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Licensee Event Report (LER) Compilation

For month of March 1991

Oak Ridge National Laboratory

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Oak Ridge National Laboratory
Nuclear Operations Analysis Center
Oak Ridge, TN 37831

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Abstract

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

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

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[1] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 90-020
DESIGN DEFICIENCY RESULTS IN POTENTIAL FOR STRUCTURAL DAMAGE OR FAILURE OF CONTAINMENT POLAR CRANE DURING DESIGN BASIS ACCIDENT CONDITIONS.
EVENT DATE: 103090 REPORT DATE: 011091 NSSS: BW TYPE: PWR

(NSIC 220660) ON 10/30/90 DURING A REFUELING OUTAGE, DESIGN ENGINEERING PERSONNEL DETERMINED THAT AN ANALYSIS HAD NEVER BEEN PERFORMED TO ENSURE THAT CERTAIN STRUCTURAL COMPONENTS OF THE ANO-1 CONTAINMENT BUILDING POLAR CRANE COULD WITHSTAND THE EFFECTS OF A RAPID INCREASE IN CONTAINMENT PRESSURE DURING A LOSS OF COOLANT ACCIDENT WITHOUT SUSTAINING STRUCTURAL DAMAGE. THE POTENTIAL CONCERN WAS THAT INADEQUATE VENTING COULD RESULT IN A LARGE DIFFERENTIAL PRESSURE ACROSS THE CRANE'S GIRDERS WHICH MIGHT CAUSE THE GIRDERS TO YIELD OR COLLAPSE ALLOWING THE CRANE OR A PART OF THE CRANE STRUCTURE TO FALL FROM ITS STORED POSITION. FURTHER INVESTIGATIONS DETERMINED THAT THE ANO-2 POLAR CRANE, WHICH IS SIMILARLY DESIGNED, HAD BEEN MODIFIED DURING THE CONSTRUCTION PHASE OF THE UNIT TO ADDRESS THE SAME CONCERN. THE ANO-1 POLAR CRANE VENDOR WAS CONSULTED AND AN ANALYSIS WAS PERFORMED WHICH INDICATED THAT MODIFICATIONS WERE NECESSARY TO ENSURE THE CRANE COMPONENTS WERE ADEQUATELY VENTED. THE CRANE WAS MODIFIED BY CUTTING VENT HOLES IN THE BRIDGE GIRDERS, TROLLEY SIDES AND END TRUCKS. THE ROOT CAUSE WAS DETERMINED TO BE AN OVERSIGHT BY THE ANO-1 ARCHITECT ENGINEER DURING THE CONSTRUCTION PHASE OF ANO-1.

[2] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 90-019
INADVERTENT ACTUATION OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM DUE TO A SPURIOUS TRIP CAUSED BY AN UNSOLDERED CONNECTION WHICH RESULTED FROM A MANUFACTURING/PRODUCTION DEFECT.
EVENT DATE: 120990 REPORT DATE: 010891 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: ARKANSAS NUCLEAR 2 (PWR)

(NSIC 220659) ON 12/9/90, AT APPROXIMATELY 2228, AN INADVERTENT ACTUATION OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM (CREVS) OCCURRED. AT THE TIME OF THE ACTUATION, OPERATIONS PERSONNEL OBSERVED THAT THE INDICATION OF THE ANO-2 CONTROL ROOM (CR) VENTILATION RADIATION MONITOR (2RE-8750-1) FAILED LOW, THEN INCREASED TO THE TRIP SETPOINT AND ACTUATED THE CREVS. 2RE-8750-1 WAS RESET AND THE VENTILATION LINEUP WAS RETURNED TO NORMAL. HOWEVER, AT 2235, THE MONITOR WAS DECLARED INOPERABLE AND THE CR WAS ISOLATED AND VENTILATION WAS PLACED IN THE RECIRCULATION MODE. THE IMMEDIATE CAUSE OF THIS EVENT WAS DETERMINED TO BE AN UNSOLDERED ELECTRICAL CONNECTION ON THE RADIATION MONITOR OPERATION SELECTOR SWITCH. THE SWITCH WAS REPAIRED TO NORMAL AT 0855 ON 12/12/90. A REVIEW OF THE MAINTENANCE RECORDS WAS CONDUCTED AND NO DOCUMENTATION WAS FOUND TO INDICATE THAT PREVIOUS MAINTENANCE HAD BEEN PERFORMED ON THE SWITCH. THEREFORE, IT WAS CONCLUDED THAT THE MOST LIKELY ROOT CAUSE OF THIS EVENT WAS A MANUFACTURING/PRODUCTION DEFECT. OTHER ANO-2 TECH SPECS RADIATION MONITORS WERE VISUALLY INSPECTED. SINCE ONLY ONE ADDITIONAL UNSOLDERED CONNECTION WAS IDENTIFIED, IT WAS CONCLUDED THAT THIS CONDITION WAS NOT A GENERIC PROBLEM.

[3] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 90-022
REACTOR TRIP DURING PLANT HEATUP DUE TO PERSONNEL ERROR WHILE SHIFTING REACTOR COOLANT PUMPS.
EVENT DATE: 121890 REPORT DATE: 011791 NSSS: BW TYPE: PWR

(NSIC 220815) ON DECEMBER 18, 1990, WHILE CONDUCTING A PLANT HEATUP IN PREPARATION FOR STARTUP, AN AUTOMATIC REACTOR TRIP WAS INITIATED BY THE REACTOR PROTECTION SYSTEM (RPS) UPON SENSING NO REACTOR COOLANT PUMPS (RCPS) RUNNING IN THE "B" REACTOR COOLANT SYSTEM (RCS) LOOP. AT THE TIME OF THE TRIP, RCPS P-32C AND P-32D WERE RUNNING IN RCS LOOP 'A' AND P-32A WAS RUNNING IN LOOP 'B'. RCPS WERE BEING BALANCED TO REDUCE VIBRATION IN ACCORDANCE WITH AN APPROVED PROCEDURE. THE OPERATORS WERE REQUESTED TO SHIFT FROM P-32A TO P-32B IN RCS LOOP 'B'. AFTER REVIEWING THE RCP OPERATING PROCEDURE, THE INVOLVED OPERATORS ASKED THE SHIFT SUPERVISOR (SS) IF HE WISHED TO STOP P-32A AND START P-32B. THE SS GAVE AN AFFIRMATIVE RESPONSE. AT THAT TIME, A TRAINEE UNDER THE SUPERVISION OF A SENIOR REACTOR OPERATOR, STOPPED P-32A. A REACTOR TRIP THEN OCCURRED DUE TO ZERO PUMPS RUNNING IN THE 'B' RCS LOOP. THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR. AN INADEQUATE PROCEDURE WAS A CONTRIBUTING FACTOR. THE RCP OPERATING PROCEDURE

CONTAINED NO CAUTIONS REGARDING THE POSSIBILITY OF INITIATING TRIP WHEN STOPPING RCPS. A CREW BRIEFING WAS HELD WITH THE CREW INVOLVED TO DISCUSS THIS EVENT AND ITS SIGNIFICANCE. THE RCP OPERATING PROCEDURE WILL BE REVISED TO INCLUDE ADDITIONAL GUIDANCE REGARDING SHIFTING RCPS.

[4] ARKANSAS NUCLEAR 2 DOCKET 50-368 LER 86-003 REV 02
UPDATE ON INOPERABLE FIRE DAMPERS RESULT IN TECHNICAL SPECIFICATION VIOLATION DUE TO FAILURE TO PERFORM FUNCTIONAL TESTING FOLLOWING INSTALLATION.
EVENT DATE: 031286 REPORT DATE: 021591 NSSS: CE TYPE: PWR
VENDOR: AMERICAN WARMING & VENTILATING INC.
RUSKIN MANUFACTURING COMPANY

(NSIC 220991) DURING THE INITIAL PERFORMANCE OF PERIODIC FUNCTIONAL TESTING OF FIRE DAMPERS, A TOTAL OF 19 FIRE DAMPERS WERE IDENTIFIED AS INOPERABLE. THE TESTING INVOLVES REMOVAL OF THE FIRE DAMPER FUSIBLE LINK AND VERIFYING THAT THE FIRE DAMPER COMPLETELY CLOSES IN THE PRESENCE OF NORMAL VENTILATION AIR FLOW. OF THE 19 INOPERABLE FIRE DAMPERS, 9 FAILURES WERE ATTRIBUTED TO MECHANICAL INTERFERENCE AND 10 WERE ATTRIBUTED TO A DESIGN DEFICIENCY OF THE FIRE DAMPER. THE CAUSE OF THIS EVENT WAS INADEQUATE FUNCTIONAL TESTING OF INSTALLED FIRE DAMPERS IN THAT THE ABILITY OF THE FIRE DAMPERS TO COMPLETELY CLOSE WITH NORMAL VENTILATION AIR FLOW HAD NOT BEEN PREVIOUSLY VERIFIED. AS A RESULT OF THIS EVENT, THE FIRE DAMPERS THAT FAILED TO COMPLETELY CLOSE DUE TO MECHANICAL INTERFERENCE WERE REPAIRED AND SUCCESSFULLY TESTED. A PLANT MODIFICATION HAS BEEN IMPLEMENTED TO REPLACE THE FIRE DAMPERS THAT FAILED TO COMPLETELY CLOSE UNDER NORMAL VENTILATION AIR FLOW. PERIODIC FUNCTIONAL TESTING WILL BE DISCONTINUED TO ELIMINATE THE POTENTIAL FOR PERSONNEL INJURY OR EQUIPMENT DAMAGE. THE PERFORMANCE OF FUNCTIONAL TESTS FOLLOWING MAINTENANCE OR MODIFICATION ACTIVITIES COMBINED WITH THE TECH SPEC REQUIRED VISUAL INSPECTIONS WILL ENSURE THE CONTINUED OPERABILITY OF THE FIRE DAMPERS.

[5] ARKANSAS NUCLEAR 2 DOCKET 50-368 LER 90-024
INADEQUATE PREVENTIVE MAINTENANCE PROGRAM FOR STEAM TURBINE DRIVEN EMERGENCY FEEDWATER PUMP RESULTS IN DEGRADED TURBINE GOVERNOR SYSTEM AND SUBSEQUENT OVERSPEED TRIPS OF TURBINE.
EVENT DATE: 120590 REPORT DATE: 010491 NSSS: CE TYPE: PWR
VENDOR: WOODWARD GOVERNOR COMPANY

(NSIC 220677) ON DECEMBER 5, 1990 BASED ON EVALUATIONS OF TWO PREVIOUS EVENTS INVOLVING OVERSPEED TRIPS OF THE STEAM TURBINE DRIVEN EMERGENCY FEEDWATER PUMP, IT WAS CONCLUDED THAT THE CAUSE OF THE TRIPS HAD BEEN WATER SLUGGING OF THE TURBINE ON STARTUP DUE TO CONDENSATE ACCUMULATION IN THE STEAM SUPPLY LINE TO THE TURBINE. FOLLOWING ANOTHER OVERSPEED TRIP ON DECEMBER 6, 1990, THE ACTUAL CAUSE FOR THE TURBINE TRIPS WAS FOUND TO BE SLUGGISH RESPONSE OF THE TURBINE GOVERNOR VALVE DUE TO A CONTAMINATED CONTROL OIL SYSTEM. THE ROOT CAUSE WAS CONSIDERED TO BE INADEQUACIES IN THE PREVENTIVE MAINTENANCE PROGRAM. THE PROGRAM DID NOT APPROPRIATELY ADDRESS AND MINIMIZE THE POTENTIAL EFFECTS OF OIL CONTAMINATION AND DEGRADATION OF GOVERNOR COMPONENTS OVER TIME. FOLLOWING THE LAST OVERSPEED TRIP, THE OIL AND OIL FILTER ASSEMBLY WERE CHANGED, A HYDRAULIC ACTUATOR WAS REPLACED AND A REMOTE SERVO VALVE AND CONTROL OIL TUBING WERE CLEANED. THE TURBINE IS BEING TESTED ON AN INCREASED FREQUENCY AND THE OIL QUALITY IS BEING MONITORED TO ENSURE IT IS NOT DEGRADING. LONG TERM ACTIONS INCLUDE PROCEDURE REVISIONS TO INCLUDE PERIODIC CLEANING AND/OR REPLACEMENT OF CONTROL OIL SYSTEM COMPONENTS. ADDITIONALLY, THE TURBINE OIL SYSTEM WILL BE CLEANED TO REMOVE VARNISH AND HARDENED OIL DEPOSITS DURING THE NEXT REFUELING OUTAGE.

[6] ARNOLD DOCKET 50-331 LER 91-001
MANUAL SCRAM SHUTDOWN OF PLANT DUE TO STEAM LINE IN THE HEATER BAY.
EVENT DATE: 010691 REPORT DATE: 013091 NSSS: GE TYPE: BWR

(NSIC 220941) ON 1/6/91, WITH THE PLANT OPERATING AT 95% POWER, A CONTROLLED REACTOR SHUTDOWN WAS INITIATED DUE TO A STEAM LEAK IN THE HEATER BAY. AT 1107 OPERATORS RAPIDLY REDUCED RECIRCULATION FLOW TO MINIMUM IN PREPARATION FOR INSERTION OF A MANUAL SCRAM. THE DECISION TO MANUALLY SCRAM WAS CONSERVATIVELY

MADE TO ENSURE HEATER BAY TEMPERATURES WOULD NOT CHALLENGE THE MAIN STEAM LINE ISOLATION SETPOINT OF 200F. AT 1113, WITH REACTOR POWER AT APPROX. 60%, A MANUAL SCRAM WAS INSERTED. THE INTERMEDIATE CAUSE OF THIS EVENT WAS A BREAK IN A TWO INCH EXTRACTION STEAM DRAIN LINE JUST BELOW THE WELDED JOINT WHICH ATTACHES THE DRAIN LINE TO A 12 INCH EXTRACTION STEAM LINE. THE CAUSE FOR THE BREAK IN THE PIPE WAS CYCLIC FATIGUE DUE TO THE RELATIVE MOVEMENT OF THE TWO INCH AND TWELVE INCH PIPE DURING PLANT OPERATION. THE SECTION OF TWO INCH PIPE, WHERE THE BREAK OCCURRED, WAS REPLACED WITH A NEW PIPE. THE NEW PIPE WAS CONSTRUCTED WITH AN EXPANSION LOOP TO COMPENSATE FOR RELATIVE MOVEMENT BETWEEN THE TWELVE AND TWO INCH PIPES. THIS EVENT HAD NO EFFECT ON THE SAFE OPERATION OF THE PLANT. FOLLOWING THE SCRAM THE PLANT WAS QUICKLY BROUGHT TO A STABLE CONDITION.

[7] BEAVER VALLEY 1 DOCKET 50-334 LER 90-020
COMPUTER FAILURE CAUSES INOPERABLE FLUX DIFFERENCE MONITOR.
EVENT DATE: 121790 REPORT DATE: 011791 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 220819) ON 12/17/90 AT 1030 HOURS, A REVIEW OF THE P250 PLANT COMPUTER LOGS DETERMINED THAT ON 12/13/90 AT 1030 HOURS, A MEMORY ALLOCATION ERROR HAD OCCURRED IN COMPUTER OPERATION. THIS ERROR HAD PERSISTED UNTIL THE COMPUTER WAS REBOOTED ON 12/15/90 AT 0207 HOURS. DURING THIS TIME, ALL THE COMPUTER'S AVERAGED PLANT PARAMETER VALUES WERE INACCURATE. PLANT PROCEDURES USE SEVERAL OF THESE AVERAGED VALUES TO VERIFY COMPLIANCE WITH TECHNICAL SPECIFICATION REQUIREMENTS FOR THE SECONDARY HEAT BALANCE AND THE REACTOR'S AXIAL FLUX DIFFERENCE. ON 12/17/90, OPERATORS VERIFIED THAT THE ROUTINES TO CALCULATE THESE AVERAGED VALUES WERE FUNCTIONING PROPERLY. OPERATORS HAD ONLY PERFORMED ONE SECONDARY HEAT BALANCE CALCULATION DURING THE PERIOD WHEN THE AVERAGED DATA WAS INACCURATE. THE STATION REEVALUATED THIS HEAT BALANCE USING LOGGED ACCURATE DATA FOR THAT PERIOD AND VERIFIED THAT NO THERMAL OR OPERATIONAL LIMITS WERE EXCEEDED. A SIMILAR REVIEW OF LOGGED VALUES FOR AXIAL FLUX VERIFIED A STABLE FLUX DISTRIBUTION DURING THIS EVENT. THE STATION HAS INITIATED A PROCEDURE TO VERIFY PROPER COMPUTER OPERATION PRIOR TO USING DATA FROM THE P250 FOR TECHNICAL SPECIFICATION REQUIRED CALCULATIONS. THERE WERE NO SAFETY IMPLICATIONS DUE TO THIS EVENT AS ALL PLANT PARAMETERS REMAINED WITHIN THEIR TECHNICAL SPECIFICATION REQUIRED RANGES.

[8] BEAVER VALLEY 1 DOCKET 50-334 LER 90-019
CLOSURE OF MAIN STEAM TRIP VALVE DURING PARTIAL STROKE TESTING.
EVENT DATE: 122690 REPORT DATE: 012491 NSSS: WE TYPE: PWR
VENDOR: NAMCO CONTROLS

(NSIC 220818) ON 12/26/90, UNIT LOAD WAS REDUCED TO 30% TO SUPPORT BORIC ACID ADDITIONS TO THE SECONDARY PLANT. WHILE THE STATION WAS AT REDUCED LOAD, OPERATORS PERFORMED THE PARTIAL STROKE TEST FOR THE B MAIN STEAM ISOLATION VALVE (MSIV). THIS PROCEDURE VERIFIES FREE MOVEMENT OF THE MSIV BY CYCLING IT OPENED AND CLOSED 3 DEGREES USING THE INSTALLED TEST CIRCUITRY. AT 1110 HOURS, DURING THE PERFORMANCE OF THIS TEST, THE VALVE FULLY CLOSED, ISOLATING STEAM FROM THE B STEAM GENERATOR. OPERATORS INITIATED THE EMERGENCY SHUTDOWN PROCEDURE AND PERFORMED A MANUAL PLANT SHUTDOWN. BY 1140 HOURS, THE PLANT WAS IN OPERATIONAL MODE 3 (HOT STANDBY) WITH ALL CONTROL RODS INSERTED. THE REMAINING SHUTDOWN RODS WERE MANUALLY INSERTED AND THE REACTOR TRIP BREAKERS WERE MANUALLY OPENED AT 1404 HOURS. INVESTIGATION DETERMINED THAT THE CAUSE OF THIS EVENT WAS INTERMITTENT BINDING OF THE 3 DEGREE PARTIAL CLOSURE LIMIT SWITCH IN THE MSIV TEST CIRCUIT. THIS LIMIT SWITCH HAS BEEN REPAIRED AND A DESIGN CHANGE HAS BEEN INITIATED TO IMPROVE LIMIT SWITCH RELIABILITY. THERE WERE NO SAFETY IMPLICATIONS DUE TO THIS EVENT. THE FAILED LIMIT SWITCH IS ONLY USED BY THE TEST CIRCUIT AND HAS NO SAFETY RELATED FUNCTION.

[9] BEAVER VALLEY 2 DOCKET 50-412 LER 90-006 REV 01
UPDATE ON CONTAINMENT ISOLATION VALVES EXCEED STROKE TIME LIMIT.
EVENT DATE: 060590 REPORT DATE: 021491 NSSS: WE TYPE: PWR
VENDOR: AUTOMATIC SWITCH COMPANY (ASCO)
FISHER CONTROLS CO.

(NSIC 220962) ON 6/5/90 AT 1315 HOURS, WITH THE UNIT AT 86% REACTOR POWER, SURVEILLANCE TESTING OF THE LETDOWN ORIFICE ISOLATION VALVES, 2CHS*AOV200A,B,C WAS IN PROGRESS. THIS TESTING IDENTIFIED THAT ONE SET OF SERIES SOLENOID VALVES, FOR 2CHS*AOV200A,B,C FAILED TO MEET THE TECHNICAL SPECIFICATION (TS) 3.6.3.1 CONTAINMENT ISOLATION PHASE A (CIA) CLOSURE LIMIT OF LESS THAN 10 SECONDS. THE VALVES WERE DECLARED INOPERABLE. NORMAL REACTOR COOLANT SYSTEM CHARGING AND LETDOWN WERE ISOLATED AND EXCESS LETDOWN WAS PLACED IN SERVICE. THE OUTSIDE CONTAINMENT ISOLATION VALVE WAS CLOSED ISOLATING THIS PENETRATION. THE CAUSES FOR THIS EVENT WERE A DEFICIENCY IN THE SURVEILLANCE PROGRAM AND DIFFERENCES BETWEEN THE DESIGN CONFIGURATION AND THE INSTALLED SOLENOID VALVE ARRANGEMENT FOR 2CHS*AOV200A,B,C. A TEMPORARY WAIVER OF COMPLIANCE FOR TS 3.6.3.1 WAS OBTAINED FROM THE NUCLEAR REGULATORY COMMISSION ALLOWING CONTINUED OPERATION WITH NORMAL CHARGING AND LETDOWN IN SERVICE. THERE WERE NO SAFETY IMPLICATIONS AS A RESULT OF THIS EVENT. THE AS FOUND CLOSURE TIMES OF 2CHS*AOV200A,B,C ARE BOUNDED BY THE CLOSURE TIME LIMIT ESTABLISHED FOR THE OUTSIDE CONTAINMENT ISOLATION VALVE, ENSURING THAT THE PENETRATION IS ISOLATED WITHIN THE TIME REQUIREMENTS STATED IN THE UPDATED FINAL SAFETY ANALYSIS.

[10]   BEAVER VALLEY 2   DOCKET 50-412   LER 90-007 REV 01
UPDATE ON OPERATION WITH REFUELING CAVITY DRAIN FLANGES INSTALLED.
EVENT DATE: 062290   REPORT DATE: 012391   NSSS: WE   TYPE: PWR

(NSIC 220971) DURING REVIEW OF IE NOTICE 90-19, POTENTIAL LOSS OF EFFECTIVE VOLUME FOR CONTAINMENT RECIRCULATION SPRAY AT PWR FACILITIES, NO DOCUMENTATION COULD BE FOUND VERIFYING THE REMOVAL OF UNIT 2'S REFUELING CAVITY TRANSFER CANAL DRAIN FLANGES. THE UFSAR ASSUMES THESE FLANGES ARE REMOVED DURING POWER OPERATION TO PREVENT THE TRANSFER CANAL FROM ACTING AS A CATCH BASIN IN THE EVENT OF A CONTAINMENT SPRAY ACTUATION. ON 6/22/90, A CONTAINMENT ENTRY WAS PERFORMED TO DETERMINE IF THE FLANGES WERE INSTALLED. THE FLANGES WERE FOUND TO BE INSTALLED AND WERE IMMEDIATELY REMOVED. THE FLANGES WERE LEFT INSTALLED DUE TO A DEFICIENCY IN THE TRANSFER CANAL DRAINING PROCEDURE. THIS PROCEDURE HAS BEEN REVISED TO VERIFY THAT THE FLANGES ARE REMOVED AFTER THE CANAL IS DRAINED. THERE WERE NO SAFETY IMPLICATIONS DUE TO THIS EVENT. AN ENGINEERING EVALUATION HAS VERIFIED THAT THE CONTAINMENT SPRAY SYSTEM WAS STILL OPERABLE WHILE THE FLANGES WERE INSTALLED.

[11]   BRAIDWOOD 1   DOCKET 50-456   LER 90-021
REACTOR TRIP AS A RESULT OF A FEEDWATER PUMP TRIP DUE TO LOW OIL PRESSURE.
EVENT DATE: 120190   REPORT DATE: 123190   NSSS: WE   TYPE: PWR

(NSIC 221097) AT 1645 ON 12/1/90 THE 1B FEEDWATER PUMP (FW), ONE OF THE TWO ON LINE FW PUMPS, TRIPPED. THE NUCLEAR STATION OPERATOR (NSO) INITIATED A TURBINE RUNBACK TO 60% POWER IN ACCORDANCE WITH PROCEDURE. STEAM GENERATOR (SG) LEVELS DECREASED DUE TO BOTH THE REDUCTION IN FW FLOW FROM THE LOSS OF THE FW PUMP AND THE "SHRINK" EFFECT ON THE SG LEVEL INSTRUMENTATION FROM THE REDUCTION IN STEAM FLOW AS A RESULT OF THE TURBINE RUNBACK. AT 1647 THE LEVEL IN THE 1B SG REACHED THE LO-2 REACTOR TRIP SETPOINT AND A REACTOR TRIP, TURBINE TRIP, FEEDWATER ISOLATION, AND AUX. FW AUTOMATIC INITIATION OCCURRED AS DESIGNED. ALL COMPONENTS ASSOCIATED WITH THESE ACTUATIONS FUNCTIONED AS DESIGNED. THE CAUSE OF THE FW PUMP TRIP WAS LOW OIL PRESSURE. IT IS BELIEVED THAT THE SUCTION OF THE HIGH PRESSURE OIL PUMP BECAME PARTIALLY PLUGGED FROM A SLUDGE BURST. THE STANDBY OIL PUMP STARTED BUT THE LOW PRESSURE "DIP" THAT OCCURRED WAS OF SUFFICIENT MAGNITUDE TO REACH THE LOW OIL PRESSURE TRIP SETPOINT. THE TURBINE RUNBACK WAS INITIATED, BUT EQUILIBRIUM FW FLOW/STEAM FLOW WAS NOT ACHIEVED PRIOR TO REACHING THE REACTOR TRIP SETPOINT. A CONTRIBUTING CAUSE TO THE EVENT WAS A PROCEDURAL DEFICIENCY. THE PROCEDURE DID NOT ADDRESS CLOSING THE RECIRCULATION VALVE ON THE TRIPPED FW PUMP. THE OIL SYSTEM HAS BEEN CLEANED AND IS BEING MONITORED. THE PROCEDURE HAS BEEN REVISED. NO PREVIOUS OCCURRENCES.

[12]   BRAIDWOOD 1   DOCKET 50-456   LER 90-023
REACTOR TRIP CAUSED BY MAIN GENERATOR PHASE C GROUND FAULT.
EVENT DATE: 123090   REPORT DATE: 012491   NSSS: WE   TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 220853) ON 12/30/90, UNIT 1 WAS LOAD FOLLOWING TO ACCOMODATE SYSTEM DEMAND FOR THE COMMONWEALTH EDISON COMPANY SYSTEM LOAD DISPATCHER. AT 0815 THE UNIT COMPLETED A POWER ASCENSION TO FULL CAPABILITY. AT 0821 A GENERATOR NEUTRAL GROUND OVERCURRENT PROTECTIVE RELAY ACTUATED AND TRIPPED THE UNIT 1 MAIN GENERATOR. A TURBINE AND REACTOR TRIP FOLLOWED AS DESIGNED. MEGGER TESTING REVEALED A GROUND ON THE "C" PHASE OF THE MAIN GENERATOR. THE CAUSE OF THE GROUND WAS AN INTERNAL GENERATOR DEFECT. UPON APPLICATION OF HI POTENTIAL ALTERNATING CURRENT TO THE "C" PHASE, OBSERVATION OF SMOKE AND ELECTRICAL ARCING REVEALED THAT THE FAULT WAS IN THE BOTTOM COIL IN SLOT 29 OF THE STATOR. DAMAGE TO THE COIL WAS ATTRIBUTED TO A VENT SPACER THAT CAME LOOSE. VIBRATION DURING NORMAL OPERATION ALLOWED THE SPACER TO RUB AND WEAR DOWN INSULATION PROTECTING THE COIL. THE GROUND WAS CREATED AS A RESULT OF INSULATION BREAKDOWN. THE COIL WAS REMOVED FROM THE STATOR AND SENT TO THE VENDOR. THIS COMPONENT FAILURE IS CONSIDERED TO BE AN ISOLATED EVENT. THERE HAVE BEEN PREVIOUS OCCURRENCES OF A REACTOR TRIP CAUSED BY A GENERATOR TRIP. PREVIOUS CORRECTIVE ACTIONS AND CONTRIBUTING ROOT CAUSE ARE NOT APPLICABLE TO THIS EVENT.

[13]   BROWNS FERRY 1   DOCKET 50-259   LER 90-015
UNPLANNED ENGINEERED SAFETY FEATURES ACTUATION CAUSED BY PERSONNEL ERROR.
EVENT DATE: 091690   REPORT DATE: 101590   NSSS: GE   TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
                      BROWNS FERRY 3 (BWR)

(NSIC 220886) ON SEPTEMBER 16, 1990 AT 1235 HOURS AN UNPLANNED ENGINEERED SAFETY FEATURES (ESF) ACTUATION (JE) OCCURRED WHEN UNIT 1/2 DIESEL GENERATOR (DG) "D" AUTOMATICALLY STARTED AFTER RECEIVING AN EMERGENCY CORE COOLING SYSTEM (ECCS) INITIATION SIGNAL OF LOW-LOW-LOW REACTOR WATER LEVEL. OTHER ESF EQUIPMENT THAT WOULD HAVE STARTED UPON RECEIPT OF THIS SIGNAL DID NOT START BECAUSE IT HAD BEEN TAGGED OUT OF SERVICE. THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR RESULTING FROM AN INADEQUATE EVALUATION OF THE CONSEQUENCES OF WORK ACTIVITIES ON TWO REACTOR VESSEL WATER LEVEL TRANSMITTERS. THE INDIVIDUAL RESPONSIBLE FOR PERFORMING THIS EVALUATION FAILED TO IDENTIFY THAT THE TWO LEVEL TRANSMITTERS WERE A PART OF THE ECCS INITIATION LOGIC. IMMEDIATE CORRECTIVE ACTION WAS TAKEN TO REMOVE THE RELAYS FOR THESE TRANSMITTERS AND RESET THE ECCS INITIATION LOGIC. THE INDIVIDUAL RESPONSIBLE FOR PERFORMING THE EVALUATION RECEIVED DISCIPLINARY ACTION FROM PLANT MANAGEMENT. IN ADDITION, THIS EVENT WILL BE REVIEWED WITH PERSONNEL RESPONSIBLE FOR EVALUATING THE IMPACT OF WORK ACTIVITIES TO STRESS THE IMPORTANCE OF PROPERLY AND THOROUGHLY COMPLETING THE IMPACT EVALUATION SHEET FOR EACH WORK ACTIVITY TO BE PERFORMED.

[14]   BROWNS FERRY 1   DOCKET 50-259   LER 90-021
HOURLY FIRE WATCH COULD NOT ENTER VITAL AREA, PLACING THE PLANT IN VIOLATION OF TECHNICAL SPECIFICATIONS.
EVENT DATE: 092790   REPORT DATE: 012291   NSSS: GE   TYPE: BWR

(NSIC 220749) ON DECEMBER 22, 1990, IT WAS DETERMINED THAT ON SEPTEMBER 27, 1990, PROPER COMPENSATORY ACTIONS HAD NOT BEEN TAKEN FOR FIRE PROTECTION DETECTION SYSTEMS OUT OF SERVICE WHEN AN HOURLY ROVING FIRE WATCH COULD NOT ENTER "A" 4160V SHUTDOWN BOARD ROOM TO PERFORM A VISUAL INSPECTION OF THE AREA, THUS VIOLATING TECHNICAL SPECIFICATIONS. TVA IS PRESENTLY CONDUCTING AN INVESTIGATION OF THIS EVENT. TVA WILL REPORT THE RESULTS IN A SUPPLEMENT TO THIS LER. THE SUPPLEMENT WILL BE SUBMITTED BY FEBRUARY 14, 1991.

[15]   BROWNS FERRY 1   DOCKET 50-259   LER 90-018 REV 01
UPDATE ON NINE FIREWATCH OBSERVATIONS WERE PERFORMED LATE DUE TO PERSONNEL ERROR THEREBY EXCEEDING TECHNICAL SPECIFICATION REQUIREMENTS.
EVENT DATE: 121190   REPORT DATE: 021391   NSSS: GE   TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
                      BROWNS FERRY 3 (BWR)

(NSIC 221034) ON DECEMBER 11, 1990 AT 2148, THE SECURITY MICRO ACCESS COMPUTER (MAC)-540 MALFUNCTIONED AND AUTOMATICALLY SHUTDOWN IN THE DEGRADE MODE. THIS RESULTED IN REQUIRING A SECURITY OFFICER TO BE DISPATCHED TO LOCK VITAL AREA

DOORS FOR ENSURING UNAUTHORIZED ACCESS DID NOT OCCUR. THE LOCKING OF VITAL AREA DOORS DETAINED A FIRE WATCH FROM ENTERING BATTERY BOARD ROOM NUMBER 1 ELEVATION 593 IN THE CONTROL BAY BUILDING. THIS CONTRIBUTED TO A REQUIRED TECHNICAL SPECIFICATION (TS) HOURLY FIREWATCH TOUR FROM BEING PERFORMED ON TIME. THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR DUE TO AN INADEQUATE JOB PLAN (I.E., INADEQUATE TIME SCHEDULED FOR FIREWATCH TOUR CONTINGENCIES). CONTRIBUTING CONTINGENCIES WERE: (1) TIME SPENT BY THE SECURITY OFFICER TO FIND THE FIREWATCH, (2) THE SECURITY OFFICER'S AND FIREWATCH'S COMMUNICATION WAS NOT PROACTIVE, (3) THE ATTEMPTS TO CONTACT THE SHIFT OPERATING SUPERVISOR WERE UNSUCCESSFUL, AND (4) THE CURRENT MAC-540 SYSTEM IS SUSCEPTIBLE TO ELECTRICAL SPIKES AND FAILURES DUE TO AGING OF THE ELECTRONIC COMPONENTS. THE IMMEDIATE CORRECTIVE ACTIONS WERE TO REPROGRAM THE MAC-540 COMPUTER. THE SECURITY OFFICER SEARCHED AND FOUND THE FIREWATCH TO ALLOW THE FIREWATCH TO CONTINUE THE HOURLY FIREWATCH TOUR.

[16] BROWNS FERRY 1 DOCKET 50-259 LER 90-019 REV 01
UPDATE ON LOSS OF POWER ON INSTRUMENT AND CONTROL BUS 1A RESULTED IN LOSS OF VARIOUS FIRE PROTECTION PANELS PLACING THE PLANT OUTSIDE TECHNICAL SPECIFICATIONS.
EVENT DATE: 121290 REPORT DATE: 021491 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 221035) ON DECEMBER 12, 1990 AT 1245 HOURS, AN HOURLY FIRE WATCH WAS NOT ESTABLISHED WHEN INSTRUMENT AND CONTROL BUS 1A TRIPPED, RESULTING IN A LOSS OF POWER TO VARIOUS FIRE PROTECTION SYSTEM FIRE DETECTION PANELS. THE ROOT CAUSE OF THIS EVENT IS THAT THE PLANT PROCEDURE FOR LOSS OF INSTRUMENT AND CONTROL BUS 1A DID NOT GIVE CLEAR GUIDANCE FOR WATCH AREAS. THE CORRECTIVE ACTIONS TAKEN DURING THE EVENT INCLUDED THE RETURN OF INSTRUMENT AND CONTROL BUS 1A TO SERVICE AND AN ATTEMPT WAS MADE TO ESTABLISH COMPENSATORY ACTIONS TO MEET TECHNICAL SPECIFICATIONS FOR LOSS OF FIRE DETECTION EQUIPMENT. TO PREVENT RECURRENCE, OPERATIONS WILL CREATE A LIST THAT IDENTIFIES AREAS THAT WILL NEED A FIREWATCH UPON LOSS OF INSTRUMENT AND CONTROL BUS A.

[17] BROWNS FERRY 1 DOCKET 50-259 LER 90-020 REV 01
UPDATE ON ENGINEERED SAFETY FEATURE ACTUATION DURING RELAY TESTING CAUSED BY PROCEDURE INADEQUACY AND PERSONNEL ERROR.
EVENT DATE: 121490 REPORT DATE: 013191 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 220956) ON DECEMBER 14, 1990 AT 0400 HOURS, 480V SHUTDOWN BOARD 1B WAS DEENERGIZED WHEN THE NORMAL FEEDER BREAKER TO THE BOARD WAS TRIPPED DURING A TIME DELAY RELAY TEST. THE DEENERGIZATION OF THE SHUTDOWN BOARD IN TURN DEENERGIZED REACTOR PROTECTION SYSTEM BUS 1B AND THE PRIMARY CONTAINMENT ISOLATION SYSTEM LOGIC RELAYS POWERED BY THE BUS, RESULTING IN THE ISOLATION OF GROUP 2 VALVES. THE COMPLETION OF GROUP 2 ISOLATION LOGIC IS CONSIDERED A PLANT ENGINEERED SAFETY FEATURE. THIS EVENT OCCURRED WHEN TEST EQUIPMENT (SYNCHRONOUS TIMER) WAS PLACED ACROSS THE SHUTDOWN BOARD'S BREAKER CONTROL TRANSFER SWITCH. THIS CAUSED THE NORMALLY CLOSED FEEDER BREAKER TO TRIP. THE ROOT CAUSE OF THIS EVENT IS PROCEDURE INADEQUACY AND PERSONNEL ERROR. THE TIME DELAY RELAY TESTING PROCEDURE DID NOT PROVIDE ANY DIRECTION ON A PROPER DC POWER SOURCE TO BE USED FOR THE SYNCHRONOUS TIMER. MAINTENANCE PERSONNEL INVOLVED IN THE TEST DID NOT APPLY A SUFFICIENT DEGREE OF ATTENTION IN THOROUGHLY DETERMINING THE IDENTIFICATION OF THE DC SOURCE. ELECTRICAL MAINTENANCE PERSONNEL WILL REVIEW THIS LICENSEE EVENT REPORT. PERSONNEL CORRECTIVE ACTION IN ACCORDANCE WITH TVA POLICY HAS BEEN GIVEN TO THE INDIVIDUALS INVOLVED IN THIS EVENT. THE APPLICABLE ELECTRICAL MAINTENANCE PROCEDURE WILL BE REVISED TO REQUIRE A PORTABLE DC SUPPLY FOR TIME DELAY RELAY SETPOINT CHECKS.

[18] BROWNS FERRY 1 DOCKET 50-259 LER 91-001
DIESEL GENERATOR VOLTAGE RELAYS FAIL TO STAY ENERGIZED.
EVENT DATE: 010291 REPORT DATE: 020891 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

VENDOR: GENERAL ELECTRIC CO.

(NSIC 220977) ON DECEMBER 27, 1990, DURING THE PERFORMANCE OF DIESEL GENERATOR "A" LOAD ACCEPTANCE TEST THE SHUTDOWN BOARD "A" DIESEL GENERATOR VOLTAGE AVAILABLE RELAYS (DGVA) FAILED TO STAY ENERGIZED. ON DECEMBER 31, 1990, DURING THE PERFORMANCE OF DIESEL GENERATOR "D" LOAD ACCEPTANCE TEST, THE SHUTDOWN BOARD "D" DGVA RELAYS FAILED TO ENERGIZE. INVESTIGATIONS SHOWED THE BREAKER STATIONARY AUXILIARY SWITCH CONTACTS FAILED TO CLOSE WHEN THE DIESEL GENERATOR BREAKER CLOSED. THE COMBINATION OF THESE TWO EVENTS COULD HAVE RESULTED IN A FAILURE OF BOTH CORE SPRAY PUMPS TO START, RESULTING IN THE POTENTIAL LOSS OF BOTH LOOPS OF A CRITICAL SAFETY FUNCTION. THE ROOT CAUSE OF THIS EVENT WAS AN UNEXPECTED FAILURE AFFECTING THE RELIABILITY OF THE EQUIPMENT. CORRECTIVE ACTIONS INCLUDE THE ADDITION OF PARALLEL REDUNDANT CONTACTS ON THE UNIT 1 AND UNIT 2 SHUTDOWN BOARDS AND AN EVALUATION OF THE UNIT 3 SHUTDOWN BOARDS. IN ADDITION, PREVENTATIVE MAINTENANCE PROCEDURES WILL BE MODIFIED TO INCLUDE A VISUAL INSPECTION OF THESE SWITCHES.

[19] BROWNS FERRY 3 DOCKET 50-296 LER 90-005
WHEN THE REACTOR BUILDING VENT EXHAUST MONITOR WAS REMOVED FROM SERVICE, A COMPENSATORY SAMPLE WAS ISOLATED, THEREBY CAUSING A TECHNICAL SPECIFICATION MONITORING REQUIREMENT TO BE EXCEEDED.
EVENT DATE: 123190 REPORT DATE: 012991 NSSS: GE TYPE: BWR

(NSIC 220925) ON DECEMBER 31, 1990 AT 0530 HOURS, TVA DISCOVERED THAT DURING THE REACTOR BUILDING VENT EXHAUST MONITOR SOURCE CALIBRATION AND FUNCTIONAL TEST, MONITOR 3-RM-90-250 (MON) WAS VALVED OUT SEVERAL TIMES FOR PERIODS OF LESS THAN THREE HOURS. THE AGGREGATE TIME THAT THE MONITOR WAS OUT OF SERVICE WAS ESTIMATED TO BE 12 HOURS. THE CLOSURE OF THESE VALVES ISOLATED THE TEMPORARY CONTINUOUS MONITORING SYSTEM THAT HAD BEEN INSTALLED TO SATISFY PLANT TECHNICAL SPECIFICATION REQUIREMENTS. THE CAUSE OF THE EVENT WAS AN INADEQUATE DESIGN REVIEW. WITH THE DESIGN AS IT WAS, THE CALIBRATION SURVEILLANCE INSTRUCTION (SI) AND A CHEMISTRY SAMPLE PROCEDURE COULD NOT BE PERFORMED SIMULTANEOUSLY AS REQUIRED. A CONTRIBUTING CAUSE WAS A MISINTERPRETATION OF THE TECHNICAL SPECIFICATION REQUIREMENTS. THE IMMEDIATE CORRECTIVE ACTION WAS TO ISOLATE THE AFFECTED VENTS, THEREBY INVALIDATING THE NEED FOR A COMPENSATORY SAMPLE AND THE SI WAS COMPLETED. ADDITIONAL CORRECTIVE ACTIONS TAKEN TO PREVENT RECURRENCE WERE: (1) UPSTREAM CHEMISTRY SAMPLING POINTS WERE INSTALLED AND (2) RESPONSIBLE ORGANIZATION WILL REVISE CALIBRATION AND FUNCTIONAL TEST SIS FOR THE 10 EFFLUENT CAMS TO CLARIFY SAMPLING REQUIREMENTS WHEN MONITOR IS INOPERABLE PRIOR TO UNIT 2 RESTART.

[20] BRUNSWICK 1 DOCKET 50-325 LER 90-026 REV 01
UPDATE ON SECONDARY CONTAINMENT MANUAL ISOLATION DURING A DRYWELL FIRE IN A TEMPORARY CABLE TRAY PENETRATING THROUGH THE PERSONNEL AIRLOCK.
EVENT DATE: 120390 REPORT DATE: 021291 NSSS: GE TYPE: BWR

(NSIC 220987) ON 12/3/90, UNIT 1 WAS IN A SCHEDULED OUTAGE. REACTOR WAS DRAINED DOWN FOR RECIRCULATION PIPING REPLACEMENT. DRYWELL "PERSONNEL AIRLOCK" WAS BEING USED AS A TEMPORARY PENETRATION POINT. VENTILATION DUCT AND TRAY CONSTRUCTED OF FIRE RETARDANT PLAYWOOD AND SCAFFOLDING WERE ROUTED THROUGH THE AIRLOCK. THE TRAY ORGANIZED AND SUPPORTED TEMPORARY ELECTRICAL CABLES, HYDRAULIC LINES, AIR HOSES AND LEADS. EACH ITEM WAS INDIVIDUALLY WRAPPED IN A THIN PLASTIC. HEAT UP FOR POST WELD HEAT TREATMENT (PWHT) OF RECIRCULATION NOZZLE "G" WAS IN PROGRESS. AT APPROX. 0400, INDICATIONS OF SMOKE WERE NOTED. BETWEEN 0415 AND 0419 AN ELECTRICAL FIRE IN THE PWHT CABLES ON THE TEMPORARY TRAY WAS DISCOVERED. CONTROL ROOM WAS NOTIFIED, FIRE BRIGADE WAS DISPATCHED, NON-ESSENTIAL PERSONNEL IN REACTOR BLDG. (RB) WERE EVACUATED AT 0420. AN UNUSUAL EVENT (UE) WAS DECLARED AT 0429. RB VENTILATION SYSTEM WAS MANUALLY ISOLATED TO PREVENT VENTILATING THE FIRE. FIRE FIGHTING EFFORTS WERE FROM OUTSIDE OF THE PERSONNEL AIRLOCK WITH DRY CHEMICAL EXTINGUISHERS AND THEN WITH WATER AFTER THE TEMPORARY POWER WAS SECURED. 0552 UE WAS TERMINATED. CAUSE OF THE FIRE WAS UNDER SIZING OF THE PWHT CABLES FOR THE GIVEN APPLICATION AND CONFIGURATION.

[21] BRUNSWICK 1 DOCKET 50-325 LER 90-028
UNEXPECTED AUTOMATIC CLOSURE OF THE HPCI TURBINE EXHAUST VACUUM BREAKER ISOLATION VALVE WHEN THE ALTERNATE SAFE SHUTDOWN SUPPLY BREAKER WAS OPENED.
EVENT DATE: 121590 REPORT DATE: 011091 NSSS: GE TYPE: BWR

(NSIC 220708) ON 12-15-90, AT APPROXIMATELY 0530, THE HIGH PRESSURE COOLANT INJECTION SYSTEM TURBINE EXHAUST VACUUM BREAKER, 1-E41-F079, STROKED CLOSED WHEN ITS ALTERNATE SAFE SHUTDOWN POWER SUPPLY BREAKER WAS OPENED TO RESTORE IT TO ITS REQUIRED NORMAL SUPPLY BREAKER. WHEN NO REASON WAS FOUND FOR THE ALTERNATE BREAKER BEING ON, AN AUXILIARY OPERATOR WAS DIRECTED TO OPEN THE BREAKER. COINCIDENT WITH THE OPENING OF THE BREAKER THE VALVE STROKED CLOSED. THIS EVENT IS BELIEVED TO HAVE RESULTED FROM A SEALED IN CLOSURE SIGNAL IN THE VALVE'S NORMAL SUPPLY BREAKER LOGIC. WITH THE ALTERNATE BREAKER SUPPLYING POWER TO THE VALVE MOTOR, NO AUTOMATIC CLOSURE SIGNAL OR MANUAL OPERATION OF THE VALVE FROM THE CONTROL ROOM OR LOCAL OPERATION FROM THE NORMAL SUPPLY BREAKER IS AVAILABLE, PER DESIGN. THE EVENT IS STILL BEING INVESTIGATED AND A SUPPLEMENT WILL BE SUBMITTED BY APRIL 15, 1991. THE VALVE WAS RESTORED TO THE OPEN POSITION. CORRECTIVE ACTIONS WELL BE DETERMINED. THIS IS AN ISOLATED EVENT WITH NO SAFETY SIGNIFICANCE AS THE UNIT WAS SHUTDOWN AND PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) OPERABILITY WAS NOT REQUIRED. IF PCIS OPERABILITY HAD BEEN REQUIRED THE REDUNDANT ISOLATION VALVE AND SECONDARY CONTAINMENT WOULD HAVE BEEN AVAILABLE TO MITIGATE THE EVENT.

[22] BRUNSWICK 1 DOCKET 50-325 LER 90-029 REV 01
UPDATE ON CBEAF SYSTEM ACTUATION RESULTING FROM THE FAILURE OF THE 1-D22A-K2 RELAY COIL.
EVENT DATE: 121690 REPORT DATE: 021591 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BRUNSWICK 2 (BWR)
VENDOR: GENERAL ELECTRIC CO.

(NSIC 220990) ON 1/16/90, CONTROL BLDG. VENTILATION SYSTEM WAS OPERATING IN ITS NORMAL MODE. AT 0623, THE ALARM "CONTROL BLDG. HIGH RADIATION" (UA-3 6-7) ANNUNCIATED AND THE CONTROL BLDG. EMERGENCY AIR FILTRATION (CBEAF) SYSTEM AUTOMATICALLY INITIATED. AREA RADIATION MONITORS (ARM) WERE CHECKED AND TRIP INDICATIONS WERE NOT FOUND, THE FAINT SMELL OF OVER HEATED ELECTRONICS WAS DETECTED IN UNIT 1 ARM CABINET LOCATED IN PANEL H12-0600 OF THE CONTROL ROOM. CAUSE OF THE ALARM WAS DETERMINED TO BE A FAILED RELAY COIL. THE RELAY IS A NORMALLY ENERGIZED, GENERAL ELECTRIC MODEL CR-120, 115 VAC RELAY AND ITS COIL FAILURE RESULTED IN THE RELAY OPENING, CAUSING THE ALARM AND AUTOMATIC ACTUATION OF THE CBEAF SYSTEM. RELAY FAILURE WAS DETERMINED TO BE A NORMAL END OF LIFE FAILURE DUE TO AGING. RELAY COILS ARE ELECTRICALLY INSULATED BY A THIN EPOXY COATING, ON A CONTINUOUSLY ENERGIZED COIL THE EPOXY COATING CAN CRACK RESULTING IN A SHORT CIRCUIT ACROSS THE COIL AND THE OBSERVED "BURNED UP" COIL. THIS IS TERMED AN END OF COIL "LOFE" FAILURE. ONLY 5 FAILURES OF THIS TYPE RELAY HAVE BEEN EXPERIENCED AT BRUNSWICK IN THE PAST 25 MONTHS, OF THE 5 ONLY 2 ARE DIRECTLY ATTRIBUTABLE TO A COIL FAILURE. THE RELAY COIL HAS BEEN REPLACED. CONTROL BLDG. VENTILATION SYSTEM WAS RESTORED TO NORMAL OPERATION. AN INVESTIGATION UTILIZING THE REPETITIVE FAILURE PROGRAM MAY BE REQUIRED TO AVOID RECURRENCE OF THIS SITUATIO

[23] BRUNSWICK 2 DOCKET 50-324 LER 90-019
RWCU ISOLATION DURING HPCI MAINTENANCE SURVEILLANCE TESTING DUE TO SPURIOUS ACTUATION OF STEAM LEAK DETECTION INSTRUMENTATION.
EVENT DATE: 122690 REPORT DATE: 012491 NSSS: GE TYPE: BWR

(NSIC 220870) AT 0836 ON 12/26/90 AN "A" LOGIC GROUP 3 ISOLATION SIGNAL CAUSED THE REACTOR WATER CLEANUP (RWCU) INLET INBOARD ISOLATION VALVE (2-G31-F001) TO AUTOMATICALLY CLOSE. THIS OCCURRED WHEN SCHEDULED MAINTENANCE ON THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM CREATED AN ELECTRICAL DISTURBANCE WHICH TRIPPED THE RILEY SCAM MODULE (TEMPERATURE SWITCHES) OF THE RWCU STEAM LEAK DETECTION SYSTEM (SLDS) AND RESULTED IN A SPURIOUS ISOLATION SIGNAL. THE CONTROL OPERATOR BECAME AWARE OF THIS THROUGH APPROPRIATE CONTROL ROOM INDICATION AND ANNUNCIATION. THE ISOLATION VALVE WAS REOPENED AT 0858 AND THE RWCU SYSTEM WAS RETURNED TO SERVICE. THE SUSCEPTIBILITY OF THESE SWITCHES TO SPURIOUS TRIPS HAS

BEEN DOCUMENTED IN GENERAL ELECTRIC SERVICE INFORMATION LETTERS (SILS 416 AND 443) AND IE INFORMATION NOTICE 86-69. CP&LS ORIGINAL PLANS WERE TO INSTALL TIME-DELAY RELAYS IN THE SLDS CIRCUITRY TO REDUCE THE PROBABILITY OF SPURIOUS TRIPS; HOWEVER, IT HAS SINCE BEEN DETERMINED THIS WAS ONLY A PARTIAL REMEDY FOR BROADER PROBLEMS WITH THE CURRENT SLDS. CURRENT PLANS INCLUDE REPLACING THE EXISTING RILEY SCAM TEMPERATURE SWITCH UNITS WITH MORE MODERN NUMAC HARDWARE. DURING THIS EVENT THE REACTOR WAS AT 100% POWER. PRIOR EVENTS INCLUDE: LERS 2-83-58, 2-84-17, 1-86-008, 1-86-014, 2-87-007, 2-87-009, 1-88-002, 2-88-020, 1-89-004 AND 1-89-005.

[24]   BRUNSWICK 2   DOCKET 50-324   LER 90-020
HPCI SYSTEM MINIMUM FLOW VALVE INOPERABLE BECAUSE OF A BLOWN CONTROL POWER FUSE WHEN MOTOR CONTROL CENTER COMPARTMENT POSITION INDICATING LIGHT BULB SHORTED.
EVENT DATE: 122690   REPORT DATE: 012491   NSSS: GE   TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 220725) ON DECEMBER 26, 1990, THE UNIT 2 REACTOR WAS AT 100% POWER; THE EMERGENCY CORE COOLING SYSTEMS WERE OPERABLE. THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM WAS REMOVED FROM SERVICE FOR SURVEILLANCES AND MAINTENANCE. WHILE RETURNING THE SYSTEM TO SERVICE, THE MINIMUM FLOW VALVE CONTROL POWER FUSE WAS DETERMINED TO BE BLOWN. THIS WOULD PREVENT THE VALVE FROM OPENING AUTOMATICALLY. THE HPCI SYSTEM WAS PREVENTED FROM AUTOMATICALLY INITIATING AND THE FUSE WAS REPLACED. THE FUSE BLEW BECAUSE OF A SHORT WITHIN THE VALVES MOTOR CONTROL CENTER COMPARTMENT POSITION INDICATING BULB WHEN THE FILAMENT TWISTED WITHIN THE GLASS WHILE THE BULB BASE REMAINED STATIONARY. THE FILAMENT EVENTUALLY CROSSED OVER ITSELF CAUSING THE SHORT. AN INTERVIEW WITH THE CO DETERMINED THAT VALVE INDICATION WAS PRESENT ON THE CONTROL BOARD PRIOR TO THE HANGING OF THE CLEARANCE VERIFYING THAT THE CONTROL POWER FUSE WAS NOT BLOWN PRIOR TO THIS EVENT. THE FUSE WAS REPLACED AND THE SYSTEM WAS RETURNED TO SERVICE. THE SYSTEM WAS OUT OF SERVICE FOR 13 HOURS AND 33 MINUTES, APPROXIMATELY 4 HOURS WERE AS A RESULT OF THIS EVENT. THE EVENT WILL BE REVIEWED WITH APPROPRIATE PERSONNEL, EXAMPLES OF BULBS WHICH HAVE SEPARATED FROM THEIR BASE WILL BE PROVIDED TO HELP IN THE IDENTIFICATION OF POTENTIAL PROBLEMS.

[25]   BYRON 1   DOCKET 50-454   LER 90-007 REV 01
UPDATE ON MAIN STEAMLINE ISOLATION SYSTEM INOPERABLE DUE TO FAILURE TO TEST MANUAL INITIATION HANDSWITCH.
EVENT DATE: 061290   REPORT DATE: 022091   NSSS: WE   TYPE: PWR
OTHER UNITS INVOLVED: BYRON 2 (PWR)

(NSIC 221041) ON 06/12/90, DURING A REVIEW OF THE BYRON UNIT 1 MAIN STEAM ISOLATION VALVE (MSIV) FULL STROKE TEST SURVEILLANCES, IT WAS FOUND THAT THE STEAM LINE ISOLATION HANDSWITCH ON MAIN CONTROL BOARD PANEL 1PM06J HAD NOT BEEN TESTED DURING THE PAST REFUELING OUTAGES FOR BOTH UNITS. TECHNICAL SPECIFICATION 3.3.2, TABLE 4.3-2 ITEM 4.A.2, REQUIRES TESTING OF THE STEAM LINE ISOLATION HANDSWITCH ON A REFUELING OUTAGE INTERVAL AND REQUIRES THAT TWO TRAINS BE OPERABLE IN MODES 1, 2 AND 3. AT 1100 ON 6-12-90, THE MSIV MANUAL ISOLATION SYSTEM WAS DECLARED INOPERABLE FOR BOTH UNITS. DUE TO AN UNRELATED PROBLEM, UNIT 1 WAS SHUTDOWN (MODE 4) ON 06/13/90. THEREFORE , IT WAS DECIDED TO PERFORM THE TEST ON UNIT 1 FIRST. THE HANDSWITCHES WERE TESTED AND THE SYSTEM WAS DECLARED OPERABLE AT 1545. ON 06/13/90, AT 2345, WITH UNIT 2 OPERATING IN MODE 1, THE UNIT ENTERED TECHNICAL SPECIFICATION 3.0.3 BECAUSE BOTH TRAINS OF MANUAL ISOLATION WERE INOPERABLE DURING THE TEST. ON 06/14/90 AT 0001, THE TEST WAS COMPLETED AND TECHNICAL SPECIFICATION 3.0.3 WAS EXITED. THE SYSTEM WAS DECLARED OPERABLE AT 0730. AS PREVENTATIVE ACTION, BYRON PLANS TO HAVE A REVIEW PERFORMED ON ALL SIMILAR EQUIPMENT TO ENSURE THAT PROPER TESTING IS BEING DONE. IN ADDITION, THE STATION WILL REVIEW RELATED PROCEDURES AND THE PROCEDURE REVIEW PROCESS.

[26]   BYRON 2   DOCKET 50-455   LER 90-010
MANUAL REACTOR TRIP AND MAIN STEAM ISOLATION DUE TO SAMPLE PROBE WELD FAILURE.
EVENT DATE: 122090   REPORT DATE: 011791   NSSS: WE   TYPE: PWR
OTHER UNITS INVOLVED: BYRON 1 (PWR)

(NSIC 220832) ON DECEMBER 20, 1990 AT 0400, A SEVERE STEAM LEAK WAS REPORTED IN THE UNIT 2 MAIN STEAM TUNNEL. AFTER VERIFYING THE SIZE OF THE LEAK, THE REACTOR WAS MANUALLY TRIPPED. BY ELIMINATING STEAM GENERATOR BLOWDOWN AND FEEDWATER AS CAUSES, IT WAS DETERMINED THAT THE LEAK WAS ON THE MAIN STEAM SIDE. THE MAIN STEAM ISOLATION VALVES WERE THEN CLOSED WHICH ISOLATED THE LEAK. THE MAIN STEAM DUMPS WERE OPENED TO DEPRESSURIZE THE MAIN STEAM HEADER. UPON ENTRY INTO THE MAIN STEAM TUNNEL, THE 2C MAIN STEAM SAMPLE PROBE WAS FOUND LYING ON THE FLOOR. THE WELD FOR THE PROBE HAD BEEN IMPROPERLY REPAIRED DURING THE PREVIOUS REFUELING OUTAGE CAUSING THE PROBE AND ITS ISOLATION VALVE TO BE EJECTED LEAVING A ONE INCH HOLE IN THE MAIN STEAM LINE. SINCE THIS PROBE WAS NEEDED ONLY FOR INITIAL START-UP TESTING, THE NOZZLE WAS CAPPED. THIS EVENT IS REPORTABLE PURSUENT TO 10CFR50.73(A)(2)(IV) ANY EVENT THAT RESULTS IN A MANUAL OR AUTOMATIC ACTUATION OF THE ENGINEERING SAFETY FEATURES INCLUDING REACTOR PROTECTION SYSTEM.

[27] CALLAWAY 1 DOCKET 50-483 LER 90-017
A REACTOR TRIP DUE TO A FAILURE OF A CONTROLLER/DRIVER CARD FOR THE 'B' FEEDWATER REGULATING VALVE.
EVENT DATE: 123090 REPORT DATE: 012591 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELEC CORP.-NUCLEAR ENERGY SYS

(NSIC 220729) ON 12/30/90, AT 1152 CST, A REACTOR TRIP OCCURRED DUE TO THE FAILURE OF A CONTROLLER/DRIVER CARD FOR THE 'B' FEEDWATER REGULATING VALVE (FRV). THE FRV FAILED CLOSED AND COULD NOT BE RE-OPENED BY THE LICENSED OPERATORS FROM THE MAIN CONTROL BOARD IN EITHER THE AUTOMATIC OR MANUAL MODE. THE SUBSEQUENT 'B' STEAM GENERATOR LOW WATER LEVEL ACTUATED THE REACTOR TRIP SIGNAL. AS A RESULT OF THE RPS ACTUATION, A FEEDWATER ISOLATION (FWIS) AND AN AUXILIARY FEEDWATER ACTUATION (AFAS) WERE GENERATED BY DESIGN. THE PLANT WAS IN MODE 1 - POWER OPERATIONS AT 100 PERCENT REACTOR POWER. THE REACTOR COOLANT SYSTEM TEMPERATURE WAS 588 DEGREES F AND THE PRESSURE WAS 2235 PSIG. THE LICENSED OPERATORS RECOVERED FROM THE TRIP AND ENGINEERED SAFETY FEATURE (ESF) ACTUATIONS VIA PLANT PROCEDURES. THE FRV CONTROLLER/DRIVER CARD WAS REPLACED AT 1330 ON 12/30/90. THE MANUFACTURER OF THE CARD IS WESTINGHOUSE (COMPONENT #2837A16G03). THE PLANT WAS RETURNED TO MODE 1 - POWER OPERATIONS AT 0134 ON 12/31/90. A CAPACITOR IN THE POWER SUPPLY SECTION OF THE CARD WAS THE CAUSE OF THE CARD FAILURE. THE FAILURE OF THE CAPACITOR IS INDETERMINATE. CORRECTIVE ACTIONS INCLUDE THE FAILURE HISTORY AND OTHER APPLICATIONS OF THE USE OF THIS CAPACITOR WILL BE EVALUATED; AND AN EVALUATION WILL BE PERFORMED TO DETERMINE THE FEASIBILITY OF ADDING A REDUNDANT CONTROLLER/DRIVER CARD TO THE CIRCUIT FOR THE FRV'S.

[28] CALVERT CLIFFS 1 DOCKET 50-317 LER 88-015 REV 01
UPDATE ON MOVEMENT OF HEAVY LOADS OVER THE SPENT FUEL POOL.
EVENT DATE: 123088 REPORT DATE: 020491 NSSS: CE TYPE: PWR

(NSIC 220930) IN MAY, 1983, THE NRC ISSUED A SAFETY EVALUATION DOCUMENTING THE STAFF'S APPROVAL OF BALTIMORE GAS AND ELECTRIC'S (BG&ES) RESOLUTION OF GENERIC ISSUE A-36, CONTROL OF HEAVY LOADS NEAR SPENT FUEL. THIS RESOLUTION INCLUDED THE USE OF ADMINISTRATIVE CONTROLS TO PREVENT THE MOVEMENT OF HEAVY LOADS OVER THE SPENT FUEL POOL. ON DECEMBER 30, 1988, IT WAS DISCOVERED THAT ONE OF THESE ADMINISTRATIVE CONTROLS WAS NOT BEING PROPERLY MAINTAINED AND THAT A HEAVY LOAD (THE SPENT FUEL CASK CRANE LOAD BLOCK) HAD BEEN MOVED OVER THE SPENT FUEL POOL. IMMEDIATE CORRECTIVE ACTION WAS TAKEN TO PREVENT RECURRENCE BY NOT ALLOWING THE CRANE TO TRAVEL OVER SPENT FUEL. THE CAUSE OF THIS EVENT WAS PERSONNEL RESPONSIBLE FOR CONTROLLING CRANE MOVEMENT WERE UNAWARE THAT THE CRANE LOAD BLOCK, IN AND OF ITSELF, CONSTITUTED A HEAVY LOAD. PREVENTIVE MEASURES WERE TO REVISE THE CONTROLLING PROCEDURES TO ENSURE THE SAFE MOVEMENT OF HEAVY LOADS OVER THE SPENT FUEL POOL.

[29] CALVERT CLIFFS 1 DOCKET 50-317 LER 89-016 REV 01
UPDATE ON RESISTANCE TEMPERATURE DETECTORS NOT ENVIRONMENTALLY QUALIFIED DUE TO UNSEALED HOUSING.
EVENT DATE: 090889 REPORT DATE: 012591 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: CALVERT CLIFFS 2 (PWR)

(NSIC 220880) ON SEPTEMBER 8, 1989 IT WAS DETERMINED THAT THE RESISTANCE TEMPERATURE DETECTORS (RTDS) INSTALLED IN THE HOT AND COLD LEGS OF THE REACTOR COOLANT SYSTEM IN BOTH UNITS HAD NOT BEEN PROPERLY SEALED TO PREVENT MOISTURE INTRUSION. THIS CONDITION WAS ASSUMED TO INVALIDATE THE ENVIRONMENTAL QUALIFICATION (EQ) OF THE RTDS. SUBSEQUENT ANALYSIS HAS SHOWN THE RTDS WERE CAPABLE OF FUNCTIONING IN A POST-ACCIDENT ENVIRONMENT. THE ROOT CAUSE OF THIS EVENT WAS AN INADEQUATE PROCEDURE. THE EQ DESIGN MANUAL DID NOT SPECIFICALLY REQUIRE IDENTIFICATION AND REVIEW OF MECHANICAL INTERFACES SUCH AS THE NIPPLE TO BASE INTERFACE IN THE PRIMARY RTDS. WE HAVE IDENTIFIED, EVALUATED, AND OBTAINED A QUALIFIED THREAD SEALANT TO BE USED SPECIFICALLY IN APPLICATIONS WHERE A MECHANICAL INTERFACE MUST BE ENVIRONMENTALLY SEALED. FIELD ENGINEERING CHANGES HAVE BEEN COMPLETED FOR UNIT 1 TO PROPERLY SEAL THE INTERFACE WITH AN ENVIRONMENTALLY QUALIFIED SEALANT. A FACILITY CHANGE REQUEST HAS BEEN ISSUED TO REPLACE THE UNIT 2 RTDS. THE EQ DESIGN MANUAL WAS REVISED TO SPECIFICALLY REQUIRE IDENTIFICATION AND REVIEW OF MECHANICAL INTERFACES.

[30] CALVERT CLIFFS 1 DOCKET 50-317 LER 90-002 REV 01
UPDATE ON FUEL REPAIR ACTIVITIES PERFORMED ON MORE THAN ONE ASSEMBLY DURING THE SAME TIME PERIOD IN VIOLATION OF FUEL HANDLING INCIDENT SAFETY ANALYSIS.
EVENT DATE: 011690 REPORT DATE: 012491 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: CALVERT CLIFFS 2 (PWR)

(NSIC 220869) ON DECEMBER 11, 1989, WITH UNIT 1 IN MODE 5 AND UNIT 2 DEFUELED, WE DETERMINED THAT OUR PREVIOUS PRACTICE OF RECONSTITUTING MORE THAN ONE SPENT FUEL ASSEMBLY AT A TIME WAS NOT BOUNDED BY THE ASSUMPTIONS OF THE FUEL HANDLING INCIDENT SAFETY ANALYSIS, WHICH ASSUMES THAT ONLY ONE FUEL ASSEMBLY COULD BE DAMAGED IN THE EVENT OF A FUEL HANDLING INCIDENT. ON JANUARY 16, 1990 THIS PRACTICE WAS DETERMINED TO PLACE THE PLANT IN A CONDITION OUTSIDE ITS DESIGN BASIS AS DESCRIBED IN THE UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR). THE ROOT CAUSE OF THIS CONDITION WAS A DEFICIENT PROCEDURE WHICH DID NOT PREVENT MOVEMENT AND PLACEMENT OF FUEL ASSEMBLIES SUCH THAT MORE THAN ONE COULD BE DAMAGED IN THE EVENT OF A FUEL HANDLING INCIDENT. FUEL HANDLING PROCEDURE FH-48 HAS BEEN REVISED TO REQUIRE THAT ONLY ONE ASSEMBLY AT A TIME BE PLACED IN A DESIGNATED LOCATION FOR RECONSTITUTION. THIS CHANGE ENSURES THAT NO MORE THAN ONE ASSEMBLY COULD BE DAMAGED IN A FUEL HANDLING INCIDENT. ALL FUEL HANDLING PROCEDURES HAVE BEEN REVIEWED AGAINST THE UFSAR WITH NO DISCREPANCIES IDENTIFIED. FUTURE CHANGES TO FUEL HANDLING PROCEDURES WILL BE REVIEWED AGAINST UFSAR ACCIDENT ANALYSIS ASSUMPTIONS.

[31] CALVERT CLIFFS 1 DOCKET 50-317 LER 90-018 REV 01
UPDATE ON AXIAL SHAPE INDEX NOT CONTINUOUSLY MONITORED AS REQUIRED BY TECHNICAL SPECIFICATIONS DUE TO INCORRECT LABELING OF POWER RANGE DETECTOR CONNECTORS.
EVENT DATE: 052990 REPORT DATE: 013191 NSSS: CE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 220931) ON MAY 29, 1990, IT WAS DETERMINED THAT THE UPPER AND LOWER CABLE CONNECTIONS FOR THE UNIT-1 Y-CHANNEL EXCORE POWER RANGE DETECTORS WERE REVERSED, MAKING THE POWER RATIO CALCULATOR (PRC) INOPERABLE. THE PRC IS REQUIRED FOR CONTINUOUS AXIAL SHAPE INDEX MONITORING IN MODE 1 WHEN THE PLANT COMPUTER IS NOT AVAILABLE. THERE HAVE BEEN FOUR INSTANCES WHEN THE PLANT WAS IN MODE 1 AND THE PLANT COMPUTER WAS NOT AVAILABLE. THIS CONDITION IS NOT IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS. THE ROOT CAUSE OF THE EVENT WAS THE IMPROPER LABELING OF THE DETECTOR CONNECTIONS. A CONTRIBUTING FACTOR WAS THAT THE PRE-INSTALLATION TEST OF THE SYSTEM DID NOT DETECT THE PROBLEM. CORRECTIVE ACTIONS INCLUDE CORRECTION OF THE DETECTOR CONNECTION LABELING AND CONNECTING SYSTEM LEADS TO THE CORRECT DETECTOR CONNECTIONS. PRE-INSTALLATION TESTING OF THE EXCORE DETECTORS HAS BEEN REVIEWED AND WILL BE REVISED. DATA WILL BE COLLECTED FOR UNIT-2 X- AND Y-CHANNEL EXCORE DETECTORS DURING THE NEXT UNIT-2 STARTUP TO CONFIRM THAT THE PROBLEM DOES NOT APPLY TO UNIT-2. THE DESIGN SPECIFICATION FOR THE DETECTORS HAS BEEN IMPROVED TO ENSURE PROPER LABELING IN THE FUTURE.

[32] CALVERT CLIFFS 1 DOCKET 50-317 LER 90-030
FATIGUE CRACKING IN THE SAFETY INJECTION TANK NITROGEN VENT LINE WELDS DUE TO DESIGN ENGINEERING PERSONNEL ERROR.
EVENT DATE: 120490 REPORT DATE: 021491 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: CALVERT CLIFFS 2 (PWR)

(NSIC 220986) ON 12/4/90 WITH UNIT 1 IN MODE 5 AT 170 DEGREES AND 180 PSIA, THE NITROGEN VENT LINES ON TOP OF TANKS 11B AND 12B WERE INSPECTED AND CRACKS WERE FOUND IN THE STRUCTURAL WELD AT THE COUPLING-TO-VENT LINE JUNCTION. NO SIMILAR CRACKS IN THE OTHER UNIT 1 SITS (11A AND 12A) OR THE UNIT 2 SITS WERE FOUND. THE ABILITY OF THE CRACKED LINES TO WITHSTAND A SAFE SHUTDOWN EARTHQUAKE CONCURRENT WITH A LOSS OF COOLANT ACCIDENT COULD NOT BE DETERMINED. THIS PLACED THE PLANT OUTSIDE ITS DESIGN BASIS. THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR IN THAT DESIGN ENGINEERS FAILED TO RECOGNIZE A CANTILEVER CONDITION DURING REPLACEMENT OF THE SIT RELIEF VALVES. THE RELIEF VALVES ON ALL FOUR TANKS FOR BOTH UNITS WERE RELOCATED TO THE NITROGEN SUPPLY LINE AT THE BASE OF THE TANKS AND ADEQUATELY SUPPORTED TO PREVENT RECURRENCE OF THIS EVENT. A SPECIAL TRAINING SESSION WILL BE HELD WITH DESIGN ENGINEERING PERSONNEL TO REVIEW THE MAJOR ISSUES ASSOCIATED WITH THIS MODIFICATION AND TO REINFORCE THE PROGRAM REQUIREMENTS REGARDING PRE- AND POST-DESIGN WALKDOWNS. THIS EVENT WILL BE REVIEWED WITH ENGINEERING PERSONNEL IN FUTURE CODE TRAINING CLASSES.

[33] CALVERT CLIFFS 1 DOCKET 50-317 LER 90-027
UNDERSIZED FUSES COULD HAVE PREVENTED SAFETY-RELATED EQUIPMENT FROM PERFORMING ITS INTENDED SAFETY FUNCTION CAUSED BY A CALCULATIONAL ERROR.
EVENT DATE: 120690 REPORT DATE: 010791 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: CALVERT CLIFFS 2 (PWR)

(NSIC 220663) ON DECEMBER 6, 1990 A CONDITION WAS DISCOVERED AT CALVERT CLIFFS WHICH COULD HAVE PREVENTED VARIOUS SAFETY-RELATED (SR) MOTORS FROM PERFORMING THEIR INTENDED SAFETY FUNCTIONS. IF A DEGRADED VOLTAGE CONDITION EXISTED AT THE MOTOR CONTROL CENTER POWER SOURCE WHILE AN ENGINEERED SAFETY FEATURE ACTUATION SIGNAL (ESFAS) WAS ATTEMPTING TO START SR LOADS, FUSES FOR THE CONTROL CIRCUITRY OF SR MOTORS SUPPLIED BY THE MOTOR CONTROL CENTERS WOULD HAVE BLOWN BEFORE THE DEGRADED-VOLTAGE RELAYS COULD TIME OUT AND SWITCH THE POWER SOURCE TO THE EMERGENCY DIESEL GENERATORS (EDGS). AT THE TIME UNIT 1 WAS IN A COLD SHUTDOWN CONDITION AND UNIT 2 WAS IN A REFUELING CONDITION. THE CAUSE OF THE CONDITION WAS A CALCULATIONAL ERROR IN THE DESIGN PACKAGE THAT INSTALLED THE 4160-VOLT BUS POWER VOLTAGE PROTECTION RELAYS IN 1977. THE CALCULATION ERROR RESULTED IN ALLOWING SOME UNDERSIZED FUSES IN SR MOTOR CONTROL CIRCUITS TO REMAIN INSTALLED. ALL UNDERSIZED FUSES FOR BOTH UNITS HAVE BEEN REPLACED WITH PROPERLY RATED FUSES. FUSING SIZES FOR ALL ESFAS AND EDG LOADS HAVE BEEN VERIFIED AS CORRECT. THE CALCULATION HAS BEEN REVISED. APPROPRIATE DRAWINGS WILL BE REVISED. A PROBLEM REPORT HAS BEEN INITIATED TO ADDRESS ONE OF THE UNDERLYING CAUSES OF THE CALCULATIONAL ERROR.

[34] CATAWBA 1 DOCKET 50-413 LER 91-001 REV 01
UPDATE ON BOTH TRAINS OF RESIDUAL HEAT REMOVAL AND AUXILIARY CONTAINMENT SPRAY INOPERABLE DUE TO DEFECTIVE PROCEDURES AND INAPPROPRIATE ACTION.
EVENT DATE: 111290 REPORT DATE: 021391 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: CATAWBA 2 (PWR)

(NSIC 221048) IN OCTOBER 1990, PERFORMANCE BECAME AWARE OF A MCGUIRE NUCLEAR STATION INCIDENT THAT RENDERED BOTH TRAINS OF THE RESIDUAL HEAT REMOVAL (ND) SYSTEM INOPERABLE WH ILE STROKE TESTING VALVES NI136B, NS38S, AND NS43A. WITH ANY OF THESE VALVES OPE N DURING A LOSS OF COOLANT ACCIDENT, ND INJECTION FLOW COULD BE DEGRADED. THESE VALVES HAVE BEEN TESTED QUARTERLY AT CATAWBA SINCE UNIT STARTUP PERFORMANCE REVI SED TEST PROCEDURES FOR NS38B AND NS43A TO CLOSE DOWNSTREAM VALVES SO THAT QUART ERLY TESTING COULD CONTINUE. WHILE REVIEWING THE FINAL SAFETY ANALYSIS REPORT (F SAR), PERFORMANCE MISINTERPRETED INFORMATION RELATED TO THE AUXILIARY CONTAINMEN T SPRAY HEADERS AND ALLOWED THE TEST TO BE COMPLETED WITHOUT DECLARING ND/NS INO PERABLE. ON NOVEMBER 12, WITH UNIT 1 IN MODE 1, POWER OPERATION, LNS38B WAS TEST ED WITH TRAIN A ND INOPERABLE; THEREBY, RENDERING BOTH TRAINS OF AUXILIARY CONTA INMENT SPRAY INOPERABLE. THIS INCIDENT

IS ATTRITUTED TO DEFECTIVE PROCEDURES THA T ALLOWED QUARTERLY TESTING OF NI136B, NS38B, AND NS43A WITHOUT REGARD TO POSSIB LE DEGRADED ND INJECTION FLOW. THE AUXILIARY CONTAINMENT SPRAY HEADERS WERE INOP ERABLE DUE TO AN INAPPROPRIATE ACTION DUE TO MISINTERPRETED INFORMATION IN THE F SAR. CORRECTIVE ACTIONS INCLUDE RESCHEDULING TESTS FOR COLD SHUTDOWN AND FSAR EN HANCEMENTS.

[35] CATAWBA 1 DOCKET 50-413 LER 91-003
TECHNICAL SPECIFICATION VIOLATION FOR EMERGENCY PERSONNEL HATCH BEING UNLATCHED DUE TO EQUIPMENT FAILURE/MALFUNCTION.
EVENT DATE: 011091 REPORT DATE: 020691 NSSS: WE TYPE: PWR

(NSIC 220918) ON JANUARY 10, 1991, AT 1530 HOURS, WITH UNIT 1 IN MODE 3, HOT STANDBY, THE EMERGENCY PERSONNEL HATCH, ALSO KNOWN AS THE SUBMARINE HATCH, WAS FOUND IN THE DOWN OR CLOSED POSITION BUT NOT LATCHED. THE TAMPER SEAL WAS IN PLACE AROUND THE HANDWHEEL AND THE HINGE. THE AREA IN THE IMMEDIATE PROXIMITY OF THE HATCH WAS HIGHLY TRAVELED ONE WEEK PRIOR TO FINDING THE PROBLEM FOR WORK BEING CONDUCTED IN THE LOWER ICE CONDENSER. THE EXACT TIME THE HATCH BECAME UNLATCHED IS UNKNOWN; HOWEVER, THE OPPORTUNITY FOR THE HATCH TO BECOME UNLATCHED WAS VERY HIGH DURING ICE CONDENSER WORK ACTIVITIES. CONSTANT STEPPING ON AND OFF THE HATCH CAUSES THE LATCHING MECHANISMS TO RETRACT. THIS SITUATION WAS ALSO ENHANCED BY ONE OF THE FOUR LATCHING MECHANISMS BEING OUT OF ADJUSTMENT. TECHNICAL SPECIFICATION 3.6.5.5 REQUIRES THIS HATCH TO BE OPERABLE AND CLOSED IN MODE 1, POWER OPERATION, MODE 2, STARTUP, MODE 3, HOT STANDBY, AND MODE 4, HOT SHUTDOWN. THIS INCIDENT IS ATTRIBUTED TO EQUIPMENT FAILURE/MALFUNCTION. CORRECTIVE ACTIONS INCLUDE MODIFYING THE HATCH BY INSTALLING A TAB TO MATCH MARK THE CORRECT CLOSED POSITION OF THE HANDWHEEL AND TO PROVIDE A CLOSER AND MORE SECURE POSITION TO INSTALL THE TAMPER SEAL. THE LATCHING MECHANISMS WILL BE ADJUSTED SUCH THAT ALL MAKE SIMULTANEOUS CONTACT WITH THE LATCHING LIP.

[36] CATAWBA 2 DOCKET 50-414 LER 91-002
TECHNICAL SPECIFICATION VIOLATION DUE TO A VALVE BEING RETURNED TO SERVICE WITH AN EXPIRED SURVEILLANCE.
EVENT DATE: 010691 REPORT DATE: 013191 NSSS: WE TYPE: PWR

(NSIC 220920) ON DECEMBER 8, 1990, WITH UNIT 2 IN MODE 1, POWER OPERATION, 2NM-197B, S/G UPPER SHELL SAMPLE INSD CONT ISOL VLV, WOULD NOT OPEN DURING AUXILIARY SAFEGUARDS TESTING. THE VALVE WAS DEENERGIZED AND TAGGED CLOSED TO ENSURE CONTAINMENT INTEGRITY, AND A WORK REQUEST WAS GENERATED FOR ITS REPAIR. SINCE 2NM-197B WOULD NOT OPEN, THE REQUIRED STROKE TIME TEST WAS NOT COMPLETED BY THE LATE TEST DATE, DECEMBER 18, 1990. HOWEVER, A TECHNICAL SPECIFICATION OPERABILITY NOTIFICATION SHEET (TSONS) WAS NOT GENERATED. ON JANUARY 6, 1991, WITH UNIT 2 IN MODE 3, HOT STANDBY, WORK WAS COMPLETED ON 2NM-197B AND AT 1731 HOURS THE VALVE WAS DECLARED OPERABLE, EVEN THOUGH THE SURVEILLANCE HAD EXPIRED ON DECEMBER 18, RESULTING IN A VIOLATION OF TECHNICAL SPECIFICATIONS. ON JANUARY 7, AT APPROXIMATELY 0800 HOURS, WITH UNIT 2 IN MODE 3, A PERFORMANCE TEST SUPERVISOR NOTED WHILE REVIEWING THE SHIFT MANAGER'S LOGBOOK THAT THE REPAIR OF 2NM-197B WAS COMPLETE. PERFORMANCE PERSONNEL SATISFACTORILY STROKE TIME TESTED THE VALVE BY 0910 HOURS. THIS INCIDENT IS ATTRIBUTED TO AN INAPPROPRIATE ACTION, BECAUSE PERFORMANCE PERSONNEL DID NOT ENSURE THAT A TSONS WAS SUBMITTED TO DOCUMENT THE SURVEILLANCE DUE ON VALVE 2NM-197B. AS CORRECTIVE ACTIONS, THIS INCIDENT HAS BEEN DISCUSSED WITH PERFORMANCE PERSONNEL AND A PROGRAM HAS BEEN IMPLEMENTED TO ENSURE THAT A TSONS IS GENERATED WHEN A SURVEILLANCE IS DUE.

[37] CLINTON 1 DOCKET 50-461 LER 90-018
FAILURE TO CONSIDER ALL POTENTIAL CONTAINMENT ATMOSPHERE LEAKAGE PATHWAYS REQUIRING TESTING IN ACCORDANCE WITH 10CFR50, APPENDIX J.
EVENT DATE: 121890 REPORT DATE: 011791 NSSS: GE TYPE: BWR

(NSIC 220855) ON DECEMBER 18, 1990, DURING THE PLANT'S SECOND REFUELING OUTAGE, WITH THE PLANT IN MODE 5 (REFUELING), ILLINOIS POWER COMPANY DETERMINED THAT THE MECHANICAL JOINTS ON THE DISCHARGE SIDE OF 1E12-F055A AND 1E12-F055B, RESIDUAL HEAT REMOVAL (RHR) 'A' AND 'B' HEAT EXCHANGER (HX) RELIEF LINE VALVES, HAD NOT BEEN LOCAL LEAK RATE TESTED AS REQUIRED BY 10CFR50, APPENDIX J. INVESTIGATION

INTO THIS EVENT IDENTIFIED OTHER PENETRATIONS FOR WHICH NOT ALL POTENTIAL CONTAINMENT ATMOSPHERE LEAK PATHS IN ACCORDANCE WITH THE REQUIREMENTS OF 10CFR50, APPENDIX J WERE CONSIDERED; AND IDENTIFIED THREE PENETRATIONS WHICH DO NOT TERMINATE IN THE SUPPRESSION POOL AT OR BELOW THE DESIGN LEVEL REQUIRED TO MAINTAIN THE PENETRATION LINES WATER SEALED FOLLOWING A LOSS OF COOLANT ACCIDENT (LOCA). THE CAUSE OF THIS EVENT IS STILL UNDER INVESTIGATION. CORRECTIVE ACTIONS INCLUDE: A REVIEW OF ALL CONTAINMENT PENETRATIONS TO ENSURE THAT PROPER TEST CRITERIA ARE BEING IMPLEMENTED; IDENTIFICATION OF MECHANICAL JOINTS THAT COULD BE POTENTIAL CONTAINMENT ATMOSPHERE LEAKAGE PATHS; AND MODIFICATION OF THE IDENTIFIED PENETRATIONS TO FACILITATE CONFORMANCE WITH THE REQUIREMENTS OF 10CFR50, APPENDIX A, GENERAL DESIGN CRITERIA 55 AND 56, AND APPENDIX J.

[38] COMANCHE 1 DOCKET 50-445 LER 90-023
REACTOR TRIP ON LOSS OF FEEDWATER FLOW CAUSED BY A LOOSE CONTROL POWER FUSE.
EVENT DATE: 080890 REPORT DATE: 090590 NSSS: WE TYPE: PWR

(NSIC 220895) ON 8/8/90, COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1 WAS IN MODE 1, POWER OPERATIONS, WITH REACTOR POWER AT 17%. A LOOSE FUSE IN THE MAIN FEEDWATER CONTROL POWER CIRCUIT CAUSED CLOSURE OF A VALVE IN THE FEEDWATER FLOW PATH TO STEAM GENERATOR NUMBER 4. WATER LEVEL IN STEAM GENERATOR NUMBER 4 DECREASED TO THE LO-LO LEVEL SETPOINT, INITIATING A REACTOR TRIP SIGNAL. CORRECTIVE ACTIONS INCLUDED INSPECTION OF SIMILAR COMPONENTS IN OTHER APPLICATIONS, MAINTENANCE ON THE MALFUNCTIONING COMPONENT, AND PERSONNEL TRAINING.

[39] COMANCHE 1 DOCKET 50-445 LER 90-036 REV 01
UPDATE ON PERSONNEL ERROR RESULTING IN FAILURE TO COMPLY WITH TECHNICAL SPECIFICATION ACTION REQUIREMENTS.
EVENT DATE: 101590 REPORT DATE: 021491 NSSS: WE TYPE: PWR

(NSIC 220988) ON 10/15/90, COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1 WAS IN MODE 1, POWER OPERATION, WITH THE REACTOR AT 65% OF RATED THERMAL POWER. WHILE PERFORMING A ROUTINE SYSTEM/AREA WALKDOWN, THE SYSTEM ENGINEER FOR THE PRIMARY PLANT VENTILATION SYSTEM DISCOVERED THAT ONE OF TWO TRAIN "B" ENGINEERED SAFETY FEATURES (ESF) FILTRATION UNITS WAS OUT OF SERVICE. THE UNIT REMAINED OUT OF SERVICE UNTIL 10/23/90, WHEN CONTROL ROOM PERSONNEL AGAIN DISCOVERED THE CONDITION WHILE ATTEMPTING TO PERFORM REQUIRED SURVEILLANCE TESTING. IT WAS DETERMINED THAT THE ESF FILTRATION UNIT INOPERABILITY EXCEEDED THE TIME LIMIT PRESCRIBED BY THE ASSOCIATED TECH SPEC. THE CAUSE OF THE EVENT WAS DETERMINED TO BE PERSONNEL ERROR. CORRECTIVE ACTION INCLUDES EVENT REVIEW AND A SYSTEM DESIGN MODIFICATION.

[40] COMANCHE 1 DOCKET 50-445 LER 90-037
BLACKOUT SEQUENCER ACTUATION DUE TO PERSONNEL ERROR.
EVENT DATE: 110590 REPORT DATE: 120590 NSSS: WE TYPE: PWR

(NSIC 220896) AT APPROXIMATELY 0300 ON 11/5/90, ELECTRICAL MAINTENANCE PERSONNEL WERE CONDUCTING A TEST OF THE NUCLEAR SAFETY RELATED TRAIN B 6.9 KV SWITCHGEAR UNDERVOLTAGE (UV) RELAYS. AT 0443, WHILE ATTEMPTING TO RE-LAND A WIRE, THE ELECTRICIAN INADVERTENTLY MADE CONTACT WITH AN ENERGIZED POINT ON THE UV RELAY, RESULTING IN A UV ACTUATION. AS A RESULT OF THE UV ACTUATION, THE REACTOR OPERATOR (RO) OBSERVED THE TRANSFER OF TRAIN B 6.9 KV SWITCHGEAR TO THE ALTERNATE POWER SUPPLY, AND ACTUATION OF THE TRAIN B BLACKOUT SEQUENCER (BOS). AT 0454, THE RO RESET THE BOS, AND RESTORED ACTUATED COMPONENTS TO THEIR ORIGINAL CONFIGURATION. AT 0652 ON 11/5/90, WITH RESTORATION COMPLETE, THE RO RESTORED THE NORMAL POWER SUPPLY TO TRAIN B 6.9 KV SWITCHGEAR. THE ROOT CAUSE WAS DETERMINED TO BE PERSONNEL ERROR. A CONTRIBUTING FACTOR WAS DETERMINED TO BE THE UV RELAY TEST PROCEDURE NOT WRITTEN TO MINIMIZE THE RISK OF HUMAN ERROR. CORRECTIVE ACTIONS INCLUDE A MEMO TO ELECTRICAL MAINTENANCE PERSONNEL ADDRESSING THESE CONCERNS, AND A PROCEDURE CHANGE.

[41] COMANCHE 1 DOCKET 50-445 LER 90-038
GAS CHANNEL ALARM INITIATED A CONTAINMENT VENTILATION ISOLATION DUE TO STAGNANT AIR POCKETS IN CONTAINMENT.
EVENT DATE: 110590 REPORT DATE: 120590 NSSS: WE TYPE: PWR

(NSIC 220897) ON 11/5/90, COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1 WAS IN MODE 5, COLD SHUTDOWN, WITH A CONTAINMENT PURGE IN PROGRESS. THE CONTAINMENT AIR COOLING AND RECIRCULATION SYSTEM (CACRS) FANS WERE RUNNING, RECIRCULATING AIR INSIDE CONTAINMENT. APPROXIMATELY THREE HOURS AFTER SHIFTING THE CACRS FANS, A CONTAINMENT AIRBORNE CONTAMINATION HIGH RADIATION ALARM WAS RECEIVED WHICH INITIATED A CONTAINMENT VENTILATION ISOLATION. THE INCREASE IN NOBLE GAS ACTIVITY WAS ATTRIBUTED TO STAGNANT AIR POCKETS IN CONTAINMENT THAT WERE SUBSEQUENTLY MIXED WHEN THE CACRS FANS WERE SHIFTED, CAUSING AN INCREASE IN DETECTED AIRBORNE ACTIVITY. THE ROOT CAUSE WAS DETERMINED TO BE A PHENOMENON IN WHICH STAGNANT AIR WITH A HIGHER NOBLE GAS CONTENT EXISTED DURING A CONTAINMENT PURGE ALONG WITH HAVING THE HIGH ALARM SET POINT ON THE GASEOUS MONITOR SET TOO CONSERVATIVELY FOR THE EXISTING CONDITIONS. CORRECTIVE ACTIONS ARE TO REVISE CONTAINMENT PURGE PROCEDURES TO ENSURE A GOOD MIXING OF THE CONTAINMENT ATMOSPHERE PRIOR TO PURGING AND TO REVIEW THE GUIDELINES FOR RADIOACTIVE EFFLUENT RELEASES, TO ENSURE THAT SET POINTS ARE NOT SET SO CONSERVATIVELY THAT AUTOMATIC SAFETY FUNCTIONS ARE ACTUATED UNNECESSARILY.

[42] COMANCHE 1 DOCKET 50-445 LER 90-040
MISSED TECHNICAL SPECIFICATION SURVEILLANCE RESULTING FROM FAILURE TO FOLLOW PROCEDURAL REQUIREMENTS.
EVENT DATE: 111390 REPORT DATE: 121390 NSSS: WE TYPE: PWR

(NSIC 220909) ON 11/13/90, COMANCHE PEAK STEAM ELECTRIC STATION UNIT 1 WAS IN MODE 3, HOT STANDBY. WHILE UPDATING THE SURVEILLANCE SCHEDULING SYSTEM DATABASE, THE OPERATIONS SURVEILLANCE TEST COORDINATOR DISCOVERED THAT TWO OPERATIONS TEST PROCEDURES WERE NOT INCLUDED IN THE DATABASE. FURTHER INVESTIGATION REVEALED THAT THE TESTS HAD NOT BEEN PERFORMED AS REQUIRED TO DEMONSTRATE OPERABILITY OF THE MAIN STEAM DRIP POT ISOLATION VALVES. FAILURE TO PERFORM THE SURVEILLANCE CONSTITUTES A FAILURE TO MEET THE OPERABILITY REQUIREMENTS FOR A LIMITING CONDITION FOR OPERATION. THE CAUSE OF THE EVENT WAS DETERMINED TO BE FAILURE TO FOLLOW PROCEDURES. CORRECTIVE ACTIONS INCLUDED PROCEDURE REVISIONS AND DATABASE REVIEW.

[43] COMANCHE 1 DOCKET 50-445 LER 91-001
INADVERTENT ACTUATION OF CONTROL ROOM AIR CONDITIONING ENGINEERED SAFETY FEATURE CAUSED BY SENSITIVITY OF RADIATION MONITORING DEVICE TO OVERCURRENT CONDITIONS.
EVENT DATE: 010391 REPORT DATE: 020491 NSSS: WE TYPE: PWR

(NSIC 220852) ON 1/3/91, AT APPROXIMATELY 2348 CST, AN AUXILIARY OPERATOR WAS ATTEMPTING TO CHANGE A BURNED-OUT BULB ON THE LOCAL MICROPROCESSOR ASSOCIATED WITH ONE OF THE RADIATION MONITORS IN THE CONTROL ROOM AIR CONDITIONING AIR INTAKE. WHEN THE BULB WAS UNSCREWED, A SHORT PIECE OF THE BULB'S LOOP FILAMENT FELL ACROSS THE TWO TERMINAL POSTS INSIDE THE BULB. THE MOMENTARY CURRENT SURGE EXCEEDED THE CAPACITY OF THE POWER SUPPLY OUTPUT FUSE, RESULTING IN A LOSS OF POWER TO THE MONITOR. THE CONTROL ROOM AIR CONDITIONING SYSTEM AUTOMATICALLY REALIGNED TO THE EMERGENCY RECIRCULATION MODE. THE CAUSE OF THE EVENT WAS DETERMINED TO BE EQUIPMENT SENSITIVITY TO OVERCURRENT CONDITIONS. CORRECTIVE ACTIONS INCLUDED TRAINING AND ADMINISTRATIVE CONTROLS OVER BULB REPLACEMENT IN MONITORS WITH AUTOMATIC ESF FUNCTIONS.

[44] CONNECTICUT YANKEE DOCKET 50-213 LER 90-009
DESIGN DEFICIENCY IDENTIFIED IN CONTAINMENT AIR RECIRCULATION FAN DAMPER SOLENOID VALVES.
EVENT DATE: 071190 REPORT DATE: 080890 NSSS: WE TYPE: PWR
VENDOR: AUTOMATIC SWITCH COMPANY (ASCO)

(NSIC 220899) ON 7/11/90, AT 0800 HOURS, WITH THE PLANT IN COLD SHUTDOWN (MODE 5) AN EINGINEERING EVALUATION DETERMINED THAT A DESIGN DEFICIENCY EXISTED THAT COULD

DISABLE ALL FOUR CONTAINMENT AIR RECIRCULATION (CAR) FANS DURING A LOSS OF COOLANT ACCIDENT (LOCA). SPECIFICALLY, A FAILURE OF THE PRESSURE REGULATOR WHICH SUPPLIES AIR TO THE CAR FAN DAMPER SOLENOID VALVES COULD RESULT IN A DIFFERENTIAL PRESSURE AT THE VALVES THAT IS GREATER THAN DESIGN. THE RESULTING AIR PRESSURE COULD HAVE PREVENTED THE SOLENOID VALVES FROM VENTING AIR AND, AS A RESULT, THE CAR FAN DAMPERS MAY NOT HAVE ALIGNED TO THEIR ACCIDENT POSITION. THE ROOT CAUSE OF THIS EVENT WAS A DESIGN DEFICIENCY. CORRECTIVE ACTION CONSISTED OF INSTALLING QUALIFIED SOLENOID VALVES. THE EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(V)(C) SINCE THIS CONDITION ALONE COULD HAVE PREVENTED THE FULFILLMENT OF THE SAFETY FUNCTION OF A SYSTEM NEEDED TO CONTROL THE RELEASE OF RADIOACTIVE MATERIALS.

[45] CONNECTICUT YANKEE DOCKET 50-213 LER 90-032
LOAD REDUCTION DUE TO INOPERABILITY OF CONTAINMENT FAN COOLERS.
EVENT DATE: 122790 REPORT DATE: 012591 NSSS: WE TYPE: PWR

(NSIC 220876) ON 12/27/90, AT 1335 HOURS, WITH THE PLANT IN MODE 1 AT 100% POWER, ALL 4 CONTAINMENT AIR RECIRCULATION (CAR) FANS WERE DECLARED INOPERABLE DUE TO THE FOULING OF BOTH SERVICE WATER FILTERS THAT SUPPLY THE FAN COOLERS. THIS CONDITION WAS IDENTIFIED DURING AUXILIARY OPERATOR SURVEILLANCE OF FILTER DIFFERENTIAL PRESSURES. A LOAD REDUCTION WAS IMMEDIATELY INITIATED IN ACCORDANCE WITH TECH SPEC 3.0.3. THE ROOT CAUSE OF THE EVENT WAS EXCESSIVE SILT SUSPENSION IN THE CONNECTICUT RIVER. ONE OF THE FILTERS WAS CLEANED AND RETURNED TO SERVICE AND THE LOAD REDUCTION WAS TERMINATED AT 1447 HOURS AT APPROXIMATELY 70% POWER. CORRECTIVE ACTION CONSISTS OF PROVIDING OPERATORS WITH ADDITIONAL GUIDANCE FOR MONITORING INCREASING DIFFERENTIAL PRESSURE ACROSS THE FILTERS. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(I)(B) SINCE IT RESULTED IN A CONDITION PROHIBITED BY THE PLANT'S TECH SPECS.

[46] COOK 1 DOCKET 50-315 LER 90-016
FAILURE OF TWO PRESSURIZER SAFETY VALVES TO MEET TECHNICAL SPECIFICATION REQUIRED SURVEILLANCE TEST CRITERIA.
EVENT DATE: 122190 REPORT DATE: 012191 NSSS: WE TYPE: PWR
VENDOR: CROSBY VALVE & GAGE CO.

(NSIC 220724) ON DECEMBER 21, 1990 IT WAS DETERMINED THAT TWO OF THREE PRESSURIZER SAFETY VALVES SENT TO A TEST LABORATORY OFF SITE FOR TESTING REQUIRED BY TECHNICAL SPECIFICATION 4.4.3 HAD BEEN FOUND WITH LIFT SETTINGS OUTSIDE OF THE ACCEPTANCE CRITERIA. ACCEPTABLE SETTINGS ARE BETWEEN 2461 PSIG AND 2509 PSIG. VALVE 1-SV-45B LIFTED AT 2451 PSIG AND VALVE 1-SV-45C AT 2548 PSIG. AN EVALUATION OF THE TEST DATA REVEALED THAT THE VALVE WITH A LIFT SETTING OF 10 PSI BELOW ACCEPTANCE CRITERIA WOULD HAVE HAD NO IMPACT ON THE OPERABILITY OF AFFECTED COMPONENTS, AS IT WOULD HAVE LIFTED PREMATURELY. THE VALVE WITH A LIFT SETTING EXCEEDING THE ACCEPTABLE RANGE DID SO BY 39 PSI. CALCULATIONS PERFORMED INDICATE THAT IF THE REACTOR COOLANT SYSTEM PRESSURE HAD REACHED 2548 PSIG, NO DAMAGE WOULD HAVE BEEN INCURRED BY PIPE, FITTINGS, VALVES AND/OR THE PRESSURIZER. THE UFSAR WAS REVIEWED FOR IMPACT OF THE AS-FOUND SETPOINTS. THE HIGH SETPOINT WOULD HAVE BEEN OFFSET BY THE ADDITIONAL CAPACITY OF POWER OPERATED RELIEF VALVES AND FULL SAFETY VALVE CAPACITY IS NOT REQUIRED. ALSO REVIEWED WERE CRITERIA ASSOCIATED WITH DEPARTURE FROM NUCLEATE BOILING. NO ADVERSE CONSEQUENCES APPLIED TO SAFETY ANALYSES. AN INVESTIGATION INTO THE CAUSE IS BEING CONDUCTED BY THE VALVE MANUFACTURER.

[47] COOPER DOCKET 50-298 LER 91-001
UNPLANNED ACTUATION OF GROUP VI ISOLATION DURING SURVEILLANCE TESTING DUE TO PERSONNEL ERROR.
EVENT DATE: 010791 REPORT DATE: 020191 NSSS: GE TYPE: BWR

(NSIC 220926) ON JANUARY 7, 1991, AT 1:38 PM, WITH THE PLANT IN OPERATION AT FULL POWER, AN UNPLANNED ACTUATION OF THE GROUP VI ISOLATION LOGIC OCCURRED, RESULTING IN ISOLATION OF THE SECONDARY CONTAINMENT AND INITIATION OF THE STANDBY GAS TREATMENT (SGT) SYSTEM. THE ACTUATION OCCURRED DUE TO INCORRECT PLACEMENT OF A JUMPER DURING PERFORMANCE OF SURVEILLANCE PROCEDURE 6.3.7.5, REACTOR BUILDING VENTILATION RADIATION MONITOR SOURCE CHECK, WHICH WAS BEING PERFORMED BY I&C

TRAINEES, UNDER THE SUPERVISION OF QUALIFIED I&C TECHNICIANS. THE PROCEDURE WAS BEING CONDUCTED AS A REGULARLY SCHEDULED SURVEILLANCE TEST AND AS AN ON-THE-JOB TRAINING (OJT) EXERCISE. THE CAUSE OF THE UNPLANNED ACTUATION WAS THE FAILURE OF BOTH THE QUALIFIED TECHNICIAN, ACTING AS AN OJT INSTRUCTOR, AND THE TRAINEE TO REFER TO THE PROCEDURE STEP PRIOR TO INSTALLATION OF THE JUMPER. CORRECTIVE ACTION TAKEN INCLUDED TEST TERMINATION, JUMPER REMOVAL, RESET OF THE GROUP VI ISOLATION, AND RESTORATION OF REACTOR BUILDING VENTILATION TO NORMAL. THE PROCEDURE WAS REVIEWED AND DETERMINED TO BE SATISFACTORY FROM A CONTENT AND HUMAN FACTORS STANDPOINT. BOTH THE QUALIFIED I&C TECHNICIAN AND THE TRAINEE WERE COUNSELED. FURTHER CORRECTIVE ACTION TO BE TAKEN INCLUDES INCORPORATION OF THIS EVENT IN INDUSTRY EVENTS TRAINING FOR I&C TECHNICIANS.

[48]   CRYSTAL RIVER 3   DOCKET 50-302   LER 89-022 REV 01
UPDATE ON UNKNOWN CAUSE LEADS TO ERRONEOUS INDICATION OF LOSS OF MAIN FEEDWATER PUMPS AND RESULTS IN MANUAL ENGINEERED SAFETY FEATURE ACTUATION.
EVENT DATE: 061489   REPORT DATE: 012191   NSSS: BW   TYPE: PWR

(NSIC 220873) ON JUNE 14, 1989, CRYSTAL RIVER UNIT 3 WAS IN MODE 3 (HOT STANDBY) NEARING THE END OF A MAINTENANCE OUTAGE. ONE MAIN FEEDWATER PUMP (MFWP) WAS SUPPLYING FEEDWATER TO THE ONCE THROUGH STEAM GENERATORS (OTSG'S) WITH THE OTHER MFWP IN STANDBY (LATCHED). A REFUELING INTERVAL ENGINEERED SAFEGUARDS (ES) ACTUATION SURVEILLANCE WAS IN PROGRESS. AT 1910, DURING THE ES SURVEILLANCE, A CONTROL BOARD OPERATOR OBSERVED INDICATIONS THAT BOTH MFWP'S HAD TRIPPED. IN ACCORDANCE WITH PLANT PROCEDURES, HE MANUALLY ACTUATED EMERGENCY FEEDWATER AND COMPLIED WITH THOSE PLANT PROCEDURES. SHORTLY THEREAFTER IT WAS NOTED THAT THE MFWP'S HAD NOT ACTUALLY TRIPPED. THE STATUS OF THE MAIN FEEDWATER PUMPS WAS DETERMINED TO BE NORMAL AND EFW WAS RETURNED TO NORMAL STANDBY STATUS IN ACCORDANCE WITH PLANT PROCEDURES. THIS EVENT WAS A MANUAL OPERATOR RESPONSE TO AN INDICATED LOSS OF BOTH MFWP'S. THE LOSS OF INDICATION WAS CAUSED BY AN INTERRUPTION OF POWER TO THE SUPERVISORY CIRCUIT. THIS IS AN EXPECTED RESULT OF THE SURVEILLANCE TEST, BUT WAS NOT INCLUDED AS A NOTE OR CAUTION IN THE PROCEDURE. THE CAUTIONARY NOTE HAS BEEN ADDED. FURTHER INVESTIGATION WILL BE CONDUCTED TO DETERMINE THE CAUSE AND APPROPRIATE CORRECTIVE ACTIONS WILL BE DEVELOPED AND IMPLEMENTED FOR THE REMAINING ANOMALOUS RESPONSE IN THE FEEDWATER FLOW CIRCUIT.

[49]   CRYSTAL RIVER 3   DOCKET 50-302   LER 90-019
AN INCORRECT MOTOR INSTALLED ON A VALVE OPERATOR RESULTS IN A CONDITION OUTSIDE THE DESIGN BASIS.
EVENT DATE: 121490   REPORT DATE: 011591   NSSS: BW   TYPE: PWR

(NSIC 220811) ON DECEMBER 14, 1990, CRYSTAL RIVER UNIT 3 WAS IN MODE 5 (COLD SHUTDOWN) FOR REQUIRED MAINTENANCE. DURING THIS OUTAGE, THE PRESSURIZER SPRAY CONTROL VALVE (RCV-14), A NON SAFETY-RELATED, NON ENVIRONMENTALLY QUALIFIED (EA) VALVE, WAS BEING REPACKED. IN ORDER TO ACCOMPLISH THIS TASK, THE VALVE MOTOR OPERATOR WAS REMOVED. WHILE IT WAS REMOVED FROM THE VALVE, THE MOTOR WAS BEING REPLACE. IN THE COURSE OF REPLACEMENT, IT WAS DISCOVERED THAT THE INSTALLED MOTOR WAS NOT AS ORIGINALLY SPECIFIED FOR THIS VALVE. THE CAUSE OF THIS EVENT IS NOT SPECIFICALLY KNOWN. A REVIEW OF PLANT RECORDS REVEALED NO DOCUMENT THAT AUTHORIZED THE CHANGE. AN ENGINEERING CALCULATION WAS PERFORMED AND IT SHOWED THAT THE INSTALLED MOTOR WAS ACCEPTABLE. A PLANT MODIFICATION RECORD WAS GENERATED TO CHANGE THE DESIGN DOCUMENTS. THIS DEFICIENCY IS REPORTABLE UNDER 10 CFR 50.73 (A)(2)(II)(B). SEVERAL OTHER VALVES IN SIMILAR SERVICE IN THE PLANT WILL BE VISUALLY EXAMINED TO CONFIRM THAT THIS IS AN ISOLATED EVENT.

[50]   CRYSTAL RIVER 3   DOCKET 50-302   LER 90-020
LOSS OF AUXILIARY BUILDING VENTILATION CAUSED BY JARRING OF THE THERMAL SWITCH DURING SCAFFOLD ERECTION LEADS TO CONDITION PROHIBITED BY TECHNICAL SPECIFICATIONS.
EVENT DATE: 122790   REPORT DATE: 012391   NSSS: BW   TYPE: PWR

(NSIC 220721) ON DECEMBER 27, 1990, CRYSTAL RIVER UNIT 3 WAS IN MODE 1 (POWER OPERATIONS) AT 100% RATED THERMAL POWER. AT 1317 HOURS, ALL AUXILIARY BUILDING

VENTILATION WAS LOST WHEN THE OPERATING EXHAUST FANS TRIPPED AND THE REDUNDANT FANS WOULD NOT START. THE CAUSE WAS DETERMINED TO BE THE ACTUATION OF A TEMPERATURE SWITCH [VL,TS] IN THE EXHAUST DUCT. THE SWITCH IS INTENDED TO DETECT A FIRE IN THE CHARCOAL AND HIGH EFFICIENCY PARTICULATE (HEPA) FILTER [VL,FLT] UNITS. IF AN ELEVATED TEMPERATURE IS DETECTED, THE SWITCH TRIPS ALL OF THE AUXILIARY BUILDING FANS [VL,FAN] IN BOTH TRAINS. THE SWITCH WAS ACTUATED BY A MECHANICAL SHOCK, WHICH OCCURRED WHEN IT WAS INADVERTENTLY JARRED DURING THE ERECTION OF SCAFFOLDING. THE SCAFFOLD WAS BEING ERECTED TO CHANGE LIGHT BULBS IN THE AREA. THE SCAFFOLD ERECTION CREW WAS APPRISED OF THE SENSITIVITY OF THE SWITCH AND TOLD TO REPOSITION THE SCAFFOLD. THE SWITCH WAS RESET, THE FANS RESTARTED, AND THE VENTILATION SYSTEM RESTORED TO NORMAL AT 1344. THERE WERE NO EQUIPMENT FAILURES DURING THIS EVENT AND ALL SYSTEMS PERFORMED AS THEY WERE DESIGNED. FOR THIS REASON, THE EVENT HAD NO SAFETY CONSEQUENCES. THERE WAS NO RELEASE IN PROGRESS AT THE TIME OF THE EVENT. THIS IS AN ISOLATED EVENT AS THERE ARE NO PREVIOUS SIMILAR EVENTS AND NO OTHER CORRECTIVE ACTION IS PLANNED OR WARRANTED.

[51] DAVIS-BESSE 1 DOCKET 50-346 LER 90-015
FIRE PANEL C4720C SIX MONTH SURVEILLANCE TEST EXCEEDED LATE DATE.
EVENT DATE: 121290 REPORT DATE: 011191 NSSS: BW TYPE: PWR

(NSIC 220710) ON DECEMBER 12, 1990, AT 0700 HOURS, THE SHIFT SUPERVISOR RECOGNIZED THAT AT 0005 HOURS THE LATE DATE FOR TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENT (SR) 4.3.3.8.2 HAD BEEN EXCEEDED FOR LOCAL FIRE PANEL C4720C. THIS PANEL MONITORS THE CONTAINMENT FIRE DETECTION ZONES. THE SR DONE BY DB-MI-03806 IS A CHECK OF THE SUPERVISED CIRCUITS. THE TEST WAS RUN IN SEPTEMBER, BUT A DEFICIENCY (WHICH WAS EVALUATED AS NOT AFFECTING OPERABILITY) PREVENTED THE TEST FROM BEING SIGNED OFF AS COMPLETE. THE FAILURE TO RESOLVE THE DEFICIENCY BY THE TEST SPONSOR AND THE DESIGNATED REVIEWER CAUSED THE TEST TO EXCEED ITS LATE DATE. OVERSIGHT BY THE SHIFT SUPERVISOR IN NOT DECLARING THE PANEL INOPERABLE BEFORE THE LATE DATE WAS EXCEEDED CAUSED THIS INCIDENT TO BE REPORTABLE. THE MISSED SURVEILLANCE REQUIREMENT IS REPORTABLE AS AN LER UNDER 10CFR50.73(A)(2)(I)(B) AS A CONDITION PROHIBITED BY THE TECHNICAL SPECIFICATIONS. DB-MI-03806 WAS COMPLETED AND THE ACTION STATEMENT EXCITED AT 2142 HOURS ON DECEMBER 12, 1990. THE INCIDENT WILL BE REVIEWED WITH ALL SHIFT SUPERVISORS AS WELL AS THOSE PERSONNEL WHO HAVE RESPONSIBILITIES FOR PERFORMING TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS AND THOSE WHO HELP RESOLVE DEFICIENCIES.

[52] DAVIS-BESSE 1 DOCKET 50-346 LER 90-016
REACTOR TRIP DUE TO GROUP ROD DROP.
EVENT DATE: 121390 REPORT DATE: 011491 NSSS: BW TYPE: PWR

(NSIC 220711) ON DECEMBER 13, 1990 AT 0844 HOURS, THE PLANT EXPERIENCED A REACTOR COOLANT SYSTEM (RCS) LOW PRESSURE TRIP. A FUNCTIONAL TEST OF THE REACTOR PROTECTION SYSTEM (RPS) CHANNEL 1 REACTOR TRIP MODULE LOGIC AND REACTOR TRIP BREAKER B WAS IN PROGRESS WHEN THE TRIP OCCURRED. REACTOR TRIP BREAKER B HAD BEEN PREVIOUSLY TRIPPED AS PART OF THE RPS FUNCTIONAL TEST. SEVEN OF EIGHT CONTROL RODS IN ROD GROUP 7 DROPPED INTO THE CORE CAUSING REACTOR POWER TO INITIALLY DECREASE TO APPROXIMATELY 48 PERCENT. RCS PRESSURE AND TEMPERATURE DECREASED DUE TO A RESULTING MISMATCH BETWEEN REACTOR POWER AND FEEDWATER FLOW. RPS SUBSEQUENTLY TRIPPED THE REACTOR ON LOW RCS PRESSURE. PLANT RESPONSE TO THE REACTOR TRIP WAS NORMAL WITH KEY PARAMETERS REMAINING IN THE NORMAL POST-TRIP BAND. THE APPARENT CAUSE OF THE GROUP ROD DROP WAS A DEGRADATION IN THE "A" SIDE POWER TRAIN OF THE CONTROL ROD DRIVE (CRD) SYSTEM SUCH THAT WITH THE "B" SIDE REACTOR TRIP BREAKER OPEN, THE CURRENT SUPPLIED TO THE CRD MOTOR STATORS WAS NOT SUFFICIENT TO SUPPORT MOTION OF THE GROUP 7 CONTROL RODS. AN ACTION PLAN WAS DEVELOPED TO IDENTIFY THE SPECIFIC CAUSE. ACTION PLAN IMPLEMENTATION IS CURRENTLY IN PROGRESS. IMMEDIATE NOTIFICATION WAS MADE PER 10CFR50.72(B)(2)(II) ON DECEMBER 13, 1990, AT 0944 HOURS. THE REACTOR TRIP IS BEING REPORTED PURSUANT TO THE REQUIREMENTS OF 10CFR50.73(A)(2)(IV). A REVISION TO THIS LER WILL BE SUBMITTED WHEN THE ROOT CAUSE IS DETERMINED.

[53] DIABLO CANYON 1 DOCKET 50-275 LER 89-009 REV 01
UPDATE ON REACTOR TRIP AND SAFETY INJECTION FROM STEAM LINE DIFFERENTIAL PRESSURE SPURIOUS SIGNALS.
EVENT DATE: 100689 REPORT DATE: 022191 NSSS: WE TYPE: PWR
VENDOR: AGASTAT RELAY CO.
COPES-VULCAN, INC.
ITT-BARTON

(NSIC 221043) ON 10/6/89, AT 1302 PDT, WITH UNIT 1 IN MODE 1 (POWER OPERATION) AT 100% POWER, AN AUTOMATIC SAFETY INJECTION/REACTOR TRIP WAS ACTUATED FROM A STEAM LINE DIFFERENTIAL PRESSURE SIGNAL. AT 1321 PDT, A 1-HOUR EMERGENCY REPORT WAS MADE IN ACCORDANCE WITH 10 CFR 50.72(A)(1)(I). DURING THE REMOVAL FROM SERVICE PROCESS FOR CALIBRATION OF A PRESSURE TRANSMITTER FOR AN ATMOSPHERIC STEAM DUMP VALVE (ADV) CONTROL, PRESSURE OSCILLATIONS WERE CREATED IN THE COMMON SENSING LINE WITH A PROTECTION SET STEAM GENERATOR PRESSURE TRANSMITTER. THIS CAUSED REPEATED ACTUATIONS OF A STEAM LINE DIFFERENTIAL PRESSURE BISTABLE. BISTABLE ACTUATION, COMBINED WITH A PREVIOUSLY TRIPPED STEAM PRESSURE BISTABLE, SATISFIED THE 2/3 COINCIDENCE LOGIC TO GENERATE A STEAM LINE DIFFERENTIAL PRESSURE SI SIGNAL. THE I&C REMOVAL FROM SERVICE/CALIBRATION PROCEDURE WAS INADEQUATE IN THAT IT DID NOT VERIFY THE CONDITION OF OTHER CHANNEL BISTABLES TO ENSURE THAT COINCIDENT LOGIC COULD NOT BE SATISFIED. I&C PROCEDURES WERE REVISED TO ASSURE THAT WORK ON EQUIPMENT SHARING A COMMON PROCESS TAP IS NOT PERFORMED IF ANY OF THE SHARED INSTRUMENTATION IS IN A CONFIGURATION THAT PRODUCES A PROTECTION ACTUATION OR CONTROL FUNCTION. ACTIONS FOR ASSOCIATED EVENTS INCLUDED DEVELOPING A POLICY FOR GUIDANCE ON TRIPPED BISTABLES AND DEVELOPING A PROCEDURE TO CONTROL NON-TS EQUIPMENT CALLED UPON IN EMERGENCY CONDITIONS.

[54] DIABLO CANYON 1 DOCKET 50-275 LER 89-019
FUEL HANDLING BUILDING VENTILATION SYSTEM INOPERABLE DURING FUEL MOVEMENT DUE TO UNKNOWN CAUSE.
EVENT DATE: 101589 REPORT DATE: 021491 NSSS: WE TYPE: PWR

(NSIC 221044) ON JANUARY 1, 1991, AT 1700 PST, WITH UNIT 1 IN MODE 1 AT 100 PERCENT POWER, THE FUEL HANDLING BUILDING (FHB) VENTILATION SYSTEM WAS DECLARED INOPERABLE AFTER FAILING TO MEET THE NEGATIVE 1/8-INCH WATER GAUGE PRESSURE REQUIREMENTS SPECIFIED IN SURVEILLANCE TEST PROCEDURE (STP) M-41, "FUEL HANDLING BUILDING VENTILATION SYSTEM - DOP AND HALIDE PENETRATION." SINCE THE LAST SUCCESSFUL PERFORMANCE OF STP M-41, FUEL AND HEAVY LOADS WERE MOVED OVER THE SPENT FUEL POOL. NO WORK PERFORMED SUBSEQUENT TO THE COMPLETION OF THE LAST SUCCESSFUL STP M-41 ON SEPTEMBER 18, 1989, THAT COULD HAVE RESULTED IN TEST FAILURE HAS BEEN IDENTIFIED. THEREFORE, IT IS CONSERVATIVELY ASSUMED THAT THE FHB VENTILATION SYSTEM HAS BEEN INOPERABLE SINCE THE LAST PERFORMANCE OF STP M-41, ON SEPTEMBER 18, 1989. FUEL MOVEMENT AND RECEIPT HAVE OCCURRED SINCE THE LAST SUCCESSFUL PERFORMANCE OF STP M-41, INCLUDING FUEL MOVEMENT DURING THE UNIT 1 THIRD REFUELING OUTAGE ON OCTOBER 15, 1989. THEREFORE, IT IS CONSERVATIVELY ASSUMED THAT A POTENTIAL VIOLATION OF TECHNICAL SPECIFICATION (TS) 3.9.12.B. OCCURRED ON OCTOBER 15, 1989. TO BRING THE PRESSURE WITHIN THE ACCEPTANCE CRITERIA, LEAKS HAVE BEEN SEALED, EXHAUST FAN DUCTS HAVE BEEN CLEANED, AND THE SUPPLY FLOW TO THE FHB HAS BEEN TEMPORARILY REDUCED. THE CAUSE OF THE INOPERABLE FHB VENTILATION SYSTEM IS UNDER INVESTIGATION.

[55] DIABLO CANYON 1 DOCKET 50-275 LER 90-013
MISSED REACTOR COOLANT SYSTEM SAMPLE DUE TO PERSONNEL ERROR.
EVENT DATE: 110490 REPORT DATE: 120390 NSSS: WE TYPE: PWR

(NSIC 220889) ON NOVEMBER 4, 1990, AT 0528 PST, THE SURVEILLANCE REQUIRED BY TECHNICAL SPECIFICATION (TS) 4.4.8, INCLUDING THE ALLOWED EXTENSION OF TS 4.0.2.A, WAS EXCEEDED WHEN A REQUIRED SAMPLE COLLECTION WAS MISSED. ON NOVEMBER 3, 1990, UNIT 1 RATED THERMAL POWER CHANGE WAS GREATER THAN 15 PERCENT IN ONE HOUR AT 2158 PST. TS 4.4.8 REQUIRES THAT A REACTOR COOLANT SYSTEM (RCS) GRAB SAMPLE BE TAKEN WITHIN BETWEEN TWO TO SIX HOURS FOLLOWING EACH POWER CHANGE. ON NOVEMBER 4, 1990, AT 0710 PST, THE CHEMISTRY FOREMAN DISCOVERED THAT A SAMPLE HAD NOT BEEN COLLECTED. A SAMPLE WAS TAKEN AT 0725 PST, AND NO ABNORMAL CONDITIONS WERE NOTED. THE ROOT CAUSE OF THIS EVENT IS PERSONNEL ERROR BY PG&E SHIFT

CHEMISTRY AND RADIATION PROTECTION (CARP) TECHNICIANS. CORRECTIVE ACTIONS TO PREVENT RECURRENCE INCLUDE COUNSELING THE TECHNICIANS INVOLVED AND TAILBOARD BRIEFING OF ALL CARP TECHNICIANS REGARDING THIS EVENT.

[56] DIABLO CANYON 1 DOCKET 50-275 LER 90-015 REV 01
UPDATE ON ESF ACTUATION, P-14 (HIGH-HIGH STEAM GENERATOR LEVEL), DUE TO FEEDWATER VALVE LEAKAGE.
EVENT DATE: 120890 REPORT DATE: 012591 NSSS: WE TYPE: PWR

(NSIC 220840) ON DECEMBER 8, 1990, AT 0031 HOURS PST, WITH UNIT 1 IN MODE 2 (STARTUP) AT APPROXIMATELY 2 PERCENT REACTOR POWER, STEAM GENERATOR (SG) 1-3 LEVEL EXCEEDED THE 67 PERCENT NARROW RANGE LEVEL SETPOINT INITIATING A P-14 SIGNAL, AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION. THE ROOT CAUSE OF THE EVENT IS LEAKAGE THROUGH FEEDWATER REGULATING AND BYPASS VALVES FW-1-FCV-530 AND FW-1-FCV-1530. LEAKAGE PAST FEEDWATER CHECK VALVE FW-1-531 AND FEEDWATER RECIRCULATION CONTROL VALVE FW-1-FCV-420 CONTRIBUTED TO THIS EVENT. OTHER CONTRIBUTING CAUSES INCLUDED PLANT MANAGEMENT'S FAILURE TO RECOGNIZE THE CONSEQUENCES OF THE COMBINED LEAKAGE OF THE FEEDWATER CHECK AND REGULATING VALVES, AND TO PROVIDE DIRECTION FOR EITHER A TIMELY REPAIR OF THE FEEDWATER VALVES OR AN ADEQUATE STRATEGY FOR OPERATORS TO ENABLE THEM TO STARTUP THE UNIT WITH LEAKING FEEDWATER VALVES. OPERATING PROCEDURE OP L-3 WAS REVISED TO INCLUDE STEPS TO CHECK FOR MAIN FEEDWATER REGULATING AND BYPASS VALVE LEAKAGE DURING STARTUP, AND TO PROVIDE GUIDANCE FOR TAKING ACTION TO DEAL WITH ANY LEAKAGE IDENTIFIED. OP L-3 WAS ALSO REVISED TO REQUIRE ISOLATING FW-1-FCV-420 WHEN IT IS NOT IN SERVICE. FW-1-FCV-420 WAS ISOLATED AND WILL BE REPAIRED DURING THE NEXT REFUELING OUTAGE. CORRECTIVE MAINTENANCE WAS PERFORMED ON FW-1-FCV-530, FW-1-FCV-1530, AND FW-1-531 DURING A UNIT 1 FORCED OUTAGE.

[57] DIABLO CANYON 1 DOCKET 50-275 LER 90-018
FIRE DAMPER CARDOX ACTUATION FUSIBLE LINK ASSEMBLY INCORRECTLY INSTALLED FOR INDETERMINATE REASON.
EVENT DATE: 122190 REPORT DATE: 012191 NSSS: WE TYPE: PWR

(NSIC 220841) ON DECEMBER 12, 1990, DURING PERFORMANCE OF SURVEILLANCE TEST PROCEDURE (STP) M-39B, "ROUTINE SURVEILLANCE TEST OF CABLE SPREADING ROOM CARBON DIOXIDE FIRE SYSTEM OPERATION," UNIT 1 CABLE SPREADING ROOM VENTILATION SYSTEM SUPPLY FIRE DAMPER VAC-1-FD-220 WAS OBSERVED TO BE OPEN AFTER PRESSURIZATION OF THE CO2 HEADER, CONTRARY TO THE INTENDED CLOSED POSITION. THE FAILURE OF THE FIRE DAMPER TO CLOSE WAS DUE TO INCORRECT INSTALLATION OF THE CARDOX SYSTEM ACTUATION ROD AND FUSIBLE LINK ASSEMBLY FOLLOWING THE LAST SURVEILLANCE ACTIVITY ON THIS DAMPER ON NOVEMBER 29, 1989. THE ROOT CAUSE OF THIS EVENT WAS INDETERMINATE BUT WAS MOST LIKELY DUE TO PERSONNEL ERROR. THIS CONDITION COULD HAVE RESULTED IN DILUTION OF THE CO2 DISCHARGED IN THIS AREA IN THE EVENT OF A FIRE. THE ROD AND LINK ASSEMBLY WAS REASSEMBLED TO THE CORRECT CONFIGURATION AND SUCCESSFULLY TESTED. THE SIMILAR ASSEMBLY FOR THE UNIT 2 CABLE SPREADING ROOM DAMPER WAS INSPECTED AND FOUND ACCEPTABLE. STP M-39B WILL BE REVISED TO INCLUDE AN ILLUSTRATION OF THE CORRECT ASSEMBLED CONFIGURATION OF THE CARDOX ACTUATION ROD AND FUSIBLE LINK ASSEMBLY.

[58] DIABLO CANYON 1 DOCKET 50-275 LER 90-017
REACTOR TRIP RESULTING FROM FAILED OPEN PRESSURIZER SPRAY VALVE DUE TO INCORRECT SCREW INSTALLATION.
EVENT DATE: 122490 REPORT DATE: 012391 NSSS: WE TYPE: PWR
VENDOR: COPES-VULCAN, INC.

(NSIC 220915) ON 12/24/90, AT 0318 PST, WITH UNIT 1 IN MODE 1 (POWER OPERATION) AT 88% POWER, A REACTOR TRIP AND SAFETY INJECTION OCCURRED DUE TO LOW PRESSURIZER PRESSURE. DURING THE RECOVERY FROM THE TRIP, REACTOR COOLANT SYSTEM (RCS) COOLDOWN EXCEEDED THE ALLOWABLE RATE OF 100F PER HOUR OF TECH SPEC 3.4.9. AN UNUSUAL EVENT WAS DECLARED AT 0320 PST. A ONE-HOUR EMERGENCY REPORT REQUIRED BY 10 CFR 50.72(A)(1)(I) WAS MADE ON 12/24/90, AT 0342 PST. ON 12/24/90, AN EVENT INVESTIGATION TEAM WAS FORMED TO INVESTIGATE THE EVENT. THE CAUSE OF THE TRIP WAS A PRESSURIZER SPRAY VALVE THAT FAILED OPEN DUE TO ITS FEEDBACK LINKAGE

BECOMING DISCONNECTED. THE FEEDBACK LINKAGE BECAME DISCONNECTED BECAUSE A LOCKING DEVICE WAS NOT INSTALLED ON THE SCREW HOLDING THE LINKAGE TO THE VALVE STEM. THE FAILURE OF THE PILOT STEM OF A CONDENSER STEAM DUMP VALVE CONTRIBUTED TO THE OVERCOOLING OF THE RCS. THE ROOT CAUSE FOR THE PILOT VALVE FAILURE IS UNDER INVESTIGATION AND WILL BE REPORTED IN A SUPPLEMENTAL LER. CORRECTIVE ACTIONS FOR THE EVENT INCLUDE REVISING MAINTENANCE PROCEDURE I-2.25-1 TO ADDRESS THE USE OF APPROPRIATE LOCKING DEVICES ON FEEDBACK LINKAGES, REVISING ABNORMAL OPERATING PROCEDURE AP-13 FOR DEALING WITH FAILED OPEN PRESSURIZER SPRAY VALVES, REVISING EMERGENCY PROCEDURE E-0 ON CLOSING THE MAIN STEAM ISOLATION VALVES, REVISING DESIGN DRAWINGS, AND SENDING A LETTER TO THE VENDOR.

[59] DIABLO CANYON 1 DOCKET 50-275 LER 90-019
ACTUATION OF CONTAINMENT VENTILATION ISOLATION DUE TO PERSONNEL ERROR.
EVENT DATE: 122790 REPORT DATE: 012891 NSSS: WE TYPE: PWR

(NSIC 220842) ON DECEMBER 27, 1990, AT 1431 PST, WITH UNIT 1 IN MODE 3 (HOT STANDBY), A CONTAINMENT VENTILATION ISOLATION (CVI) ACTUATION OCCURRED. THIS EVENT CONSITITUTES AN ENGINEERED SAFETY FEATURE ACTUATION. THE 4-HOUR, NON-EMERGENCY REPORT REQUIRED BY 10 CFR 50.72 (B)(2)(II) WAS MADE TO THE NRC ON DECEMBER 27, 1990, AT 1518 PST. A CONTRACT ELECTRICIAN WAS PERFORMING DESIGN MODIFICATIONS IN AN ENERGIZED RADIATION MONITOR CABINET. AS THE ELECTRICIAN REMOVED THE PLIERS FROM THE CABINET, THE PLIERS CAME IN CONTACT WITH THE TERMINALS ON A FUSE BLOCK, CAUSING A VOLTAGE TRANSIENT ON AN INVERTER. THE TRANSIENT CAUSED ALARMS ON TWO RADIATION MONITORS SUPPLIED BY THE INVERTER, RESULTING IN THE CVI. THE ROOT CAUSE WAS DETERMINED TO BE PERSONNEL ERROR (COGNITIVE) IN THAT IF THE ELECTRICIAN HAD TAPED THE TOOL IN ACCORDANCE WITH STANDARD WORK PRACTICES FOR WORKING IN ENERGIZED CABINETS, ELECTRICAL CONTACT WITH THE FUSE BLOCK MAY NOT HAVE OCCURRED. A CONTRIBUTARY CAUSE WAS DETERMINED TO BE THAT A PREVIOUSLY ISSUED MAINTENANCE BULLETIN REGARDING WORK ON ENERGIZED EQUIPMENT, WHICH RECOMMENDS TAPING TOOLS, HAD NOT BEEN REVIEWED WITH THE ELECTRICIAN. TO PREVENT RECURRENCE OF THIS EVENT, ALL PREVIOUS MAINTENANCE BULLETINS NOT DISTRIBUTED TO GENERAL CONSTRUCTION (GC) WILL BE DISTRIBUTED, TAILBOARD MEETINGS WILL BE HELD WITH A11 GC CREWS TO REVIEW PREVIOUS MAINTENANCE BULLETINS RELEVANT TO THIS EVENT.

[60] DRESDEN 2 DOCKET 50-237 LER 90-011
UNPLANNED AUTOMATIC START OF THE DIESEL GENERATOR DUE TO PROCEDURE DEFICIENCY.
EVENT DATE: 102790 REPORT DATE: 110590 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: DRESDEN 3 (BWR)

(NSIC 220885) AT 0520 HOURS ON 10/27/90, AN UNPLANNED AUTOMATIC START OF THE UNIT 2/3 DIESEL GENERATOR (DG) OCCURRED WHILE OPERATIONS DEPARTMENT PERSONNEL WERE PREPARING TO TAKE 4 KV BUSSES 23 AND 23-1 OUT OF SERVICE (OOS) TO FACILITATE BREAKER AND CUBICLE PREVENTATIVE MAINTENANCE WORK. DRESDEN UNIT 2 WAS IN COLD SHUTDOWN FOR A REFUEL OUTAGE, WHILE UNIT 3 WAS UNDER NORMAL POWER OPERATION AT 90% RATED CORE THERMAL POWER. THE UNIT 2/3 DG PROVIDES EMERGENCY AC POWER TO EITHER OR BOTH UNITS; THE SAFETY SIGNIFICANCE OF THIS EVENT WAS MINIMAL BECAUSE THE UNIT 2 AND UNIT 3 DGS WERE OPERABLE AND THE UNPLANNED UNIT 2/3 DG AUTOSTART HAD NO AFFECT ON ITS AVAILABILITY TO POWER EMERGENCY AC LOADS IF NEEDED. THE ROOT CAUSE WAS ATTRIBUTED TO PROCEDURE DEFICIENCY DUE TO INADEQUATE PROCEDURAL GUIDANCE TO PREPARE THE OOS INSTRUCTIONS. CORRECTIVE ACTIONS WILL INCLUDE DEVELOPMENT OF IMPROVED PROCEDURES FOR DE-ENERGIZING 4KV BUSSES WHICH HAVE THE POTENTIAL FOR UNPLANNED DG STARTS, CLARIFICATION OF POLICY CONCERNING TECHNICAL ASSISTANCE FOR SUCH EVOLUTIONS, AND LABELING IMPROVEMENTS. A PREVIOUS EVENT INVOLVING AN UNPLANNED ESF ACTUATION DURING THE CURRENT UNIT 2 REFUEL OUTAGE WAS REPORTED BY LER 90-10/050237.

[61] DRESDEN 2 DOCKET 50-237 LER 90-015
INTERMEDIATE RANGE MONITOR FULL SCRAM DUE TO INDUCTIVE NOISE INPUT TO THE IRM/SRM POWER SUPPLIES.
EVENT DATE: 112390 REPORT DATE: 121890 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 221006) AT 1017 HOURS ON 11/23/90, AN AUTOMATIC REACTOR SCRAM OCCURRED DUE TO A SPURIOUS HIGH CORE FLUX SIGNAL FROM INTERMEDIATE RANGE MONITORS (IRM) 13 AND 15. DURING THE PERFORMANCE OF DRESDEN INSTRUMENT SURVEILLANCE (DIS) 1500-5, LOW PRESSURE COOLANT INJECTION LOGIC (LPCI) TEST LOGIC RELAY 2-1530-115 AY GENERATED AN ABNORMALLY LARGE VOLTAGE SPIKE ON THE 125 VOLT DC POWER SYSTEM. THROUGH ELECTROMAGNETIC INDUCTION THIS VOLTAGE SPIKE WAS TRANSFERRED TO THE 24/48 VOLT DC POWER SYSTEM. THE 24/48 VOLT DC SYSTEM PROVIDES POWER TO THE CONTROL LOGIC FOR ALL EIGHT IRM CHANNELS. THE SPIKE CAUSED THE IRM CHANNELS TO EXCEED THE HIGH CORE FLUX SCRAM SETPOINT THUS CAUSING A REACTOR SCRAM. CORRECTIVE ACTIONS INCLUDED REPLACEMENT OF THE SUSPECT RELAY AND THE INITIATION OF FURTHER INVESTIGATION. THE SAFETY SIGNIFICANCE OF THIS EVENT IS MINIMAL SINCE UNIT 2 WAS ALREADY SHUTDOWN FOR A REFUELING OUTAGE WHEN THE EVENT OCCURRED.

[62] DRESDEN 2 DOCKET 50-237 LER 90-022
UNEXPECTED CLOSURE OF 11 CONTAINMENT ISOLATION VALVES DURING SURVEILLANCE TESTING DUE TO PROCEDURE DEFICIENCY.
EVENT DATE: 120890 REPORT DATE: 012191 NSSS: GE TYPE: BWR

(NSIC 220719) AT 0626 HOURS ON 12/8/90, WITH UNIT 2 IN THE SHUTDOWN MODE, 11 OF 44 GROUP II PRIMARY CONTAINMENT ISOLATION VALVES WENT CLOSED WHILE AN ELECTRICIAN WAS PERFORMING AN ENVIRONMENTAL QUALIFICATION (EQ) SURVEILLANCE ON THE 2A MAIN STEAM ISOLATION VALVE (MSIV) SOLENOIDS. THE ELECTRICIAN INADVERTENTLY REMOVED A JUMPER INTERRUPTING CONTROL POWER TO SEVERAL SEAL-IN RELAYS CAUSING THE 11 ISOLATION VALVES TO CLOSE. NO GROUP II ISOLATION ALARM WAS RECEIVED. THE ELECTRICIAN IMMEDIATELY RELANDED THE LEAD AND TERMINATED THE SURVEILLANCE. THE UNIT 2 OPERATOR REOPENED THE 11 VALVES AND INITIATED AN INVESTIGATION. THE ROOT CAUSE IS BEING ATTRIBUTED TO A PROCEDURE DEFICIENCY WHICH FAILED TO IDENTIFY A WIRING CONFIGURATION DISCREPANCY IN CONTROL ROOM PANEL 902-3. THE EVENT HAD NO SAFETY SIGNIFICANCE AS ALL SAFETY SYSTEM FUNCTIONS WERE UNAFFECTED. CORRECTIVE ACTIONS INCLUDED A FIELD VERIFICATION OF THE SURVEILLANCE AND CORRECTION OF IDENTIFIED WIRING CONFIGURATION DISCREPANCIES BOTH ON UNIT 2 AND UNIT 3. TWO PREVIOUS EVENTS FOUND INVOLVING UNPLANNED ACTUATION OF ENGINEERED SAFETY FEATURES COMPONENTS ARE REPORTED ON LER 90-10/050237 AND LER 90-11/050237.

[63] DRESDEN 2 DOCKET 50-237 LER 90-018
LEAKAGE PATH DISCOVERED DURING PRIMARY CONTAINMENT ILRT DUE TO MANAGEMENT DEFICIENCY.
EVENT DATE: 121890 REPORT DATE: 011491 NSSS: GE TYPE: BWR

(NSIC 220745) ON DECEMBER 17, 1990, WHILE PERFORMING A PRIMARY CONTAINMENT INTEGRATED LEAK RATE TEST (ILRT) DURING THE UNIT 2 REFUEL OUTAGE, LEAKAGE IN EXCESS OF THE TECHNICAL SPECIFICATION 3.7.A.2 ILRT REQUIREMENT WAS MEASURED DUE TO A LEAKING REACTOR BUILDING TO PRESSURE SUPPRESSION CHAMBER VACUUM BREAKER VALVE A02-1601-20A INBOARD FLANGE. FURTHER REVIEW ON DECEMBER 18, 1990, INDICATED THAT THIS VACCUM BREAKER HAD BEEN REPLACED DURING THE PREVIOUS REFUEL OUTAGE WITHOUT PROPER TESTING OF THE INBOARD FLANGE CONNECTION; 10 CFR50.72 NOTIFICATION WAS THEN COMPLETED. ALTHOUGH THIS DEGRADED CONDITION POTENTIALLY EXISTED DURING THE PREVIOUS OPERATING CYCLE, THE SECONDARY CONTAINMENT WOULD HAVE MITIGATED RELEASE TO THE ENVIRONS UNDER POSTULATED DESIGN BASIS ACCIDENT CONDITIONS. ANALYSIS OF 10CFR 100 REQUIREMENTS IS IN PROGRESS, AND A SUPPLEMENTAL REPORT WILL PROVIDE FURTHER INFORMATION. THE ILRT WAS COMPLETED SATISFACTORILY AFTER THE FLANGE WAS TIGHTENED. THE UNDERLYING CAUSE FOR NOT CHALLENGING THIS PATHWAY UPON EARLIER REPLACEMENT OF THE VACUUM BREAKER WAS ATTRIBUTED TO MANAGEMENT DEFICIENCY IN THAT PERFORMANCE OF AN ILRT FOLLOWING THIS ACTIVITY WAS NOT PROPERLY SPECIFIED OR IDENTIFIED. A PREVIOUS DRESDEN UNIT 2 ILRT FAILURE DUE TO UNRELATED CAUSES IS REPORTED BY LER 83-29/050237.

[64] DRESDEN 2 DOCKET 50-237 LER 90-019
ELECTROMATIC RELIEF VALVE PRESSURE SWITCH OUTSIDE TECH SPEC DUE TO INSTRUMENT SETPOINT DRIFT.
EVENT DATE: 121990 REPORT DATE: 011691 NSSS: GE TYPE: BWR
VENDOR: DRESSER INDUSTRIAL VALVE & INST DIV

(NSIC 220863) ON 12/19/90 WHILE PERFORMING DRESDEN INSTRUMENT SURVEILLANCE (DIS) 250-3, ELECTROMATIC RELIEF VALVE (ERV)/TARGET ROCK VALVE PRESSURE SWITCH CALIBRATION, THE PRESSURE SWITCH FOR ERV 2-203-3C WAS FOUND OUTSIDE THE TECH SPEC 4.6.E SETPOINT TOLERANCE LIMIT OF +/- 1%. DURING RECALIBRATION OF THE PRESSURE SWITCH, DIFFICULTY WAS ENCOUNTERED ADJUSTING THE SETPOINT TO WITHIN THE ALLOWABLE RANGE. UNIT 2 WAS IN A REFUEL OUTAGE WHEN THIS SETPOINT DISCREPANCY WAS DISCOVERED. THE ROOT CAUSE OF THIS SETPOINT DISCREPANCY HAS BEEN ATTRIBUTED TO INSTRUMENT SETPOINT DRIFT, CONTRIBUTED TO BY THE PRESSURE SWITCH BOURDON TUBE HAVING AN ABNORMALLY WIDE PRESSURE RATING. AS CORRECTIVE ACTION, THE BOURDON TUBE WAS REPLACED WITH ONE OF NORMAL PRESSURE RANGE, AND THE PRESSURE SWITCH FOR ERV 2-203-3C WAS RECALIBRATED TO WITHIN TECH SPEC LIMITS PER DIS 250-3. THE REMAINING ERV PRESSURE SWITCHES TESTED SATISFACTORILY, AND ALL UNIT 2 AND UNIT 3 ERV PRESSURE SWITCH BOURDON TUBES WERE CHECKED FOR PROPER RATING. THE SAFETY SIGNIFICANCE OF THIS EVENT WAS MINIMAL BECAUSE THE MAIN STEAM LINE SAFETY VALVES WERE AVAILABLE TO INSURE THAT THE REACTOR COOLANT SYSTEM PRESSURE LIMIT WOULD NOT BE EXCEEDED, AND THE ERV SETPOINT DISCREPANCY HAD NO EFFECT ON THE OPERATION OF THE AUTOMATIC DEPRESSURIZATION SYSTEM (ADS). A PREVIOUS OCCURRENCE OF THIS TYPE AT DRESDEN STATION WAS REPORTED BY LER 89-7/050237.

[65] DRESDEN 2 DOCKET 50-237 LER 90-017
REACTOR SCRAM ON INTERMEDIATE RANGE MONITOR HI-HI DUE TO UNKNOWN CAUSE.
EVENT DATE: 122090 REPORT DATE: 010991 NSSS: GE TYPE: BWR

(NSIC 220744) AT 0420 HOURS ON DECEMBER 20, 1990, AN AUTOMATIC REACTOR SCRAM OCCURRED DUE TO A SPURIOUS HI-HI NEUTRON FLUX SIGNAL FROM THE INTERMEDIATE RANGE MONITORS (IRMS). A SIMILAR EVENT OCCURRED ON DECEMBER 23, 1990 AT 0538 HOURS WHEN ONCE AGAIN IRM HI-HI FLUX INDICATION CAUSED AN AUTOMATIC REACTOR SCRAM. IN BOTH CASES, UNIT 2 WAS IN THE SHUTDOWN MODE WITH ALL RODS IN AT THE TIME OF THE SCRAM. ALSO, ALL SOURCE RANGE MONITORS (SRMS) AND IRMS WERE OBSERVED TO SPIKE SPURIOUSLY IN EACH CASE. AN INVESTIGATION INTO THE EVENT BY THE TECHNICAL STAFF, INSTRUMENT MAINTENANCE DEPARTMENT, AND SYSTEM OPERATIONAL ANALYSIS DEPARTMENT FAILED TO DETERMINE THE ROOT CAUSE OF THE SPIKING. THE INVESTIGATION INCLUDED, AMOUNG OTHER THINGS, A REVIEW OF PLANT CONDITIONS AT THE TIME OF EACH SCRAM, A STUDY OF THE IRM RECORDER STRIP CHARTS AND THE UNIT 2 ALARM TYPER, A REVIEW OF SRM/IRM CABLE ROUTING, AND ELECTRONIC MONITORING OF SELECTED SRM/IRM DRAWERS. SINCE THE ROOT CAUSE OF THE SPIKING HAS NOT BEEN DETERMINED AND THE SPIKING COULD NOT BE ARTIFICIALLY REPRODUCED, NO FURTHER CORRECTIVE ACTION WILL BE TAKEN. THE SAFETY SIGNIFICANCE OF THESE EVENTS IS MINIMAL SINCE THE UNIT WAS IN THE SHUTDOWN MODE WITH ALL RODS IN FOR BOTH CASES. ONE PREVIOUS OCCURRENCE OF A SIMILAR EVENT IS DOCUMENTED IN LICENSEE EVENT REPORT 90- 015/050237, INTERMEDIATE RANGE MONITOR FULL SCRAM DUE TO INDUCTIVE NOISE INPUT TO THE IRM/SRM POWER SUPPLIES.

[66] DRESDEN 2 DOCKET 50-237 LER 91-001
PARTIAL GROUP I ISOLATION DUE TO SHORTING OF 1B MSIV POSITION INDICATING LIGHT SOCKET.
EVENT DATE: 010591 REPORT DATE: 012891 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 220837) ON 1/5/91 WHILE PERFORMING DRESDEN OPERATING SURVEILLANCE (DOS) 250-2, FULL CLOSURE TIMING AND EXERCISING OF MAIN STEAM ISOLATION VALVES (MSIVS), UNIT 2 NUCLEAR STATION OPERATOR (NSO) NOTED THAT THE OPEN POSITION INDICATION FOR THE 1B MSIV WAS NOT ILLUMINATED. WHILE THE NSO WAS ATTEMPTING TO REPLACE THE BULB, THE BULB SOCKET SHORTED CAUSING FUSE 595-709-A IN CONTROL ROOM PANEL 902-3 TO BLOW. THE BLOWN FUSE, WHICH POWERS HALF THE PRIMARY CONTAINMENT GROUP I ISOLATION LOGIC, INITIATED THE CLOSURE OF BOTH THE RECIRCULATION LOOP SAMPLE VALVE 2-220-44 AND MAIN STEAM LINE DRAIN VALVE 2-220-1. THE BLOWN FUSE CAUSED THE LOSS OF POSITION INDICATION AND THE DE-ENERGIZATION OF THE DC PILOT VALVE SOLENOID FOR EACH INBOARD MSIV. UNIT 2 WAS IN THE STARTUP MODE AT 6% POWER WHEN THIS EVENT OCCURRED. THE CAUSE OF THE SHORTED SOCKET AND PARTIAL GROUP I ISOLATION WAS THE INTRODUCTION OF FOREIGN MATERIAL INTO THE LIGHT BULB SOCKET. AS CORRECTIVE ACTION, THE LIGHT SOCKET WAS REPLACED, AFFECTED CIRCUITRY WAS THEN ENERGIZED, AND THE SYSTEM WAS RETURNED TO NORMAL. THE SAFETY SIGNIFICANCE OF THIS EVENT WAS CONSIDERED MINIMAL BECAUSE NONE OF THE MSIVS EXPERIENCED AUTOMATIC CLOSURE, AND CLOSURE OF RECIRCULATION LOOP SAMPLE VALVE 2-220-44 AND MAIN STEAM

LINE DRAIN VALVE 2-220-1 HAD MINIMAL EFFECT ON PLANT OPERATION. A PREVIOUS OCCURRENCE OF THIS TYPE AT DRESDEN STATION WAS REPORTED BY LER 90-002/0500237.

[67] DRESDEN 3 DOCKET 50-249 LER 91-001
DE-ENERGIZATION OF REACTOR PROTECTION SYSTEM DUE TO PROCEDURAL DEFICIENCY.
EVENT DATE: 011491 REPORT DATE: 020691 NSSS: GE TYPE: BWR

(NSIC 220974) AT 1244 HOURS ON 1/14/91 WITH UNIT 3 IN SHUTDOWN AT 0% RATED CORE THERMAL POWER, DRESDEN TECHNICAL STAFF SURVEILLANCE (DTS) 500-2, FUNCTIONAL TESTING OF REACTOR PROTECTION SYSTEM (RPS) MOTOR GENERATOR (MG) SET AND RPS RESERVE POWER SUPPLY, WAS BEING PERFORMED. WHEN THE POWER FEED TO THE "B" RPS WAS TRANSFERRED FROM THE MG SET TO UNIT 3 BUS 39-5, AN UNPLANNED REACTOR SCRAM SIGNAL OCCURRED. DURING THE TEST A HALF SCRAM WOULD NORMALLY OCCUR WHEN THIS STEP IS PERFORMED. HOWEVER, THE REACTOR WAS ALREADY UNDER A FULL MANUAL SCRAM, AND BECAUSE THE SCRAM DISCHARGE VOLUME (SDV) HIGH WATER LEVEL RPS TRIPS WERE BYPASSED, BOTH CHANNELS OF RPS PRODUCED A SCRAM SIGNAL. THE SAFETY SIGNIFICANCE OF THIS EVENT WAS CONSIDERED MINIMAL BECAUSE THE REACTOR WAS ALREADY IN SHUTDOWN MODE, WITH ALL CONTROL BLADES FULLY INSERTED, AND A FULL SCRAM SIGNAL WAS ALREADY INSERTED. THE CAUSE ATTRIBUTED TO THE DE-ENERGIZATION OF BOTH RPS CHANNELS IS PROCEDURAL DEFICIENCY. THE PROCEDURE LACKED SUFFICIENT CONSTRAINTS CONCERNING PERFORMANCE OF THE SURVEILLANCE UNDER A PREEXISTENT SCRAM CONDITION. A REVISION TO THE FUNCTIONAL TESTING PROCEDURE, DTS 500-2, WILL BE MADE TO INCLUDE THE STEPS NECESSARY TO ADEQUATELY PERFORM THE SURVEILLANCE. THIS WAS THE FIRST OCCURRENCE OF THIS TYPE.

[68] FERMI 2 DOCKET 50-341 LER 90-013
INADEQUATE CONTROL DURING THE PRIMARY CONTAINMENT AIR GRAB SAMPLING PROCESS.
EVENT DATE: 122890 REPORT DATE: 012891 NSSS: GE TYPE: BWR

(NSIC 220867) DURING A RECENT INDUSTRY OPERATING EXPERIENCE REVIEW, IT WAS DETERMINED THAT FERMI 2 PRIMARY CONTAINMENT ATMOSPHERE SAMPLING PRACTICES COULD POTENTIALLY COMPROMISE PRIMARY CONTAINMENT INTEGRITY. THE NEED FOR PROMPT ISOLATION OF THE SAMPLING LINE IN THE EVENT OF A DESIGN BASIS ACCIDENT/LOSS OF COOLANT ACCIDENT (DBA/LOCA) WAS NOT IDENTIFIED IN A PROCEDURE. THE ROOT CAUSE OF THIS CONDITION WAS A LACK OF PROPER ADMINISTRATIVE CONTROLS IN THE SAMPLING PROCEDURE. THE TECHNICIAN COULD HAVE MANUALLY ISOLATED THE SAMPLE LINES IN 60 SECONDS WHICH IS CONSISTENT WITH THE TECH SPEC MAXIMUM ALLOWED AUTOMATIC ISOLATION TIME OF 60 SECONDS. IF THE TECHNICIAN FAILED TO TAKE ACTION, THE CONTROL ROOM OPERATOR COULD HAVE ISOLATED THE LINES. IMMEDIATE CORRECTIVE ACTION WAS TAKEN TO ADD SAMPLE SUCTION AND RETURN TAPS TO THE PRIMARY CONTAINMENT RADIATION MONITORING SKID WHICH HAS REDUNDANT AUTOMATIC CONTAINMENT ISOLATION VALVES UPSTREAM OF THE SAMPLE SUCTION TAPE AND DOWNSTREAM SAMPLE RETURN TAPS. THESE VALVES WILL ISOLATE SHOULD A DBA/LOCA SIGNAL BE RECEIVED. SAMPLING PROCEDURES HAVE BEEN REVISED TO REFLECT THE USE OF THIS MODIFIED SAMPLING SYSTEM FOR PRIMARY CONTAINMENT ATMOSPHERE SAMPLING.

[69] FITZPATRICK DOCKET 50-333 LER 90-025 REV 01
UPDATE ON FIVE SERVICE WATER TO EMERGENCY SERVICE WATER SWING CHECK VALVES FAIL TO CLOSE DURING TESTING DUE TO CORROSION AND SILT ACCUMULATION IN HINGE.
EVENT DATE: 111590 REPORT DATE: 012591 NSSS: GE TYPE: BWR
VENDOR: VELAN VALVE CORP.

(NSIC 220879) ON 11/15/90 AND 12/26/90 THE PLANT WAS OPERATING AT 100% POWER. DURING A SCHEDULED ASME SECTION XI IN-SERVICE TEST PROGRAM SURVEILLANCE TEST, THREE 3-INCH SWING CHECK VALVES FAILED TO CLOSE ON 11/15/90. TWO ADDITIONAL VALVES FAILED ON 12/26/90. FOLLOWING THE INITIAL TESTS, FOUR OF THE FIVE VALVES CLOSED WHEN TAPPED WITH A TOOL HANDLE. THE VALVES SUPPLY SERVICE WATER (SWS) (KG) TO NINE AREA VENTILATION UNIT COOLERS LOCATED IN SPACES CONTAINING SAFETY-RELATED ELECTRICAL SWITCHGEAR AND EMERGENCY CORE COOLING SYSTEM EQUIPMENT. THE VALVES ARE INTENDED TO CLOSE UPON LOSS OF SERVICE WATER PRESSURE TO PREVENT DIVERSION OF THE EMERGENCY SERVICE WATER (ESW) (BI) SUPPLY AWAY FROM THE COOLERS. ON 11/16/90 AND 12/27/90 CARBON STEEL VALVE INTERNALS WERE REPLACED WITH STAINLESS STEEL COMPONENTS TO AVOID CORROSION PROBLEMS WHICH CONTRIBUTED TO THE

STICKING CONDITION. THE AS-FOUND STUCK OPEN VALVE CONDITION WOULD NOT BE EXPECTED TO RESULT IN CONDITIONS ADVERSE TO SAFETY IN THE EVENT OF AN FSAR POSTULATED ACCIDENT. THIS IS A VOLUNTARY REPORT. RELATED LERS: 88-055, 88-009, AND 90-012.

[70] FITZPATRICK DOCKET 50-333 LER 90-026
REACTOR SCRAM DURING REACTOR WATER LEVEL INSTRUMENT SURVEILLANCE, SECOND REACTOR SCRAM DUE TO AIR DIAPHRAGM FAILURE IN FEED FLOW CONTROL VALVE.
EVENT DATE: 121290 REPORT DATE: 011191 NSSS: GE TYPE: BWR
VENDOR: MASONEILAN INTERNATIONAL, INC.

(NSIC 220668) A REACTOR SCRAM FROM FULL POWER OCCURRED AT 1352 ON 12/12/90 DURING CALIBRATION OF REACTOR WATER LEVEL INSTRUMENTATION. THE INSTRUMENT BEING CALIBRATED SHARED COMMON REFERENCE AND VARIABLE LEVEL LEGS WITH INSTRUMENTS OF THE REACTOR PROTECTION SYSTEM (JC). THE REACTOR SCRAMMED AS THE INSTRUMENT HIGH PRESSURE ISOLATION VALVE WAS BEING CRACKED OPEN DURING RETURN TO SERVICE. THE SCRAM RESULTED FROM A FALSE LOW REACTOR WATER LEVEL SIGNAL. DURING THE ACTUAL LEVEL TRANSIENT FOLLOWING THE SCRAM DIFFICULTY WAS EXPERIENCED WITH RESTARTING THE REACTOR FEEDWATER PUMPS AND A FAILURE OF THE REACTOR FEEDWATER LOW FLOW CONTROL VALVE OCCURRED. A SECOND SCRAM OCCURRED DUE TO AN ACTUAL LOW REACTOR WATER LEVEL AT 1416 DUE TO FAILURE OF THE REACTOR FEEDWATER LOW FLOW CONTROL VALVE AIR OPERATOR DIAPHRAGM. THE PLANT RETURNED TO SERVICE AT 0658 ON 12/17/90 AFTER BEING OFF LINE FOR 4 DAYS, 17 HOURS, AND 6 MINUTES. A ROOT CAUSE INVESTIGATION OF THIS SCRAM IS IN PROGRESS AND IS EXPECTED TO BE COMPLETED PRIOR TO THE END OF THE FALL 1991 REFUELING OUTAGE. UNTIL THEN, FUTURE CALIBRATIONS WILL BE CONDUCTED DURING SCHEDULED OUTAGES. RELATED LERS: 90-001 AND 90-027.

[71] FITZPATRICK DOCKET 50-333 LER 90-027
REACTOR SCRAM DURING START-UP DUE TO HIGH NEUTRON FLUX DUE TO FAILED AIR OPERATOR DIAPHRAGM ON REACTOR FEEDWATER LOW FLOW CONTROL VALVE.
EVENT DATE: 121590 REPORT DATE: 011491 NSSS: GE TYPE: BWR
VENDOR: MASONEILAN INTERNATIONAL, INC.

(NSIC 220817) A REACTOR SCRAM FROM SIX PERCENT POWER OCCURRED DURING A START-UP ON DECEMBER 15, 1990 AT 2140. THE CAUSE WAS A FAILURE OF A FABRIC-WOVEN BUNA-N DIAPHRAGM WHICH HAD BEEN IN SERVICE FOR 15 YEARS IN THE AIR OPERATOR FOR THE REACTOR FEEDWATER LOW FLOW CONTROL VALVE AND AIR LEAKAGE FROM THE OPERATOR STEM PACKING GLAND. THE FAILURE OF THE VALVE TO STROKE FULL OPEN RESULTED IN AN INABILITY TO SUPPLY SUFFICIENT FEEDWATER FLOW TO THE REACTOR. THIS RESULTED IN DECREASING REACTOR WATER LEVEL AND THE NECESSITY TO USE THE REACTOR FEED PUMP (RFP) DISCHARGE VALVE TO CONTROL WATER LEVEL. AFTER SEVERAL CONTROLLED JOGS IN THE OPEN DIRECTION OF THE RFP DISCHARGE VALVE, THE INCREASE IN WATER FLOW RESULTED IN A HIGH NEUTRON FLUX SCRAM OF THE REACTOR DUE TO EXCEEDING THE 15 PERCENT POWER LIMIT WHILE THE MODE SWITCH WAS IN THE START-UP MODE POSITION. CORRECTIVE ACTIONS INCLUDED REPAIRING THE REACTOR FEEDWATER LOW FLOW CONTROL VALVE OPERATOR, REVISING THE START-UP PROCEDURE TO VERIFY FULL STROKE CAPABILITY OF THE VALVE, AND ADDING A CAUTION LIMITING THE ACCEPTABLE OPEN DEMAND SIGNAL FOR THE VALVE TO 70 PERCENT. THE PLANT WAS RESTORED TO SERVICE AT 0658 ON 12/17/90. RELATED LER: 90-026.

[72] FITZPATRICK DOCKET 50-333 LER 90-028
PARTIAL ISOLATION OF REACTOR BUILDING VENTILATION SYSTEM DUE TO PERSONNEL ERROR CONTRIBUTED TO BY STEP LOCATION IN PROCEDURE.
EVENT DATE: 122790 REPORT DATE: 012891 NSSS: GE TYPE: BWR

(NSIC 220933) THE PLANT WAS OPERATING AT FULL POWER ON DECEMBER 27, 1990. THE REACTOR BUILDING VENTILATION EXHAUST RADIATION MONITOR (IL) WAS BEING REMOVED FROM SERVICE AS A PREREQUISITE TO PREVENTIVE MAINTENANCE ON THE MONITOR SAMPLE PUMP. THE TECHNICIAN TOUCHED A VOLT METER PROBE TO THE MONITOR HIGH VOLTAGE CIRCUIT TEST CONNECTION TO DETERMINE THE VOLTAGE IN ACCORDANCE WITH PROCEDURE. THE MONITOR SPIKED HIGH RESULTING IN ISOLATION OF THE REACTOR BUILDING VENTILATION SYSTEM (VA) AND PRIMARY CONTAINMENT ATMOSPHERE SAMPLING SYSTEM (BB) AND START OF THE STANDBY GAS TREATMENT SYSTEM (BH) AT 0830. THE SYSTEMS WERE

RESET AT 0835. THE RADIATION MONITOR AND VOLT METER WERE CHECKED AND DETERMINED TO BE OPERATING PROPERLY. VOLTAGE SPIKES ARE NOT AN UNUSUAL OCCURRENCE WHEN MAINTAINING RADIATION MONITOR ELECTRONICS. CORRECTIVE ACTION WILL BE TO GENERATE A SPECIFIC PROCEDURE FOR THIS MAINTENANCE EVOLUTION. THIS PROCEDURE WILL DESCRIBE THE POTENTIAL FOR SPURIOUS ISOLATION AS THE RADIATION MONITOR IS REMOVED AND RETURNED TO SERVICE.

[73] FITZPATRICK DOCKET 50-333 LER 91-001
SPURIOUS PARTIAL ACTIVATION OF PRIMARY CONTAINMENT ISOLATION SYSTEM DUE TO ELECTRICAL NOISE WHILE CALIBRATING ADJACENT INSTRUMENT.
EVENT DATE: 010991 REPORT DATE: 020791 NSSS: GE TYPE: BWR

(NSIC 220944) A FALSE PRIMARY CONTAINMENT HIGH RADIATION ISOLATION (JM) SIGNAL OCCURRED AT 1105 ON 1/9/91 WHILE THE PLANT WAS OPERATING AT FULL POWER. THE SIX PRIMARY CONTAINMENT ISOLATION VALVES WHICH ARE ACTIVATED BY THIS SIGNAL WERE ALREADY IN THE CLOSED POSITION. THE TRIP WAS RESET AT 1109. REDUNDANT INSTRUMENTATION CONFIRMED THAT RADIATION LEVELS WERE NORMAL. AN ELECTRICAL NOISE SIGNAL WAS GENERATED WHEN CONTACTING A VOLT METER PROBE WITH THE HIGH VOLTAGE TEST CONNECTION DURING A CALIBRATION OF AN UNRELATED INSTRUMENT. THE UNRELATED INSTRUMENT WAS PHYSICALLY LOCATED ABOUT ONE FOOT ABOVE THE ELECTRONIC DRAWER FOR THE PRIMARY CONTAINMENT HIGH RADIATION MONITOR WHICH GENERATED THE FALSE SIGNAL. CALIBRATION PROCEDURES FOR RADIATION MONITORING INSTRUMENTS WILL BE REVISED TO NOTE THAT THE POSSIBILITY OF SPURIOUS TRIP SIGNAL BEING TRANSMITTED TO ADJACENT INSTRUMENTS EXISTS. DURING THE 1991 REFUELING OUTAGE THE PANEL CONTAINING THE RADIATION MONITORS WILL BE THOROUGHLY INSPECTED AND ANALYZED TO DETERMINE IF IMPROVEMENTS IN NOISE SUPPRESSION ARE POSSIBLE. RELATED LER: 90-028.

[74] FT. CALHOUN 1 DOCKET 50-285 LER 90-022 REV 02
UPDATE ON NONFUNCTIONAL FIRE BARRIER PENETRATIONS DUE TO LACK OF DOCUMENTATION.
EVENT DATE: 090790 REPORT DATE: 020691 NSSS: CE TYPE: PWR

(NSIC 220959) ON SEPTEMBER 7, 1990, APPROXIMATELY 460 FIRE BARRIER PENETRATION SEALS, 60 FIRE DAMPERS AND 6 FIRE DOORS ASSOCIATED WITH 25 FIRE AREAS WERE DECLARED NONFUNCTIONAL DUE TO EITHER LACK OF DOCUMENTATION TO QUALIFY NON-VERIFIABLE PENETRATION CRITICAL PARAMETERS, OR PENETRATION "AS BUILT" CONFIGURATIONS WHICH DID NOT MATCH TYPICAL CONFIGURATIONS PREVIOUSLY QUALIFIED BY FIRE TESTS. THESE NONFUNCTIONAL PENETRATIONS WERE DISCOVERED THROUGH A SPECIAL DESIGN BASIS VERIFICATION WALKDOWN PROMPTED BY NRC INFORMATION NOTICE 88-04. AS REQUIRED BY TECHNICAL SPECIFICATIONS, THE APPROPRIATE COMPENSATORY MEASURES WERE IMPLEMENTED. ON NOVEMBER 27, 1990 AND ON JANUARY 2, 1991, ADDITIONAL BARRIER PENETRATIONS WERE DETERMINED TO BE NONFUNCTIONAL AND THE REQUIRED COMPENSATORY MEASURES WERE ESTABLISHED. THE COMPENSATORY MEASURES WILL REMAIN IN PLACE UNTIL THE AFFECTED FIRE BARRIER PENETRATIONS ARE RESTORED TO FUNCTIONAL STATUS THROUGH ENGINEERING EVALUATION, REPAIR, OR REPLACEMENT. THIS REPORT IS SUBMITTED PURSUANT TO TECHNICAL SPECIFICATION 2.19(7) BECAUSE MOST OF THE NONFUNCTIONAL FIRE BARRIER PENETRATIONS WERE NOT RESTORED TO FUNCTIONAL STATUS WITHIN 7 DAYS. IT IS BEING SUBMITTED ALSO AS A VOLUNTARY LER DUE TO POTENTIAL REGULATORY AND INDUSTRY INTEREST.

[75] FT. CALHOUN 1 DOCKET 50-285 LER 90-028
LEAKAGE THROUGH CONTROL ELEMENT DRIVE MECHANISM HOUSING.
EVENT DATE: 121490 REPORT DATE: 011491 NSSS: CE TYPE: PWR
VENDOR: COMBUSTION ENGINEERING, INC.

(NSIC 220703) ON DECEMBER 14, 1990, AN INVESTIGATION OF UNKNOWN REACTOR COOLANT SYSTEM (RCS) 1LEAKAGE IDENTIFIED THE SOURCE AS INSTALLED SPARE CONTROL ELEMENT DRIVE MECHANISM (CEDM) HOUSING NUMBER 9. SUBSEQUENT REMOVAL AND INSPECTION IDENTIFIED TWO AXIAL CRACKS IN AN INSIDE DIAMETER WELD OVERLAY REGION APPROXIMATELY TWO FEET FROM THE BOTTOM FLANGE OF THE HOUSING. SIMILAR INSTALLED SPARE CEDM HOUSING NUMBER 13 WAS ALSO REMOVED AND INSPECTED, REVEALING TWO SIMILAR CRACKS IN THE WELD OVERLAY REGION. THE CAUSE OF THIS EVENT WAS LACK OF VENTING, WHICH CREATED CONDITIONS CONDUCIVE TO TRANSGRANULAR STRESS CORROSION CRACKING (TGSCC) IN THE SPARE HOUSINGS. THIS REPORT IS SUBMITTED VOLUNTARILY DUE

TO POTENTIAL NRC AND INDUSTRY INTEREST. BLANK FLANGES WERE INSTALLED IN PLACE OF CEDM HOUSINGS 9 AND 13. A PROCEDURE CHANGE HAS BEEN IMPLEMENTED TO ASSURE COMPLETE VENTING OF TWO OTHER SIMILAR HOUSINGS. OTHER APPROPRIATE CEDM HOUSINGS HAVE BEEN EXAMINED WITH NO CRACKS FOUND. AN ENHANCED RCS LEAKAGE MONITORING PROGRAM HAS BEEN IMPLEMENTED.

[76] FT. CALHOUN 1 DOCKET 50-285 LER 91-001
CONTAINMENT TENDON SURVEILLANCE DETERMINED NOT IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS.
EVENT DATE: 010991 REPORT DATE: 020891 NSSS: CE TYPE: PWR

(NSIC 220937) ON 1/9/91, WITH THE REACTOR IN MODE 1 AT APPROXIMATELY 26% POWER, IT WAS DETERMINED THAT PRIOR TO THE PERFORMANCE OF THE CONTAINMENT TENDON SURVEILLANCES IN 1981 AND 1985, THE TEST PROCEDURES WERE CHANGED TO REFLECT THE GUIDANCE OF REVISION 3 TO REGULATORY GUIDE 1.35. TECH SPEC 3.5(7)A, WHICH INCORPORATES THE GUIDANCE IN REVISION 1 OF REGULATORY GUIDE 1.35, IS MORE RESTRICTIVE RELATIVE TO DETENSIONING OF TENDONS. BY USING GUIDANCE NOT REFLECTED IN THE TECH SPECS, BOTH THESE TESTS FAILED TO COMPLETELY DETENSION ALL THE REQUIRED TENDONS. THIS VIOLATION OF THE TECH SPEC IS REPORTABLE PURSUANT TO 10CFR 50.73(A)(2)(I)(B). THE ROOT CAUSE FOR THIS EVENT WAS INADEQUATE ADMINISTRATIVE CONTROL OF THE TENDON SURVEILLANCE PROGRAM, WITH INAPPROPRIATE ACTIONS BY PERSONNEL AS A CONTRIBUTING FACTOR. CORRECTIVE ACTIONS INCLUDE: ENHANCEMENT OF STATION ENGINEERING SUPPORT, INCLUDING IMPLEMENTATION OF A CONTAINMENT TESTING PROGRAM PLAN; IMPROVEMENTS IN THE SAFETY EVALUATION PROCESS FOR PROCEDURE CHANGES; AND SUBMITTAL OF A PROPOSED LICENSE AMENDMENT TO INCORPORATE APPROPRIATE REGULATORY GUIDE 1.35, REVISION 3 GUIDANCE INTO THE TECHNICAL SPECIFICATIONS.

[77] GINNA DOCKET 50-244 LER 90-018
DROPPED CONTROL ROD DURING ROD CONTROL EXERCISE CAUSES AUTOMATIC ACTUATION OF RADIATION PROTECTION SYSTEM.
EVENT DATE: 122090 REPORT DATE: 012191 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 220862) ON DECEMBER 20, 1990 AT 1321 EST, WITH THE REACTOR AT APPROXIMATELY 22% FULL POWER, A TURBINE RUNBACK OCCURRED DUE TO A DROPPED CONTROL ROD. THE CONTROL ROOM OPERATORS PERFORMED THE APPROPRIATE ACTIONS OF ABNORMAL PROCEDURES AP-TURB.2 (AUTOMATIC TURBINE RUNBACK) AND AP-RCC.2 (RCC/RPI MALFUNCTION) TO STABILIZE THE PLANT. THE CONTROL ROOM OPERATORS MANUALLY TRIPPED THE TURBINE TO PREVENT REVERSE POWER TO THE GENERATOR. SUBSEQUENTLY THE REACTOR WAS TAKEN SUBCRITICAL TO ACCOMMODATE ANTICIPATED CORRECTIVE MAINTENANCE ACTIVITIES. THE MAIN STEAM ISOLATION VALVES WERE THEN CLOSED TO LIMIT A PLANT COOLDOWN. THE UNDERLYING CAUSE OF THE EVENT WAS ATTRIBUTED TO DEGRADED POWER BRIDGE THYRISTOR SUPPRESSION FILTER CAPACITORS IN THE CIRCUIT SUPPLYING POWER TO THE STATIONARY, MOVABLE AND LIFT COILS OF THE AFFECTED CONTROL ROD. CORRECTIVE ACTION WAS TO REPLACE THE DEGRADED CAPACITORS IN THE POWER CABINET SUPPLYING POWER TO THE AFFECTED ROD.

[78] GINNA DOCKET 50-244 LER 90-019
LO-LO LEVEL IN "A" STEAM GENERATOR, DURING PLANT STARTUP, DUE TO MAIN FEEDWATER PUMP TRIP, CAUSES A REACTOR TRIP.
EVENT DATE: 122190 REPORT DATE: 012191 NSSS: WE TYPE: PWR

(NSIC 220963) ON DECEMBER 21, 1990 AT 1237 EST, WITH THE REACTOR AT APPROXIMATELY 16% FULL POWER, A REACTOR TRIP OCCURRED DUE TO LO LO LEVEL (< OR = 17%) IN THE "A" STEAM GENERATOR. THE CONTROL ROOM OPERATORS IMMEDIATELY PERFORMED THE APPROPRIATE ACTIONS OF E-0 (REACTOR TRIP OR SAFETY INJECTION) AND ES-0.1 (REACTOR TRIP RESPONSE). BOTH MAIN STEAM ISOLATION VALVES WERE SUBSEQUENTLY CLOSED TO LIMIT AN RCS COOLDOWN AND THE PLANT WAS STABILIZED IN HOT SHUTDOWN. THE INTERMEDIATE CAUSE OF THE EVENT WAS THE "A" MAIN FEEDWATER PUMP TRIPPING DUE TO FEED PUMP SEAL WATER LOW DIFFERENTIAL PRESSURE CAUSED BY A CONDENSATE LOW HEADER PRESSURE TRANSIENT. THE UNDERLYING CAUSE OF THE EVENT WAS A DEFICIENCY IN THE OPERATING PHILOSOPHY FOR THE PROPER NUMBER OF CONDENSATE PUMPS RUNNING DURING LOW

POWER CONDITIONS. CORRECTIVE ACTION WILL BE TO CHANGE THE APPROPRIATE PROCEDURES INVOLVED, TO BE CONSISTENT WITH THE NEW OPERATING PHILOSOPHY.

[79] GRAND GULF 1 DOCKET 50-416 LER 90-029
REACTOR SCRAM DUE TO REACTOR FEEDWATER PUMP "A" TRIP.
EVENT DATE: 121890 REPORT DATE: 011791 NSSS: GE TYPE: BWR

(NSIC 220827) ON DECEMBER 18, 1990, DURING A CONTROLLED SHUTDOWN, A REACTOR PROTECTION SYSTEM ACTUATION OCCURRED RESULTING IN AN AUTOMATIC PLANT SHUTDOWN. THE ACTUATION OCCURRED DUE TO LOW REACTOR WATER LEVEL WHICH WAS CAUSED BY A REACTOR FEEDWATER PUMP TRIP. THE HIGH DISCHARGE PRESSURE TRIP OF THE FEEDWATER PUMP WAS A RESULT OF THE INTERACTION BETWEEN THE STARTUP LEVEL CONTROL SYSTEM AND THE MASTER FEEDWATER CONTROL SYSTEM. THE INVESTIGATION, WHICH FOLLOWED THE SCRAM, IDENTIFIED THE AIR SUPPLY VALVE FOR THE STARTUP LEVEL CONTROL VALVE NOT BEING FULLY OPEN AND THE 'A' ELECTRIC AUTOMATIC POSITIONER'S HIGH SPEED STOP LIMIT SWITCH FOR THE REACTOR FEEDPUMP TURBINE NOT BEING PROPERLY SET. THE COMBINED EFFECT OF THESE PROBLEMS ARE THE MOST PROBABLE CAUSE OF THE EVENT. ALL SAFETY SYSTEMS PERFORMED AS DESIGNED DURING THE TRANSIENT. THE MINIMUM WATER LEVEL REACHED WAS -25 INCHES AS INDICATED ON THE WIDE RANGE LEVEL INSTRUMENTATION. THE MINIMUM LEVEL WAS APPROXIMATELY 142 INCHES ABOVE THE TOP OF ACTIVE FUEL.

[80] GRAND GULF 1 DOCKET 50-416 LER 91-001
DELINQUENT LCO ACTION FOR DIESEL GENERATOR 11 DUE TO PERSONNEL ERROR.
EVENT DATE: 011491 REPORT DATE: 021391 NSSS: GE TYPE: BWR

(NSIC 221006) A TECHNICAL SPECIFICATION LIMITING CONDITION OF OPERATION (LCO) ACTION WAS NOT SATISFIED AFTER THE DIVISION 1 DIESEL GENERATOR (DG 11) WAS REMOVED FROM STANDY SERVICE TO PERFORM PREVENTIVE MAINTENANCE. A TECHNICAL SPECIFICATION LCO REPORT HAD BEEN GENERATED ON THE PREVIOUS SHIFT WHEN DG 11 WAS MADE OPERABLE. THE REPLACEMENT SHIFT SUPERINTENDENT AND SHIFT SUPERVISOR LCO REPORT HAD BEEN GENERATED ON THE PREVIOUS SHIFT WHEN DG 11 WAS MADE INOPERABLE. THE REPLACEMENT SHIFT SUPERINTENDENT AND SHIFT SUPERVISOR WERE MADE AWARE OF REQUIRED ACTIONS. THE MISSED SURVEILLANCE WAS DUE TO PERSONNEL ERROR BY PLANT LICENSED OPERATORS. INATTENTION TO DETAIL WAS DETERMINED TO BE THE CAUSE OF THIS EVENT. STATION PROCEDURE 06-OP-1000-D-0001 WAS AMENDED TO INCORPORATE A SURVEILLANCE REQUIREMENT TRIGGER FOR THE DIESEL GENERATORS. THE PROCEDURE PROVIDES A METHOD OF COMPLETING AND TRACKING SURVEILLANCES REQUIRED ON A DAILY OR MORE FREQUENT SCHEDULE. THE PROCEDURE CHANGE SHOULD PREVENT RECURRENCE OF SIMILAR DELINQUENT DIESEL GENERATOR LCO ACTIONS. THE LATE VERIFICATION OF OFFSITE A.C. POWER SOURCES DID NOT RESULT IN A COMPROMISE TO PLANT SAFETY. ALL EMERGENCY CORE COOLING SYSTEMS WERE OPERABLE AND AVAILABLE TO PERFORM THE REQUIRED SAFETY FUNCTIONS.

[81] HATCH 1 DOCKET 50-321 LER 90-024
OFFGAS SAMPLES NOT COLLECTED AND ANALYZED AS REQUIRED BY TECHNICAL SPECIFICATIONS.
EVENT DATE: 122890 REPORT DATE: 012791 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: HATCH 2 (BWR)

(NSIC 220871) ON 12/28/90 AT APPROXIMATELY 0800 CST, UNIT 1 AND UNIT 2 WERE IN THE RUN MODE AT AN APPROXIMATE POWER LEVEL OF 2436 CMWT (APPROXIMATELY 100% RATED THERMAL POWER). AT THAT TIME, IT WAS DETERMINED THE REQUIREMENTS OF UNIT 1 TECHNICAL SPECIFICATIONS SECTION 4.15.2.7.2.B AND UNIT 2 TECHNICAL SPECIFICATIONS SECTION 4.11.2.7.2.B HAD NOT BEEN MET. SPECIFICALLY, REPRESENTATIVE SAMPLES OF GASES AT THE PRETREATMENT MONITORING STATION WERE NOT ALWAYS COLLECTED AND ANALYZED WITHIN 4 HOURS OF AN INCREASE OF GREATER THAN 50% IN A NOMINAL STEADY-STATE FISSION GAS RELEASE VALUE AS REQUIRED BY THE AFOREMENTIONED TECHNICAL SPECIFICATIONS. IT IS HIGHLY UNLIKELY ANY TECHNICAL SPECIFICATIONS OFFGAS PRETREATMENT ACTIVITY LIMITS WERE EXCEEDED. THE FISSION GAS RELEASE VALUES WERE A SMALL FRACTION OF THE LIMIT, AND SAMPLES WERE ROUTINELY COLLECTED AND ANALYZED TO CONFIRM THE CALIBRATION OF THE PRETREATMENT MONITORS. THE CAUSE OF THIS EVENT WAS A MISINTERPRETATION OF THE TECHNICAL SPECIFICATIONS REQUIREMENTS. CONTRIBUTING TO THE EVENT WERE AMBIGUOUS TECHNICAL SPECIFICATIONS.

CORRECTIVE ACTIONS INCLUDE PROVIDING THE CORRECT INTERPRETATION OF THE REQUIREMENTS TO CHEMISTRY PERSONNEL, ISSUING A TECHNICAL SPECIFICATIONS CLARIFICATIONS AND REVIEWING CHEMISTRY PROCEDURES.

[82]   HATCH 2   DOCKET 50-366   LER 89-007 REV 01
UPDATE ON SAFETY RELIEF VALVES EXPERIENCE SETPOINT DRIFT DUE TO CORROSION INDUCED BONDING.
EVENT DATE: 092689   REPORT DATE: 020791   NSSS: GE   TYPE: BWR
VENDOR: TARGET ROCK CORP.

(NSIC 220951) REVISION 1 TO THIS LER PROVIDES UPDATED INFORMATION REGARDING THE CURRENT INDUSTRY UNDERSTANDING OF THE ROOT CAUSE OF THE SETPOINT DRIFT EXHIBITED BY SOME OF THE PRESSURE VESSEL SAFETY RELIEF VALVES (SRVS) AND SUMMARIZES THE CURRENT BOILING WATER REACTOR OWNERS' GROUP (BWROG) ACTIVITIES TO REDUCE SETPOINT DRIFT. ON 9/26/89, AT APPROXIMATELY 1200 EDT, UNIT 2 WAS IN THE REFUEL MODE AT AN APPROXIMATE POWER LEVEL OF 0 MWT (APPROX. 0% OF RATED THERMAL POWER). AT THAT TIME PLANT ENGINEERING PERSONNEL RECEIVED WRITTEN NOTIFICATION OF THE RESULTS OF OFF-SITE TESTING OF THE SRVS. OF THE ELEVEN SRVS, FOUR HAD EXHIBITED DRIFT IN THE MECHANICAL LIFT SETPOINTS IN EXCESS OF THE +/- 3% TOLERANCE SPECIFIED BY IN-SERVICE TESTING (IST) REQUIREMENTS. THIS VOLUNTARY REPORT IS BEING SUBMITTED DUE TO THE POTENTIAL INDUSTRY INTEREST IN THIS EVENT IN VIEW OF THE REFERENCED BWROG ACTIVITIES. THE EXPERIENCED SETPOINT DRIFT WAS WELL WITHIN THE ANALYTICAL LIMITS EXISTING FOR REACTOR VESSEL OVER-PRESSURE PROTECTION. THE ROOT CAUSE OF THE EVENT IS CORROSION-INDUCED BONDING OF THE SURFACE BETWEEN THE PILOT VALVE DISC AND SEAT. THE EXPERIENCED SETPOINT DRIFT IN THIS EVENT IS CONSISTENT WITH CURRENT INDUSTRY DATA DEMONSTRATING THAT BOTH PH13-8MO DISCS AND STELLITE DISCS CAN OCCASIONALLY FORM CORROSION BONDS WITH THE STELLITE SEAT RESULTING IN SETPOINT DRIFT.

[83]   HATCH 2   DOCKET 50-366   LER 90-013
OFFGAS RADIATION MONITORS INOPERABLE DUE TO INCORRECT VALVE LINEUP.
EVENT DATE: 122090   REPORT DATE: 011891   NSSS: GE   TYPE: BWR

(NSIC 220823) ON 12/20/90, AT APPROX. 0950 CST, UNIT 2 WAS IN THE RUN MODE AT APPROX. 2436 CMWT (APPROX. 100% RATED THERMAL POWER). WHILE PREPARING TO OBTAIN AN OFFGAS SAMPLE, A NONLICENSED CHEMISTRY TECHNICIAN DISCOVERED 3 VALVES MISALIGNED IN THE OFFGAS POST-TREATMENT RADIATION MONITORING SYSTEM. BECAUSE OF THE VALVE MISALIGNMENT, OFFGAS POST-TREATMENT RADIATION MONITORS 2D11-K615A AND B WERE NO LONGER MONITORING OFF GAS SYSTEM EFFLUENT AND WERE, THEREFORE, INOPERABLE. UNIT 2 TECH SPECS SECTION 3.3.6.1 REQUIRES THE MONITORS TO BE OPERABLE IN THE RUN MODE. THE TECHNICIAN, IN OBTAINING THE SAMPLE PER PROCEDURE 64CH-SAM-001-OS, "OFFGAS SAMPLING," IMMEDIATELY CONFIGURED THE SYSTEM SUCH THAT OFFGAS SAMPLE FLOW WAS RESTORED TO THE MONITORS. AT THIS TIME, THE MONITORS WERE OPERABLE AND THE PLANT WAS IN COMPLIANCE WITH THE TECH SPECS. A DEFICIENCY WAS INITIATED AND THE SHIFT SUPERVISOR WAS NOTIFIED OF THE AS-FOUND VALVE MISALIGNMENT. IT IS NOTED THAT OFFGAS POST-TREATMENT ACTIVITY REMAINED WELL BELOW TECH SPECS LIMITS DURING THE EVENT. THE CAUSE OF THE EVENT IS INCONCLUSIVE. HOWEVER, THE MOST LIKELY CAUSE WAS COGNITIVE PERSONNEL ERROR IN THAT NONLICENSED PERSONNEL FAILED TO RESTORE THE VALVE LINEUP DURING PRIOR SAMPLING. A CONTRIBUTING FACTOR TO THE EVENT WAS A LESS THAN OPTIMAL SAMPLING PROCEDURE IN THAT IT DID NOT REQUIRE APPROPRIATE DOCUMENTATION AND VERIFICATION OF AS-LEFT VALVE POSITIONS.

[84]   HOPE CREEK 1   DOCKET 50-354   LER 90-012
PREPLANNED ENTRY INTO TECHNICAL SPECIFICATION 3.0.3 TO REPLACE SUSPECT ROSEMOUNT TRANSMITTER IN ACCORDANCE WITH NRC BULLETIN 90-01.
EVENT DATE: 081190   REPORT DATE: 091290   NSSS: GE   TYPE: BWR
VENDOR: ROSEMOUNT ENGINEERING COMPANY

(NSIC 220907) ON 8/11/90 AT 0905, HOPE CREEK MADE A PREPLANNED VOLUNTARY ENTRY INTO TECH SPEC 3.0.3 TO REPLACE A SUSPECT ROSEMOUNT MODEL 1153 LEVEL TRANSMITTER PER PSE&G'S RESPONSE TO NRC BULLETIN 90-01. PRIOR TO ENTRY INTO TECH SPEC 3.0.3, AN ENGINEERING REVIEW WAS CONDUCTED TO DETERMINE THE MOST FEASIBLE METHOD OF

AFFECTING REPLACEMENT OF THE SUBJECT TRANSMITTER. THE METHOD CHOSEN REQUIRED VALVING OUT A VESSEL LEVEL REFERENCE LEG DURING REPLACEMENT OF THE TRANSMITTER. THIS METHOD ISOLATED TRANSMITTERS AFFECTING VESSEL LEVEL INPUTS TO CHANNEL "B" REACTOR PROTECTION SYSTEM AND ENGINEERED SAFETY FEATURES (ESF) SYSTEMS, CHANNEL "D" AUTOMATIC DEPRESSURIZATION SYSTEM, "B" LOOP OF THE CORE SPRAY SYSTEM, AND "B" LOOP OF THE LOW PRESSURE COOLANT INJECTION SYSTEM. PRIOR TO VALVING OUT THE REFERENCE LEG FOR REPLACEMENT OF THE SUBJECT TRANSMITTER, ALL TRANSMITTERS ON THE REFERENCE LEG WERE PLACED IN A "TEST" CONDITION. THE ENTRY INTO 3.0.3 WAS REQUIRED BECAUSE INSTRUMENTS FOR THE ABOVED DESCRIBED ECCS SYSTEM CHANNELS/LOOPS WERE CONCURRENTLY INOPERABLE. IT SHOULD BE NOTED THAT ALTERNATE CHANNELS/LOOPS WERE AVAILABLE FOR THE ABOVE SYSTEMS. TECH SPEC 3.0.3 WAS EXITED AT 0942 WHEN THE INSTRUMENT RACK WAS RETURNED TO SERVICE.

[85] HOPE CREEK 1 DOCKET 50-354 LER 90-013
VIOLATION OF TECHNICAL SPECIFICATION 4.0.5 DUE TO INADEQUATE REVIEW OF WORKORDER SCOPE BY SRO LICENSED SUPERVISORY PERSONNEL.
EVENT DATE: 081390 REPORT DATE: 091290 NSSS: GE TYPE: BWR

(NSIC 220908) ON 8/13/90 AT 1045, THE STATION INSERVICE TEST ENGINEER ADVISED THE WORK CONTROL CENTER NUCLEAR SHIFT SUPERVISOR (NSS, SRO LICENSED) THAT A RE-BASELINE TEST, AS REQUIRED BY ASME SECTION XI, HAD NOT BEEN PERFORMED ON THE "B" SERVICE WATER SYSTEM SPRAY WASH PUMP FOLLOWING CORRECTIVE MAINTENANCE IN JULY, 1990. THIS CIRCUMSTANCE WAS DETERMINED TO BE A VIOLATION OF THE REQUIREMENTS OF TECH SPEC 4.0.5. AS SUCH, THE SUBJECT PUMP WAS DECLARED INOPERABLE. AT THE DIRECTION OF THE NSS, A RE-BASELINE TEST OF THE PUMP WAS IMMEDIATELY COMPLETED. FOLLOWUP INVESTIGATION DETERMINED THE ROOT CAUSE OF THIS EVENT TO BE AN INADEQUATE SUPERVISORY REVIEW OF THE JULY, 1990 CORRECTIVE MAINTENANCE WORKORDER. THE NSS WHO WAS RESPONSIBLE FOR CLOSING OUT THE WORKORDER INCORRECTLY DELETED THE REQUIREMENT FOR A RE-BASELINE RUN OF THE PUMP AS PART OF THE RETEST FOLLOWING WORK COMPLETION BECAUSE, WHEN REVIEWING THE WORKORDER, HE DID NOT RECOGNIZE THAT THE PUMP HAD BEEN DISASSEMBLED DURING MAINTENANCE. CORRECTIVE ACTIONS NICLUDED COUNSELLING FOR THE NSS, AND IMMEDIATELY COMPLETING THE REQUIRED TESTING.

[86] HOPE CREEK 1 DOCKET 50-354 LER 90-025 REV 01
UPDATE ON RECIRCULATION SYSTEM INSTRUMENT LINE CRACK IN WELDED JOINT DUE TO VIBRATION INDUCED FATIGUE AND A NONSYMMETRICALLY INSTALLED PIPE.
EVENT DATE: 110490 REPORT DATE: 011691 NSSS: GE TYPE: BWR

(NSIC 220989) ON 11/4/90 AT 1515, WITH THE REACTOR IN OPERATIONAL CONDITION 3 (HOT SHUTDOWN), AND WHILE CONDUCTING AN INVESTIGATION TO DETERMINE THE SOURCE OF UNIDENTIFIED DRYWELL LEAKAGE, A LEAK WAS DISCOVERED AT A WELD ON A REACTOR RECIRCULATION INSTRUMENT LINE. SUBSEQUENT TO THE DISCOVERY OF THE LEAK, THE SUBJECT INSTRUMENT LINE WAS CUT OUT AND REPLACED WITH A NEW SECTION OF PIPING. THE FAILED LINE WAS FORWARDED TO AN INDEPENDENT LABORATORY FOR FAILURE ANALYSIS OF THE JOINT WELD. ANALYSIS RESULTS HAVE BEEN RECEIVED WHICH INDICATE THAT THE ROOT CAUSE OF THE LEAKING WELD WAS VIBRATION INDUCED FATIGUE. STRESS CONCENTRATION AT THE WELD ROOT COUPLED WITH A NONSYMMETRICALLY POSITIONED PIPE CONTRIBUTED TO THE INITIATION OF THE CRACK. OTHER SIMILAR PIPING AND CONNECTIONS IN THE REACTOR RECIRCULATION SYSTEM WERE VISUALLY INSPECTED AND A SAMPLING OF SIMILAR WELDS WERE NON-DESTRUCTIVELY EXAMINED TO ENSURE PRESSURE BOUNDARY INTEGRITY. NO FURTHER PROBLEMS WERE NOTED. ONE DESIGN CHANGE WAS IMPLEMENTED TO INSTALL INSTRUMENTATION TO MONITOR VIBRATION LEVELS OF THE REACTOR RECIRCULATION INSTRUMENT LINES AND ANOTHER DESIGN CHANGE WAS IMPLEMENTED TO REDUCE THE POTENTIAL FOR FATIGUE INDUCED FAILURE DUE TO VIBRATION BY ADDING PIPE SUPPORTS AND MODIFYING SOME EXISTING SUPPORTS. THIS EVENT WILL BE DISCUSSED WITH THE APPROPRIATE PERSONNEL.

[87] HOPE CREEK 1 DOCKET 50-354 LER 90-033
ENGINEERED SAFETY FEATURES ACTUATION DUE TO TRIPPING OF REACTOR PROTECTION SYSTEM CHANNEL "A" ELECTRICAL PROTECTION ASSEMBLY DUE TO EQUIPMENT MALFUNCTION.
EVENT DATE: 121990 REPORT DATE: 011691 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 220835) ON 12/19/90 AT 0710, THE CONTROL ROOM RECEIVED INDICATION OF A HALF SCRAM AND ISOLATION OF THE INBOARD REACTOR WATER CLEANUP (RWCU) ISOLATION VALVE. THE ABOVE ACTIONS OCCURRED AS A RESULT OF A LOSS OF POWER TO THE CHANNEL "A" REACTOR PROTECTION SYSTEM (RPS) ELECTRICAL BUS WHEN THE ALTERNATE POWER SUPPLY ELECTRICAL PROTECTION ASSEMBLY (EPA) EXPERIENCED A SPURIOUS TRIP (THE NORMAL POWER "A" RPS MOTOR GENERATOR SET WAS OUT OF SERVICE FOR MAINTENANCE). THE CHANNEL "A" RPS BUS WAS RE-POWERED FROM ITS NORMAL POWER SOURCE AFTER COMPLETION OF MAINTENANCE ON THE ASSOCIATED MOTOR GENERATOR SET, AND THE HALF SCRAM AND RWCU ISOLATION WERE RESET. FOLLOWUP TROUBLESHOOTING BY THE MAINTENANCE DEPARTMENT DETERMINED THAT A FAULTY LOGIC CARD CAUSED THE TRIP OF THE EPA. IMMEDIATE CORRECTIVE ACTIONS CONSISTED OF REPLACING THE FAULTY LOGIC CARD. LONG TERM CORRECTIVE ACTIONS CONSIST OF COMPLETING A DESIGN CHANGE TO ENHANCE LOGIC CARD PERFORMANCE WHEN LOGIC CARD UPGRADE KITS ARE RECEIVED FROM THE VENDOR.

[88] HOPE CREEK 1 DOCKET 50-354 LER 90-034
AUTOMATIC START OF THE FILTRATION, RECIRCULATION, AND VENTILATION SYSTEM DUE TO DESIGN INADEQUACY IN CONTROL CIRCUITRY.
EVENT DATE: 122490 REPORT DATE: 011791 NSSS: GE TYPE: BWR

(NSIC 220821) ON 12/24/90 AT 1830, THE "F" FILTRATION, RECIRCULATION, AND VENTILATION SYSTEM (FRVS) RECIRCULATION FAN WAS DISCOVERED RUNNING DURING SHIFT TURNOVER. A REVIEW OF STRIP CHARTS INDICATED THAT THE FAN AUTOMATICALLY STARTED AT ABOUT 1545. ON 1/7/91 AT 2215, CONTROL ROOM OPERATORS OBSERVE THE "E" FRVS RECIRCULATION FAN AUTO-START FOR NO APPARENT REASON. IN BOTH CASES, AFTER VERIFYING THAT THE FAN WAS NOT RUNNING FOR TESTING OR OTHER REASONS, THE NUCLEAR SHIFT SUPERVISOR (NSS, SRO LICENSED) DIRECTED THAT THE FANS BE STOPPED AND RETURNED TO A NORMAL (STANDBY) STATUS. SUBESEQUENT INVESTIGATION DETERMINED THAT THE CAUSE OF THE "F" AND "E" FRVS RECIRCLUATION FAN STARTS WAS AN APPARENTLY SPURIOUS LOW FLOW SIGN FROM THE RESPECTIVE FRVS VENT FANS. THE PRIMARY CAUSE OF THIS AND SIMILAR RECENT SPURIOUS FRVS AUTO-STARTS IS THE LESS THAN ADEQUATE DESIGN OF THE "E" AND "F" FRVS FAN AUTO-INITIATION CIRCUITRY. THE INHERENT SENSITIVITY OF FRVS FLOW INSTRUMENTATION RENDERS THE CIRCUITRY SUSCEPTIBLE TO SPURIOUS FAN STARTS UNDER NORMAL OPERATING CONDITIONS, WITH ONLY MINOR FLUCTUATIONS IN SYSTEM OPERATING PARAMETERS. ADDITIONALLY, IT WAS DETERMINED THAT THE CONDENSATION ACCUMULATION IN FRVS INSTRUMENT TUBING MAY CONTRIBUTE TO THE FLUCTUATIONS.

[89] HOPE CREEK 1 DOCKET 50-354 LER 90-035
SERVICE WATER PIPING THROUGH WALL FLAW DUE TO PIPE CORROSION.
EVENT DATE: 122790 REPORT DATE: 012391 NSSS: GE TYPE: BWR

(NSIC 220866) ON 12/27/90 AT 1515, THE SYSTEM ENGINEER RESPONSIBLE FOR THE SERVICE WATER SYSTEM (SSWS) REPORTED TO THE CONTROL ROOM THAT A 30" PIPE SECTION ON THE "A" SSWS LOOP HAD DEVELOPED A MINOR THROUGH WALL FLAW. ULTRASONIC TEST SUBSEQUENT TO DISCOVERY OF THE FLAW DETERMINED THAT A 5" X 4" AREA SURROUNDING THE FLAW HAD DEGRADED TO LESS THAN REQUIRED MINIMUM WALL THICKNESS, AND THAT THE FLAW WAS ABOUT 1.25" IN TOTAL LENGTH. ACTUAL LEAKAGE THROUGH THE FLAW WAS MINIMAL (MEASURED IN DROPS PER MINUTE). DEFICIENCY REPORT AND SAFETY EVALUATION WERE INTITIATED IMMEDIATELY TO DOCUMENT THE FINDINGS. CALCULATIONS PERFORMED BY THE NUCLEAR ENGINEERING DEPARTMENT DETERMINED THAT THE PIPING WAS ACCEPTABLE FOR CONTINUED USE, PENDING REPLACEMENT OF THE PIPING PRIOR TO PLANT STARTUP FOLLOWING STATIONS 3RD REFUELING OUTAGE. THE INITIATING CAUSE OF THIS OCCURENCE WAS A FLAW IN THE INTERNAL PIPING EPOXY COATING WHICH ALLOWED PIPE CORROSION TO BEGIN AND SUBSEQUENT EROSION TO OCCUR. CORRECTIVE ACTIONS INCLUDE VISUAL INSPECTION OF SELECTED SECTIONS OF THE SAFETY RELATED SS PIPING IN ACCORDANCE WITH THE INSPECTION PROGRAM DEVELOPED IN RESPONSE NRC GENERIC LETTER 89-13 AND REPLACEMENT OF THE SUBJECT PIPE SECTION PRIOR TO THE END OF THE PRESENT REFUELING OUTAGE.

[90] HOPE CREEK 1 DOCKET 50-354 LER 91-001
PRIMARY CONTAINMENT ISOLATION DURING PERFORMANCE OF SURVEILLANCE TEST DUE TO MULTIPLE CAUSES.
EVENT DATE: 011091 REPORT DATE: 020691 NSSS: GE TYPE: BWR

(NSIC 220949) ON JANUARY 11, 1991 AT 0125, AN ACTUATION OF THE CHANNEL "B" PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) OCCURRED DURING THE PERFORMANCE OF A MONTHLY FUNCTIONAL TEST OF CHANNEL "B" EMERGENCY CORE COOLING SYSTEM (ECCS) REACTOR VESSEL LEVEL INSTRUMENTATION. FOLLOW-UP INVESTIGATION DETERMINED THAT A CHANNEL "B" PCIS ACTUATION SIGNAL WAS INPUT DURING THE SURVEILLANCE AT THE SAME TIME THAT A SEALED IN ISOLATION FROM THE NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NS4) EXISTED DUE TO A PREVIOUS RPS BUS TRANSFER FROM ALTERNATE TO NORMAL POWER. THE CONCURRENT EXISTENCE OF BOTH SIGNALS SATISFIED THE CHANNEL "B" PCIS LOGIC, AND THE REQUIRED SYSTEM ACTUATIONS OCCURRED. IT WAS DETERMINED THAT THE PREREQUISITES OF THE ECCS FUNCTIONAL TEST PROCEDURE DID NOT REQUIRE RESETTING OF POTENTIALLY SEALED IN ISOLATION SIGNALS FROM THE NS4 PRIOR TO INITIATING A PCIS SIGNAL. ADDITIONALLY, A NUCLEAR CONTROL OPERATOR (NCO, RO LICENSED) DID NOT RESET THE NS4 CIRCUIT AS REQUIRED BY THE RPS OPERATING PROCEDURE FOLLOWING THE RPS BUS TRANSFER AND NO INDICATIONS EXIST TO ALERT CONTROL ROOM PERSONNEL WHEN A SINGLE CHANNEL NS4 ISOLATION SIGNAL IS PRESENT. CORRECTIVE ACTIONS INCLUDED MODIFYING THE ECCS FUNCTIONAL TEST PROCEDURE TO REQUIRE THE RESETTING OF POTENTIALLY SEALED IN NS4 ISOLATION SIGNALS AND COUNSELLING FOR THE NCO WHO FAILED TO RESET THE NS4 ISOLATION PER PROCEDURE.

[91] INDIAN POINT 2 DOCKET 50-247 LER 90-017
TOXIC GAS MONITOR RESULTS IN AN ESF ACTUATION.
EVENT DATE: 122790 REPORT DATE: 012591 NSSS: WE TYPE: PWR
VENDOR: WISCONSIN BRIDGE & IRON

(NSIC 220983) ON 12/27/90 AT APPROXIMATELY 0330 HOURS, WITH THE UNIT OPERATING AT 96% POWER, THE TOXIC GAS MONITOR HYDROGEN CYANIDE (HCN) CHANNEL 2 INADVERTENTLY ALARMED, RESULTING IN THE TRANSFER OF THE CENTRAL CONTROL ROOM (CCR) VENTILATION SYSTEM FROM THE NORMAL MODE TO THE INCIDENT MODE. FOUR ADDITIONAL INADVERTENT ACTUATIONS OF THE CCR VENTILATION SYSTEM OCCURRED SUBSEQUENTLY, WITHIN THE PERIOD OF JANUARY 6 THROUGH 13, 1991, AS A RESULT OF FALSE ALARMS GENERATED BY THE TOXIC GAS MONITOR CHLORINE (CL(2)), AMMONIA (NH(3)), AND HCN CHANNELS. AS DESIGNED, THE DETECTION OF THE TOXIC GAS MONITOR CHANNELS OF EITHER CL(2), NH(3) OR HCN WILL GENERATE AN ALARM IN THE CCR AND ISOLATE THE CCR VENTILATION SYSTEM. THE TOXIC GAS MONITOR SYSTEM AND THE CCR ISOLATION SYSTEM ARE ENGINEERED SAFETY FEATURES. NO TECHNICAL SPECIFICATION OR NRC LIMIT WAS EXCEEDED. LIKEWISE THERE WAS NO IMPACT ON PUBLIC HEALTH AND SAFETY.

[92] INDIAN POINT 3 DOCKET 50-286 LER 90-010
PRESSURIZER PRESSURE TRANSMITTERS OUT OF TECHNICAL SPECIFICATION RANGE DUE TO INSTRUMENT DRIFT.
EVENT DATE: 112190 REPORT DATE: 122190 NSSS: WE TYPE: PWR
VENDOR: FOXBORO CO., THE

(NSIC 220580) ON OCTOBER 6, 1990, WITH THE PLANT IN COLD SHUTDOWN FOR THE CYCLE 7/8 REFUELING OUTAGE, TECHNICIANS PERFORMING THE "PRESSURIZER PRESSURE CONTROL SYSTEM TRANSMITTER CHECK AND CALIBRATION" FOUND THAT CALIBRATION DATA FOR THE FOUR (4) PRESSURIZER PRESSURE TRANSMITTERS WERE OUT OF THE ACCEPTANCE RANGE. THIS RESULTED IN THE HIGH PRESSURIZER PRESSURE TRIP BEING GREATER THAN THAT ALLOWED BY THE TECHNICAL SPECIFICATIONS. THE CAUSE OF THIS EVENT IS THE WEARING OF THE ZERO SUPPRESSION SPRING IN THE PRESSURIZER PRESSURE TRANSMITTERS. ALL "AS-LEFT" VALUES WERE WITHIN THEIR RESPECTIVE TOLERANCE BANDS.

[93] INDIAN POINT 3 DOCKET 50-286 LER 91-003
MANUAL REACTOR TRIP FROM LOSS OF ALL CONDENSER COOLING WATER.
EVENT DATE: 122790 REPORT DATE: 012191 NSSS: WE TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.
WESTINGHOUSE ELECTRIC CORP.

(NSIC 220720) ON DECEMBER 27, 1990, WITH THE REACTOR AT 48 PERCENT POWER, A MANUAL UNIT TRIP WAS INITIATED BECAUSE CONTROL ROOM OPERATORS OBSERVED ALL CIRCULATING WATER PUMPS HAD TRIPPED OFF. PLANT SYSTEMS FUNCTIONED PROPERLY FOLLOWING THE TRIP, WITH THE EXCEPTION OF 32 REACTOR COOLANT PUMP, WHICH TRIPPED DURING 6.9KV TRANSFER FROM ONSITE TO OFFSITE POWER. THE CAUSE OF THE LOSS OF

CIRCULATING WATER PUMPS WAS A FAULT IN A NON-SAFETY-RELATED TRANSFORMER. FOLLOWING REPAIRS, THE UNIT WAS RETURNED TO SERVICE ON DECEMBER 28, 1990.

[94] KEWAUNEE DOCKET 50-305 LER 90-010
RUPTURE RESTRAINT NOT INSTALLED AS DESIGNED RESULTS IN A CONDITION OUTSIDE THE PLANT'S DESIGN BASIS.
EVENT DATE: 110290 REPORT DATE: 120390 NSSS: WE TYPE: PWR

(NSIC 220890) AT 1225 ON NOVEMBER 2, 1990, WITH THE PLANT AT 100% POWER, A MANAGEMENT REVIEW OF ANALYSIS OF A RUPTURE RESTRAINT ON THE PRESSURIZER SURGE LINE DETERMINED THAT THE RESTRAINT HAD NOT MET ITS ORIGINAL DESIGN BASIS. NINE OF THE ORIGINAL THIRTY BASE PLATE WALL ANCHORS HAD NOT BEEN INSTALLED AS ORIGINALLY DESIGNED. THE INCIDENT WAS DISCOVERED ON MARCH 11, 1990, DURING KEWAUNEE'S ANNUAL REFUELING OUTAGE. KEWAUNEE ARCHITECT ENGINEER WAS CONTACTED FOR ASSISTANCE. THEY DETERMINED THAT FUTURE OPERABILITY COULD BE ASSURED BY INSTALLING 3 OF THE MISSING ANCHORS AND REPAIRING ONE NONFUNCTIONAL ANCHOR. WESTINGHOUSE WAS CONTRACTED TO ANALYZE THE AS-FOUND CONDITION. WESTINGHOUSE'S REPORT WAS RECEIVED AT APPROXIMATELY 1200 ON NOVEMBER 2, 1990. THE ANALYSIS DETERMINED THAT, HAD A HYPOTHETICAL RUPTURE OCCURRED ON THE PRESSURIZER SURGE LINE, THE RESTRAINT AS ORIGINALLY INSTALLED WOULD NOT HAVE RESTRAINED THE PIPE. THE RUPTURE RESTRAINT WAS INSTALLED DURING ORIGINAL PLANT CONSTRUCTION. IT APPEARS THAT THE RESTRAINT WAS IMPROPERLY INSTALLED AT THIS TIME. DUE TO THE LENGTH OF TIME BETWEEN PLANT CONSTRUCTION AND DISCOVERY OF THIS EVENT (APPROXIMATELY 15 YEARS), THE CAUSE CANNOT BE CONCLUSIVELY DETERMINED. THREE OF THE MISSING ANCHORS WERE INSTALLED AND ONE OF THE NONFUNCTIONAL ANCHORS WAS REPAIRED PRIOR TO THE END OF THE 1990 REFUELING OUTAGE.

[95] KEWAUNEE DOCKET 50-305 LER 90-014
FAILURE TO FULLY UNDERSTAND THE DESIGN OF THE RADIATION MONITORING SYSTEM RESULTS IN INCORRECT ASSUMPTION WHICH CAUSES AN INADVERTENT ESF ACTUATION DURING PIPE RADIOGRAPHY.
EVENT DATE: 121390 REPORT DATE: 011491 NSSS: WE TYPE: PWR

(NSIC 220813) THIS REPORT DESCRIBES AN UNPLANNED AUTOMATIC ACTUATION OF THE STEAM GENERATOR BLOWDOWN ISOLATION VALVES, AN ENGINEERED SAFETY FEATURE (ESF). THE VALVES CLOSED AND ISOLATED STEAM GENERATOR BLOWDOWN AND BLOWDOWN SAMPLING AT 1910 AND 1956 ON DECEMBER 13, 1990, WITH THE PLANT AT 100% POWER. THE VALVES CLOSED AS DESIGNED ON A HIGH RADIATION SIGNAL FROM R-15, CONDENSER AIR EJECTOR GAS MONITOR. THE HIGH RADIATION SIGNAL WAS GENERATED AS A RESULT OF RADIOGRAPHY IN THE TURBINE BUILDING IN THE VICINITY OF THE DETECTOR FOR R-15. PRIOR TO STARTING THE RADIOGRAPHY, THE "OPERATION SELECTOR" SWITCH ON THE R-15 DRAWER WAS ROTATED FROM THE "OPERATE" POSITION TO THE "RESET" POSITION. IT WAS BELIEVED THAT PLACING THE SWITCH IN "RESET" BLOCKED ALL SIGNALS TO THE ACTUATION CIRCUITRY FOR R-15. HOWEVER THIS ASSUMPTION, WHICH WAS BASED ON EXISTING OPERATING PROCEDURES AND PAST OPERATING PRACTICES FOR R-15, WAS INCORRECT. PLACING THE SWITCH TO "RESET" DOES NOT BLOCK THE SIGNAL FROM A SATURATED DETECTOR. TO PREVENT RECURRENCE OF THIS EVENT A TEMPORARY CHANGE HAS BEEN APPROVED TO JUMPER OUT THE ACTUATIONS ASSOCIATED WITH R-15 DURING RADIOGRAPHY. OPERATIONS WILL REVIEW AND REVISE THEIR PROCEDURES AS NECESSARY TO ENSURE THEY ACCURATELY REFLECT THE DESIGN OF R-15.

[96] LA SALLE 1 DOCKET 50-373 LER 90-014
REACTOR CORE ISOLATION COOLING SYSTEM INOPERABLE DUE TO LOW 250 VDC BATTERY ELECTROLYTE TEMPERATURE CAUSED BY THE INABILITY OF THE TEMPERATURE CONTROLLER TO MAINTAIN PROPER TEMPERATURES.
EVENT DATE: 122590 REPORT DATE: 012491 NSSS: GE TYPE: BWR

(NSIC 220985) ON 12/25/90 AT 1140 HOURS, WITH UNIT 1 IN OPERATIONAL CONDITION 1 (RUN) AT 97% POWER, THE AVERAGE ELECTROLYTE TEMPERATURE OF THE UNIT 1 250V BATTERY WAS FOUND TO BE 62F. ALTHOUGH THERE ARE NO TECH SPEC TEMPERATURE REQUIREMENTS ON THIS BATTERY, LASALLE OPERATING SURVEILLANCE LOS-AA-D1 "UNIT DAILY SURVEILLANCES" REQUIRES THE UNIT 1 250V BATTERY TO BE DECLARED INOPERABLE SHOULD AVERAGE ELECTROLYTE TEMPERATURE FALL BELOW 65 DEGREES F. SINCE THIS

BATTERY ALSO PROVIDES EMERGENCY POWER TO THE REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM. THE RCIC SYSTEM WAS ALSO DECLARED INOPERABLE AND A 14-DAY TIMECLOCK INITIATED IN ACCORDANCE WITH TECH SPECS. THE APPARENT CAUSE OF THIS EVENT APPEARS TO BE THE INABILITY OF THE UNIT 1 DIVISION I SWITCHGEAR HEAT REMOVAL SYSTEM (VX) TEMPERATURE CONTROLL"R TO MAINTAIN PROPER TEMPERATURE IN THE 250V BATTERY ROOM DUE TO OUTSIDE SEASONAL TEMPERATURE CHANGES. THE CONSEQUENCES OF THIS EVENT APPEARS TO BE THE INABILITY OF THE UNIT 1 DIVISION I SWITCHGEAR HEAT REMOVAL SYSTEM (VX) TEMPERATURE CONTROLLER TO MAINTAIN PROPER TEMPERATURE IN THE 250V BATTERY ROOM DUE TO OUTSIDE SEASONAL TEMPERATURE CHANGES. THE CONSEQUENCES OF THIS EVENT WERE MINIMAL. THE HIGH PRESSURE CORE SPRAY SYSTEM, WHICH IS THE ALTERNATE HIGH PRESSURE INJECTION SYSTEM, WAS FULLY OPERABLE THROUGHOUT THIS EVENT.

[97] LA SALLE 2 DOCKET 50-374 LER 90-013
MAIN STEAM ISOLATION VALVE INITIATION SIGNAL DURING SURVEILLANCE TEST DUE TO PERSONNEL ERROR.
EVENT DATE: 121790 REPORT DATE: 011691 NSSS: GE TYPE: BWR

(NSIC 220825) ON DECEMBER 17, 1990 AT 1316 HOURS, UNIT 2 WAS IN HOT SHUTDOWN WITH THE MAIN STEAM ISOLATION VALVES (MSIV'S) CLOSED. TECHNICIANS FROM THE INSTRUMENT MAINTENANCE DEPARTMENT WERE PERFORMING LASALLE INSTRUMENT SURVEILLANCE LIS-MS-406 "UNIT 2 CONDENSER LOW VACUUM MSIV ISOLATION FUNCTIONAL TEST." WHILE IN THIS MODE THE MSIV ISOLATION SIGNALS ARE BYPASSED BY MEANS OF A KEY SWITCH. IN THIS EVENT, THE FUNCTIONAL TEST OF THE SWITCH AND ASSOCIATED ALARMS WAS SATISFACTORILY COMPLETED ON THE INITIAL CHANNEL AND IT WAS RETURNED TO BYPASS, HOWEVER, THE MSIV HALF ISOLATION RECEIVED FROM THE TEST WAS NOT RESET BEFORE THE SECOND CHANNEL WAS REMOVED FROM BYPASS, AND A FULL ISOLATION SIGNAL OCCURRED. THE EVENT OCCURRED AS A RESULT OF PERSONNEL ERROR WHERE THE INSTRUMENT MAINTENANCE TECHNICIAN DID NOT FOLLOW THE STEPS OF THE INSTRUMENT SURVEILLANCE PROCEDURE IN SEQUENTIAL ORDER. THE ISOLATION SIGNAL WAS IMMEDIATELY RESET AND THE SURVEILLANCE FOR THAT CHANNEL COMPLETED SATISFACTORILY. ALL INSTRUMENT MAINTENANCE DEPARTMENT PERSONNEL WERE INFORMED OF THIS EVENT AND WERE TAILGATED ON THE PERFORMANCE OF PROCEDURE. THIS EVENT IS BEING REPORTED TO THE NRC PURSUANT TO THE REQUIREMENTS OF 10CFR50.73(A)(2)(IV) DUE TO ACTUATING AND AN ENGINEERING SAFETY FEATURE SYSTEM SIGNAL.

[98] LA SALLE 2 DOCKET 50-374 LER 90-014
INSULATION FIRE ON THE 2A TURBINE DRIVEN REACTOR FEEDWATER PUMP DUE TO OIL LEAK ON INSULATION.
EVENT DATE: 122290 REPORT DATE: 012191 NSSS: GE TYPE: BWR

(NSIC 220728) AT APPROX. 0745 HOURS ON 12/22/90 WITH UNIT 2 IN OPERATIONAL CONDITION 1 (RUN) AT 80% POWER, SECURITY PERSONNEL NOTIFIED THE CONTROL ROOM THAT A FIRE WATCH ON ELEVATION 815 OF THE AUX. BLDG. NOTICED A SMELL OF SOMETHING BURNING ON THAT ELEVATION. THE CENTER DESK DISPATCHED AN OPERATOR TO INVESTIGATE. AT THE SAME TIME A SHIFT FOREMAN WAS IN THE AUX. BLDG. AND NOTICED THE SMELL. THE SHIFT FOREMAN PROCEEDED TO THE TURBINE DRIVEN REACTOR FEEDWATER PUMP (TDRFP) ROOMS TO INVESTIGATE. UPON ARRIVING AT THE 2A TDRFP, HE NOTICED SMOKE COMING FROM THE FRONT AREA OF THE TDRFP. HE THEN NOTIFIED THE CONTROL ROOM. A FIRE WATCH WAS ESTABLISHED AND ADDITIONAL FIRE SUPPRESSION EQUIPMENT WAS BROUGHT TO THE AREA. REACTOR POWER WAS REDUCED TO 900 MWE AT 0845 HOURS IN PREPARATION FOR TAKING THE 2A TDRFP OFF-LINE. AT 0915 HOURS THE MOTOR DRIVEN REACTOR FEEDWATER PUMP (MDRFP) WAS ON-LINE AND THE 2A TDRFP WAS SHUT DOWN. AT 0925 HOURS THERE WAS A REPORT OF FLAMES IN THE 2A TDRFP ROOM. THE FIRE BRIGADE RESPONDED AT 0927 HOURS. AT THIS TIME, THE CONTROL ROOM OPERATORS CLOSED THE HIGH PRESSURE AND LOW PRESSURE OIL SUPPLIES TO THE TURBINE IN ORDER TO PREVENT THE FIRE FROM SPREADING. AT 0945 HOURS THE FIRE WAS EXTINGUISHED. THE FIRE BRIGADE WAS NOT ABLE TO ESTINGUISH THE FIRE WITHIN TEN MINUTES OF ARRIVING ON THE SCENE AND A GENERATING STATION EMERGENCY PLAN (GSEP) UNUSUAL EVENT WAS DECLARED AT 0940 HOURS.

[99]   LA SALLE 2   DOCKET 50-374   LER 91-001
HIGH PRESSURE CORE SPRAY PUMP ROOM FIRE RATED BARRIER FOUND DEGRADED DURING INSPECTION.
EVENT DATE: 011091   REPORT DATE: 020891   NSSS: GE   TYPE: BWR

(NSIC 221004) AT 0830 HOURS ON 1/10/91, WITH UNIT 2 IN OPERATIONAL CONDITION 1 (RUN) AT 100% POWER, THE STATION TECHNICAL STAFF WAS PERFORMING LASALLE TECH SPEC LTS-1000-42 "FIRE ASSEMBLY INTEGRITY INSPECTIONS" AND FOUND 3 OPEN PENETRATIONS IN A TECH SPEC RELATED FIRE RATED WALL. A ONE HOUR FIRE WATCH WAS INITIATED IN ACCORDANCE WITH LASALLE TECH SPEC 3.7.6 ACTION REQUIREMENT A. A WORK REQUEST WAS INITIATED TO SEAL THE PENETRATION AND WAS COMPLETED ON 1/15/91. A DESIGN BASIS FIRE ON EITHER SIDE OF THESE PENETRATIONS COULD RENDER THE HIGH PRESSURE CORE SPRAY (HPCS) SYSTEM INOPERATIVE. HOWEVER, THE EMERGENCY CORE COOLING SYSTEM DIVISIONS I AND II, AND THE REACTOR CORE ISOLATION COOLING SYSTEM WERE AVAILABLE IN THE EVENT OF AN EMERGENCY. IONIZATION DETECTORS ARE PROVIDED IN THESE FIRE ZONES AND ANNUNCIATE AN ALARM LOCALLY AND IN THE MAIN CONTROL ROOM. BECAUSE THE DEGRADATION OF THE FIRE BARRIER WOULD NOT HAVE IMPAIRED SAFE SHUTDOWN OF UNIT 2, THE SAFETY SIGNIFICANCE OF THIS IS CONSIDERED TO BE MINIMAL. SINCE LTS-1000-42 HAS BEEN PERFORMED SEVERAL TIMES ON THIS WALL WITHOUT DETECTING THESE OPENINGS, THE ROOT CAUSE OF THIS EVENT IS ATTRIBUTED TO THE FAILURE OF TECHNICAL SUPPORT PERSONNEL TO PERFORM ADEQUATE SURVEILLANCE INSPECTIONS. A CONTRIBUTING CAUSE IS THE LOCATION OF THE OPENINGS, BEING APPROXIMATELY 17 FEET OFF THE FLOOR AND ABOVE VENTILATION DUCTWORK.

[100]   LIMERICK 1   DOCKET 50-352   LER 91-001
LOSS OF REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY DUE TO A REACTOR ENCLOSURE OVERPRESSURIZATION TRANSIENT CAUSING A BLOWOUT PANEL TO ACTUATE.
EVENT DATE: 010891   REPORT DATE: 020691   NSSS: GE   TYPE: BWR

(NSIC 220947) ON 01/07/91, AT 1545 HOURS, WHILE PLACING THE NORMAL REACTOR ENCLOSURE (RE) HEATING, VENTILATION, AND AIR CONDITIONING (HVAC) SYSTEM IN OPERATION, THREE OF THE FOUR OPERATING RE HVAC SUPPLY AND EXHAUST AIR FANS TRIPPED. HOWEVER, THE 'A' RE HVAC SUPPLY AIR FAN CONTINUED TO OPERATE, RESULTING IN A RE HIGH POSITIVE AIR PRESSURE CONDITION. THE POWER SUPPLY BREAKER FOR THE 'A' RE HVAC SUPPLY AIR FAN WAS MANUALLY TRIPPED AND THE FAN THEN SHUTDOWN. AT 1630 HOURS, OPERATIONS PERSONNEL RESTARTED THE RE HVAC SYSTEM, HOWEVER, THE SYSTEM EXPERIENCED DIFFICULTY IN MAINTAINING THE RE NEGATIVE DIFFERENTIAL PRESSURE. THE THIRD RE HVAC EXHAUST AIR FAN WAS PLACED IN OPERATION AND RE NEGATIVE DIFFERENTIAL PRESSURE WAS REESTABLISHED. AT 1900 HOURS, AN OPEN RE BLOWOUT PANEL WAS DISCOVERED. THE TECH SPECS (TS) ACTION FOR LOSS OF RE SECONDARY CONTAINMENT INTEGRITY (SCI) WAS ENTERED AND REQUIRED RESTORING RE SCI OR TO COMPLETE A PLANT SHUTDOWN BY 1100 HOURS ON 01/08/91. ON 01/08/91, AT 0920 HOURS, REINSTALLATION OF THE BLOWOUT PANEL WAS COMPLETE AND RE SCI WAS RESTORED. FOLLOWING THIS EVENT ON 01/08/91, PLANT PERSONNEL DETERMINED THAT RE SCI ACTUALLY BECAME INOPERABLE AT 1545 HOURS AND NOT AT 1900 HOURS, WHICH REQUIRED THE PLANT TO BE IN HOT SHUTDOWN BY 0745 HOURS ON 01/08/91. CAUSE OF THE RE HVAC SYSTEM PROBLEMS WAS DUE TO EQUIPMENT FAILURES.

[101]   LIMERICK 1   DOCKET 50-352   LER 91-003
INOPERABILITY OF THE LOOSE-PART MONITORING SYSTEM FOR MORE THAN 30 DAYS DUE TO A FAULTY ELECTRICAL COMPONENT IN ONE SYSTEM.
EVENT DATE: 011391   REPORT DATE: 012391   NSSS: GE   TYPE: BWR
VENDOR: BABCOCK & WILCOX COMPANY

(NSIC 220727) ON JANUARY 13, 1991, THE LIMERICK GENERATING STATION (LGS) UNIT 1 LOOSE- PART MONITORING SYSTEM (LPMS) WAS INOPERABLE FOR MORE THAN 30 DAYS REQUIRING SUBMITTAL OF A SPECIAL REPORT IN ACCORDANCE WITH THE REQUIREMENTS OF TECHNICAL SPECIFICATIONS (TS) SECTION 6.9.2. ON DECEMBER 5, 1990, DURING AN INDEPENDENT VERIFICATION OF A SURVEILLANCE TEST (ST) PROCEDURE, ST-2-036-640-1, "LOOSE PART MONITORING SYSTEM CHANNEL FUNCTIONAL TEST," THE SYSTEM ENGINEER IDENTIFIED THAT THE DIGITAL LOOSE PART LOCATOR (DLPL) CIRCUIT OF THE LPMS WAS IN A "LOCKUP" CONDITION AND COULD NOT BE RESET. IT WAS THEN CONCLUDED THAT THE LPMS WAS INCAPABLE OF RESPONDING TO A POTENTIAL LOOSE PART EVENT. THE LPMS HAD ENTERED A 30 DAY ACTION REQUIREMENT AS REQUIRED BY TS SECTION 3.3.7.10 ON

DECEMBER 14, 1990, WHEN THE UNIT 1 OPERATIONAL CONDITION (OPCON) WAS CHANGED FROM OPCON 4 (COLD SHUTDOWN) TO OPCON 2 (STARTUP). AS A RESULT, ON JANUARY 13, 1991, THE 30 DAY TIME LIMIT FOR RETURNING THE LPMS TO OPERABLE STATUS EXPIRED REQUIRING THE SUBMITTAL OF A SPECIAL REPORT IN ACCORDANCE WITH TS SECTIONS 3.3.7.10 AND 6.9.2. THE CAUSE OF THE MALFUNCTION HAS NOT YET BEEN DETERMINED WITH CERTAINTY BEYOND RECOGNIZING THAT A FAULTY COMPONENT IN ONE OF THE DLPL PRINTED CIRCUIT BOARDS IS THE PROBABLE CAUSE. A PURCHASE REQUISITION HAS BEEN INITIATED TO REPLACE ALL THE BOARDS IN THE DLPL CIRCUIT.

[102] LIMERICK 2 DOCKET 50-353 LER 90-021
SPECIAL REPORT FOR AN INVALID DIESEL GENERATOR START FAILURE DURING A VALID TEST.
EVENT DATE: 120690 REPORT DATE: 010291 NSSS: GE TYPE: BWR

(NSIC 220673) ON DECEMBER 6, 1990, PLANT PERSONNEL WERE PERFORMING SURVEILLANCE TEST (ST) PROCEDURE ST-6-092-311-2, "D21 DIESEL GENERATOR OPERABILITY TEST RUN." AFTER RUNNING FOR APPROXIMATELY 90 MINUTES, THE D21 EMERGENCY DIESEL GENERATOR (EDG) OUTPUT BREAKER TRIPPED ON REVERSE POWER. THE D21 EDG WAS DECLARED INOPERABLE AT 1228 HOURS ON DECEMBER 6, 1990. THE CAUSE OF THIS EVENT WAS INTERMITTANT CLOSURE OF CROSS CURRENT CONTROL RELAY (CCCR) CONTACTS WHICH CAUSED THE D21 EDG CONTROL CIRCUITS TO GO INTO AND OUT OF THE ISOCHRONOUS MODE. THE D21 EDG CCCR WAS REPLACED. THE D21 EDG WAS DECLARED OPERABLE AT 2340 HOURS ON DECEMBER 7, 1990, AFTER THE SUCCESSFUL PERFORMANCE OF PROCEDURE ST-6-092-361-2. THE TERMINATED D21 EDG TEST WAS CLASSIFIED AS AN INVALID FAILURE IN ACCORDANCE WITH THE GUIDANCE OF REGULATORY GUIDE 1.108, "PERIODIC TESTING OF DIESEL GENERATOR UNITS USED AS ONSITE ELECTRIC POWER SYSTEM AT NUCLEAR POWER PLANTS," REVISION 1. IN THE EVENT OF AN ACTUAL LOSS OF OFFSITE POWER, THE THREE REMAINING OPERABLE UNIT 2 EDGS WOULD HAVE PROVIDED ADEQUATE POWER TO ACHIEVE THE SAFE SHUTDOWN OF THE REACTOR.

[103] LIMERICK 2 DOCKET 50-353 LER 91-001
A CONDITION PROHIBITED BY TECH SPECS DUE TO AN INOPERABLE OVERCURRENT PROTECTIVE DEVICE AND THE REQUIRED TECH SPEC ACTION WAS NOT PERFORMED WITHIN THE REQUIRED TIME PERIOD.
EVENT DATE: 010791 REPORT DATE: 020691 NSSS: GE TYPE: BWR

(NSIC 220948) ON 1/7/91, I&C PERSONNEL IDENTIFIED A CONDITION WHICH WAS PROHIBITED BY TECH SPECS (TS). A UNIT 2 PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICE WAS NOT CALIBRATED TO THE TS (SECTION 3.8.4.1) REQUIRED TOLERANCE WHEN IT WAS INSTALLED AS A REPLACEMENT FOR A DEFECTIVE OVERCURRENT PROTECTIVE DEVICE ON 12/3/90 AND THE REQUIRED TS ACTIONS WERE NOT TAKEN. THE OVERCURRENT PROTECTIVE DEVICE IS AN INSTANTANEOUS MAGNETIC CIRCUIT BREAKER (D234-R-E-10) THAT IS ALIGNED TO THE UNIT 2 A2 DRYWELL UNIT COOLER FAN AND PROTECTS THE ELECTRICAL PENETRATION IN THE PRIMARY CONTAINMENT WALL. THE REPLACEMENT BREAKER FOR THE A2 DRYWELL UNIT COOLER FAN WAS INCORRECTLY CALIBRATED ON 12/3/90 IN ACCORDANCE WITH PROCEDURE PMQ-093-007, "PREVENTIVE MAINTENANCE PROCEDURE FOR TESTING AND CALIBRATION OF 480V MOLDED CASE CIRCUIT BREAKERS." THIS PROCEDURE DID NOT DIFFERENTIATE BETWEEN TS AND NON-TS REQUIRED CIRCUIT BREAKERS WHICH HAVE DIFFERENT ALLOWABLE TOLERANCES. IN ADDITION, MAINTENANCE PERSONNEL AND OPERATIONS PERSONNEL FAILED TO PERFORM THE SURVEILLANCE TEST (ST) WHICH REQUIRES THE OPERABILITY OF THE CIRCUIT BREAKER. THE PMQ-093-007 PROCEDURE WILL BE REVISED TO INCLUDE A NOTE SPECIFYING THAT ST PROCEDURE ST-8-093-320-1 OR 2 SHALL BE PERFORMED FOR THE TS REQUIRED CIRCUIT BREAKERS.

[104] MAINE YANKEE DOCKET 50-309 LER 90-012
PLANT SHUTDOWN DUE TO STEAM GENERATOR NO. 1 TUBE LEAK.
EVENT DATE: 121790 REPORT DATE: 011791 NSSS: CE TYPE: PWR
VENDOR: EAST MOLINE METAL PROD CO

(NSIC 220844) ON 12/17/90 AT 0450, MAINE YANKEE COMMENCED A REACTOR SHUTDOWN FROM 95% POWER WHEN REACTOR COOLANT SYSTEM LEAKAGE THROUGH A TUBE IN STEAM GENERATOR #1 EXCEEDED ADMINISTRATIVE LIMITS OF 0.035 GPM. THE CALCULATED LEAK RATE AT 0521 WAS 0.105 GPM. BY 0636, LEAK RATE HAD INCREASED TO APPROXIMATELY 1.4 GPM. THE TECH SPEC LIMIT FOR PRIMARY TO SECONDARY LEAKAGE IS 0.15 GPM FROM ANY ONE STEAM

GENERATOR. MAINE YANKEE PERFORMED INSPECTIONS AND IDENTIFIED ONE TUBE WITH AN AXIAL CRACK AT THE APEX OF THE U-BEND. THIS TUBE IS LOCATED IN A REGION, CALLED THE STEAM BLANKET REGION, WHERE GENERATOR SUPPORTS DEPRESS FLOW CREATING A STEAM VOID AT THE APEX OF THE U-TUBE BEND. WE BELIEVE THAT MOISTURE ENTERING THIS REGION FLASHES TO STEAM AND ANY CONTAMINANTS ARE DEPOSITED ON THE TUBE SURFACE. THE BUILDUP OF CONTAMINANTS COMBINED WITH RESIDUAL STRESS IN THE TUBE MATERIAL FORM CONSTRUCTION RESULT IN TUBE DEGRADATION WHICH WE BELIEVE IS DUE TO SOME FORM OF CORROSION CRACKING. ALL TUBES IN ROWS 1-11 OF STEAM GENERATOR #1 WERE INSPECTED. MAINE YANKEE ALSO INSPECTED THE TUBES IN ROWS 3-10 IN STEAM GENERATOR #2 AND ROWS 3-11 IN STEAM GENERATOR #3 FOR DEFECTS. AS NECESSARY, THE IDENTIFIED DEFECTED TUBES WERE PLUGGED.

[105] MCGUIRE 1 DOCKET 50-369 LER 90-026 REV 01
UPDATE ON REMOVAL OF THE EMERGENCY AIR PENETRATION ACCESS PLATE RENDERED THE ANNULUS VENTILATION SYSTEM INOPERABLE BECAUSE OF A DESIGN DEFICIENCY.
EVENT DATE: 090989 REPORT DATE: 111290 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 220911) ON 10/12/90, DESIGN ENGINEERING DETERMINED THAT DURING PERFORMANCE OF PROCEDURE PT/1,2/A/4200/01F, LOWER CONTAINMENT PERSONNEL AIR LOCK LEAK RATE TEST, BOTH TRAINS OF THE ANNULUS VENTILATION (VE) SYSTEM WERE RENDERED INOPERABLE AND WERE JUDGED TO BE UNABLE TO CREATE THE REQUIRED NEGATIVE PRESSURE IN THE ANNULUS WITHIN THE REQUIRED TIME ACCORDING TO TECH SPECS. ALSO, THE ANNULUS COULD NOT BE MAINTAINED WITHIN THE SPECIFIED PRESSURES FOR THE REQUIRED TIME WITH THE VE SYSTEM IN THE RECIRCULATION MODE. TO PERFORM THIS PROCEDURE, PERFORMANCE PERSONNEL WERE REQUIRED TO REMOVE THE EMERGENCY AIR PENETRATION ACCESS PLATE. NEXT, IT WAS NECESSARY FOR PERFORMANCE PERSONNEL TO OPEN THE BYPASS LEAKAGE ENCLOSURE CONTROL ACCESS DOOR (CAD) TO FACILITATE REMOVAL OF THE EMERGENCY AIR PENETRATION FLANGE AND INSTALL THE LEAK RATE TEST RIG. BOTH TRAINS OF THE VE SYSTEM WERE DETERMINED TO BE INOPERABLE DURING THE SUBJECT TESTING DUE TO EXCESS INLEAKAGE THROUGH THE OPEN TEST PORT AND ACCESS DOOR. THIS EVENT IS ASSIGNED A ROOT CAUSE OF DESIGN DEFICIENCY BECAUSE OF UNANTICIPATED INTERACTION OF SYSTEMS DUE TO A DESIGN OVERSIGHT. UNITS 1 AND 2 HAVE BEEN IN VARIOUS MODES OF OPERATION DURING PERFORMANCE OF THE PERSONNEL AIR LOCK (PAL) LEAK RATE TEST. APPROPRIATE MODIFICATIONS HAVE BEEN SUBMITTED TO PREVENT RECURRENCE OF THIS EVENT.

[106] MCGUIRE 1 DOCKET 50-369 LER 90-024 REV 01
UPDATE ON INLEAKAGE ON TRAIN "A" OF THE CONTROL ROOM VENTILATION SYSTEM EXCEEDED ACCEPTABLE LEVELS BECAUSE OF AN UNKNOWN CAUSE.
EVENT DATE: 082290 REPORT DATE: 092190 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 220953) ON 8/22/90, AT APPROXIMATELY 1445, MECHANICAL MAINTENANCE (MNT) PERSONNEL DISCOVERED AIR INLEAKAGE ON THE CONTROL ROOM VENTILATION (VC) SYSTEM TRAIN A AIR HANDLING UNIT (AHU). THIS INLEAKAGE WAS DISCOVERED AFTER MNT PERSONNEL NOTICED AUDIBLE LEAKAGE ON THE VC SYSTEM TRAIN A AHU. THE INLEAKAGE WAS ESTIMATED BY MNT PERSONNEL TO BE SEVERAL HUNDRED CUBIC FEET PER MINUTE (CFM). THIS UNFILTERED AIR INLEAKAGE VIOLATED THE LEAKAGE ASSUMPTIONS FOR THE POST-ACCIDENT OPERATOR DOSE ACCORDING TO TECH SPEC (TS) REQUIREMENTS. THIS EVENT IS ASSIGNED A CAUSE OF UNKNOWN BECAUSE IT CANNOT BE DETERMINED HOW THE AHU FAN HOUSING BECAME DAMAGED, CAUSING THE UNFILTERED AIR INLEAKAGE TO OCCUR. ALSO, IT COULD NOT BE DETERMINED WHEN THE UNFILTERED AIR INLEAKAGE ON TRAIN "A" OF THE VC SYSTEM AHU FAN HOUSING BECAME UNACCEPTABLE. THEREFORE, THE VC SYSTEM TRAIN "A" WAS INOPERABLE FOR AN INDETERMINABLE TIME PERIOD. UNIT 1 AND 2 WERE IN MODE 1 (POWER OPERATION) AT 100% POWER WHEN THIS VC SYSTEM INLEAKAGE WAS DISCOVERED. MNT PERSONNEL IMMEDIATELY SEALED THE AHU FAN HOUSING WITH RTV SEALANT. MAINTENANCE ENGINEERING SERVICES (MES) PERSONNEL WILL INVESTIGATE FURTHER A PROBABLE FAILURE OF DEFICIENCY CAUSE FOR THE FAN HOUSING DAMAGE.

[107] MCGUIRE 1 DOCKET 50-369 LER 90-030 REV 01
UPDATE ON A TECHNICAL SPECIFICATION SURVEILLANCE COULD NOT BE PERFORMED DUE TO INADEQUATE GROUP INTERFACE.
EVENT DATE: 111090 REPORT DATE: 121090 NSSS: WE TYPE: PWR

OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 220834) ON 11/9/90, THE ALTERNATE POWER SUPPLY (SHARED LOAD CENTER 2SLXC, BREAKER 5D) TO SHARED MOTOR CONTROL CENTER SMXR, WAS DEENERGIZED FOR IMPLEMENTATION OF MODIFICATIONS BY CONSTRUCTION AND MAINTENANCE DEPARTMENT - NORTH (CMD-N) PERSONNEL. ON 11/10/90, AT 0913, WHILE THE MODIFICATIONS WERE IN PROGRESS, THE NORMAL POWER SUPPLY TO SMXR (2SLXI, BREAKER 4D), TRIPPED. THIS RESULTED IN A POWER FAILURE TO EMF-53, RADIATION MONITOR FOR THE WASTE HANDLING AREA (WHA) VENTILATION SYSTEM. HOWEVER, THE WHA VENTILATION SYSTEM CONTINUED TO OPERATE, RESULTING IN A POTENTIAL UNMONITORED RELEASE TO THE ENVIRONMENT. OPERATIONS (OPS) PERSONNEL SHUT DOWN THE VENTILATION SYSTEM AT 1130. POWER WAS RESTORED EMF-53 AT 1257. UNITS 1 AND 2 WERE IN MODE 5 (COLD SHUTDOWN) AT THE TIME OF THIS EVENT. THIS EVENT HAS BEEN ASSIGNED A CAUSE OF INADEQUATE GROUP INTERFACE, WHICH RESULTED IN OPERATION OF NEW EQUIPMENT PRIOR TO THE ACCEPTANCE OF OPERATIONAL CONTROL BY THE STATION. A CONTRIBUTORY CAUSE OF POSSIBLE INAPPROPRIATE ACTION HAS ALSO BEEN ASSIGNED. SUBSEQUENT SAMPLING AND ANALYSIS BY RADIATION PROTECTION (RP) PERSONNEL INDICATED THERE WAS NO MEASURABLE RADIOLOGICAL RELEASE.

[108] MCGUIRE 1 DOCKET 50-369 LER 90-033
DAILY SURVEILLANCE REQUIREMENT FOR THE POWER RANGE DETECTORS WAS NOT COMPLETED WITHIN THE REQUIRED TIME FRAME DUE TO PROCEDURE DEFICIENCIES.
EVENT DATE: 120690 REPORT DATE: 010791 NSSS: WE TYPE: PWR

(NSIC 220678) ON DECEMBER 6, 1990, AT 0700, OPERATIONS PERSONNEL ENTERED TECHNICAL SPECIFICATION (TS) 3.0.3. OPERATIONS PERSONNEL WERE UNABLE TO VERIFY THE READINGS FROM THE POWER RANGE DETECTORS TO THE THERMAL POWER BEST ESTIMATE (TPBE) VALUE IN ACCORDANCE WITH TEHCNICAL SPECIFICATION 4.3.1.1.(2). OPERATIONS PERSONNEL COULD NOT VERIFY THE READINGS FROM THE POWER RANGE DETECTORS BECAUSE OF THE TPBE VALUE BEING INVALID. THIS WAS DUE TO A PARTIAL LOSS OF THE OPERATOR AID COMPUTER (OAC). PERFORMANCE PERSONNEL WERE CALLED IN TO PERFORM A MANUAL CALCULATION OF PRIMARY POWER TO VERIFY THE READINGS FROM THE POWER RANGE DETECTORS. PERFORMANCE PERSONNEL INITIATED PROCEDURE PT/0/A/4150/03, THERMAL POWER OUTPUT MEASUREMENT, FOR CALCULATING PRIMARY POWER AT APPROXIMATELY 0600 ON DECEMBER 6, 1990; HOWEVER, THE RESULTS WERE NOT OBTAINED UNTIL APPROXIMATELY 0750 ON DECEMBER 6, 1990. THIS EVENT IS ASSIGNED CAUSES OF PROCEDURE DEFICIENCIES ON THE LOSS OF OAC PROCEDURE AND THE THERMAL POWER OUTPUT MEASUREMENT PROCEDURE. THIS EVENT IS ALSO ASSIGNED A CAUSE OF A POSSIBLE MANUFACTURING DEFICIENCY BECAUSE THE PARTIAL LOSS OF THE OAC WAS DUE TO INADEQUATE SOLDERING. A CONTRIBUTORY CAUSE OF A MANAGEMENT DEFICIENCY IS ALSO ASSIGNED DUE TO INADEQUATE COMMUNICATIONS BETWEEN GROUPS. UNIT 1 WAS IN MODE 1 (POWER OPERATION) AT 100 PERCENT POWER AT THE TIME OF THIS EVENT.

[109] MCGUIRE 2 DOCKET 50-370 LER 89-010 REV 02
UPDATE ON A LEAK OCCURRED ON CONTAINMENT SPRAY HEAT EXCHANGER 2A AFTER VALVE STROKE TIMING BECAUSE OF A DEFICIENT PROCEDURE AND AN INAPPROPRIATE ACTION.
EVENT DATE: 090589 REPORT DATE: 102389 NSSS: WE TYPE: PWR
VENDOR: DELTA SOUTHERN CO.
GARLOCK, INC.
WESCHLER ELECTRIC CORP.

(NSIC 220881) ON 9/5/89, AT APPROX. 1609, CONTROL ROOM (CR) PERSONNEL OPENED VALVE 2NS-18A, CONTAINMENT SPRAY (NS) PUMP 2A SUCTION FROM CONTAINMENT SUMP BLOCK, FOR STROKE TIME MEASUREMENT. CR PERSONNEL THEN NOTICED PRESSURIZER LEVEL DECREASING AND PRESSURIZER RELIEF TANK LEVEL INCREASING AND CLOSED VALVE 2NS-18A. AT APPROXIMATELY 1655, LEAK WAS REPORTED IN THE AUX. BLDG. 716' ELEVATION PIPE CHASE. ABOUT A MINUTE LATER LEVEL BEGAN DECREASING IN THE REFUELING WATER STORAGE TANK. AFTER SEVERAL VALVE MANIPULATIONS, CR PERSONNEL DETERMINED THE PROBLEM WAS WITH NS HEAT EXCHANGER (HX) 2A. A VISUAL INSPECTION AT 1900 REVEALED A LEAK ON THE HX AT THE LOWER HEAD GASKET. CR PERSONNEL CLOSED VALVE 2NS-20A, NS PUMP 2A SUCTION FROM REFUELING WATER STORAGE TANK BLOCK, WHICH ISOLATED THE LEAK. LATER ESTIMATES CONCLUDED THAT APPROX. 10,000 GALLONS OF WATER LEAKED THROUGH THE GASKET. THIS EVENT IS ASSIGNED A CAUSE OF PROCEDURE DEFICIENCY BECAUSE OF AMBIGUOUS AND INCOMPLETE INFORMATION CONCERNING STROKING OF VALVE 2NS-18A. THIS

EVENT IS ALSO ASSIGNED A CAUSE OF INAPPROPRIATE ACTION BECAUSE OF LACK OF ATTENTION TO DETAIL. THE GASKET WAS REPLACED AND THE AFFECTED NS SYSTEM PIPING WAS ANALYZED AND RETURNED TO SERVICE. UNIT 2 WAS IN MODE 5, COLD SHUTDOWN, WITH THE REACTOR COOLANT SYSTEM AT 305 PSI AND 139.5 DEGREES-F, AND RHR (ND) PUMP 2A IN OPERATION.

[110] MCGUIRE 2 DOCKET 50-370 LER 90-008
A MANUAL REACTOR TRIP WAS INITIATED DUE TO A CONTROL ROD FAILURE CAUSED BY A POSSIBLE EQUIPMENT FAILURE.
EVENT DATE: 122790 REPORT DATE: 012891 NSSS: WE TYPE: PWR

(NSIC 220848) ON 12/27/90 OPERATIONS (OPS) AND PERFORMANCE (PRF) REACTOR GROUP PERSONNEL WERE PERFORMING ROUTINE ROD MOVEMENT TESTING ASSOCIATED WITH ZERO POWER PHYSICS TESTING (ZPPT). OPS PERSONNEL ATTEMPTED TO INSERT SHUTDOWN BANK E FROM THE FULLY WITHDRAWN POSITION. SHUTDOWN BANK E FELL INTO THE CORE AT 0141:25, TAKING THE REACTOR SUBCRITICAL. AT 0142:15, OPS PERSONNEL INITIATED A MANUAL REACTOR TRIP. OPS PERSONNEL THEN IMPLEMENTED THE REACTOR TRIP AND UNIT FAST RECOVERY PROCEDURES TO RECOVER FROM THE TRANSIENT. AT 0225, OPS PERSONNEL MADE THE REQUIRED NOTIFICATION TO THE NRC. SINCE IT COULD NOT BE DETERMINED WHAT CAUSED SHUTDOWN BANK E TO DROP, AN INDEPENDENT TECHNICAL REVIEW WAS PERFORMED ON THE EVENT AND, CONSEQUENTLY, ON 12/28/90 AT 0918, A DECISION WAS MADE BY STATION MANAGEMENT PERSONNEL TO RESTART THE REACTOR. UNIT 2 WAS RETURNED TO MODE 2 (STARTUP) OPERATION ON 12/28/90, AT APPROXIMATELY 1234. THIS EVENT IS ASSIGNED A CAUSE OF POSSIBLE EQUIPMENT FAILURE/MALFUNCTION BECAUSE IT COULD NOT BE DETERMINED DURING THE COURSE OF THIS INVESTIGATION WHAT CAUSED THE SHUTDOWN BANK E TO DROP INTO THE CORE. MAINTENANCE ENGINEERING SUPPORT (MES) PERSONNEL WILL DEVELOP A PLAN TO PERFORM FURTHER TESTING IN AN ATTEMPT TO FIND A CAUSE.

[111] MILLSTONE 1 DOCKET 50-245 LER 90-014 REV 01
UPDATE ON LOW PRESSURE COOLANT INJECTION HEAT EXCHANGER FLOW RATES.
EVENT DATE: 090790 REPORT DATE: 011591 NSSS: GE TYPE: BWR
VENDOR: PERFEX, INC.

(NSIC 220861) ON 9/7/90, AT 1845 HOURS, WITH THE PLANT AT 100% POWER (530F AND 1030 PSIG), AN INCONSISTENCY BETWEEN PROCEDURAL AND DESIGN PARAMETERS ASSOCIATED WITH THE LOW PRESSURE COOLANT INJECTION (LPCI) HEAT EXCHANGER FLOW RATES WAS IDENTIFIED. THE INCONSISTENCY WAS ASSOCIATED WITH THE MAXIMUM LPCI FLOW PERMITTED THROUGH THE HEAT EXCHANGER TO PRECLUDE FAILURE DUE TO FLOW-INDUCED VIBRATION, AND THE HEAT EXCHANGER FLOW RATES REQUIRED BY THE EMERGENCY OPERATING PROCEDURES (EOP'S). AFTER REVIEW OF THE PROCEDURES, THE DESIGN BASIS, AND DISCUSSIONS WITH THE HEAT EXCHANGER MANUFACTURER, IT WAS DETERMINED THAT OPERABILITY OF THE CONTAINMENT COOLING SYSTEM COULD NOT BE ASSURED DUE TO POTENTIAL MECHANICAL LIMITATIONS OF THE HEAT EXCHANGER. BOTH CONTAINMENT COOLING SUBSYSTEMS WERE DECLARED INOPERABLE AND A PLANT SHUTDOWN TO COLD SHUTDOWN WAS IMMEDIATELY INITIATED AS REQURED BY TECH SPECS. COLD SHUTDOWN WAS ACHIEVED ON 9/8/90 AT 1705 HOURS. NO SAFETY SYSTEMS WERE REQUIRED TO FUNCTION AS A RESULT OF THIS EVENT AND NO SAFETY CONSEQUENCES RESULTED FROM THIS EVENT.

[112] MILLSTONE 2 DOCKET 50-336 LER 89-010 REV 01
UPDATE ON SERVICE WATER STRAINERS NOT QUALIFIED TO THE APPROPRIATE SEISMIC CRITERIA.
EVENT DATE: 110989 REPORT DATE: 013191 NSSS: CE TYPE: PWR
VENDOR: ADAMS, R. P. CO., INC.

(NSIC 220942) ON 11/9/89 AT APPROX. 1845 HOURS WITH THE REACTOR PLANT IN MODE 5 (0% POWER, 91F, 0 PSIG), IT WAS ANALYTICALLY DETERMINED THAT ALL 3 SERVICE WATER PUMP DISCHARGE STRAINERS WERE NOT QUALIFIED TO THE APPROPRIATE SEISMIC CRITERIA. AS A RESULT OF THIS CONDITION, BOTH SERVICE WATER HEADERS, HENCE BOTH EMERGENCY DIESEL GENERATORS AND BOTH SHUTDOWN COOLING LOOPS, WERE DECLARED INOPERABLE. AN UNUSUAL EVENT WAS DECLARED AT 1845 HOURS. ALL OPERATIONS INVOLVING CORE ALTERATIONS OR POSITIVE REACTIVITY CHANGES WERE SUSPENDED AS REQUIRED BY THE PLANT'S TECH SPECS. A TOTAL OF SEVEN LIMITING CONDITIONS FOR OPERATION WERE ENTERED. A REQUEST FOR ENFORCEMENT DISCRETION WAS REQUESTED AND RECEIVED IN

ORDER TO TAKE EXCEPTION TO THE TECH SPEC REQUIREMENT TO ESTABLISH CONTAINMENT INTEGRITY IN ACCORDANCE WITH ACTION STATEMENT 3.8.2.2. A NEW PIPE SUPPORT WAS DESIGNED AND INSTALLED AT THE OUTLET FLANGE OF EACH OF THE 3 SERVICE WATER STRAINERS. ALL LIMITING CONDITIONS FOR OPERATION WERE MET AND THE UNUSUAL EVENT WAS TERMINATED ON 11/14/89 AT 0310 HOURS.

[113] MILLSTONE 2 DOCKET 50-336 LER 90-004 REV 01
UPDATE ON CONTROL ROOM VENTILATION TECHNICAL SPECIFICATION VIOLATION.
EVENT DATE: 022890 REPORT DATE: 020591 NSSS: CE TYPE: PWR

(NSIC 220943) ON 2/28/90 WITH THE UNIT IN MODE 6, REFUELING, IT WAS DETERMINED THAT IT WAS NOT SATISFYING TECH SPECS 3.7.6.1, FOR CONTROL ROOM AIR CONDITIONING, (CRAC). THIS SPECIFICATION REQUIRES THAT 2 INDEPENDENT EMERGENCY VENTILATION SYSTEMS BE OPERABLE, WHILE IN MODES 5 OR 6. WITH 1 EMERGENCY VENTILATION SYSTEM INOPERABLE, RESTORE THE INOPERABLE SYSTEM TO OPERABLE STATUS WITHIN 7 DAYS OR INITIATE AND MAINTAIN THE OPERABLE CONTROL ROOM EMERGENCY AIR CLEAN UP SYSTEM IN THE RECIRCULATION MODE. THE DEFINITION OF OPERABILITY REQUIRES AN EMERGENCY DIESEL GENERATOR TO BE AVAILABLE. AFTER 7 DAYS OF PERFORMING MAINTENANCE ON THE EMERGENCY DIESEL GENERATOR FOR THE "A" FACILITY, "B" CRAC WAS PLACED IN THE RECIRCULATION MODE OF OPERATION, ON 2/27 AT 22:26 HOURS. 2 HOURS LATER ON 2/28 AT 00:25 HOURS, "B" CRACK WAS TAKEN OUT OF THE RECIRCULATION MODE OF OPERATION AND PLACED IN NORMAL UNTIL 16:31 HOURS ON 2/28. CRAC WAS TAKEN OUT OF RECIRCULATION ON THE BASIS OF TECH SPEC 3.05 WHICH STATES THAT WHEN A SYSTEM IS DECLARED INOPERABLE SOLELY BECAUSE OF ITS EMERGENCY OR NORMAL POWER SUPPLY BEING INOPERABLE, IT MAY BE CONSIDERED OPERABLE IF THE NORMAL OR EMERGENCY POWER SUPPLY IS OPERABLE AND ALL REDUNDANT SYSTEMS ARE OPERABLE. AFTER APPROX. 16 HOURS OF OPERATION WITH THE CRAC IN THE NORMAL MODE IT WAS DETERMINED THAT TECH SPEC 3.0.5 DID NOT APPLY FOR MODE 5 OR 6 AND THE CRAC WAS PLACED IN THE RECIRCULATION MODE.

[114] MILLSTONE 2 DOCKET 50-336 LER 90-018
CONTAINMENT INTEGRITY VIOLATED DURING FUEL MOVEMENT WHEN THE S/G ATMOSPHERIC DUMP VALVE WAS OPENED WITH S/G HANDHOLE OFF.
EVENT DATE: 100290 REPORT DATE: 110190 NSSS: CE TYPE: PWR

(NSIC 220906) ON 10/2/90, AT 1950 HOURS, OPERATIONS DEPARTMENT DETERMINED THAT REQUIREMENTS OF LCO 3.9.4 WERE NOT SATISFIED WHILE PERFORMING REFUELING OPERATIONS. A VENT PATH HAD BEEN ESTABLISHED FROM THE NO. 1 SG ATMOSPHERIC DUMP VALVE (ADV) TO ALLOW DRAINING OF THE NO. 1 SG. THE OPERATOR INADVERTENTLY FORGOT THE SECONDARY MANWAY WAS REMOVED FROM THE NO. 1 SG. UPON DISCOVERY OF THE SITUATION, TSAS 3.9.4.C WAS IMMEDIATELY ENTERED: ALL REFUELING OPERATIONS IN CONTAINMENT WERE SUSPENDED. NO. 1 SG ADV WAS CLOSED TO RE-ESTABLISH CONTAINMENT INTEGRITY. THERE WERE NO SAFETY IMPLICATIONS BECAUSE NO FUEL HANDLING ACCIDENTS OR FUEL DAMAGE OCCURRED WHILE THE CONTAINMENT INTEGRITY WAS VIOLATED. THE ROOT CAUSE OF THE EVENT WAS PERSONNEL ERROR. THE OPERATOR FORGOT THE NO. 1 SG SECONDARY MANWAY WAS REMOVED WHICH PROVIDED A DIRECT PATH TO THE ENVIRONMENT FROM CONTAINMENT WHEN THE NO. 1 SG ADV WAS OPENED. CONSEQUENTLY, HE DID NOT REALIZE LCO 3.9.4 WAS NOT SATISFIED WHEN THE ADV WAS INITIALLY OPENED. TO PREVENT RECURRENCE, AN EXISTING CAUTION IN OP 2316A HAS BEEN RE-LOCATED TO A MORE APPROPRIATE OPERATIONAL SEQUENCE OF STEPS TO PROVIDE CLEAR GUIDANCE TO OPERATORS. IN ADDITION, OPERATIONS DEPARTMENT SUPERVISORS HAVE BEEN COUNSELED ON THEIR KEY ROLE IN ENSURING THAT ATTENTION TO DETAIL DOES NOT LAPSE DURING PERIODS OF EXTENSIVE MAINTENANCE WORK AND CHANGING PLANT CONDITIONS.

[115] MILLSTONE 2 DOCKET 50-336 LER 90-022
SERVICE WATER HEADERS CROSS-TIE VALVE OPEN DUE TO UNKNOWN CAUSE.
EVENT DATE: 111590 REPORT DATE: 011491 NSSS: CE TYPE: PWR
VENDOR: FISHER CONTROLS CO.

(NSIC 220820) ON NOVEMBER 15, 1990, AT 1330, WITH THE PLANT IN MODE 1 (75% POWER, 565F, 2270 PSIG), SERVICE WATER HEADER CROSS-TIE VALVE, 2-SW-97A, WAS FOUND OPEN BY A PLANT ENGINEERING TECHNICIAN PERFORMING A ROUTINE INTAKE STRUCTURE INSPECTION. THE CONTROL ROOM WAS NOTIFIED AND AN OPERATOR MANUALLY CLOSED THE VALVE. NO EMERGENCY OPERATIONS WERE PERFORMED. NO EQUIPMENT WAS CYCLED TO ITS

ACCIDENT POSITION. THE SPECIFIC CAUSE OF THE EVENT IS UNKNOWN BUT THE MOST LIKELY CAUSE IS EITHER IMPROPER POSITIONING OF THE CROSS-TIE VALVE OR THE VALVE WAS INADVERTENTLY RE-POSITIONED DURING MAINTENANCE ACTIVITIES. THIS EVENT IS BEING REPORTED PURSANT TO THE REQUIREMENTS OF PARAGRAPH 50.73(A)(2)(I), REPORTING ANY OPERATION OR CONDITION PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATIONS. SIMILAR LERS: NONE.

[116] MILLSTONE 2 DOCKET 50-336 LER 91-001
ELECTRO-HYDRAULIC CONTROL SYSTEM FAILURE CAUSED REACTOR TRIP.
EVENT DATE: 011091 REPORT DATE: 020891 NSSS: CE TYPE: PWR
VENDOR: DENISON DIVISION

(NSIC 220945) ON 1/10/91 AT 1612 HOURS, WITH THE UNIT OPERATING AT 92% POWER, THE MAIN TURBINE TRIPPED ON LOW ELECTRO-HYDRAULIC CONTROL (EHC) PRESSURE. THE REACTOR PROTECTION SYSTEM INITIATED A REACTOR TRIP FOLLOWING THE TURBINE TRIP. NORMAL POST-TRIP PROCEDURES WERE FOLLOWED. THERE WERE NO SAFETY IMPLICATIONS SINCE THE UNIT EXPERIENCED A NORMAL REACTOR TRIP SHUTDOWN.

[117] MILLSTONE 3 DOCKET 50-423 LER 90-026 REV 01
UPDATE ON HYDROGEN MONITOR TEMPERATURE PROFILE/OPERATING SPECIFICATION INCONSISTENCY DUE TO INADEQUATE DESIGN ENGINEERING INTERFACE.
EVENT DATE: 062590 REPORT DATE: 013091 NSSS: WE TYPE: PWR

(NSIC 220934) AT 1530 HOURS ON 6/25/90, WHILE OPERATING IN MODE 1, AT 100% POWER, 587F AND 2260 PSIA, THE "B" TRAIN CONTAINMENT HYDROGEN MONITOR FAILED A CALIBRATION SURVEILLANCE. THE APPLICABLE TECH SPECS LIMITED CONDITION FOR OPERATION (LCO) HAD BEEN ENTERED AT 0720 HOURS ON 6/25/90, AND THE PLANT REMAINED IN THE LCO UNTIL THE "B" TRAIN HYDROGEN MONITOR CALIBRATION WAS SUCCESSFULLY COMPLETED AT 1630, ON 6/26/90. ON 7/22/90, AN INCONSISTENCY WAS DISCOVERED BETWEEN THE ANTICIPATED AMBIENT TEMPERATURE PROFILE AT THE CONTAINMENT HYDROGEN MONITORS, AND THE OPERATING TEMPERATURE REQUIREMENTS OF THE MONITORS. DURING SOME CONDITIONS, THE AMBIENT TEMPERATURE COULD EXCEED THAT RECOMMENDED BY THE MANUFACTURER. IT WAS IMMEDIATELY DIRECTED THAT THE RECOMBINER BUILDING VENTILATION SYSTEM BE MAINTAINED IN CONTINUOUS SERVICE. THE CONTAINMENT HYDROGEN MONITORS WERE REVIEWED FOR OPERABILITY AND WERE DETERMINED TO BE OPERABLE PROVIDED THAT THE AMBIENT TEMPERATURE IS MAINTAINED BELOW 90F. THIS FINDING WAS DOCUMENTED IN A JUSTIFICATION FOR CONTINUED OPERATION ON 7/25/90. THE ROOT CAUSE OF THE EVENT IS INADEQUATE ENGINEERING INTERFACE BETWEEN DESIGN ORGANIZATIONS FOR THE RECOMBINER BUILDING AND THE CONTAINMENT HYDROGEN MONITORS.

[118] MILLSTONE 3 DOCKET 50-423 LER 90-030
MANUAL REACTOR TRIP DUE TO MOISTURE SEPARATOR REHEATER PIPING LINE BREAKS.
EVENT DATE: 123190 REPORT DATE: 013091 NSSS: WE TYPE: PWR

(NSIC 220935) ON 12/31/90, AT 1636 HOURS WITH THE PLANT IN MODE 1 AT 86% POWER, 580F AND 2250 PSIA, A MANUAL REACTOR TRIP WAS INITIATED DUE TO TWO SIX INCH MOISTURE SEPARATOR DRAIN LINE (DSM) PIPING BREAKS IN THE TURBINE BLDG. FOLLOWING THE TRIP A MAIN STEAM LINE ISOLATION WAS INITIATED TO MINIMIZE THE RELEASE OF STEAM INTO THE TURBINE BLDG. CAUSE OF THE EVENT WAS FAILURE OF THE 2 DSM LINES DOWNSTREAM OF THE RESPECTIVE LEVEL CONTROL VALVES. BOTH LINES APPEARED TO BURST, FAIL LONGITUDINALLY, THEN UNZIP CIRCUMFERENTIALLY AT THE MINIMUM WALL THICKNESS LOCATION. THE WALL THICKNESS AT THE RUPTURE WAS APPROX. 0.020 INCHES. THE CAUSE OF THE SEVERE WALL LOSS WAS SINGLE PHASE EROSION/CORROSION. THE COMBINATION OF TEMPERATURE, HIGH FLUID VELOCITY AND EXTREMELY LOW OXYGEN CONTENT ARE THE CAUSATIVE FACTORS. THE WALL LOSS WAS LOCALIZED. THE MINIMUM THICKNESS OCCURRED ADJACENT TO THE CONTROL VALVE(S) AND INCREASED AT 0.011 INCHES PER INCH DOWNSTREAM FROM THE VALVE(S). THE ROOT CAUSE ANALYSIS OF THE EVENT HAS NOT BEEN COMPLETED. AS IMMEDIATE CORRECTIVE ACTION CONTROL ROOM OPERATORS PERFORMED THE ACTIONS REQUIRED BY THE APPLICABLE EMERGENCY OPERATING PROCEDURES. THE RUPTURED PIPES WERE CAPPED PENDING REPAIRS DURING THE UPCOMING OUTAGE. THE DSM PUMPS AND PIPING WERE ISOLATED. POTENTIAL LONG TERM CORRECTIVE ACTIONS WILL BE EVALUATED WHEN THE ROOT CAUSE ANALYSIS IS COMPLETE.

[119] MONTICELLO DOCKET 50-263 LER 91-001
LOSS OF POWER TO RADIATION MONITOR RESULTING IN UNPLANNED EMERGENCY FILTRATION TREATMENT SYSTEM ACTUATION TO HIGH RADIATION MODE.
EVENT DATE: 010991 REPORT DATE: 020891 NSSS: GE TYPE: BWR

(NSIC 221008) ON JANUARY 9, 1991 WITH THE PLANT OPERATING AT 100% POWER, THE CONTROL ROOM EMERGENCY FILTRATION TREATMENT SYSTEM TRANSFERRED TO THE HIGH RADIATION MODE DURING PERFORMANCE OF THE MONTHLY SURVEILLANCE TEST. A LOSS OF POWER TO THE "A" RADIATION MONITOR RESULTED IN THE TRANSFER BUT THE CAUSE OF THE LOSS OF POWER CANNOT BE DETERMINED. A THOROUGH INVESTIGATION INTO THE EVENT HAS NOT IDENTIFIED ANY UNUSUAL CONDITIONS WHICH MAY HAVE CAUSED THE LOSS OF POWER. FOLLOWING VERIFICATION OF NORMAL SYSTEM OPERATION, THE HIGH RADIATION SIGNAL WAS RESET AND THE SYSTEM RETURNED TO NORMAL OPERATION. THE MONTHLY SURVEILLANCE TEST WAS COMPLETED WITHOUT FURTHER UNPLANNED ACTUATIONS. NO FURTHER CORRECTIVE ACTIONS ARE PLANNED.

[120] MONTICELLO DOCKET 50-263 LER 91-002
CONTROL POWER CIRCUIT BREAKER FOR FILTRATION UNIT HEATER FOUND OPEN RESULTING IN INOPERABILITY OF CONTROL ROOM EMERGENCY FILTRATION SYSTEM.
EVENT DATE: 010991 REPORT DATE: 020891 NSSS: GE TYPE: BWR

(NSIC 220978) ON JANUARY 9, 1991 WITH THE PLANT OPERATING AT 97% POWER, ENGINEERING PERSONNEL DISCOVERED THE 120VAC CONTROL POWER CIRCUIT BREAKER FOR THE "A" EMERGENCY FILTRATION TREATMENT SYSTEM FILTER HEATER IN THE "OFF" POSITION. THIS EVENT WAS REPORTABLE SINCE IT RESULTED IN OPERATION IN A CONDITION PROHIBITED BY THE TECHNICAL SPECIFICATIONS, WHICH REQUIRE THE EMERGENCY FILTRATION SYSTEM BE OPERABLE WHENEVER REACTOR WATER TEMPERATURE IS GREATER THAN 212 DEGREES FAHRENHEIT. THE CONTROL POWER CIRCUIT BREAKER WAS RETURNED TO THE "ON" POSITION AND THE CONTROL ROOM EMERGENCY FILTRATION TREATMENT SYSTEM RETURNED TO NORMAL OPERATION FOLLOWING SUCCESSFUL COMPLETION OF THE MONTHLY SURVEILLANCE TEST. THE PLANT WORK CONTROL PROCESS WILL BE REVISED TO REQUIRE THAT FOLLOWING WORK IN A SAFETY-RELATED POWER PANEL OR LIGHTING PANEL A VERIFICATION OF ALL BREAKER POSITIONS IN THAT PANEL BE COMPLETED.

[121] NINE MILE POINT 1 DOCKET 50-220 LER 90-014
ALTERNATE ROD INJECTION DUE TO PERSONNEL ERROR DURING SURVEILLANCE TESTING.
EVENT DATE: 072290 REPORT DATE: 083190 NSSS: GE TYPE: BWR

(NSIC 220884) ON JULY 22, 1990 WITH THE NINE MILE UNIT #1 (NMP1) IN COLD SHUTDOWN, REACTOR PRESSURE AT AMBIENT, REACTOR COOLANT TEMPERATURE AT 170 DEGREES FAHRENHEIT, WHILE PERFORMING A SURVEILLANCE TEST OF THE ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS), LOGIC CIRCUITS, AN ALTERNATE ROD INJECTION (ARI) SIGNAL WAS GENERATED. THE SCRAM AIR HEADER DEPRESSURIZED AND ALL CONTROL RODS RECEIVED SCRAM INSERT PRESSURE FROM THEIR ACCUMULATORS. THE ROOT CAUSE OF THIS EVENT WAS DETERMINED TO BE HUMAN ERROR IN THAT A COMMUNICATIONS BREAKDOWN BETWEEN A PROCEDURE CONTROLLER AND THE PERFORMERS RESULTED IN A JUMPER NOT BEING REMOVED. IMMEDIATE CORRECTIVE ACTION WAS TO RESET THE ATWS/ARI SIGNAL, REPRESSURIZE THE SCRAM AIR HEADER AND RETURN THE SYSTEM TO NORMAL. SUBSEQUENTLY, THE SURVEILLANCE TEST PROCEDURE WAS REPERFORMED SEVERAL TIMES TO CONFIRM THAT THERE HAD BEEN NO EQUIPMENT FAILURES OR MALFUNCTION THAT CAUSED THE EVENT. OTHER CORRECTIVE ACTION INCLUDED ADDING A VERIFICATION STEP IN THE PROCEDURE TO MINIMIZE THE POTENTIAL FOR PERSONNEL ERROR.

[122] NINE MILE POINT 1 DOCKET 50-220 LER 90-019
REACTOR SCRAM DUE TO SPURIOUS TRIP OF NEUTRON MONITOR CAUSED BY NOISE.
EVENT DATE: 122990 REPORT DATE: 012491 NSSS: GE TYPE: BWR
VENDOR: LIMITORQUE CORP.

(NSIC 220836) ON 12/29/90, AT 1719 HOURS, NINE MILE POINT UNIT 1 (NMP1) EPXERIENCED A FULL REACTOR SCRAM WHEN A TRIP SIGNAL WAS RECEIVED ON REACTOR PROTECTION SYSTEM (RPS) CHANNEL 11 DUE TO A SPIKE ON INTERMEDIATE RANGE MONITOR (IRM) 12. AT THE TIME OF THE EVENT THE MODE SWITCH WAS IN THE "STARTUP" POSITION, REACTOR POWER WAS AT APPROXIMATELY 10% AND A RPS TRIP HAD BEEN INSERTED

ON CHANNEL 12 DUE TO THE FAILURE OF A MAIN STEAM ISOLATION VALVE (MSIV) TO MEET ITS SURVEILLANCE TEST REQUIREMENTS. WHEN THE SCRAM OCCURRED THE UNIT WAS CONDUCTING A FORCED SHUTDOWN DUE TO THE MSIV FAILURE AND INCREASED DRYWELL LEAKAGE. THE CAUSE OF THE EVENT WAS A SPIKE ON IRM 12 (RPS CHANNEL 11) COINCIDENT WITH A HALF-TRIP INSERTED ON RPS CHANNEL 12 DUE TO THE MSIV FAILURE. THE CORRECTIVE ACTION FOR THE IRM SPIKE IS THE CONTINUATION OF THE NEUTRON MONITORING TROUBLESHOOTING AND UPGRADE EFFORTS.

[123] NINE MILE POINT 1 DOCKET 50-220 LER 91-001
REACTOR SCRAM DUE TO SPURIOUS NON-COINCIDENT LOGIC TRIP SIGNAL.
EVENT DATE: 010891 REPORT DATE: 020191 NSSS: GE TYPE: BWR

(NSIC 220993) ON 1/8/91, AT 0327 HOURS, NINE MILE POINT UNIT 1 (NMP1) EXPERIENCED A FULL REACTOR SCRAM AND A MAIN STEAM ISOLATION VALVE (MSIV) ISOLATION. THE MODE SWITCH WAS IN THE REFUEL POSITION, ALL CONTROL RODS WERE AT POSITION 00, REACTOR COOLANT TEMPERATURE WAS 163F, AND REACTOR PRESSURE WAS 0 PSIG. WATER LEVEL WAS BEING LOWERED FOLLOWING THE PERFORMANCE OF A REACTOR VESSEL INSERVICE LEAK TEST. THE INSIDE MSIV'S CLOSED DUE TO THE ISOLATION SIGNAL AND THE OUTSIDE MSIV'S WERE ALREADY CLOSED DUE TO THE HYDRO. THE CAUSE OF THE EVENT WAS DETERMINED TO BE A SPURIOUS ACTUATION OF REACTOR PROTECTION SYSTEM (RPS) NON-COINCIDENT LOGIC, SPECIFICALLY CONDENSER LOW VACUUM AND MAIN STEAM ISOLATION CLOSED BYPASS LOGIC. CORRECTIVE ACTIONS TAKEN WERE TO PERFORM CHECKS, TROUBLESHOOTING, AND MONITORING OF COMPONENTS THAT MAKE UP THE CONDENSER LOW VACUUM AND MAIN STEAM ISOLATION CLOSED BYPASS LOGIC. THE RESULTS OF THESE CORRECTIVE ACTIONS DID NOT INDICATE ANY EQUIPMENT MALFUNCTION OR DEGRADATION THAT COULD HAVE CONTRIBUTED TO A SPURIOUS ACTUATION.

[124] NINE MILE POINT 2 DOCKET 50-410 LER 89-003 REV 01
UPDATE ON UNANALYZED CONDITION DUE TO DESIGN DEFICIENCY.
EVENT DATE: 021689 REPORT DATE: 021591 NSSS: GE TYPE: BWR

(NSIC 221045) ON FEBRUARY 16, 1989, WITH NINE MILE POINT UNIT 2 (NMP2) IN COLD SHUTDOWN, IT WAS NOTED THAT THE SURVEILLANCE TEST PROGRAM DID NOT INCLUDE FOUR SAFETY RELATED VALVES IN THE SERVICE WATER SYSTEM (SWP). AN ENGINEERING EVALUATION ALSO IDENTIFIED THAT THE ASME SECTION XI PUMP AND VALVE INSERVICE TEST (IST) PROGRAM CONTAINED VALVE STROKE TIME LIMITS FOR VARIOUS SWP VALVES WHICH WERE DIFFERENT THAN THOSE USED IN THE SWP FLUID TRANSIENT ANALYSES. FURTHER REVIEW OF THE SWP LOGIC INDICATED THAT TWO PAIRS OF SAFETY RELATED ISOLATION VALVES, WHICH ISOLATE SWP SAFETY RELATED SECTIONS FROM NON-SAFETY RELATED SECTIONS, DID NOT MEET SINGLE FAILURE CRITERIA. THE ROOT CAUSES FOR THIS EVENT WERE DESIGN DEFICIENCY DUE TO AN INADEQUATE UNDERSTANDING OF THE SYSTEM DESIGN BASES, AND A PROGRAMMATIC DEFICIENCY REGARDING THE IST PROGRAM. INITIAL CORRECTIVE ACTIONS INCLUDED REVISION OF THE NMP2 IST PROGRAM, AND INITIATION OF ENGINEERING EVALUATIONS. ADDITIONAL CORRECTIVE ACTIONS INCLUDED: MODIFICATION TO THE SWP SYSTEM, THE INITIATION OF FURTHER ENGINEERING EVALUATIONS, CORRECTIONS TO IDENTIFIED NMP2 IST PROGRAM DEFICIENCIES, AND AN IST PROGRAM REVIEW OF OTHER SAFETY SYSTEMS.

[125] NINE MILE POINT 2 DOCKET 50-410 LER 90-021 REV 01
UPDATE ON TECHNICAL SPECIFICATION VIOLATION DUE TO SAFETY SYSTEM VALVE TESTING REQUIREMENTS NOT INCLUDED IN THE INSERVICE TESTING PROGRAM PLAN.
EVENT DATE: 102690 REPORT DATE: 013191 NSSS: GE TYPE: BWR

(NSIC 220917) ON 10/26/90, IT WAS DETERMINED THAT THE NINE MILE POINT UNIT 2 (NMP2) INSERVICE TESTING (IST) PROGRAM PLAN, DURING ITS DEVELOPMENT, HAD CERTAIN RESIDUAL HEAT REMOVAL SYSTEM (RHS) VALVE TESTS INCORRECTLY OMITTED. SUBSEQUENT REVIEWS OF PLANT SAFETY SYSTEMS HAVE IDENTIFIED ADDITIONAL ASME VALVES WITH INCORRECT IST PROGRAM PLANT TEST REQUIREMENTS, PLUS VALVES WHICH WERE OMITTED FROM THE PROGRAM PLAN WITHOUT ADEQUATE JUSTIFICATION OR DOCUMENTATION. NMP2 WAS IN A REFUELING OUTAGE AT THE TIME THESE CONDITIONS WERE DISCOVERED AND REPORTED. THE ROOT CAUSE FOR THESE CONDITIONS WAS DETERMINED TO BE INADEQUATE DESIGN ANALYSIS. CORRECTIVE ACTIONS INCLUDED: REVISING THE IST PROGRAM PLAN; REVISING

THE IST IMPLEMENTING PROCEDURES; COMPLETING INSERVICE TESTING IN ACCORDANCE WITH REVISED PROCEDURES; AND REVISING THE UPDATED SAFETY ANALYSIS REPORT (USAR).

[126] NINE MILE POINT 2 DOCKET 50-410 LER 90-026
INSTRUMENTATION NOT ENVIRONMENTALLY QUALIFIED DUE TO PERSONNEL ERROR.
EVENT DATE: 122690 REPORT DATE: 012591 NSSS: GE TYPE: BWR
VENDOR: PYCO

(NSIC 220848) ON DECEMBER 26, 1990, AT 1050 HOURS, FOLLOWING A REVIEW OF THE EQUIPMENT QUALIFICATION REQUIRED MAINTENANCE (EQRM) FOR TEMPERATURE ELEMENTS USED IN THE CONTAINMENT ATMOSPHERE MONITORING SYSTEM (CMS), IT WAS FOUND THAT THE EQUIPMENT QUALIFICATION CRITERION STATED IN REGULATORY GUIDE 1.89 AND THE ALTERNATE APPROACH COMMITTED TO IN THE NMP2 FINAL SAFETY ANALYSIS REPORT (FSAR) WERE NOT BEING MET. NINE MILE POINT UNIT 2 (NMP2) WAS IN A REFUELING OUTAGE AT THE TIME THIS CONDITION WAS DISCOVERED. THE ROOT CAUSE IS PERSONNEL ERROR DUE TO POOR WORK PRACTICES IN THAT THE PURCHASER (ARCHITECT/ENGINEER FOR THE PLANT, STONE & WEBSTER ENGINEERING CORP.) DID NOT RECOGNIZE THAT THE COMPONENTS QUOTED BY THE SUPPLIER DID NOT AGREE WITH THE PURCHASE SPECIFICATION. THE CORRECTIVE ACTIONS TAKEN WERE: TO DECLARE THE INSTRUMENTATION INOPERABLE; TO INSPECT AND INSTALL ENVIRONMENTAL SEALS ON ALL AFFECTED TEMPERATURE ELEMENTS, FUNCTIONALLY TEST AND RETURN THE INSTRUMENTS TO SERVICE; AND TO CONDUCT A REVIEW OF ALL SAFETY RELATED TEMPERATURE ELEMENTS AND OTHER ELECTRICAL EQUIPMENT TO VERIFY SIMILAR CONDITIONS DID NOT EXIST.

[127] NORTH ANNA 1 DOCKET 50-338 LER 90-009 REV 02
UPDATE ON MISSED PRESSURIZER HIGH LEVEL REACTOR TRIP RESPONSE TIME SURVEILLANCES DUE TO INADEQUATE LICENSE AMENDMENT REVIEW UPDATED FOR NONCONSERVATIVE RESPONSE TIME TEST METHODS.
EVENT DATE: 071990 REPORT DATE: 021991 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: NORTH ANNA 2 (PWR)

(NSIC 221037) AT 1622 HOURS ON 7/19/90, WITH UNIT 1 OPERATING AT 100% POWER AND UNIT 2 OPERATING AT 85% POWER, IT WAS DISCOVERED THAT RESPONSE TIME TESTING OF THE PRESSURIZER HIGH LEVEL REACTOR TRIP FUNCTION HAD NOT BEEN PERFORMED IN ACCORDANCE WITH TECH SPEC (TS) 3.3.1.1, TABLE 3.3-2. THIS EVENT IS REPORTABLE PURSUANT TO 10 CFR 50.73 (A)(2)(I)(B). THE CAUSE OF THE EVENT WAS PERSONNEL ERROR IN IMPLEMENTING AN APPROVED TS CHANGE. A TEMPORARY WAIVER OF COMPLIANCE WAS GRANTED 7/20/90 TO ALLOW CONTINUED OPERATION WITHOUT TESTING UNTIL THE NEXT REFUELING OUTAGE. PARTIAL RESPONSE TIME TESTING WAS CONDUCTED 7/24/90. RESULTS FOR TRANSMITTERS WERE WELL BELOW THE TS LIMIT. FULL RESPONSE TESTING ON UNIT 2 WAS CONDUCTED 10/25/90 WITH SIMILAR RESULTS. ACCIDENT ANALYSES INDICATE THAT CERTAIN PLANT CONDITIONS MUST EXIST COINCIDENT WITH GREATER THAN TWO SECOND RESPONSE TIME IN ORDER TO FILL THE PRESSURIZER BEFORE A REACTOR TRIP. NORTH ANNA UNITS 1 AND 2 WILL NOT MEET THESE CONDITIONS PRIOR TO PERFORMING THE SURVEILLANCES. ON 9/14/90, 2 POTENTIALLY NONCONSERVATIVE TESTING METHODS WERE DISCOVERED. ON 9/27/90, A THIRD SUCH METHOD WAS DISCOVERED. A COMPONENT WAS NOT INCLUDED IN THE TESTED PORTION OF AN INSTRUMENT CIRCUIT. CAUSE WAS PERSONNEL ERROR DUE TO NON-CONSERVATIVE TS INTERPRETATION.

[128] OCONEE 1 DOCKET 50-269 LER 91-001
POTENTIAL SINGLE FAILURE DURING A LOCA/LOOP EVENT MAY RESULT IN THE LOSS OF EMERGENCY POWER DUE TO DESIGN DEFICIENCY.
EVENT DATE: 010891 REPORT DATE: 020791 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: OCONEE 2 (PWR)
OCONEE 3 (PWR)

(NSIC 220999) ON JANUARY 8, 1991, AT 10:00 HOURS, WHILE ALL THREE OCONEE UNITS WERE AT 100% FULL POWER, IT WAS DISCOVERED THAT A SINGLE FAILURE COULD RESULT IN DAMAGE TO BOTH TRAINS OF OCONEE'S EMERGENCY POWER SOURCE RESULTING IN THE LOSS OF EMERGENCY POWER DURING A POSTULATED DESIGN BASIS EVENT. KEOWEE HYDRO STATION, OCONEE'S EMERGENCY POWER SOURCE, PROVIDES POWER TO OCONEE BY TWO INDEPENDENT PATHS, ONE OF WHICH IS THROUGH AIR CIRCUIT BREAKERS AND AN UNDERGROUND FEEDER. THE POSTULATED SCENARIO IS A LOSS OF COOLANT ACCIDENT/LOSS OF OFFSITE POWER

DESIGN BASIS EVENT WHEN ONE KEOWEE UNIT IS IN OPERATION AND THE OTHER UNIT SHUTDOWN CONCURRENT WITH A SINGLE FAILURE CAUSING A KEOWEE AIR CIRCUIT BREAKER TO CLOSE, SIMULTANEOUSLY CONNECTING THE TWO KEOWEE GENERATORS TOGETHER. AS THE IDLE GENERATOR AUTOMATICALLY STARTS DUE TO THE EMERGENCY SIGNAL, THE SINGLE FAILURE WOULD CONNECT THE TWO GENERATORS TOGETHER WHILE THEY WERE OUT OF PHASE RESULTING IN PROBABLE DAMAGE TO THE GENERATORS AND PROBABLE LOSS OF EMERGENCY POWER TO OCONEE. THE DAMAGE COULD ALSO OCCUR, THOUGH LESS LIKELY, IF BOTH KEOWEE GENERATORS ARE IDLE. THE ROOT CAUSE OF THIS EVENT IS CLASSIFIED AS DESIGN DEFICIENCY, UNANTICIPATED INTERACTION OF SYSTEMS. ON JANUARY 9, 1991, OCONEE IMPLEMENTED ADMINISTRATIVE PROCEDURAL CONTROLS PREVENTING THE DETRIMENTAL EFFECTS OF THIS POSTULATED FAILURE.

[129] OYSTER CREEK DOCKET 50-219 LER 89-001 REV 01
UPDATE ON POSSIBLE LOSS OF MAIN STEAM LINE ISOLATION CAPABILITY DUE TO EXCESSIVE MAIN STEAM ISOLATION VALVE CONTROL AIR LEAKAGE.
EVENT DATE: 011189 REPORT DATE: 011691 NSSS: GE TYPE: BWR

(NSIC 220742) IN RESPONSE TO NRC GENERIC LETTER 88-14, "INSTRUMENT AIR SUPPLY SYSTEM PROBLEMS AFFECTING SAFETY-RELATED EQUIPMENT", AND CONCERNS IDENTIFIED DURING THE EOP INSPECTION AND INTERNAL REVIEW OF THE AIR SYSTEM, TESTING WAS PERFORMED ON THE MAIN STEAM ISOLATION VALVE (MSIV) CONTROL AIR SYSTEM ON JANUARY 3, 1989. THE PURPOSE OF THE TESTING WAS TO CHECK THE MSIV ACCUMULATORS FOR WATER AND TO TEST CONTROL AIR CHECK VALVE LEAK TIGHTNESS. THE TESTING PLAN COULD NOT BE COMPLETED DUE TO EXCESSIVE AIR LEAKAGE FROM CONTROL AIR PIPING CONNECTIONS. THIS CONDITION WAS DETERMINED REPORTABLE ON JANUARY 11, 1989. THE MSIV CONTROL AIR SYSTEM LEAKAGE IS ATTRIBUTED TO THE SYSTEM CONSTRUCTION. THE CONTROL AIR SYSTEM PIPING FROM THE ACCUMULATOR CHECK VALVES TO THE MSIV ACTUATORS WAS REPLACED WITH STAINLESS STEEL PIPING. SOCKET WELDED CONNECTIONS WERE USED WHERE PRACTICAL TO MINIMIZE AIR LEAKAGE. IN ADDITION, THE ACCUMULATOR CHECK VALVES WERE REPLACED WITH A SOFT SEAT CHECK VALVE TO REDUCE LEAKAGE ON THE LOSS OF INSTRUMENT AIR. POST MAINTENANCE TESTING DEMONSTRATED THAT AIR LEAKAGE RATE ON LOSS OF INSTRUMENT AIR WAS ACCEPTABLE FOR THE MSIV TO PERFORM ITS SAFETY FUNCTIONS. FUTURE TESTING OF AIR ACCUMULATORS AND RELATED PIPING WILL BE PERFORMED IN ACCORDANCE WITH THE TESTING PROGRAM DEVELOPED BY GPUN IN RESPONSE TO GENERIC LETTER 88-14.

[130] OYSTER CREEK DOCKET 50-219 LER 90-016
TECHNICAL SPECIFICATION VIOLATION DUE TO ABSENCE OF AN SRO IN THE CONTROL ROOM CAUSED BY PERSONNEL ERROR.
EVENT DATE: 122090 REPORT DATE: 012191 NSSS: GE TYPE: BWR

(NSIC 220875) ON 12/20/90 AT 0326 HOURS, THE SRO LICENSED GROUP SHIFT SUPERVISOR (GSS) LEFT THE GSS OFFICE AFTER INFORMING THE SRO LICENSED GROUP OPERATNIG SUPERVISOR (GOS). AT 0349 HOURS THE GOS LEFT THE CONTROL ROOM WITHOUT ENSURING THAT THE GSS HAD RETURNED TO THE CONTROL ROOM. AS A RESULT, THERE WAS NO SRO IN THE CONTROL ROOM FOR A FOUR MINUTE PERIOD, UNTIL APPROXIMATELY 0353 HOURS. THIS IS A VIOLATION OF TECH SPEC 6.2.2.C WHICH REQUIRES THAT AN SRO BE PRESENT IN THE CONTROL ROOM UNDER THE PLANT CONDITIONS WHICH EXISTED AT THE TIME OF THE OCCURRENCE. CORRECTIVE ACTIONS INCLUDED THE IMMEDIATE RETURN OF THE GOS AND GSS TO THE CONTROL ROOM WHEN IT WAS DISCOVERED THAT THERE WAS NO SRO IN THE CONTROL ROOM. PLANT OPERATIONS MANAGEMENT COUNSELED THE INDIVIDUAL INVOLVED ABOUT THE OCCURRENCE.

[131] OYSTER CREEK DOCKET 50-219 LER 90-017
BOTH STANDBY GAS TREATMENT SYSTEMS DECLARED INOPERABLE DUE TO COMMON DUCT FAILURE.
EVENT DATE: 122090 REPORT DATE: 012191 NSSS: GE TYPE: BWR

(NSIC 220718) ON DECEMBER 20, 1990 AT APPROXIMATELY 1415 HOURS A DEGRADATION IN DUCTWORK WAS DISCOVERED THAT CAUSED BOTH STANDBY GAS TREATMENT SYSTEMS TO BECOME INOPERABLE. THIS CONDITION IS CONSIDERED REPORTABLE IN ACCORDANCE WITH 10CFR50.73(A)(2)(V). THE DUCT IS CONSTRUCTED OF 1/8 INCH SHEET ALUMINUM AND HAS A CROSS SECTIONAL MEASUREMENT OF 14 INCHES BY 14 INCHES. THE DEGRADATION CONSISTED OF A SIDE PANEL SEPARATING FROM THE TOP AND BOTTOM CORNERS FOR A SPAN

OF APPROXIMATELY THREE FEET. THE CAUSE OF THE DUCT FAILURE IS STILL UNDER INVESTIGATION. THE DEGRADATION OF THE DUCT IS A POTENTIALLY SIGNIFICANT CONDITION AS IT COULD HAVE AFFECTED THE OPERATION OF BOTH TRAINS OF THE SGTS. IMMEDIATE CORRECTIVE ACTION CONSISTED OF DECLARING BOTH STANDBY GAS TREATMENT SYSTEMS INOPERABLE AND COMMENCING AN ORDERLY SHUTDOWN IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS. CONCURRENT WITH THE PLANT SHUTDOWN, TEMPORARY REPAIRS WERE MADE TO RESTORE THE INTEGRITY OF THE DUCTWORK.

[132] OYSTER CREEK DOCKET 50-219 LER 90-015
TWO ISOLATION CONDENSER PIPE BREAK SENSORS FOUND OUT OF SPECIFICATION DUE TO EXCESSIVE DRIFT.
EVENT DATE: 122190 REPORT DATE: 011691 NSSS: GE TYPE: BWR
VENDOR: ITT-BARTON

(NSIC 220743) DURING A SURVEILLANCE TEST ON DECEMBER 21, 1990 AT 0940 HOURS, BOTH OF THE "B" ISOLATION CONDENSER CONDENSATE RETURN LINE PIPE BREAK SENSORS WERE FOUND TO TRIP AT A DIFFERENTIAL PRESSURE GREATER THAN THE MAXIMUM ALLOWABLE TRIP SETPOINT SPECIFIED IN THE TECHNICAL SPECIFICATIONS. AT THE TIME OF THE OCCURRENCE THE PLANT WAS IN THE RUN MODE AT 55% POWER. THE CAUSE OF THE EVENT IS EXCESSIVE COMPONENT DRIFT EXPERIENCED SINCE A 1980 FIELD MODIFICATION ON THE SWITCHES. THE SAFETY SIGNIFICANCE IS MINIMAL DUE TO THE OPERABILITY OF THE PIPE BREAK SENSORS AT A SLIGHTLY HIGHER SETPOINT. IMMEDIATE CORRECTIVE ACTION WAS TAKEN TO ADJUST THE SWITCHES TO TRIP WITHIN TECHNICAL SPECIFICATION LIMITS. LONG TERM CORRECTIVE ACTION WILL BE TO REPLACE THESE SENSORS AS PRESENTLY COMMITTED IN THE INTEGRATED SCHEDULE.

[133] PALISADES DOCKET 50-255 LER 91-003
FAILURE TO TEST DIESEL AUTO START CIRCUITS.
EVENT DATE: 052090 REPORT DATE: 012891 NSSS: CE TYPE: PWR

(NSIC 220839) ON 12/19/90, DURING THE REVIEW OF TESTING REQUIREMENTS, IT WAS DETERMINED THAT, ON 5/20/90 AND AGAIN ON 6/17/90, THE EMERGENCY DIESEL GENERATORS (K6A AND K6B) AUTO START INITIATING CIRCUITS HAD NOT BEEN TESTED PRIOR TO START-UP AS REQUIRED BY TECHNICAL SPECIFICATIONS. AT THE TIME OF DISCOVERY THE PLANT WAS IN COLD SHUTDOWN FOR REFUELING AND STEAM GENERATOR REPLACEMENT. THIS EVENT WAS CAUSED BY A PROCEDURAL INADEQUACY IN OPERATIONS CHECK LIST (CL) 36. THE PROCEDURAL INADEQUACY RESULTED WHEN THE RESPONSIBLE ENGINEER AND PROCEDURE REVIEWERS FAILED TO IDENTIFY THAT A RECENT FACILITY CHANGE WHICH CAUSED A REVISION TO CL 36 DELETED A REQUIRED TS TEST. NO PROVISION WAS MADE TO ENSURE THAT THIS REQUIRED TEST WAS INCORPORATED INTO OTHER PROCEDURES. CORRECTIVE ACTION FOR THIS EVENT INCLUDES THE DEVELOPMENT OF TECHNICAL SPECIFICATIONS TEST PROCEDURES TO TEST THE AUTO START CIRCUIT OF THE DIESEL GENERATORS. ADDITIONALLY, A REVIEW OF TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS WILL BE CONDUCTED TO IDENTIFY THE TESTING REQUIREMENTS WHICH ARE NOT PERFORMED BY THE TECHNICAL SPECIFICATION SURVEILLANCE PROCEDURES. THE TEST REQUIREMENTS IDENTIFIED IN THIS REVIEW WILL BE INCORPORATED INTO TS SURVEILLANCE TEST PROCEDURES OR FLAGGED IN THE NON-SURVEILLANCE TYPE PROCEDURES TO REDUCE THE POSSIBILITY OF INADVERTENT DELETION. THIS EVENT DID NOT INVOLVE THE FAILURE OF ANY SYSTEMS OR COMPONENTS.

[134] PALISADES DOCKET 50-255 LER 90-018 REV 01
UPDATE ON INADEQUATE FLOWS THROUGH HOT LEG INJECTION CHECK VALVES.
EVENT DATE: 093090 REPORT DATE: 021991 NSSS: CE TYPE: PWR

(NSIC 221033) ON SEPTEMBER 30, 1990, AT 1632 HOURS THE PLANT WAS SHUTDOWN AND ON SHUTDOWN COOLING. THE SURVEILLANCE TEST RO-65, "HPSI/RHPSI CHECK VALVE TEST", WHICH PROVIDES FOR FULL STROKE TESTING OF CERTAIN HIGH PRESSURE SAFETY INJECTION (HPSI) PUMP SUCTION AND DISCHARGE CHECK VALVES AND THE HOT LEG INJECTION (HLI) CHECK VALVES WAS BEING CONDUCTED. TECHNICAL SPECIFICATION 4.0.5A REQUIRES TESTING OF VALVES IN ACCORDANCE WITH THE ASME BOILER AND PRESSURE VESSEL CODE, SECTION XI, EDITION AND ADDENDA AS SPECIFIED BY 10CFR50.55A(G). THE CODE REQUIRES THAT CHECK VALVES BE EXERCISED "TO THE POSITION REQUIRED TO FULFILL THEIR FUNCTION." THE NRC HAS INTERPRETED THIS TO MEAN FULL STROKE TESTING IS REQUIRED OR, IF FULL STROKING OF THE DISC CANNOT BE VERIFIED, FULL FLOW TESTING. THE RESULTANT FLOW

RATES FOR EACH OF THE TWO PRIMARY COOLANT SYSTEM (PCS) HOT LEGS WERE LESS THAN THE ACCEPTANCE CRITERIA OF 250 GALLONS PER MINUTE (GPM). THE FAILURE OF THE TEST TO SATISFY THE ACCEPTANCE CRITERIA WAS REPORTED TO THE SYSTEM ENGINEER AND THE SHIFT MANAGER AND A CORRECTIVE ACTION DOCUMENT WAS ISSUED. SUBSEQUENT ANALYSIS ON OCTOBER 12, 1990 DETERMINED THAT THE FLOW DELIVERED TO THE HOT LEGS WAS LESS THAN THAT REQUIRED BY THE PLANT SAFETY ANALYSIS. THE HLI CHECK VALVES (CK-ES-3408, 3409 AND 3410) WERE REPLACED WITH SWING CHECK VALVES. THIS EVENT IS REPORTABLE AS A CONDITION OUTSIDE THE DESIGN BASIS OF THE PLANT.

[135] PALISADES DOCKET 50-255 LER 91-002
NON-QUALIFIED CABLE SPLICES INSIDE CONTAINMENT.
EVENT DATE: 122890 REPORT DATE: 012891 NSSS: CE TYPE: PWR
VENDOR: RAYCHEM CORP.

(NSIC 220838) ON 12/28/90, THE PLANT WAS IN COLD SHUTDOWN AND THE CORE WAS TOTALLY DEFUELED FOR THE REPLACEMENT OF THE STEAM GENERATORS. WHILE WORKING ON A DC GROUND IN THE REACTOR HEAD VENT VALVE CIRCUITRY, IT APPEARED THAT A RAYCHEM CABLE SPLICE ON A CONTAINMENT PENETRATION CONNECTOR INSIDE THE CONTAINMENT FOR REACTOR HEAD VENT VALVE PRV-1067 WAS IMPROPERLY INSTALLED. FURTHER INVESTIGATION REVEALED THAT THE CABLE SPLICE INNER SLEEVE WAS TOO SMALL AND THE OUTER SLEEVE WAS SHRUNK DIRECTLY ONTO BRAIDED CABLE JACKET. NO IMMEDIATE ACTIONS WERE REQUIRED AS THE PLANT WAS SHUTDOWN FOR REFUELING AND STEAM GENERATOR REPLACEMENT. A PLANT CORRECTIVE ACTION DOCUMENT WAS INITIATED TO ASSURE THAT THE PROBLEM WAS RESOLVED PRIOR TO STARTUP FROM THE REFUELING OUTAGE. FURTHER INVESTIGATION OF ALL THE (127 TOTAL) CONTAINMENT PENETRATION CONNECTOR PIGTAILS ON PENETRATIONS INSIDE CONTAINMENT, REVEALED EIGHT ADDITIONAL CONTAINMENT PENETRATION CONNECTOR PIGTAILS WITH OUT OF SPECIFICATION SPLICES. THIS RESULTED IN EQUIPMENT BEING IN A CONDITION OUTSIDE ITS DESIGN BASIS. THE CAUSE OF THE EVENT IS ATTRIBUTABLE TO INADEQUATE MODIFICATION OR DESIGN CONTROL DURING THE TIME THE EQUIPMENT WAS INSTALLED OR MODIFIED. ALL IDENTIFIED OUT OF SPECIFICATION SPLICES WILL BE REWORKED PRIOR TO START-UP FROM THE PRESENT REFUELING OUTAGE.

[136] PALISADES DOCKET 50-255 LER 91-004
CONTAINMENT EMERGENCY AIR LOCK LEAKAGE RESULTS IN CONTAINMENT CUMULATIVE LOCAL LEAK RATE IN EXCESS OF TECHNICAL SPECIFICATIONS LIMITS.
EVENT DATE: 010591 REPORT DATE: 020691 NSSS: CE TYPE: PWR

(NSIC 220976) ON 1/5/91, AT 2100 HOURS, THE PLANT WAS IN COLD SHUTDOWN AND THE REACTOR WAS TOTALLY DEFUELED FOR THE REPLACEMENT OF THE STEAM GENERATORS. CONTAINMENT LOCAL LEAK RATE TESTING WAS BEING PERFORMED ON THE CONTAINMENT EMERGENCY AIRLOCK (MZ-50). DURING THIS TESTING, PLANT OPERATORS NOTICED THAT THE INNER DOOR VIEWING PORT GASKET WAS A SOURCE OF LEAKAGE. THE CALCULATED AIRLOCK LEAK RATE WAS DETERMINED TO BE 42,905 CUBIC CENTIMETERS PER MINUTE (CC/MIN). THIS RESULTED IN AN AS FOUND CUMULATIVE LOCAL LEAK RATE TESTING TOTAL OF 98,994 CC/MIN WHICH EXCEEDED THE TECHNICAL SPECIFICATIONS LIMIT OF 65,200 CC/MIN. THE GASKET WAS LATER FOUND TO BE DEGRADED AND WAS REPLACED. THE CONTAINMENT EMERGENCY AIRLOCK WAS SUBSEQUENTLY TESTED SATISFACTORILY. THIS OCCURRENCE IS ATTRIBUTABLE TO INADEQUATE PREVENTATIVE MAINTENANCE PROCEDURES FOR INSPECTION AND REPLACEMENT OF THE ESCAPE LOCK DOOR VIEWING PORT GASKET, AND IS REPORTABLE AS A DEVIATION FROM THE PLANT'S TECHNICAL SPECIFICATIONS. PREVENTATIVE MAINTENANCE PROCEDURES TO REPLACE VIEWING PORT GASKETS ON BOTH THE EMERGENCY AND PERSONNEL AIRLOCKS WILL BE DEVELOPED BY JULY 1, 1991.

[137] PALO VERDE 1 DOCKET 50-528 LER 90-0[illegible]
REACTOR THERMAL POWER LICENSE LIMIT EXCEEDED.
EVENT DATE: 120690 REPORT DATE: 010791 NSSS: CE TYPE: PWR

(NSIC 220698) ON DECEMBER 6, 1990 UNIT 1 WAS OPERATING IN MODE 1 (POWER OPERATIONS) AT APPROXIMATELY 100 PERCENT POWER. OPERATIONS PERSONNEL PLACED A PURIFICATION ION EXCHANGER IN SERVICE WHICH RESULTED IN AN APPROXIMATE 3 PPM DILUTION OF THE REACTOR COOLANT SYSTEM BORON CONCENTRATION. DURING THIS TIME STEAM GENERATOR HIGH RATE BLOWDOWNS WERE ALSO PERFORMED. THESE TWO EVOLUTIONS CAUSED AN INCREASE IN REACTOR POWER. WHEN OPERATORS ATTEMPTED TO INSERT CONTROL

ELEMENT ASSEMBLIES (CEAS) IN THE MANUAL SEQUENTIAL MODE OF OPERATION TO TERMINATE THE POWER INCREASE, THE CEAS DID NOT MOVE. MANUAL GROUP CEA CONTROL WAS SELECTED AND THE CEAS WERE INSERTED AND POWER RETURNED TO 100 PERCENT. ALTHOUGH THE CORE OPERATING LIMIT SUPERVISORY SYSTEM (COLSS) INDICATED THERMAL POWER DID NOT EXCEED 102 PERCENT, ENGINEERING CALCULATIONS SHOW ACTUAL THERMAL POWER MAY HAVE REACHED 104 PERCENT DURING THE TRANSIENT. THE CAUSE OF THE EVENT WAS INSUFFICIENT PROCEDURAL CONTROLS FOR PLACING AN ION EXCHANGER IN SERVICE. ADDITIONALLY, THIS EVENT COULD HAVE BEEN AVOIDED HAD THE SHIFT SUPERVISOR PRESCRIBED TIGHTER RCS TEMPERATURE CONTROLS DURING A PRE-JOB BRIEFING HELD TO DISCUSS THIS EVOLUTION WITH CONTROL ROOM PERSONNEL. A COMPLETE INVESTIGATION INTO THE EVENT WAS PERFORMED IN ACCORDANCE WITH PLANT PROCEDURES. THERE HAVE BEEN NO PREVIOUS SIMILAR EVENTS REPORTED.

[138] PALO VERDE 1 DOCKET 50-528 LER 90-012
LIGHT-EMITTING DIODE LIGHT SOURCE USED TO PERFORM SOURCE CHECK.
EVENT DATE: 122690 REPORT DATE: 012491 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: PALO VERDE 2 (PWR)
PALO VERDE 3 (PWR)

(NSIC 220858) ON DECEMBER 26, 1990, AT APPROXIMATELY 1200 MST, PALO VERDE UNITS 1 AND 2 WERE OPERATING AT APPROXIMATELY 100 PERCENT POWER AND UNIT 3 WAS OPERATING AT APPROXIMATELY 40 PERCENT POWER WHEN A DESIGN ENGINEER REPORTED HIS CONCERN THAT THE PERFORMANCE OF THE RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION (RGEMI) SOURCE CHECKS REQUIRED BY TECHNICAL SPECIFICATION (TS) 4.3.3.8 DID NOT CONFORM WITH THE TS DEFINITION OF SOURCE CHECK. THE SOURCE CHECKS HAD BEEN PERFORMED USING AN INSTALLED LIGHT EMITTING DIODE (LED) LIGHT SOURCE RATHER THAN A RADIOACTIVE SOURCE. ON JANUARY 8, 1991 APS DETERMINED THAT, THOUGH THE RGEMI LOW RANGE AND RADWASTE NOBLE GAS ACTIVITY MONITORS WERE CAPABLE OF PERFORMING THEIR SPECIFIED FUNCTION, THE SOURCE CHECKS USING AN LED LIGHT SOURCE DID NOT MEET THE LITERAL TS DEFINITIONS AND THOSE MONITORS WERE DECLARED ADMINISTRATIVELY INOPERABLE. THE ROOT CAUSE OF THIS EVENT WAS THE LACK OF RECOGNITION OF THE LITERAL REQUIREMENTS OF THE TS DEFINITION OF SOURCE CHECK BY COGNIZANT PERSONNEL. AN ENGINEERING EVALUATION CONCLUDED THAT THE LED LIGHT SOURCE IS AN ACCEPTABLE RADIOACTIVE SOURCE EQUIVALENT. AS CORRECTIVE ACTION, A TS CHANGE REQUEST WILL BE SUBMITTED TO CHANGE THE TS DEFINITION TO ALLOW THE USE OF AN LED LIGHT SOURCE. THE MONITORS WILL BE SOURCE CHECKED USING A RADIOACTIVE SOURCE PENDING THE APPROVAL OF THE TS DEFINITION CHANGE.

[139] PEACH BOTTOM 2 DOCKET 50-277 LER 90-034 REV 01
UPDATE ON INOPERABLE EMERGENCY DIESEL GENERATOR DUE TO PERSONNEL ERROR AND PROCEDURAL DEFICIENCY RESULTS IN A TECHNICAL SPECIFICATION VIOLATION.
EVENT DATE: 111290 REPORT DATE: 020491 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: PEACH BOTTOM 3 (BWR)

(NSIC 220958) ON 11/12/90, AT 1615 HOURS, WITH UNIT 2 OPERATING AT APPROXIMATELY 80% POWER AND UNIT 3 IN SHUTDOWN, IT WAS DISCOVERED THAT THE E3 EMERGENCY DIESEL GENERATOR (EDG) DROOP SELECTOR SWITCHES WERE IN THE PARALLEL POSITION. THESE SWITCHES MUST BE IN THE UNIT POSITION IN ORDER TO ALLOW THE EDG TO PERFORM ITS REQUIRED SAFETY FUNCTION. THIS CONDITION EXISTED SINCE 11/9/90, DURING WHICH TIME TESTING OF THE OTHER EDG'S WAS NOT PERFORMED AS REQUIRED BY TECHNICAL SPECIFICATIONS. THE CAUSES OF THE EVENT WERE PERSONNEL ERROR AND PROCEDURAL DEFICIENCY. THERE WERE NO ACTUAL SAFETY CONSEQUENCES AS A RESULT OF THIS EVENT. THE DROOP SELECTOR SWITCHES WERE RETURNED TO THE UNIT POSITION ON 11/12/90 AND THE E3 EDG WAS SUBSEQUENTLY VERIFIED TO OPERABLE. THE PLANT OPERATOR INVOLVED IN THIS EVENT WAS COUNSELED ON THE IMPORTANCE OF ATTENTION TO DETAIL. SO 52A.8.A, "DIESEL GENERATOR DAILY SHUTDOWN/PRE-STARTUP INSPECTION", WAS REVISED TO REQUIRE VERIFICATION THAT THE DROOP SELECTOR SWITCHES ARE IN THE UNIT POSITION. THERE WERE NO PREVIOUS SIMILAR EVENTS.

[140] PERRY 1 DOCKET 50-440 LER 90-018
FAILURE TO RECOGNIZE INVALID PROCESS COMPUTER DATA FOR CONTAINMENT HUMIDITY RESULTS IN TECHNICAL SPECIFICATION VIOLATION.
EVENT DATE: 081790 REPORT DATE: 091490 NSSS: GE TYPE: BWR

(NSIC 220892) 8/17/90 AT 1720, IT WAS DETERMINED THAT CONTAINMENT AVERAGE TEMPERATURE/RELATIVE HUMIDITY HAD NOT BEEN VERIFIED TO BE WITHIN TECH SPECS (TS) LIMITS AS SHOWN IN TS FIGURE 3.6.5.2-1 FOR A PERIOD OF APPROX. SIX DAYS. FOR DETERMINED REASONS, PROCESS COMPUTER DATA ACQUISITION FOR CONTAINMENT HUMIDITY HAD BEEN SECURED ON 8/12/90; HOWEVER, THE FAILURE OF THE COMPUTER DISPLAY TO UPDATE WAS NOT RECOGNIZED BY CONTROL ROOM PERSONNEL. THE PROCESS COMPUTER POINT WAS IMMEDIATELY RESTORED TO NORMAL, AND AN I&C TECHNICIAN WAS DISPATCHED TO MEASURE CONTAINMENT RELATIVE HUMIDITY USING A PSYCHROMETER TO CONFIRM TECHNICAL SPECIFICATION COMPLIANCE. ROOT CAUSE OF THE FAILURE TO COMPLY WITH TS WAS DETERMINED TO BE A PROGRAMMATIC WEAKNESS IN THE USE OF THE PROCESS COMPUTER SYSTEM INFORMATION TO DETERMINE TECHNICAL SPECIFICATION COMPLIANCE. ALTHOUGH INSTRUCTIONS SPECIFICALLY REQUIRE THE PROCESS COMPUTER TO BE USED TO OBTAIN CONTAINMENT HUMIDITY DATA, FORMAL RAINING AND PROCEDURAL GUIDANCE IS INADEQUATE TO ENSURE EFFECTIVE AND CONSISTENT USE BY THE CONTROL ROOM OPERATORS. A CONTRIBUTING FACTOR WAS THE DEGREE OF COLOR CHANGE IN THE PROCESS COMPUTER SYSTEM DISPLAY WHEN DATA INPUT BECOMES INVALID. TO PREVENT RECURRENCE, INITIAL AND REQUALIFICATION TRAINING PROGRAMS FOR LICENSED OPERATORS WILL BE ENHANCED TO INCLUDE ADEQUATE TRAINING ON THE PROCESS COMPUTER.

[141] PERRY 1 DOCKET 50-440 LER 90-019
LIQUID RADWASTE DISCHARGES NOT MADE IN ACCORDANCE WITH TECHNICAL SPECIFICATION 3.3.7.9 DUE TO PROGRAM DEFICIENCIES AND PERSONNEL ERROR.
EVENT DATE: 081790 REPORT DATE: 091490 NSSS: GE TYPE: BWR

(NSIC 220893) ON 8/17 AND 9/1/90, LIQUID RADIOLOGICAL EFFLUENT RELEASES WERE PERFORMED WITH RADIOACTIVE EFFLUENT MONITORING INSTRUMENTATION INOPERABLE WITHOUT COMPLETING INDEPENDENT VERIFICATION OF VALVE LINEUPS REQUIRED BY TECH SPEC 3.3.7.9. ON 8/20/90, DURING THE INVESTIGATION INTO THE AUGUST 17 EVENT, IT WAS DISCOVERED THAT THE RADWASTE HIGH FLOW DISCHARGE HEADER FLOW INSTRUMENTATION CHANNEL WAS IMPROPERLY CALIBRATED BETWEEN APRIL 7, 1987 AND 7/8/90. DISCHARGES DURING THAT PERIOD WERE PERFORMED WITH NONCONSERVATIVE INDICATIONS OF DISCHARGE FLOW. THE ROOT CAUSES OF THESE EVENTS INCLUDE PERSONNEL ERROR AND PROCEDURAL INADEQUACY. THE FAILURES TO PERFORM INDEPENDENT VERIFICATION ARE ATTRIBUTED TO FAILURE TO FOLLOW PROCEDURE BY RADWASTE OPERATORS, PARTIALLY ATTRIBUTED TO LACK OF CLARITY IN THE DISCHARGE SURVEILLANCE PROCEDURE. THE FAILURE TO APPROPRIATELY CALIBRATE THE FLOW INSTRUMENTATION, DISCOVERED ON 8/20, IS ATTRIBUTED TO A PROCEDURAL DEFICIENCY. THE SURVEILLANCE INSTRUCTION FOR INSTRUMENT CALIBRATION DID NOT INCLUDE PROVISIONS TO ENSURE THE PROPER CALIBRATION FACTOR WAS UTILIZED. APPROPRIATE MODIFICATIONS ARE BEING MADE TO SURVEILLANCE INSTRUCTIONS FOR RADWASTE DISCHARGES TO ENSURE INDEPENDENT VERIFICATION OF VALVE LINEUPS ARE REVIEWED BY THE SHIFT SUPERVISOR PRIOR TO INITIATION OF A DISCHARGE.

[142] PERRY 1 DOCKET 50-440 LER 90-025 REV 02
UPDATE ON LOCAL LEAK RATE TESTS RESULT IN EXCEEDING ALLOWABLE PRIMARY CONTAINMENT LEAKAGE RATES FOR MAIN STEAM LINES A, B, C AND D.
EVENT DATE: 091690 REPORT DATE: 021991 NSSS: GE TYPE: BWR
VENDOR: ATWOOD & MORRILL CO., INC.
BORG-WARNER CORP.
ROCKWELL MANUFACTURING COMPANY

(NSIC 220992) DURING THE PERIOD SEPT. 16-17, 1990, FOLLOWING SHUTDOWN AND COOLDOWN OF UNIT 1 FOR THE SECOND REFUELING OUTAGE, LOCAL LEAK RATE TESTING (LLRT) OF THE MAIN STEAM ISOLATION VALVES (MSIV) WAS CONDUCTED. ALL FOUR OF THE MAIN STEAM LINE (MSL) PENETRATIONS EXHIBITED LEAKAGE IN EXCESS OF THE TECHNICAL SPECIFICATION 3.6.1.2(C) LIMIT OF 25 SCFH WHEN TESTED AT PA (11.31 PSIG). SEVERAL OF THE MSIV'S HAD NOT BEEN FAST CLOSED FOLLOWING THE REACTOR SHUTDOWN. THE MSL'S WERE TESTED AGAIN FOLLOWING THE OPENING AND FAST CLOSING OF EACH MSIV AND WERE STILL FOUND TO BE LEAKING IN EXCESS OF TECHNICAL SPECIFICATION REQUIREMENTS. THE CAUSE OF THESE FAILURES WAS INADEQUATE SEATING CONTACT ON THE OUTBOARD MSIV'S, PILOT VALVE SEAT DAMAGE ON THE A AND D INBOARD MSIV'S, LEAKBY ON ALL FOUR OUTBOARD MSIV DRAIN VALVES AND LEAKBY ON THE B AND D MSL MSIV LEAKAGE CONTROL SYSTEM (LCS) ISOLATION VALVES. AS A RESULT OF THESE FAILURES, SIX OF EIGHT MSIV'S, ALL FOUR OUTBOARD MSIV DRAIN VALVES AND THE B AND D MSL MSIV-LCS

ISOLATION VALVES WERE REWORKED. ALL OF THE MSL'S HAVE BEEN TESTED SATISFACTORILY. MODIFICATIONS TO ENHANCE PROPER SEATING OF THE MSIV'S ARE BEING PURCHASED.

[143] PERRY 1 DOCKET 50-440 LER 90-037
OPERATION OF WRONG SLIDING LINK WHILE PERFORMING SURVEILLANCE TESTING RESULTS IN INADVERTENT START OF THE RHR "B" PUMP.
EVENT DATE: 120990 REPORT DATE: 010791 NSSS: GE TYPE: BWR

(NSIC 220829) ON DECEMBER 9, 1990 AT 0643 DURING PERFORMANCE OF SURVEILLANCE TESTING, TWO UNEXPECTED AUTOMATIC STARTS OF RESIDUAL HEAT REMOVAL (RHR) PUMP "B" OCCURRED DURING THE TIMING PORTION OF THE TEST. FOR EACH START, OPERATORS VERIFIED ADEQUATE WATER LEVEL AND SECURED THE PUMP IN ACCORDANCE WITH APPROVED OPERATING INSTRUCTIONS. AT THE TIME OF OCCURRENCE, THE PLANT WAS IN OPERATIONAL CONDITION 4 (COLD SHUTDOWN). REACTOR PRESSURE VESSEL (RPV) TEMPERATURE WAS 135 DEGREES FAHRENHEIT AND REACTOR PRESSURE WAS ATMOSPHERIC. THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR. THE SURVEILLANCE INSTRUCTION REQUIRES A SLIDING LINK TO BE OPENED PRIOR TO THE TIMING PORTION OF THE TEST, TO PREVENT AN AUTOMATIC PUMP START. ALTHOUGH THE SLIDING LINKS ARE ADEQUATELY IDENTIFIED, AND THE TECHNICIANS WERE FOLLOWING THE PROCEDURES AS WRITTEN, THE TECHNICIANS IDENTIFIED AND OPENED THE WRONG SLIDING LINK. THE CORRECTIVE ACTIONS TAKEN FOR THIS EVENT INCLUDE APPROPRIATE COUNSELING AND DISCIPLINARY ACTION FOR THE I&C TECHNICIANS INVOLVED. ADDITIONALLY, THIS EVENT WILL BE DISCUSSED DURING I&C SECTION CONTINUING TRAINING AND ALL LICENSED OPERATORS WILL REVIEW THIS EVENT DURING OPERATOR REQUALIFICATION TRAINING.

[144] PERRY 1 DOCKET 50-440 LER 90-039
INADEQUATE SURVEILLANCE INSTRUCTIONS RESULT IN TECHNICAL SPECIFICATION VIOLATION.
EVENT DATE: 121190 REPORT DATE: 011091 NSSS: GE TYPE: BWR

(NSIC 220830) ON DECEMBER 11, 1990, AT 1300, IT WAS DISCOVERED THAT INADEQUATE INSTRUCTIONS RESULTED IN THE INOPERABILITY OF BOTH LOOPS OF THE CONTAINMENT SPRAY MODE OF THE RESIDUAL HEAT REMOVAL (RHR) SYSTEM IN VIOLATION OF TECHNICAL SPECIFICATION 3.6.3.2. WHILE PERFORMING REVIEW OF SURVEILLANCE INSTRUCTION (SVI-E12-T1182B), "RHR B LPCI VALVE LINEUP VERIFICATION AND SYSTEM VENTING," IT WAS DISCOVERED THAT THE POSITION OF THE SECOND ISOLATION VALVE IN THE RHR "B" LOOP OF CONTAINMENT SPRAY WAS NOT VERIFIED BY THE SVI AS REQUIRED BY TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENT 4.6.3.2.A. FURTHER, RESEARCH ALSO REVEALED THAT THE POSITION OF THE SECOND ISOLATION VALVE POSITION FOR THE RHR "A" LOOP OF CONTAINMENT SPRAY WAS NOT VERIFIED BY THE APPLICABLE SVI AND THAT NO OTHER SVI'S MET THE REQUIREMENT FOR THESE VALVES. THE CAUSE OF THIS EVENT WAS INADEQUATE INSTRUCTIONS. SVI-E12-T1182A AND B WERE INITIALLY WRITTEN IN 1986 TO SATISFY THE SURVEILLANCE REQUIREMENTS OF TECHNICAL SPECIFICATION 4.6.3.2.A BUT VERIFICATION OF THE SECOND ISOLATION VALVES FOR EACH LOOP WAS NOT INCLUDED. ADDITIONALLY, PERIODIC REVIEWS OF THE INSTRUCTIONS FAILED TO IDENTIFY THE DEFICIENCY. TO PREVENT RECURRENCE, SVI-E12-T1182A AND B WERE REVISED TO INCLUDE THE APPROPRIATE ISOLATION VALVE FOR POSITION VERIFICATION. A REVIEW WAS PERFORMED TO ENSURE THAT THESE VALVES WERE NOT OMITTED FROM OTHER SURVEILLANCE REQUIREMENTS.

[145] PERRY 1 DOCKET 50-440 LER 90-040
REACTOR WATER CLEANUP SYSTEM ISOLATION CAUSED BY A BLOWN FUSE WHILE TROUBLESHOOTING ISOLATION CIRCUITRY.
EVENT DATE: 121890 REPORT DATE: 011791 NSSS: GE TYPE: BWR

(NSIC 220831) ON DECEMBER 18, 1990 AT 0108 WHILE PERFORMING A WORK ORDER TO CORRECT THE CAUSE OF THE REACTOR WATER CLEANUP (RWCU) SYSTEM OUTBOARD ISOLATION VALVES [ISV] NOT OPENING WITH THE RWCU LD (LEAK DETECTION) ISOLATION BYPASS SWITCH IN "NORMAL", A DIVISION 1 RWCU OUTBOARD ISOLATION OCCURRED. AT THE TIME OF OCCURRENCE, THE PLANT WAS IN OPERATIONAL CONDITION 4 (COLD SHUTDOWN). REACTOR PRESSURE VESSEL (RPV) TEMPERATURE WAS 120 DEGREES F AND REACTOR PRESSURE WAS ATMOSPHERIC. IMMEDIATE CORRECTIVE ACTION WAS TAKEN TO SUSPEND WORK AND TO VERIFY THAT ALL AFFECTED VALVES HAD ISOLATED PROPERLY. THE ROOT CAUSE OF THIS EVENT IS INDETERMINATE PERSONNEL ERROR. AN I&C TECHNICIAN WAS PERFORMING A WORK ORDER

THAT REQUIRED LIFTING A WIRE AND RELUGGING IT. THE WIRE BEING RELUGGED HAD BEEN VERIFIED TO BE DEENERGIZED AT BOTH ENDS PRIOR TO COMMENCING WORK. PRECAUTIONS TO PREVENT INADVERTENT SHORTING WERE TAKEN. HOWEVER, THE RWCU ISOLATION OCCURRED AT ABOUT THE SAME TIME AS THE LIFTING OF THE LUGS, INDICATING THAT THE EVENTS WERE PROBABLY RELATED. WIRING TO THE TWO FUSES THAT BLEW WAS IN CLOSE PROXIMITY ON BOTH THE TEST SWITCH AND ON THE RELAY CORRESPONDING TO THE TWO LUGS THAT WERE LIFTED. THE CORRECTIVE ACTIONS TAKEN FOR THIS EVENT INCLUDED INSPECTING THE PANEL AND TOOLS FOR INDICATIONS OF SHORTING, REPLACING THE FUSES AND RESTORING THE RWCU SYSTEM TO NORMAL.

[146] PERRY 1 DOCKET 50-440 LER 90-041
INOPERABLE INSTRUMENTATION RESULTS IN TECHNICAL SPECIFICATION VIOLATION AND HIGH PRESSURE CORE SPRAY SYSTEM INOPERABILITY.
EVENT DATE: 122890 REPORT DATE: 012591 NSSS: GE TYPE: BWR

(NSIC 220865) BETWEEN 12/12 AND 12/28/90, INOPERABLE REACTOR PRESSURE VESSEL LEVEL INSTRUMENTATION CHANNELS RESULTED IN HIGH PRESSURE CORE SPRAY (HPCS) SYSTEM BEING INOPERABLE AND IN VIOLATIONS OF TS 3.3.1, 3.3.2, AND 3.3.3. ON 12/28, TECHNICIANS DISCOVERED THAT A CHANNEL "D" REFERENCE LEG ISOLATION VALVE WAS CLOSED WHICH HAD PREVENTED THE ASSOCIATED INSTRUMENTATION FROM SENDING THE PROPER REACTOR LEVEL/PRESSURE WHICH RESULTED IN HPCS INOPERABILITY. ON 12/24 AND 12/25/90 A LEAKING EQUALIZING VALVE RESULTED IN THE INOPERABILITY OF ALL INSTRUMENTATION ASSOCIATED WITH THE CHANNEL "B" REFERENCE LEG. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR. THE INITIAL POSITIONING OF THIS VALVE WAS DIRECTED BY A VALVE LINEUP INSTRUCTION PERFORMED ON 12/12/90. BECAUSE NO WORK WAS PERFORMED ON OR AROUND THE AFFECTED VALVE, IT IS ASSUMED THAT THE MISPOSITIONING WAS THE RESULT OF AN UNINTENTIONAL ERROR BY UNIDENTIFIED PERSONNEL DURING THE PERFORMANCE OF REFUEL OUTAGE WORK. PLANT MANAGEMENT HAS GENERATED A MEMO TO ALL PLANT PERSONNEL TO ADDRESS THE IMPORTANCE OF MAINTAINING THE INTEGRITY OF VALVE LINEUPS, AND NECESSITY OF EQUIPMENT OPERATION BY ONLY QUALIFIED PERSONNEL WITH PROPER AUTHORIZATION. THESE EVENTS WILL BE DISCUSSED WITH ALL I&C TECHNICIANS, PLANT OPERATORS, CHEMISTRY AND HEALTH PHYSICS TECHNICIANS TO STRESS IMPORTANCE OF PROCEDURAL COMPLIANCE AND ATTENTION TO DETAIL.

[147] PERRY 1 DOCKET 50-440 LER 91-001
RWCU ISOLATION ON HIGH DELTA FLOW WHILE ATTEMPTING TO SHIFT PUMPS.
EVENT DATE: 010191 REPORT DATE: 013191 NSSS: GE TYPE: BWR

(NSIC 220877) ON JANUARY 1, 1991 AT 1935, WHILE ATTEMPTING TO SHIFT REACTOR WATER CLEANUP (RWCU) SYSTEM FLOW FROM THE B PUMP TO THE A PUMP, A DELTA-FLOW HIGH RWCU ISOLATION OCCURRED. AT THE TIME OF DISCOVERY, THE PLANT WAS IN OPERATIONAL CONDITION 2 (STARTUP). REACTOR PRESSURE VESSEL (RPV) TEMPERATURE WAS 170 DEGREES FAHRENHEIT AND REACTOR PRESSURE WAS ATMOSHPERIC. THE OPERATOR STARTED THE A RWCU PUMP FOR A 24 HOUR RUN AND SECURED THE B RWCU PUMP. RWCU FLOW DROPPED OFF CONSIDERABLY AND THE B RWCU PUMP WAS RESTARTED APPROXIMATELY 3 MINUTES LATER. THE A RWCU PUMP WAS THEN SECURED. A DELTA-FLOW HIGH RWCU ISOLATION OCCURRED WITHIN 1 MINUTE BECAUSE OF FLOW IMBALANCES WITHIN THE RWCU SYSTEM RESULTING FROM THIS EVOLUTION. UPON INVESTIGATION, IT WAS DISCOVERED THAT THE A RWCU PUMP DISCHARGE VALVE WAS CLOSED. RWCU FLOW WAS THEN PROPERLY RE-ESTABLISHED IN ACCORDANCE WITH SYSTEM OPERATING INSTRUCTION SOI-G33, "REACTOR WATER CLEANUP SYSTEM", USING THE B RWCU PUMP. THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR, INATTENTION TO DETAIL, BY THE CONTROL ROOM OPERATOR. THE OPERATOR STARTED THE A RWCU PUMP WITHOUT FIRST VERIFYING THAT THE DISCHARGE VALVE HAD BEEN OPENED. THE CORRECTIVE ACTIONS TAKEN FOR THIS EVENT INCLUDE COUNSELING THE OPERATOR INVOLVED AND DISCUSSING THE EVENT DURING OPERATOR REQUALIFICATION TRAINING.

[148] PERRY 1 DOCKET 50-440 LER 91-002
UNEXPECTED TURBINE STOP VALVE CLOSURE DURING PERFORMANCE OF A SURVEILLANCE RESULTS IN A REACTOR PROTECTION SYSTEM ACTUATION.
EVENT DATE: 010191 REPORT DATE: 013191 NSSS: GE TYPE: BWR

(NSIC 220878) ON JANUARY 1, 1991, AT 2050, WHILE PERFORMING ACTIVITIES FOR TURBINE STOP VALVE AND REACTOR PROTECTION SYSTEM (RPS) TESTING, AN INADVERTENT

TURBINE STOP VALVE (TSV) CLOSURE SIGNAL RESULTED IN A FULL SCRAM SIGNAL BEING GENERATED. AT THE TIME OF THE EVENT THE PLANT WAS IN OPERATIONAL CONDITION 2 WITH ALL OF THE CONTROL RODS INSERTED. THE ROOT CAUSE OF THIS EVENT IS INDETERMINATE. A POSSIBLE ELECTRICAL MALFUNCTION WAS INVESTIGATED BY TROUBLESHOOTING THE SPEED CONTROL LOGIC AND TROUBLESHOOTING EFFORTS DID NOT IDENTIFY ANY EQUIPMENT PROBLEMS. THE TWO SURVEILLANCES WERE REVIEWED FOR POSSIBLE INTERACTION EFFECTS. ALTHOUGH SVI-N31-T1151, "MAIN TURBINE VALVE EXERCISE TEST", COULD HAVE PROVIDED MORE DETAIL FOR ADJUSTING CONTROL VALVE POSITION, THE OPERATOR'S CHOSEN METHODS WOULD NOT HAVE INITIATED THE RPS ACTUATION AND INTERACTION BETWEEN THE SURVEILLANCES WAS NOT FOUND. CONTROL ROOM AND I&C PERSONNEL WERE INTERVIEWED AND THE EXACT ACTIONS PERFORMED PRIOR TO THE SCRAM SIGNAL WERE REVIEWED. ALTHOUGH NONE OF THE INDIVIDUALS COULD DISTINCTLY REMEMBER DEPRESSING OR INADVERTENTLY TOUCHING THE CLOSE VALVES BUTTON, THIS ACTION WOULD HAVE CAUSED THE TSV TO CLOSE AND A SCRAM TO OCCUR. THE CLOSE VALVES CONTROL BUTTON IS LOCATED IN CLOSE PROXIMITY TO THE LOAD SET BUTTON AND WAS OBSERVED TO BE LIT FOLLOWING THE SCRAM.

[149] PERRY 1 DOCKET 50-440 LER 91-003
PERFORMANCE OF INADEQUATE PROCEDURE RESULTS IN MAIN STEAM LINE DRAIN ISOLATIONS.
EVENT DATE: 010691 REPORT DATE: 020491 NSSS: GE TYPE: BWR

(NSIC 220923) ON JANUARY 6, 1991 AT 1311, THE PERFORMANCE OF AN INADEQUATE PROCEDURE RESULTED IN MAIN STEAM LINE DRAIN ISOLATIONS. CONTROL ROOM OPERATORS WERE IN THE PROCESS OF RESTORING FEEDWATER HEATER 6A TO SERVICE IN ACCORDANCE WITH SYSTEM OPERATING INSTRUCTION (SOI-N27) "FEEDWATER SYSTEM (UNIT ONE)." AT 1257, THE CONTROL ROOM OPERATORS BYPASSED THE TRIP FUNCTIONS FOR ALL FOUR MAIN STEAM LINE (MSL) RADIATION MONITORS AND, IN ADDITION, PLACED THE NSSSS MSL DRAIN ISOLATION LOGIC TEST SWITCHES TO THE "TEST" POSITION. THE RESULTING LOGIC CONFIGURATION RESULTED IN THE GENERATION OF ISOLATION SIGNALS TO THE INBOARD AND OUTBOARD MSL DRAIN ISOLATION VALVES. THE ROOT CAUSE OF THIS EVENT IS PROCEDURE DEFICIENCY. SOI-N27 INSTRUCTS OPERATORS TO PERFORM APPLICABLE STEPS OF APPROPRIATE SURVEILLANCE INSTRUCTIONS TO BYPASS THE MSL RADIATION MONITORS. THESE SURVEILLANCE INSTRUCTIONS ARE USED TO MEET TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS AND DO CONTAIN STEPS THAT IN EFFECT "BYPASS" MSL RADIATION MONITOR TRIP FUNCTIONS; HOWEVER, THESE STEPS ARE NOT READILY DISCERNIBLE OR PLACED IN A SPECIFIC SECTION SEPARATED FROM OTHER NON-APPLICABLE STEPS. AS A RESULT, THE CONTROL ROOM PERSONNEL HAD TO DETERMINE, WITHOUT PROCEDURAL GUIDANCE, THE "APPLICABLE STEPS" TO BE PERFORMED. TO PREVENT RECURRENCE, SOI-N27 IS BEING REVISED TO INCLUDE OR REFERENCE THE APPROPRIATE STEPS TO BYPASS THE MSL RADIATION MONITORS.

[150] PILGRIM 1 DOCKET 50-293 LER 87-002 REV 01
UPDATE ON LOGIC SYSTEM FUNCTIONAL TEST PROCEDURAL INADEQUACIES.
EVENT DATE: 012387 REPORT DATE: 013191 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CORP. (NUCLEAR ENG DIV)

(NSIC 220843) ON 1/23/87 DURING AN EXTENDED OUTAGE, CERTAIN INADEQUACIES INVOLVING LOGIC SYSTEM FUNCTIONAL TEST (LSFT) PROCEDURES WERE DISCOVERED. THE DISCOVERY RESULTED FROM A COMPREHENSIVE ASSESSMENT OF THE COMPLETENESS AND TECHNICAL ADEQUACY OF LSFT PROCEDURES THAT WAS PROMPTED BY FINDINGS IDENTIFIED IN NRC INSPECTION 50-293/86-21. THE ROOT CAUSE WAS THE PREVIOUS INTERPRETATION OF TECH SPECS REQUIREMENTS FOR LSFT. WHEN THE ORIGINAL TECH SPECS WERE APPROVED, THE APPROACH FOR INTERPRETING FUNCTIONAL TEST REQUIREMENTS WAS CONSISTENT WITH EXISTING (C.1972) INDUSTRY PRACTICE. THE ASSESSMENT APPLIED A MORE CONSERVATIVE INTERPRETATION OF LSFT REQUIREMENTS THAT MORE CLOSELY REFLECTS CURRENT INDUSTRY PRACTICE. CORRECTIVE ACTION TAKEN CONSISTED OF REVISING (THEN) EXISTING PROCEDURES AND/OR WRITING NEW TEST PROCEDURES. THE PROCEDURES WERE THEN PERFORMED PRIOR TO REFUELING, AND PRIOR TO STARTUP AND DURING SUBSEQUENT OPERATION. CORRECTIVE ACTION ALSO INCLUDED A CHANGE TO TECH SPECS THAT LENGTHENED THE LSFT INTERVAL TO ONCE PER 18 MONTHS. LONG TERM CORRECTIVE ACTION INCLUDES A STUDY FOR POSSIBLE HARDWARE IMPROVEMENTS RELATED TO TESTS (E.G. LSFT) THAT INVOLVE THE INSTALLATION OF JUMPERS, BLOCKING RELAY CONTACTS, LIFTING WIRES, OR THE REMOVAL OF FUSES. THIS REPORT IS SUBMITTED IN ACCORDANCE WITH 10 CFR 50.73(A)(2)(I)(B).

[151] PRAIRIE ISLAND 1 DOCKET 50-282 LER 90-019
DISCOVERY THAT FILTER TESTING REQUIREMENTS ARE NOT BEING MET DUE TO PERSONNEL OVERSIGHT IN DEVELOPING PROCEDURES.
EVENT DATE: 121890 REPORT DATE: 011691 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: PRAIRIE ISLAND 2 (PWR)

(NSIC 220812) ON 12/18/90 BOTH UNITS WERE AT 100% POWER. A QUALITY ASSURANCE AUDIT HAD BEEN PERFORMED TO EVALUATE COMPLIANCE WITH SELECTED TECHNICAL SPECIFICATIONS. VENTILATION FILTER TESTING SPECIFICATIONS REQUIRE THAT TESTS "...BE PERFORMED AT LEAST ONCE PER OPERATING CYCLE, OR ONCE EVERY 18 MONTHS WHICHEVER OCCURS FIRST, OR AFTER EVERY 720 HOURS OF SYSTEM OPERATION OR FOLLOWING PAINTING, FIRE OR CHEMICAL RELEASE IN ANY VENTILATION ZONE COMMUNICATING WITH THE SYSTEM THAT COULD CONTAMINATE THE HEPA FILTERS OR CHARCOAL ADSORBERS." THE AUDIT VERIFIED THAT ROUTINE TESTS WERE BEING CONDUCTED EACH OPERATING CYCLE, AND THAT PROCEDURES EXISTED TO ADDRESS TESTING AFTER PAINTING, BUT REVEALED THAT NO MEANS EXISTED TO INSURE TESTING WAS DONE "...AFTER EVERY 720 HOURS OF SYSTEM OPERATION OR FOLLOWING FIRE OR CHEMICAL RELEASE...." THE AUDIT SHOWED THAT THE 720-HOUR LIMIT WAS EXCEEDED WITHOUT ATTENDANT TESTING ON BOTH THE SHIELD BUILDING VENTILATION AND SPENT FUEL POOL SPECIAL VENTILATION SYSTEMS. CAUSE OF THE EVENT WAS OVERSIGHT BY THOSE SYSTEM ENGINEERS CHARGED WITH DEVELOPING TESTING PROCEDURES TO SATISFY THE SURVEILLANCE REQUIREMENTS OF TECHNICAL SPECIFICATIONS. PROCEDURE REVISIONS HAVE BEEN INITIATED TO ADDRESS THE NEED FOR SPECIAL TESTING.

[152] PRAIRIE ISLAND 2 DOCKET 50-306 LER 90-012
REACTOR TRIP AS A RESULT OF ROD CONTROL SYSTEM FAILURES.
EVENT DATE: 122990 REPORT DATE: 012891 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 220872) ON 12/29/90, UNIT 2 WAS AT 100% POWER, AT 2234, UNIT 2 TRIPPED. CONTROL ROOM ANNUNCIATORS, AND THE SEQUENCE OF EVENTS GENERATED BY THE PLANT PROCESS COMPUTER, INDICATED THAT THE CAUSE OF THE REACTOR TRIP WAS A NEGATIVE FLUX RATE TRIP FROM THE REACTOR PROTECTION SYSTEM. AN INVESTIGATION INTO THE CAUSE OF THE NEGATIVE FLUX RATE TRIP REVEALED THAT TWO CIRCUIT CARDS IN THE ROD CONTROL SYSTEM POWER CABINET 1BD HAD FAILED. THE FIRST FAILURE WAS OF A TRANSISTOR IN THE URGENT FAILURE ALARM CIRCUIT. THE SECOND FAILURE WAS THE OPENING OF A SOLDER CONNECTION ON THE STATIONARY GRIPPER REGULATION CARD, CAUSING THE REFERENCE VOLTAGE TO GO TO ZERO. IN RESPONSE TO THE REFERENCE VOLTAGE GOING TO ZERO, THE URGENT FAILURE ALARM CIRCUIT SHOULD HAVE GENERATED AN URGENT FAILURE ALARM AND A "HOLD" CURRENT THAT WOULD BE APPLIED TO THE STATIONARY GRIPPERS FOR ALL THE RODS SUPPLIED BY THAT POWER CABINET. THIS HOLD CURRENT WOULD HAVE PREVENTED THE RODS FROM DROPPING INTO THE REACTOR. BUT SINCE THE URGENT FAILURE ALARM CIRCUIT HAD ALSO FAILED, NO CURRENT WAS SUPPLIED TO THE STATIONARY GRIPPERS IN CONTROL ROD BANK D AND THEY FELL INTO THE CORE, CAUSING THE NEGATIVE FLUX RATE TRIP. THE FAILED CARDS IN THE ROD CONTROL SYSTEM WERE REPLACED. UNIT 2 WAS RETURNED TO SERVICE AT 1330 ON 12/30/90.

[153] QUAD CITIES 1 DOCKET 50-254 LER 89-022 REV 01
UPDATE ON HPCI INOPERABLE DUE TO INADVERTENT DELUGE SYSTEM ACTUATION DUE TO UNKNOWN CAUSE.
EVENT DATE: 112889 REPORT DATE: 020691 NSSS: GE TYPE: BWR

(NSIC 220954) AT 0910 HOURS ON NOVEMBER 28, 1989, THE UNIT ONE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM WAS DECLARED INOPERABLE FOLLOWING AN UNEXPECTED ACTUATION OF THE HPCI PUMP ROOM DELUGE SYSTEM. THE ACTUATION CAUSED DC SYSTEM GROUNDS DUE TO MOISTURE INTRUSION IN THE VARIOUS TURBINE AUXILIARY ELECTRICAL EQUIPMENT. THE ACTUATION OCCURRED WHILE OPERATING PERSONNEL WERE IN THE PROCESS OF RETURNING THE DELUGE SYSTEM TO SERVICE. THE ROOT CAUSE OF THE HPCI DELUGE ACTUATION IS NOT KNOWN. THE ELECTRICAL EQUIPMENT AFFECTED BY THE DELUGE ACTUATION WAS TESTED AND DRIED AS NECESSARY TO REMOVE THE DC GROUNDS. THE HPCI SYSTEM WAS SUCCESSFULLY TESTED AND RETURNED TO SERVICE AT 1045 HOURS ON DECEMBER 1, 1989. CORRECTIVE ACTIONS WILL INCLUDE A PROCEDURE REVISION AND OPERATOR TRAINING. THE DELUGE SYSTEM REMAINS OUT OF SERVICE TO ALLOW INSTALLATION OF A LINEAR DETECTION SYSTEM TO REDUCE THE POTENTIAL FOR FUTURE INADVERTENT ACTUATIONS. FIRE WATCH FREQUENCY WILL BE INCREASED AND A TEMPORARY PROCEDURE

WILL BE INITIATED. ON DECEMBER 11, 1989, THE PERIOD OF TIME THAT THE DELUGE SYSTEM WAS INOPERABLE EXCEEDED THE 14-DAY REPORTING REQUIREMENT OF TECHNICAL SPECIFICATION 3.12.C.3. THIS EVENT IS BEING SUBMITTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(V)(D) AND TECHNICAL SPECIFICATION 3.12.C.3.

[154] QUAD CITIES 1 DOCKET 50-254 LER 90-014
MISSED RADIOLOGICAL EFFLUENT SAMPLE FOLLOWING REACTOR POWER INCREASE DUE TO POOR COMMUNICATION.
EVENT DATE: 070290 REPORT DATE: 080190 NSSS: GE TYPE: BWR

(NSIC 220901) ON 7/2/90 AT 1130 HOURS, UNIT ONE WAS IN THE RUN MODE AT 100% OF RATED CORE THERMAL POWER. AT 1143 HOURS, A THERMAL POWER CHANGE OF 20.7% (519 MEGAWATT) IN ONE HOUR HAD OCCURRED DUE TO RESTARTING THE 1B REACTOR RECIRCULATION PUMP AND COMMENCING A REACTOR POWER INCREASE OF 150 MEGAWATT PER HOUR. TECH SPEC TABLE 4.8-1 REQUIRES A RADIOLOGICAL EFFLUENT (RETS) SAMPLE BE PERFORMED WITHIN 24 HOURS FOLLOWING A 20% THERMAL CHANGE. DUE TO POOR COMMUNICATIONS, THE RETS SAMPLE FOR THE POWER INCREASE WAS MISSED. A REACTOR WATER SAMPLE PERFORMED ON 7/4/90 SHOWED NO INCREASE IN ACTIVITY. CORRECTIVE ACTIONS WILL INCLUDE TRAINING AND ENHANCING PROCEDURES. THIS REPORT IS BEING SUBMITTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(I)(B).

[155] QUAD CITIES 1 DOCKET 50-254 LER 90-028
SPURIOUS ACTUATION OF PRIMARY CONTAINMENT ISOLATION VALVE, MO 1-1601-57, CAUSE IS UNKNOWN.
EVENT DATE: 110290 REPORT DATE: 120390 NSSS: GE TYPE: BWR

(NSIC 220900) ON 11/2/90, AT 1021 HOURS, UNIT ONE WAS IN THE RUN MODE AT 93% OF RATED CORE THERMAL POWER. AT THIS TIME, A SPURIOUS CLOSING OF ONE OF THE TWO REDUNDANT PRIMARY CONTAINMENT ISOLATION (PCI) VALVES FOR THE NITROGEN MAKEUP SYSTEM OCCURRED. AT 1024 HOURS, THE VALVE WAS OPENED. PRELIMINARY INVESTIGATIONS FOUND NO APPARENT CAUSE FOR THIS EVENT. THE VALVE WAS SUCCESSFULLY CYCLED AND CORRECT STROKE TIMING WAS VERIFIED. AT 1229 HOURS, THE N.R.C. WAS NOTIFIED BY PHONE VIA THE EMERGENCY NOTIFICATION SYSTEM (ENS) PER THE REQUIREMENTS OF 10 CFR50.72(B)(2)(II). WORK REQUEST Q88017 WAS WRITTEN TO FURTHER INVESTIGATE POSSIBLE CAUSES OF THIS EVENT. THIS REPORT IS BEING SUBMITTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(IV).

[156] QUAD CITIES 1 DOCKET 50-254 LER 90-026
CONTROL ROOM ISOLATION ON HIGH TOXIC GAS CONCENTRATION DUE TO UNKNOWN CAUSE.
EVENT DATE: 122090 REPORT DATE: 012191 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: QUAD CITIES 2 (BWR)

(NSIC 220746) ON DECEMBER 20, 1990, AT 2306 HOURS, UNIT ONE WAS IN THE SHUTDOWN MODE AT 0 PERCENT POWER AND UNIT TWO WAS IN THE RUN MODE AT 100 PERCENT OF RATED CORE THERMAL POWER. ALARM, CONTROL ROOM STANDBY HVAC SYSTEM MAJOR TROUBLE, ANNUNCIATED AT THIS TIME. THE CONTROL ROOM VENTILATION SYSTEM (HVAC) ISOLATED ON HIGH CHLORINE GAS CONCENTRATION. THIS RESULTED IN A CONTROL ROOM HVAC ENGINEERED SAFETY FEATURE (ESF) ACTUATION. THE INSTRUMENT MAINTENANCE (IM) DEPARTMENT REFILLED THE CHLORINE PROBE WITH ELECTROLYTIC SOLUTION WHEN IT WAS DISCOVERED THAT THE PROBE HAD DRIED OUT. AN EMERGENCY NOTIFICATION SYSTEM (ENS) PHONE NOTIFICATION WAS COMPLETED AT 0114 HOURS ON DECEMBER 21, 1990, AS REQUIRED BY 10CFR50.72(B)(2)(II). ON DECEMBER 22, 1990 AT 1800 HOURS THE CONTROL ROOM VENT TOXIC GAS MONITOR WAS DECLARED OPERABLE AGAIN. THE CAUSE OF THE EVENT IS UNKNOWN AT THIS TIME. AN INSPECTION WAS COMPLETED BY THE MANUFACTURER AND A REVISED REPORT WILL BE SUBMITTED WHEN THE RESULTS HAVE BEEN RECEIVED AND REVIEWED. INITIAL CORRECTIVE ACTION WAS TO REDUCE SYSTEM AIR FLOW AS RECOMMENDED BY THE MANUFACTURER. THE MANUFACTURER COMPLETED AN INSPECTION OF THE SYSTEM, AND FURTHER CORRECTIVE ACTIONS ARE IN PROGRESS. THIS REPORT IS SUBMITTED IN ACCORDANCE WITH 10CFR50.73 (A)(2)(IV).

[157] QUAD CITIES 1 DOCKET 50-254 LER 90-033
ESF ACTUATION WHEN A STANDY GAS TREATMENT AUTO START AND REACTOR BUILDING VENTS ISOLATED DUE TO COMPONENT FAILURE.
EVENT DATE: 122390 REPORT DATE: 012191 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: QUAD CITIES 2 (BWR)
VENDOR: GENERAL ELECTRIC CO.

(NSIC 220964) ON DECEMBER 23, 1990, UNIT ONE WAS IN COLD SHUTDOWN FOR A REFUELING OUTAGE AND UNIT TWO WAS IN THE RUN MODE AT 97 PERCENT OF RATED CORE THERMAL POWER. AT 0754 HOURS, THE FOLLOWING ALARMS ANNUNCIATED IN THE CONTROL ROOM: C-16, FUEL POOL CHANNEL "A" DOWNSCALE; H-3, RX BLDG VENT CHANNEL "B" HI HI RAD; B-18, RX BLDG VENT STACK LOW FLOW; AND D-3, CONTROL ROOM VENT ISOLATED. IN ADDITION, STANDBY GAS TREATMENT (SBGT) AUTO STARTED. REACTOR BUILDING AND CONTROL ROOM VENTILATION (HVAC) ISOLATED AND THE DRYWELL TO TORUS PURGE FANS TRIPPED. AT 1028 HOURS, AN EMERGENCY NOTIFICATION SYSTEM (ENS) PHONE CALL WAS MADE AS REQUIRED BY 10CFR 50.72(B)(2)(II). THE APPARENT CAUSES FOR THIS EVENT WERE A BLOWN FUSE FOR THE CONTROL ROOM HVAC ISOLATION LOGIC AND A BURNT OUT COIL FOR THE REACTOR BUILDING VENT ISOLATION RELAY. AFTER REPAIRS WERE MADE, THE AFFECTED SYSTEMS AND COMPONENTS WERE SUCCESSFULLY TESTED, RETURNED TO THEIR NORMAL CONFIGURATION, AND RESTARTED BY 1905 HOURS. THIS REPORT IS SUBMITTED IN ACCORDANCE WITH 10 CFR 50.73(A)(2)(IV).

[158] QUAD CITIES 1 DOCKET 50-254 LER 90-034
MANUAL ISOLATION OF CONTROL ROOM HVAC DUE TO MISINTERPRETATION OF THE C1 ANALYZER INDICATION.
EVENT DATE: 122390 REPORT DATE: 012191 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: QUAD CITIES 2 (BWR)

(NSIC 220972) ON 12/23/90, UNIT ONE WAS IN THE SHUTDOWN MODE FOR A REFUELING OUTAGE AND UNIT TWO IN THE RUN MODE AT 96 PERCENT OF RATED CORE THERMAL POWER. AT 2055 HOURS, AN OPERATOR REPORTED DURING HIS ROUNDS THAT A HIGH CHLORINE CONCENTRATION INDICATION EXISTED. THE CONTROL ROOM VENTILATION (HVAC) WAS MANUALLY ISOLATED WHICH IS AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION. AT 2330 HOURS, AN EMERGENCY NOTIFICATION SYSTEM (ENS) PHONE CALL WAS COMPLETED PER 10CFR 50.72(B)(2)(II). ON 12/31/90, THE IM'S DETERMINED THE HIGH CONCENTRATION READING WAS ACTUALLY AN INSTRUMENT ERROR CODE, AND THE INDICATED CHLORINE CONCENTRATION HAD BEEN WELL BELOW THE TRIP SETPOINT. THE CAUSE OF THE EVENT WAS A MANUAL ESF ACTUATION DUE TO A MISINTERPRETATION OF THE C1 ANALYZER INDICATION. THE INDICATION WAS BELIEVED TO BE A HIGH CHLORINE CONCENTRATION. IT WAS LATER DISCOVERED THAT A HIGH CHLORINE CONCENTRATION WAS NOT PRESENT. IT IS UNKONWN WHAT CAUSED THE ALARM WHICH WAS BELIEVED TO BE A HIGH CHLORINE CONCENTRATION. AS PART OF CORRECTIVE ACTION, THE MANUFACTURER WAS CONTACTED AND AN INSPECTION OF THE SYSTEM WAS COMPLETED. THE INSPECTION RESULTS ARE PENDING. A REVISED REPORT WILL BE SUBMITTED. THIS REPORT IS SUBMITTED IN ACCORDANCE WITH 10CFR 50.73(A)(2)(IV).

[159] QUAD CITIES 1 DOCKET 50-254 LER 90-032
DIESEL FIRE PUMP OUT OF SERVICE GREATER THAN 7 DAYS FOR PERSONNEL PROTECTION TO PERFORM INSPECTION.
EVENT DATE: 122490 REPORT DATE: 012391 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: QUAD CITIES 2 (BWR)

(NSIC 220859) AT 1000 HOURS ON DECEMBER 24, 1990, UNIT ONE WAS IN THE SHUTDOWN MODE AT 0 PERCENT OF RATED CORE THERMAL POWER AND UNIT TWO WAS IN THE RUN MODE AT 97 PERCENT OF RATED CORE THERMAL POWER. AT THIS TIME, THE TECHNICAL SPECIFICATION 3.12.B.2 SEVEN DAY REPORTING CRITERIA FOR AN INOPERABLE FIRE PUMP (P) (K) WAS EXCEEDED. THE 1/2A DIESEL FIRE PUMP HAD BEEN TAKEN OUT-OF-SERVICE (OOS) ON DECEMBER 17, 1990 FOR PERSONNEL PROTECTION SO AN INSPECTION OF THE UNIT 1 CIRCULATING WATER PUMP (P) INTAKE BAY COULD BE PERFORMED. THE INSPECTION HAD BEEN COMPLETED BUT THE FIRE PUMP WAS NOT RETURNED TO SERVICE BEFORE THE SEVEN DAY TIME ALLOTMENT HAD EXPIRED. THE 1/2A DIESEL FIRE PUMP WAS SUCCESSFULLY TESTED AND RETURNED TO SERVICE ON DECEMBER 24, 1990, AT 1940 HOURS. CORRECTIVE ACTIONS WILL INCLUDE A REVISION TO QOS 4100-07 AND INCLUDING ANY OUT-OF-SERVICE DIESEL

FIRE PUMP ON THE PLAN OF THE DAY. THIS REPORT IS BEING SUBMITTED TO COMPLY WITH STATION TECHNICAL SPECIFICATIONS 3.12.B.2 AND 6.3.A.1.

[160] QUAD CITIES 1 DOCKET 50-254 LER 91-001
PARTIAL GROUP II ACTUATION FROM FUSE REMOVAL FOR OOS WORK.
EVENT DATE: 010591 REPORT DATE: 020491 NSSS: GE TYPE: BWR

(NSIC 220975) ON 1/5/91, AT 1820 HOURS, UNIT ONE WAS IN THE SHUTDOWN MODE AT 0% OF RATED CORE THERMAL POWER. AT THAT TIME SEVERAL FUSES (FU) WERE PULLED AS PART OF AN OUT OF SERVICE (OOS). PULLING THESE FUSES CAUSED SEVERAL PRIMARY CONTAINMENT ISOLATION VALVES (ISV) TO CLOSE IN ADDITION TO THOSE ADDRESSED IN THE OOS, WHICH CONSTITUTES A PARTIAL GROUP II ISOLATION. AFTER AN EVALUATION TO DETERMINE THE CAUSE OF THE ACTUATION, THE FUSES WERE REINSERTED AND THE AFFECTED VALVES WERE REOPENED. FURTHER CORRECTIVE ACTIONS WILL INCLUDE MORE DETAILED REVIEW OF EQUIPMENT OUT OF SERVICE REQUESTS. ON JANUARY 10, 1991, AT 0755 HOURS, UNIT ONE WAS IN THE REFUEL MODE. AT THAT TIME, CONDUIT CONTAINING WIRES TO THE DRYWELL FLOOR DRAIN SUMP DISCHARGE ISOLATION VALVE LIMIT SWITCHES WAS MOVED. THE MOVEMENT OF A CHAFFED WIRE INSIDE THE CONDUIT CREATED AN ELECTRICAL SHORT AND CAUSED A FUSE TO BLOW. THIS CAUSED TWO OF THE SAME VALVES TO CLOSE THAT WERE AFFECTED IN THE JANUARY 5 EVENT, WHICH CONSTITUTED ANOTHER PARTIAL GROUP II ISOLATION. THE FUSE WAS REPLACED AND THE AFFECTED VALVES WERE REOPENED. FURTHER CORRECTIVE ACTIONS WILL INCLUDE REPLACEMENT OF THE CHAFFED WIRE PRIOR TO RESTORING THE SYSTEM TO SERVICE. THIS REPORT IS BEING SUBMITTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(IV).

[161] QUAD CITIES 2 DOCKET 50-265 LER 90-011
UNIT SCRAM FROM IRM-13 AND 16 HIGH-HIGH DUE TO PERSONNEL INATTENTION.
EVENT DATE: 102790 REPORT DATE: 112890 NSSS: GE TYPE: BWR

(NSIC 221047) ON 10/27/90 AT 1559 HOURS, UNIT TWO REACTOR SCRAMMED FROM INTERMEDIATE RANGE MONITORS (IRM) 13 AND 16 HIGH-HIGH SIGNALS. THE STATION WAS IN THE PROCESS OF RETURNING TO NORMAL OPERATION FOLLOWING THE DISCONTINUATION OF A TURBINE TORSIONAL TEST. WHILE REDUCING REACTOR PRESSURE TO RETURN THE TURBINE ELECTRO-HYDRAULIC CONTROL (EHC) SYSTEM TO NORMAL, THE NUCLEAR STATION OPERATION (NSO) DID NOT REALIZE THE REACTOR HAD GONE SUBCRITICAL. AS REACTOR PRESSURE DECREASED BELOW 800 PSIG, THE NSO BEGAN WITHDRAWING CONTROL RODS TO INCREASE REACTOR PRESSURE. THE ROD WITHDRAWALS RESULTED IN A SHORT PERIOD AND THE IRM SCRAM. THE PRIMARY CAUSE OF THE EVENT WAS PERSONNEL ERROR. CONTRIBUTING CAUSES WERE INEFFECTIVE COMMUNICATIONS AND MANAGEMENT OVERSIGHT, INSUFFICIENT TRAINING, AND THE ON-SITE REVIEW PROCESS. CORRECTIVE ACTIONS COMPLETED INCLUDED: AN IN-DEPTH DISCUSSION OF THE EVENT, ADDITIONAL MANAGEMENT OVERSIGHT, REMEDIAL TRAINING, AND AN INDEPENDENT IN-DEPTH INVESTIGATION OF THE EVENT. FURTHER CORRECTIVE ACTIONS WILL INCLUDE: TRAINING ON THIS EVENT DURING LICENSE REQUALIFICATION, PROCEDURE ENHANCEMENT, PERSONNEL COUNSELING, ASSESSMENT OF REACTIVITY MANAGEMENT TRAINING, COMMUNICATIONS ENHANCEMENT, PROCEDURALIZED TURN-OVER CHECKLISTS, AND A COMMITTEE TO ADDRESS PROCEDURE ADHERENCE. THIS REPORT IS BEING SUBMITTED IN ACCORDANCE WITH 10CFR 50.73(A)(2)(IV).

[162] QUAD CITIES 2 DOCKET 50-265 LER 90-013
TORUS LEVEL BELOW TECHNICAL SPECIFICATION LIMIT DUE TO PROCEDURE DEFICIENCY.
EVENT DATE: 121490 REPORT DATE: 011491 NSSS: GE TYPE: BWR

(NSIC 220701) ON 12/14/90, AT 2015 HOURS, UNIT TWO WAS IN THE RUN MODE AT 98% OF RATED CORE THERMAL POWER. AT THIS TIME, THE SHIFT ENGINEER (SE) NOTICED THAT SUPPRESSION POOL (TORUS) LEVEL WAS LOW. PROCEDURE QOS 2300-1, HPCI MONTHLY AND QUARTERLY TEST, WAS IN PROGRESS AND REQUIRED THE SUPPRESSION POOL LEVEL BE REDUCED AS A PREREQUISITE. AFTER REFERRING TO THE GRAPH OF SUPPRESSION POOL LEVEL VERSUS DRYWELL TO SUPPRESSION CHAMBER DIFFERENTIAL PRESSURE GRAPH, IT WAS DISCOVERED THAT LEVEL WAS BELOW THE TECHNICAL SPECIFICATION REQUIRED LEVEL. A LEVEL INCREASE WAS IMMEDIATELY INITIATED AND AT 2045 HOURS WAS WITHIN THE ACCEPTABLE LIMITS. DUE TO THE SUPPRESSION POOL LOW LEVEL, TECHNICAL SPECIFICATION 3.0 WAS ENTERED. THE CAUSE OF THIS EVENT WAS DETERMINED TO BE DUE TO A PROCEDURE DEFICIENCY. THE PROCEDURE STATED TO MAINTAIN SUPPRESSION POOL

LEVEL WITHIN TECHNICAL SPECIFICATION LIMITS; HOWEVER, IT DID NOT REFER THE OPERATOR TO THE GRAPH THAT WAS TO BE UTILIZED TO ENSURE COMPLIANCE WITH THE REQUIREMENT. A CONTRIBUTING CAUSE WAS PERSONNEL ERROR. THE NUCLEAR STATION OPERATOR (NSO), WAS AWARE OF THE TECHNICAL SPECIFICATION LIMIT, HOWEVER, HE DID NOT CONSIDER THE EFFECT THAT THE DIFFERENTIAL PRESSURE WOULD HAVE ON LEVEL. CORRECTIVE ACTIONS WILL INCLUDE TRAINING, AND PROCEDURES ENHANCEMENT. THIS REPORT IS BEING SUBMITTED IN ACCORDANCE WITH 10 CFR 50.73(A)(2)(I)(B).

[163] QUAD CITIES 2 DOCKET 50-265 LER 90-014
TORUS LEVEL SIGHTGLASS LEFT VALVED IN DUE TO PERSONNEL ERROR CAUSING A CONTAINMENT INTEGRITY VIOLATION.
EVENT DATE: 122090 REPORT DATE: 012191 NSSS: GE TYPE: BWR

(NSIC 220748) ON DECEMBER 20, 1990, AT 0500 HOURS UNIT TWO WAS IN THE RUN MODE AT 94 PERCENT OF RATED CORE THERMAL POWER. OPERATING PERSONNEL DETERMINED THAT A PRIMARY CONTAINMENT VIOLATION EXISTED BECAUSE THE TORUS LEVEL SIGHTGLASS WAS LEFT VALVED IN SINCE 0040 HOURS ON DECEMBER 17, 1990. THE VALVING ERROR WAS CORRECTED UPON DISCOVERY. THE SIGHTGLASS IS NOT SEISMICALLY QUALIFIED AND THEREFORE CANNOT FUNCTION AS A CONTAINMENT BOUNDARY. AN EMERGENCY NOTIFICATION SYSTEM (ENS) PHONE NOTIFICATION WAS COMPLETED AT 0558 HOURS UNDER 10CFR50.72(B)(1)(I)(B). AFTER FURTHER REVIEW, THE ENS SHOULD HAVE BEEN COMPLETED IN ACCORDANCE WITH 10CFR50.72(B)(1)(II)(B). THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR DURING A VALVING EVOLUTION. ALTHOUGH NOT CONTRIBUTING CAUSES TO THIS EVENT, INCORRECT LABELING OF THE SIGHTGLASS, AND POOR LIGHTING IN THE AREA WERE FOUND. CORRECTIVE ACTION WILL INCLUDE ENHANCING TRAINING AND STRESSING THE IMPORTANCE OF PERFORMING ADEQUATE INDEPENDENT VERIFICATIONS. A DISCUSSION WITH THE INDIVIDUALS INVOLVED WILL BE HELD AND DISCIPLINARY ACTION WILL BE CONSIDERED. FURTHER CORRECTIVE ACTION WILL BE TO CORRECT THE LACK OF LABELING AND LABELING ERRORS, AND INVESTIGATE THE NEED FOR ADDITIONAL LIGHTING NEAR THE SIGHTGLASS. THIS REPORT IS BEING SUBMITTED AS REQUIRED BY 10CFR50.73(A)(2)(II)(B).

[164] QUAD CITIES 2 DOCKET 50-265 LER 90-015
HPCI STEAM DRAIN LINES AND SUPPORTS FOUND OUTSIDE FSAR COMPLIANCE DUE TO AN INADEQUATE ORIGINAL SEISMIC EVALUATION.
EVENT DATE: 122190 REPORT DATE: 012191 NSSS: GE TYPE: BWR

(NSIC 220750) ON DECEMBER 21, 1990 AT 1030 HOURS, UNIT TWO WAS IN THE RUN MODE AT APPROXIMATELY 96 PERCENT OF RATED CORE THERMAL POWER. AT THIS TIME, SARGENT AND LUNDY (S&L) INFORMED THE STATION THAT THE UNIT TWO HIGH PRESSURE COOLANT INJECTION (HPCI) STEAM LINE DRAIN PIPING AND SUPPORTS DID NOT MEET THE FINAL SAFETY ANALYSIS REPORT (FSAR) ALLOWABLES FOR THERMAL AND SEISMIC LOADINGS. OPERABILITY WAS VERIFIED FOR UNIT TWO, THUS NO IMMEDIATE ACTION WAS REQUIRED BY THE STATION. AN EMERGENCY NOTIFICATION SYSTEM (ENS) PHONE NOTIFICATION WAS COMPLETED AT 1058 HOURS AS REQUIRED BY 10CFR50.72(B)(1)(II)(B). THE CAUSE OF THIS EVENT IS DUE TO AN INADEQUATE ORIGINAL SEISMIC EVALUATION OF THE DRAIN LINE. CORRECTIVE ACTIONS FOR THIS EVENT ARE TO INSTALL ADDITIONAL SUPPORTS TO COMPLY WITH FSAR THERMAL AND SEISMIC ALLOWABLES. THE UNIT ONE HPCI STEAM LINE DRAIN PIPING AND SUPPORTS WERE ANALYZED AND FOUND TO MEET FSAR ALLOWABLES. THIS REPORT IS BEING SUBMITTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(II)(B).

[165] QUAD CITIES 2 DOCKET 50-265 LER 91-001
HPCI DECLARED INOPERABLE DUE TO SIGNIFICANT VALVE PACKING LEAK OF MOTOR OPERATED VALVE MO 2-2301-5.
EVENT DATE: 010291 REPORT DATE: 013091 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: QUAD CITIES 1 (BWR)
VENDOR: CRANE PACKING CO.

(NSIC 220979) ON JANUARY 2, 1991, AT 0143 HOURS, UNIT TWO WAS IN THE RUN MODE AT 92 PERCENT RATED CORE THERMAL POWER. WHILE ON ROUNDS, AN EQUIPMENT ATTENDANT (EA) DISCOVERED A SEVERE PACKING LEAK ON THE HIGH PRESSURE COOLANT INJECTION (HPCI) TURBINE (TRB) STEAM SUPPLY ISOLATION MOTOR OPERATED (MO) VALVE (ISV) 2-2301-5. THE NUCLEAR STATION OPERATOR (NSO) CLOSED THE VALVE. HPCI WAS DECLARED INOPERABLE AND AN OUTAGE REPORT WAS INITIATED. MECHANICAL MAINTENANCE

(MM) REPLACED THE PACKING ON THE VALVE AND DETERMINED A STEAM CUT IN THE VALVE STEM HAD WORN THE PACKING AWAY. AT 2005 HOURS, JANUARY 3, 1991 THE HPCI MO-2-2301-5 VALVE WAS SUCCESSFULLY TESTED AND DECLARED OPERABLE. THE OUTAGE REPORT WAS TERMINATED. CORRECTIVE ACTIONS INCLUDE REPAIRING AND/OR REPLACING THE VALVE STEM. NRC NOTIFICATION OF THE EVENT VIA THE EMERGENCY NOTIFICATION SYSTEM (ENS) PHONE WAS MADE AT 0407 HOURS, JANUARY 2, 1991, TO COMPLY WITH THE REQUIREMENTS OF 10CFR50.72(B)(2)(III)(D). THIS REPORT IS BEING SUBMITTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(V)(D)

[166] RANCHO SECO DOCKET 50-312 LER 90-003
EXCEEDING THE DESIGN BASIS OF THE PLANT DUE TO EXTREME LOW OUTSIDE TEMPERATURES.
EVENT DATE: 122790 REPORT DATE: 011791 NSSS: BW TYPE: PWR

(NSIC 220706) THE MINIMUM DESIGN TEMPERATURE AT RANCHO SECO IS 19F (DRY BULB). ON DECEMBER 22, 1990, FROM APPROXIMATELY 0300 HOURS, TO 0900 HOURS, THE OUTDOOR TEMPERATURE, AT THE SITE DROPPED BELOW THE 19F DESIGN BASIS TEMPERATURE TO AS LOW AS 15F. SYSTEM WALKDOWNS DETERMINED THAT, WITH THE EXCEPTION OF FIRE SUPPRESSION ZONE 35A, ALL REQUIRED PLANT SYSTEMS WERE SUITABLE FOR CONTINUED USE AND CAPABLE OF FULFILLING THEIR FUNCTION IN THE LONG-TERM DEFUELED CONDITION. APPROXIMATELY 75 LEAKS WERE IDENTIFIED, HOWEVER, THE DAMAGE DID NOT AFFECT THE ABILITY OF THE AFFECTED SYSTEMS TO PERFORM THEIR INTENDED FUNCTION. THE DISTRICT FORMED TEAMS TO CONDUCT SYSTEM WALKDOWNS AND DOCUMENT ALL SIGNS OF LEAKAGE OR PHYSICAL DAMAGE TO PRESSURE BOUNDARY PARTS. DAMAGE IS BEING CORRECTED, AS APPROPRIATE, USING THE WORK REQUEST PROCESS. IN ADDITION, A 2 GPM LEAK FROM THE RADWASTE SYSTEM RAN INTO A STORM DRAIN RESULTING IN A MINOR OFFSITE RELEASE OF RADIOACTIVE LIQUID. AS REQUIRED BY THE FIRE PROTECTION PLAN, THE SHIFT SUPERVISOR IMPLEMENTED COMPENSATORY MEASURES FOR FIRE SUPPRESSION ZONE 35A. ENGINEERING EVALUATE THE EFFECTS OF THE REDUCED TEMPERATURES TO DETERMINE IF ADDITIONAL PHYSICAL WORK IS REQUIRED FOR AFFECTED PLANT SYSTEMS. THE EXTREME LOW TEMPERATURES DID NOT RESULT IN AN UNREVIEWED SAFETY QUESTION. THERE WERE NO HEALTH OR SAFETY CONSEQUENCES AS A RESULT OF THIS EVENT.

[167] RIVERBEND 1 DOCKET 50-458 LER 90-003 REV 02
UPDATE ON INADEQUATE THERMO-LAG FIRE BARRIER ENVELOPES SURROUNDING SAFE SHUTDOWN CIRCUITS PER TS 3/4.7.7.
EVENT DATE: 020690 REPORT DATE: 020491 NSSS: GE TYPE: BWR

(NSIC 220924) DURING THE PERFORMANCE OF SURVEILLANCE TEST PROCEDURE STP-000-3602 ON 02/06/90 AND 02/08/90 WITH THE UNIT IN OPERATIONAL CONDITION 1 (FULL POWER), IT WAS FOUND THAT SEVERAL MINOR DEFICIENCIES EXISTED IN THE THERMO-LAG FIRE BARRIER ENVELOPES AROUND REDUNDANT SAFE SHUTDOWN CIRCUITS. THESE DEFICIENCIES CONSISTED OF SMALL HOLES, CRACKS AND UNFILLED SEAMS IN THE THERMO-LAG MATERIAL. A FIRE WATCH HAD ALREADY BEEN ESTABLISHED IN AREAS UTILIZING THERMO-LAG AS A FIRE BARRIER. GSU IS CURRENTLY WORKING WITH THE VENDOR TO RESOLVE THE IDENTIFIED DISCREPANCIES WHICH OCCURRED DURING CONSTRUCTION AND THE DEFICIENT THERMO-LAG BARRIERS. FIRE TESTS WERE CONDUCTED DURING NOVEMBER AND DECEMBER 1990 ON TYPICAL CONFIGURATIONS OF INSTALLED BARRIERS USED IN THE PLANT. REPAIR METHODS WERE DEVELOPED WHERE APPLICABLE. THIS REVISION TO LER 90-003 IS SUBMITTED TO PROVIDE THE RESULTS OF FIRE BARRIER TESTING. THE COMBINATION OF THE CABLE JACKET PROPERTIES, THE CONTROL OF TRANSIENT COMBUSTIBLES, THE USE OF SUPPRESSION SYSTEMS IN THE PLANT, AND THE MINOR NATURE OF THE DEFECTS IN THE BARRIERS PROVIDES ASSURANCE THAT PLANT SAFETY AND THE HEALTH AND SAFETY OF THE PUBLIC HAS NOT BEEN JEOPARDIZED.

[168] RIVERBEND 1 DOCKET 50-458 LER 90-004 REV 02
UPDATE ON ENGINEERED SAFETY FEATURE ACTUATIONS DUE TO TRIPPING OF A TOPAZ INVERTER UNIT.
EVENT DATE: 021190 REPORT DATE: 013091 NSSS: GE TYPE: BWR
VENDOR: TOPAZ ELECTRONICS

(NSIC 220854) AT 1009 ON 02/11/90, WITH THE PLANT AT 100 PERCENT POWER OPERATIONAL CONDITION 1), THE DIVISION II EMERGENCY 125 VDC BUS EXPERIENCED A VOLTAGE SPIKE WHICH CAUSED A TOPAZ INVERTER UNIT (1E12A-PS1) TO TRIP, RESULTING

IN A LOSS OF POWER TO SPECIFIC INSTRUMENTATION ON CONTROL ROOM PANEL H13-P618 (DIVISION II). THIS EVENT OCCURRED COINCIDENT WITH A SCHEDULED PREVENTIVE MAINTENANCE TASK (PM) ON DIVISION II BATTERY CHARGER (ENB*CHGR1B) WHEN THE FLOAT/EQUALIZE SWITCH ON THE CHARGER WAS MOVED FROM THE FLOAT POSITION TO THE EQUALIZE POSITION. UPON RESTORATION OF THE INVERTERS, MULTIPLE DIVISION II ENGINEERED SAFETY FEATURE (ESF) ACTUATIONS OCCURRED. THEREFORE, THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(IV). CORRECTIVE ACTIONS INCLUDED REVISING THE PREVENTIVE MAINTENANCE TASK FREQUENCIES, DEVELOPING NEW PREVENTIVE MAINTENANCE TASKS TO INCLUDE CHECKING OF THE INVERTER TRIP SETPOINTS, TROUBLESHOOTING OF THE BATTERY CHARGER, AND DEVELOPMENT OF LOAD LISTS FOR THE TOPAZ INVERTERS. OPERATIONS PERSONNEL PROPERLY RESPONDED TO THIS EVENT BY LIMITING THE NUMBER OF ESF SYSTEM ACTUATIONS. THOSE ESF SYSTEMS WHICH DID ACTUATE RESPONDED PER DESIGN. THEREFORE, THIS EVENT DID NOT ADVERSELY AFFECT THE HEALTH AND SAFETY OF THE PUBLIC.

[169] RIVERBEND 1 DOCKET 50-458 LER 90-041
OPERABILITY OF CONTAINMENT ISOLATION VALVE INDETERMINATE DUE TO IMPROPERLY INSTALLED TORQUE SWITCH.
EVENT DATE: 111390 REPORT DATE: 121390 NSSS: GE TYPE: BWR
VENDOR: LIMITORQUE CORP.
VELAN VALVE CORP.

(NSIC 220910) AT 1716 ON 11/13/90 WITH THE UNIT IN OPERATIONAL CONDITION 5 (REFUELING), DURING DIAGNOSTIC SIGNATURE TESTING OF THE REACTOR WATER CLEAN UP (RWCU) SUPPLY LINE INBOARD CONTAINMENT ISOLATION VALVE 1G33*MOVF001, IT WAS DETERMINED THAT THE VALVE WAS NOT DEVELOPING THE VENDOR CALCULATED THRUST REQUIRED TO FULLY CLOSE UNDER DESIGN BASIS CONDITIONS DUE TO IMPROPER INSTALLATION OF THE TORQUE SWITCH. THEREFORE, THE OPERABILITY OF THE VALVE AT THAT TIME IS INDETERMINATE. THIS REPORT IS SUBMITTED PURSUANT TO 10CFR50.73(A)(2)(I)(B) AS OPERATION PROHIBITED BY THE TECH SPECS. THIS EVENT WAS CAUSED BY PRIOR IMPROPER INSTALLATION OF THE TORQUE SWITCH. THE TORQUE SWITCH WAS REMOVED AND REINSTALLED PROPERLY. FUNCTIONALITY OF THE TORQUE SWITCH WAS VERIFIED BY SIGNATURE TESTING. ISOLATION OF THE OUTBOARD ISOLATION VALVE ALONE WOULD BE SUFFICIENT TO PREVENT THE RELEASE OF RADIOACTIVITY. IN ADDITION, THE CLOSURE TIME OF THE OUTBOARD ISOLATION VALVE IS SHORTER THAN THE INBOARD VALVE. THUS, IN A DESIGN BASIS ACCIDENT, THE INBOARD VALVE PROBABLY WOULD NOT HAVE BEEN SUBJECTED TO THE FULL DIFFERENTIAL PRESSURE. THEREFORE, THIS EVENT DID NOT ADVERSELY AFFECT THE HEALTH AND SAFETY OF THE PUBLIC.

[170] RIVERBEND 1 DOCKET 50-458 LER 90-042 REV 01
UPDATE ON IMPROPERLY RESTORED BARRIERS FOR HIGH RADIATION AREAS.
EVENT DATE: 111690 REPORT DATE: 010791 NSSS: GE TYPE: BWR

(NSIC 220695) ON FOUR SEPARATE DATES, 11/16/90, 11/23/90, 11/29/90 AND 12/06/90, THERE WERE FIVE OCCURRENCES IN WHICH TECHNICAL SPECIFICATION REQUIRED RADIATION AREA BARRIERS WERE DISCOVERED TO HAVE BEEN RESTORED IMPROPERLY. THE BARRIERS WERE LOCATED AT THE ENTRANCES TO FOUR HIGH RADIATION AREAS (HRAS) AND ONE VERY HIGH RADIATION AREA (VHRA). AS A RESULT, THESE AREAS WERE IN A CONDITION THAT IS PROHIBITED BY TECHNICAL SPECIFICATIONS 6.12.1 AND 6.12.2, RESPECTIVELY. THEREFORE, THIS REPORT IS SUBMITTED PURSUANT TO 10CFR50.73(A)(2)(I)(B) AS OPERATION PROHIBITED BY THE TECHNICAL SPECIFICATIONS. THE ROOT CAUSE FOR THESE INCIDENTS WAS THAT PERSONNEL FAILED TO PROPERLY RESTORE THE BARRIERS FOR UNDETERMINED REASONS. GSU HAS CONCLUDED THAT A GENERIC PROBLEM EXISTS WITH REGARD TO POSTED RADIATION PROTECTION BARRIERS. AS A RESULT, A TASK FORCE HEADED BY THE DIRECTOR-OF-RADIOLOGICAL PROGRAMS HAS BEEN ESTABLISHED TO DETERMINE THE CAUSAL FACTORS ASSOCIATED WITH THESE INCIDENTS. VERIFICATIONS OF TECHNICAL SPECIFICATION REQUIREMENTS WERE PERFORMED WITH NO VIOLATIONS. THE PROBLEM REFLECTED IN THESE INCIDENTS HAS NO OPERATIONAL IMPACT. THEREFORE, THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT ADVERSELY AFFECTED BY THESE INCIDENTS. OPERATIONAL CONDITIONS 5, 4, 2 AND 1 FOR OCCURRENCES ON 11/16, 11/23, 11/29, AND 12/6, RESPECTIVELY.

[171] RIVERBEND 1 DOCKET 50-458 LER 90-048
EQUIPMENT QUALIFICATION LIFETIMES OF TWO HYDROGEN IGNITERS LOCATED IN THE UPPER DRYWELL APPARENTLY EXCEEDED.
EVENT DATE: 122490 REPORT DATE: 020191 NSSS: GE TYPE: BWR

(NSIC 220936) ON 12/24/90, IT WAS DETERMINED THAT THE EQUIPMENT QUALIFICATION (EQ) OPERABILITY STATUS OF TWO HYDROGEN IGNITERS (1HCS*IGN28A AND 1HCS*IGN30B) APPARENTLY EXCEEDED THEIR EQ LIFETIMES DURING CYCLE 3. ASSOCIATED CABLES APPARENTLY EXCEEDED THEIR EQ LIFETIMES DURING CYCLE 2. THIS IS BECAUSE THE TEMPERATURE DATA OBTAINED FROM THERMOCOUPLES INSTALLED DURING REFUELING OUTAGE 3 INDICATED TEMPERATURES HIGHER THAN THOSE ORIGINALLY USED TO CALCULATE THE QUALIFIED LIFETIMES OF THE IGNITERS AND CABLES. THEREFORE, THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(I)(B) AS OPERATION PROHIBITED BY THE TECHNICAL SPECIFICATIONS. SIX HYDROGEN IGNITERS AND THEIR ASSOCIATED CABLES WERE REPLACED DURING THE THIRD REFUELING OUTAGE. THE REPLACEMENT EQUIPMENT IS QUALIFIED THROUGH CYCLE 4. GSU WILL CONTINUE TO MONITOR TEMPERATURES IN THE UPPER DRYWELL THROUGH CYCLE 4. EQ PREVENTIVE MAINTENANCE TASKS (EQPMS) WILL BE REVISED AS NECESSARY BASED ON THE TEMPERATURE TRENDING DATA. BASED ON THE DESIGN OF THE HYDROGEN CONTROL CAPABILITY AT RBS, ANALYSIS RESULTS, AND THE REQUIREMENTS OF EMERGENCY OPERATING PROCEDURE (EOP)-002, GSU CONCLUDES THAT ADEQUATE HYDROGEN CONTROL CAPABILITY EXISTED DURING THIS EVENT. THEREFORE, THIS EVENT DID NOT ADVERSELY AFFECT THE HEALTH AND SAFETY OF THE PUBLIC.

[172] ROBINSON 2 DOCKET 50-261 LER 90-006 REV 01
UPDATE ON BREACH OF CONTAINMENT INTEGRITY DUE TO FAILURE OF THE PERSONNEL AIR LOCK DOOR.
EVENT DATE: 031990 REPORT DATE: 020891 NSSS: WE TYPE: PWR
VENDOR: CHICAGO BRIDGE AND IRON COMPANY

(NSIC 220957) ON MARCH 18, 1990, UNIT NO. 2 WAS OPERATING AT 60 PERCENT POWER. LICENSEE OPERATIONS PERSONNEL RECEIVED AN ALARM ON "A" PENETRATION PRESSURIZATION SYSTEM (PPS) HEADER FLOW. DUE TO PREVIOUS OCCURRENCES, CONTAINMENT PERSONNEL AIR LOCK INNER DOOR LEAKAGE WAS SUSPECTED TO BE THE CAUSE. ON MARCH 19, 1990, LICENSEE MAINTENANCE PERSONNEL ENTERED THE CONTAINMENT PERSONNEL AIR LOCK TO REPAIR THE INNER DOOR SEAL. THE LICENSEE ENTERED PLANT TECHNICAL SPECIFICATION 3.0 IN ORDER TO ENTER THE AIRLOCK. REPAIRS TO THE INNER DOOR SEAL AND LATCH MECHANISM WERE MADE, AND THE AIR LOCK WAS RETURNED TO SERVICE. AN INVESTIGATION INTO THE CAUSE OF THE LEAKAGE WAS INITIATED, BUT THE ROOT CAUSE COULD NOT BE CONFIRMED AT THAT TIME DUE TO THE FACT THAT THE OPERATING STATUS OF THE PLANT LIMITED WORK ON THE AIRLOCK. RESOLUTION OF THE ISSUE WAS ACTIVELY PURSUED DURING THE 1990 REFUELING OUTAGE WHEN ACCESS TO THE AIRLOCK WAS READILY AVAILABLE. AS A RESULT OF THIS INVESTIGATION, IT WAS DETERMINED THAT THE AIRLOCK LEAKAGE WAS ATTRIBUTED TO THE CONTINUOUS APPLICATION OF PRESSURE IN THE INNERSPACE OF THE DOUBLE GASKET SEALS IN A DIRECTION WHICH IS OPPOSITE TO THAT WHICH THE AIRLOCK IS DESIGNED TO RESIST. THIS LER IS SUBMITTED PURSUANT TO 10CFR50.73(A)(2)(I)(B).

[173] ROBINSON 2 DOCKET 50-261 LER 90-012 REV 01
UPDATE ON POTENTIAL OF INADEQUATE NET POSITIVE SUCTION HEAD FOR SAFETY INJECTION PUMPS.
EVENT DATE: 092890 REPORT DATE: 012491 NSSS: WE TYPE: PWR

(NSIC 220984) ON SEPTEMBER 25, 1990, WITH H.B. ROBINSON UNIT NO. 2 IN COLD SHUTDOWN FOR A SCHEDULED REFUELING OUTAGE, A SPECIAL TEST WAS PERFORMED TO DETERMINE THE CAPABILITY OF THE ONE SAFETY INJECTION PUMP INJECTING INTO THREE COLD LEGS OF THE REACTOR COOLANT SYSTEM, WHICH COULD BE AN EXPECTED ALIGNMENT DURING CERTAIN DESIGN BASIS LOSS OF COOLANT ACCIDENTS. THE RESULTS OF THE TEST INDICATED THE POTENTIAL FOR INADEQUATE NET POSITIVE SUCTION HEAD (NPSH) AVAILABLE FOR EITHER SAFETY INJECTION PUMP. BASED ON THIS TEST, BOTH SAFETY INJECTION PUMPS WERE CONSERVATIVELY DECLARED INOPERABLE. IT SHOULD BE NOTED THAT THE PLANT WAS IN A MODE WHERE THE PUMPS WERE NOT REQUIRED TO BE OPERABLE, AND THERE WERE NO SAFETY IMPLICATIONS TO THE PUBLIC. THE CAUSE FOR THE POTENTIAL FOR PUMP RUNOUT COULD NOT BE INITIALLY DETERMINED DUE TO THE UNAVAILABILITY OF INFORMATION NECESSARY TO ADEQUATELY ASSESS THE SIGNIFICANCE OF THE CONCERN. RESOLUTION OF THIS ISSUE WAS ACTIVELY PURSUED, AND DURING NOVEMBER, 1990, ADDITIONAL TESTING

WAS CONDUCTED. UPON ANALYSIS OF THIS TESTING, IT WAS DETERMINED THAT THE SI PUMPS COULD IN FACT OPERATE PROPERLY AND PERFORM THEIR SAFETY FUNCTIONS AS REQUIRED.

[174] SALEM 1 DOCKET 50-272 LER 90-035
TECH SPEC SURVEILLANCE FREQUENCY NONCOMPLIANCE DUE TO PERSONNEL ERROR.
EVENT DATE: 100590 REPORT DATE: 110590 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 2 (PWR)

(NSIC 220888) ON 10/5/90, DURING THE COURSE OF THE TECH SPEC SURVEILLANCE AUDIT PROJECT, IT WAS DISCOVERED THAT THE POSITION INDICATION SURVEILLANCE, FOR THE PRESSURIZER PRESSURE OPERATED RELIEF VALVES (PORVS) AND THE PORV'S BLOCK VALVES, HAS NOT BEEN PERFORMED IN ACCORDANCE WITH TECH SPEC TABLE 4.3-11 PURSUANT TO A NOTATION IN TECH SPEC 3.3.3.7 TABLES 3.3-11A AND B. SUBSEQUENTLY, TECH SPEC TABLE 3.3-11 ACTION 2 WAS ENTERED. A CHANNEL FUNCTIONAL SURVEILLANCE FOR POSITION INDICATION IS REQUIRED TO BE PERFORMED QUARTERLY AS PER TECH SPEC TABLE 4.3-11. TABLE 3.3-11A AND B NOTATION ** REQUIRES ONE OF THE OPERABLE CHANNELS TO BE AN ALTERNATE MEANS OF DETERMINING PORV, PORV BLOCK, OR SAFETY VALVE POSITION BY EITHER TAILPIPE TEMPERATURES FOR THE VALVES. PRESSURIZER RELIEF TANK TEMPERATURE OR PRESSURIZER RELIEF TANK LEVEL. CONTRARY TO THE REQUIREMENTS OF THE NOTATION, THE CHANNEL FUNCTIONAL SURVEILLANCE HAS BEEN PERFORMED EVERY REFUELING, NOT QUARTERLY, FOR EITHER THE TAILPIPE TEMPERATURE FOR THE VALVES, PRESSURIZER RELIEF TANK (PRT) TEMPERATURE, OR THE PRT LEVEL. THE APPARENT CAUSE OF THIS EVENT IS ATTRIBUTED TO PERSONNEL ERROR OVER THE IMPLEMENTATION OF A 1981 TECH SPEC AMENDMENT. THE AMENDMENT ADDRESSED THE REQUIREMENT TO UTILIZE THE ALTERNATE METHODS IN INDICATING PORV AND PORV BLOCK VALVE POSITION.

[175] SALEM 1 DOCKET 50-272 LER 90-040
WRONG RELAY INSTALLED DUE TO PERSONNEL ERROR.
EVENT DATE: 111490 REPORT DATE: 013091 NSSS: WE TYPE: PWR
VENDOR: STRUTHERS DUNN, INC.

(NSIC 220981) ON 11/12/90, A GROUND FAULT INDICATION WAS RECEIVED IN THE CONTROL ROOM FOR BOTH "B" AND "C" 125 VDC VITAL BUSSES. ON 11/15/90, TROUBLESHOOTING REVEALED THE SOURCE OF THE GROUND FAULT INDICATION WAS AN INSTALLED 125V AC RELAY COIL. THIS AC RELAY COIL WAS FOUND IN THE DC CIRCUITRY FOR THE NO. 11 STEAM GENERATOR FEEDWATER PUMP (SGFP) CONTROL CIRCUIT. THE RELAY WHICH SHOULD HAVE BEEN INSTALLED WAS A 125V DC. VISUAL INSPECTION OF THE AC RELAY COIL REVEALED THAT IT HAD BEEN DAMAGED BY ITS USE IN THE DC CIRCUIT THEREBY PREVENTING ITS OPERATION. WITH THIS AC RELAY INSTALLED, THE EMERGENCY TRIPS FOR THE NO. 11 SGFP AND THE AUTO START OF THE NOS. 11 AND 12 MOTOR DRIVEN AUXILIARY FEEDWATER (MDAFW) PUMPS ON THE LOSS OF BOTH SGFPS FUNCTIONS WERE INOPERABLE (A FUNCTION REQUIRED BY TECH SPEC TABLE 3.3-3, ALTHOUGH NOT TAKEN CREDIT FOR IN THE ACCIDENT ANALYSIS). ALSO, ENGINEERING ANALYSIS HAS SHOWN THAT INSTALLATION OF A JUMPER (PER TECH SPECS) ON 11/14/90 COMBINED WITH THE INSTALLATION OF THE WRONG RELAY (AC TYPE) ESTABLISHED A CONNECTION BETWEEN THE -125V DC "C" VITAL BUS AND THE +125 VDC "B" VITAL BUS. ANALYSIS OF THIS HAS SHOWN THAT A COMMON MODE FAILURE COULD NOT OCCUR. THE ROOT CAUSE OF THIS EVENT IS ATTRIBUTED TO AN ISOLATED CASE OF PERSONNEL ERROR WHICH HAD OCCURRED IN APRIL 1990. MAINTENANCE DEPARTMENT MANAGEMENT HAS REVIEWED THIS EVENT. IT WILL BE REVIEWED WITH APPLICABLE PERSONNEL.

[176] SALEM 1 DOCKET 50-272 LER 90-026 REV 04
UPDATE ON ASME CODE CLASS 2 AND 3 PIPING LEAKAGE CAUSED BY EQUIPMENT FAILURE.
EVENT DATE: 113090 REPORT DATE: 011691 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 2 (PWR)

(NSIC 220874) THIS LER ADDRESSES SEVERAL OCCURRENCES OF ASME CODE 2 AND 3 PIPING LEAKAGE. BASED UPON DISCUSSION WITH THE NRC, NOTIFICATIONS WERE MADE FOR EACH OCCURRENCE AS AGREED IN ACCORDANCE WITH CODE OF FEDERAL REGULATIONS 10CFR 50.72. IN ALL CASES, SALEM UNIT 1 TECH SPEC 3.4.10.1 ACTIONS B AND C AND SALEM UNIT 2 TECH SPEC 3.4.11.1 ACTION C WERE COMPLIED WITH. THE SALEM UNIT 1 AND UNIT 2 TECH SPEC FOR "STRUCTURAL INTEGRITY" ARE IDENTICAL EXCEPT FOR THEIR NUMBER (I.E.,

3.4.10.1 VS. 3.4.11.1). THE ROOT CAUSE OF THE LISTED ASME CODE 2 AND 3 COMPONENT LEAKAGE HAS BEEN ATTRIBUTED TO EQUIPMENT FAILURE. THE EQUIPMENT FAILURE COMPONENT SW LEAKS WERE THE RESULT OF EROSION/CORROSION FACTORS. THE 11 RHR PUMP SUCTION PRESSURE GAGE TUBING FAILURE WAS THE RESULT OF METAL FATIGUE. THE COMPONENTS WHICH EXHIBITED LEAKAGE WERE DECLARED INOPERABLE IN ACCORDANCE WITH TECH SPECS. THE COMPONENTS WERE NOT DECLARED OPERABLE UNTIL COMPLETION OF REPAIRS, WHICH WERE DONE IN ACCORDANCE WITH THE ASME CODE FOR CLASS CODE COMPONENTS. THE REQUIREMENTS OF THE TECH SPECS WERE COMPLIED WITH IN ALL CASES. AN ONGOING PROGRAM, AT SALEM GENERATING STATION, FOR THE UPGRADE OF SERVICE WATER SYSTEM PIPING IS CONTINUING. THE SCOPE AND PRIORITIZATION OF PIPE REPLACEMENT IS REVIEWED AND MODIFIED, AS APPLICABLE, BASED UPON ROUTINE INSPECTION ACTIVITIES AND THE LEAKS IDENTIFIED IN THIS REPORT.

[177] SALEM 2 DOCKET 50-311 LER 90-029 REV 01
UPDATE ON LOSS OF BOTH STEAM GENERATOR FEEDWATER PUMPS DUE TO INADEQUATE PREVENTIVE MAINTENANCE.
EVENT DATE: 062890 REPORT DATE: 013091 NSSS: WE TYPE: PWR
VENDOR: ADD EON INC.
BROWN BOVERI

(NSIC 220939) AT 0032 HOURS ON 6/28/90, DURING SALEM UNIT 2 POWER ASCENSION OPERATIONS, A REACTOR TRIP OCCURRED FOLLOWING THE LOSS OF BOTH STEAM GENERATOR FEEDWATER PUMPS (SGFPS) RESULTING FROM FAILURE OF 2F 4160-480/277 VOLT TRANSFORMER. DUE TO THE LOSS OF BOTH SGFP'S "LOW STEAM GENERATOR LEVEL COINCIDENT WITH STEAM FLOW/FEED FLOW MISMATCH" HAD OCCURRED CAUSING THE REACTOR TRIP. A MAIN STEAMLINE ISOLATION WAS MANUALLY INITIATED TO REDUCE AN EXCESSIVE COOLDOWN RATE FOLLOWING THE REACTOR TRIP. MAIN STEAM ISOLATION VALVES (MSIVS) 21 AND 24MS167 DID NOT CLOSE ON THE INITIAL ATTEMPT; HOWEVER, THE OPERATOR AGAIN DEPRESSED THE MAIN STEAMLINE ISOLATION PUSHBUTTONS (THIS TIME FOR AN EXTENDED PERIOD) AND THE VALVES CLOSED. THE UNIT WAS STABILIZED IN MODE 3 (HOT STANDBY). THE ROOT CAUSE OF THE TRANSFORMER FAILURE WAS INADEQUATE PREVENTIVE MAINTENANCE. THE PREVENTIVE MAINTENANCE PROGRAM IS BEING REVISED TO PROVIDE ROUTINE, DOCUMENTED INSPECTION AND CLEANING OF TRANSFORMER COILS. ALSO, THE ROUTINE ELECTRICAL TRANSFORMER TESTING IS BEING REVISED TO INCLUDE LOW FREQUENCY-INDUCED TEST AND CORONA DETECTION TO BETTER CHECK INSULATION AND INTEGRITY FOR INCIPIENT FAILURES. INVESTIGATION OF THE MSIV CLOSURE CONCERN REVEALED THE MSIV ISOLATION LOGIC DID NOT MEET DESIGN BASIS AS PER IEEE STANDARD 279. THE MSIV CIRCUITRY WAS MODIFIED ON SALEM UNITS 1 AND 2 TO RESOLVE THE IDENTIFIED CONCERNS.

[178] SALEM 2 DOCKET 50-311 LER 90-042
SERVICE WATER HEADER INOPERABLE DUE TO EQUIPMENT FAILURE.
EVENT DATE: 122090 REPORT DATE: 011891 NSSS: WE TYPE: PWR

(NSIC 220722) ON 12/20/90, A SERVICE WATER (SW) SYSTEM THROUGH WALL LEAK ON THE INLET PIPE TO THE NO. 21 COMPONENT COOLING (CC) PUMP ROOM COOLER (UPSTREAM OF THE 21SW128 CC PUMP ROOM COOLER SW INLET VALVE) OCCURRED. SUBSEQUENTLY, NO. 21 SW HEADER WAS ISOLATED TO STOP THE LEAK. WITH NO. 21 SW HEADER INOPERABLE, TWO (2) GROUPS OF CONTAINMENT FAN COIL UNITS (I.E., NOS. 21 AND 22 CFCUS) AND THE NO. 21 CONTAINMENT SPRAY (CS) PUMP ROOM COOLER ARE MADE INOPERABLE (I.E., NO SW COOLING FLOW). WITH NO. 21 CS PUMP ROOM COOLER INOPERABLE, THE NO. 21 CS PUMP IS CONSIDERED INOPERABLE. SINCE TECHNICAL SPECIFICATION 3.6.2.3 ACTION B COULD NOT BE MET, WITH TWO (2) GROUPS OF CFCUS AND ONE CS PUMP INOPERABLE, TECHNICAL SPECIFICATION ACTION STATEMENT 3.0.3 WAS ENTERED. ALSO, WITH BOTH HIGH HEAD SAFETY INJECTION PUMPS INOPERABLE, THE TECHNICAL SPECIFICATION 3.5.2 ACTION STATEMENTS DO NOT APPLY. ONE OF THE TRAINS HAD BEEN DECLARED INOPERABLE SOLELY DUE TO AN INOPERABLE EMERGENCY POWER SUPPLY (MAINTENANCE OF 2B DIESEL GENERATOR); THEREFORE, TECHNICAL SPECIFICATION ACTION STATEMENT 3.0.5 APPLIED. NO. 22 CENTRIFUGAL CHARGING PUMP (CCP) WAS INOPERABLE DUE TO INOPERABILITY OF THE NO. 21 SW HEADER AND NO. 21 CCP WAS DECLARED INOPERABLE DUE TO INOPERABILITY OF THE 2B DIESEL GENERATOR. PER THE TECHNICAL SPECIFICATION ACTION STATEMENTS, A UNIT SHUTDOWN WAS INITIATED. THE ROOT CAUSE OF THIS EVENT IS ATTRIBUTED TO EQUIPMENT FAILURE.

[179] SALEM 2 DOCKET 50-311 LER 90-043
ALL ANALOG ROD CONTROL INDICATION LOST DUE TO EQUIPMENT FAILURE.
EVENT DATE: 122090 REPORT DATE: 011691 NSSS: WE TYPE: PWR

(NSIC 220723) ON 12/20/90 AT 0154 HOURS, DURING FULL POWER OPERATION, ALL CONTROL ROOM ANALOG ROD POSITION INDICATOR (ARPI) INDICATION WAS LOST. SUBSEQUENTLY, TECH. SPEC. 3.1.3.2.1. WAS REVIEWED. THE ACTION STATEMENTS DO NOT ADDRESS INOPERABILITY OF MORE THAN ONE ARPI; THEREFORE, TECH. SPEC. ACTION STATEMENT 3.0.3 WAS ENTERED. SEVERAL OVERHEAD ALARMS WERE RECEIVED IN THE CONTROL ROOM WHEN THIS EVENT OCCURRED. THE ROOT CAUSE OF THIS EVENT IS EQUIPMENT FAILURE. A CHART RECORDER'S RIBBON CABLE INSULATION WORE THROUGH TO ITS WIRE, SUBSEQUENTLY SHORTING THE EXPOSED WIRE TO GROUND. THIS RESULTED IN THE OPENING OF THE BREAKER (NO. CB-1, A 2 AMP BREAKER) WHICH REMOVED POWER TO THE ARPI INDICATORS. AT THE TIME OF THE EVENT, A MAINTENANCE-I&C TECHNICIAN WAS INSERTING THE CHART RECORDER (LOCATED ON THE CONTROL ROOM CONSOLE) INTO ITS CHASSIS. THE CUT CABLE SUPPLIES POWER FROM THE NO. 21 MISCELLANEOUS AC (MAC) 115 VAC DISTRIBUTION CABINET NO. 34 BREAKER (15 AMPS) TO THE RECORDER. WHEN THE SHORT OCCURRED, THE NO. 34 BREAKER DID NOT TRIP OPEN; HOWEVER, THE CB-1 BREAKER, LOCATED IN THE ARPI CABINET 87, TRIPPED OPEN RESULTING IN THE LOSS OF ALL ARPI INDICATION. THE POWER SOURCE TO THE CB-1 BREAKER COMES FROM BREAKER NO. 41 (15 AMPS) ALSO LOCATED IN THE NO. 21 MAC CABINET. THE CB-1 BREAKER WAS RESET AND CLOSED TO RESTORE POWER TO THE ARPI INDICATORS AND TECH. SPEC. ACTION STATEMENT 3.0.3 WAS THEN EXITED.

[180] SALEM 2 DOCKET 50-311 LER 90-044
CONTAINMENT VENTILATION ISOLATION DUE TO EQUIPMENT/DESIGN CONCERNS.
EVENT DATE: 122390 REPORT DATE: 012191 NSSS: WE TYPE: PWR
VENDOR: VICTOREEN INSTRUMENT DIVISION

(NSIC 220814) ON 12/23/90 AND 12/24/90, A CONTAINMENT PURGE/PRESSURE-VACUUM RELIEF (CP/P-VR) SYSTEM ISOLATION SIGNALS (AN ESF FUNCTION) WERE RECEIVED. THE RADIATION MONITORING SYSTEM (RMS) CHANNEL WHICH GENERATED THE SIGNALS COULD NOT BE IDENTIFIED. THE SUSPECT CHANNELS (2R12A AND 2R41C) WERE INSTRUMENTED WITH STRIP CHARTS/RECORDERS SUCH THAT IF ANOTHER SIGNAL OCCURRED, IT WOULD BE IDENTIFIED TO A SPECIFIC CHANNEL. THE CHANNELS WERE DECLARED OPERABLE AND RETURNED TO SERVICE ON DECEMBER 27, 1990 AT 1102 HOURS. AFTER APPROXIMATELY TWO (2) WEEKS OF MONITORING, WITHOUT RECURRANCE OF A CP/P-VR SYSTEM ISOLATION SIGNAL, THE MONITORING EQUIPMENT WAS REMOVED. ON 1/14/91 AT 1015 HOURS, ANOTHER CP/P-VR SYSTEM ISOLATION SIGNAL WAS RECEIVED. THE SOURCE AGAIN COULD NOT BE SPECIFICALLY IDENTIFIED. THE CP/P-VR SYSTEM ISOLATION FUNCTION WAS RESET AT 1023 HOURS ON 1/14/91. PRIOR TO BLOCKING THE SUSPECT CHANNELS, AT 1027 HOURS, ANOTHER ISOLATION SIGNAL WAS RECEIVED. SUBSEQUENTLY, THE 2R12A AND THE 2R41C CHANNELS ISOLATION SIGNAL CAPABILITY WAS BLOCKED AND APPROPRIATE ACTION STATEMENTS WERE ENTERED. THE ROOT CAUSE OF THIS EVENT IS ATTRIBUTED TO EQUIPMENT/DESIGN CONCERNS. PROBLEMS WITH THE RMS HAVE BEEN EXPERIENCED AS INDICATED IN PRIOR LERS. TESTING SHOWED THAT THE 2R12A CHANNEL WAS INTERMITTENTLY FAILING LOW. THE MICROPROCESS CHIP PIN CONNECTIONS, IN THE LOW FAIL CIRCUIT, WERE FOUND TO BE EITHER LOOSE OR DIRTY.

[181] SAN ONOFRE 1 DOCKET 50-206 LER 90-020
STATION BATTERY TERMINAL CONNECTION RESISTANCE MEASUREMENT METHODOLOGY.
EVENT DATE: 102090 REPORT DATE: 012891 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SAN ONOFRE 2 (PWR)
SAN ONOFRE 3 (PWR)

(NSIC 220955) THIS LER IS BEING SUBMITTED AS A VOLUNTARY LER BASED ON THE POTENTIAL GENERIC IMPLICATIONS INDICATED BELOW. ON 10/20/90, DURING A REVIEW OF THE POST-MAINTENANCE TEST REQUIREMENTS FOR WORK PLANNED TO REPLACE THE SCREW TYPE BATTERY JUMPER CABLE LUGS WITH STANDARD CRIMPED TYPE LUGS, THE EXISTING BATTERY TERMINAL RESISTANCE MEASUREMENT METHODOLOGY WAS QUESTIONED. THE CONCERN WAS WHETHER OR NOT TO SPECIFY MEASURING THE RESISTANCE ACROSS THE JUMPER CABLE LUG TO TERMINAL PLATE CONNECTIONS ALONG WITH THE BATTERY POST TO TERMINAL PLATE CONNECTION MEASUREMENTS WHICH ARE CURRENTLY BEING UTILIZED TO MONITOR BATTERY CONNECTION DEGRADATION. HISTORICALLY SCE, AS WELL AS OTHER UTILITIES SAMPLED, HAS INTERPRETED THE IEEE STD. 450-1980 AND TECHNICAL SPECIFICATION (TS) BATTERY

TERMINAL RESISTANCE MEASUREMENT REQUIREMENT AS BEING THE RESISTANCE MEASURED BETWEEN THE POST-TO-PLATE CONNECTION, WHERE BATTERY CORROSION IS NORMALLY OBSERVED. AN ENGINEERING REVIEW SUBSEQUENTLY DETERMINED THAT AN INCONSISTENT INTERPRETATION EXISTS WITHIN THE INDUSTRY AS TO IMPLEMENTATION OF THE IEEE AND TS REQUIREMENTS FOR MEASURING BATTERY TERMINAL RESISTANCE. SCE'S PARTICIPATION IN THE IEEE BATTERY WORKING GROUP AND DISCUSSIONS WITH OTHER UTILITIES HAVE CAUSED SCE TO CONCLUDE THAT THE ADDITIONAL CABLE LUG TO TERMINAL PLATE RESISTANCE MEASUREMENTS CAN PROVIDE USEFUL DATA.

[182] SAN ONOFRE 2 DOCKET 50-361 LER 90-015 REV 01
UPDATE ON AUXILIARY FEEDWATER PUMP GRAVITY FEED LUBE OIL COOLING SYSTEM PIPING MIS-ASSEMBLY.
EVENT DATE: 112389 REPORT DATE: 011891 NSSS: CE TYPE: PWR

(NSIC 220822) ON 12/4/90, WITH UNIT 2 AT 100% POWER, DURING A ROUTINE ANNUAL AUXILIARY FEEDWATER PUMP (AFWP) GRAVITY FEED LUBE OIL COOLING SYSTEM (GFLOCS) FUNCTIONAL TEST, THE GFLOCS STORAGE TANK CONTENTS DRAINED INTO THE AFWP 2P-504 MOTOR BEARINGS IN LESS THAN THE REQUIRED 30 MINUTES. THIS SYSTEM IS NOT REQUIRED FOR NORMAL LUBRICATION AND WAS INSTALLED PURSUANT TO FULL TERM OPERATING LICENSE (FTOL) CONDITION 2.C(25) TO ENVIRONMENTALLY QUALIFY THE MOTOR DRIVEN AFWPS FOR THE ENVIRONMENT WHICH WOULD RESULT IN THE UNLIKELY EVENT OF A HIGH ENERGY STEAM LINE BREAK (HELB) INSIDE THE AFWP ROOM. A SUBSEQUENT INVESTIGATION REVEALED THAT THE GFLOCS SUPPLY LINE TO THE AFWP OUTBOARD END MOTOR BEARING HAD BEEN INSTALLED INCORRECTLY FOLLOWING A MOTOR INSPECTION DURING THE PREVIOUS REFUELING OUTAGE (11/89). SCE'S INVESTIGATION INTO THE CAUSE OF THIS EVENT HAS IDENTIFIED SEVERAL DEFICIENCIES IN OUR MAINTENANCE, MAINTENANCE RESTORATION, AND POST-MAINTENANCE PRETEST PROCESSES. THESE DEFICIENCIES COLLECTIVELY CONTRIBUTED TO THE INCORRECT REASSEMBLY OF THE GFLOCS AND SUBSEQUENT FAILURE TO DETECT THIS CONDITION BY EITHER POST-MAINTENANCE VERIFICATION OR PRETEST. CORRECTIVE ACTIONS TO PREVENT RECURRENCE INCLUDE: TRAINING, PROGRAM AUDIT, PROGRAM AND POLICY REVIEWS/CHANGES, AND PROCEDURE CHANGES.

[183] SAN ONOFRE 2 DOCKET 50-361 LER 90-011
EROSION-CORROSION INDUCED PIPE WALL THINNING.
EVENT DATE: 060190 REPORT DATE: 022091 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: SAN ONOFRE 3 (PWR)

(NSIC 221038) THIS VOLUNTARY LICENSEE EVENT REPORT IS BEING SUBMITTED TO DESCRIBE SCE'S ACTIONS IN RESPONSE TO A RECENT OCCURRENCE OF PIPE WALL THINNING AS A RESULT OF EROSION-CORROSION (E-C) PROCESSES. SCE IMPLEMENTED AN E-C MONITORING PROGRAM TO DETECT E-C RELATED PIPE WALL THINNING PRIOR TO COMPONENT FAILURE. ALTHOUGH THIS PROGRAM HAS BEEN GENERALLY SUCCESSFUL IN PREDICTING WALL THINNING DUE TO E-C, EXCESSIVE PIPE WALL THINNING WAS OBSERVED IN JUNE 1990 IN PIPING AT A LOCATION WHICH WAS NOT INCLUDED IN THE MONITORING PROGRAM. AS A RESULT, SCE INITIATED A REEVALUATION OF THE E-C MONITORING PROGRAM. TO-DATE AS A RESULT OF THIS REEVALUATION, TWO CONCLUSIONS HAVE BEEN REACHED. FIRST, THE METHOD IN WHICH THE E-C PROGRAM WAS IMPLEMENTED AT SONGS RESULTED IN INAPPROPRIATELY LIMITING THE PIPING SYSTEMS SUBJECT TO MONITORING. SECOND, WE IDENTIFIED THAT INTERNAL GEOMETRIC DISCONTINUITIES ARE CAUSING ACCELERATED E-C AT LOCATIONS WHICH ARE NOT PREDICTED BY OUR PROGRAM AND WOULD NOT BE PREDICTED BY OTHER E-C PROGRAMS COMMONLY IN USE WITHIN THE INDUSTRY. OUR REEVALUATION IS CONTINUING. PROGRAM MODIFICATIONS WILL BE IMPLEMENTED FOLLOWING COMPLETION OF THE REEVALUATION.

[184] SEABROOK 1 DOCKET 50-443 LER 87-006 REV 01
UPDATE ON ESF ACTUATION - LOSS OF POWER TO VITAL INSTRUMENT PANEL.
EVENT DATE: 021987 REPORT DATE: 013091 NSSS: WE TYPE: PWR
VENDOR: ELGAR, CORP.

(NSIC 220851) ON 2/19/87, AT 3:20 AM EST, WHILE SEABROOK STATION WAS IN MODE 3, SEVERAL ALARMS WERE RECEIVED INDICATING A GROUND. IN ATTEMPT TO IDENTIFY THE SOURCE OF THE GROUND, WHICH APPEARED TO ORIGINATE FROM UNINTERRUPTIBLE POWER SUPPLY 1E, THE SUPPLY FROM DC BUS 11A WAS VERIFIED TO BE THE SUPPLYING SOURCE TO VITAL INSTRUMENT PANEL 1E THROUGH THE INVERTER. THE AC SUPPLY BREAKER TO UPS 1E

FROM MOTOR CONTROL CENTER E512 WAS THEN OPENED WHICH RESULTED IN THE INVERTER OUTPUT BEING INTERRUPTED FOR APPROXIMATELY 2 SECONDS. THIS LOSS OF POWER RESULTED IN MULTIPLE ESF ACTUATIONS: I.E., ISOLATION OF THE NON-NUCLEAR SAFETY PORTIONS OF THE PRIMARY COMPONENT COOLING WATER SYSTEM, ACTUATION OF THE CONTROL ROOM EMERGENCY CLEAN-UP FILTER SYSTEM, AND ISOLATION OF THE CONTAINMENT VENTILATION SYSTEM. ALL ESF SYSTEMS FUNCTIONED AS DESIGNED. THE ROOT CAUSE WAS DETERMINED TO BE THE ACTIVATION OF THE UPS 1E TRANSDUCER BOARD DC UNDERVOLTAGE OPTICAL ISOLATOR BY EXTRANEOUS PLANT ELECTRICAL NOISE CAUSED BY AN INTERMITTENT AC SYSTEM GROUND. THE DC UNDERVOLTAGE ISOLATOR IN TURN ACTIVATED THE FAULT PROTECTION CIRCUITRY WHICH INITIATED THE TWO SECOND OUTPUT INTERRUPTION. A DESIGN CHANGE WAS IMPLEMENTED TO CHANGE THE VALUE OF THE BIAS RESISTORS AND REDUCE THE SENSITIVITY OF ALL THE UPS DC UNDERVOLTAGE OPTICAL ISOLATORS. THIS IS THE FIRST OCCURRENCE OF THIS TYPE AT SEABROOK STATION.

[185] SEABROOK 1 DOCKET 50-443 LER 90-024
ACTUATION OF THE CONTROL ROOM EMERGENCY AIR CLEANUP AND FILTRATION SUBSYSTEM.
EVENT DATE: 110290 REPORT DATE: 113090 NSSS: WE TYPE: PWR

(NSIC 220894) ON 11/2/90, AT 3:32 A.M. EST, AN ENGINEERED SAFETY FEATURES (ESF) ACTUATION OCCURRED CAUSING THE CONTROL ROOM NORMAL MAKEUP AIR SUBSYSTEM (CBA) TO TRANSFER TO THE CONTROL ROOM EMERGENCY AIR CLEANUP AND FILTRATION SUBSYSTEM. THE ESF ACTUATION OCCURRED DURING THE PERFORMANCE OF A SLAVE RELAY SURVEILLANCE TEST. THE OPERATOR PERFORMING THE TEST INADVERTENTLY OPERATED THE WRONG TEST SWITCH (S911). THIS TEST SWITCH ENERGIZES SLAVE RELAY K609 WHICH, IN TURN, ACTUATES CBA-FN-16A. THE OPERATOR SHOULD HAVE OPERATED TEST SWITCH S912 AS REQUIRED BY THE PROCEDURE. TEST SWITCH S912 IS LOCATED IMMEDIATELY TO THE RIGHT OF TEST SWITCH S911. THE ROOT CAUSE HAS BEEN DETERMINED TO BE PERSONNEL ERROR INVOLVING A LACK OF ATTENTION TO DETAIL. A CONTRIBUTING CAUSE WAS THE LOCATION OF THE TEST SWITCH LABELS. THE OPERATOR INVOLVED WAS COUNSELLED ON THE EVENT AND THE NEED FOR INCREASED ATTENTION TO DETAIL. THE TEST SWITCH LABELS WILL BE RELOCATED TO AN AREA WHERE IT WILL NOT BE OBSCURED FROM VIEW DURING OPERATION OF THE TEST SWITCHES. ADDITIONALLY, A TRAINING DEVELOPMENT RECOMMENDATION (TDR) HAS BEEN GENERATED TO INCLUDE A DISCUSSION WITH THE OPERATING CREWS OF THIS EVENT AND OTHER RECENT EVENTS INVOLVING A LACK OF ATTENTION TO DETAIL.

[186] SEQUOYAH 1 DOCKET 50-327 LER 90-023
FAILURE TO ENSURE PRESERVATION OF CONTAINMENT INTEGRITY WHEN THE OPERATING VALVE WAS DIFFERENT THAN THE TESTED CONFIGURATION AS A RESULT OF INADEQUATE PROCEDURE REVISIONS.
EVENT DATE: 100490 REPORT DATE: 110590 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 220905) ON 10/4/90, WITH UNIT 1 IN MODE 1 AT 100% POWER AND UNIT 2 SHUTDOWN FOR THE CYCLE 4 REFUELING OUTAGE, IT WAS DISCOVERED THAT THE OPERATING VALVE CONFIGURATION OF THREE FLOW TRANSMITTERS IN THE RESIDUAL HEAT REMOVAL CONTAINMENT SPRAY SYSTEM WAS DIFFERENT THAN THE CONFIGURATION THAT HAD BEEN PREVIOUSLY LEAK RATE TESTED. THE FLOW TRANSMITTERS HAD BEEN INSTALLED DURING THE LAST REFUELING OUTAGE OF EACH UNIT. THE AS-TESTED VALVE CONFIGURATION HAD BEEN INCORRECTLY SPECIFIED IN THE ASSOCIATED SURVEILLANCE INSTRUCTION (SI). THE VALVE CONFIGURATIONS WERE RETURNED TO THE AS-TESTED CONFIGURATION ON UNIT 1 TO ENSURE CONTAINMENT INTEGRITY WITH THE UNIT AT POWER. THE PRESSURE INTEGRITY OF ALL THREE FLOW TRANSMITTERS WAS SUBSEQUENTLY DEMONSTRATED USING AN ALTERNATE TEST METHOD. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO INADEQUATE PROCEDURE REVISIONS FOLLOWING INSTALLATION OF THE FLOW TRANSMITTERS ON EACH UNIT. THE SI HAS BEEN REVISED TO CORRECTLY SPECIFY THE TEST CONFIGURATION FOR THE SUBJECT FLOW TRANSMITTERS.

[187] SEQUOYAH 1 DOCKET 50-327 LER 90-026 REV 01
UPDATE ON AN INADVERTENT AUXILIARY BUILDING ISOLATION (ABI) OCCURRED WHEN AN OPERATOR FAILED TO RESET THE ABI SIGNAL SEAL-IN RELAY BEFORE RETURNING A RADIATION MONITOR TO SERVICE.
EVENT DATE: 101390 REPORT DATE: 020191 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 220932) THIS LER IS BEING REVISED TO RESCHEDULE A COMPLETION DATE. ON 10/13/90, AT 1837 EDT WITH UNIT 1 IN MODE 4 FOR A MAINTENANCE OUTAGE AND UNIT 2 IN MODE 6 FOR A REFUELING OUTAGE, AN INADVERTENT AUX. BLDG. ISOLATION (ABI) OCCURRED. THE ABI SIGNAL WAS INITIATED FROM THE SPENT FUEL PIT AREA RADIATION MONITOR (RM) 0-RM-90-103, FOLLOWING THE TRANSFER OF HIGHLY CONTAMINATED TRASH THROUGH THE AREA. DURING THE TRANSFER, THE RM 0-RM-90-103 REACHED ITS HIGH RADIATION SETPOINT. THE RMS HAD BEEN REMOVED FROM SERVICE (BLOCKED) TO PREVENT A SPURIOUS ABI. ONCE THE RM HIGH RADIATION SETPOINT IS REACHED, THE HIGH RADIATION SEAL-IN RELAY RETAINS THE ABI SIGNAL UNTIL THE RELAY IS RESET. AFTER THE TRANSFER OF TRASH WAS COMPLETED, THE OPERATORS PROCEEDED TO UNBLOCK THE RMS BUT FAILED TO RESET THE SEAL-IN RELAY FOR 0-RM-90-103. THE OPERATORS UNBLOCKED THE RMS WITHOUT RESETTING THE RELAYS AND RECEIVED THE INADVERTENT ABI. OPERATIONS' PERSONNEL REALIZED THE CAUSE OF THE ABI AND PROCEEDED TO RECOVER FROM THE EVENT IN ACCORDANCE WITH PLANT PROCEDURES. CORRECTIVE ACTION WAS TO TAKE APPROPRIATE DISCIPLINARY ACTION FOR THE PERSONNEL INVOLVED. A MODIFICATION TO THE ANNUNCIATOR CIRCUITRY TO PREVENT RMS HIGH RADIATION ALARMS FROM BEING CLEARED WITHOUT RESETTING THE SEAL-IN RELAY, WILL BE MADE BY 4/1/91.

[188] SHOREHAM DOCKET 50-322 LER 90-006
INOPERABILITY OF ULTIMATE HEAT SINK.
EVENT DATE: 082090 REPORT DATE: 083190 NSSS: GE TYPE: BWR

(NSIC 220904) THIS SPECIAL REPORT IS SUBMITTED PURSUANT TO TECH SPEC 6.9.2 TO COMPLY WITH TECH SPEC 3.7.1.4. ON 1/30/90 IT WAS DETERMINED THAT TWO SAND PILES WERE WITHIN 100 FEET OF THE CREST OF THE WEST SLOPE OF THE INTAKE CANAL. TECH SPEC SURVEILLANCE REQUIREMENT 4.7.1.4.C CANNOT BE MET DUE TO THESE SAND PILES. THESE SAND PILES WERE FORMED DURING REMOVAL OF DREDGE SPOILS FROM THE ADJACENT AREA. (THE SPOILS HAD BEEN DEPOSITED DURING DREDGING OF THE INTAKE CANAL IN JANUARY OF 1989.) REMOVAL OF THE SPOILS BEGAN 1/29/90 BUT WAS STOPPED BY THE NEW YORK STATE DEPARTMENT OF ENVIRONMENTAL CONSERVATION ON 2/5. SINCE THE REACTOR HAS BEEN SHUTDOWN AND DEFUELED SINCE AUGUST OF 1989 THE PLANT OPERATIONAL CONDITION HAS NOT REQUIRED THE ULTIMATE HEAT SINK TO BE OPERABLE UNTIL 8/20/90, WHEN HANDLING OF IRRADIATED FUEL IN THE SECONDARY CONTAINMENT BEGAN. TO RETURN THE ULTIMATE HEAT SINK TO OPERABILITY THE TWO SAND PILES WILL BE MOVED TO A PERMANENT DISPOSAL SITE. THIS IS EXPECTED TO BE COMPLETED BY 3/31/91.

[189] SHOREHAM DOCKET 50-322 LER 90-009
UNPLANNED ACTUATION OF ESF SYSTEMS DUE TO ELECTRICAL PROTECTION ASSEMBLY BREAKER TRIP.
EVENT DATE: 121190 REPORT DATE: 010991 NSSS: GE TYPE: BWR

(NSIC 220707) AT 1257, ON 12/11/90, THE "A" REACTOR PROTECTION SYSTEM (RPS) BUS WAS DEENERGIZED WHEN ONE OF ITS ELECTRICAL PROTECTION ASSEMBLY (EPA) BREAKERS TRIPPED. THIS CAUSED THE UNPLANNED AUTOMATIC INITIATION OF THE ESF SYSTEMS REACTOR BUILDING STANDBY VENTILATION SYSTEM (RBSVS) "A" AND CONTROL ROOM AIR CONDITIONING (CRAC) "A" AND ALSO CLOSURE OF THE INBOARD CONTAINMENT ISOLATION VALVES FOR THE REACTOR BUILDING FLOOR DRAINS AND REACTOR BUILDING EQUIPMENT DRAINS. PLANT MANAGEMENT PERSONNEL WERE NOTIFIED OF THE EVENT AND THE NRC WAS NOTIFIED AT 1410 PER 10CFR 50.72(B)(2)(II). THE RBSVS AND CRAC INITIATIONS WERE RESET AT 1250 ON 12/13/90 AFTER INITIAL TROUBLESHOOTING HAD BEEN COMPLETED. AFTER THE EVENT, THE "A" RPS MG SET OUTPUT VOLTAGE WAS FOUND TO BE ONLY 117.4V INSTEAD OF 121.5V. 117.4V IS VERY CLOSE TO THE EPA BREAKER UNDERVOLTAGE TRIP POINT. THE CAUSE FOR THIS HAS NOT BEEN DETERMINED.

[190] SHOREHAM DOCKET 50-322 LER 90-010
UNPLANNED ACTUATION OF ENGINEERED SAFETY FEATURE SYSTEMS WHILE LIFTING A JUMPER.
EVENT DATE: 121390 REPORT DATE: 010991 NSSS: GE TYPE: BWR

(NSIC 220816) ON 12/13/90 AT 1038, AN UNPLANNED ACTUATION OF THE ESF SYSTEMS REACTOR BUILDING STANDBY VENTILATION SYSTEM (RBSVS) AND CONTROL ROOM AIR CONDITIONING (CRAC) "B" OCCURRED WHEN REMOVING A JUMPER ACROSS THE LOW REACTOR BUILDING D/P CONTACT IN AN INITIATING CIRCUIT FOR RBSVS AND CRAC. THE REACTOR BUILDING NORMAL VENTILATION SYSTEM WAS BEING OPERATED IN A RECIRCULATION MODE AT

THE TIME. THE JUMPER WAS ORIGINALLY INSTALLED IN ORDER TO PREVENT RBSVS AND CRAC INSTALLATIONS WHICH MIGHT OCCUR IF REACTOR BUILDING PRESSURE CONTROL PROVED TO BE DIFFICULT IN THIS NEWLY DEVELOPED RECIRCULATION MODE. FOLLOWING THE EVENT, PLANT MANAGEMENT PERSONNEL WERE INFORMED AND THE NRC WAS NOTIFIED AT 1150 PER 10CFR50.72 (B)(2)(II). THE CAUSE OF THIS EVENT INVOLVED AN INADEQUATE PROCEDURE AND LAPSES IN COMMUNICATIONS BETWEEN THE PERSONS INVOLVED. CORRECTIVE ACTIONS INCLUDE A PROCEDURE REVISION AND MAKING A REPORT OF THIS EVENT REQUIRED READING FOR ENGINEERS, TECHNICIANS AND MECHANICS.

[191] SOUTH TEXAS 1 DOCKET 50-498 LER 90-026
TRAIN "B" LOSS OF OFFSITE POWER ACTUATION DUE TO BREAKER MALFUNCTION.
EVENT DATE: 121990 REPORT DATE: 020191 NSSS: WE TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 220857) ON DECEMBER 19, 1990, UNIT 1 WAS IN MODE 5. AT 0821 HOURS, A TRAIN "B" LOSS OF OFFSITE POWER ACTUATION OCCURRED WHILE ATTEMPTING TO TRANSFER POWER TO THE STANDBY BUS 1G FROM THE UNIT 2 STANDBY TRANSFORMER TO THE UNIT 1 STANDBY TRANSFORMER. THE UNIT 1 STANDBY TRANSFORMER TO STANDBY BUS SUPPLY BREAKER (ST-160) WAS CLOSED. WHEN ATTEMPTING TO TRIP OPEN THE UNIT 2 STANDBY TRANSFORMER TO STANDBY BUS SUPPLY BREAKER (ST-180), THE BREAKER FAILED TO OPEN. THE STANDBY BUS WAS IN PARALLEL FOR MORE THAN EIGHT SECONDS WHICH AUTOMATICALLY TRIPPED OPEN THE ST-160 BREAKER AND THE STANDBY BUS 1G TO AUXILIARY ESF TRANSFORMER E1B FEEDER BREAKER. THE CAUSE OF THIS EVENT WAS FAILURE OF THE BREAKER TO OPEN DUE TO THE COMBINATION OF THE EXISTENCE OF HARDENED GREASE AND FRICTION BETWEEN THE TRIP ARM LINKAGE AND A METAL COVER WHICH PREVENTED THE BREAKER FROM OPENING BY IMPEDING THE FREE TRAVEL OF THE TRIP ARM LINKAGE. A STEP HAS BEEN ADDED TO THE APPROPRIATE BREAKER PERIODIC MAINTENANCE PROCEDURE TO CHECK FOR BINDING OF THE TRIPPING MECHANISM. PREVENTIVE MAINTENANCE HAS BEEN SCHEDULED TO BE PERFORMED ON THE APPROPRIATE 13.8 KV BREAKERS INCLUDING THE ST-180 BREAKER.

[192] SOUTH TEXAS 2 DOCKET 50-499 LER 90-005 REV 01
UPDATE ON TECHNICAL SPECIFICATION REQUIRED SHUTDOWN DUE TO PRIMARY COOLANT SYSTEM LEAKAGE.
EVENT DATE: 050890 REPORT DATE: 011791 NSSS: WE TYPE: PWR

(NSIC 220833) ON MAY 8, 1990, UNIT 2 WAS IN MODE 1 AT 100 PERCENT POWER. AT APPROXIMATELY 0030 HOURS, IT WAS DETERMINED THAT A SMALL PRESSURE BOUNDARY LEAK (APPROXIMATELY 10 ML/MINUTE) EXISTED ON THE ASME CLASS 2 STEAM GENERATOR C LOWER HEAD DRAIN LINE. A NOTIFICATION OF UNUSUAL EVENT WAS DECLARED AND TECHNICAL SPECIFICATION REQUIRED SHUTDOWN WAS INITIATED TO MODE 5. THE CAUSE OF THIS EVENT WAS HIGH CYCLE FATIGUE FAILURE OF THE WELD ON THE UPSTREAM SIDE OF THE DRAIN VALVE. THE DRAIN VALVES ON BOTH UNITS' STEAM GENERATORS HAVE BEEN REPLACED WITH THREADED AND WELDED CAPS WHICH ARE NOT EXPECTED TO DEVELOP HIGH ALTERNATING STRESSES. THE DESIGN OF THE DRAIN VALVE PIPING HAS BEEN REVIEWED TO ASSURE NO CYCLIC FATIGUE STRESS CONCERNS EXIST. AN EVALUATION OF THE HIGH CYCLE FATIGUE EXPERIENCED ON THE STEAM GENERATOR DRAIN LINE HAS BEEN PERFORMED FOR POTENTIAL GENERIC IMPLICATIONS.

[193] ST. LUCIE 2 DOCKET 50-389 LER 90-006
INADVERTENT ACTUATION OF AUXILIARY FEEDWATER EQUIPMENT DURING MONTHLY TESTING DUE TO TEST INSTRUMENT MALFUNCTION.
EVENT DATE: 121990 REPORT DATE: 011491 NSSS: CE TYPE: PWR
VENDOR: HEWLETT-PARKARD CO.

(NSIC 220713) ON DECEMBER 19, 1990, WITH UNIT 2 IN MODE 1, AT 100% FOLLOWING A REFUELING OUTAGE, AN INADVERTENT ACTUATION OF AUXILIARY FEEDWATER (AFW) SYSTEM EQUIPMENT OCCURRED DURING MONTHLY FUNCTIONAL TESTING. THE 2B AFW PUMP WAS STARTED AND THE STEAM ADMISSION VALVE FROM THE 2A STEAM GENERATOR TO THE 2C AFW PUMP WAS OPENED STARTING THE TURBINE-DRIVEN PUMP. TESTING WAS SUSPENDED AND THE AFW PUMPS WERE RESTORED TO THEIR NORMAL STATUS. NO AFW WAS INJECTED INTO THE STEAM GENERATORS. THE ROOT CAUSE OF THE EVENT WAS EQUIPMENT FAILURE. THE DIGITAL VOLTMETER BEING USED FOR THE TESTING DEVELOPED AN INTERMITTENT INTERNAL FAULT, CAUSING A LOSS OF THE AFW ACTUATION RELAY HOLDING CURRENT. CORRECTIVE

ACTIONS: THE DETAILS OF THE EVENT AND THE TROUBLESHOOTING PLAN WERE REVIEWED BY THE FACILITY REVIEW GROUP. THE TESTING WAS COMPLETED SATISFACTORILY USING A FLOATING, UNGROUNDED METER. THE INSTRUMENT AND CONTROL TESTING PROCEDURE WAS REVISED TO INCLUDE A CAUTIONARY NOTE FOR USING THE FLOATING, UNGROUNDED METER IN THE FUTURE. INSTRUMENT AND CONTROL PERSONNEL WERE NOTIFIED OF THE CHANGES IN THE TEST PROCEDURE.

[194] SURRY 1 DOCKET 50-280 LER 90-004
UNIT 1 REACTOR TRIP/TURBINE TRIP DUE TO DELUGE ACTUATION ON THE "A" MAIN TRANSFORMER AND UNIT 2 MANUAL REACTOR TRIP DUE TO ERRATIC INDIVIDUAL ROD POSITION INDICATIONS.
EVENT DATE: 052290 REPORT DATE: 062190 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)
VENDOR: MOLONEY ELECTRIC COMPANY
WESTINGHOUSE ELECTRIC CORP.

(NSIC 220903) ON 5/22/90 AT 1150 HOURS WITH UNITS 1 AND 2 AT 100% POWER, A FAULT OCCURRED ON THE UNIT 1 "A" MAIN TRANSFORMER AS A RESULT OF AN INADVERTENT ACTUATION OF THE TRANSFORMER'S DELUGE SYSTEM. THE FAULT INITIATED A UNIT 1 GENERATOR DIFFERENTIAL LOCKOUT WHICH IMMEDIATELY INITIATED A TURBINE TRIP/REACTOR TRIP. THE FAULT ALSO RESULTED IN THE LOCKOUT OF THE "A" RESERVE STATION SERVICE TRANSFORMER (RSST). APPROX. 10 SECONDS LATER, THE UNIT 2 CONTROL ROOM OPERATOR INITIATED A MANUAL REACTOR TRIP AFTER OBSERVING ERRATIC CONTROL ROD INDIVIDUAL ROD POSITION INDICATIONS (IRPI). OPERATORS PERFORMED THE APPROPRIATE PLANT PROCEDURES AND QUICKLY STABILIZED THE UNITS FOLLOWING THE TRIPS. THE ERRATIC UNIT 2 IRPI INDICATIONS WERE DUE TO VOLTAGE TRANSIENTS THAT OCCURRED ON BOTH UNITS' EMERGENCY BUSES WHICH WERE CAUSED BY THE UNIT 1 GENERATOR TRIP, THE "A" RSST LOCKOUT, AND SUBSEQUENT MOTOR STARTS. THE #3 EMERGENCY DIESEL GENERATOR AUTOMATICALLY RESTORED POWER TO THE UNIT 1 "J" EMERGENCY BUS WHICH WAS DE-ENERGIZED BY THE "A" RSST LOCKOUT. A FOUR HOUR NON-EMERGENCY REPORT WAS MADE TO THE NRC IN ACCORDANCE WITH 10CFR50.72.

[195] SURRY 1 DOCKET 50-280 LER 90-014 REV 01
UPDATE ON RECIRCULATION SPRAY HEAT EXCHANGERS DECLARED INOPERABLE DUE TO POTENTIALLY INADEQUATE SERVICE WATER FLOW CAUSED BY MACROFOULING.
EVENT DATE: 102390 REPORT DATE: 020691 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)

(NSIC 220982) ON 10/23/90, AT 0103 HOURS, WITH UNIT 1 AT REFUELING SHUTDOWN AND UNIT 2 AT 100% POWER, THE UNIT 1 AND UNIT 2 RECIRCULATION SPRAY HEAT EXCHANGERS (RSHXS) WERE DECLARED INOPERABLE. A SPECIAL TEST OF SERVICE WATER (SW) FLOW TO THE UNIT 1 "B" AND "C" RSHXS HAD INDICATED THAT ADEQUATE SW FLOW MAY NOT BE AVAILABLE DURING A DESIGN BASIS ACCIDENT. THIS COULD DELAY OR PREVENT DEPRESSURIZATION OF THE CONTAINMENT TO A SUBATMOSPHERIC CONDITION DURING A DESIGN BASIS ACCIDENT. IN ACCORDANCE WITH TECH SPEC 3.0.1, UNIT 2 WAS PLACED IN COLD SHUTDOWN. THE REDUCED SW FLOW WAS CAUSED BY MACROFOULING OF THE RSHXS FROM MARINE GROWTH AND FRAGMENTS OF PIPE COATING. THE DEBRIS WHICH FOULED THE RSHXS WAS PRESENT IN THE SW SYSTEM AT TEST INITIATION. FLOW REDUCTION OCCURRED IN THE EARLY STAGES OF THE TESTING AND LONG TERM MACROFOULING TRENDS WERE NOT OBSERVED. THE SW SUPPLY PIPING TO THE UNIT 2 RSHXS WAS CLEANED OF MARINE GROWTH AND LOOSE PIPE COATING. CHANGES IN OPERATION OF THE UNIT 2 SW SYSTEM WILL REDUCE THE POTENTIAL FOR MARINE GROWTH TO RETURN AND TO BE DISLODGED UPON INITIATION OF SW FLOW TO THE RSHXS. SIMILAR MEASURES WILL BE IMPLEMENTED FOR UNIT 1. AN ENHANCED INSPECTION AND MAINTENANCE PROGRAM FOR THE SW SYSTEM IS BEING IMPLEMENTED IN RESPONSE TO NRC GENERIC LETTER 89-13. THIS PROGRAM WILL INCLUDE PROVISIONS FOR BIOFOULING CONTROL.

[196] SURRY 1 DOCKET 50-280 LER 90-019
ALL SIX MAIN FEEDWATER FLOW TRANSMITTERS RENDERED INOPERABLE AS A RESULT OF FAILURE TO FOLLOW PROCEDURES DURING PERFORMANCE OF A HYDROSTATIC TEST.
EVENT DATE: 121890 REPORT DATE: 011491 NSSS: WE TYPE: PWR

(NSIC 220751) ON DECEMBER 18, 1990, WITH UNIT 1 CRITICAL AT 1 X 10-7 AMPERES

INTERMEDIATE RANGE POWER INDICATION FOLLOWING A REFUELING OUTAGE, CONTROL ROOM OPERATORS NOTED THAT ONE OF TWO REDUNDANT MAIN FEEDWATER FLOW CHANNELS FOR THE "A" STEAM GENERATOR WAS INDICATING APPROXIMATELY 0.5 MILLION LBS. MASS/HR FEEDWATER FLOW WHILE INDICATED FLOW SHOULD HAVE BEEN ZERO. INSTRUMENT TECHNICIANS DISPATCHED TO INVESTIGATE THE INDICATION FOUND ALL SIX MAIN FEEDWATER FLOW TRANSMITTERS ISOLATED, EQUALIZED, AND DRAINED. A SIX HOUR LIMITING CONDITION FOR OPERATION (LCO) WAS ENTERED AT 2040 HOURS IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS. THE TRANSMITTERS WERE RETURNED TO SERVICE AT 2131 HOURS AND THE LCO TERMINATED. THIS EVENT HAD NO SAFETY CONSEQUENCES SINCE THE PROTECTIVE FUNCTION PROVIDED BY THE FEEDWATER FLOW TRANSMITTERS (REACTOR TRIP ON LOW STEAM GENERATOR LEVEL WITH STEAM/FEEDWATER FLOW MISMATCH) WOULD STILL HAVE BEEN PERFORMED. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR BY A VIRGINIA POWER TEST ENGINEER WHO FAILED TO FOLLOW CERTAIN PROCEDURE REQUIREMENTS DURING PERFORMANCE OF A HYDROSTATIC TEST. INSTRUMENT TECHNICICANS VERIFIED THAT OTHER SAFETY RELATED SECONDARY INSTRUMENTATION WAS CORRECTLY LINED UP. A STATION POLICY WILL BE ISSUED REGARDING WHO IS RESPONSIBLE FOR PERFORMING VALVE MANIPULATIONS. A ROOT CAUSE EVALUATION IS IN PROGRESS.

[197] SUSQUEHANNA 1 DOCKET 50-387 LER 90-025
SPURIOUS ACTUATION OF THE REACTOR PROTECTION SYSTEM.
EVENT DATE: 110890 REPORT DATE: 121090 NSSS: GE TYPE: BWR

(NSIC 220891) AT 1532 HOURS ON NOVEMBER 8, 1990 WITH UNIT 1 IN CONDITION 4, AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION OCCURRED WHEN THE REACTOR PROTECTION SYSTEM (RPS) SPURIOUSLY ACTUATED. NO CONTROL ROD MOTION RESULTED SINCE ALL CONTROL RODS WERE FULLY INSERTED. THIS EVENT WAS DETERMINED TO BE REPORTABLE PER 10CFR50.73(A)(2)(IV), IN THAT THE SPURIOUS RPS ACTUATION COMPRISED AN UNPLANNED ESF ACTUATION. THERE WERE NO SAFETY CONSEQUENCES OR COMPROMISE TO THE PUBLIC HEALTH OR SAFETY DURING THIS EVENT. AN IMMEDIATE INVESTIGATION WAS CONDUCTED BY SHIFT OPERATING PERSONNEL IN THE PLANT PRIOR TO RESETTING THE RPS ACTUATION LOGIC. AN EVENT REVIEW TEAM WAS FORMED TO PERFORM A COMPREHENSIVE INVESTIGATION OF THE EVENT. POSSIBILITIES FOR BOTH ONE CROSS-DIVISIONAL EVENT AS WELL AS TWO DISCRETE YET SIMULTANEOUS EVENTS ON OPPOSITE RPS DIVISIONS WERE CONSIDERED. EXTENSIVE DOCUMENTATION WAS REVIEWED AND INTERVIEWS WERE CONDUCTED. COMPUTER DATA OR CONTROL ROOM ANNUNCIATORS PROVIDED NO INDICATION OF A LIKELY CAUSE AND THE EVENT REVIEW TEAM WAS UNABLE TO CONCLUSIVELY DETERMINE THE ROOT CAUSE FOR THIS EVENT.

[198] SUSQUEHANNA 1 DOCKET 50-387 LER 90-030 REV 01
UPDATE ON ENTRIES INTO CONDITION 2 WITHOUT COMPLETED SURVEILLANCES.
EVENT DATE: 120290 REPORT DATE: 012891 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: SUSQUEHANNA 2 (BWR)

(NSIC 220847) PLANNED SHUTDOWNS WERE PERFORMED FOR UNIT 1 (12/02/90) AND UNIT 2 (12/15/90 AND 01/05/91) TO CONDUCT CERTAIN EQUIPMENT REPAIRS. PRIOR TO THESE SHUTDOWNS, AN INTERNAL WORK GROUP REVIEW HAD IDENTIFIED THAT THE IRMS, APRMS AND SRMS COULD NOT BE SURVEILLANCE TESTED UNTIL AFTER ENTRY INTO CONDITION 2 OR 3 FROM CONDITION 1 AND NO EXCEPTIONS FOR THESE INSTRUMENTS FROM THE REQUIREMENTS OF TECHNICAL SPECIFICATION SECTIONS 3.0.4 AND 4.0.4 CURRENTLY EXISTED IN THE TECHNICAL SPECIFICATIONS. A REQUEST FOR TEMPORARY WAIVER OF COMPLIANCE FROM THESE REQUIREMENTS HAD BEEN SUBMITTED TO THE NRC BUT WAS DENIED BY THE NRC BY LETTER DATED 11/30/90 SINCE THE REQUEST DID NOT MEET THE REQUIREMENTS OF 10CFR50.91 FOR EMERGENCY OR EXIGENCY CHANGES TO THE TECH SPECS. ENTRY INTO THE TECH SPEC LCO ACTION STATEMENTS FOR THESE INSTRUMENTS RESULTS IN HALF SCRAMS AND ROD BLOCKS THAT UNNECESSARILY INCREASE THE POTENTIAL OF A SCRAM OR OTHERWISE RESTRICT UNIT OPERATION. THEREFORE, IN LIEU OF ENTERING THE LCO ACTION STATEMENTS, PP&L IS REPORTING THE ENTRIES INTO CONDITION 2 OR CONDITION 3 FROM CONDITION 1 WITH INCOMPLETE SURVEILLANCES PURSUANT TO 10CFR50.73(A)(2)(I)(B). THE SUBJECT SURVEILLANCES WERE SUCCESSFULLY COMPLETED SHORTLY AFTER ENTRY INTO CONDITION 2 FOR THE UNIT 1 12/02/90 AND UNIT 2 01/05/91 SHUTDOWNS AND AFTER ENTRY INTO CONDITION 3 FOR THE UNIT 2 12/15/90 SHUTDOWN.

[199] SUSQUEHANNA 1 DOCKET 50-387 LER 90-033
DIESEL GENERATOR 'D' PISTON PIN BUSHING FAILURE.
EVENT DATE: 121990 REPORT DATE: 011891 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: SUSQUEHANNA 2 (BWR)
VENDOR: COOPER ENERGY SERVICES

(NSIC 220826) ON 12/19/90, WITH UNIT 1 AND 2 OPERATING AT 100% POWER, A FAILED PISTON PIN BUSHING WAS FOUND ON THE NO. 7 RIGHT CYLINDER OF THE 'D' EDG WHILE PERFORMING A MAINTENANCE INSPECTION. THIS INSPECTION WAS BEING PERFORMED AFTER APPROX. 43 HOURS OF ENGINE OPERATING TIME FOLLOWING MAJOR ENGINE WORK WHICH HAD BEEN COMPLETED IN 10/90. THE INSPECTION IS A MANUFACTURER'S RECOMMENDED PRACTICE FOLLOWING ANY WORK WHICH DISTURBS A PISTON BOLT ATTACHMENT. ROOT CAUSE INVESTIGATION FOR THE BUSHING FAILURE INCLUDED AN INSTALLATION REVIEW, METALLURGICAL ANALYSES AND MANUFACTURER'S EVALUATION. IT WAS DETERMINED THAT THE MOST PROBABLE CAUSE FOR THE BUSHING FAILURE WAS EITHER AN IMPROPER FIT-UP BETWEEN THE PISTON PIN BUSHING AND PISTON BORE DURING AN EARLIER PISTON REFURBISHING BY THE MANUFACTURER OR A RESTRICTION TO LUBRICATING OIL FLOW TO THE PISTON PIN/BUSHING. THE BUSHING FAILURE WAS DETERMINED BY THE MANUFACTURER AND PP&L TO BE A UNIQUE AND ISOLATED TYPE OF FAILURE AND NOT INDICATIVE OF ANY GENERIC CONCERNS. THIS EVENT IS BEING VOLUNTARILY REPORTED FOR INFORMATION PURPOSES. CORRECTIVE ACTIONS INCLUDED INSPECTIONS TO DETERMINE EXTENT OF DAMAGE AND DEBRIS DISTRIBUTION, REPAIR WORK, DEBRIS CLEANUP AND RETESTING AND RE-INSPECTION. THE MANUFACTURER WILL INCLUDE PISTON PIN BUSHING BORE MEASUREMENT CHECKS ON ALL FUTURE PISTON REFURBISHING.

[200] SUSQUEHANNA 2 DOCKET 50-388 LER 91-002
HPCI OUTBOARD STEAM SUPPLY CONTAINMENT ISOLATION VALVE DECLARED INOPERABLE DUE TO INCORRECT TORQUE SWITCH SETTING.
EVENT DATE: 080190 REPORT DATE: 020891 NSSS: GE TYPE: BWR

(NSIC 220912) AT 1715 HOURS ON 8/1/90, WITH UNIT 2 OPERATING IN CONDITION 1 AT 100% POWER, THE HIGH PRESSURE COOLANT INJECTION (HPCI) STEAM SUPPLY OUTBOARD CONTAINMENT ISOLATION VALVE, HV-255-F003, WAS DECLARED INOPERABLE DUE TO THE CLOSING TORQUE SWITCH BEING IMPROPERLY SET. AS SUCH, LIMITING CONDITION FOR OPERATION 3.6.3 ACTION A WAS ENTERED. ACTIONS WERE IMMEDIATELY TAKEN TO CORRECT THE TORQUE SWITCH SETTING AND THE VALVE WAS DECLARED OPERABLE AT 1935 HOURS. THE ROOT CAUSE OF THE EVENT IS COGNITIVE PERSONNEL ERROR. DURING MAINTENANCE ACTIVITIES ON HV-255-F003, IN SEPTEMBER 1986, THE MINIMUM TORQUE SWITCH SETTING WAS CHANGED FROM 2 1/2 TO 2 7/8 AS A RESULT OF NRC GENERIC LETTER 85-03, HOWEVER, THE CALIBRATION CHART, LOCATED IN THE VALVE, WAS NOT CHANGED TO REFLECT THE NEW RANGE. DURING APRIL 1988, THE ACTUATOR FOR HV-255-F003 WAS OVERHAULED. AS PART OF THE WORK PLAN, DIRECTION WAS GIVEN TO RETURN THE TORQUE SWITCH SETTINGS TO THE MINIMUM SETTING. WITH THE INCORRECT CALIBRATION CHART THE TORQUE SWITCH WAS RETURNED TO 2-1/2. THE CONDITION IS BEING REPORTED UNDER THE PROVISIONS OF 10CFR50.73(A)(2)(I)(B) IN THAT CONTAINMENT ISOLATION VALVE HV-255-F003 WAS INOPERABLE FROM APRIL 9, 1988 UNTIL AUGUST 1, 1990. THIS DETERMINATION WAS MADE ON JANUARY 10, 1991 FOLLOWING RE-EVALUATION OF AN EARLIER OPERABILITY/REPORTABILITY DETERMINATION.

[201] SUSQUEHANNA 2 DOCKET 50-388 LER 91-001
INADVERTENT ISOLATION OF SHUTDOWN COOLING MODE OF RESIDUAL HEAT REMOVAL SYSTEM.
EVENT DATE: 010891 REPORT DATE: 020791 NSSS: GE TYPE: BWR

(NSIC 220950) ON JANUARY 8, 1991 WITH UNIT 1 IN COLD SHUTDOWN, THE RESIDUAL HEAT REMOVAL SYSTEM, OPERATING IN THE SHUTDOWN COOLING MODE, AUTOMATICALLY ISOLATED WHILE TECHNICIANS WERE PERFORMING INSTRUMENT MAINTENANCE. PRESSURE SWITCHES WERE BEING REPLACED DUE TO PROBLEMS WITH SETPOINT DRIFT. THE CAUSE OF THE ISOLATION WAS IMPROPER WORK INSTRUCTIONS WHICH REQUIRED OPENING LINKS AND WHEN THESE LINKS WERE OPENED, THE ISOLATION CIRCUITRY ACTUATED. THIS EVENT WAS DETERMINED TO BE REPORTABLE PER 10CFR50.73(A)(2)(IV) AS AN UNPLANNED ESF ACTUATION. THERE WERE NO SAFETY CONSEQUENCES OR COMPROMISES TO THE HEALTH OR SAFETY OF THE PUBLIC OR PLANT PERSONNEL. PERSONNEL PERFORMING THE WORK IMMEDIATELY RESTORED FROM THE WORK ACTIVITY AND SHUTDOWN COOLING WAS RESTORED TO OPERATION. CONTRIBUTING TO THE IMPROPER WORK INSTRUCTION WAS THE FAILURE TO USE ALL AVAILABLE DRAWINGS. PP&L IS

DEVELOPING A WORK PLANNING GUIDE FOR INSTRUMENT & CONTROLS PERSONNEL AND APPROPRIATE TRAINING WILL BE PERFORMED ON THIS GUIDE.

[202] TURKEY POINT 3 DOCKET 50-250 LER 89-018 REV 01
UPDATE ON PLANT OPERATING OUTSIDE OF ITS DESIGN BASIS DUE TO A DESIGN INADEQUACY OF THE SAFETY INJECTION BLOCK SWITCH.
EVENT DATE: 121289 REPORT DATE: 011791 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: TURKEY POINT 4 (PWR)
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 220860) ON DECEMBER 12, 1989, WITH TURKEY POINT UNIT 3 IN MODE 1 (POWER OPERATION) AT 100 PERCENT POWER AND UNIT 4 IN MODE 1 AT 40 PERCENT POWER, THE PLANT NUCLEAR SAFETY COMMITTEE (PNSC) DETERMINED THAT THE USE OF A SINGLE MANUAL SAFETY INJECTION (SI) BLOCK/UNBLOCK SWITCH (WESTINGHOUSE OT2 SWITCH USING A STACK OF FOUR OT2A CONTACT BLOCKS) FOR BOTH TRAINS OF SI WAS OUTSIDE THE DESIGN BASIS FOR BOTH UNITS. DURING A CONTROL ROOM DESIGN REVIEW CONCERNING PLACEMENT OF CONTROLS ON THE CONTROL ROOM CONTROL BOARDS AT THE POINT BEACH NUCLEAR PLANT, THE USE OF A SINGLE MANUAL SI BLOCK/UNBLOCK SWITCH FOR BOTH SAFETY INJECTION TRAINS WAS QUESTIONED. A SUBSEQUENT REVIEW BY WISCONSIN ELECTRIC ENGINEERING DETERMINED THAT A SINGLE MECHANICAL FAILURE OF THIS SWITCH COULD BLOCK BOTH TRAINS OF SI. ON SEPTEMBER 16, 1988, POINT BEACH ISSUED LICENSEE EVENT REPORT 88-07 DESCRIBING IN DETAIL THEIR REVIEW AND CONCLUSIONS. AFTER BEING CONTACTED BY THE NRC, WESTINGHOUSE NOTIFIED TURKEY POINT AND OTHER APPLICABLE WESTINGHOUSE FACILITIES. TURKEY POINT UNITS 3 AND 4 ARE CURRENTLY SHUT DOWN FOR THE 1991 DUAL UNIT OUTAGE. A SEPARATE SI BLOCK SWITCH WILL BE INSTALLED FOR EACH TRAIN OF SI DURING THIS DUAL UNIT OUTAGE.

[203] VERMONT YANKEE DOCKET 50-271 LER 90-017 REV 01
UPDATE ON AVERAGE POWER RANGE MONITOR MISCALIBRATION DUE TO PERSONNEL ERROR.
EVENT DATE: 101690 REPORT DATE: 021391 NSSS: GE TYPE: BWR

(NSIC 221036) ON OCTOBER 29, 1990 AN ENGINEERING REVIEW OF APRM (EIIS=IG) CALIBRATION DATA OBTAINED DURING PLANT STARTUP IDENTIFIED A MISCALIBRATION AT 1156 AND 1254 HOURS ON OCTOBER 16, 1990 WITH THE PLANT AT 20% POWER. THE AVERAGE POWER RANGE MONITORS (APRMS) WERE MISCALIBRATED LOWER THAN REQUIRED IN TECHNICAL SPECIFICATION SECTIONS 2.1.A.1.A, 2.1.B.1 AND 3.1.B. THE ROOT CAUSE OF THE MISCALIBRATION WAS DUE TO PERSONNEL ERROR ON THE PART OF THE TECHNICIAN PERFORMING THE CALIBRATIONS. THE REVIEW OF THE CALIBRATION DATA AND INITIATION OF THE PRO WERE DELAYED DUE TO STARTUP ACTIVITIES AND UNFAMILIARITY WITH THE REPORTABLE OCCURRENCE PROCESS. CORRECTIVE ACTIONS INCLUDE, TRAINING IN ATTENTION TO DETAIL FOR ALL MEMBERS OF THE DEPARTMENT INVOLVED IN SURVEILLANCE PROCEDURES. THE DEPARTMENT HAS SET UP A SPECIAL BOX FOR TECHNICAL SPECIFICATION SURVEILLANCES AND SUPERVISORS ARE REQUIRED TO REVIEW THESE SURVEILLANCES SOON AFTER COMPLETION. DEPARTMENT PERSONNEL INVOLVED WITH POTENTIALLY REPORTABLE OCCURRENCE DECISIONS HAVE BEEN INSTRUCTED ON THE URGENCY OF TAKING PROMPT ACTIONS.

[204] VERMONT YANKEE DOCKET 50-271 LER 90-015
REACTOR SCRAM DUE TO TURBINE TRIP CAUSED BY A MALFUNCTION IN THE TURBINE EMERGENCY TRIPPING SYSTEM.
EVENT DATE: 110490 REPORT DATE: 120390 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 220887) ON 11/4/90 AT 0120 HOURS, WITH REACTOR POWER AT 94%, WHILE PERFORMING WEEKLY TESTING OF THE TURBINE EMERGENCY GOVERNOR (JC*), A REACTOR SCRAM OCCURRED AS A RESULT OF A TURBINE CONTROL VALVE FAST CLOSURE SIGNAL. THE TURBINE CONTROL VALVE FAST CLOSURE SIGNAL WAS GENERATED BY THE ACCELERATION RELAY AFTER OPERATORS REMOVED A TRIP LOCKOUT FROM THE EMERGENCY TRIP VALVE. THIS OPERATOR ACTION WAS TAKEN BASED ON EMERGENCY TRIP VALVE POSITION INFORMATION LATER IDENTIFIED AS ERRONEOUS DUE TO A MISSING SET SCREW ON A COLLAR IN THE TURBINE FRONT STANDARD LINKAGE. SINCE THE COLLAR WAS NOT ATTACHED TO THE LINKAGE, A SPRING WAS NOT ABLE TO ACT UPON THE LINKAGE AND THE EMERGENCY TRIP VALVE LIMIT SWITCH DID NOT CHANGE STATE AS THE VALVE MOVED. THE PLANT WAS STABILIZED WITHOUT INCIDENT FOLLOWING THE TRANSIENT AND THE TURBINE (TA*) WAS PLACED ON THE TURNING

GEAR AT 0241 HOURS. THE TURBINE VENDOR (GE) ASSISTED PLANT PERSONNEL IN REPAIRING AND TESTING THE MALFUNCTIONING LINKAGE PRIOR TO RESTORING THE TURBINE CONTROL SYSTEM (TGX) TO SERVICE. THE REACTOR WAS RETURNED TO CRITICAL ON 11/4/90 AT 2222 HOURS. A PROCEDURE IS BEING REVISED TO PROVIDE ADDITIONAL OPERATOR ACTIONS TO BE TAKEN IF LIGHT INDICATIONS ARE ABNORMAL. TURBINE OUTAGE MAINTENANCE IS BEING EXPANDED TO INCLUDE AN INSPECTION OF SIMILAR LOCKING COLLARS AND POSITION INDICATING.

[205] VERMONT YANKEE DOCKET 50-271 LER 91-001
ENTRY INTO A HIGH RADIATION AREA BY A RADIATION PROTECTION TECHNICIAN WITHOUT A DOSE RATE MONITORING DEVICE DUE TO PERSONNEL ERROR.
EVENT DATE: 010491 REPORT DATE: 012391 NSSS: GE TYPE: BWR

(NSIC 221009) ON JANUARY 4, 1991, AT APPROXIMATELY 1115 HOURS A RADIATION PROTECTION (RP) TECHNICIAN ENTERED A POSTED HIGH RADIATION AREA, THE RADIOACTIVE WASTE CASK ROOM, TO CHECK THE CONDITION OF THE RESIN AND MOVE A RESIN CASK, WITHOUT A DOSE RATE MONITORING DEVICE, AS REQUIRED BY TECHNICAL SPECIFICATION 6.5.8.1. A RP SUPERVISOR WENT TO THE CASK ROOM TO CHECK ON THE JOB PROGRESS, DISCOVERED THE PROBLEM AND IMMEDIATELY DIRECTED THE TECHNICIAN TO RETURN TO THE STEP-OFF PAD UNTIL A DOSE RATE MONITORING DEVICE COULD BE OBTAINED AS REQUIRED. THE GENERAL AREA DOSE RATE DURING THIS EVENT WAS LESS THAN 100 MR/HR. THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR. THE TECHNICIAN FAILED TO USE A DOSE RATE METER AS REQUIRED BY PROCEDURES AND TECH SPECS WHEN ENTERING A HIGH RADIATION AREA.

[206] VOGTLE 1 DOCKET 50-424 LER 90-023
TRANSFORMER FAILURE RESULTS IN LOSS OF STEAM GENERATOR LEVEL AND MANUAL REACTOR TRIP.
EVENT DATE: 121890 REPORT DATE: 011591 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: VOGTLE 2 (PWR)
VENDOR: BROWN BOVERI
GENERAL ELECTRIC CO.

(NSIC 220828) ON 12-18-90 AT 1934 CST, UNIT WAS OPERATING AT 100% POWER WHEN A 4160/480 VOLT NON-1E TRANSFORMER (1NB10X) EXPERIENCED AN INTERNAL FAULT. THIS FAILURE RESULTED IN A LOSS OF POWER FOR THE SPEED CONTROL CIRCUITRY FOR THE 1B MAIN FEEDWATER PUMP (MFP) TURBINE AND CERTAIN SUPPORT SYSTEMS FOR EMERGENCY DIESEL GENERATOR 1B. FEEDWATER PUMP SPEED, FEEDWATER FLOW, AND STEAM GENERATOR (SG) LEVELS DECREASED. THE REACTOR OPERATOR INITIATED A MANUAL REACTOR TRIP AT 1937 CST AFTER EFFORTS TO MAINTAIN SG LEVELS WERE UNSUCCESSFUL. ALL SAFETY RELATED FUNCTIONS OCCURRED PER DESIGN FOLLOWING THE REACTOR TRIP; HOWEVER, A NON-1E 4160 VOLT BUS FAILED TO AUTOMATICALLY TRANSFER TO THE RESERVE AUXILIARY TRANSFORMERS CAUSING A TEMPORARY LOSS OF VARIOUS NON-1E HOUSE LOADS. TRANSFER OF THE 4160 VOLT BUS WAS COMPLETED MANUALLY AND NORMAL PLANT CONDITIONS WERE ESTABLISHED FOR HOT STANDBY BY 1956 CST. THE ROOT CAUSE FOR THE TRANSFORMER FAILURE IS INDETERMINATE; HOWEVER, SEVERAL SIMILAR TRANSFORMER FAILURES HAVE OCCURRED AT VEGP (REFERENCE LER 50-424/1990-016). THE INVOLVED TRANSFORMERS ARE GE CLASS AA/FA, THREE PHASE, DRY TYPE TRANSFORMERS. THE FAILED TRANSFORMER HAS BEEN REPLACED AND FURTHER STUDY OF POSSIBLE FACTORS WHICH MAY HAVE LED TO THE FAILURE IS IN PROGRESS.

[207] VOGTLE 2 DOCKET 50-425 LER 91-001
PERSONNEL ERROR RESULTS IN MISSED SPECIAL CONDITION SURVEILLANCE.
EVENT DATE: 010291 REPORT DATE: 020491 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 220922) DURING THE PERIOD BETWEEN 1518 CST ON 1-2-91 AND 0743 CST ON 1-7-91, THE ROD POSITION DEVIATION MONITOR ALARM WAS INOPERABLE AND THE SPECIAL CONDITION SURVEILLANCE REQUIREMENT OF TECHNICAL SPECIFICATION (TS) 4.1.3.2, WHICH IS APPLICABLE WHEN THE MONITOR IS INOPERABLE, WAS NOT IMPLEMENTED. THE TS REQUIRES THAT THE DEMAND POSITION INDICATION SYSTEM AND THE DIGITAL ROD POSITION INDICATION SYSTEM BE COMPARED AT LEAST ONCE PER 4 HOURS WHEN THE ROD POSITION DEVIATION ALARM IS INOPERABLE. THE CAUSE OF THIS EVENT WAS A COGNITIVE PERSONNEL

ERROR ON THE PART OF THE BALANCE OF PLANT (BOP) OPERATOR. THE BOP OPERATOR FAILED TO FOLLOW PROCEDURE ON 1-2-91 WHEN HE IMPROPERLY ENTERED A COMPUTER POINT VALUE. CONTRIBUTING TO THIS EVENT WAS THE FAILURE OF A RELAY OUTPUT CIRCUIT CARD IN THE INTERFACE BETWEEN THE DRPI AND THE PROTEUS COMPUTER. THE FAILED RELAY OUTPUT CIRCUIT CARD HAS BEEN REPLACED AND THE BOP OPERATOR HAS BEEN COUNSELED REGARDING THE IMPORTANCE OF PROCEDURAL COMPLIANCE.

[208] VOGTLE 2 DOCKET 50-425 LER 91-002
DETECTOR FAILURE RESULTS IN CONTAINMENT VENTILATION ISOLATION.
EVENT DATE: 010991 REPORT DATE: 013191 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 220850) ON 1-9-91, AT 1202 CST, AN INTERMEDIATE LEVEL ALERT ALARM WAS RECEIVED FOR CONTAINMENT AREA RADIATION MONITOR 2RE-0003. INVESTIGATION AND COMPARISON WITH A REDUNDANT MONITOR, 2RE-0002, DETERMINED THAT THE ALARM WAS DUE TO A SPURIOUS SIGNAL SPIKE. AT 1255 CST, A HIGH RADIATION LEVEL ALARM AND A CONTAINMENT VENTILATION ISOLATION (CVI) WERE RECEIVED FROM 2RE-0003. ALL EQUIPMENT RESPONDED AS DESIGNED TO ISOLATE CONTAINMENT VENTILATION AND TO MAINTAIN NEGATIVE PRESSURE FOR THE PIPING PENETRATION AREA. 220850IGATION VERIFIED THAT A FAILURE OF 2RE-0003 HAD OCCURRED. TECHNICAL SPECIFICATIONS 3.3.3.1 AND 3.3.2 ACTION REQUIREMENTS WERE ENTERED BECAUSE LESS THAN THE MINIMUM REQUIRED NUMBER OF RADIATION MONITORING CHANNELS WHICH INITIATE A CVI WERE OPERABLE. AFTER LIFTING THE ACTUATION LEADS FROM 2RE-0003, THE CVI SIGNAL WAS RESET AT 1712 CST AND NORMAL SYSTEM ALIGNMENTS WERE REESTABLISHED. TROUBLESHOOTING DETERMINED THAT THE DETECTOR ASSEMBLY FOR 2RE-0003 HAD FAILED. THE ROOT CAUSE HAS NOT BEEN DETERMINED. A SPARE DETECTOR ASSEMBLY IS NOT CURRENTLY AVAILABLE. THEREFORE, A CONTAINMENT ENTRY WILL NOT BE MADE TO COMPLETE REPAIRS UNTIL AFTER A SPARE IS OBTAINED. ANY DEFECTIVE PARTS WILL BE RETURNED TO WESTINGHOUSE FOR ANALYSIS.

[209] WATERFORD 3 DOCKET 50-382 LER 90-019
BOTH TRAINS OF CONTROL ROOM AIR CONDITIONING INOPERABLE DUE TO BREACH IN THE CONTROL ROOM ENVELOPE.
EVENT DATE: 121290 REPORT DATE: 011191 NSSS: CE TYPE: PWR

(NSIC 220682) AT 1045 HOURS ON DECEMBER 12, 1990, WITH WATERFORD STEAM ELECTRIC STATION UNIT 3 AT 100% POWER, TECHNICAL SPECIFICATION (TS) LIMITING CONDITION FOR OPERATION (LCO) 3.0.3 WAS ENTERED WHEN BOTH TRAINS OF THE CONTROL ROOM HEATING VENTILATION AND AIR CONDITIONING SYSTEM WERE DECLARED INOPERABLE DUE TO A BREACH IN THE CONTROL ROOM ENVELOPE. THE BREACH IN THE CONTROL ROOM ENVELOPE EXISTED SINCE DECEMBER 5, 1990, WHEN A PENETRATION FIRE SEAL WAS REMOVED FROM VENTILATION FIRE DAMPER FD-45 IN ACCORDANCE WITH AN APPROVED DESIGN CHANGE. THE PLANT OPERATED IN TS LCO 3.0.3 FOR A PERIOD OF 8 DAYS, THEREFORE THIS EVENT IS REPORTABLE AS OPERATION PROHIBITED BY PLANT TS. A TEMPORARY SEAL WAS INSTALLED AND THE CONTROL ROOM AIR CONDITIONING SYSTEM WAS DECLARED OPERABLE AT 1141 HOURS ON DECEMBER 12, 1990. THE ROOT CAUSE OF THIS EVENT IS LACK OF SUFFICIENT DOCUMENTATION AND DETAILS OF THE CONTROL ROOM ENVELOPE BOUNDARY SEALS. CALCULATIONS HAVE SHOWN THAT WITH THE BREACHED FIRE BARRIER, THE HABITABILITY OF THE CONTROL ROOM WOULD NOT HAVE BEEN THREATENED, DURING A HIGH RADIATION OR TOXIC CHEMICAL SCENARIO; THEREFORE, THIS EVENT DID NOT THREATEN THE HEALTH AND SAFETY OF THE GENERAL PUBLIC OR PLANT PERSONNEL.

[210] WATERFORD 3 DOCKET 50-382 LER 90-020
CHLORINE GAS RELEASE FROM LOCAL CHEMICAL PLANT.
EVENT DATE: 122790 REPORT DATE: 012891 NSSS: CE TYPE: PWR

(NSIC 220846) ON DECEMBER 27, 1990, AN ALERT WAS DECLARED AT 2116 HOURS AT WATERFORD STEAM ELECTRIC STATION UNIT 3, DUE TO A RELEASE OF CHLORINE GAS FROM A NEARBY CHEMICAL PLANT. WATERFORD 3 WAS NOTIFIED OF A SITE AREA EMERGENCY AT THE CHEMICAL PLANT AND ENTERED PROCEDURES FOR TOXIC CHEMICAL RELEASE (OPERATING PROCEDURE 901-047) AND TOXIC CHEMICAL EMERGENCY (EMERGENCY PLAN 004-010). AT 2144, THE EVENT WAS TERMINATED. THIS EVENT IS BEING REPORTED AS AN ITEM OF

POTENTIAL INDUSTRY INTEREST ALTHOUGH THIS EVENT DID NOT PRESENT A HAZARD TO PLANT EQUIPMENT OR THE HEALTH AND SAFETY OF THE GENERAL PUBLIC.

[211] WOLF CREEK 1 DOCKET 50-482 LER 90-025
BOTH SAFETY INJECTION PUMPS INOPERABLE BECAUSE OF FROZEN MINIMUM RECIRCULATION LINE TO REFUELING WATER STORAGE TANK.
EVENT DATE: 122390 REPORT DATE: 012291 NSSS: WE TYPE: PWR
VENDOR: UNITED ELECTRIC CONTROLS COMPANY

(NSIC 220864) ON 12/23/90 AT 1230 CST, IT WAS DETERMINED THAT THE COMMON MINIMUM FLOW PATH RETURN LINE FOR SAFETY INJECTION (SI) PUMPS PEM01A AND PEM01B TO THE REFUELING WATER STORAGE TANK (RWST) WAS FROZEN. PREVIOUS ACTIONS TO INVESTIGATE PROBLEMS WITH THE FREEZE PROTECTION SYSTEM WERE UNSUCCESSFUL IN PREVENTING DEVELOPMENT OF THIS CONDITION. THE TWO SI PUMPS WERE DECLARED INOPERABLE WITH THIS RETURN LINE FROZEN AND ENTRY WAS MADE INTO TECH SPEC 3.0.3. THE PUMPS WERE RETURNED TO OPERABLE STATUS AT 1757 CST. A FAULTY AMBIENT TEMPERATURE SWITCH FOR THE RWST HEAT TRACE SYSTEM PREVENTED THE HEAT TRACE FROM ACTIVATING AND WAS SUBSEQUENTLY REPLACED. IN ADDITION, ADMINISTRATIVE CONTROLS DID NOT SUFFICIENTLY RECOGNIZE THE SAFETY SIGNIFICANCE OF FLOW THROUGH THIS LINE AND THE NEED TO ENSURE FLOW CAPABILITY. ADDITONAL GUIDANCE HAS BEEN PROVIDED. ALSO, INEFFECTIVE COMMUNICATIONS BETWEEN OPERATIONS PERSONNEL AND ELECTRICAL MAINTENANCE PERSONNEL PREVENTED THOROUGH EVALUATION OF THE RWST HEAT TRACE TROUBLE ALARM. THE NEED FOR EFFECTIVE COMMUNICATIONS WILL BE REITERATED TO PLANT PERSONNEL.

[212] WOLF CREEK 1 DOCKET 50-482 LER 91-001
LEAK IN RUPTURE DISK ALLOWS AN UNPLANNED RELEASE OF WASTE GAS DECAY TANK WITHOUT PRIOR SAMPLING.
EVENT DATE: 010191 REPORT DATE: 013191 NSSS: WE TYPE: PWR
VENDOR: CONTINENTAL DISC CORP.

(NSIC 220856) ON JANUARY 1, 1991, A PRESSURE DROP OF 6.4 PSIG IN WASTE GAS DECAY TANK (WGDT) #8 WAS IDENTIFIED AT 1100 CST. THIS RELEASE DID NOT EXCEED TECHNICAL SPECIFICATION LIMITS. HOWEVER, THIS SITUATION CONSITUTES A VIOLATION OF TECHNICAL SPECIFICATION 3.11.2.1, WHICH REQUIRES, IN PART, SAMPLING OF A WGDT PRIOR TO ITS RELEASE. THE INITIAL INVESTIGATION BY MECHANICAL MAINTENANCE PERSONNEL REVEALED THAT LEAKAGE WAS OCCURRING ON THE HYDROGEN RECOMBINER SHA01A RUPTURE DISC, WHICH WAS REPLACED IN MID-DECEMBER, 1990. ATTEMPTS MADE TO CORRECT THE PROBLEM INCLUDED AN INCREASE IN THE TORQUE VALUE AND REPLACEMENT OF THE DISC. THESE ATTEMPTS FAILED TO SECURE THE LEAK. THE VENDOR WAS CONTACTED AND AN INCREASE IN THE TORQUE VALUE OF THE RUPTURE DISC WAS RECOMMENDED. ACTIVITIES TO RESOLVE THIS ISSUE WILL RECOMMENCE FOLLOWING THE COMPLETION OF AN ENGINEERING DISPOSITION WHICH FOCUSES ON THE CORRECT TORQUE TO USE AND THE POTENTIAL INSTALLATION OF A GASKET AT THE METAL TO METAL FLANGE INTERFACE. CATALYTIC HYDROGEN RECOMBINER "A" WILL REMAIN OUT OF SERVICE UNTIL RESOLUTION OF THE RUPTURE DISK LEAKAGE.

[213] WPPSS 2 DOCKET 50-397 LER 90-031
REACTOR SCRAM DUE TO MAIN GENERATOR TRIP CAUSED BY SHORTED MAIN TRANSFORMER OUTPUT LINE INSULATOR DUE TO LESS THAN ADEQUATE CORRECTIVE ACTION PLAN/PLANT DESIGN.
EVENT DATE: 120790 REPORT DATE: 010791 NSSS: GE TYPE: BWR

(NSIC 220683) ON DECEMBER 7, 1990 AT 1010 HOURS, A REACTOR SCRAM OCCURRED DUE TO ACTUATION OF THE REACTOR PROTECTION SYSTEM (RPS) LOGIC. THE INITIATING SCRAM SIGNAL WAS "TURBINE GOVERNOR VALVE FAST CLOSURE" DUE TO A MAIN TURBINE/GENERATOR TRIP WITH REACTOR POWER GREATER THAN 30 PERCENT. THE LOGIC WAS ACTUATED WHEN THE MAIN GENERATOR 500 KV OUTPUT BREAKERS TRIPPED AS A RESULT OF HIGH CURRENTS CREATED WHEN A PROCELAIN INSULATOR IN THE TRANSFORMER YARD SHORTED TO GROUND. THE INSULATOR IS ON THE OUTPUT SIDE OF 25/500 KV MAIN TRANSFORMER TR-M2 ("B" PHASE). THE ELECTRICAL FAULT (FLASHOVER) WAS DUE TO CIRCULATING WATER (CW) SYSTEM COOLING TOWER WATER CHEMICAL DEPOSITS HAVING BUILT UP ON THE INSULATOR, WITH WET AND ICING CONDITIONS CONTRIBUTING TO PROVIDE A CONDUCTIVE PATH OVER THE SURFACE OF THE INSULATOR. AT 1019 HOURS, AN "UNUSUAL EVENT" WAS DECLARED AS

DIRECTED BY THE EMERGENCY CLASSIFICATION PROCEDURE DUE TO THE RESULTING FAULT-CAUSED EXPLOSION WITHIN THE PROTECTED AREA (TRANSFORMER YARD) AT 1030 HOURS, THE REACTOR SCRAM WAS RESET AND, AT 1100 HOURS, THE "UNUSUAL EVENT WAS TERMINATED. THE ROOT CAUSES OF THIS EVENT WERE: 1) A LESS THAN ADEQUATE CORRECTIVE ACTION PLAN PERTAINING TO THE SCHEDULE AND SCOPE FOR INSPECTING AND CLEANING THE 500 KV INSULATORS, AND 2) A LESS THAN ADEQUATE PLANT DESIGN.

[214] WPPSS 2 DOCKET 50-397 LER 91-001
RCIC-V-8 AUTOMATIC CLOSURE ESF ACTUATION DUE TO FAILED ELECTRONIC COMPONENT IN LEAK DETECTION SYSTEM.
EVENT DATE: 010891 REPORT DATE: 020791 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 220913) AT 0727 ON JANUARY 8, 1991, ALARMS INDICATING HIGH REACTOR CORE ISOLATION COOLING (RCIC) TURBINE AND EQUIPMENT AREA TEMPERATURES WERE RECEIVED IN THE CONTROL ROOM. CONCURRENTLY THE OUTBOARD ISOLATION VALVE FOR THE RCIC TURBINE STEAM SUPPLY RCIC-V-8, AUTOMATICALLY CLOSED. REDUNDANT CONTROL ROOM INSTRUMENTATION AND INSPECTION OF THE RCIC LOCALE SHOWED NO ABNORMAL TEMPERATURES IN THE RCIC EQUIPMENT AREAS. THE CAUSE OF THE ISOLATION WAS TRACED TO A FAILED COMPONENT IN THE LEAK DETECTION SYSTEM, WHICH PROVIDES INPUT INTO THE RCIC ISOLATION LOGIC. OPERATIONS CORRECTLY ENTERED THE ACTION STATEMENTS FOR TECHNICAL SPECIFICATION 3.7.3 (RCIC OPERABILITY) AND 3.3.2 (ISOLATION ACTUATION INSTRUMENTS) WITHIN ONE HOUR OF THE EVENT. OPERATIONS REPORTED THE EVENT TO THE NRC AT 0850, SINCE AUTOMATIC CLOSURE OF RCIC-V-8 IS AN ESF ACTUATION, REQUIRING A 4 HOUR NOTIFICATION (50.72 (B)(2)(II)). THE DEFECTIVE COMPONENT WAS REPLACED AND THE LEAK DETECTION AND RCIC SYSTEMS RETURNED TO SERVICE. THE APPARENT CAUSE OF THIS EVENT WAS AN ISOLATED FAILURE OF A ELECTRONIC COMPONENT ASSOCIATED WITH ONE PRINTED CIRCUIT INPUT CARD IN LEAK DETECTION MONITOR LD-MON-1A. THE ROOT CAUSE IS INDETERMINATE AT THIS TIME. VENDOR ANALYSIS OF THE FAILED COMPONENT HAS BEEN REQUESTED. THIS EVENT PRESENTED NO THREAT TO PLANT PERSONNEL OR TO THE PUBLIC.

[215] WPPSS 2 DOCKET 50-397 LER 91-002
REACTOR RECIRCULATION SYSTEM JET PUMP OPERABILITY TESTING NOT IN LITERAL COMPLIANCE WITH TECHNICAL SPECIFICATIONS DUE TO LESS THAN ADEQUATE PROCEDURE PREPARATION/REVIEW.
EVENT DATE: 011091 REPORT DATE: 020891 NSSS: GE TYPE: BWR

(NSIC 220914) ON 1/10/91 IT WAS DETERMINED THAT CURRENT REACTOR RECIRCULATION (RRC) SYSTEM JET PUMP OPERABILITY SURVEILLANCE TESTING DID NOT MEET LITERAL COMPLIANCE WITH THE TECH SPECS. THE OPERABILITY OF THE JET PUMPS WAS BEING DETERMINED BY MATCHING RRC LOOP FLOWS INSTEAD OF FLOW CONTROL VALVE POSITIONS AS REQUIRED BY TECH SPEC 3.4.1.2. THIS DISCREPANCY WAS IDENTIFIED BY AN NRC INSPECTOR DURING A ROUTINE INSPECTION OF PLANT OPERATIONS ACTIVITIES. THE CAUSE OF THIS EVENT WAS LESS THAN ADEQUATE PROCEDURE PREPARATION/REVIEW TO ENSURE THAT THE PROCEDURE FOR DETERMINING JET PUMP OPERABILITY WAS IN LITERAL COMPLIANCE WITH THE TECH SPECS. THE PROCEDURE PROVIDED DIRECTION TO ADJUST DRIVE FLOWS SUCH THAT BOTH LOOPS ARE APPROXIMATELY EQUAL IN FLOW. THE TECH SPECS REQUIRE BOTH RECIRCULATION LOOPS TO BE OPERATING AT THE SAME FLOW CONTROL VALVE POSITION. IMMEDIATE CORRECTIVE ACTION CONSISTED OF REVISING THE APPROPRIATE PROCEDURE TO BE CONSISTENT WITH THE TECH SPECS REQUIREMENT, AND RE-PERFORMING THE SURVEILLANCE. FURTHER CORRECTIVE ACTIONS INCLUDE 1) SUBMITTING A TECH SPEC CHANGE REQUEST TO REMOVE THE REQUIREMENT OF EQUALIZING FLOW CONTROL POSITIONS DURING JET PUMP OPERABILITY DETERMINATIONS AND 2) IMPLEMENTING A PROCEDURE VERIFICATION AND VALIDATION PROCESS.

[216] YANKEE ROWE DOCKET 50-029 LER 90-008
INOPERABLE VAPOR CONTAINER ATMOSPHERE RECIRCULATION FAN.
EVENT DATE: 110490 REPORT DATE: 120390 NSSS: WE TYPE: PWR

(NSIC 220882) ON 11/4/90 AT 1115 HOURS WITH THE PLANT IN MODE 5, A SURVEILLANCE FLOW CHECK OF THE VC ATMOSPHERE RECIRCULATION SYSTEM SHOWED THAT FAN FN 18-3 WAS NOT PROVIDING THE MINIMUM FLOW REQUIREMENT OF 6,000 CFM. INVESTIGATION FOUND THAT THE FAN WAS ROTATING IN REVERSE. FURTHER INVESTIGATION FOUND THE LEADS AT THE

CONTACT WERE REVERSED. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO PERSONNEL ERROR IN THAT A FACILITY EMPLOYEE, AN ELECTRICIAN, INCORRECTLY FASTENED THE MOTOR LEADS TO THE CONTACT ON COMPLETION OF CONTACTOR REPLACEMENT. THE ELECTRICIAN FAILED TO IMPLEMENT A TEMPORARY CHANGE FOR THE LIFTING OF LEADS ASSOCIATED WITH THE MOTOR CONTACTOR REPLACEMENT, AND ALSO FAILED TO ADEQUATELY PERFORM THE RETEST REQUIRED BY THE MAINTENANCE PROCEDURE FOR THE CONTACTOR REPLACEMENT. THE INCORRECTLY FASTENED LEADS WERE REFASTENED CORRECTLY ON 11/4/90. AN INDEPENDENT REVIEW OF ELECTRICAL MAINTENANCE PERFORMED FROM THE BEGINNING OF THE REFUELING OUTAGE UNTIL THE TIME OF THE DISCOVERY OF THE INCIDENT HAS BEEN PERFORMED. NO OTHER DISCREPANCIES WERE FOUND. DISCIPLINARY ACTIONS HAVE BEEN TAKEN ON THE ELECTRICIAN INVOLVED IN THIS MAINTENANCE ACTIVITY.

[217] YANKEE ROWE DOCKET 50-029 LER 90-009
FAILURE TO PERFORM SURVEILLANCES REQUIRED BY TECHNICAL SPECIFICATIONS.
EVENT DATE: 110590 REPORT DATE: 120490 NSSS: WE TYPE: PWR

(NSIC 220883) ON 11/5/90, WHILE IN MODE 5 FOLLOWING A PLANT REFUELING OUTAGE, THE DETERMINATION WAS MADE THAT THE REQUIRED TECH SPEC (TS) SURVEILLANCES OF THE MAIN STEAM (MS) LINE AND PRIMARY VENT STACK (PVS) PROCESS MONITORS HAD NOT BEEN SATISFACTORILY PERFORMED BETWEEN 11/89 AND 10/90. T.S.4.3.3.1 REQUIRES THAT EACH OF THESE RADIATION MONITORING INSTRUMENTATION CHANNELS BE DEMONSTRATED OPERABLE (DURING MODES 1-4) BY THE PERFORMANCE OF CHANNEL FUNCTIONAL TESTS ON A MONTHLY BASIS. PLANT PROCEDURE OP-4816 SPECIFIED THAT THESE RADIATION MONITORS BE TESTED QUARTERLY. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO PERSONNEL ERROR. ALTHOUGH AWARE OF THE ISSUANCE OF LICENSE AMENDMENT NO. 126, RADIATION PROTECTION DEPARTMENT PERSONNEL DID NOT REVISE OP-4816 TO CHANGE THE SURVEILLANCE FREQUENCY FROM QUARTERLY TO MONTHLY. IMMEDIATE CORRECTIVE ACTION INVOLVED TESTING THE MS LINE AND PVS PROCESS MONITORS ON 11/5/90. THE RESULTS OF THESE FUNCTIONAL TESTS WERE ALL SATISFACTORY. OTHER CORRECTIVE ACTION INCLUDED REVISING OP-4816 TO REFLECT THE CORRECT SURVEILLANCE FREQUENCIES, AND ALSO, REVIEWING ALL SURVEILLANCE PROCEDURES FOR TS RADIATION MONITORS TO ENSURE THAT FREQUENCIES ARE CORRECT. THIS IS THE FIRST OCCURRENCE OF THIS NATURE AT THIS FACILITY.

[218] YANKEE ROWE DOCKET 50-029 LER 90-011
MANUAL VALVE OPERATION RESULTS IN REACTOR SCRAM.
EVENT DATE: 120590 REPORT DATE: 010491 NSSS: WE TYPE: PWR

(NSIC 221095) ON 12/5/90, AT 0702 HOURS, WITH THE PLANT IN MODE 1 AT 100% POWER, FOLLOWING PERFORMANCE OF VALVING TO SUPPORT REPAIR OF A CONTROL AIR LEAK, AN AUTOMATIC REACTOR SCRAM OCCURRED. CLOSURE OF A CONTROL AIR ISOLATION VALVE (CA-V-1239) RESULTED IN AN UNANTICIPATED LOSS OF AIR TO THE HEATER DRAIN TANK HIGH LEVEL DUMP VALVE CAUSING IT TO OPEN. THIS RESULTED IN SEQUENTIAL AUTOMATIC TRIPPING OF THE HEATER DRAIN PUMP ON LOW HEATER DRAIN TANK LEVEL AND THE BOILER FEED PUMPS (BFPS) ON LOW SUCTION PRESSURE. THE LOSS OF FEEDWATER FLOW FROM THE BFPS RESULTED IN A LOW LEVEL CONDITION IN THE STEAM GENERATORS WHICH INITIATED THE AUTOMATIC REACTOR SCRAM. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO PERSONNEL ERROR. AN INCOMPLETE NOTATION ON A PLANT DRAWING AND THE VALVE IDENTIFICATION TAG CONTRIBUTED TO THE ERROR. AS CORRECTIVE ACTIONS THE DRAWING AND VALVE TAG HAVE BEEN CORRECTED AND A WALKDOWN OF THE CONTROL AIR SYSTEM DRAWINGS IS IN PROGRESS. ALL PLANT SYSTEMS FUNCTIONED AS DESIGNED DURING THE EVENT. THERE WAS NO ADVERSE EFFECT ON THE HEALTH OR SAFETY OF THE PUBLIC AS A RESULT OF THIS EVENT.

COMPONENT INDEX

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

COMPONENT INDEX

SYSTEM INDEX

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

SYSTEM INDEX

SYSTEM INDEX

KEYWORD INDEX

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

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

VENDOR CODE INDEX

NRC FORM 335 U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE

1. REPORT NUMBER: NUREG/CR-2000, Vol.10, No. 3 ORNL/NSIC-200

2. TITLE AND SUBTITLE: Licensee Event Report (LER) Compilation

For month of March 1991

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5. AUTHOR(S): Prepared by Oak Ridge National Laboratory

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11a. TYPE OF REPORT: Monthly Report

b. PERIOD COVERED (Inclusive dates): March 1991

12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

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

14. DOCUMENT ANALYSIS - a. KEYWORDS/DESCRIPTORS

Licensee Event Report
Sequence Coding and Search System
Reactor, PWR
Reactor, BWR
Systems
Components
Operating Experience
Event Compilation

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