE VALVE PLUG. THE IMMEDIATE CORRECTIVE ACTION UPON DISCOVERY OF THE UNANALYZED CONDITION WAS TO SHUT DOWN THE PLANT AND REPAIR THE VALVE.

[297] SHEARON HARRIS 1 DOCKET 50-400 LER 87-050
TECHNICAL SPECIFICATION REQUIRED SHUTDOWN FROM MODE 1 TO MODE 3 DUE TO REACTOR COOLANT SYSTEM UNIDENTIFIED LEAKAGE GREATER THAN 1 GPM.
EVENT DATE: 083187 REPORT DATE: 093087 NSSS: WE TYPE: PWR
VENDOR: COPES-VULCAN, INC.
KEROTEST MANUFACTURING CORP.
WESTINGHOUSE ELECTRIC CORP.

(NSIC 206692) ON AUGUST 30, 1987, THE SHEARON HARRIS NUCLEAR POWER PLANT WAS OPERATING IN MODE 1 AT 100% POWER. AT 2030, THE REACTOR COOLANT SYSTEM (RCS) UNIDENTIFIED LEAKAGE EXCEEDED 1 GALLON PER MINUTE (GPM), THE TECHNICAL SPECIFICATION LIMIT. THE RCS UNIDENTIFIED LEAK RATE COULD NOT BE REDUCED BELOW 1 GPM IN 4 HOURS, AS ALLOWED BY TECHNICAL SPECIFICATION 3.4.6.2.B, AND PLANS WERE MADE TO REDUCE POWER TO HOT STANDBY. AN UNUSUAL EVENT WAS DECLARED AT 0030 ON AUGUST 31, 1987 DUE TO A TECHNICAL SPECIFICATION REQUIRED SHUTDOWN FROM MODE 1. ON AUGUST 31, A CONTAINMENT ENTRY WAS MADE TO INSPECT FOR LEAKAGE. THE INSPECTION SHOWED EVIDENCE OF ONLY MINOR LEAKS DUE TO NORMAL WEAR. THE LEAKAGE WAS REDUCED BY VALVE PACKING REPAIRS. REACTOR STARTUP COMMENCED AT 0327 ON SEPTEMBER 1, 1987.

[298] SHEARON HARRIS 1 DOCKET 50-400 LER 87-052 REV 01
UPDATE ON CONTAINMENT INTEGRITY BREACH CAUSED BY PERSONNEL OPENING INNER PERSONNEL ACCESS DOOR WHILE OUTER DOOR WAS DECLARED INOPERABLE DUE TO O-RING SEAL.
EVENT DATE: 083187 REPORT DATE: 110287 NSSS: WE TYPE: PWR
VENDOR: WOOLLEY, W. J. COMPANY

(NSIC 206677) THE PLANT WAS IN MODE 3, HOT STANDBY, AT 0% POWER ON 8/31/87 WHEN A MECHANIC AND A HEALTH PHYSICS (HP) TECHNICIAN WERE EXITING CONTAINMENT THROUGH PERSONNEL ACCESS HATCH. AS THEY OPENED THE OUTER DOOR, ONE OF THE DOOR'S O-RING SEAL FELL OUT OF ITS GROOVE. A MECHANIC NOTICED THERE WAS NO APPARENT DAMAGE TO THE O-RING SO HE RE-INSTALLED IT PRIOR TO CLOSING THE DOOR. MEANWHILE, A HP TECHNICIAN ON DUTY AT THE DOOR NOTIFIED THE SHIFT FOREMAN OF THE SITUATION, WHO IMMEDIATELY DECLARED THE PERSONNEL ACCESS HATCH INOPERABLE AND MADE AN ANNOUNCEMENT ANY PERSONNEL IN CONTAINMENT WHO NEED TO EXIT TO USE THE EMERGENCY ACCESS HATCH WHILE REPAIRS WERE BEING MADE TO THE DOOR. SHORTLY THEREAFTER, ANOTHER MECHANIC AND HP TECHNICIAN, WHO WERE INSIDE CONTAINMENT OPENED THE INNER DOOR TO EXIT CONTAINMENT WHILE THE OUTER DOOR WAS STILL DECLARED INOPERABLE. THE INNER DOOR WAS OPENED FOR APPROXIMATELY ONE MINUTE. THIS VIOLATED CONTAINMENT INTEGRITY AND TECH SPECS 3.6.1.3 ACTION A. THE IMMEDIATE CORRECTIVE ACTION WAS THAT A WORK REQUEST WAS ISSUED TO REPAIR THE DOOR . THE O-RING WAS INSPECTED AND FOUND TO BE IN SATISFACTORY CONDITION AND REINSTALLED. LOCAL LEAK RATE WAS THEN

SUCCESSFULLY COMPLETED AT 1309 HOURS AND THE AIRLOCK DECLARE OPERABLE. THE CAUSE OF THE EVENT IS PERSONNEL ERROR DUE TO COMMUNICATION BREAKDOWN.

[299] SHEARON HARRIS 1 DOCKET 50-400 LER 87-054
UNANALYZED CONDITION - ISOLATION OF AUXILIARY FEEDWATER CAUSED BY LOSS OF VITAL DC BUS 1B-SB COINCIDENT WITH LOSS OF OFF-SITE POWER.
EVENT DATE: 091587 REPORT DATE: 101587 NSSS: WE TYPE: PWR

(NSIC 206693) ON SEPTEMBER 15, 1987, AT 2000, WITH THE PLANT IN MODE 1 AT 100% POWER, IT WAS DETERMINE THAT A MORE LIMITING SINGLE FAILURE THAN HAD BEEN PREVIOUSLY ANALYZED MIGHT EXIST, AND THAT IF VALID, CONTINUED OPERATION OF THE PLANT WAS NOT JUSTIFIED. A PLANT SHUTDOWN COMMENCED AT 2015, AND THE UNIT WAS OFF-LINE AT 0040 ON SEPTEMBER 16, 1987. THE UNANALYZED FAILURE INVOLVED A LOSS OF "B" VITAL DC BUS CAUSING A FAILURE OF THE TURBINE-DRIVEN AUXILIARY FEEDWATER (AFW) PUMP AND THE "B" MOTOR-DRIVEN AFW PUMP. THE FINAL SAFETY ANALYSIS REPORT (FSAR) ACCIDENT ANALYSIS FOR MAIN FEEDWATER LINE BREAK ASSUMES THE AVAILABILITY OF BOTH MOTOR-DRIVEN PUMPS "A" AND "B". THE PLANT WAS COOLED DOWN TO MODE 4 AT 0440 ON SEPTEMBER 16, 1987, WHERE AUXILIARY FEEDWATER IS NOT REQUIRED TO BE OPERABLE BY TECHNICAL SPECIFICATIONS. ON SEPTEMBER 16, 1987, TWO ADDITIONAL POTENTIAL FAILURE MODES WERE IDENTIFIED; ONE INVOLVED THE SPURIOUS FAILURE OF A RELAY IN THE SOLID STATE PROTECTION SYSTEM (SSPS) CAUSING INADVERTENT ISOLATION OF AFW TO ONE STEAM GENERATOR. THE SECOND WAS THAT FAILURE OF "B" VITAL DC BUS COINCIDENT WITH A LOSS OF OFF- SITE POWER WOULD ISOLATE AFW TO ALL THREE STEAM GENERATORS REANALYSIS OF ACCIDENTS WITH A REDUCED AFW CAPABILITY WAS DONE.

[300] SHEARON HARRIS 1 DOCKET 50-400 LER 87-055
CONTAINMENT INTEGRITY BREACH CAUSED BY PERSONNEL OPENING INNER PERSONNEL ACCESS HATCH DOOR WHILE OUTER DOOR WAS OPEN.
EVENT DATE: 091787 REPORT DATE: 101987 NSSS: WE TYPE: PWR

(NSIC 206680) THE PLANT WAS IN MODE 4, HOT SHUTDOWN, AT 0 PERCENT REACTOR POWER ON SEPTEMBER 17, 1987. PLANT PERSONNEL HAD EXITED THE CONTAINMENT BUILDING THROUGH THE PERSONNEL ACCESS HATCH. AS THE OUTER DOOR WAS IN THE PROCESS OF CLOSING, THE DOORS' LOCKING RING DID NOT ENGAGE TO SHUT THE DOOR AFTER COMING IN CONTACT WITH THE DOOR SEALS. THIS RESULTED IN THE DOOR BEING PARTIALLY OPEN AND NOT LOCKED CLOSED AS REQUIRED. HEALTH PHYSICS PERSONNEL WHO WERE IN CONTAINMENT TRYING TO EXIT COULD NOT, AS THE INNER ACCESS DOOR WOULD NOT RESPOND TO THE NORMAL OPENING SEQUENCE WHILE THE OUTER DOOR IS NOT LOCKED CLOSED. AT 1415 HOURS, THE PERSONNEL WHO WERE IN CONTAINMENT PUSHED THE EMERGENCY STOP AND MANUALLY PUMPED THE INSIDE DOOR OPEN. BOTH PERSONNEL ACCESS HATCH DOORS WERE OPEN FOR APPROXIMATELY TWO MINUTES WHILE PERSONNEL EXITED. THIS VIOLATED CONTAINMENT INTEGRITY AND TECHNICAL SPECIFICATIONS 3.6.13 ACTION A. PERSONNEL EXITING NOTICED THE OUTER DOOR WAS OPEN AND NOTIFIED OPERATIONS PERSONNEL. ALL DOOR INTERLOCKS AND INDICATING LIGHTS WERE CHECKED FOR PROPER FUNCTION AND ALL CONDITIONS WERE FOUND NORMAL IMMEDIATELY AFTER INCIDENT OCCURRED. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR DUE TO AN INCOMPLETE UNDERSTANDING OF THE PERSONNEL ACCESS DOORS INDICATING SYSTEM.

[301] SHEARON HARRIS 1 DOCKET 50-400 LER 87-056
REACTOR TRIP CAUSED BY THE DE-ENERGIZATION OF P-13 PERMISSIVE BISTABLE DUE TO INSUFFICIENT MODIFICATION INSTALLATION INSTRUCTIONS.
EVENT DATE: 092487 REPORT DATE: 102687 NSSS: WE TYPE: PWR

(NSIC 206824) THE PLANT WAS IN MODE 4, HOT SHUTDOWN, AT 0 PERCENT REACTOR POWER ON 9/24/87. WORK WAS IN PROGRESS TO IMPLEMENT A PLANT CHANGE REQUEST WHICH MODIFIED THE CHANNEL IV INPUT TO AUXILIARY FEEDWATER ISOLATION LOGIC FROM PROCESS INSTRUMENTATION. LOSS OF CARD FRAME 8 PLACES ALL LOOPS IN PIC CABINET 4 IN TEST. OPERATIONS AND INSTRUMENT AND CONTROL PERSONNEL REVIEWED APPLICABLE DRAWINGS AND

CHECKED FOR INSTRUMENTS WHICH WOULD BE DISABLED AND ALSO VERIFIED OTHER PROTECTION CHANNELS WERE NOT IN TEST OR TRIP CONDITION TO PREVENT AN OCCURRENCE OF 2 OUT OF 3 OR 2 OUT OF 4 LOGIC. WHEN THE FUSES IN CARD FRAME 8 WERE REMOVED, BISTABLE PS-447E DE-ENERGIZED INDICATING TURBINE IMPULSE PRESSURE HAD EXCEEDED 10 PERCENT POWER LEVEL AND REMOVED THE P-13 INPUT TO P-7. SINCE THE TURBINE STOP VALVES WERE ALREADY CLOSED, LOGIC WAS COMPLETED TO REMOVE P-7 WHICH RESULTED IN A REACTOR TRIP AT 0114 HOURS. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR BASED ON INSUFFICIENT MODIFICATION INSTALLATION INSTRUCTIONS ON WHAT WOULD RESULT FROM DE-ENERGIZING CARD FRAME 8. CORRECTIVE ACTION/ACTION TO PREVENT RECURRENCE IS THAT THE PLANT PROCEDURE FOR MODIFICATION IMPLEMENTATION HAS BEEN REVISED TO PROVIDE MORE EXTENSIVE GUIDANCE IN THE OF INSTALLATION INSTRUCTIONS.

[302] SHEARON HARRIS 1 DOCKET 50-400 LER 87-049
AUXILIARY FEEDWATER ACTUATION DUE TO FAILURE OF MAIN FEEDWATER PUMP RECIRCULATION VALVE.
EVENT DATE: 092587 REPORT DATE: 102687 NSSS: WE TYPE: PWR
VENDOR: MASONEILAN INTERNATIONAL, INC.

(NSIC 206823) ON SEPTEMBER 25, 1987, WITH THE PLANT IN MODE 3 AT 475 F, THE OPERATING MAIN FEEDWATER PUMP TRIPPED AT 0930 ON LOW FLOW. THE MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS AUTOMATICALLY STARTED AS DESIGNED. AT 0955, THE STANDBY MAIN FEEDWATER PUMP WAS STARTED AND AUXILIARY FEEDWATER WAS SECURED. STEAM GENERATOR LEVELS WERE MAINTAINED THROUGHOUT THE EVENT. THE MAIN FEEDWATER PUMP RECIRCULATION VALVE STEM WAS FOUND SEPARATED FROM THE OPERATOR, CAUSING THE VALVE TO PARTIALLY CLOSE. THIS RECIRCULATION FLOWPATH IS REQUIRED TO MAINTAIN PUMP FLOW ABOVE THE MINIMUM REQUIRED FOR SAFE OPERATION OF THE PUMP, AND WHEN THE VALVE FAILED PARTIALLY CLOSE, THE PUMP TRIPPED ON LOW FLOW. WHEN REPAIRS WERE PERFORMED, THE STEM WAS ALSO FOUND TO BE BROKEN. THE CAUSE OF THE FAILURE WAS ATTRIBUTED TO HIGH VIBRATION IN THE RECIRCULATION PIPING. THE VALVE STEM WAS REPLACED AND THE RECIRCULATION VALVE RETURNED TO SERVICE ON SEPTEMBER 25, 1987. A PLANT MODIFICATION TO THE RECIRCULATION PIPING TO INSTALL BREAKDOWN ORIFICES TO REDUCE THE PRESSURE DROP IN THE PIPING IS TO BE IMPLEMENTED BY NOVEMBER 4, 1987 TO CORRECT THE VIBRATION PROBLEM.

[303] SHEARON HARRIS 1 DOCKET 50-400 LER 87-057
FAILURE TO IMPLEMENT ALL REQUIRED IN-SERVICE INSPECTION TESTS FOR DIESEL FUEL OIL TRANSFER PUMPS.
EVENT DATE: 092987 REPORT DATE: 102887 NSSS: WE TYPE: PWR

(NSIC 206825) ON SEPTEMBER 29, 1987, WITH THE PLANT IN MODE 1 AT 100% POWER, A REVIEW OF SURVEILLANCE TEST PROCEDURES COVERING THE DIESEL GENERATOR FUEL OIL TRANSFER PUMPS REVEALED A DEFICIENCY, IN THAT THE REQUIREMENTS OF THE IN-SERVICE INSPECTION PROGRAM WERE NOT FULLY IMPLEMENTED BY THE TEST PROCEDURES. THE PROGRAM REQUIRES PERIODIC PUMP DIFFERENTIAL PRESSURE MEASUREMENTS, AND REQUIRES PUMP FLOW BE ESTIMATED BY CHANGES IN THE LEVEL OF FUEL OIL IN THE TANK TO WHICH THE PUMP DISCHARGES. THE TEST PROCEDURES DID NOT RECORD THE DIFFERENTIAL PRESSURE READINGS, AND THE PUMP FLOWS WERE BEING DETERMINED USING AN ULTRASONIC FLOW METER. THE CAUSE OF THE PROCEDURAL DEFICIENCIES WAS THE INTERPRETATION THAT RELIEF GRANTED BY THE NRC REGARDING FLOW MEASUREMENTS FOR THOSE PUMPS ALLOWED CHANGES TO THE TEST METHODOLOGY AND PERMITTED ELIMINATION OF THE DIFFERENTIAL PRESSURE MONITORING REQUIREMENT. THE TEST PROCEDURES WERE REVISED TO COMPLY WITH THE IN-SERVICE INSPECTION PROGRAM, AND THE NEW PROCEDURES WERE RUN SATISFACTORILY ON OCTOBER 2, 1987. AN AUDIT OF THE PROGRAM AND THE IMPLEMENTING PROCEDURES WAS CONDUCTED TO DETERMINE IF OTHER NON-COMPLIANCES EXISTED, AND NO OTHER DISCREPANCIES WERE FOUND.

[304] SHEARON HARRIS 1 DOCKET 50-400 LER 87-061
TECHNICAL SPECIFICATION VIOLATION DUE TO MISSED FLOW RATE ESTIMATE CAUSED BY PERSONNEL ERROR.
EVENT DATE: 101687 REPORT DATE: 111287 NSSS: WE TYPE: PWR

(NSIC 207107) THE PLANT WAS IN MODE 5, COLD SHUTDOWN, AT 0 PERCENT REACTOR POWER ON 10/16/87. TECHNICAL SPECIFICATIONS (TS) 3.3.3.11 TABLE 3.3-13 ACTION 46 REQUIRES THAT FLOW RATES FROM WASTE PROCESSING BUILDING (WPB) VENT STACKS 5 AND 5A BE ESTIMATED ONCE PER FOUR HOURS WHEN THE FLOW RATE INSTRUMENT IS INOPERABLE. ON 10/16/87 THE FLOW RATE INSTRUMENTS FOR WPB VENT STACK 5 AND 5A WERE DECLARED INOPERABLE. FLOW RATE ESTIMATES WERE MADE AT 1200 HOURS AND AT 2000 HOURS, A PERIOD OF EIGHT HOURS. FLOW RATE ESTIMATES WERE NOT RECORDED FOR THE 1600 HOUR READING AS REQUIRED BY TS. THE MISSED READING WAS DISCOVERED BY RADWASTE PERSONNEL DURING A REVIEW OF THE LOG SHEETS. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR ON THE PART OF THE RADWASTE CONTROL OPERATOR AS HE FAILED TO RECORD THE 1600 HOUR FLOW RATE ESTIMATE ON THE LOG SHEET. CORRECTIVE ACTIONS INCLUDE A) A MEETING WAS HELD FOR ALL SHIFT FOREMAN TO DISCUSS THIS AND SIMILAR EVENTS, B) THE OPERATOR INVOLVED WAS COUNSELED ON THIS ITEM, C) THE STACK FLOW RATE ESTIMATES ARE NOW LOGGED AT TWO HOUR INTERVALS, D) A COPY OF THIS LER WILL BE ROUTED TO ALL RADWASTE FOREMAN TO BE REVIEWED BY EACH SHIFT FOREMAN, AND E) PROBLEMS WITH STACK FLOW INSTRUMENTS WILL BE RESOLVED SUCH THAT STACK FLOW ESTIMATES ARE NO LONGER REQUIRED TO BE LOGGED. THIS EVENT IS BEING REPORTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(I)(B) AS A VIOLATION OF TECH SPECS.

[305] SHOREHAM DOCKET 50-322 LER 87-029
RESULTS OF LLRT OF PENETRATION SHOWED LEAKAGE THAT, WHEN COMBINED WITH ALL TYPE B AND C PENETRATIO L[illegible]ES, EXCEEDED THE TECH SPEC LIMIT OF 0.6 LA.
EVENT DATE: 09038[illegible] REPORT DATE: 100287 NSSS: GE TYPE: BWR
VENDOR: VELAN VALVE CORP.

(NSIC 206595) ON SEPTEMBER 8, 1987 AT 1330, IT WAS DETERMINED BY THE MAINTENANCE SECTION THAT THE RESULTS OF A LOCAL LEAK RATE TEST (LLRT) OF CHECK VALVE 1T48*01V-0016A (BD) (POST ACCIDENT SAMPLING CONTAINMENT ATMOSPHERE SAMPLE RETURN), SHOWED LEAKAGE THAT, WHEN COMBINED WITH ALL "B" AND "C" PENETRATION LEAKAGES EXCEEDED THE TECH. SPEC. LIMIT OF 0.8 LA. THE PLANT WAS IN OPERATIONAL CONDITION 4 (COLD SHUTDOWN) WITH THE MODE SWITCH IN SHUTDOWN AND ALL RODS INSERTED IN THE CORE. THE PENETRATION WAS TESTED ON SEPTEMBER 3, 1987 TO SATISFY TECH. SPEC. REQUIREMENT 3.6.1.2.B. UPON IDENTIFICATION OF THE EXCESSIVE LEAKAGE, A MAINTENANCE WORK REQUEST WAS GENERATED TO INVESTIGATE THE CAUSE. THE CAUSE OF THE LEAKAGE WAS DETERMINED TO BE NORMAL VALVE DEGRADATION (EXCESSIVE VALVE DISC TO SEAT LEAKAGE). A MAINTENANCE WORK REQUEST WAS GENERATED TO REMOVE THE DISC FROM THE VALVE LT48-OIV-0016A. THE DISC AND SEAT WERE LAPPED FOR CLEANING PURPOSES ONLY. UPON COMPLETION OF THESE ACTIONS ON SEPTEMBER 15, 1987, THE VALVE WAS RETESTED WITH ACCEPTABLE RESULTS.

[306] SHOREHAM DOCKET 50-322 LER 87-030
PRIMARY CONTAINMENT VENT/PURGE FILTER OPERATED GREATER THAN 720 HOURS WITHOUT SAMPLING.
EVENT DATE: 091187 REPORT DATE: 100987 NSSS: GE TYPE: BWR

(NSIC 206676) ON SEPTEMBER 11, 1987 AT 1100, THE RADIOCHEMISTRY SECTION DETERMINED THAT THE PRIMARY CONTAINMENT PURGE FILTER, 1T41-FLT-003 HAD BEEN OPERATING 1398 HOURS SINCE THE LAST CHARCOAL SAMPLE WAS REMOVED. THE PLANT WAS IN OPERATIONAL CONDITION 4 (COLD SHUTDOWN) WITH THE MODE SWITCH IN SHUTDOWN AND ALL RODS INSERTED IN THE CORE. THIS IS A VIOLATION OF TECH SPEC 4.11.2.8.3.C. THE TECH SPEC REQUIRES THAT AFTER EVERY 720 HOURS OF CHARCOAL ADSORBER OPERATION, A REPRESENTATIVE CARBON SAMPLE BE REMOVED AND ANALYZED WITHIN 31 DAYS TO VERIFY THAT THE CRITERIA OF REGULATORY GUIDE 1.52 IS MET. THE LACK OF AN ACCURATE METHOD FOR TRACKING RUN TIME HOURS THROUGH 1T41-FLT-003 WAS THE CAUSE OF THE

EVENT. IN ORDER TO ENSURE COMPLIANCE WITH TECH SPEC 4.11.2.8.3.C, A PREVENTIVE MAINTENANCE ACTIVITY HAS BEEN INITIATED TO ENSURE AN ACCURATE ACCOUNT OF FILTER RUN TIME. IN ADDITION, AN ENGINEERING EVALUATION AND ASSISTANCE REQUEST (EEAR 85-081) HAS BEEN EXPEDITED WHICH CALLS FOR THE INSTALLATION OF A RUN TIME METER AT THE PRIMARY CONTAINMENT PURGE FILTER EXHAUST FAN.

[307] SOUTH TEXAS 1 DOCKET 50-498 LER 87-003
ACTUATOR MOTOR SHAFT-TO-PINION KEYS SHEARED DUE TO INCORRECT AND DEFECTIVE MATERIAL.
EVENT DATE: 082887 REPORT DATE: 092687 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SOUTH TEXAS 2 (PWR)
VENDOR: HILLS-MCCANNA COMPANY

(NSIC 206506) ON AUGUST 17, 1987 AT APPROXIMATELY 1735 HOURS A NORMAL ESSENTIAL COOLING WATER (ECW) SYSTEM STARTUP WAS BEING PERFORMED WHEN THE ECW PUMP 1B TRIPPED AFTER HAVING BEEN STARTED FOR APPROXIMATELY TEN (10) SECONDS. IMMEDIATE INVESTIGATION INDICATED THAT THE MOTOR OPERATED DISCHARGE VALVE DID NOT FULLY OPEN DUE TO A SHEARED ACTUATOR MOTOR SHAFT-TO-PINION KEY. THE ACTUATOR IS A LIMITORQUE SMB-0-25 MOTOR OPERATOR. DURING APRIL OF 1987 A SIMILAR KEY FAILURE OCCURRED IN THE C TRAIN ECW DISCHARGE VALVE MOTOR OPERATOR. METALLURGICAL ANALYSTS INDICATED THAT ECW VALVE KEYS FAILED AS A RESULT OF THE KEY MATERIALS NOT BEING CONSISTENT WITH THE MATERIAL SPECIFIED BY LIMITORQUE. THE PROJECT JUDGED THAT OTHER VALVE OPERATORS OF THIS TYPE WERE SUSPECT AND AN EVALUATION HAS BEEN PERFORMED OF OTHER LIMITORQUE ACTUATOR MODELS USED IN SAFETY RELATED APPLICATIONS AT SOUTH TEXAS. THE KEYS HAVE BEEN REPLACED IN EACH SMB-0-25 MOTOR OPERATOR IN UNIT 1 (A TOTAL OF TWELVE OPERATORS).

[308] SOUTH TEXAS 1 DOCKET 50-498 LER 87-004
CONTROL ROOM VENTILATION ACTUATION TO RECIRCULATION MODE AS A RESULT OF LOSS OF SAMPLE FLOW TO THE CONTROL ROOM VENTILATION RADIATION MONITOR DUE TO OPERATOR ERROR.
EVENT DATE: 090387 REPORT DATE: 100287 NSSS: WE TYPE: PWR

(NSIC 206603) AT APPROXIMATELY 0411 HOURS ON SEPTEMBER 3, 1987 WITH UNIT 1 IN MODE 6, A CONTROL ROOM VENTILATION ACTUATION TO RECIRCULATION MODE OCCURRED AS A RESULT OF LOSS OF SAMPLE FLOW TO A CONTROL ROOM VENTILATION RADIATION MONITOR (RT- 8033). AN OPERATOR HAD INADVERTENTLY PUSHED THE "FLOW" PUSHBUTTON FOR MONITOR RT-8033 AT THE RADIATION MONITOR CONTROL ROOM OPERATOR CONSOLE (ZCP-023), THIS TURNED OFF THE SAMPLE PUMP FOR RT-8033 AND THE CONTROL ROOM VENTILATION ACTUATION OCCURRED AS DESIGNED. FOLLOWING THE EVENT THE CONTROL ROOM OPERATORS TURNED ON THE SAMPLE PUMP FOR RT-8033 AND REALIGNED THE CONTROL ROOM VENTILATION TO ITS NORMAL CONFIGURATION. CORRECTIVE ACTIONS TO PREVENT SIMILAR OCCURRENCES INCLUDE INSTALLATION OF PROTECTIVE COVERS OVER THE ACTUATION BUTTONS, RED WARNING TAPE BY THE ACTUATION PUSHBUTTONS AND WARNING LABELS ON PANEL ZCP-023. AN EVALUATION IS BEING PERFORMED TO CONSIDER THE POSSIBILITY A) OF MAKING TWO SPECIFIC ACTIONS NECESSARY TO STOP MONITOR SAMPLES FLOW FROM PANEL ZCP-023, B) CHANGING THE NAME OF THE PUSHBUTTONS FROM "FLOW" TO "PUMP ON/OFF" AND C) MODIFYING THE ESF ACTUATION LOGIC TO REQUIRE BOTH CONTROL ROOM VENTILATION RADIATION MONITORS TO BE INOPERABLE TO CAUSE AN ESF ACTUATION.

[309] SOUTH TEXAS 1 DOCKET 50-498 LER 87-007
CONTROL ROOM VENTILATION ACTUATION TO RECIRCULATION MODE DUE TO A TOXIC GAS MONITOR DEFECTIVE FLOW SWITCH.
EVENT DATE: 090687 REPORT DATE: 100687 NSSS: WE TYPE: PWR
VENDOR: DWYER INSTRUMENTS INC.
FOXBORO CO., THE

(NSIC 206687) AT APPROXIMATELY 1206 ON SEPTEMBER 6, 1987 WITH UNIT 1 IN MODE 5,

AN AUTO- ACTUATION OF THE CONTROL ROOM VENTILATION TO RECIRCULATION MODE OCCURRED AS A RESULT OF LOSS OF SAMPLE FLOW SIGNAL FROM A TOXIC GAS MONITOR. THE CONTROL ROOM VENTILATION ACTUATION TO RECIRCULATION MODE IS AN, ENGINEERED SAFETY FEATURE (ESF). THS CONTROL ROOM OPERATORS VERIFIED THE RECIRCULATION MODE DAMPER LINEUP AND INITIATED AN INVESTIGATION OF THE EVENT. THE INVESTIGATION DETERMINED THAT THE EVENT WAS CAUSED BY A DEFECTIVE FLOW SWITCH ON A TOXIC GAS MONITOR. ON SEPTEMBER 9, 1987 THE DEFECTIVE FLOW SWITCH WAS REPLACED AND THE SYSTEM WAS RESTORED TO NORMAL OPERATION. NO OTHER CORRSCTIVE ACTIONS ARE PLANNED.

[310] SOUTH TEXAS 1 DOCKET 50-498 LER 87-008
QUALIFIED DISPLAY PROCESSING SYSTEM SELF-DIAGNOSTIC ERROR INITIATES A REATOR TRIP AND AUXILIARY FEEDWATER ACTUATION.
EVENT DATE: 091087 REPORT DATE: 100987 NSSS: WE TYPE: PWR

(NSIC 206686) AT APPROXIMATELY 2020 HOURS ON SEPTEMBER 10, 1987 REACTOR TRIP AND AUXILIARY FEEDWATER (AFW) ACTUATION SIGNALS WERE GENERATSD DUE TO A STEAM GENERATOR (SG) LO-LO LEVEL TRIP. PRIOR TO THE EVENT ONE PROTECTION CHANNEL OF SG 1A LEVEL HAD BEEN INTENTIONALLY PLACED IN THE TRIPPED CONDITION. A SECOND PROTECTION CHANNEL OF THE SG 1A LEVEL SPURIOUSLY TRIPPED INITIATING AFW ACTUATION AND REACTOR TRIP SIGNALS. THE AFW PUMPS WERE OUT OF SERVICE AND THE ROD CONTROL MOTOR-GENERATOR POWER SUPPLY BREAKERS WERE OPEN; THEREFORE, ONLY 1 PART OF THE AFW AND CONTROL ROD DRIVE SYSTEMS WERE ACTUATED. THE CAUSE OF THE SECOND PROTECTION CHANNEL TRIP WAS A SELF-DIAGNOSTIC ROUTINE IN THE QUALIFIED DISPLAY PROCESSING SYSTEM (QDPS) COMPUTER PROVIDED BY WESTINGHOUSE WHICH CAUSED THE SYSTEM TO RESET THE SG LEVEL OUTPUTS TO 0% AND PRODUCED A TRIP SIGNAL TO THE REACTOR PROTECTION SYSTEM (RPS). THE DIAGNOSTIC ERRORS WHICH CAUSED THE SYSTEM TO RESET THE SG LEVEL OUTPUTS DO NOT MAKE THE QDPS INOPERABLE AND SHOULD NOT GENERATE A TRIP SIGNAL TO THE RPS; THEREFORE, A FIRMWARE CHANCE HAS BEEN INCORPORATED TO PREVENT FUTURE OCCURRENCES OF THIS EVENT. THIS EVENT WAS DETERMINED NOT TO BE REPORTABLE SINCE THE SPURIOUS ACTUATION OF PART OF A SYSTEM WHICH IS OUT OF SERVICE IS NOT A REPORTABLE EVENT. ADDITIONALLY THE AFW PUMPS DID NOT START AND NO MOVEMENT OF THE CONTROL RODS OCCURRED.

[311] SOUTH TEXAS 1 DOCKET 50-498 LER 87-009
SURVEILLANCE DEFICIENCY DUE TO A PROCEDURAL INADEQUACY RESULTING IN A TECH SPEC VIOLATION.
EVENT DATE: 091887 REPORT DATE: 101687 NSSS: WE TYPE: PWR

(NSIC 206688) ON SEPTEMBER 18, 1987 WITH THE UNIT IN MODE 5, DURING THE SYSTEM ENGINEER'S REVIEW OF THE REACTOR CONTAINMENT BUILDING (RCB) ATMOSPHERE MONITOR SURVEILLANCE TEST, IT WAS DETERMINED THAT, CONTRARY TO THE TECHNICAL SPECIFICATIONS, THE TEST DID NOT INCLUDE THE MONITOR'S IODINE CHANNEL. THE PERFORMANCE OF THE SURVEILLANCE TEST IS REQUIRED FOR ALL MODES OF OPERATION, THEREFORE THE IODINE CHANNEL OF THE RCB ATMOSPHERE MONITOR WAS DECLARED TO BE INOPERABLE. SAMPLING OF THE RCB ATMOSPHERE WAS INITIATED IMMEDIATELY AS REQUIRED BY THE TECHNICAL SPECIFICATIONS. THE RCB ATMOSPHERE MONITOR SURVEILLANCE PROCEDURE WAS REVISED TO INCLUDE THE TEST OF THE IODINE CHANNEL AND THE SURVEILLANCE WAS SATISFACTORILY PERFORMED ON SEPTEMBER 19, 1987. THE TECHNICAL SPECIFICATION HAD BEEN REVISED BETWEEN THE FINAL DRAFT AND THE TECHNICAL SPECIFICATIONS ISSUED WITH THE OPERATING LICENSE. THE CAUSE OF THIS EVENT WAS A PROCEDURAL INADEQUACY WHICH RESULTED IN A SURVEILLANCE DEFICIENCY. TO PREVENT RECURRENCE, THE SURVEILLANCE TEST PROCEDURES ARE BEING REVIEWED TO ENSURE THAT ADEQUATE PROCEDURAL REQUIREMENTS EXIST FOR EACH TECHNICAL SPECIFICATION ITEM.

[312] SOUTH TEXAS 1 DOCKET 50-498 LER 87-010
FUEL HANDLING BUILDING VENTILATION SYSTEM AUTO-ACTUATION TO FILTRATION MODE DUE TO AN APPARENT EQUIPMENT FAILURE OF A RADIATION MONITOR.
EVENT DATE: 092687 REPORT DATE: 102687 NSSS: WE TYPE: PWR

(NSIC 206844) ON SEPTEMBER 26, 1987 AT APPROXIMATELY 1610 HOURS WITH THE PLANT IN MODE 5, A FUEL HANDLING BUILDING (FHB) VENTILATION SYSTEM AUTO-ACTUATION TO FILTRATION MODE OCCURRED DUE TO AN APPARENT EQUIPMENT FAILURE OF A FHB ATMOSPHERE RADIATION MONITOR. THE FHB VENTILATION SYSTEM AUTO-ACTUATION TO THE FILTRATION MODE IS AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION. THE RADIATION MONITOR IS A NOBLE GAS MONITOR PROVIDED BY G. A. TECHNOLOGIES AS PART OF THE DIGITAL RADIATION MONITORING SYSTEM. FOLLOWING THE EVENT, THE OPERATORS VERIFIED THE PROPER FHB VENTILATION SYSTEM DAMPER ALIGNMENT. SUBSEQUENT INVESTIGATION OF THE EVENT INDICATED THAT THE RADIATION MONITOR HAD A "LOSS OF COUNTS" MAKING THE RADIATION MONITOR INOPERABLE: HOWEVER, THE INVESTIGATION OF THE CAUSE OF THE "LOSS OF COUNTS" WAS INCONCLUSIVE. THE RADIATION MONITOR RECORDED NO COUNTS FOR APPROXIMATELY ONE HOUR ON SEPTEMBER 26, 1987, BUT HAS OPERATED SATISFACTORILY SINCE THAT TIME. THE RADIATION MONITOR'S PRINTED CIRCUIT BOARDS HAVE BEEN INSPECTED, THEIR CONTACTS CLEANED, THE INPUT/OUTPUT AND PREAMPLIFIER PRINTED CIRCUIT BOARDS HAVE BEEN REPLACED AND THE CALIBRATION SURVEILLANCE PROCEDURE HAS BEEN PERFORMED IN AN EFFORT TO PREVENT RECURRENCES OF THE EVENT.

[313] ST. LUCIE 1 DOCKET 50-335 LER 87-014
UNIDENTIFIED REACTOR COOLANT SYSTEM LEAKAGE GREATER THAN TECHNICAL SPECIFICATION LIMIT RESULTS IN REACTOR SHUTDOWN.
EVENT DATE: 100887 REPORT DATE: 110687 NSSS: CE TYPE: PWR
VENDOR: BYRON JACKSON PUMPS, INC.

(NSIC 206927) ON OCTOBER 8, 1987, ST. LUCIE UNIT 1 WAS AT MODE 1, 100% POWER AND AT STEADY STATE CONDITIONS. WHILE PERFORMING A ROUTINE 2-HOUR LEAK RATE SURVEILLANCE, IT WAS DISCOVERED THAT THERE WAS AN UNIDENTIFIED REACTOR COOLANT SYSTEM (RCS) LEAKAGE GREATER THAN ONE GALLON PER MINUTE (GPM) (1.09 GPM). AT 0529, THE NUCLEAR PLANT SUPERVISOR (NPS) DECLARED AN UNUSUAL EVENT. OPERATIONS PERSONNEL MADE A CONTAINMENT ENTRY TO INVESTIGATE THE SOURCE OF LEAKAGE. AT 0612 HOURS, A CONTROLLED REACTOR SHUTDOWN WAS STARTED SO FURTHER INVESTIGATIONS COULD BE PERFORMED. DURING THIS INVESTIGATION, THE SOURCES OF THE LEAKAGE WERE IDENTIFIED AND DETERMINED TO BE LESS THAN THE 10 GPM ALLOWED BY THE TECHNICAL SPECIFICATIONS. THE UNUSUAL EVENT WAS TERMINATED AT 1405 ON OCTOBER 8, 1987. CAUSE OF THE EVENT WAS DUE TO LEAKING CHECK VALVE BONNET AND A CRACKED PIPE IN THE HEAT AFFECTED ZONE ON THE 1A1 REACTOR COOLANT PUMP (RCP) LOWER CAVITY SEAL NOZZLE WELD. THE ROOT CAUSE OF THE WELD JOINT FAILURE WAS DUE TO THE PREVIOUS MISALIGNMENT OF THE SEAL INJECTION PIPING FLANGE AND THE RCP LOWER CAVITY SEAL NOZZLE FLANGE. FOR CORRECTIVE ACTIONS, THE UNIT WAS SHUTDOWN AND REQUIRED REPAIRS WERE PERFORMED AND COMPLETED. THE REMAINING REACTOR COOLANT PUMPS, 1A2, 1B1, AND 1B2 WERE INSPECTED FOR PROPER ALIGNMENT OF THE SEAL INJECTION AND LOWER CAVITY SEAL NOZZLE FLANGES.

[314] ST. LUCIE 2 DOCKET 50-389 LER 87-006
MISSED SURVEILLANCE ON THE 2A AND 2B EMERGENCY DIESEL GENERATOR AUTOMATIC LOAD SEQUENCE RELAYS DUE TO A COGNITIVE PERSONNEL ERROR.
EVENT DATE: 091787 REPORT DATE: 101687 NSSS: CE TYPE: PWR

(NSIC 206684) AT 1900 ON SEPTEMBER 17, 1987, IT WAS DETERMINED THAT THE 12 MONTH TEST OF THE 2A AND 2B EMERGENCY DIESEL GENERATOR (EDG) AUTOMATIC LOAD SEQUENCE RELAYS WAS INCOMPLETE. A REVIEW OF THE SURVEILLANCE PROCEDURE, PERFORMED ON MAY 4, 1987, INDICATED THAT ONLY THE COMPONENTS ON THE 4160 VOLT 2A3 AND 2B3 BUSSES WERE TESTED AND THAT THE TEST DID NOT INCLUDE THE COMPONENTS ON THE 4160 VOLT 2AB BUS AND THE 480 VOLT LOAD CENTERS (LC). THE ROOT CAUSE OF THE EVENT WAS A COGNITIVE PERSONNEL ERROR BY UTILITY PERSONNEL IMPLEMENTING THE MAINTENANCE SURVEILLANCE PROCEDURE AND THE QUALITY CONTROL PERSONNEL REVIEWING THE PLANT WORK ORDER AND THE MAINTENANCE SURVEILLANCE PROCEDURE FOR COMPLETENESS AND COMPLIANCE TO THE SURVEILLANCE REQUIREMENTS. FOR CORRECTIVE ACTIONS, THE ELECTRICAL MAINTENANCE PERSONNEL IMMEDIATELY PERFORMED THE REQUIRED SURVEILLANCE TEST IN ACCORDANCE WITH THE APPROVED PROCEDURE. PLANT MANAGEMENT HAS RE-EMPHASIZED THE

IMPORTANCE OF ADEQUATE REVIEW OF PLANT WORK ORDERS AND SURVEILLANCE PROCEDURES. IN ADDITION, THE SURVEILLANCE PROCEDURE WAS REVISED FOR CLARITY.

[315] SUMMER 1 DOCKET 50-395 LER 87-017 REV 01
UPDATE ON SEISMIC INSTRUMENTATION SETPOINTS.
EVENT DATE: 072487 REPORT DATE: 100187 NSSS: WE TYPE: PWR
VENDOR: ENGDAHL ENTERPRISES

(NSIC 206592) ON JULY 24, 1987, AT 0930 HOURS, A LICENSEE INVESTIGATION DETERMINED THAT SEISMIC EVENT ANNUNCIATOR IYA-1780 LOCATED IN THE MAIN CONTROL PANEL HAD INDICATION SETPOINTS GREATER THAN REQUIRED BY DESIGN. THIS ANNUNCIATOR MONITORS SEISMIC ACCELERATIONS AT TWELVE (12) FREQUENCY POINTS IN EACH ORTHOGONAL DIRECTION AS RECORDED BY TRIAXIAL RESPONSE SPECTRUM RECORDER IYM- 1783, WHICH IS LOCATED ON THE REACTOR BUILDING FOUNDATION MAT. FOR EACH FREQUENCY, THE ANNUNCIATOR HAS A "YELLOW" LIGHT SET TO A VALUE OF ONE-HALF (1/2) THE SAFE SHUTDOWN EARTHQUAKE - DESIGN BASES RESPONSE SPECTRA (SSE-DBRS), AND A "RED" LIGHT SET TO A VALUE OF THE OPERATING BASIS EARTHQUAKE - DESIGN BASES RESPONSE SPECTRA (OBE-DBRS). THESE SETPOINTS, WHICH REPRESENT THE ACTUAL DESIGN RESPONSE SPECTRA FOR THE REACTOR BUILDING FOUNDATION MAT, WERE PRE-SET IN 1978 BY THE INSTRUMENT SUPPLIER FROM DATA PROVIDED BY THE ARCHITECT ENGINEER. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO THE ARCHITECT ENGINEER MISREADING THE SCALE OF THE OBE-DBRS CURVES AND TABULATING ACCELERATION "G" VALUES FOR THE SETPOINTS CONSIDERABLY HIGHER THAN DESIGN. THE SSE-DBRS VALUES WERE MATHEMATICAL CONVERSIONS MADE FROM THE INITIAL MISREAD OBE-DBRS, AND ARE ALSO CORRESPONDINGLY HIGH. BOTH THE ANNUNCIATOR AND RECORDER ARE PRESENTLY DECLARED INOPERABLE AWAITING AVAILABILITY OF THE INSTRUMENT SUPPLIER TO INCORPORATE CORRECTED SETPOINT VALUES AND CALIBRATIONS.

[316] SUMMER 1 DOCKET 50-395 LER 87-020
BOTH SI PUMPS INOPERABLE DUE TO DESIGN INTERLOCK.
EVENT DATE: 082187 REPORT DATE: 091587 NSSS: WE TYPE: PWR
VENDOR: PACIFIC PUMPS

(NSIC 206079) AT 0910 HOURS, AUGUST 21, 1987, VOLUNTARY ENTRY WAS MADE INTO SECTION 3.0.3 OF THE TECHNICAL SPECIFICATIONS (TS) TO TEST "A" CHARGING/SAFETY INJECTION PUMP (C/SIP). ENTRY INTO SECTION 3.0.3 OF TS WAS REQUIRED BECAUSE "B" C/SIP WAS INOPERABLE AND IN ORDER TO TEST "A" C/SIP, BOTH "A" AND "C" BREAKERS WERE REQUIRED TO BE RACKED UP TO THE SAME TRAIN. THIS CONDITION MAKES BOTH PUMPS INOPERABLE DUE TO A DESIGN INTERLOCK DURING A SAFETY INJECTION FOLLOWED BY A BLACKOUT. AT 0913 HOURS, "A" C/SIP BREAKER OPERABILITY TESTING WAS COMPLETE. AT 0920 HOURS, "A" C/SIP WAS PLACED IN-SERVICE ON 'A' TRAIN; AT 0921 HOURS, "C" C/SIP WAS REMOVED FROM SERVICE ON 'A' TRAIN AND TS SECTION 3.0.3 WAS EXITED. THE CAUSE OF THIS EVENT WAS DUE TO A DESIGN INTERLOCK AS PREVIOUSLY DISCUSSED AND THE DECISION TO VOLUNTARILY ENTER TS SECTION 3.0.3 AS A MEANS TO MINIMIZE THE TIME WITH ONLY ONE OPERABLE C/SIP. THE CONSEQUENCES OF THE EVENT WERE MINIMAL BECAUSE OF THE SHORT DURATION (11 MINUTES) UNDER THE REQUIREMENTS OF TS 3.0.3 AND THE ABILITY TO MANUALLY START "C" C/SIP IF REQUIRED. NO SPECIFIC CORRECTIVE ACTION FOR THIS EVENT IS WARRANTED.

[317] SUMMER 1 DOCKET 50-395 LER 87-021
REACTOR TRIP/TURBINE TRIP DUE TO MAIN GENERATOR STATOR GROUND.
EVENT DATE: 090287 REPORT DATE: 093087 NSSS: WE TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 206593) ON SEPTEMBER 2, 1987, AT 2109 HOURS, WHILE THE PLANT WAS OPERATING AT 100% POWER THE REACTOR TRIPPED DUE TO A TURBINE TRIP CONDITION ABOVE THE 50% POWER P-9 PERMISSIVE. A MAIN GENERATOR STATOR GROUND RESULTED IN AN ELECTRICAL FAULT WHICH CAUSED A MAIN TURBINE TRIP. A FAILED MAIN GENERATOR BUSHING WAS THE

CAUSE OF THE GENERATOR STATOR GROUND CONDITION. THE PLANT RESPONDED TO THE REACTOR TRIP AS EXPECTED WITH NO SAFETY LIMITS EXCEEDED. ALIGNED TO THE CONDENSATE STORAGE TANK, BOTH MOTOR DRIVEN EMERGENCY FEEDWATER PUMPS STARTED TO MAINTAIN STEAM GENERATOR LEVEL WHEN MAIN FEEDWATER WAS ISOLATED ON THE REACTOR TRIP. THE MAIN STEAM POWER OPERATED RELIEF VALVES WERE USED TO MAINTAIN RCS TEMPERATURE BECAUSE THE MAIN CONDENSER VACUUM WAS BROKEN TO ALLOW INSPECTIONS AND REPAIRS TO BE MADE ON THE HEATERS AND HEATER DRAIN VALVES. BECAUSE THE CONDENSATE STORAGE TANK CONTAINED TRACES OF TRITIUM, THE STEAM RELEASED FROM THE MAIN STEAM POWER OPERATED RELIEF VALVES RESULTED IN AN UNMONITORED RELEASE WHICH WAS CALCULATED TO BE 1.008E-04% OF THE TECHNICAL SPECIFICATIONS SECTION 3.11.2 LIMIT. THE FAILED MAIN GENERATOR "C" PHASE BUSHING WAS REPLACED. THE "A" AND "B" PHASE BUSHINGS WERE INSPECTED, AND MEGGAR READINGS WERE TAKEN REVEALING NO ABNORMALITIES.

[318] SUMMER 1 DOCKET 50-395 LER 87-022
AUTO-START OF MOTOR DRIVEN EMERGENCY FEEDWATER PUMPS.
EVENT DATE: 091187 REPORT DATE: 100887 NSSS: WE TYPE: PWR

(NSIC 206690) ON SEPTEMBER 11, 1987, AT 0742 HOURS WITH THE PLANT IN MODE 3 AND MAINTENANCE BEING PERFORMED ON THE MAIN GENERATOR AND MAIN CONDENSER, THE VIRGIL C. SUMMER NUCLEAR STATION EXPERIENCED AN ENGINEERED SAFETY FEATURES (ESF) ACTUATION. THE TWO MOTOR DRIVEN EMERGENCY FEEDWATER PUMPS AUTO-STARTED ON A SIGNAL FROM THE LOSS OF ALL THREE MAIN FEEDWATER PUMPS. PRIOR TO THE EVENT, "B" AND "C" MAIN FEEDWATER PUMPS WERE ALREADY IN THE TRIPPED CONDITION AND "A" WAS RESET. "B" MOTOR DRIVEN EMERGENCY FEEDWATER PUMP WAS SUPPLYING MAKEUP TO THE STEAM GENERATORS AND "A" WAS IN STANDBY. "A" MAIN FEEDWATER PUMP TRIPPED UPON RECEIVING A DEAERATOR LOW LEVEL SIGNAL, AND THUS WITH ALL THREE MAIN FEEDWATER PUMPS TRIPPED, A START SIGNAL WAS SENT TO THE TWO MOTOR DRIVEN EMERGENCY FEEDWATER PUMPS. THERE WAS NO SAFETY SIGNIFICANCE TO THIS EVENT AND OPERATIONS PERSONNEL HAVE BEEN REINDOCTRINATED TO ANTICIPATE RESPONSES TO CHANGING PLANT CONDITIONS.

[319] SUMMER 1 DOCKET 50-395 LER 87-023
CONTROL ROOM VENTILATION ALIGNS TO RECIRCULATION MODE DUE TO SPURIOUS HIGH RADIATION SIGNAL.
EVENT DATE: 092287 REPORT DATE: 101987 NSSS: WE TYPE: PWR

(NSIC 206821) OPERATING MODE 1 - REACTOR POWER LEVEL 100% AT 1733 HOURS, SEPTEMBER 22, AND 1753 HOURS, SEPTEMBER 23, 1987, THE CONTROL ROOM VENTILATION ALIGNED TO THE EMERGENCY RECIRCULATION MODE AS A RESULT OF SPURIOUS HIGH RADIATION SIGNALS ON THE GASEOUS CHANNEL OF THE CONTROL ROOM RADIATION MONITOR (RMA-1). IMMEDIATELY FOLLOWING EACH EVENT, GRAB SAMPLES WERE TAKEN TO VERIFY THE ABSENCE OF ANY ACTIVITY. FOLLOWING THE EVENT ON SEPTEMBER 22, INSTRUMENT TECHNICIANS FOUND OIL IN THE DETECTOR HOUSING. THE HOUSING WAS CLEANED; A SATISFACTORY SOURCE CHECK WAS PERFORMED AND THE OPERABILITY OF THE UNIT WAS MONITORED FOR APPROXIMATELY SIXTEEN HOURS PRIOR TO DECLARING THE UNIT OPERABLE. FOLLOWING THE EVENT ON SEPTEMBER 23, ALL CARD (CIRCUIT BOARD) EDGES WERE THOROUGHLY CLEANED TO ENSURE POSITIVE CONTINUITY. A SATISFACTORY OPERATIONAL SURVEILLANCE TEST WAS PERFORMED, AND THE OPERATION OF THE UNIT WAS MONITORED FOR APPROXIMATELY SEVENTY-TWO HOURS PRIOR TO DECLARING THE UNIT OPERABLE. AT PRESENT, THE LICENSEE PLANS NO ADDITIONAL CORRECTIVE ACTION OTHER THAN SCHEDULED PREVENTATIVE MAINTENANCE AND SURVEILLANCE TESTING. LER 87-016 DATED AUGUST 12, 1987, DOCUMENTS A SIMILAR EVENT.

[320] SUMMER 1 DOCKET 50-395 LER 87-024
NEGATIVE RATE REACTOR TRIP DUE TO TECHNICIAN DEENERGIZING ROD CONTROL POWER SUPPLY.
EVENT DATE: 092487 REPORT DATE: 101987 NSSS: WE TYPE: PWR

VENDOR: LAMBDA ELECTRONICS

(NSIC 206822) MODE 1 - REACTOR POWER 100% AT 2210 HOURS, SEPTEMBER 24, 1987, A NEGATIVE RATE REACTOR TRIP OCCURRED WHEN AN INSTRUMENT AND CONTROL (I&C) TECHNICIAN INADVERTENTLY DEENERGIZED TWO ROD CONTROL POWER SUPPLIES WHICH RESULTED IN TWELVE CONTROL RODS INSERTING INTO THE REACTOR CORE. THE CAUSE OF THIS EVENT IS DUE TO PERSONNEL ERROR IN THAT THE I&C TECHNICIAN USED A SCHEMATIC DIAGRAM FROM AN APPROVED TECHNICAL MANUAL IN LIEU OF THE CIRCUIT WIRING DIAGRAM WHILE PERFORMING THIS REPAIR. ALTHOUGH THE TECHNICIAN WAS IN ERROR IN THAT HE WAS WORKING FROM A SCHEMATIC DIAGRAM, CONTRIBUTING TO HIS ERROR WAS THE IDENTIFICATION OF THE POWER SUPPLIES AS "PRIMARY" AND "REDUNDANT." THE SCHEMATIC DIAGRAM OF THE CIRCUITRY AND THE NOMENCLATURE LED THE TECHNICIAN TO BELIEVE THAT THE POWER SUPPLIES WERE IN PARALLEL AND THAT REPLACEMENT OF EITHER POWER SUPPLY COULD BE PERFORMED WITHOUT AFFECTING OPERATION OF THE PLANT. WHEN THE POSITIVE VOLTAGE LEAD WAS LIFTED, BOTH POWER SUPPLIES DEENERGIZED RESULTING IN THE REACTOR TRIP. AS A RESULT OF THIS EVENT, ALL I&C PERSONNEL ARE TO BE TRAINED IN THE PROPER APPLICATION OF LOGIC/SCHEMATIC DIAGRAMS. TRAINING WILL ALSO EMPHASIZE THE DIFFERENCE BETWEEN LOGIC/SCHEMATIC AND WIRING DIAGRAMS. (THIS TRAINING WILL BE COMPLETED BY NOVEMBER 20, 1987.) THERE HAVE BEEN NO SIMILAR EVENTS.

[321] SURRY 1 DOCKET 50-280 LER 87-024
REACTOR TRIP ON LOW RCS FLOW DUE TO REACTOR COOLANT PUMP TRIP.
EVENT DATE: 092087 REPORT DATE: 102087 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 206785) ON SEPTEMBER 20, 1987 AT 2028 HOURS, WITH UNIT 1 AT 100% POWER, A LOW REACTOR COOLANT SYSTEM (RCS) FLOW REACTOR TRIP OCCURRED WHEN THE 'B' REACTOR COOLANT PUMP (RCP) (EIIS-AB P) TRIPPED. APPROXIMATELY 35 SECONDS AFTER THE REACTOR TRIP, A HIGH STEAM FLOW WITH LOW RCS TAVG SAFETY INJECTION OCCURRED. OPERATORS PERFORMED THE APPROPRIATE EMERGENCY AND FUNCTION RESTORATION PROCEDURES AND QUICKLY STABILIZED THE UNIT. THE 'B' REACTOR COOLANT PUMP BREAKER TRIPPED ON INSTANTANEOUS GROUND FAULT. INSPECTION OF THE MOTOR LEADS REVEALED A COMPLETE SEPARATION OF THE 'A' PHASE MAIN LOAD CONNECTION BUS BAR. AN ENGINEERING EVALUATION IS BEING PERFORMED TO DETERMINE THE CAUSE OF THIS FAILURE. THE FAILED BUS BAR WAS REPLACED. THE OTHER BUS BARS ON A, B AND C RCP'S WERE VISUALLY INSPECTED AND MEGGERED TO VERIFY THEIR INTEGRITY. THE HIGH STEAM FLOW SAFETY INJECTION INITIATION WAS DETERMINED TO BE SPURIOUS.

[322] SURRY 1 DOCKET 50-280 LER 87-025
EXCESSIVE REACTOR COOLANT SYSTEM LEAKAGE DUE TO VALVE SEAT LEAKAGE.
EVENT DATE: 100887 REPORT DATE: 110587 NSSS: WE TYPE: PWR

(NSIC 206914) ON OCTOBER 8, 1987 AT 1208 HOURS, WITH UNIT 1 AT 100% POWER, A 30 MINUTE REACTOR COOLANT SYSTEM (RCS) (EIIS-AB) LEAK RATE TEST INDICATED TOTAL RCS LEAKAGE OF 4.06 GPM, OF WHICH 1.13 GPM WAS UNIDENTIFIED. THIS EXCEEDED TECHNICAL SPECIFICATION 3.1.C.2 WHICH LIMITS UNIDENTIFIED RCS LEAKAGE TO LESS THAN 1 GPM. SUSPECTING THAT THE LEAKAGE ORIGINATED IN THE LETDOWN SYSTEM (EIIS-CB), THE UNIT WAS PLACED ON EXCESS LETDOWN; NORMAL LETDOWN WAS SECURED, AND A 14 MINUTE LEAK RATE CALCULATION WAS PERFORMED AT 1443 HOURS. THE RESULTS INDICATED TOTAL RCS LEAKAGE OF 13 GPM OF WHICH .24 GPM WAS UNIDENTIFIED. THIS EXCEEDED TECHNICAL SPECIFICATION 3.1.C.5 WHICH LIMITS TOTAL RCS LEAKAGE TO LESS THAN 10 GPM. EXCESS LETDOWN WAS THEN SECURED AND NORMAL LETDOWN REESTABLISHED. AT 1702 HOURS, A 67 MINUTE LEAK RATE CALCULATION SHOWED TOTAL RCS LEAKAGE OF 4.02 GPM OF WHICH .24 GPM WAS UNIDENTIFIED. SINCE A RCS WALKDOWN DISCOVERED NO LEAKAGE AND SUBSEQUENT LEAK RATES INDICATED UNIDENTIFIED RCS LEAKAGE TO BE ACCEPTABLE, IT IS SUSPECTED THAT THE ORIGINAL LEAK RATE CALCULATION WAS IN ERROR. THE PROCEDURE HAS BEEN REVISED TO MORE ACCURATELY ACCOUNT FOR FLUCTUATIONS IN RCS TEMPERATURE DURING LEAK RATE CALCULATIONS. IT WAS SUBSEQUENTLY DETERMINED THAT THE EXCESS LETDOWN

HEAT EXCHANGER BYPASS VALVE (1-RC-107) (EIIS-V) WAS LEAKING PAST THE SEAT. THE VALVE WAS TIGHTENED AND THE SEAT LEAKAGE WAS REDUCED.

[323] SURRY 1 DOCKET 50-280 LER 87-026
MAIN CONTROL ROOM VENTILATION ISOLATION DUE TO HIGH VOLTAGE OUTPUT ON CHLORINE GAS DETECTOR.
EVENT DATE: 101887 REPORT DATE: 110687 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)
VENDOR: CAPITAL CONTROLS CO.

(NSIC 206915) ON OCTOBER 18, 1987 AT 1621 HOURS AND 1728 HOURS, THE MAIN CONTROL ROOM (MCR) VENTILATION WAS ISOLATED DUE TO A HIGH VOLTAGE OUTPUT ON CHLORINE GAS DETECTOR CLA-VS-100A (EIIS-DET). AS A RESULT OF THE VOLTAGE SPIKE, THE CONTROL ROOM EXHAUST FAN, 1-F-VS-15 (EIIS-FAN) TRIPPED, AND SUPPLY DAMPER, 1-MOD-VS-103A (EIIS-DMP) AND EXHAUST DAMPER, 1-MOD-VS-103D (EIIS-DMP) CLOSED. THE CHEMISTRY DEPARTMENT VERIFIED THAT THERE WERE NO UNUSUAL CHLORINE CONCENTRATIONS PRESENT IN THE CONTROL ROOM. AT 1914 HOURS, THE 'A' TRAIN OF CHLORINE DETECTION WAS DECLARED OUT OF SERVICE AND THE 'A' DETECTOR WAS TURNED OFF. THE CONTROL ROOM VENTILATION WAS THEN REALIGNED TO NORMAL STATUS. THE 'A' DETECTOR WAS RETURNED TO SERVICE ON OCTOBER 20 AT 0807 HOURS. AT THE TIME OF THESE OCCURRENCES, BOTH UNITS 1 AND 2 WERE AT 100% POWER. THE MCR VENTILATION ISOLATED DUE TO HIGH SENSOR VOLTAGE ON THE CHLORINE GAS DETECTOR. AN INVESTIGATION PERFORMED BY THE DETECTOR VENDOR DETERMINED THAT THE DETECTOR SENSOR OPERATION IS AIR FLOW DEPENDENT DUE TO MOUNTING OF THE SENSOR IN THE VENTILATION DUCT. THE AIR FLOW HAS A COOLING EFFECT ON THE SENSOR WHICH INCREASES THE VOLTAGE OUTPUT, THUS INDUCING A SPURIOUS SIGNAL. THE CHLORINE GAS TREATMENT SYSTEM AT THE SEWAGE TREATMENT PLANT WILL BE REPLACED BY AN ULTRAVIOLET SYSTEM. AT THAT TIME, A TECHNICAL SPECIFICATION REVISION WILL BE SUBMITTED TO REMOVE THE CHLORINE GAS DETECTORS.

[324] SURRY 2 DOCKET 50-281 LER 87-004
CONTAINMENT ISOLATION VALVE INOPERABLE DUE TO RUPTURED DIAPHRAGM IN AIR PILOT RELAY.
EVENT DATE: 083187 REPORT DATE: 093087 NSSS: WE TYPE: PWR
VENDOR: ITT HAMMEL DAHL CONOFLOW

(NSIC 206588) ON AUGUST 31, 1987 AT 1130 HOURS, WITH UNIT 2 OPERATING AT 100% POWER, A UNIT NO. 2 CONTAINMENT COMPONENT COOLING WATER (CCW) (EIIS-CC OUTLET HEADER ISOLATION TRIP VALVE (EIIS ISV)) WAS DETERMINED TO BE INOPERABLE DURING THE PERFORMANCE OF A ROUTINE SURVEILLANCE PROCEDURE. THE VALVE, 2-CC-TV-209B, FAILED TO CLOSE WHEN A MANUAL SIGNAL WAS INITIATED FROM THE CONTROL ROOM. THIS IS CONTRARY TO TECHNICAL SPECIFICATION 3.8.A.1 WHICH REQUIRES THAT ALL AUTOMATIC CONTAINMENT ISOLATION VALVES BE OPERABLE. THE VALVE WAS SUCCESSFULLY CLOSED TWO HOURS LATER FOLLOWING VALVE ACTUATOR MAINTENANCE. THE FAILURE OF THE VALVE TO CLOSE WAS CAUSED BY A MALFUNCTIONING AIR PILOT RELAY. THE RELAY WAS DETERMINED TO HAVE A RUPTURED DIAPHRAGM WHICH PREVENTED THE RELAY FROM FUNCTIONING PROPERLY. THE RELAY WAS REPLACED.

[325] SURRY 2 DOCKET 50-281 LER 87-005
ENGINEERED SAFETY FEATURE ACTUATION DUE TO PERSONNEL ERROR.
EVENT DATE: 101487 REPORT DATE: 111287 NSSS: WE TYPE: PWR

(NSIC 206916) ON OCTOBER 14, 1987, UNIT 2 WAS OPERATING AT 100% POWER. AT 1037 HOURS, AN ELECTRICIAN INVESTIGATING A SPURIOUS FLOOD SENSOR ALARM ON THE TRAIN "A' TURBINE BUILDING FLOOD CONTROL (EIIS-VK) PANEL (TBFT-A) INADVERTENTLY COMPLETED THE FLOOD PROTECTION CIRCUIT. THIS CAUSED TWO OF THE FOUR CONDENSER CIRCULATING WATER INLET VALVES, MOV-CS-206A AND C (EIIS-BS V), TO BEGIN AUTOMATICALLY CLOSING AS DESIGNED. OPERATIONS PERSONNEL, RESPONDING TO A

DECREASING CONDENSER VACUUM, IDENTIFIED THE AFFECTED VALVES AND FULLY OPENED THEM. THE SPURIOUS FLOOD SENSOR ALARM WAS CAUSED WHEN A LEAK DEVELOPED ON A LEVEL SENSING LINE ABOVE ONE OF THE TURBINE BUILDING FLOOD LEVEL PROBES AND DAMPENED THE PROBE. THE PROBE WAS DRIED, AND THE FLOOD CONTROL SYSTEM WAS RETURNED TO NORMAL STATUS. THE LEAK ON THE SENSING LINE ABOVE THE PROBE WAS REPAIRED. A PROCEDURE WILL BE DEVELOPED TO PROVIDE INSTRUCTIONS AND PRECAUTIONS TO MAINTENANCE PERSONNEL WHEN WORKING ON THE TURBINE BUILDING FLOOD CONTROL CIRCUITS.

[326] SUSQUEHANNA 1 DOCKET 50-387 LER 86-018 REV 01
UPDATE ON DIVISION I LOCA ISOLATION WHEN "A" REACTOR PROTECTION SYSTEM BUS POWER LOST.
EVENT DATE: 041586 REPORT DATE: 110587 NSSS: GE TYPE: BWR

(NSIC 206954) ON APRIL 15, 1986 WITH UNIT 1 IN A REFUELING OUTAGE, A DIVISION I LOCA ISOLATION AND HALF SCRAM OCCURRED AT 1834 HOURS WHEN POWER WAS LOST ON THE 'A' REACTOR PROTECTION SYSTEM (RPS) BUS. THIS CAUSED A LOSS OF SHUTDOWN COOLING, DIVISION I ISOLATION OF THE ZONE I AND III HEATING, VENTILATION AND AIR CONDITIONING (HVAC) SYSTEMS AND INITIATION OF THE 'A' STANDBY GAS TREATMENT SYSTEM AND 'A' CONTROL ROOM EMERGENCY OUTSIDE AIR SUPPLY SYSTEM. THE ZONE I AND III HVAC SYSTEMS WERE ALREADY IN THE RECIRCULATION MODE DUE TO A PREVIOUS DIVISION II LOCA ISOLATION AT 1820 HOURS (SEE LER 86-015-00). WHILE INVESTIGATING THE CAUSE OF THE DIVISION II ISOLATION, A TEST LEAD WAS BUMPED OFF ITS TERMINAL INSIDE THE DIVISION I RPS PANEL AND CAUSED AN ARC. THIS CAUSED THE 'A' RPS BUS TO DE-ENERGIZE CAUSING THE EVENTS DESCRIBED ABOVE. ALL AFFECTED SYSTEMS WERE RESTORED AND SHUTDOWN COOLING WAS RE-ESTABLISHED AT 1910 HOURS. AN ENGINEERING REVIEW OF THE EVENT HAS DETERMINED THAT THE RPS SYSTEM RESPONDED CORRECTLY TO THE INADVERTENTLY APPLIED FAULT.

[327] SUSQUEHANNA 1 DOCKET 50-387 LER 87-026
REACTOR BUILDING HEATING, VENTILATING AND AIR CONDITIONING ZONES I AND III CROSS-TIED.
EVENT DATE: 090487 REPORT DATE: 100587 NSSS: GE TYPE: BWR

(NSIC 206600) ON SEPTEMBER 4, 1987 AT 0330 HOURS, WITH UNIT 1 OPERATING AT 90% POWER, IT WAS DISCOVERED THAT THE REACTOR BUILDING HEATING, VENTILATING AND AIR CONDITIONING (HVAC) SYSTEM ZONES I AND III HAD BEEN CROSS-TIED FROM 8/31/87 THRU 9/3/87. A SIMILAR OCCURRENCE, DISCOVERED ON AUGUST 10, 1987, WAS ADDRESSED IN LER 87- 025-00. THE 9/4/87 OCCURRENCE WAS DETERMINED TO HAVE BEEN CAUSED BY PERSONNEL ERROR AND POOR COMMUNICATIONS, BOTH WRITTEN AND VERBAL. WRITTEN INSTRUCTIONS ON AN EQUIPMENT RELEASE FORM WERE MISLEADING, AND OPERATIONS PERSONNEL FAILED TO SPECIFY THE USE OF AND UTILIZE THE OPERATING PROCEDURE WHILE ALIGNING THE REACTOR BUILDING HVAC SYSTEM TO FACILITATE THE TRANSFER OF EQUIPMENT FROM THE RAILROAD BAY, VIA REMOVABLE WALLS, INTO THE UNIT 1 REACTOR BUILDING. TO PREVENT RECURRENCE, THE WORK PLANNING GROUP WAS INSTRUCTED ON THE CORRECT WORDING TO BE USED WHEN REQUESTING WALL REMOVAL BETWEEN THE REACTOR BUILDING HVAC ZONES AND THE RAILROAD BAY. ALSO, THE REACTOR BUILDING HVAC OPERATING PROCEDURE WAS REVISED TO REQUIRE STEP-BY-STEP CONFIRMATION, VERIFICATION AND ADMINISTRATIVE TAGGING OF ISOLATION DAMPERS DURING RAILROAD BAY EVOLUTIONS. ALTHOUGH THE PROCEDURE DID NOT CONTRIBUTE TO THIS INCIDENT, IT IS FELT THAT THIS ENHANCEMENT ENSURES MORE POSITIVE CONTROL OVER RAILROAD BAY EVALUATIONS.

[328] SUSQUEHANNA 1 DOCKET 50-387 LER 87-027
ENTRY INTO L.C.O. FOR MODIFICATION IMPLEMENTATION.
EVENT DATE: 091087 REPORT DATE: 101287 NSSS: GE TYPE: BWR

(NSIC 206689) ON SEPTEMBER 10, 1987, WITH UNIT 1 OPERATING AT 100% POWER, TECHNICAL SPECIFICATION L.C.O. 3.0.3 WAS ENTERED FOR TWENTY-THREE MINUTES FROM 1009 TO 1032 HOURS. THIS WAS A PLANNED EVOLUTION FOR THE PURPOSE OF PERFORMING

WIRING MODIFICATIONS WHICH INCLUDED TEMPORARILY DECLARING THE DIVISION I ESSENTIAL 4160 VOLT BUSES INOPERABLE DUE TO DE-ENERGIZATION OF CONTROL POWER TO THEIR 84% DEGRADED GRID VOLTAGE PROTECTION CIRCUITRY. THESE MODIFICATIONS WERE PERFORMED TO INCREASE THE RELIABILITY OF THE PLANT AUXILIARY LOCA LOAD SHED LOGIC CIRCUITRY. IT WAS SUBSEQUENTLY RECOGNIZED THAT AN ALTERNATE METHOD OF PERFORMING THE MODIFICATIONS COULD HAVE PRECLUDED ENTRY INTO STS SECTION 3.0.3. SINCE BOTH DIVISION I BUSES WERE AFFECTED AND LOSS OF THE 84% PROTECTION IS NOT DESCRIBED IN THE TECHNICAL SPECIFICATIONS FOR TWO BUSES, L.C.O. 3.0.3 WAS ENTERED. TO REINFORCE THE INTENDED USE OF STS 3.0.3, A SUMMARY OF THE BASIS AND USE OF STS 3.0.3 WILL BE REVIEWED WITH SENIOR STATION MANAGERS AND SUPERVISORS AND WITH THE PLANT OPERATIONS REVIEW COMMITTEE, WHICH IS COMPOSED OF REPRESENTATIVES FROM ALL THE MAJOR PLANT STAFF FUNCTIONAL UNIT AREAS.

[329] SUSQUEHANNA 1 DOCKET 50-387 LER 87-028
PRIMARY CONTAINMENT ISOLATION VALVE CLOSES DUE TO SPURIOUS HIGH SHUTDOWN COOLING FLOW SIGNAL.
EVENT DATE: 091387 REPORT DATE: 101387 NSSS: GE TYPE: BWR

(NSIC 206672) ON SEPTEMBER 13, 1987, WHILE SHUTDOWN IN CONDITION 4, UNIT 1 EXPERIENCED AN UNANTICIPATED ENGINEERED SAFETY FEATURE (ESF) ACTUATION WITH THE CLOSURE OF A PRIMARY CONTAINMENT ISOLATION VALVE DUE TO A SPURIOUS FLOW SIGNAL. WHILE IN THE SHUTDOWN COOLING MODE, OPERATORS ATTEMPTED TO SWAP OPERATING RESIDUAL HEAT REMOVAL (RHR) PUMPS. WITH THE "A" PUMP IN OPERATION, FLOW WAS REDUCED AND THE "C" PUMP STARTED. AT 0605 HOURS, THE "A" PUMP WAS MANUALLY TRIPPED FROM THE CONTROL ROOM. AT APPROXIMATELY THE SAME TIME, THE OUTBOARD ISOLATION VALVE (F008), A PRIMARY CONTAINMENT ISOLATION VALVE ON THE SHUTDOWN COOLING SUCTION OF THE RHR PUMPS, STARTED TO CLOSE DUE TO A SPURIOUS HIGH FLOW SIGNAL. THIS CAUSED AN AUTOMATIC TRIP OF THE "C" RHR PUMP. FOLLOWING FILLING AND VENTING OF THE RHR LOOP BY OPERATIONS PERSONNEL, RHR PUMP "C" WAS RESTARTED AND SHUTDOWN COOLING WAS RE-ESTABLISHED AT 0625 ON 9/13/87. REVIEWS OF PLANT DATA AND DISCUSSIONS WITH OPERATIONS PERSONNEL CONCERNING PREVIOUS RHR PUMP SWAPS WHILE OPERATING IN THE SAME MODE CONCLUDED THAT SUCCESS OF THE TRANSFER DEPENDS HEAVILY UPON FLOW RATE. TO PREVENT RECURRENCE, SYSTEM OPERATING PROCEDURES WILL BE REVISED REQUIRING LOOP FLOW RATES TO BE REDUCED TO THE APPROPRIATE RATE PRIOR TO SWAPPING RHR PUMPS IN THE SHUTDOWN COOLING MODE. ALSO CALIBRATION OF PRESSURE SWITCHES WILL BE CHECKED.

[330] SUSQUEHANNA 2 DOCKET 50-388 LER 87-011
COMMON LOADS ON 125 VDC BATTERY WERE NOT TRANSFERRED PER PROCEDURE DUE TO PERSONNEL ERROR.
EVENT DATE: 092487 REPORT DATE: 102687 NSSS: GE TYPE: BWR

(NSIC 206854) ON SEPTEMBER 24, 1987 AT 1905 HOURS, IT WAS DISCOVERED THAT THE COMMON LOADS ON 125 VDC BATTERY 1D640 HAD NOT BEEN TRANSFERRED TO THEIR UNIT 2 SOURCE AS IS NECESSARY WHEN BATTERY 1D620 CAPACITY SURVEILLANCE TESTING IS PERFORMED UNIT 1 WAS IN SHUTDOWN FOR REFUELING AND UNIT 2 WAS OPERATING AT 100% POWER. THE TEST HAD BEGUN ON 9/22/87. UPON DISCOVERY, THE COMMON LOADS ON 1D640 WERE TRANSFERRED TO THEIR ALTERNATE SOURCES ON 9/24/87, EXCEPT THAT AN ERROR IN THE OPERATING PROCEDURE RESULTED IN THE 125VDC SUPPLY TO "D" DIESEL GENERATOR NOT BEING PROPERLY TRANSFERRED. THIS MIS-ALIGNMENT WAS DISCOVERED ON 9/29/87 AND WAS CORRECTED. THE CAUSE OF THIS INCIDENT HAS BEEN ATTRIBUTED TO PERSONNEL ERROR AND PROCEDURAL INADEQUACIES. THE OPERATING PROCEDURE ERRORS WERE CORRECTED. ADDITIONALLY, THE 18 MONTH BATTERY SURVEILLANCE TEST PROCEDURES ARE BEING REVISED TO REQUIRE DIRECT VERIFICATION SIGNOFFS FOR TRANSFER OF THE APPLICABLE 125 VDC LOADS DURING BATTERY TESTING.

[331] THREE MILE ISLAND 1 DOCKET 50-289 LER 86-013 REV 01
UPDATE ON REACTOR BUILDING PURGE EXHAUST AND PENETRATION PRESSURIZATION VALVE LEAKAGE.
EVENT DATE: 110786 REPORT DATE: 100387 NSSS: BW TYPE: PWR
VENDOR: HANCOCK CO.

(NSIC 206866) ON NOVEMBER 7, 1986, TMI-1 DECLARED A FAILURE AS-FOUND ON THE REACTOR BUILDING INTEGRATED LEAK RATE TEST (ILRT). THE FAILURE WAS DECLARED WHEN TRENDING CALCULATED BUILDING MASS-CHANGES INDICATED A LEAK RATE OF APPROXIMATELY 0.1 WEIGHT PERCENT PER DAY. THE ACCEPTANCE CRITERIA FOR AS-LEFT TESTING IS 0.075 WEIGHT PERCENT PER DAY. LEAKAGE FROM THE REACTOR BUILDING PURGE EXHAUST VALVE INTERSPACE INTO THE VENTED PENETRATION PRESSURIZATION SYSTEM WAS MEASURED BY LOCAL LEAK DETECTION METHODS AS BEING APPROXIMATELY 21% OF THE TOTAL OBSERVED BUILDING LEAKAGE. THE OUTER PURGE VALVE LEAKAGE WAS STOPPED BY READJUSTING THE VALVE SEATS AND LEAKAGE THROUGH THE CHECK VALVES IN THE PENETRATION PRESSURIZATION SYSTEM WAS ISOLATED. A SECOND ILRT WAS PERFORMED AND THE RESULTANT AS-LEFT BUILDING LEAK RATE WAS ACCEPTABLE. THE LOCAL LEAK RATE PROCEDURE PERFORMED PRIOR TO THE ILRT DID NOT IDENTIFY THE PENETRATION PRESSURIZATION SYSTEM CHECK VALVES AS REQUIRING TESTING.

[332] THREE MILE ISLAND 1 DOCKET 50-289 LER 87-004 REV 01
UPDATE ON REACTOR TRIP ON HIGH PRESSURE DUE TO OPERATOR ERROR DURING TESTING.
EVENT DATE: 050287 REPORT DATE: 110387 NSSS: BW TYPE: PWR

(NSIC 206874) TMI-1 WAS OPERATING AT 90% POWER WITH THE INTEGRATED CONTROL SYSTEM (ICS) IN FULL AUTOMATIC AND WITH NO COMPONENTS OR SYSTEMS INOPERABLE. A LICENSED OPERATOR PERFORMING THE "POWER RANGE CALIBRATION" SURVEILLANCE ON THE REACTOR PROTECTION SYSTEM (RPS), ACCIDENTLY TRANSFERRED THE "FLOW" SIGNAL INSTEAD OF THE "FLUX" SIGNAL DURING THE TEST. THE ICS RESPONDED TO THIS CHANGE AS DESIGNED BY REDUCING MAIN FEEDWATER FLOW AND WITHDRAWING RODS. THE RPS RESPONDED AS DESIGNED BY TRIPPING THE REACTOR ON HIGH PRESSURE AT 0751 HOURS ON MAY 2, 1987. NO OTHER SAFETY SYSTEMS WERE CALLED UPON DURING THIS TRANSIENT. THIS AUTOMATIC UNPLANNED RPS ACTUATION IS REPORTABLE UNDER 10CFR50.73A.2.IV. ROOT CAUSE WAS THE FAILURE OF THE OPERATOR TO RECOGNIZE THAT THE WRONG SWITCH HAD BEEN SELECTED. A SIMILAR INCIDENT OCCURRED MAY 27, 1976 DURING CYCLE 2 STARTUP PHYSICS TESTING. THE "FLUX" SIGNAL TO THE ICS WAS SELECTED WITHOUT PLACING THE PROPER STATIONS IN MANUAL. THE ICS WITHDREW CONTROL RODS AS DESIGNED. THE RPS TRIPPED THE REACTOR ON HIGH FLUX. CORRECTIVE ACTIONS ARE BEING TAKEN WHICH ARE TO: (1) IMPROVE AFFECTED PROCEDURES BY REQUIRING ADDITIONAL VERIFICATION THAT THE PROPER SWITCH HAS FUNCTIONED AS DESIRED. (2) PROVIDE GUIDANCE AND TRAINING TO ALL OPERATORS TO ENSURE THAT LESSONS LEARNED ARE CONSTRUCTIVELY USED TO HELP PREVENT FUTURE OCCURRENCES OF THIS NATURE.

[333] THREE MILE ISLAND 1 DOCKET 50-289 LER 87-007 REV 01
UPDATE ON INADVERTENT REACTOR PROTECTION SYSTEM ACTUATION ON HIGH PRESSURE DURING HEATUP DUE TO COGNITIVE OPERATOR ERROR.
EVENT DATE: 062587 REPORT DATE: 110287 NSSS: BW TYPE: PWR

(NSIC 206875) ON JUNE 25, 1987, TMI-1 HAD AN INADVERTENT REACTOR PROTECTION SYSTEM (RPS) ACTUATION ON HIGH PRESSURE DURING HEATUP. THE RPS ACTUATION WAS CAUSED BY EXCEEDING THE SHUTDOWN BYPASS REACTOR COOLANT SYSTEM (RCS) PRESSURE TRIP SETPOINT OF 1720 PSIG WHILE OPERATING 2 REACTOR COOLANT PUMPS (RCPS) IN ONE RCS LOOP AND 1 RCP IN THE OTHER RCS LOOP. THE RCS LOOP WITH 1 OPERATING RCP HAD A HIGHER PRESSURE DUE TO THE KNOWN DYNAMICS OF THE SYSTEM. THE ROOT CAUSE OF THIS EVENT IS THAT THE OPERATING CREW WAS NOT ATTENTIVE TO PRESSURE GAGE READINGS IN THE LOOP WITH 1 OPERATING RCP. THE RESULT WAS AN AUTOMATIC TRIP ON HIGH PRESSURE IN THIS LOOP WHICH CAUSED THE ONLY ROD GROUP WITHDRAWN AT THAT TIME TO FULLY INSERT INTO THE CORE. THE RPS WAS RESTORED TO ITS NORMAL HIGH PRESSURE TRIP SETPOINT AND THE RCS HEATUP WAS COMPLETED WITHOUT ANY FURTHER COMPLICATIONS.

AS CORRECTIVE ACTION, ALL OPERATORS ARE BEING COUNSELLED AND ADDITIONAL STEPS ARE BEING ADDED TO THE HEATUP PROCEDURE TO EMPHASIZE THIS CONDITION.

[334] THREE MILE ISLAND 1 DOCKET 50-289 LER 87-008
REACTOR TRIP FROM TURBINE TRIP DUE TO HIGH MOISTURE SEPARATOR LEVEL.
EVENT DATE: 091687 REPORT DATE: 101687 NSSS: BW TYPE: PWR
VENDOR: FISHER CONTROLS CO.

(NSIC 206759) TMI-1 WAS OPERATING AT 100% POWER WITH THE INTEGRATED CONTROL SYSTEM IN FULL AUTOMATIC. NO SYSTEMS OR COMPONENTS WERE INOPERABLE THAT AFFECTED THE POST TRIP RESPONSE. AT 1704 HOURS ON 9/16/87, THE TURBINE TRIPPED ON HIGH MOISTURE SEPARATOR LEVEL FOLLOWED BY A REACTOR TRIP. THE REACTOR PROTECTION SYSTEM (RPS) FUNCTIONED AS DESIGNED. THE PLANT POST TRIP RESPONSE WAS NORMAL. NO OTHER SAFETY SYSTEMS WERE ACTUATED. THE APPARENT ROOT CAUSE WAS THE FAILURE OF THE NON-SAFETY RELATED LEVEL CONTROLLER LC77E. LC77E CONTROLS THE HIGH LEVEL DUMP VALVE MO-V-2E AND MAY NOT HAVE OPENED THE VALVE FAR ENOUGH TO MAINTAIN WATER LEVEL. SIMILAR REACTOR TRIPS RESULTING FROM HIGH MOISTURE SEPARATOR LEVEL OCCURRED ON 1/4/86 AND 5/9/75. SEE THE TEXT FOR MORE DETAIL. CORRECTIVE AND PREVENTATIVE ACTIONS BEING TAKEN ARE: 1. HIGH LEVEL TRIP TIME DELAY RELAY WAS REPLACED. 2. DUMP VALVE LEVEL CONTROLLER (LC77E) WAS REPAIRED. 3. ALL SIX HIGH LEVEL TRIP SWITCHES WERE TESTED PRIOR TO STARTUP. 4. ALL SIX HIGH LEVEL DUMP VALVES WERE TESTED PRIOR TO STARTUP. 5. ALL SIX HIGH LEVEL ALARM SWITCHES WERE TESTED PRIOR TO STARTUP. 6. LEVELS OF ALL SIX MOISTURE SEPARATORS WERE MONITORED DURING STARTUP. 7. THE MOISTURE SEPARATOR DRAIN SYSTEM IS BEING EVALUATED FOR IMPROVEMENTS. 8. THE PREVENTATIVE MAINTENANCE PERFORMED ON THE HIGH LEVEL DUMP VALVE CONTROLLERS AND ALARM SWITCHES WILL BE IMPROVED.

[335] THREE MILE ISLAND 1 DOCKET 50-289 LER 87-009
RELEASE WITH LIQUID RELEASE MONITOR RM-L-6 DEFEATED DUE TO PERSONNEL ERROR.
EVENT DATE: 092787 REPORT DATE: 102787 NSSS: BW TYPE: PWR

(NSIC 206788) TMI-1 WAS OPERATING AT 100% POWER ON SEPTEMBER 27, 1987. AT 1658, SETPOINTS AND INTERLOCK FUNCTIONS FOR THE LIQUID RELEASE MONITOR RM-L-6 WERE BEING CHECKED BY I&C PERSONNEL IN ACCORDANCE WITH APPROVED LIQUID RELEASE PAPERWORK. RM-L-6 MONITORS DISCHARGE FROM THE WASTE EVAPORATOR CONDENSATE STORAGE TANKS. SUBSEQUENT TO THE I&C CHECKS, AT 1830, THE RELEASE FROM THE "A" TANK WAS INITIATED IN ACCORDANCE WITH THE OPERATING PROCEDURE. AT 1835, "ALERT" AND "ALARM" CONDITIONS WERE RECEIVED ON RM- L-6, AND CLEARED ABOUT 5 SECONDS LATER. UPON INVESTIGATING, IT WAS DISCOVERED THAT RM-L-6 HAD BEEN IN "DEFEAT" AND THEREFORE DID NOT AUTOMATICALLY TERMINATE THE RELEASE. THE DEFEAT SWITCH WAS PLACED IN "ENABLE" AT 1840 BY SHIFT SUPERVISOR DIRECTION AND THE SWITCH POSITION WAS INDEPENDENTLY VERIFIED. THE ROOT CAUSE FOR THE EVENT WAS PERSONNEL ERROR. SEVERAL PERSONNEL ERRORS CONTRIBUTED TO THIS EVENT WHICH WERE COMPOUNDED BY INADEQUATE PROCEDURAL GUIDANCE. THERE WERE NO SAFETY CONSEQUENCES FROM THIS EVENT. ENVIRONMENTAL IMPACT WAS MINIMAL BECAUSE THE RELEASE WAS SIGNIFICANTLY LESS THAN TECH. SPEC. QUARTERLY RELEASE LIMITS AND 10CFR20 MPC LIMITS, BASED ON SAMPLE ANALYSES OF THE TANK'S CONTENT PRIOR TO INITIATING THE RELEASE. REVIEW OF THE INCIDENT WITH I&C TECHNICIANS AND OPERATIONS PERSONNEL HAS BEEN COMPLETED. PROCEDURE ENHANCEMENTS ARE PLANNED.

[336] THREE MILE ISLAND 2 DOCKET 50-320 LER 87-010
FAILURE TO PROPERLY SURVEIL THE EPICOR II VENTILATION SYSTEM RADIATION MONITOR.
EVENT DATE: 071587 REPORT DATE: 110487 NSSS: BW TYPE: PWR
VENDOR: EBERLINE INSTRUMENT CORP.

(NSIC 207147) ON JULY 15, 1987, EPICOR II VENTILATION SYSTEM RADIATION MONITOR (ALC-RMI-18) WAS REPLACED WITH AN EBERLINE PING 2A TYPE MONITOR. THIS MODIFICATION RESULTED IN THE SAMPLE FLOWRATE MONITOR BEING CHANGED FROM A

PHOTOHELIC TYPE MONITOR TO A ROTAMETER AND PRESSURE GAUGE COMBINATION. TABLE 2.1-3B OF THE APPENDIX B TECH SPECS REQUIRES THE SAMPLE FLOWRATE MONITOR TO BE CALIBRATED SEMI-ANNUALLY. THESE SURVEILLANCES HAD BEEN PERFORMED PREVIOUSLY IN ACCORDANCE WITH SURVEILLANCE PROCEDURE 4221-SUR-3526.01. HOWEVER, THE ENGINEERING CHANGE AND IMPLEMENTING DOCUMENTATION DID NOT SPECIFY THE PERFORMANCE OF THIS PROCEDURE AS PART OF THE CRITERIA FOR PLACING THE REPLACEMENT MONITOR IN SERVICE. THE FAILURE TO DO SO RESULTED IN ALC-RMI-18 BEING IN AN INOPERABLE CONDITION, PER THE TECH SPECS, AND SHOULD HAVE REQUIRED COMPLIANCE WITH ACTION STATEMENT 36 OF TABLE 2.1-3A. ON OCTOBER 6, 1987, DURING THE PREPARATION OF A PROCEDURE CHANGE TO 4221-SUR-3526.01, THE SAFETY REVIEW GROUP (SRG) WAS CONSULTED AND IDENTIFIED THE ABOVE CONDITION. COMPLIANCE WITH THE ABOVE REFERENCED ACTION STATEMENT WAS THE FAILURE TO RECOGNIZE AND COMPLY WITH THE ACTION STATEMENT UPON THE INSTALLATION OF THE REPLACEMENT MONITOR. THIS EVENT IS REPORTABLE PER 10 CFR 50.73(A)(3)(I)(B) DUE TO A CONDITION PROHIBITED BY THE PLANT'S TECH SPECS.

[337] TROJAN DOCKET 50-344 LER 87-018 REV 01
UPDATE ON VALVE PACKING LEAKAGE EXCEEDED FSAR ASSUMED LEAKAGE.
EVENT DATE: 050987 REPORT DATE: 100587 NSSS: WE TYPE: PWR
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 206640) DURING LOCAL LEAK RATE TESTING (LLRT), ON MAY 9, 1987, THE CONTAINMENT SPRAY AND RESIDUAL HEAT REMOVAL RECIRCULATION SUCTION VALVES OUTSIDE CONTAINMENT (MO2052B AND MO-8811B) EXHIBITED PACKING LEAKS. THE LEAKAGE EXCEEDED THE 1580 CUBIC CENTIMETERS PER HOUR ASSUMED IN THE FINAL SAFETY ANALYSIS REPORT FOR POST-ACCIDENT RECIRCULATION LEAKAGE. THE CAUSE OF THE VALVE PACKING LEAKS WAS ATTRIBUTED TO NORMAL PACKING DEGRADATION. THE VALVE PACKINGS WERE TIGHTENED AND TESTED SATISFACTORILY. THE VALVE PACKING PROGRAM FOR VALVES IN THE RECIRCULATION FLOW PATH WILL BE REVIEWED FOR CHANGES NECESSARY TO PREVENT RECURRENCE. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY. THE LLRT OF THIS PENETRATION WAS PERFORMED WITH AIR AT 60 PSIG. ACTUAL LEAKAGE IN THE EVENT OF OPERATION OF THE RECIRCULATION SUMP FLOW PATH WOULD HAVE BEEN SIGNIFICANTLY LESS BECAUSE: (1) THE LEAKAGE WOULD HAVE BEEN LIQUID INSTEAD OF GAS, AND (2) THE DURATION OF THE PEAK POST-ACCIDENT CONTAINMENT PRESSURE OF 60 PSIG IS BRIEF, FOLLOWED BY A RAPID DECREASE IN PRESSURE.

[338] TROJAN DOCKET 50-344 LER 87-014 REV 01
UPDATE ON CHARGING PUMP CORROSION DUE TO MANUFACTURING DEFICIENCY.
EVENT DATE: 052387 REPORT DATE: 103087 NSSS: WE TYPE: PWR
VENDOR: PACIFIC PUMPS

(NSIC 206803) ON MAY 23, 1987 DURING INSPECTION OF THE "A" CENTRIFUGAL CHARGING PUMP (CCP), A PORTION OF THE STAINLESS STEEL CLADDING ON THE INSIDE SURFACE OF THE PUMP CASING EXHIBITED CORROSION. CORROSION OF THE PUMP CASING WAS THROUGH THE STAINLESS STEEL CLADDING INTO THE CARBON STEEL BASE MATERIAL. INSPECTION OF THE "B" CCP REVEALED SIMILAR CORROSION. THE CAUSE OF THIS EVENT WAS A MANUFACTURING DEFICIENCY. CORROSION OBSERVED AT THE PUMP CASING DISCHARGE NOZZLE WAS ATTRIBUTED TO A CLADDING BREAKTHROUGH DURING FINAL MACHINING. CORROSION OBSERVED AT THE PUMP CASING INLET END WAS ATTRIBUTED TO EITHER OVERMACHINING OF THE CLADDING OR INADEQUATE OVERLAY OF TWO ADJACENT WELD BEADS. THE "A" AND "B" CCP CASINGS WERE REPLACED. THERE ARE NO OTHER SAFETY-RELATED PUMPS IN THE EMERGENCY CORE COOLING SYSTEM WITH STAINLESS STEEL CLADDING. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[339] TROJAN DOCKET 50-344 LER 87-027
BORON INJECTION TANK RELIEF VALVE LEAKED GREATER THAN FSAR ASSUMED LIMITS.
EVENT DATE: 092787 REPORT DATE: 102787 NSSS: WE TYPE: PWR
VENDOR: CROSBY VALVE & GAGE CO.

(NSIC 206804) ON SEPTEMBER 27, 1987, BORON INJECTION TANK (BIT) OUTLET RELIEF VALVE PSV-8852 WAS EXHIBITING SEAT LEAKAGE OF 207 CUBIC CENTIMETERS (CC)/MINUTE (12,420 CC/HOUR). THE VALVE DISCHARGE WAS DIRECTED TO AN OPEN FLOOR DRAIN IN THE AUXILIARY BUILDING. VALVE PSV-8852 IS LOCATED IN THE EMERGENCY CORE COOLING SYSTEM FLOW PATH AND THE LEAKAGE EXCEEDED THE 1850 CC/HOUR ASSUMED IN THE FINAL SAFETY ANALYSIS REPORT FOR SYSTEMS OUTSIDE CONTAINMENT WHICH COULD CONTAIN RADIOACTIVE WATER FOLLOWING A DESIGN BASIS ACCIDENT (DBA). THIS LEAKAGE COULD HAVE RESULTED IN THYROID DOSES TO CONTROL ROOM OPERATORS FOLLOWING A DBA EXCEEDING THOSE STATED IN THE FSAR. THE CAUSE OF THE EVENT WAS LEAKAGE PAST THE SEAT OF VALVE PSV-8852. A CONTRIBUTING CAUSE WAS THAT THE DISCHARGE OF PSV-8852 WAS DIRECTED TO AN OPEN FLOOR DRAIN INSTEAD OF A CLOSED SYSTEM. THE PLANT WAS SHUT DOWN AND PSV-8852 WAS REMOVED AND REPLACED WITH A BLIND FLANGE. A PRELIMINARY REVIEW FOR OTHER LEAK-PATHS IN SYSTEMS OUTSIDE CONTAINMENT WHICH COULD CONTAIN RADIOACTIVE WATER FOLLOWING A DBA WAS PERFORMED. NO SIMILAR LEAK PATHS WERE IDENTIFIED. A DETAILED REVIEW WILL BE COMPLETED BY OCTOBER 30, 1987. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[340] TROJAN DOCKET 50-344 LER 87-028
CONTAINMENT VENTILATION ISOLATION DUE TO SPURIOUS ACTUATION OF PROCESS RADIATION MONITOR.
EVENT DATE: 100787 REPORT DATE: 110687 NSSS: WE TYPE: PWR
VENDOR: VICTOREEN INSTRUMENT DIVISION

(NSIC 206930) ON OCTOBER 7, 1987 DURING PERIODIC OPERATING TEST (POT) 20-1, "CONTROL ROOM EMERGENCY VENTILATION PERFORMANCE", CONTAINMENT AIR PARTICULATE PROCESS RADIATION MONITOR (PRM) 1A SPIKED HIGH, CAUSING A CONTAINMENT VENTILATION ISOLATION. THE CAUSE OF THE SPURIOUS ACTUATION OF PRM 1A WAS HIGH TEMPERATURE IN CONTROL ROOM PANEL C41 WHERE PRM 1A IS LOCATED. THE HIGH TEMPERATURE IN C41 WAS DUE TO THE ELEVATED AMBIENT CONTROL ROOM TEMPERATURE DURING POT-20-1 (83 F DURING THE POT AS OPPOSED TO 75F NORMALLY). THE DOORS TO PANEL C41 WERE OPENED TEMPORARILY TO IMPROVE THE VENTILATION IN C41. LONG TERM CORRECTIVE ACTION CONSISTS OF ADDING CHILLERS TO THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM DURING THE 1988 REFUELING OUTAGE. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[341] TURKEY POINT 3 DOCKET 50-250 LER 87-023
SAFETY INJECTION AND REACTOR TRIP DUE TO FAILED HIGH STEAM FLOW CHANNELS AND AN ACTUAL LOW AVERAGE REACTOR COOLANT TEMPERATURE AFTER A LOAD REDUCTION FOR TURBINE TESTING.
EVENT DATE: 091387 REPORT DATE: 101387 NSSS: WE TYPE: PWR
VENDOR: ROSEMOUNT, INC.
TARGET ROCK CORP.

(NSIC 206663) ON SEPTEMBER 13, 1987, UNIT 3 EXPERIENCED A SAFETY INJECTION (SI) SYSTEM ACTUATION AND SUBSEQUENT REACTOR TRIP FROM APPROXIMATELY 5% POWER. ON SEPTEMBER 12, 1987, 1 OF 2 HIGH STEAM FLOW PROTECTION CHANNELS ON THE 3A STEAM GENERATOR (SG), FT-3-475, FAILED LOW AND WAS TAKEN OUT OF SERVICE AS PER PROCEDURE. ON SEPTEMBER 13, 1987, A LOAD REDUCTION WAS COMMENCED FOR UNIT 3 TO PERFORM A TURBINE OVERSPEED TEST AS PER OPERATING PROCEDURE (OP) 8004.1, TURBINE GENERATOR OVERSPEED TRIP TEST. THE TEST WAS BEGUN WITH THE LOW REACTOR COOLANT SYSTEM (RCS) AVERAGE TEMPERATURE (TAVE) INPUT TO THE SI LOGIC FOR A SI SIGNAL ON LOW TAVE ON 2 OUT OF 3 RCS LOOPS COINCIDENT WITH HIGH STEAM FLOW ON 2 OUT OF OUT OF 3 SGS, ALARMED ON THE A, B AND C RCS LOOPS. UPON COMPLETION OF THE TEST A HIGH STEAM FLOW TRANSMITTER FOR THE 3C SG, SPIKED, BEGAN CYCLIC OPERATION AND SUBSEQUENTLY FAILED WHICH LOCKED IN THE HIGH STEAM FLOW SIGNAL FOR THE 3C SG AND COMPLETED THE LOGIC FOR A SI ACTUATION AND SUBSEQUENT REACTOR TRIP. THE CAUSE OF THE TRIP WAS COMPONENT FAILURE ALONG WITH PERSONNEL ERROR. AN EVENT RESPONSE TEAM WAS FORMED TO DETERMINE ROOT CAUSE AND CORRECTIVE ACTIONS. A REVIEW WAS PERFORMED

TO ASSESS PROPER PLANT OPERATION WHICH DETERMINED THAT THE PLANT RESPONSE TO THE TRANSIENT WAS AS EXPECTED.

[342] TURKEY POINT 3 DOCKET 50-250 LER 87-024 REV 01
UPDATE ON REACTOR CONTROLS MANIPULATED BY A NON-LICENSED PERSON UNDER THE DIRECT SUPERVISION OF A LICENSED OPERATOR.
EVENT DATE: 091387 REPORT DATE: 102387 NSSS: WE TYPE: PWR

(NSIC 206778) ON SEPTEMBER 13, 1987, AT 0300, WITH UNIT 3 AT 30% POWER, A NON-LICENSED PERSON UNDER THE DIRECT SUPERVISION OF A REACTOR CONTROL OPERATOR (RCO), TURNED THE REACTOR CONTROL MAKE-UP SWITCH TO START ON TWO OCCASIONS. THIS RESULTED IN A 30 GALLON, THEN A 20 GALLON DILUTION OF THE REACTOR COOLANT SYSTEM (RCS). AT THIS TIME, A FLUX MAP WAS BEING RUN. NEGATIVE REACTIVITY WAS BEING ADDED DUE TO XENON BUILDUP. IN ORDER TO MINIMIZE FLUX DISTORTION, CONTROL ROD MOTION WAS BEING MINIMIZED, AND TO COUNTER THE NEGATIVE REACTIVITY ADDITION, THE RCS WAS BEING DILUTED OFTEN. THE RCO DIRECTLY SUPERVISED BOTH DILUTIONS. UNEVALUATED INFORMATION REGARDING OTHER CONTROL MANIPULATIONS BY THE NON-LICENSED PERSON IN QUESTION IS BEING INVESTIGATED BY FPL. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR, IN THAT THE RCO FAILED TO COMPLY WITH THE REQUIREMENTS OF 10CFR50.54(I) AND 10CFR55.3. CONTRIBUTING TO THE EVENT WERE INADEQUATE PROCEDURES AND TRAINING ON THE REQUIREMENTS OF 10CFR50.54(I) AND 10CFR55.3. THE PLANT SUPERVISOR-NUCLEAR (PSN) COUNSELED THE RCO AND THE NON-LICENSED PERSON. THE PSN DISCUSSED THE EVENT AT THE SHIFT TURNOVER MEETING. MEMOS EXPLAINING THE REQUIREMENTS OF 10CFR50.54(I) WERE ISSUED. ALL LICENSED OPERATORS HAVE BEEN REQUIRED TO READ AND SIGN THE MEMO PRIOR TO ASSUMING SHIFT RESPONSIBILITY.

[343] TURKEY POINT 3 DOCKET 50-250 LER 87-025
UNDOCUMENTED SURVEILLANCE OF COOLANT LOOP OPERABILITY DUE TO INADEQUATE PROCEDURE.
EVENT DATE: 091487 REPORT DATE: 101387 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: TURKEY POINT 4 (PWR)

(NSIC 206706) ON SEPTEMBER 14, 1987, THE QUALITY ASSURANCE DEPARTMENT NOTED THAT THE SURVEILLANCE REQUIREMENTS OF TECH SPEC 4.1-2 ITEM 18 HAD NOT BEEN MET. THIS TECH SPEC REQUIRES THAT WITH AVERAGE COOLANT TEMPERATURE LESS THAN 350 DEGREES F, A SECOND COOLANT LOOP IS TO BE VERIFIED OPERABLE ONCE EVERY 7 DAYS. TECH SPEC 3.4.1.E REQUIRES THAT "WITH AVERAGE COOLANT TEMPERATURE LESS THAN 350 DEGREES F, AT LEAST 2 COOLANT LOOPS SHALL BE OPERABLE OR IMMEDIATE CORRECTIVE ACTION MUST BE TAKEN TO RETURN TWO COOLANT LOOPS TO OPERABLE AS SOON AS POSSIBLE." EVEN THOUGH NO DOCUMENTED SURVEILLANCE IN ACCORDANCE WITH TECH SPEC 4.1-2 ITEM 18 WAS BEING PERFORMED, COMPLIANCE WITH TECH SPEC 3.4.1.E ASSURES AN ONGOING OBSERVATION OF COOLANT LOOP OPERABILITY FOR TWO COOLANT LOOPS. THE CAUSE OF THE UNDOCUMENTED SURVEILLANCE WAS AN INADEQUATE PROCEDURE. PROCEDURES 3/4-OSP-201.1, "RCO DAILY LOGS", WERE UPGRADES OF THE RCO LOG SHEET. UPON COMPLETION OF 3/4-OSP-201.1, THE OLD RCO LOG SHEETS, WHICH ADDRESSED THIS SURVEILLANCE, WERE CANCELLED. THE UPGRADED PROCEDURES FAILED TO INCLUDE THIS SURVEILLANCE REQUIREMENT. TRANSITION DOCUMENTS (TD) NOT GENERATED, AS THE OLD RCO LOG SHEETS WERE NOT CONSIDERED TO BE PROCEDURES.

[344] TURKEY POINT 3 DOCKET 50-250 LER 87-027
TRANSPORT OF CONTAMINATED SEAL INJECTION FILTER CAUSES ACTUATION OF PROCESS RADIATION MONITOR R-11 RESULTING IN CONTAINMENT VENT AND CONTROL ROOM VENTILATION ISOLATION.
EVENT DATE: 093087 REPORT DATE: 103087 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: TURKEY POINT 4 (PWR)
VENDOR: TRACER LAB

(NSIC 206779) ON SEPTEMBER 30, 1987, AT 2115, UNIT 3 WAS IN COLD SHUTDOWN (MODE 5). THE 3B SEAL WATER INJECTION FILTER, WHICH WAS CONTAMINATED, WAS REPLACED AND

TRANSPORTED THROUGH THE AUXILIARY BUILDING PAST THE SAMPLING CABINET FOR THE RADIOACTIVE PARTICULATE CONTAINMENT RADIATION MONITOR (PRM) R- 11. THIS CAUSED R-11 TO INCREASE ABOVE ITS SETPOINT WHICH RESULTED IN ACTUATION OF THE CONTAINMENT VENT AND CONTROL ROOM VENTILATION ISOLATION LOGIC (EIIS: IL,JN). THE CONTAINMENT VENT ISOLATED AND THE CONTROL ROOM VENTILATION ISOLATED AND SWITCHED OVER TO THE RECIRCULATION MODE, PER DESIGN. A NEW HEALTH PHYSICS PROCEDURE FOR REMOVAL AND TRANSPORTATION OF USED CHEMICAL AND VOLUME CONTROL SYSTEM FLUID FILTERS HAS BEEN WRITTEN WHICH INCORPORATES INSTRUCTIONS FOR TRANSPORTING USED FILTERS VIA A TRAVEL PATH THAT DOES NOT PASS BY THE R-11 SAMPLE CABINET TO AVOID THE ACTUATION OF THE R-11 SETPOINT TRIP. ADDITIONALLY, MAINTENANCE PROCEDURE 0-PMM-047.10, CHEMICAL AND VOLUME CONTROL CHARGING AND LETDOWN SYSTEMS FLUID FILTER REPLACEMENT. HAS BEEN REVISED TO INCLUDE A SIGN OFF STEP REQUIRING NOTIFICATION OF THE CONTROL ROOM BEFORE TRANSPORTING THE USED FILTER FROM THE AREA.

[345] TURKEY POINT 3 DOCKET 50-250 LER 87-026
INADEQUATE TESTING OF THE HYDROGEN RECOMBINER LINE AFTER PIPING.
EVENT DATE: 100287 REPORT DATE: 110287 NSSS: WE TYPE: PWR

(NSIC 206872) DURING THE UNIT 3 REFUELING OUTAGE WHILE THE UNIT WAS IN MODE 6, PLANT CHANGE MODIFICATION (PC/M) 81-157 WAS IMPLEMENTED TO CHANGE THE MOTOR OPERATOR ROTOR FOR MOV-3-1426, FROM A 2 POLE ROTOR TO A 4 POLE ROTOR. THE MODIFICATION REQUIRED A LARGER MOTOR COVER TO BE INSTALLED BUT IT COULD NOT BE SET IN PLACE DUE TO INTERFERENCE FROM THE HYDROGEN RECOMBINER LINE IN CLOSE PROXIMITY TO MOV-3-1426. THE LINE WAS REROUTED PER CHANGE REQUEST 15. THE WELDED JOINT WAS PENETRANT TESTED AND VISUALLY INSPECTED. THE LINE IS OPEN ENDED AND CANNOT BE ISOLATED. IT WAS INCORRECTLY DETERMINED THAT A PRESSURE TEST WAS NOT REQUIRED. UNIT 3 ENTERED MODE 4 ON AUG 21, 1987 AND CONTINUED THE UNIT STARTUP INTO POWER OPERATION. ON SEPT 25, 1987 THE UNIT WAS SHUTDOWN DUE TO OPERATING PROBLEMS UNRELATED TO THIS EVENT. THE ERROR REGARDING EXEMPTION FROM PRESSURE TESTING WAS DISCOVERED DURING THIS SHUTDOWN. THE HYDROGEN RECOMBINER LINE WAS SATISFACTORILY PRESSURE TESTED DURING THE SHUTDOWN. THE PLANT CONSTRUCTION ADMINISTRATIVE SITE PROCEDURE (ASP) 2, HAS BEEN REVISED TO INCLUDE THE IN SERVICE INSPECTION COORDINATOR IN THE REVIEW CYCLE OF PLANT CHANGE MODIFICATIONS TO CLASS A, B, C SYSTEMS TO ENSURE ADEQUATE TESTING IS PERFORMED AFTER ANY MODIFICATION.

[346] TURKEY POINT 3 DOCKET 50-250 LER 87-028
MISSED SURVEILLANCE OF CONTROL ROD POSITIONS DUE TO PERSONNEL ERROR.
EVENT DATE: 101587 REPORT DATE: 111687 NSSS: WE TYPE: PWR

(NSIC 206906) ON OCTOBER 15, 1987, DURING A QA REVIEW OF THE REACTOR CONTROL OPERATOR (RCO) LOGSHEETS, IT WAS DISCOVERED THAT THE TECHNICAL SPECIFICATION (TS) REQUIRED SURVEILLANCE OF CONTROL ROD POSITIONS (TS TABLE 4.1-1, ITEMS 9 AND 10) WAS MISSED DURING PEAK SHIFT ON SEPTEMBER 12, 1987. TS TABLE 4.1-1, ITEMS 9 AND 10 REQUIRE A SHIFTLY CHECK OF THE ANALOG ROD POSITIONS AND ROD POSITION BANK COUNTERS. THIS SURVEILLANCE WAS LAST PERFORMED AT 0730 ON SEPTEMBER 12, 1987 AND WAS PERFORMED AGAIN WITH SATISFACTORY RESULTS AT 0125 ON SEPTEMBER 13, 1987. THE CAUSE OF THE MISSED SURVEILLANCE WAS PERSONNEL ERROR. AT THE TIME THE SURVEILLANCE WAS MISSED, UNIT 3 WAS BEING BROUGHT ON LINE AND. POWER INCREASED TO 30%. BECAUSE THESE EVOLUTIONS REQUIRED THE RCO'S CONSTANT ATTENTION, HE DID NOT PAY ADEQUATE ATTENTION TO HIS NORMAL LOGS. A CONTRIBUTING CAUSE OF THE MISSED SURVEILLANCE WAS THE ASSISTANT PLANT SUPERVISOR-NUCLEAR (APSN) DID NOT NOTICE THE OMISSION DURING HIS NORMAL REVIEW OF THE RCO LOGS. THIS IN TURN WAS DUE PARTIALLY TO THE LARGE NUMBER OF LOGS SHEETS REQUIRING APSN REVIEW AND HIS REQUIRED ATTENTION TO THE EVOLUTIONS IN PROGRESS. THE RCO AND APSN WERE COUNSELED CONCERNING THE NEED FOR PROPER ATTENTION TO NORMAL DUTIES DURING PLANT EVOLUTIONS AND THE RCO LOGS ARE BEING REVIEWED TO SEE WHERE IMPROVEMENTS CAN BE MADE IN THE AREAS OF FORMAT AND VOLUME REDUCTION TO FACILITATE EASIER REVIEW.

[347] TURKEY POINT 3 DOCKET 50-250 LER 87-029
WASTE GAS SYSTEM OPERATED IN A CONFIGURATION NOT DESCRIBED IN THE FINAL SAFETY ANALYSIS REPORT DURING VOLUME CONTROL TANK PURGE OPERATIONS.
EVENT DATE: 101587 REPORT DATE: 111687 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: TURKEY POINT 4 (PWR)

(NSIC 206907) ON 10/15/88, A SAFETY EVALUATION OF THE PAST OPERATION OF THE WASTE GAS SYSTEM DURING VOLUME CONTROL TANK (VCT) PURGE OPERATIONS DETERMINED THAT THE CONFIGURATION WAS NOT AS DESCRIBED IN THE FSAR. INVESTIGATIONS REVEALED THAT AN AUX. BUILDING GASSING PROBLEM HAD EXISTED DUE TO A LEAK PATH THROUGH THE A GAS STRIPPER. WHEN THE GAS STRIPPERS WERE TAKEN OUT OF SERVICE IN 1978, THE GAS STRIPPER TO VENT HEADER ISOLATION VALVES WERE LEFT OPEN. WHEN THE AIRBORNE CONTAMINATION CONDITIONS WERE EXPERIENCED, PROCEDURES WERE REVISED TO ALLOW THE OPENING OF THE VENT HEADER TO COVER GAS HEADER CROSS-CONNECT LINE VALVE (4627) DURING VCT PURGING. THIS HELPED ALLEVIATE THE PROBLEM BUT BYPASSED THE VENT HEADER REGULATOR, THUS PERMITTING THE POTENTIAL PRESSURIZATION OF THE VENT HEADER BEYOND THE FSAR BASIS OF 3 PSIG. PROCEDURES WERE REVISED TO ENSURE THAT VALVE 4627 WAS CLOSED. IN ADDITION, THE A GAS STRIPPER TO VENT HEADER ISOLATION VALVE WAS ENERGIZED TO CLOSE AND ISOLATE THE LEAKAGE PATH, THUS RESTORING THE ORIGINAL FLOW PATH FOR THE VCT PURGING MODE OF OPERATION. THE IMPROPER OPERATION OF THE WASTE GAS SYSTEM WAS DUE TO AN INADEQUATE PROCEDURE REVISION. THE REVISED PROCEDURE DID NOT ASSURE THAT THE FSAR REQUIREMENTS WERE BEING MET. THE PROCEDURE UPGRADE PROJECT HAS GENERATED ADDITIONAL ADMINISTRATIVE CONTROL PROCEDURES WHICH INCREASE THE RETURN.

[348] TURKEY POINT 4 DOCKET 50-251 LER 87-022
MISINTERPRETATION OF TECH SPEC 3.6.B.3 AND 3.6.C.3 WHICH ALLOWED A BORIC ACID STORAGE TANK TO BE CLASSIFIED AS IN SERVICE WHEN IT WAS BELOW THE TECH SPEC LEVEL LIMIT.
EVENT DATE: 091487 REPORT DATE: 101687 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: TURKEY POINT 3 (PWR)

(NSIC 206707) AT 0545 SEPT. 14, 1987 WITH UNIT 4 OPERATING AT APPROXIMATELY 90% POWER AND UNIT 3 IN HOT STANDBY, SURVEILLANCE TEST OSP-204.2 WAS PERFORMED ON THE BORIC ACID STORAGE TANKS (BAST) LEVEL INDICATION. THE A AND B BAST WERE FOUND TO HAVE A LEVEL MISMATCH BETWEEN THE LOCAL AND THE CONTROL ROOM INDICATION IN EXCESS OF THE ALLOWABLE 10% VALUE. THE C BAST WAS FOUND TO CONTAIN ONLY 3000 GALLONS WHICH WAS BELOW THE 3080 GALLON VALUE REQUIRED BY TECHNICAL SPECIFICATION (TS) 3.6.D. ALL 3 BASTS WERE DECLARED INOPERABLE PLACING UNIT 4 IN T S 3.0.1 AND PREVENTING UNIT 3 FROM ENTERING MODE 2. AT 0845 THE C BAST WAS VERIFIED TO BE WITHIN THE LIMITS OF TS 3.6 FOR LEVEL AND BORON CONCENTRATION AND UNIT 4 POWER ASCENSION WAS RECOMMENCED. AT 1045 THE A AND B BAST LEVEL INDICATORS WERE RETURNED TO SERVICE. FURTHER INVESTIGATION INTO THIS INCIDENT ON SEPT. 30 ,1987, REVEALED THAT THE C BAST HAD ACTUALLY BEEN BELOW THE TS LEVEL LIMIT SINCE 1200 SEPT. 9, 1987, DUE TO A MISINTERPRETATION OF TS 3.6 THAT RESULTED IN CONSIDERING A TANK TO BE OPERABLE WHEN IT WAS BELOW THE 3080 GALLON LEVEL AS LONG AS THE CUMULATIVE TOTAL BETWEEN ALL THREE BASTS MET THE TS REQUIREMENT. THE PROPER TANK LINEUP ON UNIT 4 WAS IMMEDIATELY VERIFIED AND CLARIFICATION LETTER ISSUED TO OPERATIONS.

[349] TURKEY POINT 4 DOCKET 50-251 LER 87-023
PROCESS RADIATION MONITOR TRENDS HIGH DUE TO JAMMED PAPER DRIVE CAUSING CONTROL ROOM VENTILATION AND CONTAINMENT VENT ISOLATION.
EVENT DATE: 091587 REPORT DATE: 101587 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: TURKEY POINT 3 (PWR)
VENDOR: TRACER LAB

(NSIC 206664) ON SEPTEMBER 15, 1987, AT 0410, WITH UNIT 4 AT 100% POWER, PROCESS RADIATION MONITOR (PRM) R-11, THE RADIOACTIVE PARTICULATE CONTAINMENT RADIATION

MONITOR, TRENDED HIGH OVER A SHORT PERIOD, ACTUATING THE CONTAINMENT VENT AND CONTROL ROOM VENTILATION ISOLATION LOGIC (EIIS:IL,JN). THE CONTROL ROOM VENTILATION ISOLATED AND SWITCHED INTO THE RECIRCULATION MODE, PER DESIGN. AT THIS TIME, THE CONTAINMENT VENT WAS ALREADY ISOLATED. JUST PRIOR TO THIS INCIDENT, THE FILTER PAPER WAS REPLACED. SHORTLY AFTER THIS, THE PAPER DRIVE JAMMED. THIS CAUSED THE SENSED LEVEL OF RADIOACTIVITY TO TREND UP, RESULTING IN CONTROL ROOM VENTILATION ISOLATION. THE PAPER WAS REPLACED, AND THE PAPER DRIVE TENSION WAS ADJUSTED. AT THIS TIME, THE DETECTOR WAS FOUND TO BE READING LOW, AND IT WAS REPLACED. R-11 WAS RETURNED TO SERVICE ON SEPTEMBER 21, 1987. ENHANCEMENTS TO IMPROVE PAPER DRIVE RELIABILITY ARE BEING CONSIDERED. BASED ON THE ABOVE, THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED. ADDITIONAL DETAILS: SIMILAR OCCURRENCES: LERS 251-86-030, 250-87-005, 250- 87-008, 251-87-021 MANUFACTURER: R-11: TRACER LAB INSTRUMENT. MODEL NUMBER: MAP-1B.

[350] VERMONT YANKEE DOCKET 50-271 LER 87-007 REV 01
UPDATE ON 1987 APPENDIX J TYPE B AND C LEAK RATE FAILURE DUE TO SEAT LEAKAGE.
EVENT DATE: 081287 REPORT DATE: 103087 NSSS: GE TYPE: BWR
VENDOR: ALLIS CHALMERS
WALWORTH COMPANY

(NSIC 206894) ON 8/12/87, 8/18/87 AND 8/23/87 WHILE PERFORMING TYPE C LEAK RATE TESTING WITH THE PLANT SHUT DOWN FOR THE 1987 REFUEL OUTAGE MAIN STEAM DRAIN VALVE MSD-77, LIQUID RADWASTE VALVES LRW-83, LRW-94, LRW-95, AND PRIMARY CONTAINMENT ATMOSPHERIC CONTROL VALVES PCAC-8,9,10,23 AND PCAC-6,7,6A,6B,7A,7B (EIIS-SB,WK,BB) WERE FOUND TO HAVE SEAT LEAKAGE ABOVE THAT PERMITTED BY TECH SPECS SECTION 3.7.A.4 (NOTE: THE PCAC VALVES ARE TESTED IN THE GROUPS LISTED. ONE VALVE IN THE FIRST GROUP, PCAC-8, AND TWO VALVES IN THE SECOND GROUP PCAC-6 AND PCAC-7, CAUSED THE GROUP LEAKAGE TO EXCEED THE ACCEPTANCE CRITERIA. ALSO ON 8/20/87 THE SUM TOTAL TYPE B (PENETRATIONS) AND TYPE C (VALVES) LEAKAGE EXCEEDED THAT ALLOWED BY 10CFR50 APPENDIX J. APPENDIX J LIMITS ALLOWABLE TOTAL B AND C PENETRATION LEAKAGE TO 0.60 LA. VERMONT YANKEE HAS PERFORMED MAINTENANCE ON ALL OF THE VALVES THAT WERE FOUND TO BE LEAKING AND DETERMINED THE CAUSE OF FAILURE. THE VALVES WERE RETESTED TO VERIFY THAT SEAT LEAKAGE WAS WITHIN ALLOWABLES PRIOR TO PLANT STARTUP FOLLOWING THE 1987 REFUELING OUTAGE.

[351] VERMONT YANKEE DOCKET 50-271 LER 87-008 REV 01
UPDATE ON LOSS OF NORMAL POWER DURING SHUTDOWN DUE TO ROUTING ALL OFF-SITE POWER SOURCES THROUGH ONE BREAKER.
EVENT DATE: 081787 REPORT DATE: 110487 NSSS: GE TYPE: BWR

(NSIC 206957) AT APPROXIMATELY 1400 HOURS ON 8/17/87 WHILE THE PLANT WAS IN A REFUELING OUTAGE AND ALL OFF-SITE POWER WAS BEING ROUTED THROUGH ONE SET OF BREAKERS, AN INTERRUPTION ON THE GRID CAUSED THE PLANT TO LOSE NORMAL POWER SUPPLIES. THE EMERGENCY DIESEL GENERATORS (EDG) RESPONDED AS REQUIRED AS DID OTHER ENGINEERED SAFETY SYSTEMS. WHEN THE EDG'S STARTED, THEY WERE ABLE TO SUPPLY POWER TO ALL NECESSARY SYSTEMS. THREE PUMPS THAT STARTED IMMEDIATELY AFTER THE EDG'S WERE 2 SERVICE WATER PUMPS AND THE ELECTRIC FIRE PUMP. THE STARTING OF THESE THREE PUMPS, IN ADDITION TO THE DIESEL DRIVEN FIRE PUMP CAUSED A PRESSURE SURGE WHICH RUPTURED A TEMPORARY PIPING SYSTEM FABRICATED FROM 2" SCHEDULE 80 PVC PIPING (EIIS = KP). THE PVC PIPING WAS MADE BY ESLON. THE RUPTURED PIPE SPILLED ABOUT 2000 GALLONS OF RIVER WATER ONTO THE REFUELING FLOOR OF THE REACTOR BUILDING. AS A RESULT OF THE SPILL, THIS WATER WAS COMMUNICATED THROUGH THE FLOOR DRAIN SYSTEM WHICH RESULTED IN CONTAMINATING LOCAL AREAS OF THE REACTOR BUILDING. MINOR SEEPAGES THROUGH THE INTERFACE BETWEEN THE REACTOR BUILDING REFUEL FLOOR PANELING AND THE REACTOR BUILDING EXTERIOR WALLS WERE DETECTED. NO EQUIPMENT WAS DAMAGED AS A RESULT OF THE SPILL. PRECAUTIONS WILL BE ADDED TO PROCEDURES.

[352] VERMONT YANKEE DOCKET 50-271 LER 87-009 REV 01
UPDATE ON RELIEF VALVE ACCUMULATOR FAILED DUE TO SOLENOID VALVE LEAKAGE.
EVENT DATE: 081887 REPORT DATE: 103087 NSSS: GE TYPE: BWR
VENDOR: ASCO VALVES
NUPRO COMPANY

(NSIC 206712) ON 8/18/87, WITH THE PLANT SHUT DOWN FOR THE 1987 REFUELING OUTAGE, THE "C" MAIN STEAM (EIIS=SB) RELIEF VALVE ACCUMULATOR ASSEMBLY (REFER TO ATTACHED SKETCH NO. 1) WAS FOUND TO HAVE SEAT LEAKAGE THAT EXCEEDED THE ACCEPTANCE CRITERIA SPECIFIED IN THE SURVEILLANCE PROCEDURE. THE OTHER THREE MAIN STEAM RELIEF VALVE ACCUMULATOR ASSEMBLIES PASSED THE INITIAL LEAK TESTS. INITIALLY, THE LEAKAGE WAS THOUGHT TO BE THROUGH THE CHECK VALVE IN THE ASSEMBLY. ON 9/4/87 THE CHECK VALVE WAS DISASSEMBLED AND HAD A SMALL DEPOSITE OF CORROSION PRODUCT ON THE SEAT. THE VALVE INTERNALS WERE CLEANED AND ON 9/4/87 THE ASSEMBLY WAS UNSUCCESSFULLY RETESTED. DURING THE RETEST, EXCESSIVE LEAKAGE WAS OBSERVED THROUGH THE SOLENOID VALVE'S EXHAUST PORT. THE SOLENOID VALVE WAS CYCLED DURING SYSTEM FUNCTIONAL TESTING AND THE LEAKAGE REDUCED TO WITHIN ACCEPTABLE LIMITS. THE ROOT CAUSE OF THIS LEAKAGE IS SUSPECTED TO BE DIRT/CORROSION PRODUCT ON THE SOLENOID VALVE SEAT THAT PRECLUDED PROPER SEATING OF THE VALVE. THIS DIRT/CORROSION PRODUCT IS FROM THE CONTAINMENT ATMOSPHERE SYSTEM (EIIS-LE) CARBON STEEL PIPING, THAT EXISTS FROM WHEN THE SYSTEM WAS SUPPLIED WITH AIR. THE SYSTEM IS CURRENTLY SUPPLIED WITH CLEAN, DRY NITROGEN. THE ACCUMULATOR ASSEMBLY WAS SUCCESSFULLY RETESTED ON 09/28/87, PRIOR TO PLANT STARTUP.

[353] VERMONT YANKEE DOCKET 50-271 LER 87-013
INADVERTENT SCRAM DUE TO INCORRECT LISTING OF ELECTRICAL LOADS CAUSED BY OPERATOR ERROR.
EVENT DATE: 090487 REPORT DATE: 093087 NSSS: GE TYPE: BWR

(NSIC 206609) AT APPROXIMATELY 1851 HOURS ON 9/4/87 WITH THE REACTOR IN THE COLD SHUTDOWN CONDITION, A FULL REACTOR SCRAM AND PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) (JM*) ACTUATION OCCURRED. (THE PCIS ACTUATION INCLUDED GROUPS I, II, AND III). THE SCRAM AND PCIS ACTUATION OCCURRED AS A RESULT OF A LOSS OF POWER TO THE "A" REACTOR PROTECTION SYSTEM (RPS) (JE*) AT THE SAME TIME THAT POWER TO THE "B" REACTOR PROTECTION SYSTEM WAS SECURED. THE RPS IS DIVIDED INTO TWO SEPARATE SYSTEMS, EACH POWERED FROM A SEPARATE MOTOR-GENERATOR (MG) (MG*) SET. THE LOSS OF POWER TO THE "A" RPS OCCURRED WHEN ITS BUS (BU*) WAS DE-ENERGIZED BY A LOSS OF THE "A" RPS MG SET. THE MOTOR CONTROL CENTER (MCC) (SWGR*) THAT POWERS THE MG SET WAS DE-ENERGIZED TO CONDUCT MAINTENANCE ON ITS SUPPLY TRANSFORMER (XFMR*). AT THE TIME OF THIS EVENT, THE "B" RPS MG SET HAD BEEN INCORRECTLY DE-ENERGIZED. THIS ERROR WAS THE RESULT OF USING A LIST THAT WAS GENERATED, BY AN OPERATOR, TO SHOW WHICH LOADS WERE TO BE SECURED FOR WORK ON THE TRANSFORMER. THE WRONG MG SET WAS LISTED. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO PERSONNEL ERROR. AT APPROXIMATELY 1910 HOURS, THE REACTOR SCRAM AND PCIS ISOLATIONS WERE PROMPTLY RESET AND SYSTEMS WERE RETURNED TO NORMAL.

[354] VERMONT YANKEE DOCKET 50-271 LER 87-014
INADVERTENT SCRAMS DUE TO UNEXPECTED LACK OF INPUTS TO THE POWER RANGE NEUTRON MONITORS DUE TO INADEQUATE OPERATOR TRAINING.
EVENT DATE: 090887 REPORT DATE: 100787 NSSS: GE TYPE: BWR

(NSIC 206713) EVENT 1: AT 2050 HOURS ON 09/08/87 WITH THE REACTOR IN THE COLD SHUTDOWN CONDITION, THE FEEDER BREAKER (BU*) FOR 480 VAC BUS 8 (SWGR*) WAS OPENED, TO RETURN BUS 8 TO ITS NORMAL SOURCE OF POWER, AND A FULL REACTOR SCRAM OCCURRED. BUS 8 SUPPLIES POWER TO MOTOR CONTROL CENTER 8A (MCC) (SWGR*) WHICH IS THE POWER SOURCE FOR "A" REACTOR PROTECTION SYSTEM (RPS) (JE*) MG (*MG) SET AND IN TURN POWERS "A" RPS. LOSS OF POWER TO EITHER RPS (A OR B) WILL CAUSE A HALF-SCRAM. AT 2053, THE SCRAM WAS RESET AND SYSTEMS WERE RETURNED TO NORMAL. EVENT 2: AT 0257 HOURS ON 09/10/87 WITH THE REACTOR IN THE COLD SHUTDOWN

CONDITION THE FEEDER BREAKER (BU*) FOR MOTOR CONTROL CENTER 8B(MCC) (SWGR*) WAS OPENED, TO FACILITATE TESTING THE BREAKER, AND A FULL REACTOR SCRAM OCCURRED. DURING THIS EVENT, THE "A" RPS WAS POWERED FROM ITS ALTERNATE POWER SUPPLY; MCC 8B. AT 0307, THE SCRAM WAS RESET AND SYSTEMS WERE RETURNED TO NORMAL. BOTH EVENTS: FOR BOTH PLANNED EVOLUTIONS, OPERATIONS INTENTIONALLY REMOVED POWER FROM "A" RPS AND THE EXPECTED HALF-SCRAM OCCURRED. THE FULL SCRAM WAS UNANTICIPATED. OPERATIONS INVESTIGATION, FOLLOWING THE EVENT ON 09/10/87 REVEALED THAT THE AVERAGE POWER RANGE MONITOR SYSTEM (APRM)(*JE) CHANNELS A AND C ALSO SHARE THEIR LOCAL POWER RANGE MONITOR (LPRM)(DETI) INPUTS WITH CHANNELS D AND F.

[355] VERMONT YANKEE DOCKET 50-271 LER 87-015
REACTOR SCRAM DUE TO TRANSIENT IN TURBINE CONTROL OIL SYSTEM.
EVENT DATE: 100387 REPORT DATE: 110287 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 206858) ON 10/03/87 AT 1356, WITH REACTOR AT 18% POWER, WHILE BRING THE TURBINE (TA*) UP TO SPEED IN PREPARATION FOR PLACING THE GENERATOR (TB*) IN SERVICE, A SCRAM OCCURRED AS A RESULT OF A TURBINE CONTROL VALVES RAPID OPENING AND SUBSEQUENT FAST CLOSURE. THE TURBINE CONTROL VALVES WERE NOT OPERATING PROPERLY DUE TO A SUSPECTED PROBLEM IN THE TURBINE CONTROL OIL SYSTEM (TG*). IT APPEARS THAT SUFFICIENT OIL PRESSURE WAS NOT AVAILABLE TO POSITION THE CONTROL VALVES TO THE LOAD LIMITER POSITION. WHEN OIL PRESSURE CONTROL SWITCHED FROM THE AUXILIARY OIL PUMP (TG*) TO THE SHAFT DRIVEN OIL PUMP (TG*), AN OIL PRESSURE TRANSIENT CAUSED THE CONTROL VALVES TO QUICKLY OPEN TO THE LOAD LIMITER POSITION. THE SUBSEQUENT SHORT-TERM INCREASE IN STEAM FLOW THROUGH THE TURBINE WAS EQUAL TO APPROXIMATELY 50%, AS DETERMINED BY TURBINE FIRST STAGE PRESSURE. THIS RAPID INCREASE IN TURBINE FIRST STAGE PRESSURE ABOVE 30%, REMOVED THE AUTOMATIC SCRAM BYPASS FOR THE TURBINE CONTROL VALVE FAST-CLOSURE LOGIC. SINCE THE CONTROL VALVE FAST-CLOSURE SCRAM SIGNAL (JC*) IS NOT NORMALLY RESET (ARMED) UNTIL TURBINE FIRST STAGE PRESSURE IS APPROACHING 30%, THIS PREVIOUSLY BYPASSED SCRAM SIGNAL CAUSED THE REACTOR TO SCRAM, AT APPROXIMATELY 1356.

[356] VOGTLE 1 DOCKET 50-424 LER 87-057
PROCEDURE DEFICIENCY RESULTS IN FAILURE TO TRIP OVERTEMP DELTA T REACTOR TRIP BISTABLE.
EVENT DATE: 080887 REPORT DATE: 101487 NSSS: SS TYPE: PWR
VENDOR: WESTON ELECTRIC INSTRUMENT

(NSIC 206726) ON AUGUST 8, 1987, WITH UNIT 1 OPERATING IN MODE 1 AT 90% REACTOR POWER, IT WAS DISCOVERED THAT A PRESSURIZER PRESSURE CHANNEL WOULD NOT PASS CHANNEL CHECK REQUIREMENTS. A LIMITING CONDITION FOR OPERATION (LCO) WAS ENTERED AT 1920 CDT AND THE ASSOCIATED HIGH PRESSURE REACTOR TRIP, LOW PRESSURE REACTOR TRIP, AND LOW PRESSURE SAFETY INJECTION ACTUATION BISTABLES WERE SUBSEQUENTLY TRIPPED PER PROCEDURE 18001-1 "PRIMARY SYSTEMS INSTRUMENTATION MALFUNCTION". AT 0600 CDT ON AUGUST 9, 1987, THE SHIFT SUPERVISOR OBSERVED THAT THE OVERTEMPERATURE DELTA T (OTDT) REACTOR TRIP BISTABLE WAS NOT TRIPPED AND CONCLUDED THE LCO ACTION REQUIREMENTS OF TECHNICAL SPECIFICATION 3.3.1 ITEM 7 HAD NOT BEEN COMPLIED WITH. A REVIEW OF PROCEDURE 18001-1 REVEALED A PROCEDURAL DEFICIENCY HAD RESULTED IN THE FAILURE TO TRIP THE OTDT BISTABLE. THE ASSOCIATED OTDT BISTABLE WAS TRIPPED AT 0830 CDT ON AUGUST 9, 1987. THE FAILED PRESSURIZER PRESSURE TRANSMITTER WAS REPLACED AND THE LCO WAS EXITED AT 0530 CDT ON AUGUST 10, 1987. PROCEDURE 18001-1 HAS BEEN REVISED TO ADD THE OTDT REACTOR TRIP BISTABLES TO THE LIST OF BISTABLES TO BE TRIPPED FOR A PRESSURIZER PRESSURE INSTRUMENT FAILURE.

[357] VOGTLE 1 DOCKET 50-424 LER 87-056
TECHNICAL SPECIFICATION NOT MET DUE TO INCOMPLETE VENDOR SOFTWARE FOR DOSE CALCULATIONS.
EVENT DATE: 091687 REPORT DATE: 101587 NSSS: SS TYPE: PWR
VENDOR: NUCLEAR DATA, INC.

(NSIC 206725) ON SEPTEMBER 16, 1987, IT WAS IDENTIFIED THAT THE CUMULATIVE DOSE CALCULATIONS PROGRAM FOR GASEOUS EFFLUENT RELEASES TO THE ATMOSPHERE FOR RADIOIODINES DID NOT INCLUDE THE IODINE ISOTOPE 133 (I-133), AS REQUIRED BY TECHNICAL SPECIFICATIONS (T.S.). THE DOSE CALCULATIONS ARE PERFORMED UTILIZING EFFLUENT MANAGEMENT SOFTWARE (EMS) WHICH WAS PROGRAMMED TO MEET VOGTLE'S T.S. REQUIREMENTS. DOSE CALCULATIONS WERE BEGUN IN MARCH 1987, FOLLOWING INITIAL CRITICALITY, AND WERE PERFORMED EACH MONTH AS A SURVEILLANCE REQUIREMENT OF THE T.S. THE DIRECT CAUSE OF THIS EVENT WAS THAT IODINE (I-133) WAS NOT INCLUDED IN THE SOFTWARE PACKAGE SUPPLIED TO PLANT VOGTLE BY A QUALIFIED VENDOR AND WAS NOT CORRECTED DURING INDEPENDENT VERIFICATION BY GPC. CORRECTIVE ACTIONS WERE TAKEN TO HAVE THE EMS SOFTWARE PACKAGE CORRECTED. BY UTILIZING TEST CASES SUPPLIED BY AN INDEPENDENT THIRD SOURCE, EMS TESTING WILL BE CONDUCTED TO ENSURE THAT OTHER ISOTOPES WERE NOT OMITTED FROM THE DOSE CALCULATIONS.

[358] VOGTLE 1 DOCKET 50-424 LER 87-058
FALSE SIGNAL FROM A RADIATION MONITOR LEADS TO CONTROL ROOM ISOLATION.
EVENT DATE: 092187 REPORT DATE: 102187 NSSS: SS TYPE: PWR
VENDOR: AMPEREX ELECTRONIC CORP.

(NSIC 206837) ON SEPTEMBER 21, 1987, UNIT 1 WAS IN MODE 1 (POWER OPERATIONS AT 100% RATED THERMAL POWER. AT 0204 CDT, A CONTROL ROOM ISOLATION (EMERGENCY MODE) (CRI) OCCURRED AS A RESULT OF RADIATION MONITOR 1RE-12116 SPURIOUSLY OR FALSELY GENERATING A HIGH RADIATION SIGNAL. ANOTHER CONTROL ROOM AIR INTAKE RADIATION MONITOR, 1RE-12117, WAS CHECKED, AND DISPLAYED NORMAL READINGS. AIR SAMPLES WERE TAKEN WHICH VERIFIED THAT A VALID HIGH RADIATION CONDITION DID NOT EXIST. BY 0328 CDT, CONTROL ROOM ISOLATION WAS DISCONTINUED AND NORMAL HVAC OPERATION RESUMED. THE CAUSE OF THIS EVENT WAS A FALSE HIGH RADIATION SIGNAL RECEIVED FROM RADIATION MONITOR 1RE-12116. CORRECTIVE ACTION(S) ARE YET TO BE DETERMINED.

[359] VOGTLE 1 DOCKET 50-424 LER 87-059
CHANNEL CHECKS MISSED DUE TO AN INADEQUATE PROCEDURE.
EVENT DATE: 101387 REPORT DATE: 111287 NSSS: SS TYPE: PWR

(NSIC 207119) THIS REPORT IS BEING SUBMITTED BECAUSE CERTAIN TECHNICAL SPECIFICATION (T.S.) REQUIRED SURVEILLANCES (CHANNEL CHECKS) WERE NOT PERFORMED WITHIN THEIR ALLOWABLE TIME INTERVAL. ON OCTOBER 13,1987 IT WAS DISCOVERED THAT CHANNEL CHECKS, WHICH VERIFY OPERABILITY OF THE CONTAINMENT ISOLATION VALVES (CIV) POSITION INDICATION, WERE NOT PERFORMED FOR THREE (3) VALVES. CHANNEL CHECKS HAD BEEN PERFORMED FOR THE MONTHS PRIOR TO AND AFTER THE TIME PERIOD EACH VALVE SURVEILLANCE WAS NOT PERFORMED. ON EACH OCCASION, WHEN THE CHANNEL CHECKS WERE NOT PERFORMED, UNIT 1 WAS IN MODE 1 AT APPROXIMATELY 100% OF RATED THERMAL POWER. THE CAUSE OF THIS EVENT WAS AN INADEQUATE PROCEDURE. WHEN THE SURVEILLANCE PROCEDURE WAS WRITTEN ALL CIVS WERE NOT INCLUDED. CORRECTIVE ACTION INCLUDED REVISING THE SURVEILLANCE PROCEDURE TO INCLUDE A CHANNEL CHECK FOR THESE ADDITIONAL CIVS. A COMPLETE REVIEW OF THE CIV SURVEILLANCE REQUIREMENTS AND THE SURVEILLANCE PROCEDURES WERE PERFORMED TO ENSURE THAT CIV SURVEILLANCES HAVE BEEN ADDRESSED.

[360] WATERFORD 3 DOCKET 50-382 LER 87-018 REV 02
UPDATE ON CONTAINMENT PRESSURE INSTRUMENT ISOLATED DUE TO PERSONNEL ERROR.
EVENT DATE: 071687 REPORT DATE: 102387 NSSS: CE TYPE: PWR

(NSIC 206820) AT APPROXIMATELY 1100 HOURS JULY 16, 1987, TECHNICIANS PERFORMING TROUBLESHOOTING FOUND CONTAINMENT PRESSURE TRANSMITTER CB-IPT-6702 SMD ISOLATED. THE TRANSMITTER WAS UNISOLATED AND THE INSTRUMENT CHANNEL WAS DECLARED OPERABLE JULY 17, 1987. THE TRANSMITTER WAS EVIDENTLY NOT RESTORED PROPERLY FOLLOWING CALIBRATION DECEMBER 20, 1986 DUE TO ERRORS BY TWO TECHNICIANS WHO INITIALED FOR PERFORMANCE AND VERIFICATION OF THIS ALIGNMENT. THE PLANT WAS THEREFORE IN A CONDITION PROHIBITED BY TECHNICAL SPECIFICATIONS FROM ITS ENTRY INTO MODE 3 JANUARY 28, 1987 UNTIL THE CHANNEL WAS DECLARED INOPERABLE JULY 12. THE OTHER SEVEN CONTAINMENT PRESSURE TRANSMITTERS WERE VERIFIED TO BE ALIGNED CORRECTLY. THE TECHNICIAN WHO EVIDENTLY FAILED TO REALIGN THE INSTRUMENT WAS COUNSELED AND DISCIPLINED. ADDITIONAL GUIDANCE WAS PROVIDED TO I&C PERSONNEL CONCERNING PROCEDURAL COMPLIANCE. IN THE FUTURE THESE INSTRUMENTS WILL BE CHECKED DURING AND AFTER CONTAINMENT PURGE SYSTEM OPERATION PRIOR TO HEATUP FROM COLD SHUTDOWN. THIS SYSTEM INDUCES SUFFICIENT PRESSURE CHANGES TO VERIFY THE INSTRUMENTS ARE RESPONDING TO CHANGES IN ACTUAL PRESSURE. SINCE THE OTHER THREE CHANNELS PROVIDING CONTAINMENT SPRAY ACTUATION WERE OPERABLE THROUGHOUT THIS PERIOD AND ONLY TWO CHANNEL TRIPS ARE REQUIRED TO PRODUCE AN ACTUATION, THERE WAS NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[361] WOLF CREEK 1 DOCKET 50-482 LER 86-043 REV 01
UPDATE ON INDETERMINATE WIRING IN MOTOR OPERATOR VALVE OPERATORS CAUSED A CONDITION WHICH COULD HAVE PREVENTED FULFILLMENT OF A SAFETY FUNCTION.
EVENT DATE: 080586 REPORT DATE: 110287 NSSS: WE TYPE: PWR
VENDOR: LIMITORQUE CORP.

(NSIC 206897) IN MAY OF 1986, DURING A ROUTINE NRC INSPECTION TO VERIFY COMPLIANCE TO ENVIRONMENTAL QUALIFICATION REQUIREMENTS AND ALSO DURING UNRELATED MAINTENANCE ACTIVITIES, DISCREPANCIES IN THE INTERNAL WIRING AND TERMINAL BLOCKS IN SAFETY RELATED MOTORIZED VALVE ACTUATORS WERE IDENTIFIED. THE DECISION WAS MADE TO REPLACE THE VENDOR INSTALLED WIRING AND TERMINAL LUGS IN THESE ACTUATORS WITH COMPONENTS HAVING VERIFIABLE QUALITY AND MEETING ALL APPLICABLE DESIGN REQUIREMENTS. DURING THIS TIME PERIOD, UNIT CONDITIONS VARIED FROM HOT STANDBY THROUGH POWER OPERATION. FOLLOWING COMPLETION OF THIS REWORK EFFORT AN ENGINEERING EVALUATION OF THE DISCREPANCIES DISCOVERED DURING THIS ACTIVITY WAS INITIATED. THIS SUPPLEMENT (REVISION 01) PROVIDES THE RESULTS OF THE ENGINEERING EVALUATION AS ADDITIONAL INFORMATION. A THOROUGH EVALUATION OF THIS EVENT HAS BEEN CONDUCTED. HOWEVER, BECAUSE OF THE COMPLEXITY OF THE INTERACTIONS INVOLVED DURING THE VALVE OPERATOR MANUFACTURING PROCESS, DETERMINATION OF WHERE IN THE MANUFACTURING PROCESS THIS PROBLEM OCCURRED HAS NOT BEEN POSSIBLE. ACTIONS TAKEN TO PRECLUDE RECURRENCE INCLUDE INSPECTION/REWORK OF THE ACTUATORS PRESENTLY IN STORAGE AND ENHANCEMENTS TO THE PROCUREMENT PROCESS WHEN NEW ACTUATORS ARE PROCURED IN THE FUTURE.

[362] WOLF CREEK 1 DOCKET 50-482 LER 87-035
ENGINEERED SAFETY FEATURES ACTUATION - CONTROL ROOM VENTILATION ISOLATION SIGNALS CAUSED BY MALFUNCTIONS OF THE CHLORINE MONITORS.
EVENT DATE: 090387 REPORT DATE: 100287 NSSS: WE TYPE: PWR
VENDOR: M D A SCIENTIFIC, INC.

(NSIC 206738) ON SEPTEMBER 3, 1987, AT 0952 CDT, AND ON SEPTEMBER 20, AT 2214 CDT, CONTROL ROOM VENTILATION ISOLATION SIGNALS (CRVIS) OCCURRED DUE TO CHLORINE MONITOR GK-AITS-3 INDICATING HIGH CHLORINE LEVEL IN THE OUTSIDE AIR MAKEUP TO THE CONTROL BUILDING HEATING, VENTILATING AND AIR CONDITIONING SYSTEM. NO CHLORINE WAS PRESENT AS EVIDENCED BY NORMAL READINGS ON THE REDUNDANT CHLORINE MONITOR. EXAMINATION OF THE MONITOR AFTER THE SEPTEMBER 3 EVENT REVEALED THAT THE VIBRATION OF THE WISA PUMP (AIR SAMPLE PUMP) CAUSED A SPURIOUS SIGNAL SPIKE TO BE SENT TO THE CHLORINE MONITOR, WHICH INITIATED A CRVIS. THE CRVIS SIGNAL WAS RESET AND THE MONITOR WAS PLACED IN BYPASS FOR TROUBLESHOOTING AT APPROXIMATELY 0953 CDT. NO FURTHER PROBLEMS WITH THE MONITOR WERE FOUND AND IT WAS RETURNED TO

OPERATION AT 1324 CDT ON SEPTEMBER 8, 1987. EXAMINATION OF THE MONITOR AFTER THE SEPTEMBER 20 EVENT REVEALED THAT THE CHEMICALLY SENSITIVE PAPER TAPE USED TO DETECT CHLORINE HAD BROKEN, RESULTING IN A CHANGE IN OPACITY READING SUFFICIENT FOR THE MONITOR TO INITIATE A CRVIS. THE CRVIS SIGNAL WAS RESET AND THE MONITOR WAS PLACED IN BYPASS FOR TROUBLESHOOTING AT APPROXIMATELY 2215 CDT. NO FURTHER PROBLEMS WITH THE MONITOR WERE FOUND AND IT WAS RETURNED TO OPERATION AT 0120 ON SEPTEMBER 21, 1987.

[363] WOLF CREEK 1 DOCKET 50-482 LER 87-037
HIGH VOLTAGE TRANSMISSION LINE FAILURE CAUSES GENERATOR TRIP/REACTOR TRIP AND SUBSEQUENT ESF ACTUATION DURING RECOVERY AND RESTART.
EVENT DATE: 091087 REPORT DATE: 101287 NSSS: WE TYPE: PWR

(NSIC 206755) ON SEPTEMBER 10, 1987, AT APPROXIMATELY 2009 CDT, A REACTOR TRIP, FEEDWATER ISOLATION (FWIS), AND AUXILIARY FEEDWATER ACTUATION (AFAS) OCCURRED AS A RESULT OF A FAILED 345 KILOVOLT TRANSMISSION LINE CONDUCTOR BETWEEN THE 'B' PHASE MAIN TRANSFORMER AND THE UNIT SWITCHYARD. SUBSEQUENTLY, A CONTAINMENT PURGE ISOLATION AND CONTROL ROOM VENTILATION ISOLATION OCCURRED AT APPROXIMATELY 2107 CDT DURING INVESTIGATION OF A LOSS OF FLOW TO THE CONTAINMENT ATMOSPHERE RADIATION MONITORS. AT APPROXIMATELY 2157 CDT, A SECOND TURBINE DRIVEN AFAS OCCURRED DUE TO TWO LO-LO STEAM GENERATOR LEVELS. A FWIS AND AFAS OCCURRED DURING RESTART DUE TO HI-HI STEAM GENERATOR LEVEL AT APPROXIMATELY 2200 CDT ON SEPTEMBER 11, 1987, AND AGAIN AT APPROXIMATELY 0207 CDT ON SEPTEMBER 12, 1987. THE PROBABLE CAUSE OF THE LINE CONDUCTOR FAILURE WAS HIGH RESISTANCE AND HEATING AT THE BOLTED T-PAD CONNECTION. THE LINE CONDUCTOR HAS BEEN REPLACED. THE BOLTED T-PAD HAS BEEN REPLACED WITH A COMPRESSION T-PAD. IT WILL BE REEMPHASIZED TO CONTROL ROOM OPERATORS THAT POWER ASCENSION SHOULD BE A SLOW EVOLUTION SO THAT FEW PARAMETERS WILL BE VARIED SIMULTANEOUSLY TO AVOID FUTURE FWIS/AFAS ACTUATIONS DUE TO FEEDWATER CONTROL PROBLEMS.

[364] WOLF CREEK 1 DOCKET 50-482 LER 87-039
ACCIDENTAL MISPOSITIONING OF BREAKER SWITCH CAUSES INOPERABILITY OF ONE POWER OPERATED RELIEF VALVE WHILE THE OTHER WAS ISOLATED DUE TO SEAT LEAKAGE.
EVENT DATE: 091687 REPORT DATE: 101687 NSSS: WE TYPE: PWR
VENDOR: GARRET AIR RESEARCH MFG CO.

(NSIC 206739) ON SEPTEMBER 16, 1987, AT APPROXIMATELY 2100 CDT, WHILE PERFORMING A ROUTINE WALKDOWN OF THE MAIN CONTROL ROOM PANELS, CONTROL ROOM OPERATORS DISCOVERED AN ABSENCE OF POSITION INDICATION FOR PRESSURIZER POWER OPERATED RELIEF VALVE (PORV) BB-PCV-456A BLOCK VALVE BB HV-8000B. IT WAS SUBSEQUENTLY DETERMINED THAT THE LOCAL BREAKER ISOLATION SWITCH FOR BB HV-8000B WAS IN THE "ISOLATE" POSITION, CAUSING THE VALVE TO BE CLOSED WITH CONTROL POWER REMOVED. THE SWITCH WAS RETURNED TO THE "NORMAL" POSITION AND BB HV-8000B WAS REOPENED, THEREBY RESTORING THE PORV TO OPERABLE STATUS AT 2110 CDT. AT THIS TIME, BLOCK VALVE BB HV-8000A WAS ALSO CLOSED DUE TO SEAT LEAKAGE ON ITS ASSOCIATED PORV. THIS CONDITION, WHICH WOULD HAVE PREVENTED PROPER PORV FUNCTIONING IN THE ABSENCE OF OPERATOR INTERVENTION, IS BEING REPORTED PURSUANT TO 10CFR50.73(A)(2)(V). IT WAS DETERMINED THAT THE SWITCH HAD BEEN MISPOSITIONED AT 1030 CDT ON SEPTEMBER 16, MOST LIKELY BY CONTRACTOR PERSONNEL UNKNOWINGLY BUMPING THE SWITCH WITH THEIR EQUIPMENT. DISCUSSIONS HAVE BEEN HELD WITH THESE PERSONNEL AND THEIR SUPERVISORS.

[365] WOLF CREEK 1 DOCKET 50-482 LER 87-038
FAILURE TO PROPERLY VERIFY OPERABILITY OF FIRE PUMPS DUE TO PROCEDURAL INADEQUACY.
EVENT DATE: 091887 REPORT DATE: 101987 NSSS: WE TYPE: PWR
VENDOR: MERCOID CORP.
NATIONAL CONTROLS INC.

(NSIC 206771) ON SEPTEMBER 18, 1987, AT APPROXIMATELY 0730 CDT, IT WAS

DETERMINED, FOLLOWING AN INTERNAL AUDIT, THAT THE OPERABILITY OF THE ELECTRIC AND DIESEL DRIVEN FIRE PUMPS HAD NOT BEEN PROPERLY VERIFIED IN ACCORDANCE WITH THE REQUIREMENTS OF TECH SPEC 4.7.10.1.1.F. THE CRITERIA IN THE SURVEILLANCE PROCEDURE, STS FP-004, "FIRE SYSTEM FLOW TEST, PUMP SEQUENTIAL START, AND ANNUAL FIRE PUMP TEST", DID NOT FULLY VERIFY THAT THE PUMPS MET THE TECH SPEC REQUIREMENTS PRIOR TO SEPTEMBER 23, 1987. THE SURVEILLANCE PROCEDURE WAS REVISED ON SEPTEMBER 23, 1987, AND COMPLETED AS SCHEDULED ON SEPTEMBER 26, 1987, AT APPROXIMATELY 0626 CST. SINCE THE FIRE PUMPS HAD NOT BEEN PROPERLY VERIFIED TO BE OPERABLE BETWEEN RECEIPT OF THE OPERATING LICENSE ON MARCH 11, 1985, AND SEPTEMBER 26, 1987, THIS REPORT IS BEING SUBMITTED PURSUANT TO 10CFR 50.73(A)(2)(I)(B) AS A CONDITION PROHIBITED BY THE PLANT'S TECH SPECS. THE ROOT CAUSE OF THIS EVENT WAS DETERMINED TO BE A PROCEDURAL DEFICIENCY. THE SURVEILLANCE PROCEDURE DID NOT CONTAIN ALL OF THE TECH SPEC REQUIREMENTS, AND THEREFORE, WHEN THE SURVEILLANCE TEST WAS CONDUCTED, THE TECH SPEC CRITERIA WERE NOT SATISFIED. THE SURVEILLANCE PROCEDURE WAS REVISED AND THE SURVEILLANCE TEST WAS PERFORMED SATISFACTORILY ON SEPTEMBER 26, 1987.

[366] WOLF CREEK 1 DOCKET 50-482 LER 87-040
PERSONNEL OVERSIGHT RESULTS IN NONCONSERVATIVE ERROR IN CONTAINMENT PURGE RADIATION MONITORS SETPOINT FOR ISOLATING CONTAINMENT PURGE.
EVENT DATE: 091887 REPORT DATE: 101987 NSSS: WE TYPE: PWR

(NSIC 206849) ON SEPTEMBER 18, 1987, IT WAS DETERMINED THAT THE ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS) BISTABLE TRIP SETPOINT, WHICH CONTROL THE ISOLATION FUNCTION OF THE CONTAINMENT PURGE RADIATION MONITORS, WERE SET AT A VALUE LESS CONSERVATIVE THAN THE TRIP SETPOINT CALCULATED IN ACCORDANCE WITH THE OFFSITE DOSE CALCULATION MANUAL (ODCM). THIS CONDITION IS CONTRARY TO TECHNICAL SPECIFICATION REQUIREMENTS. THE ESFAS BISTABLE TRIP SETPOINTS WERE ORIGINALLY SET DURING STARTUP TESTING AND DESIGN DRAWINGS REFLECTED THIS VALUE. THESE BISTABLE TRIP SETPOINTS HAD NOT BEEN ALTERED BEEN THE ALERT AND HIGH ALARMS FOR THESE MONITORS WERE ADJUSTED BASED ON ODCM CALCULATIONS PRIOR TO EACH CONTAINMENT PURGE. THIS EVENT HAS BEEN ATTRIBUTED TO COGNITIVE PERSONNEL ERROR BY NONLICENSED PERSONNEL IN VARIOUS DISCIPLINES, WHO WERE AWARE THAT A SETPOINT DISCREPANCY EXISTED, BUT WERE UNAWARE OF THE CONSEQUENCES. THE SFAS BISTABLE TRIP SETPOINTS HAVE BEEN SET TO A MORE CONSERVATIVE VALUE. THE APPROPRIATE PROCEDURES HAVE BEEN REVISED TO REFLECT THE FACT THAT ADJUSTMENT OF THE ESFAS BISTABLE TRIP SETPOINTS IS NECESSARY IN ADDITION TO ADJUSTMENT OF THE ALERT/HIGH ALARM SETPOINTS. IN ADDITION, THE DESIGN DRAWINGS WILL BE REVISED TO REFLECT THE FACT THAT THE ESFAS BISTABLE TRIP SETPOINTS ARE CONTROLLED BY ODCM CALCULATIONS.

[367] WOLF CREEK 1 DOCKET 50-482 LER 87-041
PERSONNEL ERRORS RESULT IN LOSS OF POWER TO CONTROL ROD MOVEABLE GRIPPER COILS WHICH CAUSES A REACTOR TRIP WHEN CONTROL ROD MOVEMENT WAS ATTEMPTED.
EVENT DATE: 092787 REPORT DATE: 102787 NSSS: WE TYPE: PWR

(NSIC 206848) ON SEPTEMBER 27 1987, AT APPROXIMATELY 0232 CDT, A REACTOR TRIP OCCURRED FROM APPROXIMATELY 86 PERCENT POWER. POWER LEVEL WAS BEING REDUCED IN PREPARATION FOR THE REFUEL II OUTAGE. THE REACTORS TRIP OCCURRED AS A RESULT OF POWER RANGE FLUX HIGH NEGATIVE RATE WHEN THE BANK 'D', GROUP 'A' CONTROL RODS DROPPED WHILE ATTEMPTING TO MOVE RODS OUT ONE STEP. THIS EVENT IS BEING REPORTED PERSUANT TO 10CFR 50.73(A)(2)(IV) AS AN UNPLANNED ACTUATION OF ENGINEERED SAFETY FEATURES EQUIPMENT. THE ROOT CAUSE OF THE EVENT WAS COGNITIVE PERSONNEL ERROR BY TWO PERSONNEL. A CONTRACTOR FOR INADVERTENTLY BUMPING THE FUSIBLE LINK DISCONNECTS COIL SWITCH ON THE CONTROL ROD MOVEABLE GRIPPER COIL CABINET AND A LICENSED UTILITY OPERATOR FOR ASSUMING AN ALARM WOULD GENERATED IF THE SWITCH WAS MISPOSITIONED. THIS CAUSED THE PROBLEM NOT TO BE FULLY INVESTIGATED. IT WILL BE REEMPHASIZED TO OPERATIONS PERSONNEL TO REQUEST APPROPRIATE ASSISTANCE INVESTIGATING AN EQUIPMENT PROBLEM IF THE PROBLEM CANNOT BE POSITIVELY RESOLVED

BY OPERATORS. AN EVALUATION TO REMOVE THE SWITCH HANDLES TO PREVENT THEM FROM BEING ACCIDENTLY MISPOSITIONED WILL BE CONDUCTED.

[368] WOLF CREEK 1 DOCKET 50-482 LER 87-043
SURVEILLANCE OF POWER RANGE LOW SETPOINT AND P-8, P-9 AND P-10 INTERLOCKS NOT PERFORMED PER TECH SPECS DUE TO PROCEDURAL DEFICIENCIES.
EVENT DATE: 092987 REPORT DATE: 102887 NSSS: WE TYPE: PWR

(NSIC 206847) ON SEPTEMBER 29, 1987, IT WAS DISCOVERED THAT SURVEILLANCE TESTING OF THE POWER RANGE (PR) NEUTRON FLUX LOW SETPOINT AND THE P-8, P-9, AND P- 10 REACTOR TRIP-SYSTEM INTERLOCKS WAS NOT BEING PERFORMED PER THE TECHNICAL SPECIFICATION (T/S) SURVEILLANCE REQUIREMENTS. IN APRIL, 1987, THE APPLICABLE ANALOG CHANNEL OPERATIONAL TEST (ACOT) PROCEDURES WERE REVISED TO NO LONGER REQUIRE THAT THE DETECTOR CABLES BE DISCONNECTED FOR TESTING WHILE AT POWER OPERATION DURING PERFORMANCE OF MONTHLY ACOT'S. THEREFORE, DURING POWER OPERATION, THE ACOT'S CONSISTED OF MONTHLY VERIFICATION THAT THE PERMISSIVE ANNUNCIATOR WINDOW WAS IN ITS REQUIRED STATE (AS ALLOWABLE PER T/S). CREDIT WAS BEING TAKEN FOR PERFORMANCE OF THE ACOT'S WITHOUT CONSIDERATION OF THE POWER LEVEL AT WHICH THE ACOT WAS PERFORMED. AS A RESULT, TWO PLANT STARTUPS OCCURRED WITHOUT A COMPLETE ACOT ON THE PR LOW SETPOINT AND THE P-8, P-9, AND P10 INTERLOCKS HAVING THEN PERFORMED WITHIN THE PREVIOUS THIRTY-ONE DAYS. THIS EVENT HAS BEEN ATTRIBUTED TO A PERSONNEL ERROR DURING THE PROCEDURE REVISION PROCESS. THE MODE CHANGE CHECKLIST IS BEING REVISED TO ENSURE THE REQUIRED ACOT'S HAVE BEEN PERFORMED PRIOR TO MODE 2 (STARTUP) ENTRY. AS DISCUSSED IN LICENSEE EVENT REPORT 87-029, ENHANCEMENTS TO THE PROCEDURE CHANGE REVIEW PROCESS HAVE BEEN IMPLEMENTED.

[369] WOLF CREEK 1 DOCKET 50-482 LER 87-042
PERSONNEL ERROR LEADS TO HIGH-HIGH STEAM GENERATOR LEVEL RESULTING IN FEEDWATER ISOLATION SIGNAL.
EVENT DATE: 093087 REPORT DATE: 102787 NSSS: WE TYPE: PWR

(NSIC 206884) ON SEPTEMBER 30, 1987, AT APPROXIMATELY 0722 CDT, A FEEDWATER ISOLATION SIGNAL (FWIS) WAS INITIATED BY HIGH-HIGH WATER LEVEL IN STEAM GENERATOR 'D'. CONSEQUENTLY, MAIN TURBINE TRIP AND MAIN FEEDWATER PUMP TRIP SIGNALS WERE ALSO INITIATED. UPON RECEIPT OF THE FWIS, ALL ENGINEERED SAFETY FEATURES EQUIPMENT REQUIRED TO OPERATE RESPONDED PROPERLY. THE MAIN FEEDWATER PUMPS AND MAIN TURBINE HAD BEEN SECURED PREVIOUS TO THE EVENT. PLANT SYSTEMS WERE RESTORED TO NORMAL CONFIGURATION AT APPROXIMATELY 0755 CDT. THE HIGH-HIGH LEVEL CONDITION IN STEAM GENERATOR 'D' WHICH INITIATED THIS EVENT WAS THE RESULT OF A COGNITIVE PERSONNEL ERROR BY LICENSED PERSONNEL WHO FAILED TO COMPENSATE FOR THE INCREASING WATER LEVEL IN CONJUNCTION WITH THE DECREASING STEAMING RATE IN STEAM GENERATOR 'D'. THE INDIVIDUAL INVOLVED IN THE EVENT WERE COUNSELED TO RE-EMPHASIZE THE IMPORTANCE OF CLOSE ATTENTION TO DETAIL AND THE NECESSITY FOR BEING ALERT TO DEVELOPING CONDITION AND FOR TAKING PROMPT ACTIONS TO AVOID UNNECESSARY CHALLENGES TO PLANT SAFETY SYSTEMS.

[370] WOLF CREEK 1 DOCKET 50-482 LER 87-044
PERSONNEL ERROR LEADS TO OMISSION OF SNUBBER FROM INSPECTION PROCEDURE RESULTING IN TECHNICAL SPECIFICATION VIOLATION.
EVENT DATE: 093087 REPORT DATE: 102787 NSSS: WE TYPE: PWR

(NSIC 206846) ON SEPTEMBER 30, 1987, IT WAS DISCOVERED THAT SNUBBER AE04-R005 HAD NOT BEEN VISUALLY INSPECTED DURING THE FIRST REFUELING OUTAGE AS REQUIRED BY TECHNICAL SPECIFICATION (T/S) SURVEILLANCE REQUIREMENT 4.7.8. THE FAILURE TO MEET THE T/S SURVEILLANCE REQUIREMENT WITHOUT SATISFYING THE ASSOCIATED ACTION SETTLEMENT REQUIREMENTS CONSTITUTES A CONDITION PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATIONS. THEREFORE THIS EVENT IS BEING REPORTED PURSUANT TO

10CFR 50.73(A)(2)(I). THE PLANT WAS IN MODE 5, COLD SHUTDOWN, AT THE TIME OF DISCOVERY OF THE EVENT. A REVIEW OF THE SNUBBER VISUAL INSPECTION PROCEDURE REVEALED THAT PROCEDURAL ERRORS ATTRIBUTABLE TO NONLICENSED PERSONNEL ERRORS, DURING PROCEDURE REVISIONS HAD INADVERTENTLY EXCLUDED SNUBBER AE04-R005 FROM THE PROCEDURE. A DETAILED REVIEW WAS CONDUCTED TO VERIFY THAT NO OTHER REQUIRED SNUBBERS WERE INADVERTENTLY EXCLUDED FROM THE PROCEDURE. THE SNUBBER VISUAL INSPECTION PROCEDURE IS BEING REVISED TO CORRECT THE ERROR BY INCLUDING SNUBBER AE04-R005 TO SATISFY THE SURVEILLANCE REQUIREMENTS. SNUBBER AE04-R005 WAS INSPECTED SATISFACTORILY ON OCTOBER 11, 1987.

[371] WOLF CREEK 1 DOCKET 50-482 LER 87-045
TECHNICAL SPECIFICATION VIOLATION - INADEQUATE HYDROSTATIC PRESSURE TESTS DUE TO PROCEDURAL INADEQUACY.
EVENT DATE: 100287 REPORT DATE: 103087 NSSS: WE TYPE: PWR
VENDOR: DRAGON VALVE, INC.

(NSIC 206887) ON FOUR OCCASION BETWEEN NOVEMBER 18, 1986, AND DECEMBER 7, 1986, AMERICAN SOCIETY OF MECHANICAL ENGINEERS BOILER AND PRESSURE VESSEL CODE PIPING WAS NOT PROPERLY PRESSURE TESTED FOLLOWING REPAIR IN ACCORDANCE WITH SECTION XI OF THE CODE. ON EACH OCCASION, A HYDROSTATIC TEST WAS CONDUCTED BUT THE PRESSURE DROP. BETWEEN THE TEST GAUGE AND THE COMPONENT BEING TESTED WAS NOT ACCOUNTED FOR BY THE TEST PROCEDURE. SUBSEQUENT ANALYSIS HAS FOUND INSUFFICIENT INFORMATION TO DETERMINE THE SPECIFIC PRESSURE THAT WAS APPLIED TO EACH COMPONENT. THESE EVENTS WERE DETERMINED TO BE REPORTABLE FOLLOWING THE DESIGN ENGINEERING GROUP DISPOSITION ON OCTOBER 2, 1987, OF A CORRECTIVE WORK REQUEST CONCERNING THE CONFIGURATION OF THE TEST EQUIPMENT. THE EVENTS WERE CAUSED BY THE FAILURE OF THE GOVERNING PROCEDURE TO COVER THE POSSIBILITY OF A PRESSURE DROP BETWEEN THE TEST GAUGE AND THE COMPONENT BEING TESTED. THE PROCEDURE HAS TEEN REVISED AND THE MAINTENANCE ENGINEERS HAVE BEEN TRAINED ON THE NEW REQUIREMENTS. THE HYDROSTATIC TESTS WILL BE CORRECTLY COMPLETED BEFORE THE PLANT ENTERS MODE 4 FOLLOWING THE CURRENT REFUELING OUTAGE.

[372] WOLF CREEK 1 DOCKET 50-482 LER 87-048
IMPROPER MAINTENANCE ACTIONS CAUSE FATALITY AND RESULTS IN ENGINEERED SAFETY FEATURES ACTUATIONS AND LOSS OF RESIDUAL HEAT REMOVAL.
EVENT DATE: 101487 REPORT DATE: 110387 NSSS: WE TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 207152) ON OCTOBER 14, 1987 AT APPROXIMATELY 2037 CDT, AN UNUSUAL EVENT (UE) WAS DECLARED DUE TO A FIRE BEING REPORTED IN THE ENGINEERED SAFETY FEATURES (ESF) SWITCHGEAR ROOM. IT WAS DISCOVERED THAT A WORKER HAD COME IN CONTACT WITH AN ENERGIZED PART OF THE B TRAIN SAFETY-RELATED 4160 VOLT ESF BUS. SUBSEQUENT OPERATOR ACTION (DEENERGIZING THE 'A' TRAIN 4160 VOLT ESF BUS TO DEENERGIZE THE CROSS-TIE TO THE 'B' TRAIN ESF BUS) RESULTED IN A LOSS OF THE RESIDUAL HEAT REMOVAL (RHR) SYSTEM FOR APPROXIMATELY 17 MINUTES AND AN AUTOMATIC ACTUATION A DIESEL GENERATOR. THE UE WAS EXITED AT APPROXIMATELY 2111 CDT AFTER RHR WAS RESTORED. THE DIESEL START AND SHUTDOWN SEQUENCER ACTUATION ARE BEING REPORTED PER 10CFR 50.73(A)(3)(IV). THE LOSS OF RHR IS BEING REPORTED PER 10CFR 50.73(A)(2)(V) AND 10CFR 50.73(A)(2)(VII). THE FIRE AND THE FATALITY ARE BEING REPORTED PER 10CFR 50.73(A)(2)(X). THE DIESEL GENERATOR FAILURE IS BEING REPORTED TO SATISFY THE SPECIAL REPORT REQUIREMENTS OF REGULATORY GUIDE 1.108. DETAILED INVESTIGATION DETERMINED THAT THE UTLIMATE CAUSE OF THE ACCIDENT WAS THE FAILURE OF THE QUALIFIED ELECTRICIAN TO FOLLOW THE MAINTENANCE PROCEDURE GOVERNING THE WORK WHICH REQUIRED HIM TO CHECK THE STATIONARY DISCONNECTS FOR HIGH VOLTAGE POTENTIAL.

[373] WPPSS 2 DOCKET 50-397 LER 87-028
SPURIOUS PRIMARY CONTAINMENT ISOLATION CAUSED BY TEMPERATURE MONITORING SYSTEM COMPONENT FAILURE.
EVENT DATE: 091687 REPORT DATE: 101687 NSSS: GE TYPE: BWR
VENDOR: PANALARM COMPANY

(NSIC 206628) ON SEPTEMBER 16, 1987 AT 1707 HOURS, AN INADVERTENT ESF ACTUATION OCCURRED WHEN A LEAK DETECTION (LD) SYSTEM TEMPERATURE MONITORING UNIT FAILED CAUSING A NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NS4) GROUP 8 ISOLATION. THIS FAILURE RESULTED IN CLOSURE OF RCIC-V-8, THE REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) OUTBOARD STEAM SUPPLY ISOLATION VALVE FOR THE RCIC TURBINE. AFTER THOROUGH INVESTIGATION IN ACCORDANCE WITH THE PLANT ABNORMAL CONDITIONS PROCEDURES, THE PLANT OPERATORS RESET THE ISOLATION LOGIC AND RETURNED THE RCIC SYSTEM LINEUP TO NORMAL. THE FAILED TEMPERATURE MONITORING MODULE WAS REPLACED WITH A NEW UNIT, THE TEMPERATURE MONITOR CHANNEL WAS RECALIBRATED AND PLACED BACK INTO SERVICE. THE ROOT CAUSE OF THE FAILURE WAS IDENTIFIED AS MODULE DEFICIENCIES PREVIOUSLY NOTED IN NRC IE INFORMATION NOTICE NO. 86-69. AN ENGINEERING STUDY, IN PROGRESS PRIOR TO THIS EVENT, CONCLUDED THAT THE TEMPERATURE MONITORING SYSTEM CIRCUITY IS TO BE MODIFIED TO IMPROVE ITS RELIABILITY. WNP-2 IS ALSO PURSUING REPLACEMENT OF THIS SYSTEM WITH A MORE RELIABLE DESIGN. THERE IS NO SAFETY SIGNIFICANCE ASSOCIATED WITH THIS EVENT IN THAT THERE WAS NO ACTUAL INITIATING PLANT CONDITION AND ALL EQUIPMENT FUNCTIONED CORRECTLY. THIS EVENT POSED NO THREAT TO THE HEALTH AND SAFETY OF PLANT PERSONNEL OR THE PUBLIC.

[374] WPPSS 2 DOCKET 50-397 LER 87-029
PLANT TECHNICAL SPECIFICATION FIRE-RATED FLOOR PENETRATION IMPAIRED DURING PLANT DESIGN MODIFICATION - INADEQUATE TRAINING.
EVENT DATE: 091687 REPORT DATE: 101687 NSSS: GE TYPE: BWR

(NSIC 206691) DURING AN INSPECTION ON SEPTEMBER 16, 1987 IT WAS DISCOVERED THAT A PLANT TECHNICAL SPECIFICATION FIRE-RATED FLOOR PENETRATION HAD BEEN IMPAIRED SINCE MAY 10, 1986. THE PENETRATION WAS IMPAIRED IN THAT TWO HOLES EXISTED AS A RESULT OF CABLES BEING PULLED BACK THROUGH THE SEAL DURING INSTALLATION OF THE ALTERNATE REMOTE SHUTDOWN SYSTEM IN THE FIRST REFUELING OUTAGE (SPRING, 1986). THE PENETRATION IS LOCATED BETWEEN FIRE AREAS RC-11 (ZONE A - CABLE SPREADING ROOM) AND RC-IX (REMOTE SHUTDOWN ROOM). THE PENETRATION WAS IMMEDIATELY PLACED ON THE HOURLY FIRE TOUR AND A MAINTENANCE WORK REQUEST (MWR) WAS WRITTEN TO HAVE THE SEAL REPAIRED. THE CAUSE OF THIS EVENT HAS BEEN DETERMINED TO BE INADEQUATE TRAINING ON FIRE PROTECTION BARRIER IDENTIFICATION AND IMPAIRMENT REQUIREMENTS PROVIDED TO PLANT TECHNICAL ENGINEERS, CONTRACTOR FIELD ENGINEERS, CONTRACTOR CRAFT AND PLANT QUALITY CONTROL (QC) INSPECTORS. FURTHER CORRECTIVE ACTIONS INCLUDE 1) PROVIDING TRAINING ON FIRE-RATED PENETRATION REQUIREMENTS TO PLANT TECHNICAL ENGINEERS, PLANT MAINTENANCE ENGINEERS, PLANT QC INSPECTORS, PLANT CRAFT AND CONTRACTOR FIELD ENGINEERS AND 2) REVISING THE CONTRACTOR CRAFT TRAINING PROGRAM TO BETTER EXPLAIN FIRE PROTECTION SYSTEM IMPAIRMENT PERMIT REQUIREMENTS.

[375] ZION 2 DOCKET 50-304 LER 87-008
CONTAINMENT ISOLATION TECH SPEC VIOLATION DUE TO OUT OF SERVICE AUXILIARY FEEDWATER DISCHARGE STOP VALVE.
EVENT DATE: 090887 REPORT DATE: 100887 NSSS: WE TYPE: PWR
VENDOR: LIMITORQUE CORP.

(NSIC 206770) FOLLOWING THE COMPLETION OF THE AUXILIARY FEEDWATER (AFW) PERIODIC TEST (PI-7), IT WAS DISCOVERED THAT THE DISCHARGE STOP VALVE 2MOV-FW0055 TO THE 2A STEAM GENERATOR WAS INDICATING CLOSED BUT WAS ACTUALLY THROTTLED OPEN TO THE REQUIRED STANDBY FLOW. ON SEPTEMBER 8, 1987, A WORK REQUEST TO REPAIR WAS INITIATED, AND THE VALVE WAS TAKEN OUT OF SERVICE IN THE THROTTLED POSITION. THE

UNIT WAS IN MODE 1 (POWER OPERATIONS) AT THE TIME. PER ZION TECH SPECS, TABLE 3.9-3D, 2MOV-F20055 IS REQUIRED TO BE OPERABLE OR CLOSED IN MODE 1. ALTHOUGH THE LICENSED SHIFT SUPERVISOR DID CONSULT THE TECH SPEC ON THE AFW SYSTEM, HE DID NOT CONSULT THE TECH SPECS ON CONTAINMENT ISOLATION BEFORE AUTHORIZING THE OUT OF SERVICE. ON 9/8/87 AT 1447 HRS, THE VALVE WAS DE-ENERGIZED IN ITS THROTTLED POSITION, RENDERING THE VALVE INOPERABLE, CONTRARY TO THE REQUIREMENTS OF TECH SPECS 3.9.3. THIS CONDITION LASTED UNTIL 9/9/87 WHEN THE VALVE WAS RETURNED TO OPERABLE STATUS AT 1230 HRS. THERE WAS NO SAFETY SIGNIFICANCE, BECAUSE THIS VALVE IS REQUIRED TO BE OPEN TO SUPPLY AUXILIARY FEEDWATER FOLLOWING A REACTOR TRIP, AND BECAUSE THE VALVE COULD BE CLOSED LOCALLY IF NEEDED. THIS VALVE HAS CONFLICTING DESIGN REQUIREMENTS IN THAT IT PERFORMS AN ENGINEERED SAFETY FEATURE (ESF) FUNCTION AND A CONTAINMENT ISOLATION FUNCTION.

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

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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.

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SYSTEM INDEX

SYSTEM INDEX

SYSTEM INDEX

KEYWORD INDEX

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KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

VENDOR CODE INDEX

VENDOR CODE INDEX

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Monthly Report

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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-1061, 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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