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Licensee Event Report (LER) Compilation

For month of February 1988

Oak Ridge National Laboratory

Prepared for
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Oak Ridge National Laboratory
Nuclear Safety Information Center
Oak Ridge, TN 37831

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Abstract

This monthly report contains Licensee Event Report (LER) operational information that was processed into the LER data file of the Nuclear Safety Information Center (NSIC) during the one month period identified on the cover of the document. The LERs, from which this information is derived, are submitted to the Nuclear Regulatory Commission (NRC) by nuclear power plant licensees in accordance with federal regulations. Procedures for LER reporting for revisions to those events occurring prior to 1984 are described in NRC Regulatory Guide 1.16 and NUREG-1061, Instructions for Preparation of Data Entry Sheets for Licensee Event Reports. For those events occurring on and after January 1, 1984, LERs are being submitted in accordance with the revised rule contained in Title 10 Part 50.73 of the Code of Federal Regulations (10 CFR 50.73 - Licensee Event Report System) which was published in the Federal Register (Vol. 48, No. 144) on July 26, 1983. NUREG-1022, Licensee Event Report System - Description of Systems and Guidelines for Reporting, provides supporting guidance and information on the revised LER rule.

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

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[1] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 87-005
REACTOR TRIP DUE TO MAIN TURBINE GENERATOR TRIP CAUSED BY PERSONNEL ERROR DURING SWITCHYARD BREAKER TESTING.
EVENT DATE: 082587 REPORT DATE: 092487 NSSS: BW TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 206559) WITH THE UNIT OPERATING AT 70 PERCENT POWER, ONE OF TWO PARALLEL GENERATOR OUTPUT BREAKERS WAS REMOVED FROM SERVICE FOR MAINTENANCE. WHEN THE BREAKER WAS TRIPPED DURING POST MAINTENANCE TESTING, ONE OF THREE PHASES FAILED TO OPEN. A PHASE DISCORDANT SIGNAL WAS PRODUCED DUE TO THE PERCEIVED FAULT RESULTING IN A BREAKER FAILURE LOCKOUT. A BREAKER FAILURE MODULE IN THE BREAKER CIRCUITRY SHOULD HAVE BEEN PULLED PRIOR TO TESTING AS THE BREAKER WAS ISOLATED FROM THE SWITCHYARD. THE MODULE WAS NOT PULLED DUE TO PERSONNEL ERROR. THIS CAUSED OTHER SWITCHYARD BREAKERS TO OPEN, ONE OF WHICH WAS THE OTHER GENERATOR OUTPUT BREAKER RESULTING IN A LOAD REJECTION. THE BREAKER FAILURE LOCKOUT ALSO CAUSED A TURBINE TRIP AND REACTOR TRIP ON LOSS OF TURBINE ANTICIPATORY TRIP. ONLY ONE OF THE STATION 6900 VOLT BUSSES ACCOMPLISHED A FAST TRANSFER TO OFFSITE POWER. THE OTHER BUSSES PERFORMED A SLOW TRANSFER. THE SLOW TRANSFER OF THE BUSSES RESULTED IN LOSS OF BOTH MAIN FEEDWATER PUMPS AND SUBSEQUENT ACTUATION OF EMERGENCY FEEDWATER ON LOW STEAM GENERATOR WATER LEVEL. MOMENTARY ENGINEERED SAFEGUARD (ES) BUSSES UNDERVOLTAGE DUE TO THE SLOW TRANSFER CAUSED THE EMERGENCY DIESEL GENERATORS TO START BUT NOT TO TIE ON TO THE BUSSES. THE PLANT WAS STABILIZED AT HOT SHUTDOWN.

[2] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 87-006
FAILURE TO PERFORM PENETRATION ROOM VENTILATION SYSTEM SURVEILLANCE PRIOR TO ESTABLISHING REACTOR BUILDING INTEGRITY DUE TO INADEQUATE PROCEDURAL GUIDANCE AND PERSONNEL ERROR.
EVENT DATE: 110387 REPORT DATE: 120387 NSSS: BW TYPE: PWR

(NSIC 207337) A HEATUP OF THE REACTOR COOLANT SYSTEM (RCS) WAS COMMENCED AT 1839 HOURS ON 11/2/87 FOLLOWING AN OUTAGE. AT 1935 HOURS ON 11/2/87, THE RCS REACHED 200 DEGREES FAHRENHEIT, A CONDITION WHICH REQUIRES REACTOR BUILDING INTEGRITY BE ESTABLISHED AND THE PENETRATION ROOM VENTILATION SYSTEM (PRVS) BE OPERABLE. ON 11/3/87 AT 1000 HOURS WITH THE RCS AT 424 DEGREES FAHRENHEIT, IT WAS DISCOVERED BY OPERATIONS PERSONNEL REVIEWING THE HEATUP PACKAGE DOCUMENTATION THAT THE SURVEILLANCE INTERVAL FOR THE STANDBY PRVS HAD BEEN EXCEEDED MAKING THE STANDBY PRVS TECHNICALLY INOPERABLE. THE LEAD PRVS WAS VERIFIED OPERABLE. TESTING OF THE STANDBY PRVS WAS SATISFACTORILY COMPLETED AT 1139 HOURS ON 11/3/87. OPERATIONS AND WORK CONTROL CENTER PERSONNEL WHO HAD AUTHORIZED THE HEATUP HAD FAILED TO ADEQUATELY REVIEW THE DOCUMENTATION USED IN VERIFYING THAT THE SURVEILLANCE HAD NOT BEEN COMPLETED. ALSO, THE PROCEDURE GOVERNING PREPARATIONS FOR HEATUP AND CRITICALITY CONTAINED INADEQUATE GUIDANCE FOR USE BY OPERATIONS PERSONNEL OF A FORM WHICH DENOTES THE STATUS OF SURVEILLANCES REQUIRED TO BE COMPLETED PRIOR TO HEATUP. THE EVENT WILL BE REVIEWED WITH THE PERSONNEL RESPONSIBLE FOR AUTHORIZING HEATUP AND CRITICALITY AND THEIR RESPONSIBILITIES WILL BE CLARIFIED AND RE-EMPHASIZED.

[3] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 87-007
SINGLE FAILURE DESIGN CRITERIA NOT MET FOR THE STANDBY PENETRATION ROOM VENTILATION SYSTEM.
EVENT DATE: 110687 REPORT DATE: 120787 NSSS: BW TYPE: PWR
VENDOR: BUFFALO FORGE COMPANY

(NSIC 207349) ON NOVEMBER 6, 1987, DURING PERFORMANCE OF VERIFICATION REVIEWS OF THE SYSTEM DESIGN DESCRIPTIONS IN THE SAFETY ANALYSIS REPORT (SAR) AS PART OF AN SAR UPGRADE PROJECT, A DISCREPANCY IN THE ORIGINAL DESIGN OF THE PENETRATION ROOM VENTILATION SYSTEM (PRVS) WAS IDENTIFIED. THE PRVS HAS TWO REDUNDANT FANS THAT RECEIVE AN ENGINEERED SAFEGUARDS (ES) ACTUATION. ONE FAN WAS DESIGNATED THE LEAD

FAN WHICH WAS DESIGNED TO START UPON RECEIPT OF THE ES SIGNAL. THE STANDBY FAN WAS DESIGNED NOT TO START UNLESS THE LEAD FAN FAILED AFTER STARTING AND A TIME DELAY RELAY ACTUATED THE STANDBY FAN ON LOW LEAD FAN FLOW. THE SAR AND TECH SPECS INDICATE THAT THE PRVS IS CAPABLE OF AUTOMATIC INITIATION AND DESIGNED TO MEET SINGLE FAILURE CRITERIA. A LOSS OF POWER TO THE LEAD SYSTEM WAS APPARENTLY NOT CONSIDERED IN THE SINGLE FAILURE DESIGN. THE STANDBY SYSTEM WOULD NOT HAVE AUTOMATICALLY STARTED IF POWER HAD BEEN LOST TO THE LEAD SYSTEM. THIS CONDITION WAS CORRECTED BY A DESIGN MODIFICATION SUCH THAT UPON RECEIPT OF AN ES SIGNAL, THE STANDBY SYSTEM WILL AUTOMATICALLY START WITHIN 5 SECONDS IF POWER IS LOST TO THE LEAD SYSTEM. MANUAL ACTUATION OF THE STANDBY SYSTEM WAS POSSIBLE EVEN IF THE LEAD SYSTEM LOST POWER. NO OTHER DESIGN DISCREPANCIES IN THE APPLICATION OF SINGLE FAILURE CRITERIA OR AUTOMATIC INITIATION WERE IDENTIFIED FOR THIS SYSTEM.

[4] ARKANSAS NUCLEAR 2 DOCKET 50-368 LER 87-007
REACTOR TRIP ON HIGH REACTOR COOLANT SYSTEM PRESSURE CAUSED BY LOSS OF POWER TO MAIN TURBINE CONTROLS DUE TO FAILURE OF CABLE TERMINAL LUG ON INSTRUMENTATION TRANSFORMER.
EVENT DATE: 090987 REPORT DATE: 100787 NSSS: CE TYPE: PWR
VENDOR: COPES-VULCAN, INC.

(NSIC 206654) ON 9/9/87 WITH THE UNIT OPERATING AT FULL POWER, A REACTOR TRIP ON HIGH REACTOR COOLANT SYSTEM (RCS) PRESSURE OCCURRED. FAILURE OF AN ALUMINUM SETSCREW TYPE TERMINAL LUG, CONNECTING ONE PHASE OF THE SECONDARY SIDE OF A STEP DOWN TRANSFORMER TO AN INSTRUMENTATION DISTRIBUTION PANEL, CREATED A PARTIAL LOSS OF POWER TO THE MAIN TURBINE ELECTROHYDRAULIC CONTROL SYSTEM RESULTING IN RAPID CLOSURE OF THE TURBINE CONTROL VALVES. THE RESULTING PRIMARY TO SECONDARY POWER MISMATCH CAUSED RCS PRESSURE TO INCREASE RAPIDLY TO THE REACTOR PROTECTION SYSTEM TRIP SETPOINT. EMERGENCY FEEDWATER ACTUATED AS DESIGNED TO RESTORE AND MAINTAIN STEAM GENERATOR WATER LEVELS. TWO TURBINE BYPASS VALVES FAILED TO RESPOND PROPERLY DURING THE TRANSIENT DUE TO FAILURE OF POSITIONERS ASSOCIATED WITH THEIR ACTUATORS. THE CAUSE OF THE TERMINAL LUG FAILURE WAS ELECTRICAL ARCING BETWEEN THE LUG AND CABLE CONDUCTORS AS A RESULT OF POOR ELECTRICAL CONTACT DUE TO LOOSENING OF THE LUG AROUND THE CONDUCTORS. THE FAILED LUG WAS REPLACED AND THE SYSTEM RETURNED TO SERVICE AFTER TESTING TO VERIFY A GOOD CONNECTION IN THE REPAIRED AREA AND AT OTHER LOCATIONS WITH SIMILAR CONNECTIONS. THE TURBINE BYPASS VALVES WERE REPAIRED, TESTED AND RETURNED TO SERVICE.

[5] ARKANSAS NUCLEAR 2 DOCKET 50-368 LER 87-008
REACTOR TRIP ON HIGH REACTOR COOLANT SYSTEM PRESSURE DUE TO MAIN TURBINE TRIP CAUSED BY INACCURATE BEARING VIBRATION INDICATION AND INCORRECT TURBINE TRIP SETPOINT.
EVENT DATE: 111487 REPORT DATE: 121187 NSSS: CE TYPE: PWR

(NSIC 207508) ON 11/14/87 WITH THE UNIT OPERATING AT FULL POWER, INDICATED HIGH BEARING VIBRATIONS ON NUMBER NINE MAIN TURBINE GENERATOR JOURNAL BEARING INITIATED AN AUTOMATIC TRIP OF THE MAIN TURBINE. THE RESULTING PRIMARY TO SECONDARY POWER MISMATCH CAUSED AN AUTOMATIC REACTOR TRIP ON HIGH REACTOR COOLANT SYSTEM PRESSURE WITHIN A FEW SECONDS OF THE MAIN TURBINE TRIP. EMERGENCY FEEDWATER AUTOMATICALLY ACTIVATED AS DESIGNED TO RESTORE AND MAINTAIN STEAM GENERATOR WATER LEVELS. THE MAIN TURBINE TRIP WAS DUE TO THE TRIP SETPOINT FOR NUMBER NINE BEARING HIGH VIBRATION BEING SET AT A VALUE LOWER THAN SPECIFIED IN PLANT PROCEDURES IN COMBINATION WITH INACCURATE INDICATIONS OF NUMBER NINE BEARING VIBRATION PROVIDED TO THE TRIP SYSTEM (HIGH BEARING VIBRATION WAS NOT SHOWN USING LOCAL VIBRATION MONITORING EQUIPMENT). DOCUMENTATION COULD NOT BE FOUND TO SHOW THAT THE VIBRATION MONITORING INSTRUMENTS AND PORTIONS OF THE MAIN TURBINE TRIP SYSTEM UTILIZING THESE INSTRUMENTS HAD BEEN ROUTINELY CALIBRATED. THE NUMBER NINE BEARING HIGH VIBRATION TRIP SETPOINT WAS INCREASED TO THE CORRECT VALUE AND A ROUTINE CALIBRATION PROGRAM FOR THE VIBRATION MONITORING SYSTEM IS BEING DEVELOPED. ON 11/15/87, THE MAIN TURBINE GENERATOR WAS STARTED AND LOADED.

VIBRATION INDICATION AT THE NUMBER NINE BEARING HAD RETURNED TO AN ACCEPTABLE LEVEL.

[6] ARNOLD DOCKET 50-331 LER 86-025 REV 01
UPDATE ON REACTOR SCRAM FROM A SPURIOUS HIGH INTERMEDIATE RADIATION MONITOR TRIP.
EVENT DATE: 121086 REPORT DATE: 121087 NSSS: GE TYPE: BWR

(NSIC 207399) AT 0338 HOURS ON DECEMBER 10, 1986, A REACTOR SHUTDOWN WAS IN PROGRESS. WHILE SUBCRITICAL AT APPROXIMATELY 1-2% THERMAL POWER, AN AUTOMATIC SCRAM WAS RECEIVED WHEN THE B, C AND D INTERMEDIATE RANGE MONITORS (IRMS) TRIPPED UPSCALE. THE CAUSE OF THE UPSCALE TRIPS IS BELIEVED TO BE NOISE GENERATED FROM SOURCE RANGE MONITOR (SRM) INSERTION. THE MOST PROBABLE ROOT CAUSE OF THE SCRAM IS THIS SPURIOUS SIGNAL. A POSITIVE REACTIVITY ADDITION DUE TO A FEEDWATER TRANSIENT HAS BEEN INVESTIGATED, BUT IS NOT CONSIDERED AS THE LIKELY ROOT CAUSE. TWO SRMS WERE INSERTED SIMULTANEOUSLY WHICH CONFLICTS WITH SOME PROCEDURAL GUIDELINES. THEREFORE, THE ROOT CAUSE OF THE SCRAM COULD BE ATTRIBUTED TO PERSONNEL ERROR IN NOT EXPLICITLY FOLLOWING PROCEDURES, BUT A PROCEDURAL INCONSISTENCY MUST BE CONSIDERED AS STRONGLY CONTRIBUTING TO THE ERROR. AS A SHORT TERM CORRECTIVE ACTION, THE OPERATOR WAS COUNSELED ON THE NECESSITY OF EXPLICITLY FOLLOWING PROCEDURES OR IDENTIFYING AND CLARIFYING PROCEDURAL REQUIREMENTS WHEN INCONSISTENCIES EXIST. THE EFFECTED PROCEDURES HAVE ALSO BEEN CORRECTED. THE REACTOR EXPERIENCED NO SIGNIFICANT TRANSIENTS, NOR WERE ANY EMERGENCY SYSTEMS INITIATED OR REQUIRED. THEREFORE, THERE WAS NO SIGNIFICANT EFFECT ON THE SAFE OPERATION OF THE PLANT FROM THE SCRAM.

[7] ARNOLD DOCKET 50-331 LER 87-028
RWCU ISOLATION DUE TO A TEMPERATURE SWITCH REACHING ITS END-OF-LIFE USE.
EVENT DATE: 101987 REPORT DATE: 111887 NSSS: GE TYPE: BWR
VENDOR: FENWALL, INC.

(NSIC 207199) ON OCTOBER 19, 1987 WITH THE PLANT IN COLD SHUTDOWN, A REACTOR WATER CLEANUP (RWCU) SYSTEM ISOLATION OCCURRED ON A SIGNAL FROM THE RWCU NONREGENERATIVE HEAT EXCHANGER OUTLET TEMPERATURE INDICATING SWITCH. THE TEMPERATURE SWITCH SETPOINT WAS FOUND TO HAVE DRIFTED CONSERVATIVELY LOW. THE ROOT CAUSE WAS DETERMINED TO BE THAT THE SWITCH HAD REACHED ITS END-OF- LIFE USE. THIS SAME TEMPERATURE SWITCH WAS RESPONSIBLE FOR TWO RWCU ISOLATIONS IN JULY, 1987. A NEW SWITCH WAS ORDERED IN EARLY 1987 DUE TO PREVIOUS SETPOINT DRIFTING PROBLEMS. THIS SWITCH ARRIVED ON-SITE IN AUGUST, 1987. OPERATIONS PERSONNEL RETURNED THE RWCU SYSTEM TO SERVICE LATER IN THE DAY AFTER THE SWITCH WAS ELECTRICALLY REMOVED FROM THE ISOLATION CIRCUIT. THE NEW SWITCH WAS INSTALLED AND TESTED ON OCTOBER 21, 1987.

[8] ARNOLD DOCKET 50-331 LER 87-029
HALF GROUP III ISOLATION AND STANDBY GAS INITIATION DURING PREVENTIVE MAINTENANCE.
EVENT DATE: 111187 REPORT DATE: 120487 NSSS: GE TYPE: BWR

(NSIC 207366) ON NOVEMBER 11, 1987, WITH THE PLANT AT POWER OPERATION A PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) HALF GROUP 111 ('B' SIDE) ISOLATION SIGNAL WAS RECEIVED ALONG WITH THE INITIATION OF THE 'B' STANDBY GAS TREATMENT SYSTEM. THE ISOLATION OCCURRED WHILE PERFORMING PREVENTIVE MAINTENANCE. DURING CONNECTION OF A VOLTMETER TO A POWER SUPPLY WHICH FEEDS TWO RADIATION MONITORS A NOISE SPIKE WAS PRODUCED CAUSING THE MONITORS' TRIP POINTS TO BE EXCEEDED. WHILE VERIFYING THE STATUS OF AUTOMATIC FUNCTIONS IT WAS FOUND THAT ONE OF THE REACTOR BUILDING HEATING AND VENTILATION (H&V) ISOLATION DAMPERS HAD NOT FULLY CLOSED. AT THIS POINT THE OPERATORS CLOSED THE REDUNDANT DAMPER WHICH IS OPERATED BY THE 'A' SIDE (UNTRIPPED) LOGIC. THE ISOLATION WAS RESET FOLLOWING AUTOMATIC FUNCTION VERIFICATION AND THE STANDBY GAS TREATMENT SYSTEM WAS RETURNED TO STANDBY. THIS EVENT HAD NO AFFECT ON THE SAFE OPERATION OF THE PLANT. THE CAUSE OF THIS EVENT

WAS AN ELECTRICAL NOISE SPIKE. THE ROOT CAUSE FOR THIS SPIKE, WHICH WAS PRODUCED WHEN THE METER WAS CONNECTED, COULD NOT BE DETERMINED. AS A CORRECTIVE ACTION, SIMILAR PREVENTIVE MAINTENANCE ACTIVITIES WILL BE SCHEDULED DURING REFUEL OUTAGES. IN ADDITION, A NOTE WILL BE ADDED TO THE APPLICABLE PROCEDURES STATING THE POSSIBILITY OF AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION.

[9] BEAVER VALLEY 1 DOCKET 50-334 LER 87-018 REV 01
UPDATE ON VITAL BUS SPIKE RESULTS IN ESF ACTUATION.
EVENT DATE: 102287 REPORT DATE: 112387 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 207264) ON 10/22/87 AT 0457 HOURS, DURING NORMAL FULL POWER OPERATION, A MAIN FILTER BANK AND CONTROL ROOM PRESSURIZATION ACTUATION OCCURRED. WHILE CHANGING THE GENERATOR WATT RECORDER CHART PAPER, THE REACTOR OPERATOR MOMENTARILY SHORTED THE RECORDERS POWER SUPPLY CAUSING A 120 VAC VITAL BUS 2 SPIKE. THIS CAUSED A SPIKE ON THE "A" TRAIN AUXILIARY BUILDING VENTILATION RADIATION MONITOR INITIATING FLOW THROUGH THE MAIN FILTER BANK AND A SPIKE ON THE "B" CONTROL ROOM RADIATION MONITOR INITIATING CONTROL ROOM PRESSURIZATION. THE OPERATORS RESET THE RADIATION MONITORS AND INITIATED ACTIONS TO RETURN THE SYSTEMS TO NORMAL OPERATION. THIS EVENT WAS CAUSED BY THE DESIGN OF THE RECORDERS' PAPER DRIVE PLUG WHICH HAS INSUFFICIENT PROTECTION AGAINST THE PLUG CONTACT TOUCHING GROUND. THERE WERE NO SAFETY IMPLICATIONS RESULTING FROM THIS EVENT SINCE THE SYSTEMS FUNCTIONED AS DESIGNED ON RECEIPT OF A HIGH-HIGH RADIATION SIGNAL.

[10] BEAVER VALLEY 2 DOCKET 50-412 LER 87-013 REV 01
UPDATE ON INADVERTENT REALIGNMENT OF MAIN FILTER BANK DAMPERS.
EVENT DATE: 080687 REPORT DATE: 110687 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: BEAVER VALLEY 1 (PWR)

(NSIC 207136) AT 0510 HOURS ON 8/6/87, WITH THE UNIT IN THE STARTUP MODE AT 1.0 E-6 AMPS ON THE INTERMEDIATE RANGE NEUTRON FLUX INSTRUMENTS, AN INADVERTENT REALIGNMENT OF THE SUPPLEMENTARY LEAK COLLECTION AND RELEASE SYSTEM (SLCRS) DAMPERS TO THE MAIN FILTER BANK OCCURRED. THE ACTUATION WAS INITIATED WHEN A LICENSED OPERATOR PERFORMING A CIRCUIT BREAKER CLEARANCE INADVERTENTLY RACKED THE BREAKER ONTO AUXILIARY 4KV BUS G, WHICH RESULTED IN A SUPPLY BREAKER OVERCURRENT TRIP AND LOCKOUT. THE DEENERGIZATION OF THE 4KV BUS LED TO AN AUTOMATIC BUS TRANSFER OF THE 480 V BUSSES SUPPLIED BY BUS G. THE BRIEF LOSS OF VOLTAGE DURING THE TRANSFER ACTUATED THE INSTANTANEOUS SOLID STATE UNDERVOLTAGE PROTECTION OF SUPPLEMENTARY LEAK COLLECTION RADIATION MONITOR RMR-RQI-301, WHICH SIMULATED A HIGH RADIATION CONDITION AND REALIGNED THE DAMPERS. THEREFORE, THIS REPORT IS BEING SUBMITTED UNDER 10 CFR 50.73.A.2.IV. NO SAFETY IMPLICATIONS RESULTED BECAUSE ALL EQUIPMENT FUNCTIONED PROPERLY AND NO ACTUAL RADIATION WAS PRESENT. THE RADIATION MONITOR WAS REENERGIZED AND NORMAL POWER ALIGNMENT RESTORED BY 0550 HOURS, 8/6/87. THE OPERATOR INVOLVED WAS REPRIMANDED AND ALL OPERATIONS PERSONNEL WILL REVIEW THIS EVENT AS A REMINDER OF PROPER BREAKER RACKING PROCEDURES.

[11] BEAVER VALLEY 2 DOCKET 50-412 LER 87-029
TURBINE TRIP/REACTOR TRIP DUE TO ERRATIC MAIN FEEDWATER REGULATING VALVE OPERATION.
EVENT DATE: 101587 REPORT DATE: 111687 NSSS: WE TYPE: PWR

(NSIC 207137) ON 10/15/87 AT APPROXIMATELY 1514 HOURS, WITH THE UNIT AT 65% REACTOR POWER, ERRATIC OPERATION OF ALL THREE MAIN FEEDWATER REGULATING VALVES (MFRV) WAS EXPERIENCED. THE OPERATORS TOOK MANUAL CONTROL OF THE "A" AND "B" MFRVS TO ATTEMPT TO RESTORE CONTROL. LEVEL STABILITY COULD NOT BE ACHIEVED BECAUSE ADJUSTMENT OF ONE MFRV CAUSED ADVERSE RESPONSES ON THE OTHER FEEDWATER LINES. AT 1515 HOURS, THE LEVEL IN THE 21A STEAM GENERATOR INCREASED TO 75%,

CAUSING A TURBINE TRIP. SINCE REACTOR POWER WAS ABOVE THE P-9 PROTECTION INTERLOCK (49%), THE TURBINE TRIP INITIATED A REACTOR TRIP. THE OPERATORS STABLIIZED THE PLANT IN HOT SHUTDOWN UTILIZING THE EMERGENCY OPERATING PROCEDURES. THE CAUSE FOR THIS EVENT WAS THE OVER-RESPONSIVENESS OF THE MFRVS TO SMALL LEVEL CHANGES IN THE STEAM GENERATORS. INSTRUMENT AND CONTROL PERSONNEL WERE CALLED IN TO ADJUST THE ELECTRONIC GAINS WITHIN THE MFRV CONTROLLERS. AS A PERMANENT CORRECTIVE ACTION, THE RESPONSE TIMES OF THE MFRVS WILL BE ADJUSTED TO ALLOW STABLE LEVEL CONTROL OF THE STEAM GENERATORS. THERE WERE NO SAFETY IMPLICATIONS TO THE PUBLIC AS A RESULT OF THIS EVENT. THIS EVENT HAS BEEN PREVIOUSLY ANALYZED IN FSAR SECTION 15.1.2, "FEEDWATER SYSTEM MALFUNCTIONS CAUSING AN INCREASE IN FEEDWATER FLOW". THIS EVENT IS BEING REPORTED IN ACCORDANCE WITH 10 CFR 50.73.A.2.IV AS AN EVENT INVOLVING AN AUTOMATIC ACTUATION OF THE REACTOR PROTECTION SYSTEM.

[12] BEAVER VALLEY 2 DOCKET 50-412 LER 87-030
MANUAL REACTOR TRIP DUE TO FIRE IN #2 TURBINE BEARING.
EVENT DATE: 101687 REPORT DATE: 111687 NSSS: WE TYPE: PWR

(NSIC 207111) ON 10/16/87 AT 2215 HOURS, THE CONTROL ROOM RECEIVED A PHONE CALL FROM A MECHANIC INDICATING THAT SMOKE WAS ISSUING FROM THE HIGH PRESSURE TURBINE ENCLOSURE. THE EMERGENCY SQUAD WAS DISPATCHED TO INVESTIGATE. AT 2217 HOURS, THE EMERGENCY SQUAD REPORTED A FIRE IN THE VICINITY OF THE #2 TURBINE BEARING. THE NUCLEAR SHIFT SUPERVISOR IMMEDIATELY ORDERED THE REACTOR MANUALLY TRIPPED. AT 2223 HOURS, THE EMERGENCY SQUAD REPORT THE FIRE OUT. THE FIRE WAS CAUSED BY OIL SOAKED INSULATION ON THE #2 GLAND PIPING. LONG TERM CORRECTIVE ACTION IS PENDING RESULTS OF VENDOR INVESTIGATION. IT HAS BEEN DETERMINED THE #2 BEARING LEAKS OIL WHENEVER THE LID ON THE LUBE OIL RESERVOIR IS REMOVED AND NEGATIVE PRESSURE IS LOST IN THE PEDESTAL. THE TIME THE LID IS REMOVED IS BEING KEPT TO A MINIMUM TO MAINTAIN A NEGATIVE PRESSURE AT THE PEDESTAL. THERE WERE NO INJURIES OR SAFETY IMPLICATIONS RESULTING FROM THIS INCIDENT.

[13] BEAVER VALLEY 2 DOCKET 50-412 LER 87-031
UNUSUAL EVENT-REACTOR COOLANT SYSTEM LEAKAGE IN EXCESS OF TECHNICAL SPECIFICATIONS.
EVENT DATE: 102087 REPORT DATE: 111987 NSSS: WE TYPE: PWR
VENDOR: KEROTEST MANUFACTURING CORP.

(NSIC 207238) DURING THE MORNING OF 10/20/87, WITH THE UNIT AT 97% REACTOR POWER, A REACTOR COOLANT SYSTEM (RCS) INVENTORY TEST WAS PERFORMED. AT 11108 HOURS, PERSONNEL ENTERED CONTAINMENT TO SEARCH FOR SUSPECTED RCS LEAKS SINCE A PREVIOUS TEST HAD INDICATED INCREASED LEAKAGE. THE RESULTS OF THE TEST YIELDED AN UNIDENTIFIED LEAKAGE RATE OF 1.33 GALLONS PER MINUTE (GPM), WHICH WAS IN EXCESS OF THE 1 GPM LIMITING CONDITION OF OPERATION (LCO) FOR TECH SPEC 3.4.6.2. AN UNUSUAL EVENT WAS DECLARED AT 1120 HOURS IN ACCORDANCE WITH THE BEAVER VALLEY EMERGENCY PREPAREDNESS PLAN, AND THE NUCLEAR REGULATORY COMMISSION WAS NOTIFIED UNDER 10 CFR 50.72.A.1.I. WHEN IT WAS DETERMINED THAT THE LEAK COULD NOT BE ISOLATED WITHIN THE FOUR HOUR LIMIT PROVIDED BY THE TECH SPEC 3.4.6.2 ACTION STATEMENT, A CONTROLLED SHUTDOWN TO HOT STANDBY WAS COMMENCED AT 1550 HOURS AND WAS COMPLETED BY 1712 HOURS. INVESTIGATION BY SUBSEQUENT CONTAINMENT ENTRY CREWS INDICATED THE CAUSE OF THE INVENTORY LOSS TO BE A BODY TO BONNET LEAK THROUGH A FURMANITE CLAMP ON THE RCS LOOP A COLD LEG BYPASS ISOLATION VALVE (2RCS*44). ADDITIONAL LEAKS WERE DISCOVERED ON TWO OTHER VALVES (2RCS*37 AND 46). AT 0510 HOURS ON 1021/87, 2RCS*44 WAS SUCCESSFULLY FURMANITED WHICH STOPPED THE LEAK. THE UNUSUAL EVENT WAS TERMINATED AT 0730 HOURS, 10/21/87.

[14] BEAVER VALLEY 2 DOCKET 50-412 LER 87-032
REACTOR TRIP DUE TO 100% LOAD REJECTION TEST.
EVENT DATE: 102487 REPORT DATE: 112387 NSSS: WE TYPE: PWR

(NSIC 207276) ON 10/24/87, THE 100% LOAD REJECTION TEST (IST 2.04.06) WAS PERFORMED. THE OPERATORS, AS PER PROCEDURE, MANUALLY OPENED THE MAIN OUTPUT BREAKERS TO INITIATE A LOSS OF LOAD TRANSIENT. CONDENSER STEAM DUMPS AUTOMATICALLY OPENED IN RESPONSE TO THE LOSS OF LOAD. THE RESULTANT STEAM FLOW TRANSIENT CAUSED ALL THREE STEAM GENERATOR LEVELS TO DROP RAPIDLY. A REACTOR TRIP OCCURRED ON LO-LO STEAM GENERATOR LEVEL. ALL AUXILIARY FEED PUMPS AUTO-STARTED TO RECOVER STEAM GENERATOR LEVELS. DUE TO TURBINE SPEED FLUCTUATIONS, ALL THREE REACTOR COOLANT PUMPS (RCPS) TRIPPED ON UNDERFREQUENCY. THIRTY SECONDS AFTER THE REACTOR TRIP, STATION LOADS WERE (AS DESIGNED) AUTOMATICALLY TRANSFERRED FROM ONSITE TO OFFSITE POWER. HOWEVER, THE "A", "B" AND "AE" 4KV BUSSES DID NOT SUCCESSFULLY TRANSFER DUE TO PHASE DIFFERENTIAL BETWEEN ONSITE AND OFFSITE POWER. THE #1 EMERGENCY DIESEL GENERATOR AUTO-STARTED AND REENERGIZED THE "AE" BUS. THE "A" AND "B" BUSSES WERE MANUALLY REALIGNED TO OFFSITE POWER. THE "A" RCP WAS RESTARTED. OPERATORS STABILIZED THE PLANT USING THE REACTOR TRIP RESPONSE PROCEDURE. THERE WERE NO SAFETY IMPLICATIONS DUE TO THIS EVENT. THIS EVENT WAS BOUNDED BY FSAR SECTION 15.2.6 (REACTOR TRIP FROM 100%, COINCIDENT WITH LOSS OF OFF-SITE POWER).

[15] BEAVER VALLEY 2 DOCKET 50-412 LER 87-033
FAILURE TO PERFORM SURVEILLANCE TEST WITHIN REQUIRED FREQUENCY.
EVENT DATE: 102887 REPORT DATE: 112587 NSSS: WE TYPE: PWR

(NSIC 207277) ON 10/28/87, THE OPERATING SURVEILLANCE TEST (OST) SCHEDULER WAS REVIEWING THE ELEVATED RELEASE WIDE RANGE GAS MONITOR CHANNEL FUNCTIONAL TEST (OST 2.43.8) PERFORMED ON 10/21/87, WHEN HE NOTICED THAT THE PERFORMANCE FREQUENCY HAD BEEN CHANGED FROM QUARTERLY TO MONTHLY. FURTHER INVESTIGATION INDICATED THAT THE WIDE RANGE GAS MONITOR CHANNEL C&D HAD NOT BEEN PERFORMED WITHIN ITS REQUIRED MONTHLY FREQUENCY. THE ROOT CAUSE OF THIS INCIDENT WAS PERSONNEL ERROR ON THE PART OF THE PROCEDURE WRITER WHO DID NOT INFORM THE TEST SCHEDULER THAT THE WIDE RANGE GAS MONITOR CHANNEL C&D FUNCTIONAL TEST HAD BEEN MOVED FROM OST 2.43.5 TO OST 2.43.8. THIS RESULTED IN THE MISSED SURVEILLANCE SINCE OST 2.43.5 WAS ON A MONTHLY FREQUENCY AND OST 2.43.8 WAS ON A QUARTERLY FREQUENCY. THE OST SCHEDULE DATA BASE WAS REVISED PLACING OST 2.43.8 ON A MONTHLY FREQUENCY. THERE WERE NO SAFETY IMPLICATIONS RESULTING FROM THIS EVENT. SATISFACTORY PERFORMANCE OF OST 2.43.8 ON 10/21/87 INDICATED THE CHANNEL WAS OPERABLE.

[16] BEAVER VALLEY 2 DOCKET 50-412 LER 87-035
LOW-LOW STEAM GENERATOR REACTOR TRIP DUE TO LOSS OF CONDENSATE PRESSURE.
EVENT DATE: 111087 REPORT DATE: 112387 NSSS: WE TYPE: PWR
VENDOR: MASONEILAN INTERNATIONAL, INC.

(NSIC 207352) ON 11/10/87, AT 2220 HOURS, WITH THE REACTOR AT 11% POWER FOLLOWING A STARTUP, AN ALARM INDICATING LOW CONDENSATE PUMP DISCHARGE PRESSURE WAS RECEIVED. AN ATTEMPT WAS MADE TO CORRECT THE PROBLEM BY OPENING THE CONDENSATE POLISHING BYPASS FLOW VALVE (2CNM-DCV100) FULL, WHILE REDUCING FLOW THROUGH THE CONDENSATE POLISHING DEMINERALIZERS. HOWEVER, THE PROBLEM PERSISTED, LEADING AT 2223 HOURS TO A TRIP OF THE RUNNING MAIN FEEDWATER PUMP (MFP) ON LOW SUCTION PRESSURE. A SUCCESSFUL RESTART OF THE PUMP WAS RETURNED, HOWEVER, TRIPPING THE MFP AGAIN AND PREVENTING THE START OF THE OTHER PUMP. ALTHOUGH BOTH MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS STARTED ON THE MFP TRIP, LOW-LOW LEVEL IN THE "C" STEAM GENERATOR CAUSED A REACTOR TRIP AT 2226 HOURS. THERE WERE NO SAFETY IMPLICATIONS AS ALL REACTOR PROTECTION AND ENGINEERED SAFEGUARDS ACTIVATED AS REQUIRED TO MITIGATE THE CONSEQUENCES OF THE EVENT. THE CAUSE OF THE LOW PRESSURE WAS FOUND TO BE A BENT DISC AND STEM ON 2CNM-DCV100, WHICH PREVENTED THE NEARLY CLOSED

VALVE FROM OPENING FURTHER IN RESPONSE TO ITS DEMAND SINGLE. THE VALVE WAS REPLACED BY A MODEL WITH AN INCREASED DIFFERENTIAL PRESSURE RATING. THE REACTOR WAS TAKEN CRITICAL AT 0345 HOURS ON 11/12/87.

[17] BEAVER VALLEY 2 DOCKET 50-412 LER 87-036
TURBINE TRIP/REACTOR TRIP DUE TO THRUST BEARING TRIP CAUSED BY PERSONNEL ERROR.
EVENT DATE: 111787 REPORT DATE: 121787 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC SUPPLY COMPANY

(NSIC 207471) ON 11/17/87 AT 1406 HOURS, AN I&C TECHNICIAN WORKING IN THE CONTROL ROOM BUMPED AND TRIPPED THE POWER SUPPLY SWITCH TO THE TURBINE THRUST BEARING CIRCUIT SUPERVISORY INSTRUMENT. THE RESULTANT SIGNAL SPIKE INITIATED A THRUST BEARING FAILURE SIGNAL WHICH CAUSED A TURBINE TRIP/REACTOR TRIP. THE 4KV BUS USST BREAKERS TRIPPED PROPERLY ON A FAST TRANSFER AND THE SSST BREAKERS CLOSED PROPERLY, HOWEVER, THE USST BREAKERS RECLOSED CAUSING THE TRIPPED GENERATOR TO BE PARALLED WITH THE SYSTEM GRID. THE USST'S WERE ALLOWED TO RECLOSE DUE TO A TRIP SIGNAL THAT DOES NOT SEAL IN. OVERCURRENT ON THE "D" USST CAUSED ALL FOUR USSTS TO RETRIP AND LOCK OPEN. WHILE THE SYSTEM GRID AND GENERATOR WERE PARALLED, THE (138KV) SYSTEM SUPPLY BREAKERS (OCB-94 & 85) TO THE "A,B,C & D" 4KV BUSSES TRIPPED OPEN. THIS RESULTED IN A TOTAL LOSS OF AC POWER WHICH PLACED THE RCS IN NATURAL CIRCULATION. BOTH EMERGENCY DIESEL GENERATORS STARTED ON AN UNDERVOLTAGE SIGNAL AND REENERGIZED THE "AE" AND "DF" EMERGENCY BUSSES. THE (138KV) SYSTEM SUPPLY BREAKER (OCB-85) RECLOSED RESTORING POWER TO THE "A & B" 4KV BUSSES. THE "A & B" REACTOR COOLANT PUMPS (RCP) WERE RESTARTED. FOLLOWING TESTING OF THE "B" SYSTEM STATION SERVICE TRANSFORMER, THE "C & D" 4KV BUSSES WERE REENERGIZED AND THE "C" RCP RESTARTED. THERE WERE NO SAFETY IMPLICATIONS RESULTING FROM THIS INCIDENT.

[18] BIG ROCK POINT DOCKET 50-155 LER 87-003 REV 03
UPDATE ON INOPERALLE PRIMARY SYSTEM SAFETY VALVES.
EVENT DATE: 012787 REPORT DATE: 112487 NSSS: GE TYPE: BWR
VENDOR: CROSBY VALVE & GAGE CO.

(NSIC 207160) ON JANUARY 27, 1987 SURVEILLANCE TESTING RESULTS SHOWED THAT THREE (3) PRIMARY SYSTEM SAFETY RELIEF VALVES INDICATED CONSISTENTLY HIGH INITIAL AS-FOUND SETPOINTS. TESTING AND EVALUATION SHOWED FAILURES OCCUR ON THE FIRST "LIFT" ONLY AND SUBSEQUENT ACTUATIONS ARE SATISFACTORY. FURTHER INVESTIGATION SHOWED THAT THE APPARENT CAUSE OF THE "STICKING" WAS DUE TO THE BONDING EFFECT OF LAPPING COMPOUND REMAINING ON THE SEAT AND DISK SURFACES FOLLOWING PREVIOUS REBUILDS. INADEQUATE PROCEDURES WITH REGARD TO CLEANING OF THE VALVE FOLLOWING LAPPING IS BELIEVED TO BE THE CAUSE OF THE FAILURES. TO CORRECT THE PROBLEMS, ALL SIX SAFETY VALVES HAVE BEEN REMOVED, REBUILT, CLEANED, AND TESTED PRIOR TO START-UP FROM THE PRESENT REFUELING OUTAGE. TO AVOID RECURRENCE, PROCEDURES WILL BE MODIFIED TO PROVIDE CLEANING REQUIREMENTS FOR THE VALVES FOLLOWING LAPPING REPAIRS. ON JUNE 1, 1987 FOLLOW-UP TESTING RESULTS INDICATE THAT "STICKING" IS STILL OCCURRING AS EVIDENCED BY HIGH "FIRST LIFT" PRESSURES. ALL SIX VALVES WERE REBUILT USING NEW 17-4 PH STAINLESS STEEL DISC INSERTS AND CLEANED USING ACETONE. VALVES WERE THEN SET USING DIRECT NITROGEN BOTTLE PRESSURE WITHOUT THE HYDRAULIC UNIT TO AVOID THE INTRODUCTION OF CONTAMINANTS DURING TESTING.

[19] BIG ROCK POINT DOCKET 50-155 LER 87-011
SPURIOUS UPSCALE/DOWNSCALE TRIP CAUSES REACTOR TRIP.
EVENT DATE: 110987 REPORT DATE: 120187 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 207242) ON NOVEMBER 09, 1987, WITH PLANT SHUTDOWN IN PROGRESS FOR A PLANNED MAINTENANCE OUTAGE, A REACTOR PROTECTION SYSTEM (JC) ACTUATION OCCURRED. POWER LEVEL AT THE TIME OF THE TRIP WAS LESS THAN .015 PERCENT AND ALL CONTROL RODS

(AA) WITHDRAWN SUCCESSFULLY INSERTED. ELECTRICAL NOISE INHERENT AT LOW POWER LEVELS CAUSED THE TRIP. ALTHOUGH NOT THE CAUSE OF THE TRIP, FOLLOWING REPAIR OF THE CHANNEL #1 PICOAMMETER, THE REACTOR PROTECTION SYSTEM WAS RETURNED TO SERVICE IN PREPARATION FOR PLANT START-UP FOLLOWING THE OUTAGE. PLANS ARE TO CHANGE OUT THE POWER RANGE INSTRUMENTATION TO THE NEW GENERAL ELECTRIC "NUMAC" LINE WHICH ARE EXPECTED TO BE LESS SUSCEPTIBLE TO NOISE PERTURBATIONS AT LOW POWER LEVELS. THIS IS NOT CONSIDERED A REACTOR TRIP WHILE CRITICAL PER INPO REPORTING CRITERIA.

[20] BIG ROCK POINT DOCKET 50-155 LER 87-012
REACTOR TRIP CAUSED BY FAILURE OF INTERMEDIATE RANGE NEUTRON MONITORING CHANNEL.
EVENT DATE: 112387 REPORT DATE: 121087 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 207356) ON NOVEMBER 23, 1987 PLANT STARTUP" WAS IN PROGRESS FOLLOWING A PLANNED MAINTENANCE OUTAGE. AT 0350 HOURS, AN INTERMEDIATE RANGE NEUTRON MONITORING CHANNEL FAILED UPSCALE CAUSING A SHORT PERIOD REACTOR TRIP. LOWER LEVEL AT THE TIME OF THE TRIP WAS LESS THAN 10E-6 PERCENT DURING AN APPROACH TO CRITICAL. ALL CONTROL RODS WITHDRAWN AT TIME OF THE TRIP SUCCESSFULLY INSERTED. CAUSE IS ATTRIBUTED TO AMPLIFIER ELECTRONIC TUBE FAILURE DUE TO AGE. FOLLOWING CHANGE-OUT OF THE AMPLIFIER AND TESTING, RESTART COMMENCED. THIS IS NOT CONSIDERED A REACTOR TRIP WHILE CRITICAL PER INPO REPORTING CRITERIA.

[21] BRAIDWOOD 1 DOCKET 50-456 LER 87-031 REV 01
UPDATE ON CONTROL ROOM VENTILATION SHIFT TO THE EMERGENCY MAKEUP MODE AS A RESULT OF SPURIOUS ACTUATION OF A RADIATION MONITOR DUE TO INADVERTENT RADIO OPERATION.
EVENT DATE: 061387 REPORT DATE: 112087 NSSS: WE TYPE: PWR

(NSIC 207334) AT 1908 ON JUNE 13, 1987, AND AGAIN ON OCTOBER 16, 1987, IT WAS DISCOVERED THROUGH MAIN CONTROL ROOM ANNUNCIATION THAT TRAIN OA OF THE CONTROL ROOM VENTILATION SYSTEM HAD SHIFTED TO ITS EMERGENCY MAKEUP MODE OF OPERATION. THESE ACTUATIONS WERE ATTRIBUTED TO INADVERTENTLY KEYING RADIOS WITHIN THE DESIGNATED EXCLUSION AREA NEAR RADIATION MONITORS OPR31J AND OPR32J. THE ELECTRIC PULSE GENERATED BY THE RADIO IS A NOISE SIGNAL WHICH ULTIMATELY GETS TRANSLATED INTO ACTIVITY BY THE MONITOR. IN BOTH OCCURRENCES ALL MONITOR CHANNEL ACTIVITY READINGS RETURNED TO NORMAL WITHIN 30 MINUTES AND THE LINEUP FOR CONTROL ROOM VENTILATION WAS SUBSEQUENTLY RESTORED. THE FIRST OCCURRENCE IS CONSIDERED TO BE AN ISOLATED EVENT AND NO ADDITIONAL ACTION WAS TAKEN. AFTER THE SECOND OCCURRENCE, THE POSSIBILITIES OF FURTHER PROTECTING THESE CONTROL ROOM VENTILATION MONITORS AGAINST ERRANT RADIO KEYS ARE BEING EXPLORED. THERE HAVE BEEN NO PREVIOUS OCCURRENCES DUE TO KEYING A RADIO CAUSING AN ENGINEERED SAFETY FEATURE ACTUATION.

[22] BRAIDWOOD 1 DOCKET 50-456 LER 87-059
MISSED SURVEILLANCE DUE TO LOST DATA SHEET.
EVENT DATE: 090587 REPORT DATE: 120487 NSSS: WE TYPE: PWR

(NSIC 207408) ON NOVEMBER 7, 1987, DURING THE REVIEW OF THE SEPTEMBER 5, 1987 UNIT ONE SHIFTLY AND DAILY OPERATING SURVEILLANCE PACKAGE, 1BWOS 0.1-1,2,3, IT WAS DISCOVERED THAT DATA SHEET 11, UNIT ONE MODE 1,2,3, TECH SPEC DATA SHEET PRIMARY PLANT AND ACCUMULATOR OPERABILITY, WAS MISSING. THE SURVEILLANCE PACKAGE HAD BEEN DISASSEMBLED TO ALLOW EASE IN COLLECTING DATA. THE PACKAGE WAS LATER REASSEMBLED, AND REVIEWED BY SHIFT PERSONNEL. WHEN THE PACKAGE WAS REVIEWED BY THE OPERATING STAFF, DATA SHEET 11 WAS NO LONGER PART OF THE PACKAGE. AN EXTENSIVE SEARCH WAS CONDUCTED TO LOCATE THE MISSING SHEET, BUT THE SEARCH WAS UNSUCCESSFUL. THE CAUSE OF THE EVENT WAS THE METHODOLOGY USED AT THE TIME OF THE SURVEILLANCE. THE IMMEDIATE CORRECTIVE ACTION WAS TO CONDUCT A SEARCH FOR THE MISSING SHEET. ACTION TO PREVENT RECURRENCE WAS TO RESTRUCTURE THE SURVEILLANCE

SO THAT DISASSEMBLY IS NO LONGER NECESSARY. THERE HAVE BEEN NO PREVIOUS OCCURRENCES OF MISSED SURVEILLANCES DUE TO THIS CAUSE.

[23] BRAIDWOOD 1 DOCKET 50-456 LER 87-048
BOTH SYSTEM AUXILIARY TRANSFORMER TRIP RESULTING IN LOSS OF OFFSITE POWER AND DELUGE SYSTEM ACTUATION.
EVENT DATE: 091187 REPORT DATE: 100787 NSSS: WE TYPE: PWR

(NSIC 206733) AT 1425 ON SEPTEMBER 11, 1987, DURING THE PERFORMANCE OF A DELUGE SYSTEM SURVEILLANCE, BOTH SYSTEM AUXILIARY TRANSFORMERS TRIPPED. THE DELUGE SYSTEM ACTUATED AS A RESULT OF A MISPOSITIONED AUXILIARY DRAIN VALVE. THIS RESULTED IN A LOSS OF OFFSITE POWER. INVESTIGATION ALSO TO THE SOURCE OF THE MISPOSITIONED VALVE REVEALED NO SPECIFIC REASON FOR THE VALVE MANIPULATION OR DOCUMENTATION OTHER THAN A PREVIOUS UNRELATED SURVEILLANCE. THE UNIT NORMAL AC POWER LINEUP WAS RESTORED AT 1518 ON SEPTEMBER 11, 1987. ALL ENGINEERED SAFETY FEATURE SYSTEMS OPERATED AS DESIGNED. TO PREVENT RECURRENCE, THE PROCEDURE FOR THE SURVEILLANCE HAS BEEN CHANGED TO ENSURE THE AUXILIARY DRAIN VALVE IS IN ITS CORRECT POSITION PRIOR TO OPENING THE MAIN DRAIN VALVE. ALSO, THE TRANSFORMER TRIP AND THE COMMON DRAIN LINE ARE BEING EVALUATED TO REMOVE THE TRIP AND PREVENT SYSTEM INTERACTION THROUGH THE COMMON DRAIN LINE. ADDITIONALLY, IT WAS DISCOVERED THAT IT IS POSSIBLE TO INADVERTENTLY ACTUATE THE DELUGE SYSTEM AT THE LOCAL ELECTRICAL SWITCH. A MECHANICAL GUARD IS BEING ADDED TO PREVENT THIS FROM OCCURRING. THERE HAVE BEEN NO PREVIOUS OCCURRENCES.

[24] BRAIDWOOD 1 DOCKET 50-456 LER 87-054
CALORIMETRIC CALCULATION SURVEILLANCES PERFORMED OUTSIDE OF THE TECHNICAL SPECIFICATION TIME LIMIT DUE TO PERSONNEL ERROR AND OPERATIONAL RESTRICTIONS.
EVENT DATE: 102187 REPORT DATE: 111987 NSSS: WE TYPE: PWR

(NSIC 207239) ON OCTOBER 7, 1987, DOCUMENTATION OF THE CALORIMETRIC CALCULATION WAS LOST AS A RESULT OF A PERSONNEL ERROR BY THE SHIFT ENGINEER AND STATION CONTROL ROOM ENGINEER (SCRE) IN THAT THE SURVEILLANCE WAS NOT COMPLETED AS REQUIRED BY THE TECH SPECS. THIS WILL BE REVIEWED WITH INDIVIDUALS INVOLVED STRESSING IMPORTANCE OF UTILIZING PROVISIONS ALLOWED TO COMPLETE SURVEILLANCES. ON OCTOBER 21 AND 27, 1987, THE CALORIMETRIC CALCULATIONS WERE PERFORMED OUTSIDE THE TIME ALLOWED BY TECH SPECS AS A RESULT OF REACTOR POWER NOT BEING CONSTANT. THIS WAS DUE TO CONDENSATE/CONDENSATE BOOSTER PUMP INLET STRAINER CLOGGING, CAUSING VARIATION IN FEEDWATER FLOW TO THE STEAM GENERATORS. THE PREREQUISITES SPECIFIED IN THE SURVEILLANCE PROCEDURE COULD NOT BE MET. ADDITIONAL GUIDANCE AND ADMINISTRATIVE CONTROLS FOR COMPLETING SURVEILLANCE REQUIREMENTS HAVE BEEN INITIATED. ON OCTOBER 29, 1987, THE CALORIMETRIC CALCULATION WAS PERFORMED OUTSIDE THE TIME ALLOWED BY TECH SPEC AS A RESULT OF AN IMPROPER TURNOVER BY THE MIDNIGHT SHIFT SCRE WITH THE DAY SHIFT SCRE. INDIVIDUALS WILL BE COUNSELLED STRESSING IMPORTANCE OF CONDUCTING A THOROUGH TURNOVER AND MAINTAINING COGNIZANCE OF SURVEILLANCE TRACKING. NO PREVIOUS REPORT ISSUED ON THIS TOPIC.

[25] BRAIDWOOD 1 DOCKET 50-456 LER 87-058
BOTH TRAINS OF CONTROL ROOM VENTILATION INOPERABLE DUE TO INCORRECT DESIGN INCORPORATION.
EVENT DATE: 110687 REPORT DATE: 120487 NSSS: WE TYPE: PWR

(NSIC 207344) ON NOVEMBER 6, 1987, DURING A PROJECT ENGINEERING DEPARTMENT REVIEW OF CONTROL ROOM VENTILATION STARTUP TEST, BWSU VC-30, AND ENGINEERING DESIGN CHANGE DC-VC041, IT WAS DETERMINED THAT THE DESIGN CHANGE WAS INCORRECT. IT WAS NOTED THAT THE 'B' TRAIN HEATERS FOR THE EMERGENCY MAKEUP FILTER UNIT DID NOT ENERGIZE WHEN THE FAN WAS STARTED. A REVIEW OF SYSTEM SCHEMATIC DIAGRAMS AND INSTRUMENT DATA SHEETS FOR THE HEATERS WERE CHECKED IN AN EFFORT TO DETERMINE THE CAUSE OF THE HEATERS NOT STARTING DURING THE TEST. IT WAS TRACED TO A HEATER

INTERLOCK LOGIC DEFICIENCY. THIS WAS CONFIRMED BY ARCHITECT ENGINEER PERSONNEL. THE CAUSE OF THE EVENT IS A PRESERVICE TESTING DEFICIENCY IN THAT THE DESIGN CHANGE WAS NOT SPECIFICALLY VERIFIED BY COMPONENT DEMONSTRATION OR RETEST. BOTH TRAINS OF CONTROL ROOM VENTILATION WERE THEN DECLARED INOPERABLE. A TEMPORARY ALTERATION WAS IMMEDIATELY PROCESSED TO CORRECT THE HEATER INTERLOCK LOGIC DEFICIENCY ALLOWING THE HEATER TO OPERATE AS DESIGNED. THERE HAVE BEEN NO PREVIOUS OCCURRENCES OF PRESERVICE TESTING DEFICIENCIES RESULTING FROM DESIGN CHANGES.

[26] BROWNS FERRY 1 DOCKET 50-259 LER 87-017 REV 01
UPDATE ON IMPROPER BYPRODUCT MATERIAL INVENTORY AND RECORDS RETENTION RESULTS IN FAILURE TO MEET TECH SPEC REQUIREMENTS.
EVENT DATE: 071787 REPORT DATE: 110387 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 207247) ON JULY 17, 1987, PLANT OPERATIONS WAS NOTIFIED THAT 18 SEALED CHECK SOURCES, EACH CONTAINING APPROXIMATELY FIVE MICROCURIES OF STRONTIUM-90, HAD BEEN FOUND IN A LOCKED STORAGE CAGE IN A REGULATED AREA. FIFTEEN OF THE SOURCES, SHIELDED INSIDE OF RADIATION MONITOR DETECTOR CASES WERE DISCOVERED BY NRC RESIDENT INSPECTORS VERIFYING THE ADEQUACY OF THE STORAGE LOCATION. THE SOURCES WERE LEAK CHECKED AND ADDED TO THE INVENTORY. ON JULY 31, 1987, PLANT PERSONNEL CONDUCTING A RECORD SEARCH DETERMINED THAT ANOTHER STRONTIUM-90 SEALED SOURCE HAD BEEN DISPOSED OF IN THE FALL OF 1984. NO RECORDS CAN BE FOUND AT THIS TIME TO DOCUMENT THE REASON FOR ITS DISPOSAL OR HOW IT WAS DISPOSED OF. ON SEPTEMBER 30, 1987, THE RECORD AND INVENTORY SEARCH FOR BYPRODUCT SOURCES WAS COMPLETED. THIS SEARCH IDENTIFIED EIGHT NONEXEMPT SOURCES NOT CURRENTLY ON THE PLANT INVENTORY. IMMEDIATELY FOLLOWING THE COMPLETION OF THIS SEARCH, EQUIPMENT WAS IDENTIFIED IN THE WAREHOUSE WHICH CONTAINED SEVEN ADDITIONAL NONEXEMPT SOURCES. THIS ADDITIONAL EFFORT WAS COMPLETED ON OCTOBER 15, 1987. VARIOUS TECH SPEC REQUIREMENTS REGARDING PROPER RADIOACTIVE MATERIAL INVENTORY AND RECORD RETENTION WERE NOT SATISFIED. ACCOUNTABILITY AND PROCEDURAL GUIDANCE FOR THE CONTROL OF RADIOACTIVE MATERIALS WILL BE REVISED.

[27] BROWNS FERRY 1 DOCKET 50-259 LER 87-029
PERSONNEL ERROR IN WRITING EQUIPMENT TAG-OUT CLEARANCE RESULTS IN ACTUATIONS OF ENGINEERED SAFETY FEATURES.
EVENT DATE: 110587 REPORT DATE: 120487 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 207316) ON NOVEMBER 5, 1987, AT 1810 HOURS, A FUSE WAS REMOVED FROM A PANEL IN ORDER TO DEENERGIZE A PRIMARY CONTAINMENT ISOLATION VALVE FOR MODIFICATION ACTIVITIES. THIS ACTION RESULTED IN A PRIMARY CONTAINMENT VENTILATION ISOLATION, AUTO START OF TWO TRAINS OF THE STANDBY GAS TREATMENT SYSTEM AND ONE TRAIN OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM. THESE ENGINEERED SAFETY FEATURE ACTUATIONS WERE THE RESULT OF IMPROPER IDENTIFICATION OF THE ISOLATION BOUNDARIES. RESEARCH INTO THE SWITCHING NECESSARY TO ELECTRICALLY ISOLATE THE VALVE WAS INADEQUATE DURING PREPARATION OF THE CLEARANCE. THE INDIVIDUALS INVOLVED HAVE BEEN CAUTIONED TO CHECK ALL PRINTS CAREFULLY TO ENSURE THAT NO UNEXPECTED EVENTS OCCUR. A CRITIQUE OF THIS INCIDENT HAS BEEN PREPARED FOR REVIEW BY OPERATIONS PERSONNEL TO REEMPHASIZE THE NEED FOR ADEQUATE RESEARCH IN WRITING A CLEARANCE.

[28] BROWNS FERRY 2 DOCKET 50-260 LER 87-011
FAILURE OF 2B REACTOR PROTECTION SYSTEM MOTOR GENERATOR SET MOTOR INITIATES ENGINEERED SAFETY FEATURES.
EVENT DATE: 110187 REPORT DATE: 112587 NSSS: GE TYPE: BWR

OTHER UNITS INVOLVED: BROWNS FERRY 1 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 207317) ON NOVEMBER 1, 1987, AT 1117 HOURS, WITH ALL THREE UNITS DEFUELED, THE UNIT 2 REACTOR PROTECTION SYSTEM (RPS) MOTOR GENERATOR (MG) SET 2B MOTOR FAILED. THIS CAUSED THE NORMAL ENERGIZED RPS BUS B TO LOSE POWER WHICH INITIATED STANDBY GAS TREATMENT, CONTROL ROOM EMERGENCY VENTILATION, AND A HALF SCRAM ON UNIT 2. THE UNIT 2 REACTOR VENTILATION ISOLATED, THE REFUELING ZONE VENTILATION ISOLATED, AND AN ISOLATION SIGNAL WAS GIVEN TO THE OUTBOARD ISOLATION VALVES ON REACTOR WATER CLEANUP, RESIDUAL HEAT REMOVAL, PRIMARY CONTAINMENT VENTILATION, AND TRAVERSING INCORE PROBE. NO VALVE MOVEMENT OCCURRED FOR REACTOR WATER CLEANUP, RESIDUAL HEAT REMOVAL AND TRAVERSING WAS REPLACED AND THE 2B RPS MG SET WAS RETURNED TO SERVICE AT 0945 HOURS, ON NOVEMBER 2, 1987. THE CAUSE OF THE MOTOR FAILURE HAS NOT BEEN DETERMINED AS OF THE DATE OF THIS REPORT.

[29] BROWNS FERRY 3 DOCKET 50-296 LER 87-004
DIESEL GENERATOR 3EB START DUE TO PERSONNEL ERROR DURING MAINTENANCE.
EVENT DATE: 101387 REPORT DATE: 111087 NSSS: GE TYPE: BWR

(NSIC 206965) ON OCTOBER 13, 1987, DIESEL GENERATOR (DG) 3EB WAS INADVERTENTLY STARTED DURING PERFORMANCE OF PREVENTATIVE MAINTENANCE. AN ELECTRICAL TECHNICIAN ACCIDENTLY SHORTED THE DG AUTO START RELAY WHILE CONNECTING TEST EQUIPMENT TO THE CONTROL CIRCUITRY. AFTER THE DG STARTED, IT TRIPPED ON OVERSPEED. THIS TRIP HAS BEEN ATTRIBUTED TO THE GOVERNOR BEING OUT OF ADJUSTMENT. TRAINING WILL BE GIVEN TO ELECTRICAL TECHNICIANS INVOLVED IN THIS TYPE OF MAINTENANCE. ALL EIGHT DGS GOVERNORS WERE ADJUSTED BY A VENDOR TECHNICIAN.

[30] BROWNS FERRY 3 DOCKET 50-296 LER 87-005
UNPLANNED DIESEL GENERATOR START DUE TO RELAY FAILURE.
EVENT DATE: 101987 REPORT DATE: 111787 NSSS: GE TYPE: BWR
VENDOR: BROWN BOVERI

(NSIC 207130) AT 1315 HOURS ON OCTOBER 19, 1987, WITH ALL THREE UNITS DEFUELED, DEGRADED VOLTAGE LOGIC WAS COMPLETED ON THE 3ED 4160V SHUTDOWN BOARD. THE 3D DIESEL GENERATOR RECEIVED AN AUTOMATIC START SIGNAL. THE NORMAL SUPPLY BREAKERS TRIPPED. THE DIESEL GENERATOR CAME UP TO SPEED AND TIED ON TO THE SHUTDOWN BOARD. THIS WAS AN UNPLANNED INITIATION OF AN ENGINEERED SAFETY FEATURE. AN INVESTIGATION WAS CONDUCTED TO DETERMINE THE CAUSE OF THE INITIATION. IT WAS DETERMINED THAT TWO OF THE THREE DEGRADED VOLTAGE RELAYS MONITORING THE 3ED 4160V SHUTDOWN BOARD HAD FAILED THEREFORE COMPLETING THE MINIMUM ACTUATION LOGIC FOR THE START OF THE DIESEL GENERATOR. AT 1352 HOURS, THE SUPPLY BREAKER FROM THE DIESEL GENERATOR WAS OPENED. BY 1405 HOURS, THE DIESEL COASTDOWN HAD BEEN COMPLETED AND THE DIESEL WAS SHUT DOWN. THE FAILED RELAYS WERE REPLACED AND THE LOGIC WAS TESTED. THE SHUTDOWN BOARD AND THE DIESEL GENERATOR WERE CHECKED AND AT 2357 HOURS THE DIESEL GENERATOR AND SHUTDOWN BOARD WERE DECLARED OPERABLE. THE INVESTIGATION IS CONTINUING INTO THE CAUSE OF THE RELAY FAILURES.

[31] BRUNSWICK 2 DOCKET 50-324 LER 87-010
INOPERABILITY OF REACTOR BUILDING FIRE HOSE STATION 2-RB-23 RESULTING FROM PERSONNEL ERROR DURING/FOLLOWING FIRE DRILL.
EVENT DATE: 110487 REPORT DATE: 120487 NSSS: GE TYPE: BWR

(NSIC 207297) EIIS COMPONENT DESCRIPTION UNAVAILABLE FOR THE TIME PERIOD OF APPROXIMATELY 1030 HOURS ON 11/4/87 UNTIL TIME OF DISCOVERY, AT 2140 HOURS ON 11/4/87, UNIT 2 REACTOR BUILDING FIRE HOSE STATION 2.RB-23 WAS INOPERABLE DUE TO REMOVAL OF THE HOSE STATION FIRE HOSE. FOLLOWING DISCOVERY OF THIS EVENT DURING ROUTINE OPERATOR SURVEILLANCE, THE HOSE STATION WAS RETURNED TO STANDBY READINESS AT 2155 HOURS ON 11/4/87. UNIT 2 WAS AT 82% POWER APPROACHING A SCHEDULED UNIT

REFUEL/MAINTENANCE OUTAGE IN JANUARY 1988. THE SUBJECT FIRE HOSE STATION WAS MADE INOPERABLE DURING THE PERFORMANCE OF A FIRE DRILL WHEN THE HOSE STATION FIRE HOSE WAS INTENTIONALLY REMOVED TO PROVIDE ADDITIONAL HOSE REACH CAPABILITY TO ANOTHER FIRE HOSE STATION DIRECTLY AFFECTED BY THE DRILL SCENARIO. FIRE BRIGADE PERSONNEL INVOLVED WITH THE FIRE DRILL FAILED TO RECOGNIZE THAT REMOVAL OF THE FIRE HOSE RENDERED THE FIRE HOSE STATION INOPERABLE. AS A RESULT OF THIS EVENT APPROPRIATE PROCEDURAL REVISIONS WILL BE IMPLEMENTED, BY 1/8/88, TO ENSURE SYSTEMS AFFECTED BY A FIRE DRILL ARE PROPERLY RESTORED FOLLOWING TERMINATION OF THE DRILL. IN ADDITION, BY 3/31/88, APPROPRIATE OPERATIONS REAL TIME TRAINING WILL BE CONDUCTED WITH THE PLANT RADWASTE/FIRE PROTECTION OPERATING SHIFTS.

[32] BRUNSWICK 2 DOCKET 50-324 LER 87-011
DIVISION I PRIMARY CONTAINMENT GROUP 5 ISOLATION OF REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM DUE TO SPURIOUS ACTUATION OF RCIC STEAM LEAK DETECTION INSTRUMENTATION.
EVENT DATE: 110587 REPORT DATE: 120487 NSSS: GE TYPE: BWR
VENDOR: CHESTERTON PACKING & SEALS

(NSIC 207365) AT 2316 HOURS ON 11/5/87, THE UNIT 2 REACTOR CORE ISOLATION COOLING (RCIC) (E51) SYSTEM TURBINE STEAM SUPPLY INBOARD PRIMARY CONTAINMENT ISOLATION VALVE (PCIV) AND THE PUMP SUPPRESSION POOL SUCTION SUPPLY INBOARD PCIV RECEIVED AUTO-CLOSE SIGNALS ON AN A LOGIC (DIVISION 1) PRIMARY CONTAINMENT GROUP 5 ISOLATION. THE SUPPRESSION POOL SUCTION SUPPLY INBOARD VALVE WAS ALREADY CLOSED IN A NORMAL CONFIGURATION. UNIT 2 WAS AT 82% POWER WHILE IN A FUEL DEPLETION COAST DOWN APPROACHING A SCHEDULED CIRCUIT REFUEL/MAINTENANCE OUTAGE IN JANUARY 1988. THE UNIT HIGH PRESSURE COOLANT INJECTION, AUTOMATIC DEPRESSURIZATION, REACTOR CORE SPRAY, AND RESIDUAL HEAT REMOVAL/LOW PRESSURE COOLANT INJECTION SYSTEMS WERE OPERABLE. FOLLOWING A DETERMINATION THAT THE EVENT RESULTED FROM A SPURIOUS ISOLATION SIGNAL, THE RCIC SYSTEM WAS RETURNED TO STANDBY READINESS AT 2335 HOURS ON 11/5/87. THE CAUSE OF THIS EVENT COULD NOT BE REVEALED. AN INVESTIGATION DETERMINED THE ISOLATION MAY HAVE RESULTED FROM SPURIOUS ACTUATION OF RCIC EQUIPMENT ROOM VENTILATION DIFFERENTIAL TEMPERATURE SWITCH (DTS) E51.DTS.N601A AND/OR ONE OR BOTH OF RCIC STEAM LINE TEMPERATURE SWITCHES (TS) E51.TS.3321 AND 3323.

[33] CALLAWAY 1 DOCKET 50-483 LER 87-013 REV 01
UPDATE ON INOPERABLE CONTROL ROOM EMERGENCY VENTILATION SYSTEM WHEN PERSONNEL ERROR CAUSED PRESSURE BOUNDARY BREACH AND FLOW BALANCING DAMPER WAS MISPOSITIONED.
EVENT DATE: 060887 REPORT DATE: 110487 NSSS: WE TYPE: PWR
VENDOR: AMERICAN WARMING & VENTILATING INC.

(NSIC 206885) 1) CONTROL ROOM (CR) EMERGENCY VENTILATION SYSTEM (CREVS) TRAIN 'A' FAILED TO MEET A POSITIVE PRESSURE TEST ON 7/9/87 DUE TO UNSEALED PENETRATIONS AT THE CR PRESSURE BOUNDARY BREACHED ON 6/8, 6/11, AND 7/6, 2) ON 8/14/87 A MISPOSITIONED DAMPER WAS DISCOVERED ON CREVS TRAIN B'. THE IMPACT ON CR DOSES AS ANALYZED IN THE FINAL SAFETY ANALYSIS REPORT WAS INITIALLY INCORRECTLY ASSESSED. SUBSEQUENT EVALUATION INDICATES TRAIN 'B' WAS INOPERABLE FOR AN UNDETERMINED PERIOD OF TIME. SINCE TRAIN 'A' HAD BEEN ROUTINELY REMOVED FROM SERVICE FOR MAINTENANCE AND TESTING, BOTH TRAINS HAVE BEEN SIMULTANEOUSLY RENDERED INOPERABLE AND TECHNICAL SPECIFICATION 3.0.3 UNKNOWINGLY ENTERED. THE UNIT WAS IN MODE 1 POWER OPERATION AT 100% POWER FOR DISCOVERY OF BOTH EVENTS. EVENT 1 WAS DUE TO UTILITY PERSONNEL NOT RECOGNIZING THAT THE PENETRATIONS WOULD AFFECT CR PRESSURIZATION DURING PLANNING OF THE DESIGN WORK. EVENT 2'S CAUSE CANNOT BE DETERMINED. PENETRATIONS WERE SEALED AND A NIGHT ORDER WAS ISSUED TO TEMPORARILY PROHIBIT CR BREACHES. PROCEDURE GUIDANCE WILL BE MADE FOR EVALUATING THE BREACHING OF BARRIERS AT THE CR BOUNDARY. TRAINS 'A' AND 'B' WERE FLOW BALANCED. DAMPER BALANCE POSITIONS WILL BE ADDED TO OPERATING PROCEDURES AND THE DAMPERS WILL BE POSITIVELY SECURED IN POSITION.

[34] CALLAWAY 1 DOCKET 50-483 LER 87-030
THREE ENGINEERED SAFETY FEATURE ACTUATIONS DUE TO UPSCALE SIGNAL SPIKES TO RADIATION MONITORS UPON LOSS OF SAFEGUARDS BUS NB01.
EVENT DATE: 102387 REPORT DATE: 112387 NSSS: WE TYPE: PWR

(NSIC 207204) ON 10/23, 10/28, AND 10/29/87, ENGINEERED SAFETY FEATURES (ESF) ACTUATIONS OCCURRED UPON LOSS OF POWER TO RADIATION (RAD) MONITORS WHICH RESULTED IN A CONTROL ROOM VENTILATION ISOLATION AND CONTAINMENT PURGE ISOLATION. THE PLANT WAS IN MODE 5 - COLD SHUTDOWN. THE ROOT CAUSE OF THESE EVENTS IS ATTRIBUTABLE TO UPSCALE SPIKES FROM RAD MONITOR SIGNAL ISOLATORS. THESE SPIKES WERE PRODUCED WHEN THE MONITOR POWER SUPPLY WAS LOST UPON LOSS OF THE 4.16 KILO-VOLT (KV) ESF BUS, NB01. IMMEDIATE ACTIONS WERE TAKEN TO RESTORE NB01. FACTORS WHICH LED TO EACH LOSS OF NB01 WERE AS FOLLOWS: 1. 10/23 - A MAIN SWITCHYARD BREAKER TRIPPED OPEN. THE CAUSE CANNOT BE DETERMINED. 2. 10/28 - UTILITY OPERATIONS PERSONNEL WERE NOT AWARE THAT FOUR LOAD SHED EMERGENCY LOAD SEQUENCER (LSELS) TRAIN 'A' BISTABLES HAD BEEN DE-ENERGIZED FOR MAINTENANCE. AS THEY POWERED UP LSELS, A DEGRADED VOLTAGE SIGNAL TRIPPED NB01'S FEEDER BREAKER. THE PROCEDURE DID NOT ADEQUATELY DEFINE POWER-UP OPERATIONS OF LSELS. 3. UTILITY RELAY PERSONNEL VERBALLY TRANSPOSED RELAY NUMBERS DURING A 345KV BUS OPERATIONAL TEST WHICH LED TO OPERATION OF THE WRONG RELAY AND TRIPPED A MAIN SWITCHYARD BREAKER. AN EVALUATION IS UNDERWAY TO CONSIDER A DESIGN CHANGE FOR THE METHOD OF ACTUATING ESF SYSTEMS FROM RAD MONITORS. PROCEDURES WERE REVISED TO PROVIDE MORE DETAIL FOR RESTORING THE LSELS.

[35] CALLAWAY 1 DOCKET 50-483 LER 87-031
TWO SAFETY INJECTION ACCUMULATORS INOPERABLE DURING TESTING WHEN NITROGEN PRESSURE RELIEF VALVE LIFTED DUE TO HIGH PRESSURE REGULATOR SETPOINT.
EVENT DATE: 110487 REPORT DATE: 120487 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 207286) FROM 0838 TO 0848 CST ON 11/4/87, TWO SAFETY INJECTION ACCUMULATORS WERE SIMULTANEOUSLY INOPERABLE AND TECHNICAL SPECIFICATION 3.0.3 WAS ENTERED. ACCUMULATOR 'B' WAS RENDERED INOPERABLE FOR CHECK VALVE SURVEILLANCE TESTING. ACCUMULATOR 'D' BECAME INOPERABLE DURING THE TEST WHEN A NITROGEN (N2) SUPPLY RELIEF VALVE LIFTED WHILE LICENSED UTILITY OPERATORS WERE REPRESSURIZING 'D' WHICH HAD DEPRESSURIZED AS A RESULT OF LEAKAGE PAST CHECK VALVE EP-8818D. FURTHER TESTING WAS SUSPENDED. THE PLANT WAS IN MODE 3 - HOT STANDBY. THE SETPOINT FOR THE N2 PRESSURE REGULATOR WAS FOUND HIGHER THAN THE DOWNSTREAM RELIEF VALVE. FURTHER INVESTIGATION REVEALED THE REGULATOR TO BE OUT OF CALIBRATION. THE CHECK VALVE FAILED TO FULLY SEAT WHICH MAY OCCUR DUE TO THE AP UTILIZED DURING TESTING. EP-8818D SEATED ITSELF UPON RE-INITIATING THE TEST. THE REGULATOR SETPOINT HAS BEEN SET BELOW THE RELIEF VALVE SETPOINT. CALIBRATING THE REGULATOR WILL BE EVALUATED FOR INCLUSION IN THE PREVENTATIVE MAINTENANCE PROGRAM. SURVEILLANCE PROCEDURE ENHANCEMENTS WERE MADE AND ARE PLANNED TO ASSIST OPERATORS IN MAINTAINING ACCUMULATOR OPERABILITY.

[36] CALLAWAY 1 DOCKET 50-483 LER 87-032
REACTOR TRIP ON LOW STEAM GENERATOR LEVEL OSCILLATIONS FOLLOWING LEAKING MAIN STEAM ISOLATION VALVE RETEST IN MODE NOT ALLOWED BY PROCEDURE.
EVENT DATE: 110887 REPORT DATE: 120887 NSSS: WE TYPE: PWR

(NSIC 207313) ON 11/8/87 AT 0252 CST, A REACTOR TRIP, FEEDWATER ISOLATION AND AUXILIARY FEEDWATER ACTUATION OCCURRED AS A RESULT OF LOW LEVEL IN STEAM GENERATOR (S/G) 'B'. THE PLANT WAS IN MODE 2 - STARTUP AT APPROXIMATELY ONE PERCENT REACTOR POWER. THE REACTOR COOLANT SYSTEM (RCS) TEMPERATURE WAS 553.4F (AVERAGE). RCS PRESSURE WAS 2188.5 PSIG. THE LOW S/G LEVEL OCCURRED AS A RESULT OF S/G LEVEL OSCILLATIONS FOLLOWING THE RETEST OF LEAKING MAIN STEAM ISOLATION VALVE (MSIV) 'B' AT 0229. THE LICENSED OPERATORS FAILED TO FOLLOW THE TEST PROCEDURE WHICH REQUIRED PERFORMANCE OF THE TEST ONLY IN MODE 3 - HOT STANDBY.

ADDITIONALLY, THE 'B' MAIN STEAM DUMP VALVE GROUP WAS CYCLING DUE TO A LOW CONTROL BAND. THIS MAGNIFIED THE PERTURBATION OF THE TEST ON THE PLANT. LICENSED OPERATORS WERE UNABLE TO STABILIZE THE OSCILLATIONS; THE REACTOR TRIPPED AT 0252. THE OPERATORS RECOVERED FROM THE TRIP VIA PLANT PROCEDURES AND STABILIZED PLANT CONDITIONS BY 0258. TO PREVENT RECURRENCE, FUTURE SIMILAR MSIV RETESTS WILL BE PERFORMED IN MODE 3. THE LEAKING MSIV WAS REPAIRED. THE CONTROL BAND FOR THE STEAM DUMP VALVE WAS CORRECTED. PROGRESSIVE DISCIPLINE WAS INITIATED WITH THE LICENSED OPERATORS INVOLVED. THE EVENT WAS ALSO DISCUSSED WITH THE OPERATORS TO EMPHASIZE THE NEED TO REVIEW RETEST REQUIREMENTS.

[37] CALVERT CLIFFS 1 DOCKET 50-317 LER 87-015
REACTOR TRIP DUE TO TRANSFORMER SHORT.
EVENT DATE: 111187 REPORT DATE: 121187 NSSS: CE TYPE: PWR
VENDOR: MOORE PRODUCTS COMPANY

(NSIC 207364) AT 1626 ON NOVEMBER 11, 1987 UNIT 1 TRIPPED FROM 100% POWER DUE TO THE FAILURE OF THE EAST SECONDARY BUSHING (H.3, 1 PHASE) OF TRANSFORMER U-25000-12. THE BUSHING APPARENTLY SHORTED DUE TO WEATHER RELATED PROBLEMS. THE TURBINE GENERATOR TRIPPED ON TRANSFORMER HIGH-SIDE LEADS DIFFERENTIAL. THE REACTOR TRIPPED ON LOSS OF TURBINE LOAD. THE PLANT WAS BROUGHT TO A SAFE SHUTDOWN CONDITION. SEVERAL UNRELATED PROBLEMS WERE ENCOUNTERED DURING IMPLEMENTATION OF THE SAFE SHUTDOWN PROCEDURES. 1. THE TURBINE BYPASS VALVE CONTROLLER DID NOT OPERATE PROPERLY. 2. THE PLANT COMPUTER WAS OUT OF SERVICE DURING THE EVENT. HOWEVER, STRIP CHART RECORDERS WERE FUNCTIONING AND PROVIDED THE DATA NECESSARY TO PROPERLY ANALYZE THE EVENT. 3 NO. 11 ATMOSPHERIC DUMP VALVE (ADV) STUCK OPEN AS IT WAS BEING MANUALLY CYCLED AFTER THE TRIP. 4. AFAS ACTUATED BUT DID NOT LOCK IN. HOWEVER, THE MOTOR DRIVEN AUXILIARY FEEDWATER PUMP (NO. 13) DID START AND WAS SECURED FOLLOWING VERIFICATION OF MAIN FEEDWATER FLOW. CORRECTIVE ACTIONS ARE AS FOLLOWS: - 1. THE ADV IS BEING SCHEDULED FOR A MAINTENANCE OVERHAUL. 2. A FACILITIES CHANGE REQUEST (FCR 84.149) HAS BEEN GENERATED TO INSTALL A TIME DELAY ON THE MOTOR DRIVEN AUXILIARY FEEDWATER PUMP TO PREVENT UNNECESSARY STARTS. 3. AN INVESTIGATION TO DETERMINE AND CORRECT ROOT CAUSE OF THE VIBRATION OF THE TURBINE BYPASS VALVES HAS BEEN PLANNED.

[38] CALVERT CLIFFS 2 DOCKET 50-318 LER 87-008
MANUAL TRIP AS A RESULT OF TWO DROPPED CONTROL ELEMENT ASSEMBLIES.
EVENT DATE: 112287 REPORT DATE: 122287 NSSS: CE TYPE: PWR
VENDOR: COMBUSTION ENGINEERING, INC.

(NSIC 207440) DURING PERFORMANCE OF THE CEA PARTIAL MOVEMENT TEST (STP-0-29-2), CONTROL ELEMENT ASSEMBLY 29 (CEA-29), DROPPED FULLY INTO THE REACTOR CORE AND WOULD NOT RAISE. DURING THE REPLACEMENT OF THE MALFUNCTIONING UPPER GRIPPER POWER SWITCH MODULE FOR CEA-29, THE SAME MODULE FOR CEA-28 WAS REMOVED INADVERTENTLY DUE TO PERSONNEL ERROR. THIS REMOVAL CAUSED CEA-28 ALSO TO DROP FULLY INTO THE REACTOR CORE THE REACTOR WAS IMMEDIATELY MANUALLY TRIPPED DUE TO TWO DROPPED CEAS. CORRECTIVE ACTIONS ARE AS FOLLOWS: 1. THE ELECTRICIANS INVOLVED WERE COUNSELLED REGARDING THIS ERROR. THE DETAILS OF THIS EVENT HAVE BEEN REVIEWED WITH ALL MAINTENANCE ELECTRICIANS. 2. A STEP-BY-STEP CEA TROUBLESHOOTING PROCEDURE WILL BE WRITTEN. 3. A MECHANICAL INTERLOCK WILL BE ADDED TO CEA TEST COMPARTMENT DOOR. 4. THE CEA NUMBER WILL BE ADDED TO MODULE DESCRIPTION LABELING. 5. THE MALFUNCTIONING GRIPPER MODULE WAS REPLACED FOR CEA-29. THE FAILED MODULE WAS RETURNED TO THE MANUFACTURER TO DETERMINE THE ROOT CAUSE OF THE FAILURE.

[39] CATAWBA 1 DOCKET 50-413 LER 85-053 REV 01
UPDATE ON DIESEL GENERATOR 1A BATTERY CHARGER INOPERABLE DUE TO BLOWN FUSES.
EVENT DATE: 072985 REPORT DATE: 120187 NSSS: WE TYPE: PWR
VENDOR: CUTLER-HAMMER

POWER CONVERSION PRODUCTS, INC.

(NSIC 207287) DURING AN INVESTIGATION ON AUGUST 19, 1985, INTO THE CAUSE OF INDICATING LIGHT SOCKET SHORTINGS ON DIESEL GENERATOR 1A BATTERY CHARGER (1DGCA) AND DIESEL ENGINE (D/E) 1A CONTROL PANEL, IT WAS DISCOVERED THAT 1DGCA WAS INOPERABLE FROM 1447 TO 2030 HOURS ON JULY 29, 1985. DURING THIS TIME PERIOD, IT WAS NOT RECOGNIZED THAT THE CHARGER WAS INOPERABLE, AND THEREFORE, THE AVAILABILITY OF ALTERNATE POWER SOURCES WAS NOT VERIFIED AS REQUIRED BY TECHNICAL SPECIFICATION 3.8.1.1. UNIT 1 WAS IN MODE 1 IN THE PROCESS OF REACTOR POWER ESCALATION AT THE TIME OF THE INCIDENT. THIS INCIDENT HAS BEEN ASSIGNED THREE EVENT CAUSE CATEGORIES. AN EVENT CAUSE OF DESIGN DEFICIENCY HAS BEEN ASSIGNED AS THE MAIN CAUSE BECAUSE THE DESIGN OF THE CHARGER DID NOT ALLOW THE OPERATOR TO RECOGNIZE THAT THE CHARGER WAS INOPERABLE. AN EVENT CAUSE OF PERSONNEL ERROR HAS BEEN ASSIGNED BECAUSE PERSONNEL DID NOT PROPERLY TAKE ACTION TO CLEAR AN ALARM SIGNIFYING TROUBLE WITH THE CHARGER. AN EVENT CAUSE OF PROCEDURAL DEFICIENCY HAS BEEN ASSIGNED BECAUSE THE ANNUNCIATOR RESPONSE PROCEDURE DID NOT ADEQUATELY AID THE OPERATOR IN IDENTIFYING THE CAUSE OF THE CHARGER TROUBLE ALARM AND DID NOT DIRECT THE OPERATOR TO BEGIN VERIFICATION OF THE AVAILABILITY OF OFFSITE POWER SOURCES.

[40] CATAWBA 1 DOCKET 50-413 LER 87-(37
FAILURE OF ITT GRINNELL MIN-STIFF PIPE CLAMPS-FIGURE 214-DUE TO MANUFA TURING DEFICIENCY.
EVENT DATE: 090987 REPORT DATE: 121587 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: CATAWBA 2 (PWR)
VENDOR: GRINNELL INDUSTRIAL PIPING, INC.

(NSIC 207472) VISUAL INSPECTION OF SNUBBERS DURING THE FIRST REFUELING OUTAGE REVEALED TWO BROKEN PIPE SUPPORTS DUE TO THE FAILURE OF THE PIPE CLAMP STRAP ASSEMBLY ON EACH SUPPORT. SUBSEQUENT ANALYSIS OF THE FRACTURED ITT GRINNELL MINI-STAFF PIPE CLAMP - FIGURE 214 - STRAPS REVEALED THAT THE MATERIAL WAS NOT PROPERLY HEAT TREATED AND COULD BE SUSCEPTIBLE TO STRESS CORROSION CRACKING. HARDNESS MEASUREMENTS TAKEN ON STRAPS FROM STOCK REVEALED ADDITIONAL STRAPS WHICH WERE IMPROPERLY HEAT TREATED. APPROXIMATELY 1500 SIMILAR MINI-STIFF PIPE CLAMPS ARE INSTALLED AT CATAWBA. ON SEPTEMBER 9, 1987, DUKE POWER DETERMINED THAT OVER SEVERAL YEARS THE ABILITY OF AFFECTED HANGERS OR SNUBBERS TO PERFORM THEIR INTENDED FUNCTION COULD BE POTENTIALLY DEGRADED. DUKE POWER PERFORMED AN OPERABILITY ANALYSIS AND DETERMINED THAT PLANT OPERATION WAS UNAFFECTED BY THE FAILED PIPE SUPPORTS. BOTH UNITS HAVE BEEN IN ALL MODES OF OPERATION SINCE THE PIPE CLAMPS WERE INSTALLED. THIS EVENT WAS ATTRIBUTED TO A MANUFACTURING DEFICIENCY DUE TO THE IMPROPER HEAT TREATMENT APPLIED TO THE SUBJECT CLAMP STRAPS. THE DAMAGED CLAMPS WERE REPLACED. A STATISTICAL SAMPLING OF STRAPS PERFORMED DURING THE SECOND REFUELING OUTAGE INDICATES THERE IS 95% CONFIDENCE THAT 95% OF THE STRAPS ARE OPERABLE. DUKE POWER IS CONSIDERING A REPLACEMENT PROGRAM.

[41] CATAWBA 1 DOCKET 50-413 LER 87-038
MISSED HOURLY FIRE WATCHES RESULT IN TECHNICAL SPECIFICATION VIOLATION DUE TO A PERSONNEL ERROR.
EVENT DATE: 101287 REPORT DATE: 111187 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: CATAWBA 2 (PWR)

(NSIC 207113) ON OCTOBER 12, 1987, AT 1940 HOURS, A SECURITY OFFICER PERFORMING A FIRE WATCH PATROL VERIFICATION DISCOVERED THAT A TECHNICAL SPECIFICATION REQUIRED HOURLY FIRE WATCH HAD NOT BEEN PERFORMED FOR ROOM 428 (FLOOR DRAIN TANK PRIMARY DEMINERALIZER ROOM) SINCE 1715 HOURS. THE HOURLY FIRE WATCH REQUIREMENT WAS IN EFFECT DUE TO THE CONCRETE HATCH COVERS BEING REMOVED EARLIER IN THE DAY. THE CONCRETE HATCH COVERS SERVE AS RADIATION SHIELDING AND FIRE BOUNDARIES BETWEEN ROOM 427 (MECHANICAL PENETRATION ROOM) AND ROOM 428. FOLLOWING THE DISCOVERY,

THE REQUIRED HOURLY FIRE WATCHES WERE PERFORMED UNTIL THE HATCH COVERS WERE REINSTALLED AT 2040 HOURS. AT THE TIME OF THIS EVENT, UNIT 1 WAS IN MODE 5, COLD SHUTDOWN, AND UNIT 2 WAS IN MODE 1, POWER OPERATION, AT 95% POWER. THIS INCIDENT IS ATTRIBUTED TO A PERSONNEL ERROR. THE RADWASTE CHEMISTRY (RDW) TECHNICIAN WHO HAD ASSUMED THE HOURLY FIRE WATCH RESPONSIBILITY, FAILED TO EITHER ENSURE THAT THE HATCH COVERS WERE REINSTALLED OR TO CONTINUE THE HOURLY FIRE WATCH AS REQUIRED. THIS INCIDENT HAS BEEN REVIEWED WITH THE RDW TECHNICIAN INVOLVED. THE IMPORTANCE OF EFFECTIVE COMMUNICATIONS TO ACCOMPLISH REQUIRED TASKS WAS STRESSED. APPROPRIATE MANAGEMENT PERSONNEL WILL BE ISSUED A TECHNICAL SPECIFICATION ALERT REEMPHASIZING THE IMPORTANCE OF PERFORMING REQUIRED FIRE WATCHES.

[42] CATAWBA 1 DOCKET 50-413 LER 87-039
AUXILIARY FEEDWATER AUTO-START WHILE IMPLEMENTING TWO NUCLEAR STATION MODIFICATIONS SIMULTANEOUSLY DURING A REFUELING.
EVENT DATE: 110287 REPORT DATE: 120287 NSSS: WE TYPE: PWR

(NSIC 207307) ON NOVEMBER 2, 1987, AT APPROXIMATELY 0950 HOURS, MOTOR DRIVEN AUXILIARY FEEDWATER (CA) PUMP 1B STARTED AUTOMATICALLY DUE TO A "LOSS OF BOTH MAIN FEEDWATER (CF) PUMPS" SIGNAL WHILE IMPLEMENTING A NUCLEAR STATION MODIFICATION (NSM) TO REPLACE ALL OF THE OIL PRESSURE SWITCHES ON CF PUMP 1B. BOTH CF PUMPS HAD BEEN SHUT DOWN PREVIOUSLY FOR THE UNIT 1 REFUELING OUTAGE, HOWEVER THE "LOSS OF BOTH CF PUMPS" SIGNAL WAS NOT PRESENT DUE TO THE OIL PRESSURE SWITCHES HAVING BEEN TEMPORARILY REMOVED FROM THE CIRCUIT. APPROXIMATELY 30 SECONDS AFTER THE LOSS OF BOTH CF PUMPS SIGNAL OCCURRED, SOLID STATE PROTECTION SYSTEM (SSPS) TRAIN B WAS PLACED IN TEST TO SUPPORT IMPLEMENTATION OF ANOTHER NSM. THIS CAUSED THE TRAIN B CA AUTO-START DEFEAT TO BE REMOVED AND ALLOWED THE CA PUMP TO START. THE UNIT WAS DEFUELED AT THE TIME OF THIS INCIDENT. THIS INCIDENT HAS BEEN ATTRIBUTED TO THE SIMULTANEOUS IMPLEMENTATION OF 2 NSMS. THE ACTIVITIES AND PROCEDURES IN PROGRESS DURING THIS INCIDENT ARE NOT CONSIDERED DEFICIENT, ALTHOUGH A BETTER PROCEDURAL EXPLANATION OF EXPECTED PLANT RESPONSE WOULD HAVE PRECLUDED REPORTING THIS ACTUATION. SIMULTANEOUS IMPLEMENTATION OF NSMS IS CONSIDERED NECESSARY TO COMPLETE REQUIRED OUTAGE ACTIVITIES ON A TIMELY MANNER.

[43] CATAWBA 1 DOCKET 50-413 LER 87-040
TURBINE DRIVEN AUXILIARY FEEDWATER PUMP AUTO-START SIGNAL/STEAM GENERATOR BLOWDOWN CONTAINMENT ISOLATION VALVES ACTUATION DUE TO PROCEDURAL DEFICIENCY.
EVENT DATE: 110587 REPORT DATE: 120487 NSSS: WE TYPE: PWR

(NSIC 207378) ON NOVEMBER 5, 1987, AT APPROXIMATELY 1135 HOURS, A STEAM SUPPLY VALVE TO THE TURBINE DRIVEN AUXILIARY FEEDWATER (CA) PUMP AUTOMATICALLY OPENED AND THE STEAM GENERATOR (S/G) BLOWDOWN CONTAINMENT ISOLATION VALVES AUTOMATICALLY CLOSED DUE TO LOW LOW LEVELS IN TWO S/GS. THE UNIT WAS SHUTDOWN IN MODE 6, REFUELING, AT THE TIME OF THIS INCIDENT. S/G 1A HAD BEEN DRAINED ON NOVEMBER 4, 1987, FOR MAINTENANCE, AND DUKE POWER PERSONNEL WERE IN THE PROCESS OF DRAINING S/G 1D. ON NOVEMBER 5, DUKE POWER TECHNICIANS PLACED SOLID STATE PROTECTION SYSTEM (SSPS) TRAIN A INTO TEST TO PERFORM A FUNCTIONAL VERIFICATION FOLLOWING CORRECTIVE MAINTENANCE. WHEN THE TECHNICIANS PLACED SSPS TRAIN A BACK INTO THE NORMAL MODE, THE LOW LOW LEVEL SIGNALS ALARMED FOR S/GS 1A AND 1D. THIS RESULTED IN A STEAM SUPPLY VALVE FOR THE TURBINE DRIVEN CA PUMP OPENING AND THE S/G BLOWDOWN CONTAINMENT ISOLATION VALVES CLOSING UNEXPECTEDLY. CONTROL ROOM OPERATORS REALIGNED THE AFFECTED VALVES TO THEIR PREVIOUS POSITIONS. THIS INCIDENT HAS BEEN ATTRIBUTED TO DEFECTIVE PROCEDURES. PROCEDURES FOR SSPS PERIODIC TESTING DID NOT PROVIDE SUFFICIENT PRECAUTIONS FOR PERFORMING THIS TEST DURING A REFUELING OUTAGE WITH TWO S/GS DRAINED. THE APPROPRIATE PROCEDURES HAVE BEEN REVISED TO INCLUDE THE REQUIRED ACTIONS TO BE TAKEN WHEN REMOVING AN SSPS TRAIN FROM TEST.

[44] CATAWBA 2 DOCKET 50-414 LER 86-045 REV 02
UPDATE ON TURBINE DRIVEN AUXILIARY FEEDWATER PUMP INOPERABLE FROM STANDBY SHUTDOWN FACILITY DUE TO DESIGN DEFICIENCY.
EVENT DATE: 101186 REPORT DATE: 112587 NSSS: WE TYPE: PWR

(NSIC 207289) ON OCTOBER 11, 1986, THE 18 MONTH TESTING OF TURBINE DRIVEN AUXILIARY FEEDWATER (CA) PUMP (PT) MANUAL START CONTROL FROM THE STANDBY SHUTDOWN FACILITY (SSF) WAS PERFORMED. THE CAPT DID NOT START BECAUSE THE VALVE CONTROLLING STEAM FLOW TO THE CAPT DID NOT OPEN. INVESTIGATION REVEALED THAT THE SOLENOID VALVE WHICH OPENS THE CONTROL VALVE WAS TUBED INCORRECTLY, ALTHOUGH IT WAS TUBED PER THE DESIGN DRAWING. BECAUSE IT CANNOT BE DETERMINED WHEN THIS VALVE WAS TUBED INCORRECTLY, IT IS ASSUMED THAT TECHNICAL SPECIFICATIONS WERE VIOLATED BETWEEN APRIL 23, 1986, AND SEPTEMBER 1, 1986. DURING THAT TIME, THE STANDBY SHUTDOWN SYSTEM WAS UNKNOWINGLY INOPERABLE WITH THE UNIT IN MODES OF OPERATIONS WHICH REQUIRED THE SYSTEM TO BE OPERABLE. THE UNIT WAS IN MODE 5, COLD SHUTDOWN, WHEN THIS INCIDENT WAS DISCOVERED, BUT HAD APPARENTLY BEEN IN ALL MODES OF OPERATION WITH CAPT CONTROL FROM THE SSF INOPERABLE. THIS INCIDENT IS ASSIGNED CAUSE CODE B, DESIGN, MANUFACTURING, CONSTRUCTION/INSTALLATION DEFICIENCY. THE SOLENOID VALVE WAS FOUND INSTALLED INCORRECTLY PER AN INCORRECT DESIGN DRAWING. THE SOLENOID VALVE WAS RETUBED AND THE ASSOCIATED INSTRUMENT DETAIL DRAWING WAS CORRECTED. A REVIEW PERFORMED BY THE DUKE POWER DESIGN ENGINEERING DEPARTMENT IDENTIFIED DOCUMENTS WHICH HAD INADEQUATE SOLENOID VALVE INFORMATION.

[45] CATAWBA 2 DOCKET 50-414 LER 87-007 REV 01
UPDATE ON REACTOR TRIP DUE TO A MANAGEMENT DEFICIENCY AND EQUIPMENT FAILURES.
EVENT DATE: 022487 REPORT DATE: 120187 NSSS: WE TYPE: PWR
VENDOR: ELECTROMAX INSTRUMENTS, INC.
WESTINGHOUSE ELECTRIC CORP.

(NSIC 207308) ON FEBRUARY 24, 1987, AT 1359:17 HOURS, A POWER PANELBOARD 2EPD UNDERVOLTAGE COMPUTER ALARM WAS RECEIVED AT THE CONTROL ROOM. DURING THE INVESTIGATION OF THE SITUATION, A BREAKER WAS DISCOVERED TO BE LOOSE WITH THE APPEARANCE OF BEING TRIPPED. THE BREAKER WAS RESET AND RESULTED IN AN ACTUAL UNDERVOLTAGE CONDITION TO 2EPD. THIS UNDERVOLTAGE RESULTED IN MAIN FEEDWATER ISOLATION AND BOTH MAIN FEEDWATER PUMPS TRIPPED ON HIGH DISCHARGE PRESSURE. THE TURBINE TRIPPED ON LOSS OF BOTH MAIN FEEDWATER PUMPS, AND A REACTOR TRIP FOLLOWED THE TURBINE TRIP. THE UNIT WAS AT 100% POWER AT THE TIME OF THIS INCIDENT. THE INCIDENT HAS BEEN ATTRIBUTED TO A MANAGEMENT DEFICIENCY DUE TO THE DECISION TO CYCLE THE BREAKER BEING MADE WITHOUT SUFFICIENTLY INVESTIGATING THE SITUATION. THE FAILURE OF AN OPTICAL ISOLATOR AND THE FAILURE OF THE FEEDER BREAKER TO INDICATE CORRECTLY ALSO CONTRIBUTED TO THIS EVENT BY GIVING THE FALSE INDICATION OF AN UNDERVOLTAGE OF 2EPD AND BY MISLEADING THE RESPONSIBLE PERSONNEL INTO BELIEVING THAT THE BREAKER HAD TRIPPED, RESPECTIVELY. THE HEALTH AND SAFETY OF THE PUBLIC WERE UNAFFECTED BY THIS EVENT.

[46] CATAWBA 2 DOCKET 50-414 LER 87-009 REV 03
UPDATE ON CONTAINMENT AIR RETURN ISOLATION DAMPERS ACTUATED DUE TO DEFECTIVE PROCEDURE.
EVENT DATE: 031287 REPORT DATE: 112487 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: CATAWBA 1 (PWR)

(NSIC 207331) ON MARCH 12, 1987, AT APPROXIMATELY 1800 HOURS, CONTROL ROOM PERSONNEL DISCOVERED THAT THE UNIT 2 CONTAINMENT AIR RETURN ISOLATION DAMPERS WERE OPEN, WHICH CONSTITUTED AN ENGINEERED SAFEGUARD FEATURE ACTUATION. AN INVESTIGATION REVEALED THAT THE ISOLATION DAMPERS HAD BEEN OPEN FOR APPROXIMATELY 8 HOURS FOLLOWING TESTING OF THE CONTAINMENT PRESSURE CONTROL SYSTEM. THE UNIT WAS AT 98% POWER DURING THIS INCIDENT. THIS INCIDENT IS CLASSIFIED AS EVENT CAUSE CODE D, DEFECTIVE PROCEDURE. THE AUXILIARY SAFEGUARDS TEST CABINET

PERIODIC TEST PROCEDURE DID NOT CONTAIN INSTRUCTIONS TO RESET THE ACTUATION SIGNAL FOR THE ISOLATION DAMPERS. THIS ALLOWED THE ISOLATION DAMPERS TO OPEN WHEN THE REQUIRED LOGIC WAS COMPLETED DURING THE PERFORMANCE OF TWO PROCEDURES. CONTROL ROOM PERSONNEL CLOSED THE ISOLATION DAMPERS UPON DISCOVERY. ALSO, A PROCEDURE CHANGE WAS INCORPORATED INTO THE AUXILIARY SAFEGUARDS TEST CABINET PERIODIC TEST PROCEDURE TO RESET THE ACTUATION SIGNAL FOR THE ISOLATION DAMPERS. ANALYSIS OF THE EVENT HAS DETERMINED THAT THE PLANT WAS ALWAYS WITHIN ITS DESIGN BASIS AND THAT THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED BY THIS INCIDENT.

[47] CATAWBA 2 DOCKET 50-414 LER 87-029
MANUAL REACTOR/AUXILIARY FEEDWATER PUMP TURBINE TRIP FOLLOWING TURBINE TRIP/MAIN FEEDWATER ISOLATION DUE TO A SPURIOUS HIGH STEAM GENERATOR LEVEL SIGNAL.
EVENT DATE: 110387 REPORT DATE: 120387 NSSS: WE TYPE: PWR
VENDOR: DRAGON VALVE, INC.
TERRY STEAM TURBINE COMPANY
WOODWARD GOVERNOR COMPANY

(NSIC 207393) ON NOVEMBER 3, 1987, AT 0836 HOURS, STEAM GENERATOR (S/G) 2C NARROW RANGE (N/R) LEVEL CHANNEL 2 INOPERABLE DUE TO ITS READING BEING OUT OF TOLERANCE. THE UNIT WAS AT 93% POWER. SINCE S/G 2C N/R LEVEL CHANNEL 4 WAS ALSO INOPERABLE, A TECHNICAL SPECIFICATION REQUIRED UNIT SHUTDOWN WAS COMMENCED AT 0926 HOURS. THE CHANNEL 2 READING BECAME INCREASINGLY ERRATIC AND AT APPROXIMATELY 1146 HOURS IT CAUSED A SPURIOUS S/G HIGH HIGH LEVEL TURBINE TRIP, A MAIN FEEDWATER (CF) ISOLATION, AND A CF PUMP TURBINE TRIP. MOTOR DRIVEN (M/D) AUXILIARY FEEDWATER (CA) PUMP 2B STARTED AUTOMATICALLY TO SUPPLY FEEDWATER TO THE S/GS (M/D CA PUMP 2A WAS ALREADY OPERATING). AT 1146:39 HOURS, THE REACTOR WAS MANUALLY TRIPPED FROM 59% POWER. APPROXIMATELY 11 SECONDS LATER, ALL FOUR S/G LEVELS DECREASED BELOW THE LOW LOW LEVEL REACTOR TRIP SETPOINT. THE TURBINE DRIVEN (T/D) CA PUMP STARTED AUTOMATICALLY, HOWEVER, IT TRIPPED WITHIN SECONDS. THE M/D CA PUMPS WERE UTILIZED TO RESTORE S/G LEVELS TO NORMAL AND TO COOL DOWN THE UNIT. THE UNIT ENTERED MODE 4, HOT SHUTDOWN, AT 2013 HOURS. THIS INCIDENT IS ATTRIBUTED TO THE FAILURE OF THE S/G 2C N/R LEVEL CHANNELS DUE TO THEIR REFERENCE LEG VENT LINE ISOLATION VALVES LEAKING STEAM PAST THE SEAT. THE REFERENCE LEG VENT LINE STEAM LEAKS WERE STOPPED, AND THE LEAKING VENT LINE ISOLATION VALVES WILL BE REPAIRED.

[48] CATAWBA 2 DOCKET 50-414 LER 87-030
FEEDWATER ISOLATION CAUSED BY A HI HI STEAM GENERATOR LEVEL SIGNAL DUE TO A PERSONNEL ERROR AND INSTALLATION DEFICIENCY.
EVENT DATE: 110587 REPORT DATE: 120487 NSSS: WE TYPE: PWR
VENDOR: KEROTEST MANUFACTURING CORP.

(NSIC 207394) ON NOVEMBER 5, 1987, AT 0214 HOURS, A FEEDWATER ISOLATION SIGNAL WAS GENERATED ON A HI HI STEAM GENERATOR (S/G) LEVEL WHILE DUKE POWER PERSONNEL WERE TROUBLESHOOTING AN INOPERABLE NARROW RANGE LEVEL TRANSMITTER. THE UNIT WAS IN MODE 4, HOT SHUTDOWN, AT THE TIME OF THE INCIDENT. S/G LEVELS WERE BEING MAINTAINED USING THE MOTOR DRIVEN AUXILIARY FEEDWATER (CA) PUMPS. THE CA AUTOSTART FUNCTION WAS DEFEATED AT THE TIME DUE TO THE STATUS OF THE UNIT. THIS INCIDENT IS ATTRIBUTED TO A PERSONNEL ERROR DUE TO THE FAILURE OF A DUKE POWER TECHNICIAN TO CORRECTLY IDENTIFY THE ROOT VALVE ASSOCIATED WITH AN INOPERABLE LEVEL TRANSMITTER. AN INSTALLATION DEFICIENCY ALSO CONTRIBUTED TO THIS INCIDENT BECAUSE THE ROOT VALVE INCORRECTLY IDENTIFIED BY THE TECHNICIAN DID NOT CONTAIN ALL THE IDENTIFICATION TAGS SPECIFIED IN THE APPROPRIATE INSTALLATION PROCEDURES. DUKE POWER TECHNICIANS REOPENED THE ROOT VALVE WHICH HAD BEEN MISTAKENLY CLOSED CLEARING THE S/G HI HI LEVEL ALARM. THIS INCIDENT AND THE IMPORTANCE OF ATTENTION TO DETAIL WERE DISCUSSED WITH THE INVOLVED PERSONNEL. DUKE POWER IS EVALUATING THE NEED OF IMPROVED LABELING OF S/G INSTRUMENTATION IN ORDER TO

PREVENT SIMILAR EVENTS. THE HEALTH AND SAFETY OF THE PUBLIC WERE UNAFFECTED BY THIS EVENT.

[49] CLINTON 1 DOCKET 50-461 LER 87-061
CHANNEL CHECK OF CONTAINMENT PRESSURE-HIGH TRIP FUNCTION FOR CONTAINMENT SPRAY MISSED DUE TO INCORRECT REQUIREMENT REMOVAL FROM PROCEDURE BY UTILITY LICENSED OPERATOR.
EVENT DATE: 101487 REPORT DATE: 110987 NSSS: GE TYPE: BWR

(NSIC 206967) ON OCTOBER 14, 1987 AT 1607 HOURS, WITH THE PLANT IN MODE 1 (POWER OPERATION) AT 95% REACTOR POWER, REVIEW OF A PROCEDURE CHANGE IDENTIFIED THAT THE REQUIREMENT FOR CHANNEL CHECKS OF THE CONTAINMENT PRESSURE-HIGH ACTUATION TRIP FUNCTION FOR THE CONTAINMENT SPRAY SYSTEM WAS INCORRECTLY DELETED FROM THE CONTROL ROOM OPERATOR SURVEILLANCE LOG MODE 1, 2, 3 DATA SHEET. A UTILITY LICENSED OPERATOR REVISED THE DATA SHEET BY A TEMPORARY CHANGE METHOD AND REMOVED A CHANNEL CHECK FOR INSTRUMENT NUMBERS THAT HE THOUGHT WERE NOT APPLICABLE. THIS REVISION RESULTED IN VIOLATION OF THE PLANT'S TECHNICAL SPECIFICATION FROM APPROXIMATELY 2300 HOURS ON OCTOBER 13 TO 1607 HOURS ON OCTOBER 14 SINCE THE REQUIRED CHANNEL CHECKS WERE NOT PERFORMED. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO UTILITY LICENSED OPERATOR ERROR DUE TO REMOVAL OF THE TECHNICAL SPECIFICATION REQUIREMENT FROM THE DATA SHEET. THE MODE 1, 2, 3 DATA SHEET WAS REVISED TO RESTORE THE CHANNEL CHECK REQUIREMENT, AND THE SUBSEQUENT CHANNEL CHECKS WERE SATISFACTORILY PERFORMED. MANAGEMENT PERSONNEL WILL BE TRAINED ON THE IMPORTANCE OF PERFORMING A THOROUGH MANAGEMENT REVIEW OF ALL REVISIONS TO PROCEDURES BY THE TEMPORARY CHANGE METHOD. THIS EVENT WAS ASSESSED AS NOT SAFETY SIGNIFICANT SINCE THE SUBSEQUENT CHANNEL CHECKS WERE SATISFACTORILY PERFORMED.

[50] CLINTON 1 DOCKET 50-461 LER 87-062
ANOMALIES IN SEATING SURFACES OF MAIN STEAM ISOLATION VALVES RESULTING IN UNACCEPTABLE LEAKAGE RATES.
EVENT DATE: 102687 REPORT DATE: 112487 NSSS: GE TYPE: BWR
VENDOR: ATWOOD & MORRILL CO., INC.

(NSIC 207223) ON OCTOBER 26, 1987, AT 1145 HOURS, WITH THE PLANT IN MODE 4 (COLD SHUTDOWN) AND THE REACTOR AT APPROXIMATELY 125 DEGREES FAHRENHEIT AND ATMOSPHERIC PRESSURE, LOCAL LEAK RATE TESTING BY TEST ENGINEERS IDENTIFIED THAT THE PRIMARY CONTAINMENT LEAKAGE RATES OF THE MAIN STEAM ISOLATION VALVES (MSIV'S) ON LINES A, B, AND C EXCEEDED TECHNICAL SPECIFICATION LIMITS OF 28 STANDARD CUBIC FEET PER HOUR (13,214 STANDARD CUBIC CENTIMETERS PER MINUTE (SCCM)) PER LINE. THE CAUSE OF THE EXCESSIVE LEAKAGE HAS BEEN ATTRIBUTED TO ANOMALIES IN THE SEATING SURFACES OF THE INBOARD AND OUTBOARD MSIV'S DUE TO COMPONENT WEAR BASED ON SERVICE SEEN DURING THE POWER ASCENSION PROGRAM. ALL SIX MSIV'S HAVE BEEN REWORKED BY LAPPING THE SEATS AND MACHINING THE POPPETS. SUBSEQUENT LEAK RATE TESTING WAS SATISFACTORILY COMPLETED. THE APPROXIMATE LEAKAGE RATES FOR THE MAIN STEAM LINES FOLLOWING REWORK WERE AS FOLLOWS: 3020 SCCM FOR THE A LINE, 420 SCCM FOR THE B LINE, AND 750 SCCM FOR THE C LINE. ANALYSIS OF THE SAFETY CONSEQUENCES AND IMPLICATIONS INDICATES THAT THIS EVENT IS CONSIDERED TO BE OF POTENTIAL SAFETY SIGNIFICANCE. THE MSIV LEAKAGE RATES EXCEEDED THE ASSUMPTIONS OF CHAPTER 15 OF THE FINAL SAFETY ANALYSIS REPORT. THIS EVENT IS REPORTABLE UNDER THE PROVISIONS OF 10CFR50.73(A)(2)(II).

[51] CLINTON 1 DOCKET 50-461 LER 87-064
INADEQUATE PROCEDURE AND INADEQUATE ELECTRICAL TECHNICIANS IMPACT MATRIX FOR UNDERVOLTAGE RELAY REMOVAL RESULTING IN DIVISION 3 DIESEL GENERATOR AUTO-START.
EVENT DATE: 110287 REPORT DATE: 112487 NSSS: GE TYPE: BWR

(NSIC 207224) ON NOVEMBER 2, 1987 AT APPROXIMATELY 0958 HOURS, WITH THE PLANT IN MODE 4 (COLD SHUTDOWN) AND THE REACTOR AT APPROXIMATELY 140 DEGREES FAHRENHEIT

AND ATMOSPHERIC PRESSURE, THE DIVISION 3 DIESEL GENERATOR (DG) AUTOMATICALLY STARTED. THE EVENT OCCURRED WHILE ELECTRICAL TECHNICIANS WERE DISCONNECTING AN UNDERVOLTAGE RELAY FROM CIRCUITRY DURING PERFORMANCE OF THE DIVISION 3 4160 VOLT UNDERVOLTAGE PROTECTING RELAY CALIBRATION. REMOVAL OF THE RELAY CAUSED THE 4160 VOLT BUS MAIN BREAKER TO TRIP OPEN AND THE DIVISION 3 DG TO AUTO- START. THE CAUSE IS ATTRIBUTED TO AN INADEQUATE PROCEDURE AND TO PERSONNEL ERROR RESULTING FROM AN INADEQUATE SURVEILLANCE IMPACT MATRIX. IN RESPONSE TO THIS EVENT, THE PROCEDURE WAS REVISED TO PROVIDE SUFFICIENT DETAIL FOR DEENERGIZING AND REMOVING THE RELAY, A REVIEW OF INFREQUENTLY PERFORMED ELECTRICAL MAINTENANCE PROCEDURES FOR SIMILAR PROBLEMS IS BEING CONDUCTED, AND ELECTRICAL TECHNICIANS ARE BEING TRAINED ON PROPER PREPARATION OF IMPACT MATRICES. THE EVENT IS ASSESSED AS NOT SAFETY SIGNIFICANT SINCE THE SYSTEM RESPONDED AS DESIGNED. THIS EVENT IS REPORTABLE UNDER THE PROVISIONS OF 10CFR50.73(A)(2)(IV) DUE TO AN AUTOMATIC ACTUATION OF AN ENGINEERED SAFETY FEATURE.

[52] CLINTON 1 DOCKET 50-461 LER 87-065
REACTOR TRIP FROM UNEXPECTED LOSS OF INSTRUMENT AIR DURING POWER SUPPLY TRANSFER SURVEILLANCE DUE TO INADEQUATE PROCEDURE AND UNTIMELY OPERATOR RESPONSE.
EVENT DATE: 110487 REPORT DATE: 120387 NSSS: GE TYPE: BWR

(NSIC 207384) ON NOVEMBER 4, 1987 AT APPROXIMATELY 2045 HOURS, WITH THE PLANT IN MODE 4 (COLD SHUTDOWN), AN AUTOMATIC ACTUATION OF THE REACTOR PROTECTION SYSTEM (RPS) OCCURRED. THE EVENT OCCURRED DURING PERFORMANCE OF A DIVISION 2 SURVEILLANCE WHICH MOMENTARILY DEENERGIZES THE DIVISION 2 BUS. WHEN THE BUS DEENERGIZED, LOGIC POWER TO INSTRUMENT AIR (IA) AND SERVICE AIR (SA) ISOLATION VALVES WAS LOST, AND THE VALVES UNEXPECTEDLY CLOSED RESULTING IN DECAY OF CONTAINMENT AIR HEADER PRESSURE, SCRAM VALVES DRIFTING OPEN, AND SCRAM DISCHARGE VOLUME (SDV) ISOLATING. THE SDV BEGAN FILLING, INITIATING A SCRAM SIGNAL ON HIGH LEVEL. THE CAUSE OF THE EVENT HAS BEEN ATTRIBUTED TO AN INADEQUATE SURVEILLANCE PROCEDURE AND UTILITY-LICENSED OPERATOR ERROR. THE PROCEDURE WILL BE REVISED TO ADD A CAUTION FOR THE IA/SA ISOLATION VALVE RESPONSE TO THE INTERRUPTED BUS POWER. OTHER SURVEILLANCE PROCEDURES THAT INTERRUPT BUS POWER WILL BE REVIEWED AND REVISED TO ADD CAUTIONS AS NECESSARY. THE OPERATORS WERE COUNSELLED FOR THE FAILURE TO RESPOND TO PLANT ALARMS IN A TIMELY MANNER. THE EVENT WAS DETERMINED TO BE NOT SAFETY SIGNIFICANT SINCE THE PLANT RESPONDED AS DESIGNED TO THE SDV HIGH LEVEL SIGNAL. THIS EVENT IS REPORTABLE UNDER THE PROVISIONS OF 10CFR50.73(A)(2)(IV).

[53] CLINTON 1 DOCKET 50-461 LER 87-066
FAILURE OF JUNCTION BOXES TO MEET ENVIRONMENTAL QUALIFICATIONS DUE TO CONSTRUCTION CONTRACTOR FAILURE TO INSTALL DRAINAGE OPENINGS.
EVENT DATE: 110587 REPORT DATE: 120487 NSSS: GE TYPE: BWR

(NSIC 207385) ON NOVEMBER 5, 1987 AT 1500 HOURS, WITH THE PLANT IN MODE 4 (COLD SHUTDOWN), A PLANT INSPECTION OF ELECTRICAL JUNCTION BOXES AND PANELS IN THE AUXILIARY, FUEL, AND CONTAINMENT BUILDINGS IDENTIFIED THAT OPENINGS TO ALLOW THE EQUIPMENT TO BREATHE AND TO DRAIN ACCUMULATED WATER WERE NOT INSTALLED IN 156 JUNCTION BOXES CONTAINING TERMINAL BLOCKS AS REQUIRED BY THE INSTALLATION SPECIFICATION. LACK OF THESE OPENINGS RESULTED IN CLASS 1E EQUIPMENT IN HARSH ENVIRONMENTS NOT MEETING ENVIRONMENTAL QUALIFICATION REQUIREMENTS AND PRESENTED A POTENTIAL IMPACT ON SAFETY RELATED EQUIPMENT. THE CAUSE OF THE EVENT IS ATTRIBUTED TO FAILURE OF THE CLINTON POWER STATION CONSTRUCTION CONTRACTOR TO IDENTIFY THE DESIGN REQUIREMENT APPLICABILITY AND TO INSTALL THE OPENINGS, AND AN UNCLEAR INSTALLATION SPECIFICATION. DRAINAGE OPENINGS HAVE BEEN ADDED TO THE 156 JUNCTION BOXES. THE INSTALLATION SPECIFICATION WILL BE REVIEWED FOR SIMILAR PROBLEMS. ASSOCIATED DRAWINGS WILL BE REVISED TO SPECIFY THE DRAINAGE OPENING REQUIREMENT. ASSESSMENT OF THE SAFETY CONSEQUENCES INDICATES THAT FAILURE OF THE TERMINALS DUE TO MOISTURE BUILDUP COULD FAIL INSTRUMENTATION NECESSARY TO SHUTDOWN THE REACTOR AND MAINTAIN IT IN A SAFE SHUTDOWN CONDITION.

[54] CONNECTICUT YANKEE DOCKET 50-213 LER 87-011 REV 01
UPDATE ON CONTAINMENT PENETRATION FAILS "AS FOUND" LOCAL LEAK RATE TEST.
EVENT DATE: 072187 REPORT DATE: 120187 NSSS: WE TYPE: PWR
VENDOR: CHAPMAN VALVE & MFG

(NSIC 207249) CONTAINMENT PENETRATION LOCAL LEAK RATE TESTING (TYPE B AND C) WAS CONDUCTED DURING THE REFUELING OUTAGE IN ACCORDANCE WITH 10CFR50 APPENDIX J AND TECHNICAL SPECIFICATION 4.4.II. ON JULY 21, 1987, WITH THE PLANT SHUTDOWN IN MODE 5, CONTAINMENT PENETRATION P-30 (SPACE HEATING STEAM SUPPLY, HS-CV-295 AND HS-CV-295A) FAILED ITS TYPE C LOCAL LEAK RATE TEST. THE TEST METHOD WAS QUESTIONED BY SITE ENGINEERING AND A RETEST WAS PERFORMED. THE PENETRATION FAILED THE RETEST, ALTHOUGH ONE OF ITS TWO SERIES INSTALLED CHECK VALVES (HS-CV-295) DID PASS. THE CAUSE OF THE FAILURE WAS ATTRIBUTED TO IMPROPER SEATING AND VALVE MISAPPLICATION. THE PENETRATION WAS SUBSEQUENTLY MODIFIED WHICH ELIMINATED HS-CV-295 AND HS-CV-295A AS CONTAINMENT ISOLATION VALVES. THIS EVENT IS REPORTABLE PER 10CFR50.73 (A)(2)(I) SINCE IT INVOLVES A CONDITION PROHIBITED BY TECHNICAL SPECIFICATIONS. THIS REVISION HAS BEEN ISSUED TO PROVIDE ADDITIONAL INFORMATION ON LONG-TERM CORRECTIVE ACTIONS TAKEN.

[55] CONNECTICUT YANKEE DOCKET 50-213 LER 87-017
INOPERABLE FIRE PROTECTION SYSTEM DUE TO PERSONNEL ERROR.
EVENT DATE: 102387 REPORT DATE: 111987 NSSS: WE TYPE: PWR

(NSIC 207162) AT 1700 HOURS ON OCTOBER 23, 1987, WITH THE PLANT SHUTDOWN IN MODE 6, A CONTINUOUS FIRE WATCH LEFT HIS POST BEFORE THE SERVICE WATER PUMP WATER CURTAIN SYSTEM WAS RETURNED TO SERVICE. SINCE THIS SYSTEM IS REQUIRED TO BE OPERABLE, OR A CONTINUOUS FIRE WATCH POSTED, THIS EVENT IS A VIOLATION OF TECHNICAL SPECIFICATION 3.22. THE FIRE WATCH NOTIFIED MAINTENANCE SUPERVISION THAT WORK ON THE SYSTEM WAS COMPLETE AND IT COULD BE RETURNED TO SERVICE. THE EVENT WAS DISCOVERED AT 1830 HOURS (AFTER ABOUT 1-1/2 HOURS) BY THE PLANT OPERATOR WHO WAS DISPATCHED TO RETURN THE SYSTEM TO SERVICE. THE ROOT CAUSE OF THIS EVENT WAS INADEQUATE PERSONNEL FAMILIARITY WITH THE STATION FIRE PROTECTION PROGRAM REQUIREMENTS. CORRECTIVE ACTIONS INCLUDE REVIEW OF THE INCIDENT FOR LESSONS LEARNED AND UPGRADES TO THE STATION TRAINING PROGRAM. THIS EVENT IS REPORTABLE PER 10CFR50.73(A)(2)(I) SINCE IT INVOLVES A DEVIATION FROM TECHNICAL SPECIFICATIONS.

[56] COOPER DOCKET 50-298 LER 87-024 REV 01
UPDATE ON FAILURE OF HPCI TURBINE OVERSPEED TRIP MECHANISM TO AUTOMATICALLY RESET DURING SURVEILLANCE TESTING DUE TO BINDING OF THE TAPPET ASSEMBLY.
EVENT DATE: 110687 REPORT DATE: 120887 NSSS: GE TYPE: BWR
VENDOR: TERRY STEAM TURBINE COMPANY

(NSIC 207389) ON NOVEMBER 6, 1987, AT 2:15 AM, DURING PERFORMANCE OF A SURVEILLANCE TEST ON THE PRESSURE COOLANT INJECTION (HPCI) TURBINE MECHANICAL/HYDRAULIC OVERSPEED TRIP DEVICE, THE TRIP TEST HANDLE DID NOT AUTOMATICALLY RETURN TO ITS RESET POSITION AFTER BEING LIFTED AND RELEASED. MAINTENANCE PERSONNEL ASSIGNED TO INVESTIGATE AND CORRECT THE CAUSE OF THE PROBLEM DETERMINED THAT THE OVERSPEED TRIP DEVICE TAPPET ASSEMBLY HEAD, MANUFACTURED OF POLYURETHANE, WAS BINDING IN THE VALVE BODY. AT THE TIME OF DISCOVERY, THE PLANT WAS IN OPERATION AT APPROXIMATELY 75 PERCENT POWER. AS A RESULT OF A SIMILAR PROBLEM EXPERIENCED AT ANOTHER PLANT, THIS SITUATION HAD BEEN THE SUBJECT OF CORRESPONDENCE FROM GENERAL ELECTRIC (GE) IN 1986 AND 1987. RECOMMENDATIONS CONTAINED THEREIN REGARDING MACHINING THE TAPPET HEAD TO AN O.D. OF 0.738 - 0.740 INCH, HAD PREVIOUSLY BEEN IMPLEMENTED. HOWEVER, A RE-CHECK OF THIS DIMENSION CONDUCTED AS A RESULT OF THIS EVENT, REVEALED ADDITIONAL GROWTH OF APPROXIMATELY 0.010 INCH SINCE IMPLEMENTATION. THE TAPPET ASSEMBLY HEAD, PER INSTRUCTIONS FROM DRESSER RAND, WAS RE-MACHINED TO AN O.D OF 0.739 INCH AND THE OVERSPEED TRIP DEVICE WAS RE-ASSEMBLED AND RE-INSTALLED ON THE TURBINE.

[57] CRYSTAL RIVER 3 DOCKET 50-302 LER 86-001 REV 03
UPDATE ON FAILURE OF RCP "A" RESULTS IN REACTOR TRIP AND EMERGENCY FEEDWATER INITIATION.
EVENT DATE: 010186 REPORT DATE: 120987 NSSS: BW TYPE: PWR
VENDOR: BYRON JACKSON PUMPS, INC.

(NSIC 207288) ON JANUARY 1, 1986, CRYSTAL UNIT 3 WAS OPERATING AT 92% REACTOR POWER WHILE GENERATING 830 MWE. AT 2334, A NUMBER OF ALARMS RELATING TO THE "A" REACTOR COOLANT PUMP (RCP) WERE RECEIVED WITHIN A 1.5 SECOND INTERVAL. THREE SECONDS AFTER THE FIRST ALARMS WERE RECEIVED, THE REACTOR TRIPPED ON NUCLEAR OVERPOWER BASED ON REACTOR COOLANT SYSTEM (RCS) FLOW AND AXIAL POWER IMBALANCE (FLUX/DELTA FLUX/FLOW). THE MOTOR FOR "A" RCP CONTINUED TO RUN FOR APPROXIMATELY TWO MINUTES UNTIL MANUALLY SECURED BY THE CONTROL BOARD OPERATOR. THE "A" RCP MOTOR INDICATIONS, ALONG WITH THE RAPID REACTOR COOLANT SYSTEM FLOW DEGRADATION, ARE INDICATIVE OF A SEPARATION BETWEEN THE "A" RCP MOTOR AND THE PUMP FOLLOWING THE EXPECTED MAIN TURBINE AUTOMATIC TRIP, THE TURBINE STOP VALVE CLOSURE CAUSED A STEAM PRESSURE SPIKE WHICH RESULTED IN A SPURIOUS ACTUATION OF THE EMERGENCY FEEDWATER (EFW) SYSTEM. THE CAUSE OF THE SEPARATION BETWEEN THE "A" RCP MOTOR AND THE PUMP WAS A FAILED REACTOR COOLANT PUMP SHAFT. THE FAILED REACTOR COOLANT PUMP SHAFT HAS BEEN REPLACED. THE CAUSE OF THE SPURIOUS EFW ACTUATION WAS THE RAPID SPIKING OF THE LEVEL TRANSMITTERS IN RESPONSE TO AN OSCILLATORY PRESSURE WAVE PHENOMENON FOLLOWING TURBINE STOP VALVE CLOSURE. FPC HAS TAKEN CORRECTIVE ACTION WHICH SHOULD PREVENT THE SPURIOUS START OF EFW FOLLOWING A MAIN TURBINE TRIP IN THE FUTURE.

[58] CRYSTAL RIVER 3 DOCKET 50-302 LER 87-025
ACCIDENTAL GROUNDING OF THE UNIT STARTUP TRANSFORMER LEADS TO INTERRUPTION OFFSITE POWER AND ENGINEERED SAFEGUARDS ACTUATION.
EVENT DATE: 101687 REPORT DATE: 121587 NSSS: BW TYPE: PWR

(NSIC 207506) ON OCTOBER 16, 1987, CRYSTAL RIVER UNIT 3 WAS SHUT DOWN IN A REFUELING OUTAGE. AT 2119, PERSONNEL WORKING IN THE VICINITY OF THE UNIT STARTUP TRANSFORMER RAISED A METAL POLE AND MADE ELECTRICAL CONTACT WITH A 230 KV FEEDER INTERRUPTING THE PLANT OFFSITE POWER SUPPLY. THE FOLLOWING SIGNIFICANT EVENTS RESULTED: THE ENGINEERED SAFEGUARDS SYSTEM ACTUATED, THE "B" DIESEL GENERATOR STARTED AND LOADED, NORMAL POWER WAS LOST TO THE SECURITY SYSTEMS, AND THE REACTOR BUILDING PURGE ISOLATED. ADDITIONALLY, POWER TO THE FOLLOWING WAS LOST: ONE NEUTRON MONITORING CHANNEL, THE AUXILIARY BUILDING VENTILATION SYSTEM EXHAUST FANS, THE CONTROL BOARD ANNUNCIATOR AND EVENT RECORDER, AND THE EMERGENCY NOTIFICATION SYSTEM PHONE. THIS EVENT WAS CAUSED BY ACCIDENTAL GROUNDING OF THE UNIT STARTUP TRANSFORMER 230 KV FEEDER RESULTING IN INTERRUPTION OF THE OFFSITE POWER SUPPLY. ELECTRICAL DISTRIBUTION SYSTEM LINEUPS WERE RESTORED TO THEIR PRE-EVENT STATUS AND THE DAMAGED 230 KV FEEDER WAS REPAIRED. WORK ACTIVITIES IN THE VICINITY OF THE UNIT STARTUP TRANSFORMER HAVE BEEN DISCONTINUED.

[59] CRYSTAL RIVER 3 DOCKET 50-302 LER 87-027
PERSONNEL ERROR LEADS TO VALVING OUT INCORRECT TRANSMITTER WHICH RESULTS IN AN "A" TRAIN ENGINEERED SAFEGUARDS ACTUATION.
EVENT DATE: 102687 REPORT DATE: 112587 NSSS: BW TYPE: PWR

(NSIC 207325) ON OCTOBER 26, 1987 CRYSTAL RIVER UNIT 3 WAS IN A REFUELING OUTAGE AND THE REACTOR WAS DEFUELED. ONE OF THE THREE CHANNELS FOR THE "A" ENGINEERED SAFEGUARDS (ES) TRAIN (CHANNEL 3A) WAS DEENERGIZED FOR MAINTENANCE. DURING PERFORMANCE OF A REACTOR PROTECTION SYSTEM (RPS) SURVEILLANCE, AN INSTRUMENT AND CONTROLS TECHNICIAN INADVERTENTLY VALVED OUT THE WRONG TRANSMITTER. INSTEAD OF VALVING OUT A REACTOR COOLANT PRESSURE TRANSMITTER FOR RPS, ONE FOR ES WAS VALVED OUT. THIS SATISFIED THE 2 OUT OF 3 LOGIC AND RESULTED IN AN "A" TRAIN ES ACTUATION. DUE TO THE PLANT LOAD CONFIGURATION AT THE TIME OF THE EVENT, ONLY THE "A" DECAY HEAT CLOSED CYCLE COOLING PUMP STARTED. THE CAUSE OF THIS EVENT IS

AN ERROR BY THE UTILITY I&C TECHNICIAN. POOR LABELLING OF THE TRANSMITTERS CONTRIBUTED TO THIS ERROR. THE ES ACTUATION WAS RESET BY CONTROL ROOM OPERATORS, AND DCP-1A WAS SHUT DOWN. THE I&C TECHNICIAN WAS COUNSELED ON THE EVENT BY HIS SUPERVISION. THE TRANSMITTERS HAVE HAD NEW LABELS ATTACHED TO THEM. IN ADDITION, ALL OTHER TRANSMITTERS OUTSIDE OF THE SECONDARY SHIELD WALL IN THE REACTOR BUILDING HAVE HAD NEW LABELS ATTACHED IF NECESSARY.

[60] CRYSTAL RIVER 3 DOCKET 50-302 LER 87-022
CIRCUIT BREAKER TRIPS DEENERGIZING SENSING CHANNEL AND ACTUATING ENGINEERED SAFEGUARDS.
EVENT DATE: 110687 REPORT DATE: 120487 NSSS: BW TYPE: PWR

(NSIC 207323) ON NOVEMBER 6, 1987 CRYSTAL RIVER UNIT 3 WAS COMPLETELY DEFUELED. PREPARATIONS WERE BEING MADE FOR PERFORMANCE OF THE CONTAINMENT INTEGRATED LEAK RATE TEST (ILRT). A MODIFICATION WAS BEING INSTALLED WHICH REQUIRED ONE OF THE SENSING CHANNELS IN THE ENGINEERED SAFEGUARDS ACTUATION SYSTEM TRAIN B TO BE DE-ENERGIZED. DE-ENERGIZING THE CHANNEL PLACED IT IN THE TRIPPED CONDITION. WHILE TECHNICIANS WERE INSTALLING JUMPER LEADS IN ANOTHER SENSING CHANNEL IN PREPARATION FOR THE ILRT, A CIRCUIT BREAKER TRIPPED DE-ENERGIZING THIS SECOND CHANNEL. THUS TWO CHANNELS WERE TRIPPED AT THE SAME TIME, COMPLETING A TWO OUT OF THREE LOGIC WHICH IS REQUIRED TO CAUSE AN ENGINEERED SAFEGUARDS ACTUATION. THIS ACTUATION RESULTED IN THE STARTING OF TWO CONTAINMENT FAN COOLER UNITS AND ONE DECAY HEAT CLOSED CYCLE COOLING PUMP MOTOR FAN COOLER UNIT. THE CAUSE OF THE CIRCUIT BREAKER TRIPPING IS NOT CONCLUSIVELY KNOWN. THE MOST LIKELY CAUSE IS THAT THE TECHNICIANS CREATED A SHORT CIRCUIT WHILE INSTALLING THE JUMPER LEAD. THE PROCEDURE FOR ILRT WILL BE REVISED TO REQUIRE USE OF A DIFFERENT TYPE OF LEAD WHICH WILL BE LESS LIKELY TO CAUSE A SHORT CIRCUIT.

[61] CRYSTAL RIVER 3 DOCKET 50-302 LER 87-026
PERSONNEL ERROR RESULTS IN DEVELOPMENT OF AN ERRONEOUS FUEL MOVE SHEET AND PLACING A FUEL ASSEMBLY IN AN INCORRECT LOCATION.
EVENT DATE: 110987 REPORT DATE: 120187 NSSS: BW TYPE: PWR

(NSIC 207324) ON NOVEMBER 9, 1987, CRYSTAL RIVER UNIT 3 WAS SHUT DOWN IN A REFUELING OUTAGE. THE REACTOR VESSEL WAS COMPLETELY DEFUELED TO FACILITATE INSPECTION OF THE CORE FLOOD VALVES. FUEL AND CONTROL ROD ASSEMBLIES WERE BEING MOVED IN THE SPENT FUEL POOLS IN PREPARATION FOR THE CORE RELOAD. AT 1715, WHILE UPDATING THE CONTROL ROOM FUEL LOCATION TAG BOARD, IT WAS NOTED THAT A NEW FUEL ASSEMBLY, WITH 3.851 PERCENT U-235 ENRICHMENT HAD BEEN PLACED IN THE "A" SPENT FUEL POOL. THE FUEL RACKS IN THE "A" SPENT FUEL POOL ARE LIMITED TO STORAGE OF FUEL ASSEMBLIES WITH 3.5 PERCENT OR LESS U-235 ENRICHMENT. THIS EVENT WAS CAUSED BY A PERSONNEL ERROR. WHEN MOVE SHEETS WERE BEING PREPARED TO MOVE A FUEL ASSEMBLY FROM LOCATION M42 IN THE "B" SPENT FUEL POOL TO THE "A" SPENT FUEL POOL, LOCATION M43 WAS INADVERTENTLY WRITTEN INSTEAD OF M42. THE MISLOCATED FUEL ASSEMBLY WAS REMOVED FROM THE "A" SPENT FUEL POOL UPON DETECTION OF ITS MISLOCATION. INDEPENDENT REVIEW OF MOVE SHEETS, PRIOR TO ACTUAL FUEL MOVEMENT, HAS BEEN IMPLEMENTED. OUTAGE.

[62] CRYSTAL RIVER 3 DOCKET 50-302 LER 87-030
UNKNOWN CAUSE LEADS TO "B" TRAIN ENGINEERED SAFEGUARDS ACTUATION.
EVENT DATE: 112087 REPORT DATE: 121887 NSSS: BW TYPE: PWR

(NSIC 207507) ON NOVEMBER 20, 1987 CRYSTAL RIVER UNIT 3 (CR-3) WAS IN A REFUELING OUTAGE AND THE REACTOR VESSEL WAS DEFUELED. ONE OF THE THREE ENGINEERED SAFEGUARDS CHANNELS FOR "B" TRAIN (CHANNEL 1B) WAS DEENERGIZED FOR MAINTENANCE. AT 1638, A "B" TRAIN HPI ACTUATION WAS RECEIVED. CHANNELS 2B AND 3B FOR HPI INDICATED THAT THEY BOTH HAD ACTUATED. AS A RESULT, THE COOLING UNIT FOR THE "B" DECAY HEAT CLOSED CYCLE COOLING PUMP STARTED AND THE REACTOR BUILDING PURGE

ISOLATED. THE CAUSE OF THE ES ACTUATION WAS RESET BY THE CONTROL ROOM OPERATORS AND THE AFFECTED COMPONENTS WERE RETURNED TO THEIR REQUIRED POSITIONS. OPERATIONS VERIFIED THAT NO WORK WAS ONGOING THAT COULD HAVE CAUSED THIS ACTUATION. PRIOR TO STARTUP FROM THE CURRENT REFUELING OUTAGE, THE SURVEILLANCES NECESSARY TO VERIFY PROPER OPERATION OF THE ES LOGIC AND RELAYS WILL BE PERFORMED.

[63] DAVIS-BESSE 1 DOCKET 50-346 LER 87-014
INOPERABLE FIRE HOSE STATION HCS30 DUE TO INADEQUATE PROCEDURAL GUIDANCE.
EVENT DATE: 112587 REPORT DATE: 122887 NSSS: BW TYPE: PWR

(NSIC 207446) ON 11/25/87 IT WAS DETERMINED THAT A DEFICIENCY THAT OCCURRED DURING PERFORMANCE OF THE FIRE HOSE HYDROSTATIC TEST, PT5116.11, MADE THE FIRE HOSE STATION HCS-30 INOPERABLE FROM OCTOBER 22, 1987 UNTIL OCTOBER 26, 1987. HCS-30 PROVIDES MANUAL FIRE FIGHTING CAPABILITY. WHEN THE 75 FT SECTION OF HOSE WAS REMOVED FOR TESTING, IT WAS TEMPORARILY REPLACED WITH A 50 FT. SECTION. THERE WERE NO SPARE 75 FT SECTIONS. IT WAS THOUGHT THAT THIS AFFECTED THE HOSE STATION'S OPERABILITY, AND THE HOSE LENGTH WAS NOT LISTED IN THE ACCEPTANCE CRITERIA OF THE TEST PROCEDURE. THE PERSONNEL ERROR OCCURRED WHEN THIS CHANGE OF HOSE WAS MADE BEFORE OPERABILITY WAS REVIEWED BY FIRE PROTECTION AND/OR THE SHIFT SUPERVISOR. HOWEVER, AFTER REVIEWING THE COMMITMENTS IN THE FIRE HAZARDS ANALYSIS REPORT AND DOING A WALKDOWN OF THE ROOM, IT WAS DETERMINED THAT A 75 FT LENGTH HOSE IS REQUIRED TO ADEQUATELY PROTECT SAFETY RELATED EQUIPMENT. WHEN THIS SITUATION WAS PRESENTED TO THE SHIFT SUPERVISOR ON OCTOBER 26, 1987 HE ASSUMED IT DID AFFECT OPERABILITY AND HAD AN ADDITIONAL LENGTH OF HOSE ADDED TO SATISFY TECHNICAL SPECIFICATION 3.7.9 3. TEST PROCEDURE PT5116.11 WILL BE CHANGED TO INCLUDE HOSE LENGTH IN THE ACCEPTANCE CRITERIA. THIS EVENT HAS BEEN REVIEWED WITH TEST PERSONNEL. THE EVENT IS BEING REPORTED PER 10CFR50.73(A)(2)(I)(B).

[64] DIABLO CANYON 1 DOCKET 50-275 LER 87-019
FAILURE TO MEET TECHNICAL SPECIFICATIONS FOR INOPERABLE ROD POSITION DEVIATION MONITOR DUE TO PERSONNEL ERROR.
EVENT DATE: 102087 REPORT DATE: 111987 NSSS: WE TYPE: PWR

(NSIC 207178) AT 0112 PDT, OCTOBER 20, 1987, WITH THE UNIT IN MODE 1 (POWER OPERATION) AT 40 PERCENT POWER, THE TIME INTERVAL REQUIREMENT SPECIFIED BY TECHNICAL SPECIFICATIONS (TS) 4.1.3.1.1 AND 4.1.3.2, INCLUDING THE ALLOWED EXTENSION OF TS 4.0.2, WAS EXCEEDED. WHEN THE ROD POSITION DEVIATION MONITOR (RPDM) IS INOPERABLE, TS 4.1.3.1.1 AND 4.1.3.2 REQUIRE VERIFICATION EVERY 4 HOURS THAT EACH FULL-LENGTH ROD IS WITHIN ITS GROUP DEMAND LIMIT. AT 2012 PDT, OCTOBER 19, 1987, THE OPERATOR HAD IMPROPERLY UPDATED THE ROD BANK POSITIONS, MAKING THE RPDM INOPERABLE. AT 0552 PDT, OCTOBER 20, 1987, THE RPDM WAS RETURNED TO SERVICE AFTER CORRECTLY UPDATING THE ROD BANK POSITIONS. SINCE OTHER INSTRUMENTATION WAS AVAILABLE TO THE OPERATORS THAT WOULD INDICATE ROD CLUSTER CONTROL ASSEMBLY OPERATION, NO ADVERSE SAFETY CONSEQUENCES OR IMPLICATIONS RESULTED FROM THIS EVENT. THIS EVENT WAS CAUSED BY PERSONNEL ERROR AND THE PROBLEM WITH THE EQUIPMENT/COMPUTER PROGRAM CONTRIBUTED AS WELL. THE RPDM WAS DECLARED INOPERABLE OCTOBER 20, 1987, AT 1257 PDT, PENDING INVESTIGATION OF THE EQUIPMENT/COMPUTER PROGRAM PROBLEM, AND THE 4-HOUR ROD BANK VERIFICATIONS REQUIRED BY TS 4.1.3.1.1 AND 4.1.3.2 WERE INITIATED. A SUPPLEMENTAL REPORT WILL BE SUBMITTED AT THE CONCLUSION OF THIS INVESTIGATION. AN OPERATION SUMMARY REPORT WILL ALSO BE ISSUED FOR REVIEW BY OPERATIONS PERSONNEL.

[65] DIABLO CANYON 1 DOCKET 50-275 LER 87-017
CONTAINMENT VENTILATION ISOLATION INITIATION DUE TO VOLTAGE SPIKE DURING TROUBLESHOOTING.
EVENT DATE: 102287 REPORT DATE: 112087 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 207179) AT 0914 PDT, OCTOBER 22, 1987, WITH THE UNIT IN MODE 1 (POWER OPERATION) AT 100 PERCENT POWER, AN AUTOMATIC INITIATION OF THE CONTAINMENT VENTILATION ISOLATION SYSTEM (CVIS) OCCURRED. THE SAMPLE LINE ISOLATION VALVES FOR CONTAINMENT AIR RADIATION MONITORS (RM) RM11 AND RM12 CLOSED AS DESIGNED. ALL OTHER CVIS VALVES THAT RECEIVE ISOLATION SIGNALS WERE ALREADY CLOSED WHEN THE EVENT OCCURRED. AS REQUIRED BY 10 CFR 50.72 (B)(2)(II), A 4-HOUR NONEMERGENCY EVENT REPORT WAS MADE AT 0945 PDT, OCTOBER 22, 1987. THIS EVENT OCCURRED WHEN AN INSTRUMENTATION AND CONTROLS (I&C) TECHNICIAN REPLACED A BLOWN FUSE WHILE TROUBLESHOOTING AN INOPERABLE CONTROL ROOM AIR PARTICULATE RADIATION MONITOR RM21 . AFTER REPLACING A FUSE, THE TECHNICIAN ENERGIZED RM21. DUE TO A SEIZED PAPER DRIVE MOTOR ON RM21, THE FUSE BLEW, RESULTING IN A VOLTAGE SPIKE AND CVIS INITIATION. THE CVIS WAS RESET AT 0916 PDT, OCTOBER 22, 1987. OPERATORS VERIFIED THAT OTHER CONTAINMENT PARAMETERS (TEMPERATURE, PRESSURE, SUMP LEVELS, AREA RADIATION) WERE NORMAL, THEN RESET THE CVIS, RESET THE ALARM, PLACED RM11 AND RM12 BACK IN SERVICE, AND VERIFIED THAT A HIGH RADIATION CONDITION DID NOT EXIST. THE PAPER DRIVE MOTOR ON RM21 WAS REPLACED. A UNIT 1 DESIGN CHANGE HAS BEEN INITIATED TO ADD TIME DELAY CIRCUITRY TO RADIATION MONITORING CHANNELS THAT INITIATE A CONTAINMENT VENTILATION ISOLATION.

[66] DIABLO CANYON 1 DOCKET 50-275 LER 87-018
FUEL HANDLING BUILDING VENTILATION SYSTEM SHIFTED TO IODINE REMOVAL MODE DUE TO AN INADEQUATE REVIEW OF A DIAGRAM CONNECTION.
EVENT DATE: 102687 REPORT DATE: 112587 NSSS: WE TYPE: PWR

(NSIC 207258) ON OCTOBER 26, 1987, AT 1400 PST, WITH THE UNIT IN MODE (POWER OPERATION) AT 100 PERCENT POWER, THE FUEL HANDLING BUILDING (FHB) VENTILATION SYSTEM SHIFTED FROM NORMAL TO THE IODINE REMOVAL MODE. THIS MODE SHIFT CONSTITUTES AN ACTUATION OF AN ENGINEERED SAFETY FEATURE. THE FHB VENTILATION SYSTEM WAS SHIFTED BACK TO THE NORMAL MODE OF OPERATION AT 1405 PST. THE 4-HOUR NONEMERGENCY REPORT REQUIRED BY 10 CFR 50.72 WAS COMPLETED BY 1455 PST. DURING INSTALLATION OF A DESIGN CHANGE, THE POWER TO RADIATION MONITOR RE-59 WAS INTERRUPTED. THIS INTERRUPTION CAUSED A SIGNAL FROM THE HIGH RADIATION ALARM RELAY FOR RE-59 TO SHIFT THE FHB VENTILATION TO THE IODINE REMOVAL MODE. THE HIGH RADIATION ALARM RELAY HAD BEEN JUMPERED TO PREVENT THIS SHIFTING, BUT THE JUMPER WAS CONNECTED TO A TERMINAL WHICH WAS DEENERGIZED IN ACCORDANCE WITH THE DESIGN CHANGE WORK INSTRUCTION. THE ROOT CAUSE OF THE EVENT IS ATTRIBUTED TO PERSONNEL ERROR (COGNITIVE) RESULTING FROM AN INADEQUATE REVIEW OF A DIAGRAM CONNECTION. THE INSTRUMENTATION AND CONTROLS FOREMAN INVOLVED HAS BEEN COUNSELED, AND LESSONS LEARNED FROM THIS EVENT WILL BE INCORPORATED INTO A TAILBOARD SESSION AND A TRAINING SESSION.

[67] DIABLO CANYON 1 DOCKET 50-275 LER 87-020
FAILURE TO MEET TECHNICAL SPECIFICATION FOR INOPERABLE QUADRANT POWER TILT RATIO ALARM DUE TO PERSONNEL ERROR.
EVENT DATE: 102887 REPORT DATE: 112587 NSSS: WE TYPE: PWR

(NSIC 207321) AT 0700 PST, OCTOBER 28, 1987, THE TIME INTERVAL REQUIREMENT SPECIFIED BY TECH SPEC 4.2.4.1, INCLUDING THE ALLOWED EXTENSION OF TECH SPEC 4.0.2, WAS EXCEEDED. WHEN THE QUADRANT POWER TILT RATIO (QPTR) ALARM IS INOPERABLE, TECH SPEC 4.2.4.1 REQUIRES CALCULATION OF THE QPTR EVERY 12 HOURS. THE QPTR ALARM BECAME INOPERABLE WHEN ONE OF FOUR NUCLEAR INSTRUMENTATION SYSTEM (NIS) UPPER AND LOWER DETECTOR CURRENT COMPARATORS WERE DISABLED. THESE DETECTOR COMPARATORS WERE DISABLED FOR 17.5 HOURS DURING WHICH TIME THE QPTR WAS NOT CALCULATED ONCE EVERY 12 HOURS AS REQUIRED BY THE TECH SPEC. DURING THE TIME OF THE QPTR INOPERABILITY, THE ROD POSITION DEVIATION MONITOR WAS ALSO INOPERABLE AND ROD POSITIONS WERE VERIFIED TO BE WITHIN GROUP DEMAND LIMITS EVERY 4 HOURS. THEREFORE, ANY ROD MISALIGNMENT WOULD HAVE BEEN DETECTED. THE QPTR WAS ALSO CALCULATED WHEN THE ALARM WAS RESTORED TO SERVICE AND FOUND TO BE WITHIN TECH SPEC LIMITS. THEREFORE, THERE WERE NO ADVERSE SAFETY CONSEQUENCES OR

IMPLICATIONS RESULTED FROM THIS EVENT. THIS EVENT WAS CAUSED BY PERSONNEL ERROR, COMPOUNDED BY A TRAINING DEFICIENCY, AND MISLEADING LABELING OF INSTRUMENTATION. TO PREVENT RECURRENCE, INSTRUMENTATION WILL BE RELABELED; AN INCIDENT SUMMARY OF THIS EVENT WILL BE REVIEWED WITH ALL OPERATIONS PERSONNEL.

[68] DIABLO CANYON 1 DOCKET 50-275 LER 87-022
REACTOR COOLANT SYSTEM CONTROL ROOM TEMPERATURE RECORDERS DECLARED INOPERABLE DUE TO INADVERTENT FAILURE TO REINSTALL SEISMIC RESTRAINTS.
EVENT DATE: 110487 REPORT DATE: 120487 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: DIABLO CANYON 2 (PWR)

(NSIC 207388) ON NOVEMBER 4, 1987, WITH UNITS 1 AND 2 IN MODE 1 (POWER OPERATION) AT 100 PERCENT POWER, INSTRUMENTATION AND CONTROLS (I&C) TECHNICIANS WERE PERFORMING I&C MAINTENANCE PROCEDURE (MP) 3.1-1, "ROUTINE MAINTENANCE ON HAGAN OPTIMAC RECORDERS." DURING THE PERFORMANCE OF THE PROCEDURE, AN I&C TECHNICIAN AND QUALITY CONTROL INSPECTOR DISCOVERED THAT THE SEISMIC RESTRAINTS FOR THE FOUR CONTROL ROOM REACTOR PLANT SYSTEM WIDE RANGE TEMPERATURE RECORDERS ON BOTH UNITS WERE MISSING. ON DEEMBER 5, 1987, IT WAS DETERMINED THAT THE RECORDERS CANNOT BE CONSIDERED OPERABLE WITHOUT THE RESTRAINTS AND THAT THE REQUIREMENTS OF TECHNICAL SPECIFICATION (TS) 3.3.3.6, TABLE 3.3-10, HAD BEEN VIOLATED. THE ORIGINAL SEISMIC RESTRAINTS FOR THE CONTROL ROOM TEMPERATURE RECORDERS WERE INSTALLED IN 1985. THE RESTRAINTS WERE REMOVED AND INADVERTENTLY NOT REINSTALLED IN PERIOD BETWEEN 1985 AND NOVEMBER 1987. NEW SEISMIC RESTRAINTS WERE FABRICATED INSTALLED ON NOVEMBER 6, 1987. THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR. TRAINING WILL BE CONDUCTED WITH THE OPERATIONS PERSONNEL AND I&C TECHNICIANS EMPHASIZING THE IMPORTANCE OF SEISMIC RESTRAINTS FOR EQUIPMENT. THE ENGINEERING DEPARTMENT WILL EVALUATE ALTERNATIVE SEISMIC RESTRAINT DESIGN TO PROVIDE EASIER ACCESS TO THE RECORDERS.

[69] DIABLO CANYON 1 DOCKET 50-275 LER 87-021
ACTUATION OF ENGINEERED SAFETY FEATURES DUE TO INADVERTENT GROUNDING OF ELECTRICAL COMPONENT.
EVENT DATE: 111387 REPORT DATE: 121187 NSSS: WE TYPE: PWR

(NSIC 207294) ON NOVEMBER 13, 1987, AT 1340 PST, WITH THE UNIT IN MODE 1 (POWER OPERATION) AT 100 PERCENT POWER, THE FUEL HANDLING BUILDING VENTILATION SYSTEM SHIFTED TO THE IODINE REMOVAL MODE, THE AUXILIARY BUILDING VENTILATION SYSTEM SHIFTED TO SAFEGUARDS-ONLY MODE AND THE CONTAINMENT VENTILATION WAS ISOLATED. THE VENTILATION SYSTEMS WERE SHIFTED BACK TO THE NORMAL MODES OF OPERATION AT 1346 PST. THE FOUR-HOUR NONEMERGENCY REPORT REQUIRED BY 10 CFR 50.72 WAS COMPLETED BY 1420 PST. DURING PERFORMANCE OF A SURVEILLANCE TEST, A PROBLEM WAS FOUND IN THE LOW VOLTAGE POWER SUPPLY FOR RADIATION MONITOR RM-17A. THE INSTRUMENT DRAWER WAS DEENERGIZED BY REMOVAL OF THE FUSES. DURING REMOVAL OF THE LOW VOLTAGE POWER SUPPLY, AN ELECTRICAL FILTER ON THE LINE SIDE OF THE FUSE WAS INADVERTENTLY SHORTED TO THE DRAWER WITH A SCREWDRIVER. THIS GROUNDING CAUSED A VOLTAGE DIP IN VITAL INSTRUMENT AC DISTRIBUTION PANEL PY-11A, WHICH IN TURN CAUSED THE ENGINEERED SAFETY FEATURES TO ACTUATE. THE ROOT CAUSE IS ATTRIBUTED TO PERSONNEL ERROR IN THAT THE TECHNICIANS FAILED TO REALIZE THAT THERE WAS POWER IN THE INSTRUMENT DRAWER WHERE THE WORK WAS BEING PERFORMED. LESSONS LEARNED FROM THIS EVENT WILL BE INCORPORATED INTO A TAILBOARD SESSION AND COVERED IN INSTRUMENT AND CONTROL DEPARTMENT TRAINING.

[70] DIABLO CANYON 2 DOCKET 50-323 LER 87-024
MANUAL REACTOR TRIP AFTER DISCOVERY OF ARCING IN ISOPHASE BUS MOTOR OPERATED DISCONNECT SWITCH DUE TO HIGH RESISTANCE ON THE CONTACTS.
EVENT DATE: 110787 REPORT DATE: 120487 NSSS: WE TYPE: PWR
VENDOR: PORTER, H. K. COMPANY, INC.

(NSIC 207296) AT 1315 PST, NOVEMBER 7, 1987, WITH THE UNIT IN MODE 1 (POWER 98 PERCENT POWER, A MANUAL REACTOR TRIP WAS INITIATED WHEN ARCING WAS DISCOVERED IN THE NON SAFETY RELATED ISOPHASE BUS MOTOR-OPERATED DISCONNECT (MOD) SWITCH. THE UNIT WAS STABILIZED IN MODE 3 (HOT SHUTDOWN) AT 1430 PST. THE FOUR-HOUR NONEMERGENCY REPORT REQUIRED BY 10 CFR 50.72 WAS MADE AT 1330 PST. ON MONDAY, NOVEMBER 9, 1987 A REPLACEMENT MOD WAS OBTAINED. THE REPLACEMENT SWITCH AND THE SWITCHES FROM THE UNAFFECTED PHASES WERE INSPECTED, CLEANED, LUBRICATED, AND ALIGNED. THE UNIT WAS RETURNED TO POWER ON NOVEMBER 14, 1987. THE MOD FAILURE WAS CAUSED BY A COMBINATION OF HIGH AMBIENT TEMPERATURE AND HIGH RESISTANCE AT THE CONTACTS. THE HIGH RESISTANCE WAS CAUSED BY DUST AND HARDENED LUBRICANT ON THE CONTACTS AND A SLIGHT MISALIGNMENT OF SOME CONTACTS. A COMPREHENSIVE PREVENTIVE MAINTENANCE PROCEDURE IS BEING DEVELOPED FOR THE ISOPHASE BUS MODS. A SUPPLEMENTAL REPORT WILL BE SUBMITTED AFTER COMPLETION OF THE EVENT INVESTIGATION TO REPORT ANY ADDITIONAL FINDINGS AND CORRECTIVE ACTIONS TAKEN TO PREVENT RECURRENCE.

[71]   DRESDEN 2   DOCKET 50-237   LER 86-019 REV 01
UPDATE ON REACTOR SCRAM FROM MAIN TURBINE TRIP ON HIGH WATER LEVEL DUE TO FAILURE OF FEEDWATER REGULATING VALVE AND PERSONNEL ERROR.
EVENT DATE: 081186   REPORT DATE: 110787   NSSS: GE   TYPE: BWR
VENDOR: COPES-VULCAN, INC.
        CRANE COMPANY

(NSIC 206898) ON 8/11/86 AT 1433 HOURS, WITH UNIT 2 OPERATING AT 91% POWER, THE MAIN TURBINE TRIPPED ON +55 INCHES OF REACTOR WATER LEVEL WHICH RESULTED IN A FULL REACTOR SCRAM FROM A TURBINE GENERATOR LOAD REJECTION SIGNAL. WHILE THE NUCLEAR STATION OPERATOR (NSO) WAS REMOVING PIECES OF A BROKEN CONTROL ROOM INDICATING LIGHT BULB ON REACTOR RECIRCULATING PUMP LUBE OIL PUMP "B1", A SHORT CIRCUIT OCCURRED AND CAUSED THE "B" RECIRCULATION PUMP TO TRIP. THE PUMP TRIP CAUSED THE REACTOR WATER LEVEL TO INCREASE TO +55 INCHES AND CAUSED THE MAIN TURBINE TO TRIP AND A SUBSEQUENT REACTOR FULL SCRAM. WHEN THE NSO TRIED TO MANUALLY CLOSE THE "A" FEEDWATER REGULATING VALVE, THE NSO NOTICED THE VALVE POSITION INDICATION SHOWED NO MOVEMENT. INSPECTION OF THE FAILED REGULATOR VALVE DISCOVERED A KEEPER FROM A FEEDWATER PUMP CHECK VALVE JAMMED UNDER THE VALVE'S SEAT. THE KEEPER WAS REMOVED AND THE REGULATING VALVE WAS CYCLED SUCCESSFULLY. THE ROOT CAUSES OF THIS EVENT ARE: 1) FAILURE OF THE NSO TO COMPLY WITH DRESDEN OPERATING PROCEDURE (DOP) 040-4, AND 2) FAILURE OF THE "A" REACTOR FEEDWATER DISCHARGE CHECK VALVE. CORRECTIVE ACTIONS INCLUDE: 1) OPERATOR TRAINING , 2) INVESTIGATION OF IMPROVED SEAT HOLD DOWN RING RETAINING METHODS, AND 3) AMENDMENT TO DOP 3200.5, REACTOR FEED PUMP SHUTDOWN.

[72]   DRESDEN 2   DOCKET 50-237   LER 87-010 REV 01
UPDATE ON CORE SPRAY SYSTEM "A" ANALYTICAL PIPING STRESSES EXCEED FINAL SAFETY ANALYSIS REPORT DESIGN REQUIREMENTS DUE TO DESIGN AND CONSTRUCTION ERRORS.
EVENT DATE: 031787   REPORT DATE: 111287   NSSS: GE   TYPE: BWR
OTHER UNITS INVOLVED: DRESDEN 3 (BWR)

(NSIC 207315) WHILE PERFORMING PIPING SYSTEM INSPECTIONS WITH DRESDEN UNIT 2 IN COLD SHUTDOWN FOR A REFUELING OUTAGE. A REVIEW OF THE CORE SPRAY PIPING WAS INITIATED AND THE FOLLOWING ADDITIONAL DISCREPANCIES WERE FOUND; 1) THE DEGRADED EMBEDMENT PLATE HAD NOT BEEN FABRICATED PROPERLY, 2) A TEST RETURN LINE TEE HAD A WELDED CONNECTION WHERE THE DESIGN ANALYSIS INDICATED A FORGED CONNECTION, 3) A SUPPORT WHICH HAD PREVIOUSLY BEEN SPECIFIED FOR REMOVAL WAS STILL IN PLACE AND A SUPPORT INTENDED TO REPLACE IT HAD BEEN INSTALLED INCORRECTLY, 4) A NON-EXISTENT SNUBBER WAS INDICATED IN THE DESIGN ANALYSIS, 5) CERTAIN PIPING SECTIONS WERE A DIFFERENT SCHEDULE TYPE THAN THE DESIGN ANALYSIS INDICATED, AND 6) ANALYSIS REVEALED TWO ADDITIONAL EMBEDMENT PLATES NEEDING REINFORCEMENT. ON MARCH 17, 1987 THE CORPORATE ENGINEERING DEPARTMENT INDICATED THAT THE AS-FOUND CONDITION WAS IN EXCESS OF FINAL SAFETY ANALYSIS REPORT (FSAR) LIMITS BUT MET OPERABILITY CRITERIA

FOR ALL DESIGN BASIS EVENTS AND THUS WAS OF MINIMAL SAFETY SIGNIFICANCE. REPAIRS WERE COMPLETED PRIOR TO UNIT STARTUP TO ENSURE FSAR COMPLIANCE. SUBSEQUENT INSPECTIONS AND ANALYSES HAVE REVEALEL OTHER INCIDENCES OF OVERSTRESSED PIPING SUPPORTS AND CONNECTIONS, WHICH ARE LISTED IN THE TEXT OF THIS REPORT.

[73] DRESDEN 2 DOCKET 50-237 LER 87-030
MAIN STEAM SAFETY VALVE SETPOINTS FOUND OUTSIDE TECH SPEC LIMITS DUE TO MISHANDLING AND SETPOINT DRIFT.
EVENT DATE: 101687 REPORT DATE: 111387 NSSS: GE TYPE: BWR
VENDOR: DRESSER INDUSTRIES, INC.

(NSIC 207140) ON OCTOBER 13, AND OCTOBER 16, 1987, TWO OF THE FOUR MAIN STEAM SAFETY VALVES, WHICH WERE REMOVED DURING THE UNIT 2 1986-1987 REFUEL OUTAGE, WERE BEING TESTED TO DETERMINE THEIR AS FOUND SETPOINTS. THE VALVES, EACH WITH A DESIGN SETPOINT OF 1260 PSIG, OPENED AT PRESSURES OF 1276 PSIG (SERIAL NUMBER BK 6290) AND 1282 PSIG (SERIAL NUMBER BK 6260). THESE SETPOINTS ARE OUTSIDE THE PLUS OR MINUS ONE PERCENT TOLERANCE REQUIRED BY TECH SPEC 4.6.E. THE CAUSE OF THE HIGH SETPOINTS HAS BEEN ATTRIBUTED TO MISHANDLING OF THE VALVES DURING TRANSPORT AND SETPOINT DRIFT. AS A CORRECTIVE ACTION, THE VALVES WILL BE OVERHAULED, RETESTED AND SET AT A PRESSURE WITHIN ONE PERCENT OF THEIR DESIGN SETPOINT. TO PREVENT RECURRENCE OF THIS TYPE EVENT, THE PROCEDURE FOR OVERHAULING THE VALVES HAS BEEN SUBSTANTIALLY IMPROVED. THE SAFETY SIGNIFICANCE OF THIS EVENT IS MINIMAL BASED ON THE EVALUATION WHICH SHOWS THAT WITH THE VALVES IN THIS CONDITION, THE MAXIMUM ALLOWABLE PRESSURE WOULD NOT HAVE BEEN EXCEEDED. THE LAST EVENT OF THIS TYPE WAS REPORTED BY LER 86-029 ON DOCKET #050237, WHICH INVOLVED A RELIEF VALVE ON THE STANDBY LIQUID CONTROL (SBLC) SYSTEM.

[74] DRESDEN 2 DOCKET 50-237 LER 87-032
REACTOR SCRAM DUE TO SPURIOUS MAIN STEAM LINE LOW PRESSURE SIGNAL CAUSED BY VIBRATION.
EVENT DATE: 102087 REPORT DATE: 110587 NSSS: GE TYPE: BWR
VENDOR: BALKSDALE VALVE COMPANY

(NSIC 207169) ON OCTOBER 20, 1987, WHILE DRESDEN UNIT 2 WAS OPERATING AT 100% RATED THERMAL POWER, DRESDEN OPERATING SURVEILLANCE (DOS) 1700-1, MAIN STEAM LINE (MSL) RADIATION MONITOR SCRAM AND ISOLATION FUNCTIONAL TEST, WAS BEING PERFORMED IN ACCORDANCE WITH TECH SPEC TABLE 4.1.1 IN ORDER TO VERIFY PROPER OPERATION OF THE MSL RADIATION MONITOR AND ISOLATION CIRCUITRY. AT 0224 HOURS, WHILE DOS 1700-1 WAS IN PROGRESS, A PRIMARY CONTAINMENT GROUP I ISOLATION AND REACTOR SCRAM OCCURRED, DUE TO A SPURIOUS HALF GROUP ONE ISOLATION OCCURRING COINCIDENT WITH THE SURVEILLANCE. ROOT CAUSE WAS DETERMINED TO BE VIBRATION OF A MSL RECURRENCE, ISOLATORS WERE INSTALLED TO MAKE THE MSL LOW PRESSURE SWITCH; THE SOURCE OF THE VIBRATION IS BELIEVED TO BE THE MAIN TURBINE. IN ORDER TO PREVENT LOW PRESSURE SWITCHES LESS SUSCEPTIBLE TO VIBRATION-INDUCED TRIPS. SAFETY SIGNIFICANCE WAS MINIMAL SINCE THE AUTOMATIC ISOLATION AND SCRAM LOGIC FUNCTIONED AS DESIGNED. ALTHOUGH PREVIOUS SPURIOUS LOW MSL PRESSURE SIGNALS HAVE BEEN OBSERVED SINCE THE LAST REFUEL OUTAGE, THESE HAVE NOT RESULTED IN AN AUTOMATIC ISOLATION OR SCRAM.

[75] DRESDEN 2 DOCKET 50-237 LER 87-033
FAILURE TO CONTINUOUSLY MONITOR REACTOR BUILDING VENTILATION GASEOUS EFFLUENT DUE TO PERSONNEL ERROR.
EVENT DATE: 111087 REPORT DATE: 113087 NSSS: GE TYPE: BWR

(NSIC 207354) ON NOVEMBER 10, 1987 WITH UNIT 2 AT 100% POWER, A RADIATION CHEMISTRY TECHNICIAN (RCT) OBSERVED THAT THE UNIT 2 REACTOR BUILDING VENTILATION SYSTEM SAMPLE HOLDER CONTAINED NO PARTICULATE SAMPLE FILTER. THE 2/3 REACTOR BUILDING SAMPLE PARTICULATE, IODINE, AND NOBLE GAS MONITOR WAS OUT OF SERVICE DURING THIS TIME PERIOD. AS SUCH, NO METHOD EXISTED FOR CONTINUOUSLY MONITORING

PARTICULATES. THE CAUSE OF THIS EVENT WAS A PERSONNEL ERROR. ON NOVEMBER 3, 1987 AN RCT WAS ASSIGNED TO CHANGE THE REACTOR BUILDING VENTILATION SAMPLER FILTERS. ON NOVEMBER 10, 1987 THE RCT WAS AGAIN ASSIGNED TO CHANGE THE FILTERS. WHEN HE OPENED THE CANNISTER CONTAINING THE FILTERS, HE OBSERVED THAT THE PARTICULATE FILTER WAS MISSING. HE INSTALLED THE NEW FILTERS, AND REPORTED THE MISSING FILTER TO RADIATION CHEMISTRY SUPERVISION. IN ORDER TO PREVENT A SIMILAR FUTURE EVENT, A PROCEDURE WILL BE WRITTEN TO DEFINE THE REQUIREMENTS OF REPLACING THE SAMPLER FILTERS, WHICH WILL ALSO INCLUDE A CHECKLIST REQUIRING THE RCT TO INITIAL THE COMPLETION OF STEPS. UNTIL THIS PROCEDURE IS APPROVED BY THE STATION, RADIATION CHEMISTRY SUPERVISION WILL VERIFY THAT SAMPLE FILTER INSTALLATIONS ARE PROPERLY ACCOMPLISHED. THE LAST EVENT OF THIS TYPE WAS REPORTED BY LER #87-025 ON DOCKET #050237.

[76] DRESDEN 3 DOCKET 50-249 LER 85-018 REV 01
UPDATE ON REACTOR SCRAM ON HIGH FLUX RESULTING FROM TURBINE CONTROL VALVE CLOSURE DUE TO PERSONNEL ERROR.
EVENT DATE: 091985 REPORT DATE: 121187 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: DRESDEN 2 (BWR)
VENDOR: ASCO VALVES
GENERAL ELECTRIC CO.

(NSIC 207514) WHILE OPERATING AT 83% POWER, UNIT 3 TRIPPED FROM AN AVERAGE POWER RANGE MONITOR (APRM) HIGH-HIGH LEVEL SCRAM. THE TRIP RESULTED FROM A PRESSURE/FLUX TRANSIENT WHICH WAS CAUSED BY CLOSURE OF THE TURBINE CONTROL VALVES. THE CAUSE OF THIS REACTOR SCRAM WAS PRESONNEL ERROR. AN INSTRUMENT MAINTENANCE MECHANIC ACCIDENTALLY MOVED A CIRCUIT CARD IN THE ELECTRO HYDRAULIC CONTROL SYSTEM CIRCUITRY WHILE REMOVING A TEST LEAD. THIS DISRUPTED THE MAXIMUM COMBINED FLOW PORTION OF THE CIRCUITRY RESULTING IN CLOSURE OF THE TURBINE CONTROL VALVES. DURING THE SCRAM RECOVERY, DIFFICULTY WAS ENCOUNTERED IN RESETTING REACTOR PROTECTION SYSTEM (RPS) CHANNEL B. DURING THE SCRAM RECOVERY, THE SCRAM DISCHARGE VOLUME (SDV) VENT AND DRAIN VALVES OPENED WHILE THE CONTROL ROD DRIVE (CRD) SCRAM INLET AND OUTLET VALVES ON EVERY CRD WERE OPEN. THIS RESULTED IN THE RELEASE OF REACTOR VESSEL WATER INVENTORY PAST THE CRD SEALS ONTO THE REACTOR BUILDING. TO PREVENT RECURRENCE OF AN EVENT OF THIS TYPE, SEVERAL CHANGES WILL BE MADE TO THE UNIT 2/3 SCRAM PROCEDURE, DGP 2-3, TO ENSURE THAT THE SDV VENT AND DRAIN VALVES REMAIN CLOSED UNTIL ALL SCRAM INLET AND OUTLET VALVES ARE CLOSED. FOLLOWING FURTHER INVESTIGATION INTO THE ROOT CAUSE OF THIS EVENT, LARGER AIR REGULATORS WILL ALSO BE INSTALLED ON THE UNIT 2 AND 3 SDV AIR HEADER SYSTEMS.

[77] DRESDEN 3 DOCKET 50-249 LER 86-018 REV 01
UPDATE ON SPURIOUS GROUP V CONTAINMENT ISOLATION DUE TO DESIGN DEFICIENCY.
EVENT DATE: 101486 REPORT DATE: 121187 NSSS: GE TYPE: BWR
VENDOR: DIETERICH STANDARD CORP.

(NSIC 207515) ON 10/14/86, UNIT 3 WAS SHUT DOWN AND THE REACTOR VESSEL WAS DEPRESSURIZED SEVERAL HOURS AFTER A REACTOR SCRAM (SEE LER 249/86-017). AT 2050 HOURS, A GROUP V (ISOLATION CONDENSER) CONTAINMENT ISOLATION SIGNAL WAS RECEIVED IN THE CONTROL ROOM. ALL THE ISOLATION CONDENSER VALVES RESPONDED TO THE CONTAINMENT ISOLATION SIGNAL AS REQUIRED. FOLLOWING THE ISOLATION, DRESDEN INSTRUMENT SURVEILLANCE (DIS) 1300-2, ISOLATION CONDENSER STEAM/CONDENSATE LINE HIGH FLOW CALIBRATION AND FUNCTIONAL TEST, WAS PERFORMED BY INSTRUMENT MAINTENANCE. ALL THE ISOLATION CONDENSER INSTRUMENT SETPOINTS WERE FOUND WITHIN THEIR TECH SPEC LIMITS. A SECOND SPURIOUS GROUP V ISOLATION OCCURRED ON DRESDEN UNIT 3 ON NOVEMBER 13, 1986 WITH THE REACTOR SHUT DOWN AND DEPRESSURIZED. ON AUGUST 7, 1987, WHILE THE ISOLATION CONDENSER WAS IN USE FOLLOWING A UNIT 3 SHUTDOWN FROM 82% RATED THERMAL POWER, A SPURIOUS GROUP V ISOLATION AGAIN OCCURRED (SEE REPORTABLE OCCURRENCE 87-013 ON DOCKET 50-249). UPON FURTHER REVIEW, A SPECIAL TEST OF THE ISOLATION CONDENSER WAS PERFORMED IN ORDER TO

PINPOINT THE ROOT CAUSE ON SEPTEMBER 5, 1987 (SEE LER 249/87 -014). THE CAUSE OF THE SPURIOUS ISOLATION CONDENSER ISOLATION WAS DETERMINED TO BE DIFFERENTIAL PRESSURE SPIKES AND/OR NOISE GENERATED BY AN ANNUBAR FLOW INSTRUMENT.

[78] DRESDEN 3 DOCKET 50-249 LER 86-020 REV 01
UPDATE ON SPURIOUS GROUP V CONTAINMENT ISOLATION DUE TO DESIGN DEFICIENCY.
EVENT DATE: 111386 REPORT DATE: 112187 NSSS: GE TYPE: BWR
VENDOR: DIETERICH STANDARD CORP.

(NSIC 207516) ON NOVEMBER 13, 1986, UNIT 3 WAS SHUT DOWN AND THE REACTOR VESSEL WAS DEPRESSURIZED SEVERAL HOURS AFTER A REACTOR SCRAM (SEE LER NO. 86-019 ON DOCKET NO. 50-249). AT 2140 HOURS, A GROUP V (ISOLATION CONDENSER) CONTAINMENT ISOLATION SIGNAL WAS RECEIVED. ALL THE ISOLATION CONDENSER VALVES RESPONDED TO THE ISOLATION SIGNAL AS REQUIRED. A PREVIOUS SIMILAR EVENT OCCURRED ON OCTOBER 14, 1986 (SEE LER NO. 86-018 ON DOCKET NO. 50-249). ON AUGUST 7, 1987, WHILE THE ISOLATION CONDENSER WAS IN USE FOLLOWING A UNIT 3 SHUTDOWN FROM 82% RATED THERMAL POWER, A SPURIOUS GROUP V ISOLATION AGAIN OCCURRED (SEE LER NO. 87-13 ON DOCKET NO. 50-249). UPON FURTHER REVIEW, A SPECIAL TEST OF THE ISOLATION CONDENSER WAS PERFORMED ON SEPTEMBER 5, 1987 IN ORDER TO PINPOINT THE ROOT CAUSE (SEE LER NO. 87-14 ON DOCKET 50-249). THE CAUSE OF THE SPURIOUS ISOLATION CONDENSER ISOLATIONS WAS DETERMINED TO BE DIFFERENTIAL PRESSURE SPIKES AND/OR NOISE GENERATED BY AN ANNUBAR FLOW INSTRUMENT INSTALLED ON THE ISOLATION CONDENSER CONDENSATE RETURN LINE DURING THE 1985 REFUELING OUTAGE. HOWEVER, THE ISOLATION CONDENSER DID OPERATE SATISFACTORILY IN THIS CONDITION PRIOR TO THE AUGUST 7, 1987 EVENT. SAFETY SIGNIFICANCE OF AN INOPERABLE ISOLATION CONDENSER IS MITIGATED BY THE HIGH PRESSURE COOLANT INJECTION SYSTEM.

[79] FARLEY 1 DOCKET 50-348 LER 87-023
FIRE DAMPERS INOPERABLE DUE TO FAILURE TO CLOSE WITH AIR FLOW.
EVENT DATE: 112787 REPORT DATE: 122287 NSSS: WE TYPE: PWR

(NSIC 207447) ON 7-13-87, FARLEY NUCLEAR PLANT SUBMITTED A SPECIAL REPORT (LER 87-011-00) CONCERNING INOPERABLE CONTROL ROOM FIRE DAMPERS. AS A RESULT, A FIRE DAMPER MAINTENANCE AND TESTING PROGRAM IS IN PROGRESS. THE FOLLOWING FIRE DAMPERS WOULD NOT CLOSE WITH AIR FLOW IN THE SYSTEM (THE DATE OF THE TEST IS SHOWN IN PARENTHESES): 1-133-01 (11-20-87), 1-133-02 (11-20-87), 1-118-04 (12-1-87), 1-118-20 (12-1-87), 1-118-21 (12-1-87), 1-131-01 (12-4-87), AND 1-131-04 (12-7-87). THESE EVENTS WERE CAUSED BY DESIGN DEFICIENCY IN THAT THE FIRE DAMPERS WILL NOT CLOSE FULLY WITH AIR FLOW. DESIGN CHANGES HAVE BEEN INITIATED TO EVALUATE THE OPTIONS AVAILABLE AND PROVIDE THE APPROPRIATE DESIGN TO ENSURE THE PROPER OPERATION OF THE FIRE DAMPERS. THESE DESIGN CHANGES ARE EXPECTED TO BE IMPLEMENTED WITHIN THE NEXT SIX MONTHS. TECHNICAL SPECIFICATION 3.7.12 REQUIRES THESE FIRE DAMPERS TO BE RETURNED TO OPERABLE STATUS WITHIN SEVEN DAYS OR A SPECIAL REPORT MUST BE SUBMITTED WITHIN THE FOLLOWING 30 DAYS. THEREFORE THIS SPECIAL REPORT IS BEING SUBMITTED. ALL TECHNICAL SPECIFICATION ACTION STATEMENT REQUIREMENTS FOR THE FIRE DAMPERS ARE BEING MET.

[80] FARLEY 2 DOCKET 50-364 LER 87-002
TRAINS OF SERVICE WATER COOLING TO THE BATTERY CHARGER ROOM COOLERS ARE CROSSED.
EVENT DATE: 090387 REPORT DATE: 100287 NSSS: WE TYPE: PWR

(NSIC 206652) AT 1645 ON 9-3-87, WITH THE UNIT OPERATING AT 100% REACTOR POWER, IT WAS DETERMINED THAT THE UNIT 2 A TRAIN BATTERY CHARGER ROOM COOLER WAS SUPPLIED WITH B TRAIN SERVICE WATER AND THE B TRAIN BATTERY CHARGER ROOM COOLER WAS SUPPLIED WITH A TRAIN SERVICE WATER. THE C BATTERY CHARGER ROOM COOLER SERVICE WATER WAS VERIFIED TO BE CORRECT. VARIOUS EQUIPMENT FAILURES WHICH COULD AFFECT BATTERY CHARGER ROOM COOLING WERE POSTULATED. IT WAS DETERMINED THAT EQUIPMENT COVERED BY TECHNICAL SPECIFICATIONS WAS NOT RENDERED INOPERABLE BY THIS

EVENT. FSAR TABLE 9.4-7 STATES THAT IF A BATTERY CHARGER ROOM COOLER FAILS, THE FAILED COOLER WILL BE ISOLATED AND REPAIRED; THE SPARE BATTERY CHARGER (C) AND THE ASSOCIATED COOLER WILL BE PLACED IN OPERATION. PROCEDURES WERE DEVELOPED AND IMPLEMENTED WHICH WOULD PROVIDE FOR CONTINUED COOLING TO BOTH TRAINS OF THE BATTERY CHARGER ROOMS IN THE EVENT OF POSTULATED EQUIPMENT FAILURES. THIS EVENT WAS CAUSED BY AN ORIGINAL DESIGN ERROR THAT RESULTED IN AN INCORRECT DESIGN DRAWING. CONSEQUENTLY, THE A AND B TRAIN SERVICE WATER BATTERY CHARGER ROOM COOLER WATER SUPPLIES WERE CROSSED. A DESIGN CHANGE WILL BE IMPLEMENTED TO PROVIDE PROPER TRAIN ALIGNMENT. THIS DESIGN CHANGE IS EXPECTED TO BE COMPLETED DURING THE UPCOMING REFUELING OUTAGE COMMENCING OCTOBER 1987.

[81] FARLEY 2 DOCKET 50-364 LER 87-003
PERSONNEL ERROR RESULTS IN TECHNICAL SPECIFICATION ACTION STATEMENT REQUIREMENTS NOT BEING MET WHEN R-29A WAS INOPERABLE.
EVENT DATE: 101787 REPORT DATE: 110987 NSSS: WE TYPE: PWR

(NSIC 207102) BETWEEN 1040 AND 1530 ON 10-17-87, CONTINUOUS IODINE AND PARTICULATE SAMPLING OF THE PLANT VENT STACK BY R-29A WAS INTERRUPTED DUE TO A TAGGING ERROR. TECHNICAL SPECIFICATION 3.3.3.11 ACTION 39 ALLOWS EFFLUENT RELEASES TO CONTINUE VIA THE VENT STACK FOR UP TO 30 DAYS PROVIDED THAT CONTINUOUS SAMPLES ARE COLLECTED WITH AUXILIARY SAMPLING EQUIPMENT. NORMAL VENT STACK EFFLUENT DID CONTINUE; HOWEVER, AUXILIARY SAMPLING WAS NOT INITIATED SINCE NO ONE RECOGNIZED THAT NORMAL SAMPLING HAD BEEN SECURED. THIS EVENT WAS CAUSED BY PERSONNEL ERROR. THE BREAKER FOR R-29A WAS INCORRECTLY SPECIFIED ON A TAGGING ORDER IN PLACE OF THE BREAKER FOR R-29B. IN ORDER TO PREVENT RECURRENCE, THE INDIVIDUAL RESPONSIBLE FOR PREPARATION OF THE TAGGING ORDER HAS BEEN COUNSELED.

[82] FARLEY 2 DOCKET 50-364 LER 87-005
PERSONNEL ERROR CAUSES ACTUATION OF ENGINEERED SAFETY FEATURE EQUIPMENT.
EVENT DATE: 111187 REPORT DATE: 120987 NSSS: WE TYPE: PWR

(NSIC 207372) AT 1231 ON 11-11-87, DURING A REFUELING OUTAGE, AN ACTUATION OF ENGINEERED SAFETY FEATURE EQUIPMENT OCCURRED WHEN THE 1-2A AND 1C DIESEL GENERATORS STARTED AUTOMATICALLY AND SUPPLIED POWER TO THE 2F, 2K AND 2H 4160 VOLT BUSES. THIS OCCURRED BECAUSE A WORKER INADVERTENTLY TRIPPED OPEN THE FEEDER BREAKER BETWEEN THE STARTUP TRANSFORMER (OFF-SITE POWER) AND THE 2F BUS (WHICH PROVIDES POWER TO THE 2K AND 2H BUSES). SUBSEQUENTLY, THE POWER SUPPLY FROM THE STARTUP TRANSFORMER WAS REESTABLISHED AND THE DIESEL GENERATORS WERE SHUT DOWN. THIS EVENT WAS CAUSED BY PERSONNEL ERROR. A WORKER ACCIDENTALLY ALLOWED THE TOOL WITH WHICH HE WAS WORKING TO CAUSE A SHORT CIRCUIT. THIS RESULTED IN TRIPPING THE BREAKER PROVIDING OFF-SITE POWER TO THE 2F 4160 VOLT BUS. TO PREVENT RECURRENCE, THE APPROPRIATE PERSONNEL HAVE BEEN COUNSELED CONCERNING THIS EVENT.

[83] FARLEY 2 DOCKET 50-364 LER 87-004
STEAM GENERATOR TUBE DEGRADATION.
EVENT DATE: 111387 REPORT DATE: 112487 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 207271) THE FOLLOWING REPORT IS BEING SUBMITTED IN ACCORDANCE WITH TECHNICAL SPECIFICATION 4.4.6.5.A AND C. PRIOR TO THE FIFTH REFUELING OUTAGE OF FARLEY UNIT 2, ALABAMA POWER COMPANY DEVELOPED AN EDDY CURRENT INSPECTION PLAN TO INSPECT ALL NON- PLUGGED TUBES IN ALL THREE STEAM GENERATORS. THIS INSPECTION WAS TO COVER THE FULL LENGTH OF TUBES ABOVE ROW 2. ROW 2 WAS TO BE INSPECTED FROM THE HOT LEG TUBE END TO THE TOP SUPPORT PLATE ON THE COLD LEG. ALL ROW 1 TUBES ARE PLUGGED. DURING THE COURSE OF THE INSPECTION, THE PROGRAM WAS EXPANDED TO INCLUDE THE COLD LEG OF ROW 2 TUBES. ALL OF THESE INSPECTIONS WERE DONE WITH THE BOBBIN COIL PROBE. PROBING PERFORMED IN A SAMPLE NUMBER OF TUBES USING THE ROTATING PANCAKE COIL (RPC) CONFIRMED THAT THE INDICATIONS FOUND AT TUBESHEET AND

SUPPORT PLATE LOCATIONS WERE CLASSIFIED CONSISTENT WITH THOSE FOUND AT THE LAST REFUELING OUTAGE (REFERENCE LER 86-004-00). NO TUBES WERE REPAIRED, 109 TUBES WERE PLUGGED AND 91 TUBES WERE DESIGNATED F*. INVESTIGATIONS AND EVALUATIONS PERFORMED IDENTIFIED FOUR AREAS WHERE TUBE DEGRADATION WAS OBSERVED: ANTIVIBRATION BAR (AVB) WEAR, PRIMARY WATER STRESS CORROSION CRACKING (PWSCC) IN THE TUBESHEET AREA, OD STRESS CORROSION CRACKING AT SUPPORT PLATES, AND RANDOM NONCORRELATABLE DEGRADATION. THESE ARE SIMILAR TO THE MECHANISMS REPORTED IN LER 86-004-00.

[84] FARLEY 2 DOCKET 50-364 LER 87-006
PERSONNEL ERROR CAUSES ACTUATION OF ENGINEERED SAFETY FEATURE EQUIPMENT.
EVENT DATE: 111587 REPORT DATE: 120987 NSSS: WE TYPE: PWR

(NSIC 207373) AT 2034 ON 11-15-87, DURING A REFUELING OUTAGE, AN ACTUATION OF ENGINEERED SAFETY FEATURE EQUIPMENT OCCURRED WHEN THE 1-2A DIESEL GENERATOR STARTED AUTOMATICALLY AND SUPPLIED POWER TO THE 2F AND 2K 416U VOLT BUSES. THIS OCCURRED BECAUSE A WORKER INADVERTENTLY TRIPPED OPEN THE FEEDER BREAKER BETWEEN THE STARTUP TRANSFORMER (OFF-SITE POWER) AND THE 2F BUS (WHICH PROVIDES POWER TO THE 2K BUS). SUBSEQUENTLY, THE POWER SUPPLY FROM THE STARTUP TRANSFORMER WAS REESTABLISHED AND THE DIESEL GENERATOR WAS SHUT DOWN. THIS EVENT WAS CAUSED BY PERSONNEL ERROR. JUMPERS WERE BEING PLACED IN PREPARATION FOR A SURVEILLANCE TEST. A WORKER ACCIDENTALLY ALLOWED A JUMPER TO CAUSE AN ARC OR SHORT CIRCUIT. THIS RESULTED IN TRIPPING THE BREAKER PROVIDING OFF-SITE POWER TO THE 2F 4160 VOLT BUS. TO PREVENT RECURRENCE, THE APPROPRIATE PERSONNEL HAVE BEEN COUNSELED CONCERNING THIS EVENT.

[85] FARLEY 2 DOCKET 50-364 LER 87-007
FIRE BARRIER PENETRATION NON-FUNCTIONAL FOR MORE THAN SEVEN DAYS.
EVENT DATE: 112387 REPORT DATE: 122287 NSSS: WE TYPE: PWR

(NSIC 207453) AT APPROXIMATELY 2330 ON 11-23-87, WHILE REVIEWING DOCUMENTATION ON NON-FUNCTIONAL FIRE BARRIER PENETRATIONS, IT WAS REALIZED THAT FIRE BARRIER PENETRATION 03-139-33 HAD BEEN NON-FUNCTIONAL FOR GREATER THAN SEVEN DAYS. TECHNICAL SPECIFICATION 3.7.12 REQUIRES THAT A SPECIAL REPORT BE SUBMITTED IF A NON-FUNCTIONAL PENETRATION IS NOT RETURNED TO FUNCTIONAL STATUS WITHIN SEVEN DAYS. THEREFORE, THIS SPECIAL REPORT IS BEING SUBMITTED. AT 1315 ON 11-16-87, A SOUND POWERED PHONE CORD WAS PLACED THROUGH THE FIRE DAMPER IN THIS PENETRATION TO FACILITATE VALVE LINEUPS FOR A CONTAINMENT INTEGRATED LEAK TEST. THE PHONE CORD WAS REMOVED AT 2345 ON 11-23-87 THUS RESTORING THE FIRE BARRIER PENETRATION TO FUNCTIONAL STATUS. TECHNICAL SPECIFICATION ACTION STATEMENT REQUIREMENTS WERE MET FOR THE NON-FUNCTIONAL FIRE BARRIER DURING THE ENTIRE TIME IT WAS NON-FUNCTIONAL.

[86] FARLEY 2 DOCKET 50-364 LER 87-008
OPENING OF REACTOR COOLANT SYSTEM PRESSURE RELIEF VALVE DUE TO PRESSURE PULSE.
EVENT DATE: 112787 REPORT DATE: 122?87 NSSS: WE TYPE: PWR
VENDOR: CROSBY VALVE & GAGE CO.

(NSIC 207454) THIS SPECIAL REPORT IS BEING SUBMITTED IN ACCORDANCE WITH TECHNICAL SPECIFICATION 3.4.10.3. AT 0445 ON 11-27-87, DURING A REFUELING OUTAGE, A RESIDUAL HEAT REMOVAL (RHR) LOOP SUCTION PRESSURE RELIEF VALVE OPENED WHEN ONE OF TWO SERIES RHR CONTAINMENT SUMP SUCTION VALVES IN THE SAME TRAIN (MOV 8812A) WAS STROKED. REACTOR COOLANT SYSTEM (RCS) PRESSURE AND PRESSURIZER LEVEL DECREASED. THE LEVEL AND PRESSURE IN THE PRESSURIZER RELIEF TANK (PRT) (TO WHICH THE RHR RELIEF VALVE RELIEVES) INCREASED AND THE PRT RUPTURE DISK RUPTURED. BASED ON THESE INDICATIONS, IT WAS BELIEVED THAT THE RELIEF VALVE MAY HAVE STUCK IN THE OPEN POSITION. TO ISOLATE THE RELIEF VALVE, THE OPERATORS STOPPED THE RHR PUMP IN THE AFFECTED TRAIN AND CLOSED THE LOOP SUCTION ISOLATION VALVES. THIS EVENT

WAS APPARENTLY CAUSED BY A LOCALIZED PRESSURE PULSE CREATED WHEN MOV 8812A WAS OPENED. THE SECTION OF PIPE BETWEEN THE TWO CONTAINMENT SUMP SUCTION ISOLATION VALVES HAD BEEN DRAINED TO PERFORM A LOCAL LEAK RATE TEST AND HAD NOT BEEN REFILLED. APPARENTLY, THE FILLING OF THIS SPACE, WHICH OCCURRED WHEN MOV 8812A WAS OPENED, CAUSED A LOCALIZED PRESSURE PULSE WHICH RESULTED IN THE LIFTING OF THE RELIEF VALVE. TO PREVENT RECURRENCE, THE APPLICABLE PROCEDURES WILL BE CHANGED TO ENSURE THAT LINES DRAINED FOR LOCAL LEAK RATE TESTING ARE SUBSEQUENTLY REFILLED.

[87] FARLEY 2 DOCKET 50-364 LER 87-009
PERSONNEL ERROR LEADS TO REACTOR TRIP DUE TO LOW STEAM GENERATOR LEVEL COINCIDENT WITH FEEDWATER FLOW LESS THAN STEAM FLOW SIGNAL.
EVENT DATE: 120387 REPORT DATE: 122387 NSSS: WE TYPE: PWR

(NSIC 207455) AT 2348 ON 12-3-87, WITH THE UNIT IN MODE 2 OPERATING AT 1.5% REACTOR POWER, THE REACTOR TRIPPED DUE TO LOW WATER LEVEL IN THE 2A STEAM GENERATOR COINCIDENT WITH FEEDWATER FLOW BEING LESS THAN STEAM FLOW. THE REACTOR TRIP WAS CAUSED BY PERSONNEL ERROR IN THAT THE OPERATOR ALLOWED THE STEAM GENERATOR LEVEL TO DECREASE TO THE TRIP SETPOINT. FOLLOWING THE REACTOR TRIP, THE UNIT WAS MAINTAINED IN A STABLE CONDITION. IN ORDER TO PREVENT RECURRENCE, THE OPERATOR INVOLVED HAS BEEN COUNSELED.

[88] FERMI 2 DOCKET 50-341 LER 87-042 REV 01
UPDATE ON CONTROL CENTER HEATING VENTILATION AND AIR CONDITIONING SYSTEM SHIFTS TO THE CHLORINE MODE WITH NO CHLORINE PRESENT.
EVENT DATE: 090287 REPORT DATE: 121487 NSSS: GE TYPE: BWR

(NSIC 207443) ON SEPTEMBER 2, 1987 AT 0557 HOURS, THE CONTROL CENTER HEATING, VENTILATION AND AIR CONDITIONING SYSTEM SHIFTED FROM NORMAL OPERATION TO THE CHLORINE MODE. THE ACTUATION OF THIS ENGINEERED SAFETY FEATURE MOST PROBABLY WAS CAUSED BY THE USE OF A TWO-WAY RADIO ON THE FOURTH FLOOR OF THE AUXILIARY BUILDING. IT IS BELIEVED THE TWO-WAY RADIO AFFECTED THE CHLORINE DETECTOR ON THE FIFTH FLOOR OF THE AUXILIARY BUILDING; CAUSING IT TO ACTUATE. A TEST WAS CONDUCTED WITH THE SUSPECT TWO-WAY RADIO TO DETERMINE IF IT COULD ACTUATE THE CHLORINE DETECTOR FROM THE FOURTH FLOOR OF THE AUXILIARY BUILDING. HOWEVER, THIS EVENT WAS NOT ABLE TO BE DUPLICATED. FURTHER TESTING WAS PERFORMED WITH VARIOUS RADIOS, BUT THE EVENT DID NOT RECUR.

[89] FERMI 2 DOCKET 50-341 LER 87-051 REV 02
UPDATE ON DIFFERENTIAL PRESSURE ACTUATION OF THE EMERGENCY EQUIPMENT COOLING WATER DUE TO A DESIGN DEFICIENCY.
EVENT DATE: 092987 REPORT DATE: 122987 NSSS: GE TYPE: BWR

(NSIC 207444) ON SEPTEMBER 29, 1987 AT 0653 HOURS, DIVISION I OF THE EMERGENCY EQUIPMENT COOLING WATER (EECW) AND EMERGENCY EQUIPMENT SERVICE WATER (EESW) SYSTEMS ACTUATED ON A LOW DIFFERENTIAL PRESSURE SIGNAL SENSED IN THE REACTOR BUILDING CLOSED COOLING WATER (RBCCW) SYSTEM. THIS OCCURRED WHEN A THIRD RBCCW PUMP WAS PUT INTO SERVICE IN PREPARATION FOR SWITCHING PUMPS. A DESIGN DEFICIENCY WAS IDENTIFIED IN THE RBCCW SYSTEM. FOR A SHORT PERIOD OF TIME WHEN THE THIRD RBCCW PUMP IS PUT INTO SERVICE, A SYSTEM PRESSURE TRANSIENT OCCURS. THE TRANSIENT IS OF SUFFICIENT DURATION TO CAUSE AN AUTOMATIC ACTUATION OF THE DIVISION I EECW SYSTEM. THE TIME DELAY FOR THE EECW/EESW SYSTEMS LOW DIFFERENTIAL PRESSURE SWITCHES WILL BE LENGTHENED. A LONGER TIME DELAY WILL PREVENT THE EECW SYSTEM FROM ACTUATING DUE TO THE PRESSURE TRANSIENT WHICH OCCURS WHEN A THIRD RBCCW PUMP IS PLACED IN SERVICE.

[90] FERMI 2 DOCKET 50-341 LER 87-048 REV 01
UPDATE ON CHANNEL CHECK NOT PERFORMED FOR DRYWELL INSTRUMENTS BECAUSE OF AN INADEQUATE PROCEDURE.
EVENT DATE: 100887 REPORT DATE: 113087 NSSS: GE TYPE: BWR

(NSIC 207368) ON OCTOBER 8, 1987 AT 1400 HOURS, IT WAS DISCOVERED THAT THE TECHNICAL SPECIFICATION REQUIREMENT TO PERFORM A CHANNEL CHECK AT LEAST ONCE PER 12 HOURS FOR REACTOR PROTECTION SYSTEM (RPS) DRYWELL HIGH PRESSURE INSTRUMENTATION WAS NOT INCLUDED IN ANY SURVEILLANCE PROCEDURE. THIS EVENT WAS CAUSED BY AN INCOMPLETE SURVEILLANCE PROCEDURE. THE "SHIFTLY, DAILY, WEEKLY AND SITUATION REQUIRED SURVEILLANCES" PROCEDURE DID NOT INCLUDE THE CHANNEL CHECK FOR RPS DRYWELL HIGH PRESSURE INSTRUMENTS. THE CORRECTIVE ACTION WAS TO REVISE THE SURVEILLANCE PROCEDURE TO INCLUDE THIS REQUIREMENT. ALL TECHNICAL SPECIFICATION SURVEILLANCE PROCEDURES ARE SCHEDULED TO BE REVIEWED BY MARCH 31, 1988 UNDER THE SURVEILLANCE PROCEDURE ENHANCEMENT PROGRAM.

[91] FERMI 2 DOCKET 50-341 LER 87-052
POTENTIAL FOR DEGRADING PRIMARY CONTAINMENT INTEGRITY THROUGH A RADIATION MONITORING SKID.
EVENT DATE: 101787 REPORT DATE: 111487 NSSS: GE TYPE: BWR

(NSIC 207225) ON OCTOBER 17, 1987 DURING MODIFICATION WORK ON THE PRIMARY CONTAINMENT RADIATION MONITORING SYSTEM (PCRMS) SKID, IT WAS DETERMINED THAT A VIOLATION OF PRIMARY CONTAINMENT INTEGRITY MAY HAVE OCCURRED. THE PCRMS WAS ISOLATED BY SINGLE AUTOMATIC ISOLATION VALVES BETWEEN THE NON-QUALIFIED SKID AND A CLOSED OUTSIDE CONTAINMENT SYSTEM. THESE VALVES ARE NOT IDENTIFIED AS CONTAINMENT ISOLATIONS IN EITHER THE TECHNICAL SPECIFICATIONS OR THE UPDATED FINAL SAFETY ANALYSIS REPORT. A LOCAL LEAK RATE TEST WAS PERFORMED ON EACH OF THE ISOLATION VALVES WHICH VERIFIED THEIR ABILITY TO MAINTAIN PRIMARY CONTAINMENT INTEGRITY. A REQUEST FOR TEMPORARY EXEMPTION FROM THE GENERAL DESIGN CRITERIA WAS REQUESTED AND HAS BEEN VERBALLY GRANTED. THE APPROPRIATE EMERGENCY OPERATING PROCEDURES HAVE BEEN REVISED TO DIRECT MANUAL ISOLATION OF THE SKID WHEN NECESSARY. COMPENSATORY TESTING WILL BE PERFORMED UNTIL THE INSTALLATION IS UPGRADED. A MODIFICATION WILL BE MADE IN ORDER TO ACHIEVE CONFORMANCE WITH THE GENERAL DESIGN CRITERIA.

[92] FERMI 2 DOCKET 50-341 LER 87-053
EXCEEDING LIMITING CONDITION FOR OPERATION FOR PRIMARY CONTAINMENT HIGH PRESSURE CHANNEL CAUSES ENTRY INTO TECHNICAL SPECIFICATION 3.0.3.
EVENT DATE: 102487 REPORT DATE: 112387 NSSS: GE TYPE: BWR

(NSIC 207233) ON OCTOBER 24, 1987, DURING THE PERFORMANCE OF THE INSTRUMENT AND CONTROLS (I&C) SURVEILLANCE PROCEDURE 44.020.015 "NSSSS DRYWELL PRESSURE DIVISION I, CHANNEL A RESPONSE TIME TEST," THE TWO HOUR TECHNICAL SPECIFICATION TIME CONSTRAINT FOR ALLOWING THE INSTRUMENT CHANNEL TO BE INOPERABLE WAS EXCEEDED. THE TIME CONSTRAINT WAS VIOLATED BECAUSE THERE WAS A FAILURE BY THE INVOLVED PERSONNEL TO RECOGNIZE THE DISCREPANCY BETWEEN THE TWO HOUR TECH SPEC LIMIT AND THE ESTIMATED SIX TO EIGHT HOURS NEEDED TO PERFORM THE TEST. A CHECKLIST WAS DEVELOPED AS A GUIDE FOR THE PERFORMANCE OF SURVEILLANCE PROCEDURE 44.020.015. THE CHECKLIST WILL BE PRESENTED TO AND DISCUSSED WITH THE NUCLEAR SHIFT SUPERVISOR/NUCLEAR ASSISTANT SHIFT SUPERVISOR AND THE I&C FOREMAN PRIOR TO THE PERFORMANCE OF SURVEILLANCES FOR APPLICATIONS, OPERATING MODES AND FREQUENCIES OTHER THAN THE ONES NORMALLY EXPECTED.

[93] FITZPATRICK DOCKET 50-333 LER 87-016
HIGH STEAM FLOW ISOLATION OF THE REACTOR CORE ISOLATION COOLING SYSTEM DUE TO FAILURE OF OPERATOR TO FOLLOW PRESCRIBED SEQUENCE OF SURVEILLANCE PROCEDURE.
EVENT DATE: 091687 REPORT DATE: 101387 NSSS: GE TYPE: BWR

(NSIC 206621) ON 9/16/87, AT 2330, WITH THE PLANT OPERATING AT 100% POWER, AN AUTOMATIC ISOLATION OF THE REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM (BN) OCCURRED. THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM (BJ) WAS INOPERABLE AND RCIC TESTING WAS IN PROGRESS AS A RESULT OF HPCI BEING INOPERABLE. WHILE PERFORMING THE RCIC PUMP AND VALVE OPERABILITY TEST PROCEDURE, THE OPERATOR INADVERTENTLY PASSED OVER A STEP TO OPEN THE RCIC INBOARD CONTAINMENT ISOLATION STEAM SUPPLY VALVE. PRIOR TO TESTING THE RCIC PUMP AND TURBINE THE OPERATOR REALIZED THE VALVE WAS SHUT AND IMMEDIATELY OPENED THE ISOLATION VALVE. THE SURGE OF STEAM PAST THE HIGH STEAM FLOW ELBOW TAPS CAUSED AN ISOLATION OF THE RCIC SYSTEM. THE ISOLATION WAS RESET AND THE RCIC SYSTEM WAS VERIFIED TO BE OPERABLE. THE NRC WAS NOTIFIED IN ACCORDANCE WITH 10 CFR 50.72. THE OPERATOR WAS COUNSELED ON THE REQUIREMENT TO FOLLOW PRESCRIBED PROCEDURE SEQUENCE. SIMILAR EVENTS HAVE BEEN REPORTED IN LERS 84-003, 84-004, 85-023 AND 86-019.

[94] FITZPATRICK DOCKET 50-333 LER 87-018
REACTOR TRIP FROM HIGH NEUTRON FLUX DUE TO ERRATIC OPERATION OF REACTOR WATER RECIRCULATION PUMP SPEED CONTROLLER.
EVENT DATE: 110887 REPORT DATE: 120787 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 207405) AT 1934 HOURS ON 1X1/8/87 WITH REACTOR THERMAL POWER AT 80% A REACTOR TRIP OCCURRED AS A RESULT OF AN AVERAGE POWER RANGE MONITOR (IG) HIGH FLUX TRIP. THE HIGH FLUX TRIP WAS INITIATED BY A SUDDEN REACTOR WATER RECIRCULATION SYSTEM (AD) PUMP SPEED INCREASE. DURING THE EVENT OPERATORS RESPONDED TO THE TRANSIENT BY UTILIZING APPROVED PLANT PROCEDURES. PLANT RESPONSE FOR THE EVENT WAS WITHIN THE BOUNDS OF TRANSIENT ANALYSIS AS DISCUSSED IN THE FINAL SAFETY ANALYSIS REPORT. THE ROOT CAUSE OF THE RECIRCULATION PUMP SPEED INCREASE IS BELIEVED TO HAVE BEEN A MALFUNCTION IN A PUMP SPEED CONTROLLER REMOTE SETPOINT/CASCADE SWITCH. THE MALFUNCTION APPEARS TO HAVE OCCURRED DUE TO THE AGE OF THE SWITCH. AFTER TROUBLESHOOTING OF THE SPEED CONTROL LOOP, THE CONTROLLER WAS REPLACED AND THE LOOP RETURNED TO NORMAL OPERATION. THE LER NUMBER OF A SIMILAR PREVIOUS EVENT IS LER 85-018-00.

[95] FT. CALHOUN 1 DOCKET 50-285 LER 87-033
WATER INTRUSION INTO AIR SYSTEM.
EVENT DATE: 070687 REPORT DATE: 112587 NSSS: CE TYPE: PWR

(NSIC 207192) AT APPROXIMATELY 1045 ON JULY 6, 1987, CLARIFIED WATER ENTERED THE INSTRUMENT AIR SYSTEM AT THE FORT CALHOUN STATION DURING A SURVEILLANCE TEST OF THE DIESEL GENERATOR ROOM DRY PIPE SPRINKLER SYSTEM. THE INTERFACING CHECK VALVES BETWEEN THE INSTRUMENT AIR (IA) AND FIRE PROTECTION (FP) SYSTEMS WERE PREVENTED FROM COMPLETELY CLOSING BY FOREIGN MATERIAL. ADDITIONALLY, INADEQUATE PROCEDURES AND INADEQUATE OPERATOR TRAINING ON THIS UNIQUE DRY PIPE VALVE CONTRIBUTED TO THIS EVENT. IMMEDIATE ACTIONS IN RESPONSE TO THIS EVENT INCLUDED ISOLATING THE WATER SUPPLY, MAINTAINING THE CHECK VALVES, PROPERLY RESETTING THE DRY PIPE VALVE, CONDUCTING BLOWDOWNS TO REMOVE THE WATER FROM THE IA SYSTEM, AND ASSIGNING ENGINEERS TO EVALUATE AND DEFINE FURTHER REQUIRED RESPONSE ACTIONS. THIS EVENT WAS A PRECURSOR TO FAILURE OF THE DG-2 EXHAUST DAMPER ON SEPTEMBER 23, 1987, WHICH IS ADDRESSED IN LER-87-025. SPECIFIC CORRECTIVE ACTIONS HAVE BEEN DEVELOPED AND ARE BEING IMPLEMENTED.

[96] FT. CALHOUN 1 DOCKET 50-285 LER 87-025 REV 01
UPDATE ON DG-2 SHUTDOWN ON HIGH COOLANT TEMPERATURE.
EVENT DATE: 092387 REPORT DATE: 121587 NSSS: CE TYPE: PWR

(NSIC 207361) ON SEPTEMBER 23, 1987, AT 0906 CDT, FOLLOWING REPAIR OF THE EXHAUST PIPE, DIESEL GENERATOR NO. 2 (DG-2) WAS STARTED AND LOADED PER OPERATING INSTRUCTION 01-DG-2 AS REQUIRED BY SURVEILLANCE TEST ST-ESF-6. APPROXIMATELY 14

MINUTES INTO THE TEST, DG-2 AUTOMATICALLY SHUTDOWN DUE TO HIGH COOLANT TEMPERATURE. INVESTIGATIONS REVEALED THAT THE AIR OPERATED EXHAUST DAMPER FOR THE DIESEL GENERATOR RADIATOR MAY NOT HAVE FULLY OPENED AUTOMATICALLY AS DESIGNED WHEN THE DIESEL WAS RUNNING, THUS RESTRICTING THE REQUIRED AIR FLOW THROUGH THE RADIATOR. THE CAUSE OF THE DAMPER MALFUNCTION WAS POSTULATED TO BE THE PRESENCE OF RESIDUE CAUSING THE PILOT VALVE THAT DIRECTS AIR FLOW TO SOMETIMES STICK. ON JULY 6, WATER WAS INTRODUCED INTO THE INSTRUMENT AIR SYSTEM DURING THE PERFORMANCE OF A SURVEILLANCE TEST ON THE FIRE PROTECTION SYSTEM DRY PIPE VALVE FOR THE DIESEL GENERATOR ROOMS. THE WATER INTRUSION WAS LIMITED TO THE AUXILIARY BUILDING AT OR BELOW ELEVATION 1025. AN EXTENSIVE PROGRAM WAS UNDERTAKEN (IN JULY) AND WAS REPEATED AS NECESSARY DURING THE MONTHS OF AUGUST AND SEPTEMBER TO BLOWDOWN AIR OPERATED DEVICES INCLUDING AIR OPERATED VALVES AND TO CYCLE THOSE VALVES AS ALLOWED DURING POWER OPERATION. AFTER THE TRIP OF DG-2, THE PILOT VALVE WAS INSPECTED AND CLEANED AND THE ACCUMULATOR DRAINED. SIMILAR ACTIONS WERE TAKEN FOR DG-1.

[97] FT. CALHOUN 1 DOCKET 50-285 LER 87-027 REV 01
UPDATE ON UNPLANNED ACTUATION OF CIAS.
EVENT DATE: 100887 REPORT DATE: 121187 NSSS: CE TYPE: PWR

(NSIC 207362) AN UNPLANNED ACTUATION OF THE CONTAINMENT ISOLATION ACTUATION SIGNAL (CIAS) OCCURRED AT 0850 HOURS ON OCTOBER 8, 1987, WHILE FORT CALHOUN STATION UNIT NO. 1 WAS OPERATING IN MODE 1 AT APPROXIMATELY 100 PERCENT POWER. ALL ENGINEERED SAFEGUARDS EQUIPMENT FUNCTIONED AS DESIGNED. ALL SYSTEMS WERE RESET TO NORMAL WITHIN THREE AND ONE HALF MINUTES. THE FOUR-HOUR REPORT AS MANDATED BY 10 CFR 50.72(B)(2)(II) WAS COMPLETED AT 0955 HOURS. DURING THE PERFORMANCE OF MONTHLY SURVEILLANCE TEST ST-ESF-2 F.2, CHANNEL "B" SAFETY INJECTION ACTUATION SIGNAL TEST, THE CHANNEL "B" PRESSURIZER PRESSURE LOW SIGNAL (PPLS) TEST SWITCH WAS INADVERTENTLY TURNED TO THE TEST POSITION PRIOR TO THE ISOLATION OF THE CHANNEL "B" CIAS, RESULTING IN "B" TRAIN CIAS ACTUATION. TO PREVENT FUTURE CIAS ACTUATIONS OF THIS TYPE, SURVEILLANCE TEST ST- ESF-2 HAS BEEN REVISED TO DIRECT PERSONNEL TO PLACE THE CONTAINMENT ISOLATION VALVES OVERRIDE SWITCH IN THE TEST POSITION EARLIER IN THE PROCEDURE.

[98] FT. CALHOUN 1 DOCKET 50-285 LER 87-029
FAILURE TO CONDUCT SURVEILLANCE TEST ST-DC-4 SECTION F.3.
EVENT DATE: 110187 REPORT DATE: 120187 NSSS: CE TYPE: PWR

(NSIC 207259) DURING REVIEW OF SURVEILLANCE TEST DATA, IT WAS DISCOVERED THAT NO DOCUMENTATION EXISTED FOR CONTAINMENT EMERGENCY LIGHTING SURVEILLANCE TEST ST-DC-4 F.3 FOR 1986. PER THE REQUIREMENTS OF TECHNICAL SPECIFICATION 3.7(3), THE CORRECT FUNCTIONING OF THE EMERGENCY LIGHTING SYSTEM SHALL BE VERIFIED AT LEAST ONCE A YEAR. THE FAILURE TO CONDUCT THE TEST DURING 1986 WAS A VIOLATION OF THE TECHNICAL SPECIFICATIONS. CORRECTIVE ACTION HAS BEEN INITIATED TO PREVENT RECURRENCE OF THIS PROBLEM. THE IMPORTANCE OF CONDUCTING SURVEILLANCE TESTS WHEN SCHEDULED, HAS BEEN REITERATED TO STATION PERSONNEL. IN THE FUTURE, ST- DC-4 F.3 WILL CONTINUE TO BE INCLUDED FOR COMPLETION DURING REFUELING OUTAGES. IN THOSE YEARS WHERE NO REFUELING OUTAGE IS SCHEDULED, ST-DC-4 F.3, WILL BE INCLUDED ON THE SURVEILLANCE TESTS TO BE PERFORMED DURING A FORCED UNIT OUTAGE. TO ENSURE THAT THE SURVEILLANCE TEST IS NOT OVERLOOKED AGAIN, ST-DC-4 F.3 WILL BE ISSUED ON THE DECEMBER SURVEILLANCE TEST SCHEDULE. FURTHER, OPPD PERSONNEL WILL ASSESS THE SAFETY IMPACT OF CHANGING THE SURVEILLANCE TEST INTERVAL TO REFUELING, RATHER THAN ANNUAL. IF IT IS DETERMINED NO SIGNIFICANT SAFETY CONCERN EXISTS, A REQUEST FOR TECHNICAL SPECIFICATION AMENDMENT WILL BE MADE.

[99] FT. CALHOUN 1 DOCKET 50-285 LER 87-036
INADVERTENT AUXILIARY FEEDWATER ACTUATION.
EVENT DATE: 111187 REPORT DATE: 121187 NSSS: CE TYPE: PWR

(NSIC 207363) ON NOVEMBER 11, 1987, WHILE PERFORMING ST-FW-3, THE OPERATOR INADVERTENTLY POSITIONED THE INCORRECT SWITCH IN OVERRIDE, SUBSEQUENTLY ENABLING THE TEST SIGNAL TO INITIATE THE AUXILIARY FEEDWATER ACTUATION SIGNAL. BOTH AUXILIARY FEEDWATER PUMPS STARTED, BUT NO AUXILIARY FEEDWATER WAS INJECTED TO THE STEAM GENERATORS. SURVEILLANCE TEST ST-FW-3, AUTO INITIATION OF AUXILIARY FEEDWATER, IS USED TO SATISFY SURVEILLANCE REQUIREMENTS OF TECHNICAL SPECIFICATION 3.1, ENSURING THE OPERABILITY OF THE AUTO INITIATION OF THE AUXILIARY FEEDWATER SYSTEM. INVESTIGATIONS OF THE EVENT REVEALED THAT THE SIMILARITY IN LABELING BETWEEN SWITCHES WAS THE OVERRIDING FACTOR OF OPERATOR ERROR. TO PREVENT A POSSIBLE RECURRENCE, LABELING CHANGES ALONG WITH REVISIONS TO ST-FW-3 WILL BE IMPLEMENTED TO REDUCE THE PROBABILITY OF HUMAN ERROR ASSOCIATED WITH THIS TEST.

[100] FT. CALHOUN 1 DOCKET 50-285 LER 87-037
DIESEL GENERATOR SURVEILLANCE TEST NOT IN CONFORMANCE WITH TECHNICAL SPECIFICATIONS.
EVENT DATE: 111187 REPORT DATE: 122387 NSSS: CE TYPE: PWR

(NSIC 207433) AS REVISED IN AMENDMENT NO. 111, TECHNICAL SPECIFICATION 3.7(1)(A)(II) REQUIRES THAT WITH THE DIESEL RUNNING AT RATED SPEED AND VOLTAGE, THE GENERATOR SHALL BE SYNCHRONIZED WITH THE 4.16 KV BUS AND THE DIESEL BREAKER MANUALLY CLOSED FROM THE ELECTRICAL CONTROL BOARD. THE GENERATOR SHALL THEN BE LOADED TO AT LEAST THE CONTINUOUS KW RATED AND RUN FOR AT LEAST 60 MINUTES BEFORE BEING OFF-LOADED AND THE DIESEL BREAKER TRIPPED. ON NOVEMBER 25, 1987, SURVEILLANCE TEST ST-ESF-6 F.2 WAS SCHEDULED TO BE PERFORMED ON DIESEL GENERATOR NO. 2 WHEN IT WAS DISCOVERED THAT THE PROCEDURE HAD NOT BEEN UPDATED TO INCLUDE THE REQUIREMENTS OF TECHNICAL SPECIFICATION AMENDMENT NO. 111. INVESTIGATION OF SURVEILLANCE TESTS RUN SINCE THE TECHNICAL SPECIFICATION AMENDMENT NO. 111 BECAME EFFECTIVE REVEALED THAT THE DIESEL GENERATOR OPERABILITY SURVEILLANCE TEST ST-ESF-6 F.2, FOR DIESEL GENERATOR NO. 1, WAS RUN ON NOVEMBER 11, 1987. DURING THE TEST THE DIESEL HAD RUN LOADED FOR ONE HOUR, BUT LOADED TO THE CONTINUOUS KW RATING FOR ONLY 21 MINUTES, AS ALLOWED BY THE TECHNICAL SPECIFICATIONS PRIOR TO AMENDMENT NO. 111 BEING ISSUED. THIS WAS CONTRARY TO TECHNICAL SPECIFICATION 3.7(1)(A)(II), IN EFFECT AT THE TIME OF THE TEST.

[101] FT. ST. VRAIN DOCKET 50-267 LER 87-017 REV 01
UPDATE ON LOOP II REHEATER PENETRATION SAMPLES NOT COLLECTED.
EVENT DATE: 071087 REPORT DATE: 111387 NSSS: GA TYPE: HTGR

(NSIC 207519) CAUSE - PERSONNEL ERROR AND DEFICIENT PROCEDURES. PER LCO 4.2.9, WHENEVER EITHER OR BOTH STEAM GENERATOR PENETRATION INTERSPACE GROUPS III AND IV ARE MAINTAINED BELOW PRIMARY COOLANT PRESSURE, THE PENETRATION WILL BE MONITORED FOR GROSS ACTIVITY BY THE CORRESPONDING LOOP ACTIVITY MONITOR, RIS-2263 OR RIS-2264. SHOULD ACTIVITY MONITOR RIS-2263 OR RIS-2264 BECOME INOPERABLE WHILE THE CORRESPONDING LOOP STEAM GENERATOR PENETRATION INTERSPACE IS BEING OPERATED BELOW PRIMARY COOLANT PRESSURE, GRAB SAMPLES WILL BE TAKEN ONCE EVERY EIGHT HOURS AND ANALYZED WITHIN TWENTY-FOUR HOURS. ON 7/10/87 AND ON 7/11/87, WITH THE REACTOR OPERATING AT 57% POWER, THE GROUP III LOOP II STEAM GENERATOR PENETRATION INTERSPACE OPERATING BELOW PRIMARY COOLANT PRESSURE, AND RIS-2264 INOPERABLE, GRAB SAMPLES WERE NOT COLLECTED FROM THE LOOP II STEAM GENERATOR PENETRATION INTERSPACE AS REQUIRED PER FORT ST. VRAIN TECH SPEC LOC 4.2.9. THIS INCIDENT OCCURRED AS A RESULT OF A PERSONNEL ERROR IN COMBINATION WITH A DEFICIENT PROCEDURE. PROCEDURE CHANGES HAVE BEEN INITIATED TO ENSURE BETTER COMMUNICATION BETWEEN THE OPERATIONS DEPARTMENT AND THE HEALTH PHYSICS DEPARTMENT CONCERNING TECH SPEC REQUIRED SAMPLING.

[102] FT. ST. VRAIN DOCKET 50-267 LER 87-023
FIRE IN TURBINE BUILDING WHEN HYDRAULIC OIL LEAKED ONTO HIGH TEMPERATURE STEAM LINE.
EVENT DATE: 100287 REPORT DATE: 110187 NSSS: GA TYPE: HTGR

(NSIC 207261) CAUSE - ROOT CAUSE ANALYSIS IN PROGRESS. ON FRIDAY, OCTOBER 2, 1987, AT 2359 HOURS, WITH THE REACTOR AT 27% POWER, A FIRE WAS IDENTIFIED IN THE TURBINE BUILDING. IT IS NOW POSTULATED THAT HYDRAULIC OIL LEAKED FROM HYDRAULIC VALVE HV-2292 ONTO AN EXPOSED HIGH TEMPERATURE STEAM LINE RESULTING IN OPEN FLAMES AND HEAVY SMOKE. A MANUAL SCRAM WAS INITIATED APPROXIMATELY TEN MINUTES LATER DUE TO PROBLEMS WITH NORMAL PRIMARY AND SECONDARY FLOW CONTROL. THE PLANT FIRE BRIGADE EXTINGUISHED THE FIRE IN LESS THAN TWENTY MINUTES WHILE THE REACTOR SIDE EQUIPMENT OPERATOR ISOLATED THE HYDRAULIC OIL SUPPLY TO THE LEAKING VALVE. OUTSIDE ASSISTANCE WAS CALLED AND ARRIVED IN TIME TO PROVIDE FIRE WATCHES TO PREVENT REFLASH FIRES. THE PLANT'S EMERGENCY RESPONSE PLAN WAS PUT INTO EFFECT AND THE EMERGENCY STATIONS STAFFED TO COORDINATE A SAFE PLANT SHUTDOWN AND TO COORDINATE THE FIRE RECOVERY ACTIVITIES. SPECIFIC TECHNICAL EVALUATIONS, INCLUDING ROOT CAUSE ANALYSIS, ARE IN PROGRESS TO DETERMINE THE SCOPE OF CORRECTIVE ACTIONS AND REPAIRS NECESSARY TO SUPPORT RESTART OF THE PLANT. THIS REPORT WILL BE SUPPLEMENTED TO IDENTIFY THE REQUIRED CORRECTIVE ACTIONS ONCE THESE EVALUATIONS ARE COMPLETED.

[103] FT. ST. VRAIN DOCKET 50-267 LER 87-024
PLANT PROTECTION SYSTEM REACTOR SCRAM LOGIC ACTUATED 11 TIMES ON NEUTRON FLUX RATE OF CHANGE HIGH.
EVENT DATE: 100987 REPORT DATE: 110887 NSSS: GA TYPE: HTGR

(NSIC 207262) CAUSE - CHATTERING RELAY. ON OCTOBER 9, 10, 11, AND 14, 1987, THE PLANT PROTECTIVE SYSTEM (PPS) REACTOR SCRAM LOGIC AND ALARM CIRCUITRY WAS ACTUATED A TOTAL OF 11 TIMES ON WIDE RANGE CHANNELS (WRC) III AND V NEUTRON FLUX RATE OF CHANGE HIGH. THE ACTUATIONS WERE CAUSED BY ELECTRICAL NOISE INDUCTION IN THE WIDE RANGE CHANNELS. THE SOURCE OF ELECTRICAL NOISE WAS REPEATED SWITCHING ("CHATTERING") OF RELAY FS-11263-X, WHICH IN TURN WAS CAUSED BY AN INTERMITTENT SHORT-TO-GROUND IN THE ASSOCIATED FLOW SWITCH FS-11263. SINCE THE REACTOR REMAINED SHUTDOWN WITH ALL 37 CONTROL ROD PAIRS FULLY INSERTED IN THE CORE AND THEIR POWER SUPPLY BREAKERS OPEN DURING THE TIME PERIOD THAT THESE ACTUATIONS OCCURRED, THE ACTUATIONS AFFECTED THE LOGIC AND ALARM CIRCUITRY ONLY. NO CONTROL ROD MOVEMENT OCCURRED. AS A TEMPORARY SOLUTION, THE LEADS TO THE COIL OF FS-11263-X HAVE BEEN LIFTED TO PREVENT RELAY CHATTERING UNTIL SUCH TIME AS FS-11263 IS REPAIRED. FLOW SWITCH FS-11263 WILL BE REPAIRED OR REPLACED AS NECESSARY TO CORRECT THE INTERMITTENT SHORT-TO-GROUND PRIOR TO EXCEEDING 100 PSIA IN THE PRESTRESSED CONCRETE REACTOR VESSEL (PCRV). TO FURTHER ENSURE THAT NORMAL SWITCHING OF THE RELAY DOES NOT INDUCE ELECTRICAL NOISE IN THE WIDE RANGE CHANNELS, A NOISE SUPPRESSION DEVICE WILL BE INSTALLED ON THE RELAY.

[104] FT. ST. VRAIN DOCKET 50-267 LER 87-025
OFF-SITE ELECTRICAL POWER LOST WHEN RAT DELUGE ACTUATED.
EVENT DATE: 103087 REPORT DATE: 113087 NSSS: GA TYPE: HTGR

(NSIC 207518) CAUSE - PERSONNEL ERROR. AT 2216 HOURS ON 10/30/87, WITH THE REACTOR SHUTDOWN, THE RESERVE AUXILIARY TRANSFORMER (RAT) FIREWATER DELUGE SYSTEM WAS INADVERTENTLY ACTUATED. THIS ACTUATION CAUSED THE RAT FEED BREAKERS TO OPEN (BY DESIGN). SINCE THE RAT WAS SUPPLYING PLANT HOUSE POWER AND THE UNIT AUXILIARY TRANSFER (UAT) WAS LINKED TO THE MAIN TURBINE GENERATOR AND THUS NOT IMMEDIATELY AVAILABLE (APPROXIMATELY TWO HOURS FOR LINK REMOVAL), LOSS OF THE RAT RESULTED IN A LOSS OF OUTSIDE ELECTRICAL POWER, AN INTERRUPTION OF FORCED CORE COOLING, AND AN AUTOMATIC START OF THE EMERGENCY DIESEL GENERATORS. AT 2250 HOURS, THE RAT DELUGE SYSTEM WAS RESET, THE RAT POWER CIRCUIT BREAKERS WERE CLOSED, AND OFFSITE HOUSE POWER WAS RETURNED TO NORMAL. AT 2350 HOURS, SECONDARY

FLOW WAS RESTORED AND AT 0115 HOURS ON 10/31/87, PRIMARY COOLANT FLOW WAS RESTORED. A JUNCTION BOX DOOR DOCUMENT HOLDER PHYSICALLY CONTACTED THE DELUGE CONTROL RELAY BAR WHEN CONTRACTOR ELECTRICIANS CLOSED THE JUNCTION BOX DOOR WHEN REPLACING CABLES. THIS ACTUATED THE RAT DELUGE SYSTEM. THIS DOCUMENT HOLDER HAS BEEN REMOVED.

[105] FT. ST. VRAIN DOCKET 50-267 LER 87-026
LOSS OF POWER TO INSTRUMENT BUS RESULTS REACTOR SCRAM LOGIC ACTUATIONS.
EVENT DATE: 111187 REPORT DATE: 121187 NSSS: GA TYPE: HTGR

(NSIC 207517) CAUSE - PERSONNEL ERROR. ON NOV. 11, 1987, SEVERAL REACTOR SCRAM ACTUATIONS OCCURRED UPON LOSS OF POWER TO INSTRUMENT BUS 2. INVERTER 1B WAS TRANSFERRING TO BACKUP POWER DUE TO VOLTAGE SWINGS ON 125 VDC BUS 1B, CAUSED BY MALFUNCTIONING OF BATTERY CHARGER 1B. THE BACKUP POWER SUPPLY BREAKER THEN TRIPPED ON OVERCURRENT, RESULTING IN LOSS OF POWER TO INSTRUMENT BUS 2. THIS CAUSED LOSS OF POWER TO STARTUP CHANNEL II. THE BISTABLE TRIP MODULES IN THE STARTUP CHANNEL FAILED IN THE SCRAM DIRECTION, AS DESIGNED. TRIP OF STARTUP CHANNEL II COMPLETED THE 1 OF 2 STARTUP CHANNEL LOGIC, AND ACTUATED THE REACTOR SCRAM LOGIC AND ALARM CIRCUITRY. THE BACKUP POWER SUPPLY BREAKER TRIP SETTING WAS TOO LOW TO SUSTAIN THE LOAD ON THE BUS DUE TO RECENTLY INCREASING BUS LOAD WITHOUT RAISING THE BREAKER TRIP SETTING. INSTRUMENTATION FOR THE STEAM LINE RUPTURE DETECTION/ISOLATION SYSTEM WAS ADDED TO THE BUS. BUT THE NEED TO RAISE THE BREAKER TRIP SETTING WAS APPARENTLY OVERLOOKED. THE TRANSFORMER IN BATTERY CHARGER 1B HAS BEEN REPLACED. THE OVERCURRENT TRIP SETTING OF THE BACKUP POWER SUPPLY BREAKER HAS BEEN RAISED SO THAT IT CAN SUSTAIN THE CURRENT LOAD ON THE BUS.

[106] GRAND GULF 1 DOCKET 50-416 LER 87-009 REV 01
UPDATE ON REACTOR SCRAM DUE TO RELAY FAILURE.
EVENT DATE: 062987 REPORT DATE: 122387 NSSS: GE TYPE: BWR
VENDOR: AGASTAT RELAY CO.

(NSIC 207473) ON JUNE 29, 1987 AGASTAT RELAY N62-R33 SUSTAINED AN INTERMITTENT FAILURE WHICH CAUSED THE MAIN STEAM INLET VALVE (N62-F001B) TO STEAM JET AIR EJECTOR (SJAE) "B" TO CLOSE. THE CLOSURE OF THIS VALVE CAUSED A LOSS OF CONDENSER VACUUM RESULTING IN A MAIN TURBINE TRIP AND REACTOR SCRAM. THE CLOSURE OF VALVE N62-F001B PREVENTED THE SJAE FROM REMOVING NON- CONDENSABLE GASES FROM THE MAIN CONDENSER. IN ADDITION, MOTOR OPERATED VALVE N62-F003B FAILED TO CLOSE ALLOWING REVERSE FLOW THROUGH THE OFF- GAS SYSTEM BACK INTO THE CONDENSER. THESE TWO FAILURES COMBINED TO DECREASE MAIN CONDENSER VACUUM RESULTING IN A MAIN TURBINE TRIP AND REACTOR SCRAM ON THE TURBINE STOP VALVE FAST CLOSURE SIGNAL. FOLLOWING THE SCRAM THE SJAES WERE SECURED WHICH CLOSED THE N62-F003B VALVE AND TERMINATED THE LOSS OF VACUUM EVENT. VACUUM STABILIZED AT APPROXIMATELY 21 INCHES MERCURY. DURING THE SCRAM RECOVERY, A DIVISION II GROUP 8 AUTOMATIC ISOLATION OCCURRED WHEN OPERATORS PREPARED TO PLACE THE REACTOR WATER CLEANUP (RWCU) SYSTEM INTO THE BLOWDOWN MODE OF OPERATION. THE RELAY WAS REPLACED. A MAINTENANCE WORK ORDER WAS INITIATED TO INVESTIGATE THE POTENTIAL PROBLEM WITH FLOW SWITCHES ASSOCIATED WITH LOW STEAM FLOW FROM THE SECOND STAGE AIR EJECTORS AND WILL BE COMPLETED DURING THE CURRENT PLANT OUTAGE.

[107] HATCH 2 DOCKET 50-366 LER 87-013
OUTSIDE AIR AND BLOWER EXHAUST CAUSE AIR TEMPERATURE DIFFERENTIAL AND ESF VALVE ISOLATION.
EVENT DATE: 101987 REPORT DATE: 111887 NSSS: GE TYPE: BWR

(NSIC 207209) ON 10/19/87 AT APPROXIMATELY 0957 CDT, PLANT PERSONNEL WERE PERFORMING A SURVEILLANCE ON THE MAIN STEAM ISOLATION VALVE (MSIV) LEAKAGE CONTROL SYSTEM (EIIS CODE IJ) WHEN THE REACTOR CORE ISOLATION COOLING (RCIC EIIS CODE BN) SYSTEM PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS EIIS CODE JM) LOGIC

ACTUATED. THIS WAS AN UNPLANNED LOGIC ACTUATION OF AN ENGINEERED SAFETY FEATURE. THE ROOT CAUSE OF THIS EVENT IS AN INCREASE IN THE SENSED TORUS CHAMBER AREA DIFFERENTIAL TEMPERATURE. THE TEMPERATURE DIFFERENTIAL WAS CAUSED BY A BLOWER BLOWING WARM AIR NEAR ONE SENSOR COMBINED WITH COLDER OUTSIDE AIR BLOWING ON ANOTHER SENSOR. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDED: 1) CHECKING FOR STEAM LEAKS AND INITIATING REQUIRED TECHNICAL SPECIFICATION ACTIONS, 2) INVESTIGATING THE EFFECT OF TEMPORARY DUCTS, 3) VERIFYING CORRECT VENTILATION SYSTEM OPERATION, 4) REVIEWING WORK IN PROGRESS, 5) DEVELOPING AND ANALYZING TREND DATA, AND 6) ANALYZING THE EVENT FOR ADDITIONAL CORRECTIVE ACTIONS. ON 11/12/87, ANOTHER LOGIC ACTUATION OCCURRED AND IS BELIEVED TO HAVE THE SAME ROOT CAUSE AS THE 10/19/87 EVENT. THE INVESTIGATION OF THE SECOND EVENT IS CONTINUING AND THE RESULTS WILL BE PRESENTED IN AN UPDATE TO THIS LER BY APPROXIMATELY 1/19/88.

[108] HOPE CREEK 1 DOCKET 50-354 LER 87-040 REV 01
UPDATE ON REACTOR WATER CLEANUP (RWCU) ISOLATIONS (2) FOLLOWING RWCU PUMP MAINTENANCE DUE TO NOT ADHERING TO PROCEDURES.
EVENT DATE: 091087 REPORT DATE: 112487 NSSS: GE TYPE: BWR

(NSIC 207246) ON SEPTEMBER 10, 1987 TWO ISOLATIONS OF THE REACTOR WATER CLEANUP SYSTEM OCCURRED DURING ATTEMPTS TO RESTORE "A" RWCU PUMP TO SERVICE FOLLOWING MAINTENANCE. AT 1046, AN RWCU ISOLATION WAS INITIATED BY AN RWCU HIGH DIFFERENTIAL FLOW SIGNAL. INVESTIGATION DETERMINED THAT TWO DRAIN VALVES ON THE DISCHARGE SIDE OF THE "A" RWCU PUMP HAD BEEN LEFT OPEN, RESULTING IN A HIGH DIFFERENTIAL FLOW CONDITION WHEN WARMING UP THE PUMP. AT 1339, ANOTHER RWCU ISOLATION OCCURRED WHEN THE "A" RWCU PUMP MECHANICAL SEAL FAILED DUE TO LACK OF COOLING WATER TO THE SEAL CAVITY COOLER. THIS RESULTED IN LOCALIZED HIGH TEMPERATURES IN THE RWCU PUMP ROOM, AND A STEAM LEAK DETECTION SYSTEM HIGH ROOM TEMPERATURE SIGNAL INITIATED THE ISOLATION. THE CAUSE OF THIS OCCURRENCE HAS BEEN ATTRIBUTED TO A COMBINATION OF FACTORS, THE MOST PREDOMINATE BEING PERSONNEL ERRORS COMMITTED WHILE PLACING THE PUMP BACK IN SERVICE. CORRECTIVE ACTIONS WERE PRIMARILY ADMINISTRATIVE IN NATURE, AND THE EQUIPMENT OPERATOR INVOLVED IN THE PERSONNEL ERROR WAS COUNSELED WITH RESPECT TO THE ERRORS MADE.

[109] INDIAN POINT 2 DOCKET 50-247 LER 87-012
ELECTRICAL POWER SUPPLY SPIKE IN CONTAINMENT CAUSES OPERATION OF ESF.
EVENT DATE: 101987 REPORT DATE: 111887 NSSS: WE TYPE: PWR

(NSIC 207255) ON OCTOBER 19, 1987, WHILE THE PLANT WAS AT COLD SHUTDOWN, A SINGLE ELECTRICAL SPIKE IN THE ELECTRICAL SUPPLY TO THE CONTAINMENT PARTICULATE RADIATION MONITOR (MON) CAUSED THE CONTAINMENT VENTILATION VALVES (ISV) TO CLOSE AUTOMATICALLY. IN ACCORDANCE WITH PLANT DESIGN, ISOLATION OF CONTAINMENT VENTILATION INITIATED PARTIAL OPERATION OF THE WELD CHANNEL AND PENETRATION PRESSURIZATION SYSTEM (WCPPS). THE WCPPS IS CLASSIFIED AS AN ENGINEERED SAFETY FEATURE (ESF). THE LOGIC FUNCTIONS OF THE ESF ACTUATION SYSTEM (ESFAS) WERE NEITHER REQUIRED NOR FULFILLED. THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED.

[110] INDIAN POINT 2 DOCKET 50-247 LER 87-013
INADVERTENT ACTUATION OF SAFETY INJECTION SYSTEM CAUSES ESF OPERATION.
EVENT DATE: 110587 REPORT DATE: 120587 NSSS: WE TYPE: PWR

(NSIC 207290) ON NOVEMBER 5, 1987, WHILE THE PLANT WAS AT COLD SHUTDOWN, A TECHNICIAN INITIATED A MANUAL SAFETY INJECTION SYSTEM (SIS) TRIP SIGNAL. THE TECHNICIAN WAS PERFORMING PREVENTIVE MAINTENANCE TO CHECK PROPER RELAY FREEDOM OF COIL MOVEMENT. AS PERMITTED BY THE TECHNICAL SPECIFICATIONS, ALMOST ALL ENGINEERED SAFETY FEATURES (ESF) WERE TAGGED OUT OF OPERATION FOR MAINTENANCE.

THE WELD CHANNEL AND PENETRATION PRESSURIZATION SYSTEM (WCPPS) DID OPERATE. THERE WAS NO IMPACT UPON THE HEALTH AND SAFETY OF THE PUBLIC.

[111] INDIAN POINT 2 DOCKET 50-247 LER 87-014
PROCEDURAL DEFICIENCY PREVENTS OBTAINING "AS FOUND" TEST DATA.
EVENT DATE: 110687 REPORT DATE: 121187 NSSS: WE TYPE: PWR

(NSIC 207416) ON NOVEMBER 6, 1987 WHILE THE PLANT WAS AT COLD SHUTDOWN FOR A REFUELING OUTAGE, TWO STEAM-GENERATOR LEVEL TRANSMITTERS WERE CONSERVATIVELY DECLARED INOPERABLE SINCE "AS FOUND" DATA COULD NOT BE OBTAINED DURING A SURVEILLANCE TEST. BOTH LEVEL TRANSMITTERS MONITOR ONE STEAM GENERATOR. "AS FOUND" DATA COULD NOT BE OBTAINED DUE TO ADJUSTMENTS WHICH WERE MADE BY A TECHNICIAN DURING THE SURVEILLANCE. THE ADJUSTMENTS WERE MADE SINCE REPRODUCIBLE MEASUREMENTS COULD NOT BE TAKEN. ALTHOUGH NOT RECOGNIZED AT THE TIME, REPRODUCIBLE RESULTS COULD NOT BE OBTAINED DUE TO WATER ENTRAPMENT. THE CAUSE OF THE EVENT IS THAT THE TEST PROCEDURE FAILED TO IDENTIFY WHETHER A "DRY" OR "WET" TEST METHOD SHOULD BE EMPLOYED IN THE SURVEILLANCE. THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED.

[112] INDIAN POINT 2 DOCKET 50-247 LER 87-015
INOPERABILITY OF BACKUP NITROGEN SUPPLY TO PORVS.
EVENT DATE: 111887 REPORT DATE: 121887 NSSS: WE TYPE: PWR

(NSIC 207417) INDIAN POINT 2 WAS AT COLD SHUTDOWN FOR A REFUELING OUTAGE. THE BACKUP NITROGEN SUPPLY SYSTEM TO THE PRESSURIZER POWER OPERATED RELIEF VALVES (PORVS), WHICH ARE USED FOR LOW TEMPERATURE OVERPRESSURE PROTECTION OF THE REACTOR COOLANT SYSTEM, WAS FOUND TO BE INOPERABLE DURING A TEST CONDUCTED ON 11/18/87. CHECK VALVES FAILED TO PREVENT BACKFLOW TO THE NON-SAFETY NITROGEN SYSTEM AND THE NITROGEN CONSUMPTION PER VALVE STROKE WAS GREATER THAN EXPECTED. THE CHECK VALVES WILL BE REPLACED. THE APPARENT HIGH NITROGEN CONSUMPTION PER VALVE STROKE IS BEING INVESTIGATED. THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED.

[113] LA SALLE 1 DOCKET 50-373 LER 87-034
AUTOMATIC START OF CONTROL ROOM VENTILATION EMERGENCY MAKE-UP TRAIN DUE TO SPURIOUS RADIATION SPIKE.
EVENT DATE: 101787 REPORT DATE: 111687 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LA SALLE 2 (BWR)

(NSIC 206966) AT 0607 HOURS ON OCTOBER 17, 1987, WITH UNITS 1 AND 2 IN OPERATIONAL CONDITION 1 (RUN) AT 89% AND 92% POWER RESPECTIVELY, THE "A" CONTROL ROOM HVAC (VC) INTAKE RADIATION MONITOR 1D18-K751D SPURIOUSLY TRIPPED (SPIKED HIGH) CAUSING AN AUTO-START OF THE "A" VC EMERGENCY MAKE-UP (EMU) TRAIN. PER DESIGN, THE MINIMUM OUTSIDE AIR DAMPERS CLOSED, ISOLATING THE "A" VC TRAIN FROM OUTSIDE AIR, AND AIR FLOW WAS RECIRCULATED THROUGH THE CHARCOAL ADSORBERS. INVESTIGATION OF THE RADIATION MONITOR REVEALED NO PROBLEM WHICH WOULD HAVE CAUSED THE SPURIOUS TRIP. IT IS BELIEVED THAT RANDOM NOISE IN THE ELECTRONICS MAY HAVE INDUCED THE SPURIOUS TRIP. THE SAFETY CONSEQUENCES OF THIS EVENT WERE MINIMAL SINCE THE "A" VC SYSTEM RESPONDED TO THE RADIATION MONITOR TRIP SIGNAL PER DESIGN. CORRECTIVE ACTIONS INCLUDED THE SUCCESSFUL PERFORMANCE OF THE CALIBRATION AND FUNCTIONAL TEST ON THE RADIATION MONITOR. SINCE BEING RETURNED TO SERVICE, THE RADIATION MONITOR HAS EXPERIENCED NO FURTHER PROBLEMS. THE STATION IS INVESTIGATING A LOGIC REVISION FOR THE VC RADIATION MONITORS. THIS EVENT IS REPORTABLE PURSUANT TO THE REQUIREMENTS OF 10CFR50.73(A)(2)(IV) DUE TO THE ACTUATION OF AN ENGINEERED SAFETY FEATURE (ESF) SYSTEM.

[114] LA SALLE 1 DOCKET 50-373 LER 87-036
SPURIOUS AMMONIA DETECTOR TRIP DUE TO BROKEN CHEMCASSETTE TAPE (DESIGN DEFICIENCY).
EVENT DATE: 111487 REPORT DATE: 120987 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LA SALLE 2 (BWR)
VENDOR: M D A SCIENTIFIC, INC.

(NSIC 207407) AT 1730 HOURS ON NOVEMBER 14, 1987, WITH UNITS 1 AND 2 IN OPERATIONAL CONDITION 1 (RUN) AT 92% AND 97% POWER RESPECTIVELY, THE "A" CONTROL ROOM HVAC SYSTEM (VC) "A" AMMONIA DETECTOR (OXY-VC125A) TRIPPED. PER DESIGN, AN ENGINEERED SAFETY FEATURE (ESF) DAMPER ACTUATION OCCURRED WHICH ISOLATED THE "A" VC TRAIN FROM OUTSIDE AIR. NO "ODOR EATER" (CHARCOAL ADSORBER) DAMPER ACTUATION OCCURRED SINCE, AT THE TIME OF THE EVENT, THE "A" VC SYSTEM WAS OPERATING IN THE RECIRCULATION MODE WITH FLOW THROUGH THE "ODOR EATER". THE INSTRUMENT MAINTENANCE DEPARTMENT INVESTIGATED THE EVENT AND FOUND THAT THE CHEMCASSETTE TAPE WAS BROKEN AT THE TAKEUP SPOOL. THE CHEMCASSETTE WAS REPLACED AND THE DETECTOR WAS DECLARED OPERABLE AT 1100 HOURS ON NOVEMBER 16, 1987. THE SAFETY CONSEQUENCES OF THIS EVENT WERE MINIMAL SINCE THE "A" VC SYSTEM RESPONDED TO THE AMMONIA DETECTOR TRIP IN ACCORDANCE WITH DESIGN. THIS EVENT IS REPORTABLE PURSUANT TO THE REQUIREMENTS OF 10CFR50.73(A)(2)(IV) DUE TO THE ACTUATION OF AN ESF SYSTEM.

[115] LA SALLE 2 DOCKET 50-374 LER 85-010 REV 01
UPDATE ON CONTROL ROD DRIVE HYDRAULIC CONTROL UNIT ACCUMULATOR PRESSURE SWITCH FAILURES DUE TO SETPOINT DRIFT.
EVENT DATE: 021585 REPORT DATE: 112087 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LA SALLE 1 (BWR)
VENDOR: BALKSDALE VALVE COMPANY

(NSIC 207398) BETWEEN 2/15/85 AND 3/1/85, THE ACCUMULATOR PRESSURE SWITCHES FOR THE UNIT 1 AND UNIT 2 CONTROL ROD DRIVE HYDRAULIC CONTROL UNITS WERE FOUND OUT OF CALIBRATION IN THE NON-CONSERVATIVE DIRECTION. ON UNIT 1, 180 OUT OF 185 SWITCHES HAD DRIFTED OUT OF CALIBRATION. ON UNIT 2, 184 OUT OF 185 SWITCHES HAD DRIFTED OUT OF CALIBRATION. BOTH UNITS WERE IN OPERATIONAL CONDITION 1 (RUN) AT THE TIME OF THE EVENT, THEREFORE SCRAM CAPABILITY WAS NOT SIGNIFICANTLY AFFECTED. THE CAUSE OF THE EVENT IS ATTRIBUTED TO SETPOINT DRIFT. THE SETPOINTS DRIFTED TO A LOWER PRESSURE. NO CAUSE FOR THE DRIFT COULD BE DETERMINED BY COMMONWEALTH EDISON COMPANY OR THE PRESSURE SWITCH MANUFACTURER (BARKSDALE). IMMEDIATE CORRECTIVE ACTIONS INCLUDED: RECHARGING OF ANY ACCUMULATORS FOUND TO HAVE LOW PRESSURE; RECALIBRATION OF ALL ACCUMULATOR PRESSURE SWITCHES; AND INCREASED MONITORING OF ALL ACCUMULATOR PRESSURES. THE TECH SPECS FOR UNIT 1 AND UNIT 2 WERE SUBSEQUENTLY AMENDED TO ALLOW FOR A MORE CONSERVATIVE PRESSURE SWITCH SETPOINT.

[116] LA SALLE 2 DOCKET 50-374 LER 87-018
REACTOR WATER CLEANUP ISOLATION CAUSED BY LOSS OF LEAKAGE DETECTION POWER RELAY DUE TO PERSONNEL ERROR.
EVENT DATE: 101787 REPORT DATE: 111387 NSSS: GE TYPE: BWR

(NSIC 207240) AT 0534 HOURS ON OCTOBER 17, 1987, WITH UNIT 2 IN OPERATIONAL CONDITION 1 (RUN) AT 92% POWER, A REACTOR WATER CLEANUP (RWCU) SYSTEM OUTBOARD ISOLATION (PARTIAL GROUP V) WAS RECEIVED DURING THE REMOVAL OF THE RESIDUAL HEAT REMOVAL (RHR) AREA AND DIFFERENTIAL TEMPERATURE RELAY FROM THE LEAK DETECTION CIRCUITRY FOR REPAIR. THE ROOT CAUSE OF THIS EVENT WAS INATTENTION TO DETAIL (INADEQUATE REVIEW) BY THE ELECTRICAL MAINTENANCE (EM) PERSONNEL INVOLVED. THE REVIEW FOR THE WORK WAS PERFORMED USING ELECTRICAL SCHEMATICS MORE EXTENSIVELY THAN WIRING DIAGRAMS. THE REVIEW FAILED TO IDENTIFY A COMMON GROUND BETWEEN SEVERAL RELAYS IN THE LEAK DETECTION CIRCUITRY. THE SAFETY CONSEQUENCES OF THE EVENT WERE MINIMAL. THE ISOLATION OCCURRED AS DESIGNED. ISOLATION OF THE RWCU

SYSTEM HAD NO ADVERSE IMPACT ON THE REACTOR COOLANT CHEMISTRY, WHICH WAS WITHIN THE TECH SPEC REQUIREMENTS DURING THIS EVENT. THE EM PERSONNEL WERE TRAINED ON THIS EVENT, AND INSTRUCTED ON THE USE OF WIRING DIAGRAMS. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(IV) DUE TO AN ACTUATION OF AN ENGINEERED SAFETY FEATURE (ESF) SYSTEM.

[117] LACROSSE DOCKET 50-409 LER 87-007
DEGRADED FIRE BARRIER.
EVENT DATE: 110987 REPORT DATE: 120387 NSSS: AC TYPE: BWR

(NSIC 207376) DURING THE 18-MONTH FIRE BARRIER INSPECTION, IT WAS DETERMINED THAT THE PLUG WAS NOT INSTALLED IN THE HOLE IN THE FLOOR BETWEEN THE MACHINE SHOP AND THE TUNNEL IN THE TURBINE BUILDING. THE HOLE'S COVER PLATE WAS INSTALLED, BUT THE PLUG WHICH CONSTITUTES THE FIRE BARRIER WAS NOT. THE HOLE IS USED FOR ROUTING WELDING CABLES AND HOSES TO THE TUNNEL. WHEN IT IS IN USE, TECHNICAL SPECIFICATIONS REQUIRE AN HOURLY FIRE PATROL, SINCE THE FIRE BARRIER IS NOT FUNCTIONAL. IT IS BELIEVED THAT THE PLUG WAS REMOVED IN AUGUST 1987 AND NOT REINSTALLED. SINCE MID-AUGUST, AN EVOLUTION WAS CONDUCTED PERIODICALLY DURING WHICH A HOSE WAS ROUTED THROUGH THE HOLE. THE PLANT IS PERMANENTLY SHUT DOWN AND THE REACTOR DEFUELED. THEREFORE, THE SAFETY SIGNIFICANCE OF THIS EVENT WAS MINIMAL. THE PLUG WAS REINSTALLED. THE TOP OF THE PLUG WAS MARKED AS A FIRE BARRIER. THE INCIDENT REPORT WAS PROMPTLY ROUTED TO THE OPERATIONS DEPARTMENT. THE PLANT SUPERINTENDENT WILL REMIND ALL PERSONNEL OF THE IMPORTANCE OF MAINTAINING ADHERENCE TO ALL REMAINING APPLICABLE REQUIREMENTS.

[118] LIMERICK 1 DOCKET 50-352 LER 87-018 REV 01
UPDATE ON EXCESSIVE PRIMARY CONTAINMENT LEAKAGE DUE TO NORMAL WEAR.
EVENT DATE: 052087 REPORT DATE: 120887 NSSS: GE TYPE: BWR
VENDOR: ATWOOD & MORRILL CO., INC.

(NSIC 207301) ON MAY 20, 1987 AT 1610 HOURS, AS A RESULT OF LOCAL LEAK RATE TESTING (LLRT), IT WAS DETERMINED THAT THE LEAKAGE RATES OF THE MAIN STEAM ISOLATION VALVES (MSIVS) AND THE TOTAL COMBINED PRIMARY CONTAINMENT PENETRATION LEAKAGE EXCEEDED THE LIMITS AS SET FORTH IN TECHNICAL SPECIFICATION 3.6.1.2 (C) AND (B) RESPECTIVELY. THERE WERE NO ADVERSE CONSEQUENCES AS A RESULT OF THE EVENT. THERE WAS NO RELEASE OF RADIOACTIVE MATERIAL AS A RESULT OF THIS EVENT. THE CAUSE OF THE EXCESSIVE LEAKAGE IS MINOR WEAR TO THE VALVE SEATING SURFACES AND THE LACK OF COMPLETE VALVE SEATING DUE TO SMALL PARTICLES OF DIRT LODGING BETWEEN THE VALVE DISC AND SEAT. THE MSIVS AND OTHER TYPE B & C PENETRATIONS HAVE BEEN REPAIRED AND WERE RETESTED TO ASSURE THAT THE ALLOWABLE LEAKAGE RATES WERE NOT EXCEEDED. THERE ARE NO ACTIONS TO PREVENT RECURRENCE AT THIS TIME, HOWEVER; RESULTS OF FUTURE LLRTS WILL BE MONITORED AND CONTINUED EXCESSIVE LEAKAGE EVALUATED IN ORDER TO DETERMINE IF ACTIONS BEYOND THE ESTABLISHED PLANT PROCEDURES ARE REQUIRED TO PREVENT RECURRENCE. THE PHILADELPHIA ELECTRIC COMPANY (PECO) IS SUPPORTING THE BOILING WATER REACTOR OWNERS GROUP (BWROG) IN AN INVESTIGATION OF MSIV LEAKAGE PROBLEMS. PECO WILL EVALUATE THE BWROG RECOMMENDATIONS AND ACT ON THEM ACCORDINGLY.

[119] LIMERICK 1 DOCKET 50-352 LER 87-023 REV 01
UPDATE ON ENGINEERED SAFETY FEATURE ACTUATION DUE TO BATTERY CHARGER FAILURE.
EVENT DATE: 061187 REPORT DATE: 121087 NSSS: GE TYPE: BWR
VENDOR: C & D BATTERIES, DIV OF ELTRA CORP.

(NSIC 207369) ON JUNE 11, 1987, THE STANDBY GAS TREATMENT AND REACTOR ENCLOSURE RECIRCULATION SYSTEMS (ENGINEERED SAFETY FEATURES) INITIATED AS A CONSEQUENCE OF ACTIONS TAKEN DUE TO FAILURE OF THE 1A1D103 STATION BATTERY CHARGER. THE 125 VDC STATION BATTERIES (1A1) WERE DISCONNECTED FROM THE BUS AT THE TIME OF THE EVENT TO ACCOMMODATE MAINTENANCE WORK. THE BATTERY CHARGER FAILURE IS BELIEVED TO BE A

RESULT OF AN INTEGRATED CIRCUIT CONTROLLER CARD FAILURE WHICH RESULTED IN DC VOLTAGE FLUCTUATION. HOWEVER, WHEN THE CARD MANUFACTURER PERFORMED A FAILURE ANALYSIS, NO DEFECT COULD BE FOUND. A TEMPORARY CIRCUIT ALTERATION (TCA) WAS INSTALLED TO PROVIDE AN ALTERNATE POWER SUPPLY TO THE DE-ENERGIZED BUS. DURING REENERGIZATION OF THE BUS, A REACTOR PROTECTION SYSTEM SERIES BREAKER TRIPPED ON A SHUNT TRIP SIGNAL FROM ITS UNDERVOLTAGE RELAYS AND CAUSED THE INBOARD INSTRUMENT GAS VALVE TO CLOSE. THE CAUSE FOR THE RPS BREAKER TRIP IS UNKNOWN AND TESTING HAS PROVEN THAT THE UNDERVOLTAGE RELAY SHUNT TRIP SIGNALS ARE REPEATABLE. THE UNDERVOLTAGE RELAYS WILL BE REPLACED WITH THE SAME MODEL AFTER SUCCESSFUL TESTING TO VERIFY THAT THE NEW RELAYS DO NOT PRODUCE SIMILAR SPURIOUS ACTUATIONS. THE CONSEQUENCES OF THIS EVENT WERE MINIMAL BECAUSE THE ENGINEERED SAFETY FEATURES INITIATED AS DESIGNED AND THE UNIT WAS SHUTDOWN WITH THE CORE OFF-LOADED AT THE TIME OF THE EVENT.

[120] LIMERICK 1 DOCKET 50-352 LER 87-046
REACTOR SCRAM AND NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NSSSS) GROUP IIB ISOLATION ON A MAIN TURBINE TRIP DUE TO MOISTURE SEPARATOR HIGH WATER LEVEL.
EVENT DATE: 090787 REPORT DATE: 101587 NSSS: GE TYPE: BWR

(NSIC 206645) ON 9/7/87 AT 1321 HOURS, THE REACTOR PROTECTION SYSTEM (RPS) INITIATED A FULL REACTOR SCRAM, WHEN THE MAIN TURBINE STOP VALVES CLOSED FOLLOWING A MAIN TURBINE TRIP ON HIGH-HIGH WATER LEVEL IN THE "C2" MOISTURE SEPARATOR DRAIN TANK. A NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NSSSS) GROUP IIB ISOLATION OCCURRED AS REACTOR WATER LEVEL DECREASED TO 12.5". THE UNIT WAS OPERATING AT 83% POWER FOLLOWING START-UP FROM THE FIRST REFUELING OUTAGE AT THE TIME OF THE EVENT. AN UNUSUAL EVENT WAS DECLARED DUE TO THE UNPLANNED SCRAM. PLANT PERSONNEL DISCOVERED THAT THE INSTRUMENT AIR SUPPLY ROOT VALVES SERVICING THE TWO "C2" MOISTURE SEPARATOR LEVEL TRANSMITTERS WERE NOT FULLY OPEN. AS A RESULT, THE LEVEL SIGNAL RECEIVED BY THE LEVEL CONTROLLERS FROM THE LEVEL TRANSMITTERS DID NOT INDICATE A TRUE LEVEL. THEREFORE, THE DRAIN VALVE AND DUMP VALVE FAILED TO PROPERLY CONTROL DRAIN TANK LEVEL. THE ROOT VALVES WERE OPENED FULLY AND THE SCRAM WAS RESET AND THE UNUSUAL EVENT TERMINATED AT 1345 HOURS. A NEW SPECIAL PROCEDURE WAS IMPLEMENTED TO VERIFY PROPER OPERATION OF ALL MOISTURE SEPARATOR DRAIN AND DUMP VALVE LEVEL CONTROL SYSTEMS. THE INSTRUMENT AIR SUPPLY PIPING AND INSTRUMENTATION DRAWINGS (P&IDS) WILL BE UPGRADED TO INCLUDE VALVES THAT CAN ISOLATE TWO OR MORE INSTRUMENTS OR DEVICES, AND A CHECK OFF LIST WILL BE DEVELOPED FOR VERIFICATION.

[121] LIMERICK 1 DOCKET 50-352 LER 87-049 REV 01
UPDATE ON TECHNICAL SPECIFICATION SURVEILLANCE TEST MISSED DUE TO PERSONNEL ERROR AND THERMAL TRANSIENT IN 1985 DUE TO PROCEDURAL ERROR DISCOVERED DURING INVESTIGATION.
EVENT DATE: 091987 REPORT DATE: 120387 NSSS: GE TYPE: BWR

(NSIC 207268) ON SEPTEMBER 19, 1987 AT APPROXIMATELY 1000 HOURS, DURING A REACTOR RECIRCULATION PUMP STARTUP, TECHNICAL SPECIFICATION 3.4.1.4 WAS NOT MET DUE TO PERSONNEL ERROR. PRIOR TO STARTING A RECIRCULATION PUMP WITH BOTH RECIRCULATION LOOPS IDLE, THE OPERATOR DID NOT PERFORM THE APPLICABLE STEPS OF ST-6-043-390-1 "REACTOR RECIRCULATION PUMP IDLE LOOP STARTUP TEMPERATURE AND FLOW CHECK" WHICH COMPARES REACTOR COOLANT TEMPERATURE TO THE IDLE RECIRCULATION LOOP TEMPERATURE. AN INVESTIGATION OF THIS EVENT COINCIDENTALLY REVEALED A PROCEDURAL ERROR IN THAT AN INCORRECT REFERENCE TEMPERATURE WAS INCORPORATED IN THE ST SINCE ITS ORIGINATION. ALL PERFORMANCES OF ST-6-043-390-1 INCLUDING THE ST PERFORMANCE IN QUESTION WERE REVIEWED BY COMPARING THE CORRECT TEMPERATURE WITH PREVIOUS TEST DATA. THAT REVIEW DETERMINED THAT ONLY ST-6-043-390-1 PERFORMED ON SEPTEMBER 16, 1985 DID NOT MEET THE ALLOWABLE TEMPERATURE DIFFERENTIAL CRITERIA. GENERAL ELECTRIC COMPANY WAS REQUESTED TO ASSESS THE SIGNIFICANCE OF THE THERMAL TRANSIENT ON SEPTEMBER 16, 1985. BASED ON THEIR REVIEW OF THE LGS THERMAL CYCLE

DIAGRAM, THEY DETERMINED THAT SUFFICIENT MARGIN EXISTS SUCH THAT EXCEEDING THE DIFFERENTIAL TEMPERATURE LIMIT ONCE HAS NOT CREATED AN UNACCEPTABLE CONDITION.

[122] LIMERICK 1 DOCKET 50-352 LER 87-055
NON-CONFORMANCE WITH FIRE SAFE SHUTDOWN REQUIREMENTS OF FIRE PROTECTION EVALUATION REPORT.
EVENT DATE: 100287 REPORT DATE: 111887 NSSS: GE TYPE: BWR

(NSIC 207236) ON OCTOBER 2, 1987 A NON-CONFORMANCE WITH THE FIRE SAFE SHUTDOWN REQUIREMENTS OF THE LIMERICK GENERATING STATION (LGS) FIRE PROTECTION EVALUATION REPORT (FPER) WAS IDENTIFIED DURING A DETAILED REVIEW OF EMERGENCY DIESEL GENERATOR (D/G) TRIP CIRCUITRY. SECTION 3.2.1, ITEM 17 OF THE LIMERICK FPER COMMITS TO IDENTIFYING AND ANALYZING ALL NON-CLASS 1E CIRCUITS WHERE FAILURE COULD AFFECT OPERATIONS OF SAFE SHUTDOWN EQUIPMENT. CONTRARY TO THE COMMITMENT, POWER SUPPLY CABLES FOR D/G FIRE SUPPRESSION FLOW SWITCHES WERE NOT PROPERLY EVALUATED DURING THE PERFORMANCE OF THE LGS UNIT 1 SAFE SHUTDOWN FIRE ANALYSIS. THE SUBJECT CABLES ARE ALL LOCATED IN FIRE AREA 75, THE SERVICE WATER PIPE TUNNEL. AN APPENDIX R DESIGN BASIS FIRE IN THAT AREA, POSTULATED TO CAUSE THE SHORTING OF THE INTERNAL CONDUCTORS OF THESE CABLES, COULD CAUSE A TRIP SIGNAL TO ALL FOUR D/G'S UNDER NON-LOCA CONDITIONS. THE CAUSE OF THIS EVENT WAS A DEFICIENCY IN A PROCEDURE USED DURING THE COMPREHENSIVE APPENDIX R SAFE SHUTDOWN FIRE ANALYSIS CONDUCTED FOR LGS UNIT 1 IN 1982. THE SUPPLY BREAKER FOR THE FLOW SWITCH HAS BEEN DE-ENERGIZED AND THE FLOW SWITCH RELAYS HAVE BEEN REMOVED. THE APPENDIX R REVIEW FOR LIMERICK GENERATING STATION WILL BE EVALUATED TO PROVIDE ASSURANCE THAT THIS IDENTIFIED DEFICIENCY IS AN ISOLATED CASE.

[123] LIMERICK 1 DOCKET 50-352 LER 87-058
REACTOR ENCLOSURE VENTILATION ISOLATION AND NSSS ISOLATION DUE TO LOW DIFFERENTIAL PRESSURE RESULTING FROM THE LOSS OF AUXILIARY STEAM HEAT.
EVENT DATE: 101987 REPORT DATE: 111987 NSSS: GE TYPE: BWR

(NSIC 207237) ON OCTOBER 19, 1987 AT 0134 HOURS A REACTOR ENCLOSURE VENTILATION ISOLATION AND A NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NSSSS) GROUPS VI A AND VI B (ENGINEERED SAFETY FEATURES) ISOLATION OCCURRED RESULTING FROM LOW DIFFERENTIAL PRESSURE, WITH BOTH THE STANDBY GAS TREATMENT SYSTEM (SGTS) AND REACTOR ENCLOSURE RECIRCULATION SYSTEM (RERS) INITIATING AS DESIGNED. THE LOW DIFFERENTIAL PRESSURE CONDITION WAS CREATED BY THE LOSS OF AUXILIARY STEAM IN THE INLET PLENUM, ALLOWING COOL OUTSIDE AIR TO ENTER, WARM, AND EXPAND IN THE REACTOR ENCLOSURE. THIS OVERWHELMED THE EXHAUST FAN'S ABILITY TO CONTROL THE REACTOR ENCLOSURE AIR PRESSURE. THE SUBSEQUENT DECREASE IN DIFFERENTIAL PRESSURE RESULTED IN THE ISOLATION. THE AUXILIARY STEAM WAS RESTORED, THE NSSSS AND REACTOR ENCLOSURE ISOLATIONS WERE RESET BY 0200 HOURS, AND NORMAL REACTOR ENCLOSURE VENTILATION WAS RESTORED. THE DURATION OF THE ISOLATIONS WAS 26 MINUTES. THERE WERE NO ADVERSE CONSEQUENCES AND NO RELEASE OF RADIOACTIVE MATERIAL TO THE ENVIRONMENT AS A RESULT OF THIS EVENT. THE ENGINEERING DIVISION IS EVALUATING WAYS OF IMPROVING THE RELIABILITY OF THE AUXILIARY BOILERS.

[124] LIMERICK 1 DOCKET 50-352 LER 87-059
REACTOR ENCLOSURE ISOLATION ON LOW DIFFERENTIAL PRESSURE DURING APPLICATION OF A BLOCKING PERMIT.
EVENT DATE: 102387 REPORT DATE: 120287 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 207302) ON 10/23/87 AT 1419 HOURS, AN ISOLATION OF THE REACTOR ENCLOSURE OCCURRED ON LOW DIFFERENTIAL PRESSURE. THE STANDBY GAS TREATMENT SYSTEM (SGTS) AND REACTOR ENCLOSURE RECIRCULATION SYSTEM (RERS), ENGINEERED SAFETY FEATURES (ESF), INITIATED AS DESIGNED. THE ISOLATION OCCURRED WHILE THE 'A' INSTRUMENT AIR COMPRESSOR/DRYER TRAIN WAS BEING REMOVED FROM SERVICE TO ALLOW REPAIR OF TWO

SUSPECTED LEAKING CHECK VALVES DOWNSTREAM OF THE 'A' INSTRUMENT AIR DRYER. THE EVENT WAS CAUSED DUE TO A FAILED CHECK VALVE ON THE 'A' INSTRUMENT AIR COMPRESSOR/DRYER TRAIN, WHICH CAUSED AIR PRESSURE IN THE INSTRUMENT AIR SYSTEM TO DECREASE WHEN THE 'A' AND THE 'B' TRAINS WERE TIED TOGETHER. THERE WERE NO ADVERSE CONSEQUENCES AND NO RELEASE OF RADIOACTIVE MATERIAL, AS A RESULT OF THIS EVENT. AT 1425 HOURS, THE ISOLATION WAS RESET AND OPERATION OF SGTS AND RERS WAS TERMINATED. THE REACTOR ENCLOSURE REMAINED ISOLATED FOR 6 MINUTES. THE SEQUENCE FOR ISOLATING THE 'A' INSTRUMENT AIR COMPRESSOR/DRYER TRAIN WAS CHANGED SO THAT THE SUSPECTED LEAKING VALVES WERE ISOLATED PRIOR TO SECURING THE 'A' COMPRESSOR, AND THE 'A' TRAIN WAS SUCCESSFULLY ISOLATED. THE FAILED VALVE WAS REPLACED AT 1605 HOURS. A SYSTEM PROCEDURE WILL BE WRITTEN TO DETAIL THE SPECIFIC STEPS FOR REMOVING AN INSTRUMENT AIR COMPRESSOR/DRYER TRAIN OF THE INSTRUMENT AIR SYSTEM FROM SERVICE.

[125] LIMERICK 1 DOCKET 50-352 LER 87-060
TWO REACTOR ENCLOSURE ISOLATIONS DURING BLOCKING DUE TO INADEQUATE SYSTEM IDENTIFICATION.
EVENT DATE: 102987 REPORT DATE: 112587 NSSS: GE TYPE: BWR

(NSIC 207155) ON OCTOBER 29, 1987 AT 0006 HOURS AND ON NOVEMBER 6, 1987 AT 0728 HOURS THE REACTOR ENCLOSURE HVAC ISOLATED AND THE STANDBY GAS TREATMENT SYSTEM (SGTS) AND THE REACTOR ENCLOSURE RECIRCULATION SYSTEM (RERS), ENGINEERED SAFETY FEATURES, INITIATED AS DESIGNED WHEN THE DIFFERENTIAL PRESSURE BETWEEN THE REACTOR ENCLOSURE SECONDARY CONTAINMENT AND THE OUTSIDE ATMOSPHERE DECREASED BELOW THE SETPOINT OF MINUS 0.1 INCHES WATER GAUGE. THE IMMEDIATE CAUSE OF BOTH OF THESE EVENTS WAS THE LOSS OF INSTRUMENT AIR SUPPLY TO THE REACTOR ENCLOSURE ISOLATION DAMPER POSITION CONTROLLERS. THE CAUSE OF THE EVENT OF OCTOBER 29, 1987 WAS A COGNITIVE PERSONNEL ERROR WHEN A UTILITY EMPLOYED NON-LICENSED OPERATOR FAILED TO CLOSE THE PROPER INSTRUMENT AIR ISOLATION VALVE WHILE REMOVING EQUIPMENT FROM SERVICE FOR MAINTENANCE. THE CAUSE OF THE EVENT OF NOVEMBER 6, 1987 WAS THE INCORRECT LABELING OF TWO INSTRUMENT AIR ISOLATION VALVES. THE ROOT CAUSE OF BOTH EVENTS IS THE LACK OF A COMPLETE AND ACCURATE IDENTIFICATION AND TAGGING SYSTEM FOR THE INSTRUMENT AIR SYSTEM TUBING AND BLOCK VALVES. THERE WERE NO ADVERSE CONSEQUENCES AS A RESULT OF THESE EVENTS. THERE WAS NO RELEASE OF RADIATION AS A RESULT OF THESE EVENTS. PLANT MODIFICATION 5561 IS BEING PREPARED TO CORRECT A NUMBER OF DEFICIENCIES OF THE INSTRUMENT AIR SYSTEM.

[126] MCGUIRE 1 DOCKET 50-369 LER 87-014 REV 01
UPDATE ON DIESEL GENERATOR 1A CONTROL POWER CIRCUIT BREAKER WAS INADVERTENTLY LEFT OPEN DUE TO PERSONNEL ERROR.
EVENT DATE: 072887 REPORT DATE: 122887 NSSS: WE TYPE: PWR

(NSIC 207457) ON JULY 30, 1987, AT 0530, OPERATIONS DISCOVERED THE CONTROL POWER CIRCUIT BREAKER FOR DIESEL GENERATOR (D/G) 1A OPEN WHICH RENDERS D/G 1A INOPERABLE, RESULTING IN A TECH SPEC VIOLATION DUE TO THE D/G BEING UNKNOWINGLY INOPERABLE FOR APPROXIMATELY 3 DAYS. THE CIRCUIT BREAKER HAD BEEN OPEN SINCE 1800 ON JULY 26, 1987. THE CIRCUIT BREAKER WAS CLOSED, AN OPERABILITY TEST WAS SUCCESSFULLY PERFORMED, AND THE D/G WAS RETURNED TO AN INOPERABLE STATUS TO PERFORM ROUTINE MAINTENANCE. D/G 1A WAS RETURNED TO OPERABLE STATUS AT 0645 ON JULY 31, 1987, UPON THE COMPLETION OF THE PERIODIC MAINTENANCE ACTIVITY. THIS EVENT HAS BEEN CLASSIFIED AS A PERSONNEL ERROR BECAUSE OPERATIONS PERSONNEL FAILED TO FOLLOW STATION AND GROUP DIRECTIVES WHILE PERFORMING A PROCEDURE TO RESTORE CONTROL POWER TO THE D/G AFTER MAINTENANCE. A CONTRIBUTORY CAUSE OF MANAGEMENT DEFICIENCY HAS BEEN ASSIGNED TO THIS EVENT BECAUSE THE INSTRUCTIONS GIVEN TO THE OPERATIONS PERSONNEL WERE NOT ADEQUATE TO ASSURE THE WORK WOULD BE DONE CORRECTLY. CORRECTIVE ACTIONS INCLUDE TAGGING COMPONENTS, POLICY CHANGES, PROCEDURE REVISIONS, INCREASED EMPHASIS ON CONTROL BOARD DEFICIENCIES, TRAINING, AND A CONTROL BOARD MODIFICATION.

[127] MCGUIRE 1 DOCKET 50-369 LER 87-024
THE REACTOR BUILDING EQUIPMENT HATCH WAS IMPROPERLY CLOSED DURING FUEL RELOADING DUE TO INADEQUATE PROCEDURAL INSTRUCTIONS AND INSUFFICIENT ADMINISTRATIVE CONTROLS.
EVENT DATE: 100287 REPORT DATE: 110287 NSSS: WE TYPE: PWR

(NSIC 207103) ON 10/02/87 AT 1250, THE REACTOR BUILDING EQUIPMENT HATCH WAS DISCOVERED TO HAVE A SMALL GAP AROUND THE TOP SEALING SURFACE. OPERATIONS (OPS) FUEL HANDLING PERSONNEL WERE NOTIFIED WHO SUSPENDED FUEL MOVEMENT. THE EQUIPMENT HATCH WAS COMPLETELY SEALED AND FUEL HANDLING RESUMED AT 1317. THE CAUSE OF THE EVENT WAS INADEQUATE PROCEDURAL INSTRUCTIONS THAT ADDRESSED THE TEMPORARY REPLACEMENT OF THE EQUIPMENT HATCH IN THE PROCEDURE USED BY THE VENDOR EQUIPMENT HANDLING CREW, AND INSUFFICIENT ADMINISTRATIVE CONTROLS DUE TO THE LACK OF CONSISTENT AND ADEQUATE TRAINING PROVIDED TO THE EQUIPMENT HANDLING CREW. ALSO, INADEQUATE CONTROLS EXISTED TO ENSURE THAT THE EQUIPMENT HATCH INSTALLATION WAS OVERSEEN BY A QUALIFIED STATION EMPLOYEE. THE APPROPRIATE PROCEDURES WILL BE REVISED TO ENSURE COMPLETE CLOSURE OF THE EQUIPMENT HATCH. MAINTENANCE WILL PERFORM AN INSPECTION FOLLOWING THE CLOSURE OF THE HATCH. IF VENDORS ARE USED TO PERFORM THIS WORK, PLANT PERSONNEL WILL PROVIDE SPONSORSHIP FOR THE VENDOR CREW TO ENSURE THEY RECEIVE MAINTENANCE ORIENTATION TRAINING.

[128] MCGUIRE 1 DOCKET 50-369 LER 87-025
PRESSURIZER DISSOLVED OXYGEN WAS NOT VERIFIED TO BE WITHIN LIMITS OF TECH SPEC PRIOR TO EXCEEDING PRZ TEMPERATURE OF 250F DUE TO MANAGEMENT DEFICIENCY.
EVENT DATE: 102587 REPORT DATE: 112487 NSSS: WE TYPE: PWR

(NSIC 207303) ON 10/25/87 AT APPROXIMATELY 1310 WHILE IN MODE 5, COLD SHUTDOWN, OPERATIONS (OPS) PERSONNEL INCREASED PRESSURIZER (PZR) TEMPERATURE (TEMP) ABOVE 250 DEGREES-F DURING UNIT STARTUP AFTER CHEMISTRY PERSONNEL REPORTED A PRESSURIZER WATER SAMPLE WITH 60 PARTS PER BILLION (PPB) DISSOLVED OXYGEN (DO). ALL PREVIOUS SAMPLE RESULTS ON THIS DAY AND 3 CONSECUTIVE SUBSEQUENT SAMPLES OBTAINED AFTER PZR TEMP INCREASED ABOVE 250 DEGREES-F EXCEED 100 PPB LIMIT FOR DO. AT APPROXIMATELY 1500, CHEMISTRY DISCOVERED THE PZR TEMP HAD BEEN INCREASED ABOVE 250 DEGREES-F PRIOR TO VERIFICATION THAT THE DO CONCENTRATION WAS WITHIN LIMITS. CHEMISTRY AND OPS DETERMINED THAT THE QUICKEST WAY TO DECREASE DO CONCENTRATION WOULD BE TO CONTINUE INCREASING PZR TEMP IN ORDER TO INCREASE REACTOR COOLANT (NC) SYSTEM PRESSURE TO PERMIT OPERATION OF AN NC PUMP TO FACILITATE HYDRAZINE ADDITIONS AND MIXING TO REDUCE DO CONCENTRATION, THUS OPS CONTINUED TO INCREASE PZR TEMP. AT 845 ON 10/26/87, PZR DO WAS CONFIRMED BY 2 CONSECUTIVE SAMPLES TO BE LESS THAN 100 PPB. THIS EVENT WAS CAUSED BY INSUFFICIENT COMMUNICATIONS TO ENSURE THAT THE PZR DO WAS VERIFIED TO BE WITHIN LIMITS PRIOR TO EXCEEDING 250 DEGREES-F PZR TEMP. APPROPRIATE PERSONNEL WILL BE TRAINED REGARDING REPORTING OF SAMPLE RESULTS.

[129] MCGUIRE 1 DOCKET 50-369 LER 87-027
ENTERED HOT SHUTDOWN WITHOUT CONTAINMENT SPRAY HEAT EXCHANGER 1B COOLING WATER INLET VALVE BEING RETESTED DUE TO PERSONNEL ERROR.
EVENT DATE: 102987 REPORT DATE: 122187 NSSS: WE TYPE: PWR
VENDOR: BASIC IN FLOW CO.

(NSIC 207458) ON 11/19/87 WHILE REVIEWING AN INSTRUMENTATION AND ELECTRICAL (IAE) WORK REQUEST FOR FINAL SIGN OFF, AN IAE STAFF PERSON DISCOVERED THAT ADDITIONAL RETEST DETERMINATIONS WERE REQUIRED ON VALVE 1RN-235B, 1B CONTAINMENT SPRAY HEAT EXCHANGER COOLING WATER INLET VALVE. WORK HAD BEEN PERFORMED ON 1RN-235B AS SPECIFIED BY 2 WORK REQUESTS. THE 2 WORK REQUESTS DOCUMENTED A COMPLETED RETEST. ADDITIONAL WORK WAS PERFORMED ON VALVE 1RN-235B AFTER THE FIRST RETEST AND THERE WAS NO RETEST DETERMINATION MADE BY PERFORMANCE (PRF). TWO WORK REQUESTS HAD BEEN SIGNED OFF AS COMPLETED WITHOUT A RETEST DETERMINATION BY PRF. ON 11/19/87, PRF SUCCESSFULLY COMPLETED RETESTING OF VALVE 1RN-235B. THIS EVENT WAS DUE TO

PERSONNEL ERROR BECAUSE THE WORK REQUESTS WERE SIGNED OFF AS COMPLETED BUT WERE NOT SENT TO PERFORMANCE FOR A RETEST DETERMINATION. A PROCEDURE CHANGE WILL INCORPORATE THE FACT THAT ONCE A WORK REQUEST IS SIGNED AS COMPLETED, IT CANNOT BE REOPENED TO DO ADDITIONAL WORK. IAE, PRF, AND OPERATIONS WILL REVIEW THIS EVENT WITH THEIR PERSONNEL.

[130] MCGUIRE 1 DOCKET 50-369 LER 87-026
UNITS 1 AND 2 ENTERED TECH SPEC 3.0.3 DUE TO BOTH TRAINS OF CONTROL ROOM VENTILATION AND CHILLED WATER INOPERABLE DUE TO LEAKING DOOR SEALS AND MANAGEMENT DEFICIENCY.
EVENT DATE: 110587 REPORT DATE: 120787 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 207374) ON 11/05/87 WHILE PERFORMING A PERIODIC TEST OF THE CONTROL AREA VENTILATION AND CHILLED WATER (VC/YC) SYSTEM, PERFORMANCE (PRF) FOUND THE CONTROL ROOM (CR) POSITIVE PRESSURE TO BE BELOW TECH SPEC LIMITS WITH VC/YC TRAIN A IN OPERATION. OPERATIONS (OPS) DECLARED VC/YC TRAIN A INOPERABLE AT 1600. AFTER INCREASING OUTSIDE AIR FLOW TO THE CR TO NEAR MAXIMUM ALLOWABLE FLOW, IT WAS DETERMINED THAT SEVERAL CR DOOR SEALS WERE ALLOWING OUTLEAKAGE. PRF INFORMED OPS THAT THIS CONDITION WOULD PROBABLY ALSO CAUSE CR POSITIVE PRESSURE TO BE BELOW LIMITS WITH C/YC TRAIN B IN OPERATION. OPS DECLARED VC/YC TRAIN B INOPERABLE AS OF 1600. WITH BOTH VC/YC TRAINS INOPERABLE, UNITS 1 AND 2 ENTERED TECH SPEC 3.0.3 RETROACTIVELY AT 1600, (ACTION STATEMENT HAD BEEN MET FROM 1600 TO 2000). AFTER REPAIRED THE SEALS, CR PRESSURIZATION MET THE TECH SPEC CRITERIA, AND VC/YC TRAIN A WAS DECLARED OPERABLE AT 0021 ON 11/06/87. VC/YC TRAIN B WAS STARTED AND THE MANUAL VOLUME DAMPERS WERE ADJUSTED TO INCREASE OUTSIDE AIR FLOW TO THE VC/YC TRAIN B THEN PASSED THE CR POSITIVE PRESSURE TEST AND WAS DECLARED OPERABLE AT 0405. THIS EVENT WAS DUE TO THE DEGRADATION OF CR DOOR SEALS AND MANAGEMENT DEFICIENCY DUE TO A LACK OF PREVENTATIVE MAINTENANCE (PM) PROGRAM FOR THE CR DOOR SEALS. A PM PROGRAM WILL BE ESTABLISHED, AND THE SEALS WILL BE CHECKED AND REPLACED AS NEEDED.

[131] MCGUIRE 1 DOCKET 50-369 LER 87-032
FIRE BARRIER INOPERABLE FOR AN UNDETERMINED AMOUNT OF TIME.
EVENT DATE: 110887 REPORT DATE: 123087 NSSS: WE TYPE: PWR

(NSIC 207463) ON 11/08/87 AT 1330, INSTRUMENTATION AND ELECTRICAL (IAE) PERSONNEL DISCOVERED A DAMAGED FIRE BARRIER ON AN AUXILIARY FEEDWATER SYSTEM VALVE ACTUATOR. IAE IMMEDIATELY NOTIFIED THE CONTROL ROOM SENIOR REACTOR OPERATOR OF THE BREACHED FIRE BARRIER. OPERATIONS THEN DECLARED THE FIRE BARRIER INOPERABLE, IMMEDIATELY INITIATED A FIRE WATCH FOR THE ROOM IN WHICH THE VALVE IS LOCATED, AND WROTE A WORK REQUEST TO REPAIR THE FIRE BARRIER. THE FIRE BARRIER WAS REPAIRED BY MECHANICAL MAINTENANCE ON 11/13/87. SIGNS WERE ATTACHED TO ALL TECHNICAL SPECIFICATION FIRE BARRIERS OF THERMALAG 303 TYPE WARNING THAT THE BARRIERS ARE TECHNICAL SPECIFICATION FIRE BARRIERS. A CAUSE OF OTHER HAS BEEN ASSIGNED TO THIS EVENT BECAUSE IT COULD NOT BE DETERMINED WHEN, WHY, OR BY WHOM THE FIRE BARRIER WAS BREACHED, OR WHY IT WAS NOT DECLARED INOPERABLE AS A FIRE BARRIER AS REQUIRED BY TECHNICAL SPECIFICATIONS. IT SHOULD ALSO BE NOTED THAT CORRECTIVE ACTIONS FROM PREVIOUS LICENSEE EVENT REPORTS ARE STILL UNDERWAY AND SHOULD PROVE EFFECTIVE IN PREVENTING FURTHER OCCURRENCES OF THIS TYPE.

[132] MCGUIRE 1 DOCKET 50-369 LER 87-029
UNIDENTIFIED REACTOR COOLANT SYSTEM LEAKAGE GREATER THAN 1 GPM CAUSED UNIT SHUT DOWN AS A RESULT OF VALVE PACKING FAILURE RESULTING FROM DEFECTIVE PROCEDURE.
EVENT DATE: 111987 REPORT DATE: 122187 NSSS: WE TYPE: PWR
VENDOR: FISHER CONTROLS CO.

(NSIC 207460) ON NOVEMBER 19, 1987 AT 0155, A UNIT 1 REACTOR COOLANT SYSTEM

LEAKAGE CALCULATION RESULT WAS GREATER THAN THE ONE GPM TECHNICAL SPECIFICATION ALLOWABLE UNIDENTIFIED LEAKAGE. AT 0652, AN UNUSUAL EVENT WAS DECLARED AND OPERATIONS COMMENCED A UNIT 1 SHUTDOWN. A SIGNIFICANT STEM PACKING LEAK WAS FOUND ON A CHARGING PUMP DISCHARGE CONTROL VALVE (1NV-238), AND AFTER IDENTIFICATION AND ADJUSTMENT, UNIDENTIFIED LEAKAGE WAS RESTORED TO A PERMISSIBLE VALUE. SUBSEQUENTLY, A TEMPORARY REPAIR WAS MADE WITH AN INJECTABLE LIQUID LEAK SEALING COMPOUND. AT 1650 ON THE SAME DAY, THE UNUSUAL EVENT WAS TERMINATED, AND BY 0837 ON NOVEMBER 21, 1987, UNIT 1 RETURNED TO FULL POWER. THE CAUSE OF THE WEAK WAS IMPROPER PACKING INSTALLATION DURING THE RECENT REFUELING OUTAGE CAUSED BY AN INADEQUATE PROCEDURE. A CONTRIBUTORY CAUSE OF MANAGEMENT DEFICIENCY HAS ALSO BEEN ASSIGNED BECAUSE INADEQUATE CONTROL WAS EXERCISED OVER CONTRACT LABOR PERSONNEL. MCGUIRE HAS BEEN WORKING ON A PACKING IMPROVEMENT PROGRAM SINCE FEBRUARY 1987 WHICH INCLUDES THIS PROCEDURE. ALSO, VALVE 1NV-238 WILL BE REBUILT.

[133] MCGUIRE 1 DOCKET 50-369 LER 87-028
MANUAL ACTUATION OF AUXILIARY FEEDWATER SYSTEM TO PREVENT A REACTOR TRIP ON LOW STEAM GENERATOR 1D LEVEL BECAUSE OF A VALVE MALFUNCTION.
EVENT DATE: 112087 REPORT DATE: 122187 NSSS: WE TYPE: PWR
VENDOR: FISHER CONTROLS CO.

(NSIC 207459) ON NOVEMBER 20, 1987 AT 0223, OPERATIONS (OPS) MANUALLY STARTED AUXILIARY FEEDWATER (CA) MOTOR DRIVEN PUMP 1B TO PREVENT THE LEVEL IN STEAM GENERATOR (S/G) 1D FROM DECREASING TO THE LOW-LOW LEVEL ALERT SETPOINT AND INITIATING A REACTOR TRIP. DURING UNIT 1 STARTUP, S/G 1D LEVEL HAD DECREASED TO APPROXIMATELY 15% ON THE NARROW RANGE LEVEL INSTRUMENTATION BECAUSE FLOW THROUGH VALVE 1CF-107, S/G FEEDWATER CONTROL VALVE BYPASS, HAD NOT INCREASED IN CONJUNCTION WITH DEMAND AS OPS BEGAN TO INCREASE REACTOR POWER FROM 3%. OPS SECURED CA MOTOR DRIVEN PUMP 1B AT 0307, AFTER ESTABLISHING ADEQUATE FEEDWATER FLOW TO S/G 1D BY OPENING VALVE 1CF-17, S/G 1D FEEDWATER CONTROL. OPS MADE NOTIFICATION OF THE ENGINEERED SAFETY FEATURES (ESF) COMPONENT ACTUATION AT 0320. OPS SUBSEQUENTLY CHECKED VALVE CF-107 AND FOUND IT TO BE FUNCTIONING PROPERLY. THIS EVENT HAS BEEN ASSIGNED A CAUSE OF OTHER, SINCE THE MANUAL ESF ACTUATION WAS TO COUNTERACT THE EFFECTS OF A MALFUNCTION OF VALVE 1CF-107. OPS WILL MONITOR THE OPERATION OF VALVE 1CF-107 TO ENSURE THE VALVE FUNCTIONS PROPERLY, AND WILL INITIATE FURTHER CORRECTIVE ACTION IF NECESSARY BASED ON THIS OBSERVATION.

[134] MCGUIRE 1 DOCKET 50-369 LER 87-031
BOTH COMPONENT COOLING HEAT EXCHANGERS DECLARED INOPERABLE DUE TO EQUIPMENT FAILURE.
EVENT DATE: 112587 REPORT DATE: 122887 NSSS: WE TYPE: PWR
VENDOR: DELTA SOUTHERN CO.
FISHER CONTROLS CO.

(NSIC 207462) ON 11/25/87 AT 1730, A PERFORMANCE TEST ON COMPONENT COOLING HEAT EXCHANGER 1B DETERMINED THAT THE DIFFERENTIAL PRESSURE (D/P) ACROSS THE HEAT EXCHANGER EXCEEDED ALLOWABLE ACCEPTANCE CRITERIA. UNIT 1 ENTERED TECHNICAL SPECIFICATION (TS) 3.0.3 AT 1730 BECAUSE COMPONENT COOLING (KC) HEAT EXCHANGER 1A HAD BEEN DECLARED INOPERABLE AT 1715 WHEN VALVE 1RN-89, KC HEAT EXCHANGER (HX) 1A OUTLET TEMPERATURE CONTROL VALVE, FAILED TO OPEN DURING A ROUTINE PLANT EVOLUTION. VALVE 1RN-89 WAS ADMINISTRATIVELY BLOCKED OPEN, AND TS 3.0.3 WAS EXITED AT 1810 ON 11/25/87. MECHANICAL MAINTENANCE CLEANED THE KC HX 1B TUBES AND ON NOVEMBER 28 IT SUCCESSFULLY PASSED THE PRESSURE DROP TEST. OPERATIONS DECLARED KC HX 1B OPERABLE ON 11/28/87 AT 1435. THIS EVENT HAS BEEN CLASSIFIED AS OTHER BECAUSE BOTH UNIT 1 KC HXS WERE MADE INOPERABLE AT THE SAME TIME BY EQUIPMENT FAILURE. TEMPORARY D/P TRANSMITTERS WILL BE INSTALLED ACROSS BOTH KC HXS ON UNITS 1 AND 2 AND WILL BE MONITORED DAILY. A PERMANENT D/P MONITORING SYSTEM WILL ALSO BE EVALUATED. ADDITIONALLY A STUDY WILL BE INITIATED FOR ENHANCEMENTS TO THE NUCLEAR SERVICE WATER SYSTEM TO REDUCE HX FOULING PROBLEMS.

[135] MCGUIRE 2 DOCKET 50-370 LER 87-019
REACTOR TRIP DUE TO LOSS OF MAIN FEEDWATER PUMP BECAUSE OF LOW CONDENSER VACUUM AS A RESULT OF A FAILURE TO FOLLOW PROCEDURE.
EVENT DATE: 110587 REPORT DATE: 120787 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 1 (PWR)

(NSIC 207375) ON 11/05/87 AT 0646, THE UNIT 2 REACTOR TRIPPED DUE TO A LOW-LOW LEVEL IN STEAM GENERATOR 2B. THE MAIN TURBINE TRIPPED BECAUSE OF THE REACTOR TRIP. OPERATIONS (OPS) DETERMINED THAT AN AIR BUBBLE FROM A UNIT 1 HEAT EXCHANGER HAD MIGRATED FROM UNIT 1 TO UNIT 2 THROUGH CROSSOVER PIPING TO THE UNIT 2 MAIN FEEDWATER PUMP TURBINE (CFPT) 2B CONDENSER WATERBOX. THE AIR BUBBLE CAUSED A LOSS OF COOLING EFFICIENCY IN THE CONDENSER RESULTING IN A LOW VACUUM IN THE CFPT 2B CONDENSER. CFPT 2B TRIPPED INITIATING A REACTOR/TURBINE RUNBACK; THE DIGITAL ELECTRO HYDRAULIC TURBINE CONTROL SYSTEM DID NOT RESPOND CORRECTLY, AND THE REACTOR TRIPPED ON A STEAM GENERATOR 2B LOW-LOW LEVEL SIGNAL. OPS VENTED THE AIR OUT OF THE MAIN FEEDWATER PUMP CONDENSER AND VACUUM WAS RESTORED TO THE CFPT 2B CONDENSER, AND UNIT 2 ENTERED MODE 1 (POWER OPERATION) AT 0945 ON 11/06/87. DURING AIR SPARGING OF UNIT 1 CONTAINMENT SPRAY (NS) HEAT EXCHANGER (HX) 1A, MECHANICAL MAINTENANCE PERSONNEL DID NOT OPEN THE HX VENT VALVE AS INSTRUCTED IN HE PROCEDURE; THEREFORE, THIS EVENT IS CLASSIFIED AS PERSONNEL ERROR. PERFORMANCE WILL INITIATE A SAFETY REVIEW OF THE NS HX CLEANING PROCESS RELATIVE TO AIR LEAKAGE INTO THE NUCLEAR SERVICE WATER (RN) HEADER AND OTHER RN COMPONENTS.

[136] MILLSTONE 1 DOCKET 50-245 LER 87-015 REV 02
UPDATE ON LOCAL LEAK RATE TEST FAILURES.
EVENT DATE: 060687 REPORT DATE: 111987 NSSS: GE TYPE: BWR
VENDOR: ALLIS CHALMERS
CHAPMAN VALVE & MFG
CRANE COMPANY
MASONEILAN INTERNATIONAL, INC.
TARGET ROCK CORP.
VELAN VALVE CORP.

(NSIC 207170) ON JUNE 6, 1987 AT 1000 HOURS, WHILE PERFORMING LOCAL LEAK RATE TESTING (LLRT) DURING THE 1987 REFUEL OUTAGE, IT WAS IDENTIFIED THAT THE "B" MAIN STEAM ISOLATION VALVES COULD NOT MEET THE REQUIRED LEAK RATE AS SPECIFIED IN TECHNICAL SPECIFICATION 4.7.F.2.C. TESTING OF ALL PRIMARY CONTAINMENT ISOLATION VALVES, CABLE PENETRATIONS AND MANWAYS AS REQUIRED BY 10CFR50 APPENDIX J REVEALED ADDITIONAL ISOLATION VALVES THAT DID NOT MEET THE LOCAL LEAK RATE TEST REQUIREMENT. ALL VALVES THAT FAILED TO MEET THE LOCAL LEAKAGE RATE TEST REQUIREMENTS WERE SATISFACTORILY RETESTED SUBSEQUENT TO REPAIRS. THERE WERE NO CONSEQUENCES.

[137] MILLSTONE 1 DOCKET 50-245 LER 87-042
FAILURE TO ESTABLISH A PAS SYSTEM SURVEILLANCE.
EVENT DATE: 102787 REPORT DATE: 112387 NSSS: GE TYPE: BWR

(NSIC 207254) ON OCTOBER 27, 1987 AT 1530 HOURS, WITH THE PLANT OPERATING AT 100% POWER, UNIT 1 ENGINEERING WAS REVIEWING A RESPONSE TO A INFORMATION NOTICE (IE-86-60), TITLED UNANALYZED POST LOCA RELEASE PATHS. IT WAS DETERMINED THAT NO SURVEILLANCE EXISTED THAT TESTED THE POST ACCIDENT SAMPLING SYSTEM (PASS) AS REQUIRED BY TECHNICAL SPECIFICATION SECTION 6.13. SPECIAL TEST PROCEDURES TO VERIFY THE INTEGRITY OF THE SYSTEM WILL BE WRITTEN AND WILL BE PERFORMED. AN UPDATE TO THIS LER WILL BE SENT PRIOR TO MAY OF 1988. THE UNIT'S SURVEILLANCE PROCEDURES WILL BE UPDATED TO INCLUDE THE PROCEDURES MENTIONED ABOVE. THERE WERE NO SAFETY CONSEQUENCES AS A RESULT OF THIS EVENT.

[138] MILLSTONE 2 DOCKET 50-336 LER 87-012
REACTOR TRIP ON LOW NUMBER STEAM GENERATOR LEVEL, FAILURE OF NO. 1 FEED REGULATING VALVE POSITIONER.
EVENT DATE: 111687 REPORT DATE: 121187 NSSS: CE TYPE: PWR
VENDOR: COPES-DUICAN INC.
GENERAL ELECTRIC CO.
MOORE PRODUCTS COMPANY

(NSIC 207367) WHILE OPERATING AT 100% POWER ON NOVEMBER 16, 1987 AT 0211 THE UNIT EXPERIENCED AN AUTOMATIC REACTOR TRIP DUE TO A LOW STEAM GENERATOR LEVEL IN THE #1 STEAM GENERATOR. FOLLOWING TURBINE VALVE TESTING, THE OPERATOR RESPONDED TO AN INDICATED HIGH LEVEL IN #1 STEAM GENERATOR BY TAKING MANUAL CONTROL OF THE #1 FEEDWATER REGULATING VALVE. THE VALVE FAILED TO RESPOND TO ALL MANUAL CLOSED SIGNALS UNTIL A HIGH STEAM GENERATOR LEVEL INITIATED A RAMP CLOSING SIGNAL, CLOSING THE VALVE. THIS RESULTED IN A LOW STEAM GENERATOR LEVEL AND SUBSEQUENT REACTOR TRIP BEFORE RAMP OVERRIDE COULD BE INITIATED. ALL SAFETY SYSTEMS RESPONDED NORMALLY WITH THE EXCEPTION THAT THE 'A' AUXILIARY FEED PUMP (AFP) FAILED TO START ON THE AUTO AUXILIARY FEED ACTUATION SIGNAL AND WAS STARTED MANUALLY. THE 'SPRING-RETURN-TO-NORMAL', 'A' AUTO AUXILIARY FEED OVERRIDE SWITCH WAS IN AN INTERMEDIATE POSITION, PREVENTING THE START. OTHER EVENTS SUBSEQUENT TO THE REACTOR TRIP INCLUDED THE LOSS OF THE A AND C REACTOR COOLANT PUMPS (RCPS) WHEN THE 25A (6.9KV) BUS FAILED TO TRANSFER FROM THE NORMAL STATION SERVICE TRANSFORMER TO THE RESERVE STATION SERVICE TRANSFORMER. THIS WAS CAUSED BY A DAMAGED COMPONENT PREVENTING FULL RACKING IN OF THE BREAKER.

[139] MILLSTONE 3 DOCKET 50-423 LER 87-036
SETPOINT DRIFT ON MAIN STEAM SAFETY VALVES.
EVENT DATE: 103187 REPORT DATE: 112587 NSSS: WE TYPE: PWR
VENDOR: DRESSER INDUSTRIAL VALVE & INST DIV

(NSIC 207279) AT 1300 ON OCTOBER 31, 1987, WITH THE PLANT AT 0% POWER IN HOT STANDBY, PRESSURE 1000 PSIG AND A TEMPERATURE 500 DEGREES FAHRENHEIT, SETPOINT DRIFT WAS DISCOVERED ON A NUMBER OF MAIN STEAM SAFETY VALVES WHILE THE VALVES WERE BEING TESTED. TECHNICAL SPECIFICATIONS DID NOT REQUIRE ACTION TO BE TAKEN SINCE THE PLANT WAS IN HOT STANDBY. THEREFORE NO IMMEDIATE CORRECTIVE ACTION WAS REQUIRED. THE CAUSE OF THE SAFETY VALVE SETPOINT DRIFT IS UNKNOWN. THE SETPOINTS ON THE VALVES WERE RESET TO TECHNICAL SPECIFICATION REQUIREMENTS. ANY SAFETY VALVES THAT WERE NOT TESTED WILL BE TESTED OR REPLACED WITH CERTIFIED TESTED SPARES PRIOR TO RESUMING POWER OPERATIONS. NO FURTHER CORRECTIVE ACTION IS REQUIRED.

[140] MILLSTONE 3 DOCKET 50-423 LER 87-037
FEEDWATER ISOLATION DUE TO HIGH STEAM GENERATOR LEVEL CAUSED BY OPERATOR ERROR.
EVENT DATE: 110187 REPORT DATE: 113087 NSSS: WE TYPE: PWR

(NSIC 207379) ON 11/1/87, AT 1153 HOURS, WHILE IN MODE 3 DURING A PLANT COOLDOWN, A FEEDWATER ISOLATION (FWI) OCCURRED DUE TO HIGH LEVEL IN THE C STEAM GENERATOR. ALL THE FEEDWATER ISOLATION VALVES CLOSED ON THE FWI SIGNAL. THE C STEAM GENERATOR LEVEL WAS ALLOWED TO DECREASE TO ITS NORMAL OPERATING RANGE. AFTER THE FEEDWATER ISOLATION WAS RESET, MAIN FEEDWATER WAS PLACED BACK IN SERVICE. IMMEDIATELY PRIOR TO THE FWI, THE STEAM GENERATORS WERE BEING FED USING THE MOTOR DRIVEN FEEDWATER PUMP WHEN THE CONTROL ROOM OPERATOR SAW THAT THE LEVEL IN THE C STEAM GENERATOR WAS DECREASING. HE OPENED THE STEAM GENERATOR FEED LINE ISOLATION VALVE 3FWS*MOV35C IN ORDER TO ALLOW LEAKAGE FLOW PAST THE FEEDWATER FLOW CONTROL VALVE 3FWS*FCV530, WHICH WAS CLOSED AT THE TIME. ONCE THE ISOLATION VALVE WAS OPENED, THE LEVEL IN THE C STEAM GENERATOR RAPIDLY INCREASED, AND THE FWI OCCURRED. THE ROOT CAUSE OF THIS EVENT WAS OPERATOR ERROR. WHEN THE MAIN FEEDWATER SYSTEM IS IN OPERATION WHILE THE PLANT IS SHUTDOWN, FEEDWATER ADDITIONS SHOULD BE MADE VIA THE FEEDWATER BYPASS LEVEL CONTROL VALVES, INSTEAD OF RELYING

ON LEAKAGE PAST THE FLOW CONTROL VALVES; HOWEVER THERE IS NO PROCEDURE THAT SPECIFICALLY PROHIBITS THE USAGE OF THE FEEDWATER FLOW PATH THROUGH 3FWS*MOV35C, RELYING ON LEAKAGE THROUGH 3FWS*FCV530 WHILE THE PLANT IS SHUTDOWN. THERE HAS BEEN ONE OTHER FEEDWATER ISOLATION DUE TO OPERATOR ERROR DURING A PLANT SHUTDOWN.

[141] MILLSTONE 3 DOCKET 50-423 LER 87-040
FIRE PROTECTION SURVEILLANCE PERFORMED LATE DUE TO HUMAN ERROR.
EVENT DATE: 110987 REPORT DATE: 120987 NSSS: WE TYPE: PWR

(NSIC 207382) ON NOVEMBER 9, 1987 AT 0900, WITH THE PLANT IN COLD SHUTDOWN FOR REFUELING, 0% POWER, 90 DEGREES, ATMOSPHERIC PRESSURE, A SHIFT SUPERVISOR REVIEWING SURVEILLANCE REQUIREMENTS NOTED THAT A WEEKLY FIRE PROTECTION WATER VALVE LINEUP HAD NOT BEEN PERFORMED. THE CAUSE OF THE EVENT WAS HUMAN ERROR. IMMEDIATE ACTION WAS TO COMPLETE THE REQUIRED SURVEILLANCE. THERE WAS NO DANGER TO THE HEALTH OR SAFETY OF THE PUBLIC DUE TO THERE NEVER HAVING BEEN A FUNCTIONAL DEGRADATION OF THE FIRE PROTECTION SYSTEM. ALL SHIFT SUPERVISORS AND SUPERVISING CONTROL OPERATORS HAVE RECEIVED A WRITTEN NOTICE REEMPHASIZING ATTENTION TO DETAIL WHEN REVIEWING THE SURVEILLANCE SCHEDULE AND PERFORMING REVIEWS AND UPDATES TO THE SCHEDULE EACH SHIFT. THE DAY SHIFT SUPERVISOR HAS BEEN ASSIGNED ACCOUNTABILITY TO ENSURE THE SURVEILLANCES FOR ANY GIVEN WEEK HAVE BEEN COMPLETED ON SCHEDULE.

[142] MILLSTONE 3 DOCKET 50-423 LER 87-038
LOSS OF 4.16 KV VITAL BUS DUE TO MECHANICAL SHOCK.
EVENT DATE: 111087 REPORT DATE: 120287 NSSS: WE TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 207380) ON NOVEMBER 10, 1987 AT 1035 HOURS, WHILE THE PLANT WAS IN COLD SHUTDOWN (RCS AT 92 DEGREES AND ATMOSPHERIC PRESSURE) FOR REFUELING OPERATIONS, THE NORMAL SUPPLY BREAKER TO TRAIN 'A' VITAL 4.16 KV BUS 34C WAS TRIPPED OPEN. THE EDG-A SEQUENCER STRIPPED BUS 34C OF MOTOR LOADS, CLOSED THE EDG-A OUTPUT BREAKER (ENERGIZING THE BUS), AND ALL VITAL LOADS THAT WERE REQUIRED FOR THE EXISTING MODE OF OPERATION WERE SUCCESSFULLY SEQUENCED ON. TRAIN 'B' VITAL 4.16 KV BUS 34D, AND NON-VITAL BUSSES 34A AND 34B REMAINED ENERGIZED, UNAFFECTED BY THIS EVENT. SUBSEQUENT INVESTIGATION DETERMINED THAT THE EVENT RESULTED FROM A SUDDEN MECHANICAL SHOCK APPLIED TO A TIMING RELAY MOUNTED ON A SWITCHGEAR DOOR WHILE CLEANING THE AREA AROUND THE BUS SWITCHGEAR. OPERATIONS RESTORED THE NORMAL SUPPLY TO BUS 34C AND SHUTDOWN EDG-A BY 1115 ON 11/10/87. THE ROOT CAUSE OF THE EVENT WAS THAT THE RESTRICTIONS PLACED ON SWITCHGEAR ACCESS DURING NORMAL PLANT OPERATIONS WERE RELAXED. DURING THE OUTAGE, THE WARNING SIGNS AND TAPE NORMALLY IN PLACE WHEN THE PLANT IS OPERATING WERE MOVED TO FACILITATE EASY ACCESS FOR MAINTENANCE AND TEST ACTIVITIES. THE WARNING SIGNS AND TAPE HAVE BEEN RETURNED TO POSITION TO PRECLUDE RECURRENCE. ADDITIONALLY, ALL PLANT DEPARTMENTS HAVE BEEN BRIEFED AS TO THE EXISTENCE OF SENSITIVE RELAYS IN THE SWITCHGEAR AREAS.

[143] MILLSTONE 3 DOCKET 50-423 LER 87-039
FAILURE TO SAMPLE EMERGENCY DIESEL GENERATOR FUEL OIL TANKS FOR PARTICULATE.
EVENT DATE: 111187 REPORT DATE: 120987 NSSS: WE TYPE: PWR
VENDOR: COLT INDUSTRIES, INC.

(NSIC 207381) AN ADMINISTRATIVE REVIEW OF SURVEILLANCE PROCEDURES REVEALED THAT THE EMERGENCY DIESEL GENERATORS' (EDG) FUEL OIL TANKS WERE NOT ANALYZED FOR PARTICULATE CONTENT IN MARCH OF 1987. TECHNICAL SPECIFICATIONS REQUIRE THAT A PARTICULATE SAMPLE OF FUEL OIL BE OBTAINED EVERY 31 DAYS. A SAMPLE WAS TAKEN OF "B" EDG FUEL OIL TANK ON MARCH 5, 1987, BUT NO PARTICULATE ANALYSIS WAS EVER PERFORMED. ALSO, THERE IS NO RECORD INDICATING THAT A FUEL OIL SAMPLE WAS TAKEN FOR "A" EDG IN MARCH, 1987. CONTRIBUTING FACTORS THAT INDICATE WHY A PARTICULATE SAMPLE WAS NOT TAKEN, OR WHY AN ANALYSIS OF THE "B" EDG SAMPLE WAS NOT PERFORMED

INCLUDE IMPROPER SCHEDULING, AND DELIVERING SAMPLES TO CHEMISTRY WITHOUT OBTAINING A FORMAL RECEIPT. CORRECTIVE ACTION TAKEN TO PREVENT A RECURRENCE OF THE EVENT INCLUDE THE ISSUANCE OF A MEMORANDUM ON SURVEILLANCE SCHEDULE REVIEWS, SCHEDULE UPDATES AND PROPER DOCUMENTATION OF SURVEILLANCE PROBLEMS, AND A PROCEDURE CHANGE THAT REQUIRES RECEIPT SIGNATURES FOR FUEL OIL SAMPLES TAKEN TO CHEMISTRY.

[144] MONTICELLO DOCKET 50-263 LER 87-016
SGBTS DUE TO WRGM POWER LOSS DURING MODIFICATION.
EVENT DATE: 103087 REPORT DATE: 112987 NSSS: GE TYPE: BWR

(NSIC 207175) A REACTOR BUILDING VENTILATION ISOLATION AND STANDBY GAS TREATMENT SYSTEM INITIATION OCCURRED DURING A ROUTINE REFUELING OUTAGE AS A RESULT OF A PROCEDURAL INADEQUACY DURING THE PERFORMANCE OF A MODIFICATION PROCEDURE. THE CONTRACT ELECTRICIANS PERFORMING THE WORK INADVERTENTLY SHORTED AN ENERGIZED LEAD TO GROUND RESULTING IN A BLOWN FUSE. PERFORMANCE OF THE PROCEDURE WAS STOPPED, THE FUSE REPLACED, THE ISOLATION RESET AND THE STANDBY GAS TREATMENT SYSTEM SHUTDOWN. THE MODIFICATION PROCEDURE WAS REVISED TO INCLUDE THE INSTALLATION OF BYPASS JUMPERS TO PREVENT SUCH AN EVENT FROM RECURRING.

[145] MONTICELLO DOCKET 50-263 LER 87-018
ENGINEERED SAFETY FEATURE ACTUATIONS DUE TO INADVERTENT TRIP OF SPENT FUEL POOL MONITOR.
EVENT DATE: 111187 REPORT DATE: 121187 NSSS: GE TYPE: BWR

(NSIC 207400) DURING A ROUTINE MONTHLY AREA RADIATION MONITOR FUNCTIONAL CHECK, A PRIMARY CONTAINMENT GROUP II ISOLATION, A SECONDARY MONTHLY CONTAINMENT ISOLATION, AND A STANDBY GAS TREATMENT SYSTEM INITIATION OCCURRED DUE TO A HIGH LEVEL TRIP OF SPENT FUEL POOL MONITOR CHANNEL A. THE TRIP OCCURRED WHEN THE RADIATION PROTECTION SPECIALIST (R.P.S.) CONDUCTING THE TEST EXPOSED THE SPENT FUEL POOL MONITOR TO THE RADIATION CHECK SOURCE, BUT HE INCORRECTLY THOUGHT HE WAS EXPOSING AN AREA RADIATION MONITOR. THE R.P.S. REALIZED HIS MISTAKE IMMEDIATELY AND THE SITUATION WAS LATER DISCUSSED WITH HIS SUPERVISOR. HE WAS COUNSELED TO EXERCISE MORE CAUTION IN THE FUTURE. THE AREA RADIATION MONITOR TEST PROCEDURE IS BEING REVISED TO BETTER IDENTIFY LOCATION OF MONITORS ALONG WITH WORDS OF CAUTION NOT TO EXPOSE A WRONG MONITOR. THE TWO SPENT FUEL POOL MONITORS HAVE SINCE BEEN MARKED TO IDENTIFY THEM AS DIFFERENT FUEL FROM THE AREA RADIATION MONITORS.

[146] MONTICELLO DOCKET 50-263 LER 87-019
ENGINEERED SAFETY FEATURE ACTUATIONS DUE TO MISPLACED JUMPER DURING SURVEILLANCE TEST.
EVENT DATE: 111187 REPORT DATE: 121187 NSSS: GE TYPE: BWR

(NSIC 207293) WHILE PERFORMING A SPENT FUEL POOL MONITOR MONTHLY FUNCTIONAL TEST, A PRIMARY CONTAINMENT GROUP II ISOLATION, A SECONDARY CONTAINMENT ISOLATION AND A STANDBY GAS TREATMENT SYSTEM INITIATION OCCURRED DUE TO A HIGH LEVEL TRIP OF FUEL POOL MONITOR CHANNEL B. THE TRIP OF THE B CHANNEL IS A PLANNED PART OF THE SURVEILLANCE TEST; HOWEVER, THE PROCEDURE HAS A STEP FOR INSTALLING A JUMPER TO PREVENT THESE ENGINEERED SAFETY FEATURE (ESF) ACTUATIONS. THE LICENSED OPERATOR ASSISTING WITH THE TEST INADVERTENTLY PLACED THE JUMPER ACROSS TERMINALS B53 AND B54 IN THE LOGIC PANEL RATHER THAN TERMINALS B53 AND B55 AS CALLED FOR IN THE PROCEDURE. THIS ALLOWED THE ESF ACTUATIONS TO OCCUR WHEN THE UPSCALE TRIP CHECK WAS PERFORMED. THE PROCEDURE DID NOT REQUIRE INDEPENDENT VERIFICATION OF JUMPER PLACEMENT. THE ESF ACTUATIONS WERE IMMEDIATELY RESET AND THE TEST WAS COMPLETED. AN ADMINISTRATIVE HOLD WAS PLACED ON THE PROCEDURE TO ASSURE INDEPENDENT VERIFICATION OF JUMPER PLACEMENT. A MODIFICATION WAS MADE THAT ELIMINATED THE NEED FOR USING JUMPERS FOR THIS ROUTINE TEST.

[147] NINE MILE POINT 1 DOCKET 50-220 LER 87-016
REACTOR SCRAM, TURBINE TRIP, AND HIGH PRESSURE COOLANT INJECTION MODE OF FEEDWATER SIGNALS DUE TO SPURIOUS TRIP OF NEUTRON MONITOR CAUSED BY NOISE.
EVENT DATE: 101987 REPORT DATE: 111887 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 207165) ON OCTOBER 19, 1987, THE NINE MILE POINT UNIT 1 REACTOR WAS IN A SHUTDOWN CONDITION AND BEING PREPARED FOR A VESSEL HYDROSTATIC TEST. THE REACTOR WAS AT 150 PSIG AND 173 DEGREES F. THE MODE SWITCH WAS IN THE "REFUEL" POSITION AND ALL CONTROL RODS WERE FULLY INSERTED. AT 1111 HOURS THE UNIT EXPERIENCED A FULL REACTOR SCRAM, TURBINE TRIP SIGNAL ACTUATION, AND HIGH PRESSURE COOLANT INJECTION (HPCI) MODE OF FEEDWATER INITIATION LOGIC ACTUATION. THE ROOT CAUSE OF THE SCRAM WAS SPURIOUS ACTUATION OF AN INTERMEDIATE RANGE MONITOR (IRM) TRIP ON REACTOR PROTECTION SYSTEM (RPS) CHANNEL 12 DUE TO NOISE. IN THIS CASE A ONE-HALF SCRAM WAS ALREADY PRESENT ON RPS CHANNEL 11 DUE TO A SURVEILLANCE BEING CONDUCTED ON THE MAIN STEAM LINE RADIATION MONITOR #111. NO ADVERSE SAFETY CONSEQUENCES RESULTED FROM THIS EVENT. THERE WAS NO FEEDWATER INJECTION INITIATED BECAUSE THE HPCI FEEDWATER PUMPS WERE LOCKED OUT. IMMEDIATE CORRECTIVE ACTIONS INCLUDED RESETTING THE SCRAM, TURBINE TRIP, AND HPCI MODE OF FEEDWATER LOGIC. SUBSEQUENT CORRECTIVE ACTIONS INVOLVED CONDUCTING A POST-SCRAM REVIEW AND INITIATING A FURTHER INVESTIGATION INTO NOISE ON IRM CHANNELS VIA A PROBLEM REPORT.

[148] NINE MILE POINT 1 DOCKET 50-220 LER 87-017
INCORRECT SYSTEM PIPING DESIGN SPECIFICATION RESULTED IN PORTION OF RAW WATER TO CORE SPRAY INTERTIE PIPING RECEIVING HYDROSTATIC TEST IN INCORRECT PRESSURE.
EVENT DATE: 102087 REPORT DATE: 111987 NSSS: GE TYPE: BWR

(NSIC 207166) ON AUGUST 31, 1987, THE NINE MILE POINT UNIT 1 INSERVICE INSPECTION DEPARTMENT INITIATED A DEFICIENCY/CORRECTIVE ACTION NOTICE (DCA) IDENTIFYING A SECTION OF RAW WATER TO CORE SPRAY INTERTIE PIPING WHICH HAD NOT BEEN HYDROSTATICALLY TESTED AS PART OF THE ASME SECTION XI 10 YEAR INSPECTION INTERVAL. ENGINEERING, IN RESPONSE TO THIS DCA ON OCTOBER 20, 1987, IDENTIFIED TWO SECTIONS OF THIS PIPING WHOSE DESIGN SPECIFICATIONS WERE INCORRECT. ASME SECTION XI REQUIRES THAT THESE PORTIONS OF PIPING BE HYDROSTATICALLY TESTED AT 590 PSIG. HOWEVER, DUE TO THE INCORRECT DESIGN SPECIFICATIONS, THIS PIPING WAS ONLY TESTED TO A MINIMUM PRESSURE OF 330 PSIG. THIS IS A VIOLATION OF TECHNICAL SPECIFICATIONS, SECTION 3.2.6. NINE MILE POINT UNIT 1 WAS SHUTDOWN AT THE TIME THE EVENT WAS DISCOVERED. NO ACTUAL OR POTENTIAL SAFETY CONSEQUENCES RESULTED. THE ROOT CAUSE WAS IDENTIFIED AS AN ENGINEERING DESIGN ERROR WHEN LOCATING NEW SYSTEM BOUNDARIES WHICH RESULTED FROM A MODIFICATION PERFORMED IN 1983. CORRECTIVE ACTIONS INCLUDE RECLASSIFICATION OF PIPING TO THE PROPER DESIGN PRESSURE AND TEMPERATURE, REVISION OF HYDRO PROCEDURES TO INCORPORATE THE CHANGE IN SYSTEM BOUNDARIES AND PERFORMANCE OF A HYDRO TEST DURING THE 1988 REFUELING OUTAGE.

[149] NINE MILE POINT 1 DOCKET 50-220 LER 87-019
TECHNICAL SPECIFICATION VIOLATION DUE TO PERSONNEL ERROR IN NOT IDENTIFYING FIRE BARRIER PENETRATIONS.
EVENT DATE: 102187 REPORT DATE: 112087 NSSS: GE TYPE: BWR

(NSIC 207168) AT 2300 HOURS ON OCTOBER 21, 1987, WITH NINE MILE POINT UNIT 1 IN STARTUP, FIRE DEPARTMENT PERSONNEL DISCOVERED FIVE AREAS IN BATTERY AND BATTERY BOARD ROOMS 11 AND 12, WHERE UNPROTECTED OPENINGS EXISTED BETWEEN RATED CEILING AND FLOOR ASSEMBLIES AND A NON-RATED WALL. THESE ARE IN TWO HOUR RATED BARRIERS AND HAVE BEEN IDENTIFIED AS BEING APPLICABLE TO TECHNICAL SPECIFICATION SECTION 3.6.10.1. IN ADDITION, SINCE THESE DEFICIENCIES WERE RECENTLY DISCOVERED, A FIRE WATCH PATROL HAD NOT BEEN PREVIOUSLY ESTABLISHED NOR HAD THE CYCLIC SURVEILLANCE TEST BEEN PERFORMED. THESE EVENTS CONSTITUTED A VIOLATION OF NMP1 TECHNICAL SPECIFICATION SECTION 3.6.10.1 AND 4.6.10.1. THE ROOT CAUSE OF THIS EVENT WAS

THE FACT THAT THE DEFICIENCIES IN THESE AREAS WERE NOT IDENTIFIED DURING PREVIOUS INSPECTIONS. A CONTRIBUTING FACTOR IS THE BARRIERS WERE PARTIALLY SEALED AND HIDDEN FROM THE LINE OF SIGHT. CORRECTIVE ACTIONS CONSISTED OF IMMEDIATELY VERIFYING DETECTION IN THE AFFECTED AREA AND ESTABLISHING A FIRE WATCH PATROL. SUBSEQUENTLY, STATION WORK REQUESTS WERE INITIATED TO PROPERLY SEAL THE FIVE AREAS.

[150] NINE MILE POINT 1 DOCKET 50-220 LER 87-018
PROCEDURE DELETION IN VIOLATION OF TECHNICAL SPECIFICATIONS DUE TO PERSONNEL ERROR.
EVENT DATE: 102287 REPORT DATE: 112087 NSSS: GE TYPE: BWR

(NSIC 207167) ON OCTOBER 22, 1987, WITH NINE MILE POINT UNIT 1 AT 0% POWER AND IN PREPARATION FOR STARTUP WITH THE MODE SWITCH IN STARTUP, A TRAINING DEPARTMENT INSTRUCTOR NOTICED THAT THE 115 KV POWER FAILURE SPECIAL OPERATING PROCEDURE WAS NOT INCLUDED IN THE PROCEDURE INDEX. A CHECK BY THE OPERATIONS DEPARTMENT ASSISTANT SUPERINTENDENT REVEALED THAT SPECIAL OPERATING PROCEDURE N1-SOP-5, "115 KV POWER FAILURE", HAD BEEN REMOVED FROM THE CONTROL ROOM MASTER PROCEDURE FILE WITHOUT AN ADMINISTRATIVE PROCEDURE 2.0, "PRODUCTION AND CONTROL OF PROCEDURES" REVIEW. THIS IS A VIOLATION OF TECHNICAL SPECIFICATION 6.8, WHICH REQUIRES PROCEDURE CHANGES TO BE APPROVED BY THE GENERAL SUPERINTENDENT. THE INITIAL CORRECTIVE ACTION TAKEN WAS TO OBTAIN AND INSERT A CONTROLLED COPY OF THE PROCEDURE INTO THE CONTROL ROOM MASTER FILE, AND TO INSTRUCT THE PROCEDURES DEPARTMENT TO DISTRIBUTE A COPY TO ALL CONTROLLED COPY HOLDERS. THE ROOT CAUSE OF THE EVENT IS ATTRIBUTED TO PERSONNEL ERROR BY THE CLERICAL STAFF OF THE PROCEDURES DEPARTMENT DURING THE DISTRIBUTION OF THE NEW SYMPTOMATIC SPECIAL OPERATING PROCEDURES AND EMERGENCY OPERATING PROCEDURES PRIOR TO THE END OF THE 1986 REFUELING OUTAGE. THE PROCEDURES DEPARTMENT HAS BEEN ADVISED OF THE NEED FOR COMPLIANCE TO PROCEDURE AND ATTENTION TO DETAIL WHILE PERFORMING JOB DUTIES.

[151] NINE MILE POINT 1 DOCKET 50-220 LER 87-020
MISSED FIRE WATCH PATROL DUE TO PERSONNEL ERROR RESULTING IN VIOLATION OF TECHNICAL SPECIFICATION.
EVENT DATE: 102787 REPORT DATE: 112587 NSSS: GE TYPE: BWR

(NSIC 207252) ON OCTOBER 27, 1987, WHILE THE NINE MILE POINT UNIT 1 NUCLEAR STATION WAS OPERATING AT 94% POWER, THE TECHNICAL SPECIFICATION REQUIREMENT OF MAINTAINING A HOUR FIRE WATCH PATROL FOR A NONFUNCTIONAL FIRE BARRIER WAS EXCEEDED. THE EVENT WAS CAUSED BY A COGNITIVE ERROR ON THE PART OF INDIVIDUALS PERFORMING THE FIRST AND SECOND FIRE WATCH PATROLS FOR THE ONCOMING SHIFT IMMEDIATELY AFTER SHIFT TURNOVER. THE FIRE DETECTION SYSTEM REMAINED OPERABLE THROUGHOUT THIS EVENT, ALONG WITH AUTOMATIC AND MANUAL SUPPRESSION IN THE AREA ADJACENT TO THE NONFUNCTIONAL BARRIER. AS A RESULT, THERE WAS NO SIGNIFICANT IMPACT ON PLANT SAFETY. CORRECTIVE ACTION INVOLVED IMMEDIATELY DISCUSSING THE EVENT WITH THE INDIVIDUALS TO DETERMINE ANY DEFICIENCIES IN THE METHODS UTILIZED IN CONTROLLING THIS ACTIVITY. ADDITIONALLY, A NEW FIRE DEPARTMENT PROCEDURE GOVERNING FIRE WATCH PATROL ACTIVITIES WAS ISSUED ON OCTOBER 29, 1987. THE REQUIREMENTS SET FORTH IN THIS PROCEDURE SHOULD PRECLUDE RECURRENCE OF AN EVENT OF THIS TYPE.

[152] NINE MILE POINT 1 DOCKET 50-220 LER 87-021
FAILURE TO REDUCE REACTOR POWER BELOW TECHNICAL SPECIFICATION LIMIT PRIOR TO ISOLATING RECIRCULATION LOOP.
EVENT DATE: 102787 REPORT DATE: 112587 NSSS: GE TYPE: BWR

(NSIC 207253) ON OCTOBER 27, 1987, WITH NINE MILE POINT UNIT 1 OPERATING AT 98.5% POWER, A REACTOR RECIRCULATION LOOP WAS TAKEN FROM THE UNISOLATED TO THE FULLY ISOLATED CONDITION. THE PLANT TECHNICAL SPECIFICATIONS (TECH. SPECS.) STATE

THAT REACTOR POWER SHALL NOT EXCEED 90.5% IF NEITHER THE FULLY ISOLATED OR UNISOLATED CONDITION CAN BE MET TO PRECLUDE THE EFFECTS OF AN INADVERTENT COLD RECIRCULATION LOOP INITIATION. THIS CONDITION WAS NOT MET FOR 37 MINUTES WHILE TRANSITIONING BETWEEN THE UNISOLATED AND ISOLATED LOOP CONFIGURATION. A PRELIMINARY INTERPRETATION OF THE TECH. SPEC. BY STATION AND LICENSING STAFF HAD CONCLUDED THAT A POWER REDUCTION BELOW 90.5% WAS NOT NECESSARY AS A COLD WATER INJECTION COULD NOT OCCUR. THE RECIRCULATION PUMP IN THAT LOOP COULD NOT BE STARTED AND THE WATER CONTAINED IN THAT LOOP WOULD NOT HAVE TIME TO COOL APPRECIABLY DURING THE TRANSITION FROM THE UNISOLATED TO FULLY ISOLATED CONDITION. ON OCTOBER 28, 1987, THE WRITTEN INTERPRETATION WAS RECEIVED FROM LICENSING WHICH STATED THAT REACTOR POWER SHALL NOT EXCEED 90.5% POWER WHEN TRANSITIONING FROM AN UNISOLATED TO ISOLATED CONDITION. BASED ON THIS INTERPRETATION, THE ACTIVITIES OF THE PREVIOUS DAY REPRESENTED A VIOLATION OF TECHNICAL SPECIFICATION AND ARE, THEREFORE, REPORTABLE UNDER THE REQUIREMENTS OF 10 CFR 50.73.

[153] NINE MILE POINT 2 DOCKET 50-410 LER 87-049 REV 01
UPDATE ON TWO STANDBY GAS TREATMENT SYSTEM INITIATIONS DUE TO A LOW FLOW CONDITION.
EVENT DATE: 082587 REPORT DATE: 112087 NSSS: GE TYPE: BWR

(NSIC 207211) ON AUGUST 25, 1987 AT 0645 WITH THE REACTOR AT APPROXIMATELY 22% POWER AND THE MODE SWITCH IN "RUN", NINE MILE POINT PIT 2 (NMP2) EXPERIENCED AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION. THIS EVENT CONSISTED OF AN AUTOMATIC INITIATION OF THE STANDBY GAS TREATMENT SYSTEM (SBGTS) AND AN EMERGENCY RECIRCULATION UNIT COOLER. AT 0834 THE SAME DAY, WITH THE REACTOR AT APPROXIMATELY 22% POWER AND THE MODE SWITCH IN "RUN", NMP2 EXPERIENCED A SECOND ESF ACTUATION. THIS EVENT ALSO CONSISTED OF AN AUTOMATIC INITIATION OF THE SBGTS SYSTEM AND AN EMERGENCY RECIRCULATION UNIT COOLER. BOTH EVENTS OCCURRED WHILE ATTEMPTING TO RESTORE NORMAL REACTOR BUILDING VENTILATION. THE CAUSE OF THE FIRST EVENT IS ATTRIBUTED TO A COGNITIVE PERSONNEL ERROR. AFTER A THOROUGH INVESTIGATION, THE CAUSE OF THE SECOND EVENT COULD NOT BE DETERMINED. CORRECTIVE ACTIONS HAVE BEEN TO: 1. CONDUCT AN INDEPENDENT EVALUATION OF THE SBGTS AUTOMATIC INITIATION LOGIC. 2. TRAIN NMP2 OPERATORS EXCLUDING SENIOR REACTOR OPERATORS (SRO), ON THE REACTOR BUILDING (RB) VENTILATION SYSTEM.

[154] NINE MILE POINT 2 DOCKET 50-410 LER 87-052 REV 01
UPDATE ON MISSED SURVEILLANCE FOR STANDBY GAS TREATMENT SYSTEM TRAIN A WITH DIVISION II DIESEL GENERATOR INOPERABLE RESULTS IN INITIATION OF REQUIRED PLANT SHUTDOWN.
EVENT DATE: 100287 REPORT DATE: 113087 NSSS: GE TYPE: BWR

(NSIC 207245) ON 10/2/87 TWO RELATED TECH SPEC (TS) VIOLATIONS OCCURRED AT NINE MILE POINT UNIT 2. THE FIRST WAS A RESULT OF EXCEEDING A SURVEILLANCE REQUIREMENT OF A STANDBY GAS TREATMENT SYSTEM (GTS) CHARCOAL ADSORBER (TS 4.6.5.3.C). THE SECOND WAS A RESULT OF EXCEEDING A LIMITING CONDITION OF OPERATION (LCO) FOR THE AC POWER SOURCES (TS 3.8.1.1 ACTION E). AT THE TIME OF THE EVENT DISCOVERY THE REACTOR WAS AT APPROXIMATELY 40% RATED THERMAL POWER. AFTER THE EVENT HAD BEEN ANALYZED A THIRD TS VIOLATION (TS 6.8.1.D) WAS ALSO DISCOVERED. THE CAUSE OF THE EVENT WAS DETERMINED TO BE INADEQUATE PROCEDURAL CONTROL OF THE CUMULATIVE OPERATING HOURS STATUS OF GTS CHARCOAL ADSORBERS. CONTRIBUTING TO THE EVENT WAS THE UNTIMELY COMPLETION OF A REVIEW TO DETERMINE THE CUMULATIVE GTS OPERATING HOURS AND THE FAILURE TO NOTIFY THE STATION SHIFT SUPERVISOR THAT THE STATUS OF A TS REQUIRED SURVEILLANCE WAS IN QUESTION. CORRECTIVE ACTIONS INCLUDE THE FOLLOWING: 1. AN INADEQUATE PROCEDURE HAS BEEN REVISED AND A NEW PROCEDURE HAS BEEN GENERATED TO TRACK GTS CUMULATIVE OPERATING TIME ON A DAILY BASIS. 2. A MODIFICATION TO ADD ELAPSED TIME METERS TO ALL SPECIAL FILTER TRAINS, INCLUDING THE GTS, HAS BEEN COMPLETED. 3. ADMINISTRATIVE

CONTROLS SMALL BE REVIEWED AND CHANGED, AS REQUIRED, TO PRECLUDE RECURRENCE OF SIMILAR EVENT.

[155] NINE MILE POINT 2 DOCKET 50-410 LER 87-062
FAILURE OF PLANT PERSONNEL TO RECOGNIZE THAT A SURVEILLANCE TEST DID NOT SATISFY ITS ACCEPTANCE CRITERIA RESULTS IN A TECHNICAL SPECIFICATION VIOLATION.
EVENT DATE: 100887 REPORT DATE: 112487 NSSS: GE TYPE: BWR

(NSIC 207273) ON OCTOBER 27, 1987 AT 1545 HOURS A LIMITING CONDITION FOR OPERATION AS DEFINED BY TECHNICAL SPECIFICATIONS (TS) WAS FOUND TO HAVE BEEN VIOLATED AT NINE MILE POINT UNIT 2. THE TS WHICH WAS VIOLATED STATES: "ENTRY INTO AN OPERATIONAL CONDITION OR OTHER SPECIFIED CONDITION SHALL NOT BE MADE UNLESS THE CONDITIONS FOR THE LIMITING CONDITIONS FOR OPERATION ARE MET WITHOUT RELIANCE ON PROVISIONS CONTAINED WITHIN THE ACTION REQUIREMENTS." AT THE TIME OF DISCOVERY OF THE EVENT THE PLANT WAS IN THE COLD SHUTDOWN CONDITION WITH THE MODE SWITCH IN THE "SHUTDOWN" POSITION. REACTOR PRESSURE AND TEMPERATURE WERE AT APPROXIMATELY 0 POUNDS PER SQUARE INCH GAUGE AND 124 DEGREES FAHRENHEIT, RESPECTIVELY. THE CAUSE OF THE EVENT WAS COGNITIVE PERSONNEL ERROR WHEN SEVERAL NIAGARA MOHAWK PERSONNEL SIGNED A COMPLETED LOW PRESSURE CORE SPRAY (LPCS) SURVEILLANCE PROCEDURE AS SATISFACTORY WHEN THE DATA RECORDED WITHIN THE PROCEDURE DID NOT MEET ITS TS "ACCEPTANCE CRITERIA". CONTRIBUTING TO THE EVENT WERE SEVERAL PROCEDURAL DEFICIENCIES AND PERSONNEL ERROR. IMMEDIATE CORRECTIVE ACTION WAS TO DECLARE THE LPCS PUMP INOPERABLE AND REPREFORM THE SURVEILLANCE TEST. ADDITIONAL CORRECTIVE ACTIONS HAVE BEEN IMPLEMENTED TO CORRECT THE MINOR PROCEDURAL DEFICIENCIES.

[156] NINE MILE POINT 2 DOCKET 50-410 LER 87-066
TECHNICAL SPECIFICATION VIOLATION DUE TO A MODE CHANGE WHILE THE STANDBY LIQUID CONTROL SYSTEM WAS INOPERABLE.
EVENT DATE: 102087 REPORT DATE: 111987 NSSS: GE TYPE: BWR

(NSIC 207214) ON OCTOBER 20, 1987 A LIMITING CONDITION FOR OPERATION (LCO) AS DEFINED BY TECHNICAL SPECIFICATION 3.0.4 WAS FOUND TO HAVE BEEN VIOLATED AT NINE MILE POINT UNIT 2 (NMP2). ON OCTOBER 17, 1987, FUSES WERE REPLACED IN THE STANDBY LIQUID CONTROL SYSTEM (SLS) FOLLOWING PERFORMANCE OF A SURVEILLANCE TEST. ON OCTOBER 18, 1987 THE REACTOR OPERATING MODE WAS CHANGED FROM MODE 4 TO MODE 2 FOR REACTOR STARTUP FOLLOWING A SHORT OUTAGE. TWO DAYS LATER ON OCTOBER 20, 1987 AT 0910 IT WAS DISCOVERED THAT INCORRECT FUSES FOR THE DIVISION I SLS WERE INSTALLED AND THE DIVISION I SLS WAS DECLARED INOPERABLE. THE LCO VIOLATION OCCURRED WHEN THE REACTOR OPERATING MODE WAS CHANGED FROM MODE 4 TO MODE 2 WITH THE DIVISION I SLS INOPERABLE. AT THE TIME OF DISCOVERY NMP2 WAS AT APPROXIMATELY 15% POWER WITH THE MODE SWITCH IN THE "RUN" POSITION. THE CAUSE OF THE EVENT HAS BEEN DETERMINED TO BE A DESIGN DOCUMENT DEFICIENCY. INCORRECT PART NUMBERS WERE ASSIGNED TO THE SLS SQUIB VALVE CONTROL POWER FUSES. THIS ERROR RESULTED IN THE ELEMENT OF INCORRECT FUSES IN THE SQUIB VALVE CIRCUIT. INITIAL CORRECTIVE ACTION WAS TO INSTALL THE CORRECT FUSES. A FIELD DEVIATION DISPOSITION REQUEST (FDDR) HAS BEEN ISSUED TO ASSIGN THE CORRECT PART NUMBER TO THE FUSES.

[157] NINE MILE POINT 2 DOCKET 50-410 LER 87-068
REACTOR WATER CLEANUP ISOLATION DUE TO PERSONNEL ERROR AND INATTENTION TO DETAIL.
EVENT DATE: 102087 REPORT DATE: 111987 NSSS: GE TYPE: BWR

(NSIC 207216) ON OCTOBER 20, 1987 AT 1854 HOURS, NINE MILE POINT UNIT 2 (NMP2) EXPERIENCED THE ACTUATION OF AN ENGINEERED SAFETY FEATURE (ESF), SPECIFICALLY, ISOLATION OF THE REACTOR WATER CLEANUP (RWCU) SYSTEM. AT THE TIME OF THE EVENT, THE PLANT WAS OPERATING AT APPROXIMATELY 9% OF ITS RATED THERMAL POWER WITH THE REACTOR MODE SWITCH IN THE "RUN" POSITION. REACTOR PRESSURE AND COOLANT

TEMPERATURE WERE APPROXIMATELY 921 POUNDS PER SQUARE INCH GAUGE (PSIG) AND 535F, RESPECTIVELY. THE ROOT CAUSE OF THIS EVENT WAS COGNITIVE PERSONNEL ERROR; FAILURE TO FOLLOW PROCEDURE. THIS FAILURE TO FOLLOW PROCEDURE WAS DUE TO AN INATTENTION TO DETAIL. IMMEDIATE CORRECTIVE ACTION FOR THE EVENT WAS FOR THE OPERATORS TO VERIFY THE AUTOMATIC RESPONSE OF THE RWCU SYSTEM, VERIFY THE PLANT STATUS AS NORMAL, RESET THE ISOLATION SIGNAL AND RESTORE THE RWCU SYSTEM TO SERVICE. ADDITIONAL CORRECTIVE ACTIONS FOR THIS EVENT ARE: 1. THE TECHNICIAN INVOLVED HAS BEEN COUNSELED. 2. A TRAINING MODIFICATION RECOMMENDATION HAS BEEN INITIATED TO ADDRESS THIS ISSUE DURING INSTRUMENT AND CONTROL (I&C) CONTINUING TRAINING. 3. THE EVENT HAS BEEN DISCUSSED IN THE I&C MONTHLY DEPARTMENTAL SAFETY MEETINGS. 4. THE I&C LESSONS LEARNED PROGRAM WILL BE REVISED TO PROVIDE MORE RAPID DISSEMINATION OF INFORMATION TO THE I&C DEPARTMENT.

[158] NINE MILE POINT 2 DOCKET 50-410 LER 87-060
FAILURE TO PERFORM SHIFT CHECKS RESULTS IN MISSED TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENTS DUE TO PERSONNEL ERRORS.
EVENT DATE: 102287 REPORT DATE: 112087 NSSS: GE TYPE: BWR

(NSIC 207212) ON OCTOBER 22, 1987 AT 1750 HOURS SEVEN TECHNICAL SPECIFICATION (TS) SURVEILLANCE TIME REQUIREMENTS WERE FOUND TO HAVE BEEN EXCEEDED AT NINE MILE POINT UNIT 2. THE SURVEILLANCE TIME REQUIREMENTS WERE EXCEEDED AS A RESULT OF THE FAILURE TO PERFORM TWELVE TS REQUIRED HOUR INSTRUMENT CHANNEL SURVEILLANCE CHECKS. AT THE TIME OF THE DISCOVERY THE REACTOR MODE SWITCH WAS IN THE "RUN" POSITION WITH THE PLANT AT APPROXIMATELY 38 PERCENT POWER. THE CAUSE OF THE TS VIOLATIONS WAS COGNITIVE PERSONNEL ERRORS. THE OPERATOR FAILED TO COMPLETE 12 SURVEILLANCE SHIFT CHECKS DURING HIS NORMAL DAILY DUTIES. IN ADDITION, THE STATION SHIFT SUPERVISOR (SSS) FAILED TO PAY PROPER ATTENTION TO DETAIL WHEN HE DID NOT ADEQUATELY REVIEW THE SHIFT CHECK LOGS BEFORE APPROVING THEM. CORRECTIVE ACTIONS FOR THIS EVENT ARE: 1. THE 12 SHIFT CHECKS WERE IMMEDIATELY PERFORMED BY THE FOLLOWING SHIFT WHEN DISCOVERED. 2. THE INDIVIDUALS INVOLVED WITH THIS EVENT PARTICIPATED IN THE LER INVESTIGATION, FOLLOW-UP REPORT, AND CORRECTIVE ACTIONS. 3. A TRAINING MODIFICATION RECOMMENDATION HAS BEEN INITIATED TO ADDRESS THIS ISSUE. 4. A MEMO WAS WRITTEN TO ALL OPERATIONS PERSONNEL ADDRESSING THIS EVENT AND OTHER EVENTS INVOLVING PERSONNEL ERROR, AS WELL AS CORRECTIVE ACTIONS TO PREVENT RECURRENCE OF THESE EVENTS.

[159] NINE MILE POINT 2 DOCKET 50-410 LER 87-064
TURBINE TRIP AND REACTOR SCRAM ON LOW CONDENSER VACUUM DUE TO IMPROPER TAGGING.
EVENT DATE: 102387 REPORT DATE: 112087 NSSS: GE TYPE: BWR

(NSIC 207213) ON OCTOBER 23, 1987 AT 1142 HOURS THE TURBINE TRIPPED ON LOW CONDENSER VACUUM, RESULTING IN A REACTOR SCRAM FROM 36% POWER. A FEEDWATER PUMP WAS TAGGED OUT OF SERVICE FOR SEAL REPAIR WORK, BUT TWO MANUAL VALVES IN THE MINIMUM FLOW LINE WERE INADVERTENTLY OMITTED FROM THE MARKUP. DUE TO AN IMPROPER REVIEW OF THIS MARKUP, THE CONTROL ROOM OPERATOR GAVE PERMISSION FOR A RELAY TO BE REMOVED FOR TROUBLESHOOTING IN THE MINIMUM FLOW VALVE CONTROL CIRCUIT. REMOVAL OF THE RELAY CAUSED THE VALVE TO FAIL OPEN, PROVIDING A PATH FOR AIR INLEAKAGE THROUGH THE PUMP BODY TO THE CONDENSER. THE LOW CONDENSER VACUUM INSTRUMENTATION INITIATED A TURBINE TRIP, WHICH SUBSEQUENTLY LED TO A REACTOR SCRAM. THE ROOT CAUSE OF THE EVENT IS COGNITIVE PERSONNEL ERROR. IMMEDIATE CORRECTIVE ACTIONS WERE TO RESET THE SCRAM AT 1206 HOURS, TO FOLLOW THE SCRAM RECOVERY PROCEDURE FOR SAFE SHUTDOWN OF THE PLANT, AND TO INVESTIGATE THE CAUSE FOR THE LOW VACUUM. FURTHER CORRECTIVE ACTIONS TAKEN WERE COUNSELING FOR THE INDIVIDUAL RESPONSIBLE FOR REVIEWING THE MARKUP, ISSUANCE OF A MEMO TO OPERATIONS PERSONNEL FROM THE OPERATIONS SUPERINTENDENT EMPHASIZING ATTENTION TO DETAIL, AND FURTHER TRAINING FOR OPERATORS VIA CONTINUED TRAINING AND THE LESSONS LEARNED PROGRAM.

[160] NINE MILE POINT 2 DOCKET 50-410 LER 87-067
LIMITING CONDITION FOR OPERATION EXCEEDED DUE TO TECHNICAL SPECIFICATION MISINTERPRETATION.
EVENT DATE: 102387 REPORT DATE: 112087 NSSS: GE TYPE: BWR
VENDOR: ROSEMOUNT, INC.

(NSIC 207215) ON OCTOBER 23, 1987 AT 1130 HOURS IT WAS DISCOVERED THAT A LIMITING CONDITION FOR OPERATION (LCO) AS DEFINED BY TECHNICAL SPECIFICATION (TS) 3.3.2 HAD BEEN EXCEEDED AT NINE MILE POINT UNIT 2 (NMP2). AT THE TIME OF THE DISCOVERY NMP2 WAS AT APPROXIMATELY 36% POWER WITH THE MODE SWITCH IN "RUN". THE LCO WAS EXCEEDED DUE TO MAIN STEAM LINE HIGH FLOW ISOLATION INSTRUMENTATION (E31-N686D) BEING INOPERABLE FOR TWO HOURS AND FORTY-FIVE MINUTES WITHOUT OBSERVING THE PROPER TS ACTION ITEM. THE CAUSE OF THIS EVENT WAS A COGNITIVE PERSONNEL ERROR DUE TO A MISINTERPRETATION OF TS 3.3.2. INITIAL CORRECTIVE ACTION WAS TO PERFORM THE APPLICABLE ACTION STATEMENT REQUIREMENTS OF TS 3.3.2. THIS CONSISTED OF INSERTING A ONE-HALF MAIN STEAM ISOLATION VALVE (MSIV) TRIP SIGNAL. ADDITIONAL CORRECTIVE ACTIONS WILL INCLUDE INCORPORATING THIS EVENT INTO THE OPERATIONS DEPARTMENT LESSONS LEARNED PROGRAM AND A MEMORANDUM WAS ISSUED BY THE OPERATIONS SUPERINTENDENT TO ALL OPERATIONS DEPARTMENT PERSONNEL ON PERSONNEL ERROR. THE MEMORANDUM DISCUSSES THIS EVENT AND OTHERS AND REINFORCES THE REQUIREMENTS FOR ATTENTION TO DETAIL AND TS ADHERENCE.

[161] NINE MILE POINT 2 DOCKET 50-410 LER 87-069
LOSS OF POWER TO A NON-CLASS 1E UNINTERRUPTIBLE POWER SUPPLY RESULTS IN ESF ACTUATIONS DUE TO DESIGN DEFICIENCY AND PERSONNEL ERROR.
EVENT DATE: 102387 REPORT DATE: 112087 NSSS: GE TYPE: BWR
VENDOR: SHAWMUT COMPANY

(NSIC 207217) WHILE IN MODE 3 ON OCTOBER 23, 1987 AT 1436 HOURS, A LOSS OF POWER TO A NON-CLASS 1E UNINTERRUPTIBLE POWER SUPPLY (UPS) RESULTED IN NUMEROUS ENGINEERED SAFETY FEATURE (ESF) ACTUATIONS. A NIAGARA MOHAWK LICENSED OPERATOR WAS RESPONDING TO A UPS3B TROUBLE ALARM AND UNKNOWINGLY TRANSFERRED POWER FROM THE NORMAL TO THE ALTERNATE POWER SUPPLY. THE ALTERNATE SUPPLY WAS INOPERABLE DUE TO A BLOWN FUSE. THUS, UPS3B MOMENTARILY LOST POWER, RESULTING IN VARIOUS SYSTEM ISOLATIONS. THE CAUSES OF THIS EVENT ARE DESIGN DEFICIENCY, PERSONNEL ERROR, AND EQUIPMENT FAILURE. CONTRIBUTING CAUSES ARE LACK OF TRAINING AND PROCEDURAL AND HUMAN-FACTORS DEFICIENCIES. IMMEDIATE CORRECTIVE ACTIONS WERE TO RESTORE POWER, TO RESET SYSTEM ISOLATION SIGNALS AND TO BEGIN RETURNING ISOLATED SYSTEMS BACK TO SERVICE. WHILE ATTEMPTING TO RESTORE THE REACTOR WATER CLEANUP SYSTEM (WCS), THE SYSTEM ISOLATED AGAIN ON HIGH DIFFERENTIAL FLOW. THE CAUSE FOR THIS ISOLATION AND CORRECTIVE ACTIONS ARE GIVEN IN LER 87-63. ALL ISOLATED SYSTEMS WERE RESTORED AT 1516 HOURS. FURTHER CORRECTIVE ACTIONS TAKEN WERE TO REPLACE THE BLOWN FUSE, TO COUNSEL THE INDIVIDUAL INVOLVED, AND TO PLACE OPERATOR AIDS ON THE LOCAL UPS CONTROL PANELS. THE OPERATING PROCEDURE WILL BE REVISED AND OPERATORS WILL BE TRAINED VIA CONTINUED TRAINING AND THE LESSONS LEARNED PROGRAM.

[162] NINE MILE POINT 2 DOCKET 50-410 LER 87-070
MANUFACTURING DEFICIENCY RESULTS IN AN INOPERABLE AIRLOCK DOOR AND TECHNICAL SPECIFICATION VIOLATION.
EVENT DATE: 102787 REPORT DATE: 112487 NSSS: GE TYPE: BWR

(NSIC 207274) WHILE IN COLD SHUTDOWN ON OCTOBER 26, 1987 AT APPROXIMATELY 1630 HOURS, A MANUFACTURING DEFICIENCY WAS DISCOVERED IN THE OPERATING MECHANISM OF THE PRIMARY CONTAINMENT EMERGENCY ESCAPE AIRLOCK. THIS DEFICIENCY WAS DISCOVERED AFTER INSTRUMENT AND CONTROL (I&C) TECHNICIANS NOTICED AIR LEAKING THROUGH THE OUTER DOOR EQUALIZING VALVE DURING PERFORMANCE OF THE OVERALL AIRLOCK LEAKAGE TEST. THE MANUFACTURING DEFICIENCY CAUSED THE DOORS AND THEIR EQUALIZING VALVES TO OPERATE OUT OF SEQUENCE WITH THE HANDWHEEL POSITION INDICATION AND MAY HAVE

RENDERED THE OUTER DOOR INOPERABLE AT VARIOUS TIMES SINCE INSTALLATION. ON OCTOBER 27, 1987 THE TECHNICAL STAFF DETERMINED THAT THE LIMITING CONDITIONS FOR OPERATION (LCO) OF TECHNICAL SPECIFICATION 3.6.1.3 MAY HAVE BEEN EXCEEDED DURING VARIOUS TIMES BETWEEN INITIAL CRITICALITY AND DISCOVERY OF THE DEFICIENCY. IMMEDIATE CORRECTIVE ACTIONS WERE TO ADJUST THE EQUALIZING VALVES' OPERATING MECHANISM AND TO CORRECT THE DEFICIENCY BY MODIFIYING THE DESIGN OF THE OPERATING MECHANISM TO PREVENT VALVE CAM ROTATION.

[163] NINE MILE POINT 2 DOCKET 50-410 LER 87-072
SECONDARY CONTAINMENT ISOLATION AND STANDBY GAS TREATMENT INITIATION CAUSED BY LACK OF PROCEDURAL GUIDANCE FOR ISOLATING DC ELECTRICAL GROUNDS.
EVENT DATE: 102887 REPORT DATE: 112487 NSSS: GE TYPE: BWR

(NSIC 207275) ON 10/28/87 AT 0430 HOURS, NINE MILE POINT UNIT 2 (NMP2) EXPERIENCED AN ESF ACTUATION. THE ESF ACTUATION CONSISTED OF A SECONDARY CONTAINMENT ISOLATION AND AN AUTOMATIC START OF THE STANDBY GAS TREATMENT SYSTEM. THE ACTUATION WAS INADVERTENTLY CAUSED BY NIAGARA MOHAWK OPERATORS WHILE TRYING TO ISOLATE A GROUND IN THE STATION BATTERY SYSTEM. THE OPERATORS DE-ENERGIZED A CIRCUIT WHICH CAUSED THE REACTOR BUILDING VENTILATION INLET AND OUTLET DAMPERS TO CLOSE WHICH INITIATED THE ESF ACTUATION. AT THE TIME OF THE EVENT, NMP2 WAS IN THE COLD SHUTDOWN CONDITION WITH THE REACTOR MODE SWITCH IN "SHUTDOWN". REACTOR TEMPERATURE AND PRESSURE WERE APPROXIMATELY 125 DEGREES FAHRENHEIT AND 0 POUNDS PER SQUARE INCH GAUGE. THE CAUSE OF THE EVENT WAS A PROCEDURAL DEFICIENCY DUE TO LACK OF GUIDANCE IN OPERATING PROCEDURES (OP) WHICH DO NOT PROVIDE A METHOD TO DETERMINE WHEN PLANT CONDITIONS PERMIT THE ISOLATION OF SPECIFIC LOADS WHEN SEARCHING FOR GROUNDS. CONTRIBUTING TO THE EVENT WAS A PERSONNEL ERROR DUE TO ISOLATING LOADS WITHOUT CORRECTLY DETERMINATING THE EFFECTS OF THE ISOLATIONS. THE FOLLOWING CORRECTIVE ACTIONS SHALL BE IMPLEMENTED: 1. PREPARATION OF A DC LOAD LIST. 2. CONTROL ROOM OPERATORS SHALL BE INSTRUCTED TO CONSULT THE NEW DC LOAD LIST WHEN ISOLATING LOADS IN SEARCH OF DC GROUNDS.

[164] NINE MILE POINT 2 DOCKET 50-410 LER 87-071
PARTIAL PRIMARY CONTAINMENT ISOLATION DUE TO AN INSTRUMENT BEING BUMPED/DESIGN DEFICIENCY AND PERSONNEL ERROR.
EVENT DATE: 111287 REPORT DATE: 121087 NSSS: GE TYPE: BWR

(NSIC 207377) ON NOVEMBER 12, 1987 AT 0122 WITH THE REACTOR MODE SWITCH IN RUN (OPERATIONAL CONDITION 1), AND AT A POWER LEVEL OF APPROXIMATELY 59% RATED THERMAL CAPACITY, NINE MILE POINT UNIT 2 EXPERIENCED A DIVISION 1 ISOLATION OF THE REACTOR CORE ISOLATION COOLING SYSTEM AND AN ISOLATION SIGNAL TO VARIOUS DIVISION 1 RESIDUAL HEAT REMOVAL (RHR) SYSTEM ISOLATION VALVES. NO MOVEMENT OF THE RHR VALVES OCCURRED SINCE THESE VALVES WERE ALREADY CLOSED. THIS ISOLATION, INITIATED BY A REACTOR BUILDING GENERAL AREA HIGH TEMPERATURE SIGNAL, OCCURRED (IT IS SURMISED) WHEN A TEMPERATURE INSTRUMENT WAS INADVERTENTLY BUMPED WHILE PERFORMING A SURVEILLANCE TEST. THE ISOLATION WAS RESET AT 0231 THAT SAME DAY. THE IMMEDIATE CAUSE FOR THIS EVENT IS PERSONNEL ERROR. THE ROOT CAUSE IS A DESIGN DEFICIENCY. THE CORRECTIVE ACTIONS FOR THIS EVENT ARE: 1. A TRAINING MODIFICATION RECOMMENDATION HAS BEEN SUBMITTED REQUESTING TECHNICIAN TRAINING ON THIS EVENT. 2. THE EVENT WILL BE DISCUSSED IN THE INSTRUMENT AND CONTROL (I&C) DEPARTMENT SAFETY MEETINGS. 3 & 6. A SUMMARY OF THE EVENT WILL BE INCLUDED IN THE I&C DEPARTMENT AND OPERATIONS DEPARTMENT LESSONS LEARNED BOOKS. THE INVOLVED CIRCUITS WILL BE MODIFIED TO FACILITATE SURVEILLANCE TESTING. APPLICABLE SURVEILLANCE PROCEDURES WILL BE REVISED AS AN INTERIM MEASURE.

[165] NORTH ANNA 1 DOCKET 50-338 LER 87-020
REACTOR TRIP GENERATED FROM 5A FEEDWATER HEATER HI-HI LEVEL SIGNAL.
EVENT DATE: 112387 REPORT DATE: 121587 NSSS: WE TYPE: PWR
VENDOR: COPES-VULCAN, INC.

CRANE VALVE CO.
INTERNATIONAL INSTRUMENTS, INC.
MERCOID CORP.

(NSIC 207441) AT 0009 HOURS ON NOVEMBER 23, 1987, UNIT 1 TRIPPED FROM 100 PERCENT POWER (MODE 1). THE INITIATING SIGNAL FOR THIS REACTOR TRIP WAS A TURBINE SOLENOID TRIP WHICH RESULTED FROM A 5A FEEDWATER HEATER HI-HI LEVEL SIGNAL. THE 5A FW HEATER HI-HI LEVEL SIGNAL WAS GENERATED WHEN A LEVEL SWITCH FAILED. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(IV). A FOUR HOUR REPORT WAS MADE IN ACCORDANCE WITH 10CFR50.72(B)(2)(II). THE CAUSE OF THE LEVEL SWITCH FAILURE WAS FATIGUE FAILURE OF A SPRING INSIDE THE MICROSWITCH. WHEN THE SPRING FAILED THE SWITCH CLOSED AND A TURBINE SOLENOID TRIP SIGNAL WAS GENERATED. AS CORRECTIVE ACTIONS, THE LEVEL SWITCH WAS REPLACED AND ALL OTHER FEEDWATER HEATER LEVEL SWITCHES WHICH INITIATE REACTOR TRIPS WERE INSPECTED WITH SATISFACTORY RESULTS. THIS EVENT POSED NO SIGNIFICANT SAFETY IMPLICATIONS BECAUSE ALL SAFETY RELATED EQUIPMENT FUNCTIONED AS DESIGNED AND KEY REACTOR PARAMETERS STABILIZED FOLLOWING THE REACTOR TRIP. ALSO, THE REACTOR TRIP SIGNAL WAS GENERATED FROM A SECONDARY PLANT PROTECTION SIGNAL FOR WHICH AN ACTUAL CONDITION DID NOT EXIST. THE HEALTH AND SAFETY OF THE GENERAL PUBLIC WERE NOT AFFECTED.

[166] NORTH ANNA 1 DOCKET 50-338 LER 87-024
VENT STACK "B" HIGH RANGE RADIATION MONITOR EXCEEDED TECHNICAL SPECIFICATION ACTION STATEMENT.
EVENT DATE: 120487 REPORT DATE: 121887 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: NORTH ANNA 2 (PWR)
VENDOR: KAMAN SCIENCES CORP.

(NSIC 207442) AT 1230 HOURS ON DECEMBER 4, 1987, WITH UNIT 1 IN MODE 5 (COLD SHUTDOWN) AND UNIT 2 AT 100% POWER (MODE 1), THE KAMAN VENT STACK "B" HIGH RANGE RADIATION MONITOR WAS DECLARED INOPERABLE BECAUSE THE MICROPROCESSOR FAILED AND COULD NOT BE RESET. ACTION STATEMENT 35 OF TECHNICAL SPECIFICATION 3.3.3.1 REQUIRES THAT THE RADIATION MONITOR BE RETURNED TO OPERABLE STATUS WITHIN 72 HOURS OR INITIATE THE ALTERNATE METHOD OF MONITORING AND PREPARE A SPECIAL REPORT. SINCE THIS ACTION STATEMENT EXPIRED AT 1230 HOURS ON DECEMBER 7, 1987 AND THE RADIATION MONITOR WAS STILL INOPERABLE, THIS EVENT IS REPORTABLE PURSUANT TO TECHNICAL SPECIFICATION 6.9.2. INVESTIGATION INTO THE CAUSE FOR THE INOPERABILITY OF THE RADIATION MONITOR REVEALED THAT THERE WAS A PROBLEM WITH THE NEW PROM'S (PROGRAMMABLE READ ONLY MEMORY) WHICH WERE INSTALLED ON DECEMBER 3, 1987. THESE NEW PROM'S WERE INSTALLED TO CORRECT AN UNRELATED PROBLEM. AS A CORRECTIVE ACTION, THE NEW PROM'S WERE INITIALLY REPLACED WITH THE ORIGINAL PROM'S AND THE CENTRAL PROCESSING UNIT CARD WAS ALSO REPLACED. THE RADIATION MONITOR HAS NOT MALFUNCTIONED SINCE THESE ACTIONS WERE PERFORMED AND WAS RETURNED TO OPERABLE STATUS ON DECEMBER 10, 1987. THIS EVENT POSED NO SIGNIFICANT SAFETY IMPLICATIONS BECAUSE THERE ARE BACKUP RADIATION MONITORS FOR THE RELEASE PATH WHICH REMAINED OPERABLE THROUGHOUT THIS EVENT.

[167] NORTH ANNA 2 DOCKET 50-339 LER 87-014
INADVERTENT DISCHARGE OF ACCUMULATOR INTO REACTOR COOLANT SYSTEM.
EVENT DATE: 102287 REPORT DATE: 112087 NSSS: WE TYPE: PWR

(NSIC 207201) ON OCTOBER 22, 1987, AT 1545 HOURS, WITH UNIT 2 IN MODE 5 (PRIMARY SYSTEM TEMPERATURE AT 90 DEGREES F AND PRIMARY SYSTEM VENTED TO ATMOSPHERE), THE ',C" ACCUMULATOR TANK INADVERTENTLY INJECTED BORATED WATER INTO THE REACTOR COOLANT SYSTEM (RCS) WHEN ITS ISOLATION VALVE (MOV-2865C) WAS CYCLED FOR POST-MAINTENANCE TESTING. RCS WATER LEVEL WAS RAISED FROM 17 INCHES ABOVE CENTERLINE TO APPROXIMATELY 96 INCHES. NO SIGNIFICANT SAFETY CONSEQUENCES RESULTED FROM THE INJECTION OF THE "C'' ACCUMULATOR INTO THE RCS. NO OVERPRESSURE CONDITION COULD HAVE OCCURRED ON THE PRIMARY SYSTEM DUE TO SUFFICIENT FREE VOLUME IN THE RCS TO ACCOMMODATE THE ACCUMULATOR INVENTORY. IN ADDITION, THE MINIMUM SHUTDOWN MARGIN

OF 1770 PCM REQUIRED BY TECHNICAL SPECIFICATION 3.1.1.2 DURING MODE 5 WAS NOT VIOLATED. THE ROOT CAUSE OF THIS EVENT WAS DUE TO THE PERSONNEL ERROR. THE PERSONNEL ERROR WAS FAILURE TO ENSURE THAT THE "C" ACCUMULATOR WAS PROPERLY VENTED PRIOR TO STROKING OF THE "C" ACCUMULATOR ISOLATION VALVE, MOV-2865C. THIS EVENT IS REPORTABLE PURSUANT TO 50.73(A)(2)(IV) DUE TO A MANUAL ACTUATION OF AN ENGINEERING SAFETY FEATURE WHICH INADVERTENTLY INJECTED INTO THE RCS.

[168] NORTH ANNA 2 DOCKET 50-339 LER 87-016
INADVERTENT RPS ACTUATION DURING TESTING.
EVENT DATE: 103187 REPORT DATE: 111887 NSSS: WE TYPE: PWR

(NSIC 207202) AT 0005 HOURS ON OCTOBER 31, 1987, WITH UNIT 2 IN MODE 4 (WITH REACTOR COOLANT SYSTEM PRIMARY TEMPERATURE AT 213 DEGREES F AND PRIMARY PRESSURE AT 341 PSIG), AN UNEXPECTED AUTOMATIC REACTOR PROTECTION SYSTEM REACTOR TRIP SIGNAL WAS GENERATED DURING THE PERFORMANCE OF AN AUXILIARY FEEDWATER PUMP TIME RESPONSE AND LOGIC TEST. SINCE THE GENERATION OF A REACTOR TRIP SIGNAL HAD NOT BEEN EXPECTED, THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(IV). DURING THE PERFORMANCE OF THE TEST, A STEAM GENERATOR (S/G) LOW LOW LEVEL SIGNAL WAS GENERATED BY PLACING TWO LEVEL COMPARATOR TEST SWITCHES FOR THE "A" S/G IN THE "TEST" POSITION. AS A RESULT, AN AUTOMATIC AFW PUMP START SIGNAL WAS GENERATED AND A REACTOR TRIP SIGNAL WAS GENERATED WHICH OPENED THE REACTOR TRIP BREAKERS. THE CAUSE OF THIS EVENT WAS PROCEDURE INADEQUACY. AS A CORRECTIVE ACTION, THE PROCEDURE WAS DEVIATED TO ADD A NOTE WHICH STATED THAT PLACING BOTH LEVEL COMPARATORS IN THE "TEST" POSITION CONCURRENTLY WILL GENERATE A REACTOR TRIP SIGNAL. ALSO, A STATEMENT WAS ADDED TO THE INITIAL CONDITIONS TO VERIFY THE REACTOR TRIP BREAKERS ARE OPEN. TO PREVENT RECURRENCE OF THIS EVENT, BOTH THE UNIT 1 AND UNIT 2 PROCEDURES WILL BE REVISED TO INCLUDE THE CONTENTS OF THE PROCEDURE DEVIATION MENTIONED ABOVE.

[169] NORTH ANNA 2 DOCKET 50-339 LER 87-017
NUCLEAR INSTRUMENT CHANNEL NOT PLACED IN TRIP WITHIN 1 HOUR.
EVENT DATE: 110387 REPORT DATE: 120287 NSSS: WE TYPE: PWR

(NSIC 207266) ON NOVEMBER 3, 1987, AT 0848 HOURS, WITH UNIT 2 IN MODE 2 (STARTUP) AT ZERO PERCENT POWER, IT WAS DISCOVERED THAT THE OUT OF SERVICE POWER RANGE NUCLEAR INSTRUMENT DETECTOR CHANNEL N-44 HAD NOT BEEN PLACED IN TRIP WITHIN 1 HOUR AFTER ENTERING MODE 2. THE POWER RANGE DETECTOR WAS TAKEN OUT OF SERVICE TO PROVIDE AN INPUT SIGNAL TO A REACTIVITY COMPUTER. THIS IS A VIOLATION OF TECHNICAL SPECIFICATION 3.3.1.1 AND IS REPORTABLE PURSUANT TO 10 CFR 50.73(A)(2)(I)(B). TO COMPLY WITH TECHNICAL SPECIFICATION 4.10.3.2, IT IS NECESSARY TO FUNCTIONALLY TEST ALL FOUR POWER RANGE CHANNELS AT LEAST 12 HOURS PRIOR TO THE COMMENCEMENT OF PHYSICS TESTING. AFTER THE FUNCTIONAL TESTING WAS SATISFACTORILY COMPLETED AT 2130 HOURS ON NOVEMBER 2, 1987, THE OUT OF SERVICE N-44 DETECTOR CHANNEL WAS NOT PLACED BACK IN TRIP. MODE 2 WAS ENTERED AT 0204 HOURS ON NOVEMBER 3, 1987, WITH THE N-44 DETECTOR CHANNEL NOT IN TRIP. THE CHANNEL WAS DISCOVERED TO NOT BE IN TRIP AT 0848 HOURS ON NOVEMBER 3, 1987. REACTOR PROTECTION WAS STILL AVAILABLE TO PREVENT ANY OVER POWER CONDITION. THEREFORE, THERE WAS NO IMPACT ON THE HEALTH AND SAFETY OF THE GENERAL PUBLIC. THE CAUSE OF THIS EVENT WAS PROCEDURE INADEQUACY. PROCEDURES WILL BE REVISED TO PREVENT RECURRENCE OF THIS EVENT.

[170] NORTH ANNA 2 DOCKET 50-339 LER 87-015
INOPERABLE REDUNDANT S/G STEAM FLOW CHANNEL EXCEEDS TECHNICAL SPECIFICATION ACTION STATEMENT.
EVENT DATE: 110487 REPORT DATE: 120487 NSSS: WE TYPE: PWR
VENDOR: RAYCHEM CORP.

(NSIC 207299) AT 2153 HOURS ON NOVEMBER 4, 1987, WITH UNIT 2 IN MODE 1 AT 27

PERCENT REACTOR POWER FOLLOWING A REFUELING OUTAGE, THE "A" STEAM GENERATOR (S/G) STEAM FLOW CHANNEL III WAS DETERMINED TO BE INOPERABLE. UPON INVESTIGATION, IT WAS DETERMINED TO HAVE BEEN INOPERABLE SINCE MAINTENANCE WAS PERFORMED ON THE FIELD WIRING DURING THE REFUELING OUTAGE. THE INOPERABLE CHANNEL HAD NOT BEEN PLACED IN THE TRIPPED CONDITION UPON ENTRY INTO MODE 3 (HOT STANDBY) AS REQUIRED BY TECHNICAL SPECIFICATION 3.3.1 AND 3.3.2. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73 (A)(2)(I)(B). THE CAUSE OF THE "A" S/G STEAM FLOW CHANNEL III FAILURE WAS DUE TO AN INSTALLATION ERROR IN THE FIELD WIRING. ALL READINGS RETURNED TO NORMAL WHEN THE LEADS WERE REVERSED. NO SIGNIFICANT SAFETY CONSEQUENCES RESULTED FROM INOPERABILITY OF THE "A" S/G STEAM FLOW CHANNEL III BECAUSE PROTECTION CHANNELS, NECESSARY FOR COMPLETING THE LOGIC TO INITIATE A REACTOR TRIP OR ESF (SAFETY INJECTION) ACTUATION, WERE OPERABLE. THE HEALTH AND SAFETY OF THE GENERAL PUBLIC WERE NOT AFFECTED.

[171] OCONEE 1 DOCKET 50-269 LER 87-008
A DIRECT PATHWAY FROM CONTAINMENT TO THE ENVIRONMENT DURING FUEL MOVEMENT RESULTS IN A VIOLATION OF TECH SPECS DUE TO MANAGEMENT DEFICIENCY AND DESIGN DEFICIENCY.
EVENT DATE: 101287 REPORT DATE: 112587 NSSS: BW TYPE: PWR

(NSIC 207176) ON OCTOBER 12, 1987, WITH UNIT 1 IN REFUELING SHUTDOWN, IT WAS IDENTIFIED THAT TECHNICAL SPECIFICATION 3.8.6 WAS VIOLATED WHEN FUEL WAS MOVED IN THE REACTOR BUILDING CONCURRENT WITH THE EXISTENCE OF A DIRECT PATHWAY FROM INSIDE CONTAINMENT TO THE OUTSIDE ENVIRONMENT. THE PATHWAY EXISTED THROUGH THE EMERGENCY HATCH AND WAS ASSOCIATED WITH A TEMPORARY MODIFICATION. THE ROOT CAUSES OF THIS INCIDENT WERE DETERMINED TO BE A MANAGEMENT DEFICIENCY IN THE PLANNING PROCESS OF THE TASK AND AN UNRELATED DESIGN DEFICIENCY. THE IMMEDIATE CORRECTIVE ACTION WAS TO STOP FUEL MOVEMENT. SUBSEQUENT CORRECTIVE ACTIONS WERE TO REPAIR THE DIRECT PATHWAY. PLANNED CORRECTIVE ACTIONS INCLUDE REVIEW OF THIS INCIDENT BY APPROPRIATE PERSONNEL. HAD A FUEL HANDLING ACCIDENT OCCURRED DURING THIS INCIDENT, THE CONSEQUENCES WOULD BE BOUNDED BY ACCIDENT ANALYSES INCLUDED IN THE OCONEE FINAL SAFETY ANALYSIS REPORT (FSAR).

[172] OCONEE 1 DOCKET 50-269 LER 87-009
TWO FUNCTIONAL UNITS OF EMERGENCY POWER SWITCHING LOGIC TAKEN OUT OF SERVICE DUE TO A MANAGEMENT DEFICIENCY.
EVENT DATE: 102887 REPORT DATE: 121587 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: OCONEE 2 (PWR)
OCONEE 3 (PWR)

(NSIC 207428) ON OCTOBER 28, 1987 AT 0415, STANDBY BUS 1 WAS REMOVED FROM SERVICE FOR MAINTENANCE INSPECTION. IN ORDER TO ISOLATE THE STANDBY BUS, THE KEOWEE FEEDER BREAKER, AND THE STANDBY BUS TO MAIN FEEDER BUS BREAKERS FOR UNITS 1, 2 AND 3 WERE TAGGED OPEN WITH THEIR RESPECTIVE CONTROL POWER FUSES REMOVED. AT FIRST IT APPEARED THAT THE REQUIREMENTS OF TECHNICAL SPECIFICATION 3.7.2(B) WERE NOT MET. HOWEVER, UPON FURTHER REVIEW, IT WAS DETERMINED THAT THE SPECIFICATION WAS NOT VIOLATED IN THAT THE CONTROLLING SPECIFICATION FOR THIS SITUATION WAS SPECIFICATION 3.7.1(B)1 WHICH WOULD ALLOW FOR THE REMOVAL OF ONE STANDBY BUS. DURING THIS INCIDENT, UNIT 1 WAS SHUTDOWN FOR REFUELING AND UNITS 2 AND 3 WERE AT 85% AND 100% FULL POWER RESPECTIVELY. THE ROOT CAUSE OF THIS INCIDENT WAS DETERMINED TO BE MANAGEMENT DEFICIENCY BECAUSE OF THE APPARENT CONFLICTS IN TECHNICAL SPECIFICATION 3.7. THE IMMEDIATE CORRECTIVE ACTION WAS TO RETURN THE STANDBY BUS TO SERVICE. THE SUBSEQUENT CORRECTIVE ACTION WAS TO SEND A SPECIAL REPORT TO THE NRC ON NOVEMBER 2, 1987. PLANNED CORRECTIVE ACTIONS INCLUDE ISSUANCE OF TRAINING PACKAGES AND A TECHNICAL SPECIFICATION INTERPRETATION, AS WELL AS SUBMITTAL OF A TECHNICAL SPECIFICATION CHANGE.

[173] OCONEE 1 DOCKET 50-269 LER 87-010
MANUAL REACTOR TRIP DUE TO A COMPONENT FAILURE DURING STARTUP PHYSICS TESTING.
EVENT DATE: 110587 REPORT DATE: 120787 NSSS: BW TYPE: PWR
VENDOR: DIAMOND POWER SPECIALTY CORP.

(NSIC 207358) ON NOVEMBER 5, 1987 UNIT 1 WAS IN STARTUP PHYSICS TESTING FOLLOWING REFUELING. AT THE TIME OF THE INCIDENT THE REACTOR WAS ABOUT 125 COUNTS PER SECOND ON THE SOURCE RANGE MONITOR. THE PRE-CRITICAL TEST PHASE HAD BEEN COMPLETED AND THE UNIT WAS IN THE ZERO POWER PHYSICS TEST PHASE. DURING THIS TESTING PHASE; CONTROL POWER WAS LOST TO THE CONTROL RODS. AT THIS TIME THE REACTOR WAS MANUALLY TRIPPED. THE RODS WERE VERIFIED TO BE INSERTED AND THE UNIT WAS STABILIZED AT HOT SHUTDOWN. ALL PARAMETERS RESPONDED TO THE TRIP AS EXPECTED AND THERE WERE NO SAFETY SYSTEM ACTUATIONS. A WORK REQUEST WAS WRITTEN TO INVESTIGATE AND REPAIR THE LOSS OF POWER PROBLEM. A LOOSE SOLDER JOINT AT PIN #1 ON A CONTROL ROD SEQUENCING CARD WAS IDENTIFIED. THE ROOT CAUSE OF THIS INCIDENT WAS DETERMINED TO BE A FAILED COMPONENT DUE TO THE LOOSE SOLDER JOINT. THE IMMEDIATE CORRECTIVE ACTION WAS TO MANUALLY TRIP THE UNIT AND STABILIZE THE UNIT IN HOT SHUTDOWN CONDITIONS. SUBSEQUENT CORRECTIVE ACTIONS WERE TO RESUME STARTUP PHYSICS TESTING AND TO IDENTIFY AND REPAIR THE LOOSE SOLDER JOINT.

[174] OYSTER CREEK DOCKET 50-219 LER 87-006 REV 01
UPDATE ON TECHNICAL SPECIFICATION VIOLATION CAUSED BY IMPROPER STORAGE OF HIGHER ENRICHMENT FUEL DUE TO PERSONNEL ERROR.
EVENT DATE: 012187 REPORT DATE: 112487 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 207248) OYSTER CREEK TECHNICAL SPECIFICATION 5.3.1(C) SPECIFIES THAT THE FUEL STORED IN THE FUEL POOL STORAGE RACKS SHALL NOT EXCEED A MAXIMUM AVERAGE PLANAR ENRICHMENT OF 3.01 WT% U-235. CONTRARY TO THE ABOVE, RELOAD FUEL BUNDLES SUPPLIED BY GENERAL ELECTRIC COMPANY (GE) HAVING AN AVERAGE PLANAR ENRICHMENT OF 3.19% U-235 WERE TEMPORARILY STORED IN THE FUEL POOL DURING THE LLR OUTAGE IN 1986. THE CAUSE OF THE EVENT IS ATTRIBUTED TO PERSONNEL ERROR IN NOT PERFORMING A THOROUGH SAFETY ANALYSIS FOR STORAGE OF THE NEW FUEL AND IN NOT RECOGNIZING A CONFLICT WITH THE TECHNICAL SPECIFICATIONS PRIOR TO FUEL STORAGE IN THE SPENT FUEL POOL. CORRECTIVE ACTIONS WILL CONSIST OF REVISING THE TECHNICAL SPECIFICATIONS TO RAISE THE ENRICHMENT LIMITATIONS ON STORED FUEL AND REVIEWING THE OCCURRENCE WITH ENGINEERING PERSONNEL.

[175] OYSTER CREEK DOCKET 50-219 LER 87-036
HIGH RADIATION AREA TECHNICAL SPECIFICATION VIOLATION DUE TO PERSONNEL ERROR AND PROCEDURAL NON-COMPLIANCE.
EVENT DATE: 101387 REPORT DATE: 111387 NSSS: GE TYPE: BWR

(NSIC 207163) ON OCTOBER 13, 1987, WITH THE REACTOR IN A COLD SHUTDOWN CONDITION AND THE DRYWELL OPEN FOR ACCESS, FOUR (4) WORKERS ENTERED A LOCKED HIGH RADIATION AREA WITHOUT THE MINIMUM RADIATION MONITORING EQUIPMENT REQUIRED BY PROCEDURE AND STATION TECHNICAL SPECIFICATIONS. IN ADDITION, THE LOCKED RADIATION DOOR TO THE AREA WAS NOT PROPERLY CONTROLLED AS REQUIRED BY PROCEDURE AND STATION TECHNICAL SPECIFICATIONS. OF THE WORKERS INVOLVED, FOUR (4) CONTRACTOR PERSONNEL, THREE (3) HAD ENTERED THE AREA ON THE PREVIOUS DAY MEETING ALL REQUIREMENTS. THE DAY OF THE VIOLATION EACH MAN ERRONEOUSLY ASSUMED ONE OF THEM OBTAINED THE REQUIRED ALARMING DOSIMETER. THEY WERE AWARE OF THE REQUIREMENT. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO PERSONNEL ERROR. THE FAILURE TO PROPERLY CONTROL THE ENTRY POINT IS ATTRIBUTED TO POOR RADIOLOGICAL CONTROLS PRACTICE WHEN CONTROLLING DRYWELL ACCESS. CONTROL OF THE ENTRY POINT WAS INFORMALLY BEING PASSED TO ANOTHER INDIVIDUAL. THIS CAUSED A BREAKDOWN OF POSITIVE CONTROL AND WAS NOT BEING PERFORMED PER STATION PROCEDURE. THE PERSONNEL WHO ENTERED THE AREA PERFORMED THEIR TASK AND RECEIVED DOSES BELOW THE EXPECTED LEVELS. THREE (3) OF THE CONTRACTORS INVOLVED WERE TERMINATED. THE FOURTH INDIVIDUAL RECEIVED A

WRITTEN REPRIMAND, AS HE HAD NOT BEEN INVOLVED ON THE PREVIOUS DAY AND WAS NOT AS FAMILIAR WITH THE JOB AS THE OTHER INDIVIDUALS.

[176] OYSTER CREEK DOCKET 50-219 LER 87-039
APPENDIX R CRITERIA NOT MET DUE TO EXISTING PLANT DRAWING DEFICIENCY DISCOVERED DURING MODIFICATION DOCUMENT UPDATE.
EVENT DATE: 101387 REPORT DATE: 111287 NSSS: GE TYPE: BWR

(NSIC 206901) ON OCTOBER 14, 1987, IT WAS DISCOVERED THAT A SIGNAL CIRCUIT FROM A LOCKOUT RELAY, WHICH PREVENTS THE EMERGENCY DIESEL GENERATOR #2 (EDG-2) BREAKER FROM CLOSING IN THE EVENT OF A 4160 VOLT ELECTRICAL BUS FAULT, DOES NOT COMPLY WITH 10CFR50, APPENDIX R. THE CIRCUIT IS NOT PROTECTED FROM A FIRE IN A FIRE ZONE FOR WHICH RESPONSE RELIES UPON THE OPERABILITY OF EDG-2. THE ABILITY TO CLOSE THE EDG-2 BREAKER COULD BE LOST IN AN APPENDIX R FIRE EVENT WITHOUT AN ACTUAL ELECTRICAL BUS FAULT. THE PLANT WAS SHUTDOWN AT THE TIME OF DISCOVERY, HOWEVER, THE CONDITION HAS EXISTED SINCE STARTUP FROM THE DECEMBER 1986 REFUELING OUTAGE. THE CAUSE OF THE CONDITION IS A DESIGN DEFICIENCY. THE ROOT CAUSE OF THE DEFICIENCY WAS A DISCREPANCY BETWEEN ACTUAL CIRCUITRY IN THE FIELD AND THAT WHICH WAS REFLECTED IN AN EXISTING PLANT DRAWING USED FOR THE DESIGN OF THE MODIFICATION. IT IS SIGNIFICANT IN THAT IT COULD HAVE PREVENTED A SAFE SHUTDOWN IN RESPONSE TO A FIRE EVENT. CORRECTIVE ACTION HAS BEEN TAKEN TO MODIFY THE CIRCUITRY TO COMPLY WITH 10CFR50 APPENDIX R CRITERIA.

[177] OYSTER CREEK DOCKET 50-219 LER 87-040
TORUS OXYGEN SAMPLE LINE DOESN'T MEET SINGLE FAILURE CRITERIA DUE TO DESIGN DEFICIENCY.
EVENT DATE: 101687 REPORT DATE: 111687 NSSS: GE TYPE: BWR

(NSIC 207164) THE PLANT'S TORUS OXYGEN SAMPLE LINE ISOLATION VALVES' LOGIC DOES NOT MEET SINGLE FAILURE CRITERIA AS REQUIRED BY NRC GENERAL DESIGN CRITERIA. THIS CONDITION WAS DETERMINED REPORTABLE ON OCTOBER 16, 1987, WHILE THE PLANT WAS SHUT DOWN FOR MAINTENANCE. THE APPARENT CAUSE OF THE OCCURRENCE IS A DESIGN DEFICIENCY WHICH HAS BEEN PRESENT SINCE INITIAL PLANT STARTUP. THIS CONDITION IS SIGNIFICANT IN THAT THE TORUS OXYGEN SAMPLE LINE COULD FAIL TO CLOSE DUE TO A SINGLE RELAY FAILURE WHICH COULD PLACE THE PLANT OUTSIDE ITS DESIGN BASIS CONTAINMENT LEAK RATE DURING AN ACCIDENT. A CONSERVATIVE ANALYSIS INDICATES THAT, UNDER THE DESIGN BASIS LOSS OF COOLANT ACCIDENT WITH THIS SINGLE FAILURE PRESENT, OPERATORS WOULD HAVE TO DIAGNOSE AND TAKE CORRECTIVE ACTION WITHIN FOUR HOURS TO PREVENT EXCEEDING 10 CFR 100 LIMITS. CORRECTIVE ACTION WILL BE TAKEN TO MODIFY THE VALVE ISOLATION CIRCUIT TO MEET SINGLE FAILURE CRITERIA IN THE NEXT REFUELING OUTAGE. UNTIL THE MODIFICATION IS INSTALLED, DIRECTION BY MEANS OF A STANDING ORDER AND PROCEDURAL INSTRUCTIONS WILL BE PROVIDED TO OPERATORS TO CLOSE THESE VALVES UPON A CONTAINMENT ISOLATION.

[178] OYSTER CREEK DOCKET 50-219 LER 87-042
POTENTIAL INOPERABILITY OF STANDBY GAS TREATMENT SYSTEM DUE TO DESIGN ERROR.
EVENT DATE: 102687 REPORT DATE: 112487 NSSS: GE TYPE: BWR

(NSIC 207250) ON OCTOBER 26, 1987 IT WAS DETERMINED THAT A CONDITION WHICH COULD PREVENT THE FULFILLMENT OF THE SAFETY FUNCTION OF THE STANDBY GAS TREATMENT SYSTEM (SGTS) EXISTS. THE PLANT'S DRYWELL PURGE SUPPLY VALVES DO NOT CLOSE ON A REACTOR BUILDING ISOLATION SIGNAL. A SECONDARY CONTAINMENT IN-LEAKAGE PATH IS CREATED WHEN THE DRYWELL IS OPEN OR BEING VENTILATED AND REACTOR BUILDING ISOLATION OCCURS WITH THE SGTS INITIATED. UNDER THESE CONDITIONS THE SGTS MAY NOT BE ABLE TO MAINTAIN THE REACTOR BUILDING VACUUM REQUIRED BY TECHNICAL SPECIFICATIONS. AT THE TIME OF DISCOVERY, THE PLANT WAS SHUT DOWN FOR MAINTENANCE; HOWEVER, THIS CONDITION HAS EXISTED SINCE INITIAL PLANT OPERATION. THE APPARENT CAUSE OF THE CONDITION IS A DESIGN DEFICIENCY. THE CONDITION IS

SIGNIFICANT IN THAT A SYSTEM NEEDED TO CONTROL THE RELEASE OF RADIOACTIVE MATERIAL COULD BE PREVENTED FROM FULFILLING ITS SAFETY FUNCTION. IMMEDIATE CORRECTIVE ACTION WAS TAKEN TO SECURE THE VALVES IN THE CLOSED POSITION. A MODIFICATION WAS INSTALLED TO PROVIDE A REACTOR BUILDING ISOLATION SIGNAL TO TWO OF THE FOUR DRYWELL PURGE SUPPLY VALVES.

[179] OYSTER CREEK DOCKET 50-219 LER 87-044
SAFETY SYSTEMS UNABLE TO FUNCTION ON LOSS OF OFFSITE POWER DURING BATTERY MAINTENANCE ACTIVITIES DUE TO PERSONNEL ERROR.
EVENT DATE: 103087 REPORT DATE: 112587 NSSS: GE TYPE: BWR

(NSIC 207251) FOR THE PERIOD OCTOBER 13, 1987 THROUGH OCTOBER 19, 1987 EMERGENCY DIESEL GENERATOR #2 AND THE 'C' MAIN STATION BATTERY, WHICH SUPPLIES CONTROL POWER TO EMERGENCY DIESEL GENERATOR (EDG) #1, DIVISION A 4160 VOLT AND 460 VOLT SWITCHGEAR, WERE BOTH OUT OF SERVICE. REDUNDANT TRAINS OF SAFETY RELATED EQUIPMENT REQUIRED TO BE OPERABLE WOULD NOT HAVE FUNCTIONED AS DESIGNED UPON A LOSS OF OFFSITE POWER DURING THIS PERIOD. THE CONDITION WAS DISCOVERED ON OCTOBER 30, 1987 AND OCCURRED DURING A MAINTENANCE OUTAGE. THE ROOT CAUSE OF THE EVENT IS PERSONNEL ERROR. THE CORE SPRAY AND STANDBY GAS TREATMENT SYSTEMS, WHICH ARE REQUIRED TO BE OPERABLE WHILE SHUT DOWN, WOULD NOT HAVE AUTOMATICALLY INITIATED DURING AN ACCIDENT WITH A LOSS OF OFFSITE POWER. THE SIGNIFICANCE OF THIS IS MINIMIZED BY: THE PRESENCE OF A FIRE WATER SYSTEM TO PROVIDE MAKEUP WATER TO THE REACTOR; THE FACT THAT THE EVOLUTIONS WHICH CAUSED THE EVENT ARE ONLY PERFORMED WHEN IN COLD SHUTDOWN; AND THE EXISTENCE OF DC POWER DIAGNOSTIC AND RESTORATION PROCEDURES. CORRECTIVE ACTIONS INCLUDE PROCEDURE REVISIONS AND REQUIRED READING.

[180] OYSTER CREEK DOCKET 50-219 LER 87-047
REACTOR SCRAM SIGNAL WHILE MOVING REACTOR MODE SWITCH FOR TESTING.
EVENT DATE: 111587 REPORT DATE: 120887 NSSS: GE TYPE: BWR

(NSIC 207409) ON NOVEMBER 15, 1987 AT 0940 HOURS, A REACTOR SCRAM SIGNAL OCCURRED WHILE MOVING THE REACTOR MODE SWITCH FROM SHUTDOWN TO REFUEL TO SUPPORT TESTING OF THE ROD WORTH MINIMIZER SYSTEM. AT THE TIME, THE PLANT WAS SHUTDOWN FOR MAINTENANCE. THE SCRAM WAS RESET AND OPERATORS ENSURED THE PLANT WAS IN A STABLE CONDITION. THE APPARENT CAUSE OF THE OCCURRENCE IS EQUIPMENT MALFUNCTION. THE MODE SWITCH WAS NOT COMPLETELY IN REFUEL, CAUSING TIME DELAY RELAYS TO DROP OUT, WHICH RESULTED IN A SCRAM SIGNAL. THE SAFETY SIGNIFICANCE IS MINIMAL, REPRESENTING ONLY AN UNNECESSARY OPERATION OF THE REACTOR PROTECTION SYSTEM. AN INVESTIGATION REVEALED NO DEFECTIVE RELAYS OR CONTACTS. FUTURE REPLACEMENT OF THE MODE SWITCH WILL BE EVALUATED.

[181] PALISADES DOCKET 50-255 LER 87-036
WASTE GAS DECAY TANK RELEASED CONTRARY TO TECHNICAL SPECIFICATION.
EVENT DATE: 101987 REPORT DATE: 111887 NSSS: CE TYPE: PWR

(NSIC 207172) ON OCTOBER 19, 1987 AT 1918 WASTE GAS DECAY TANK (WGDT) T-101B (WE;TK) WAS RELEASED TO THE ATMOSPHERE AFTER HAVING BEEN ISOLATED AT 0907 THAT SAME DAY. THIS IS CONTRARY TO PLANT TECHNICAL SPECIFICATION (TS) 3.24.6.1 WHICH STATES THAT "THE WASTE GAS DECAY TANK SYSTEM SHALL BE USED TO REDUCE RADIOACTIVE GASES BY HOLDING GASEOUS WASTE COLLECTED BY THE SYSTEM FOR A MINIMUM OF 15 DAYS UP TO 60 DAYS." THE PLANT WAS IN COLD SHUTDOWN CONDITION AT THE TIME OF THE EVENT. WGDT T-101B WAS INADVERTENTLY RELEASED WHEN A DATA TRANSPOSITION ERROR WAS MADE WHEN COMPLETING THE WASTE GAS RELEASE AUTHORIZATION FORM. WGDT T-101C WAS TO BE RELEASED AND ALL DATA ON THE AUTHORIZATION FORM WAS FOR T-101C, WITH THE EXCEPTION OF THE TANK IDENTIFICATION. THIS ERROR WAS NOT IDENTIFIED DURING RELEASE APPROVALS. BOTH T-101B AND T-101C WERE SAMPLED PRIOR TO THE RELEASE. THE ALARM SETPOINT OF PROCESS MONITOR, RIA-1113 (CALCULATED FOR T-101C) WAS

UPSCALED DURING THE MONITOR LINE PURGE AND NOT RETURNED TO ITS ORIGINAL POSITION. THE RELEASE WAS CALCULATED TO BE EQUAL TO 4.87E-4 MREM/HR OR 0.85 PERCENT OF THE VALUE LISTED IN 10CFR20.106.

[182] PALISADES DOCKET 50-255 LER 87-038
VIOLATION OF OPERATIONS DEPARTMENT OVERTIME POLICY.
EVENT DATE: 102287 REPORT DATE: 112387 NSSS: CE TYPE: PWR

(NSIC 207174) ON OCTOBER 22, 1987 AT APPROXIMATELY 2400 AN AUXILIARY OPERATOR (AO) HAD WORKED 28 HOURS IN A 48 HOUR PERIOD. THE AO IDENTIFIED THIS MATTER AND PROMPTLY REPORTED IT TO A PLANT OPERATIONS SUPPORT SUPERVISOR. THIS IS CONTRARY TO THE REQUIREMENTS SPECIFIED IN THE ORDER MODIFYING LICENSE DPR- 20/50-255 ISSUED MARCH 25, 1983. THE 1983 ORDER STATES THAT LICENSED OPERATORS "SHALL NOT BE PERMITTED TO WORK MORE THAN 16 HOURS IN ANY 24 HOUR PERIOD, NOR MORE THAN 24 HOURS IN ANY 48 HOUR PERIOD, NOR MORE THAN 72 HOURS IN ANY SEVEN DAY PERIOD (ALL EXCLUDING SHIFT TURN OVER TIME)." THE PLANT WAS IN COLD SHUTDOWN CONDITION WHEN THE DISCREPANCY WAS IDENTIFIED. THE VIOLATION OF THE OVERTIME RESTRICTIONS HAS BEEN ATTRIBUTED TO PERSONNEL ERROR IN THAT BOTH THE SHIFT SUPERVISOR OFFERING THE EXTENDED WORK HOURS AND THE AO, WHO HAS THE ULTIMATE RESPONSIBILITY FOR ASSURING REQUIREMENTS ARE MET, FAILED TO REALIZE ALL THE OPERATING LICENSE RESTRICTIONS. A MEMO IS BEING ISSUED TO OPERATIONS DEPARTMENT PERSONNEL AS PART OF THE OPERATIONS DEPARTMENT REQUIRED READING PROGRAM. THIS MEMO WILL DETAIL ALL REQUIREMENTS AS SPECIFIED IN THE PALISADES OPERATING LICENSE.

[183] PALISADES DOCKET 50-255 LER 87-039
POTENTIAL FOR OPERATION OUTSIDE OF DESIGN BASIS WITH RESPECT TO MSLB ANALYSIS.
EVENT DATE: 103087 REPORT DATE: 113087 NSSS: CE TYPE: PWR

(NSIC 207256) DURING EFFORTS TO CLOSE OUT AN NRC OPEN ITEM IDENTIFIED THROUGH THE PALISADES SYSTEM FUNCTIONAL EVALUATION (SFE) PROGRAM, IT WAS DETERMINED THAT CHARGING PUMP P-55B (CB;P) WOULD NOT AUTOMATICALLY ACTUATE UPON A PRESSURIZER (AB;PZR) LOW LEVEL SIGNAL WITH COINCIDENT SIS AS PREVIOUSLY THOUGHT. THIS DISCOVERY RESULTED IN THE POTENTIAL FOR PAST PLANT OPERATION OUTSIDE OF ITS DESIGN BASIS AS DESCRIBED IN SECTION 14.14, "STEAM LINE RUPTURE INCIDENT" OF THE PALISADES FINAL SAFETY ANALYSIS REPORT WHILE OPERATING WITHIN CURRENT PLANT TECHNICAL SPECIFICATIONS (TS). THE PLANT WAS IN COLD SHUTDOWN CONDITION WHEN THIS ITEM WAS IDENTIFIED.

[184] PALISADES DOCKET 50-255 LER 87-037
FAILURE TO TAKE COMPENSATORY MEASURES DURING TEMPORARY LOSS OF FIRE SUPPRESSION SYSTEM.
EVENT DATE: 110687 REPORT DATE: 112087 NSSS: CE TYPE: PWR

(NSIC 207173) ON NOVEMBER 6, 1987 AT 1055 AN UNDERGROUND PIPE ASSOCIATED WITH THE FIRE SUPPRESSION WATER SYSTEM (KP;PSP) RUPTURED. THIS RESULTED IN ALL THREE FIRE SYSTEM WATER PUMPS (KP;P) AUTOMATICALLY STARTING DUE TO LOW PRESSURE, WITH ALL FIRE SYSTEM PUMPS BEING SECURED AT 1100. AT 1115 THE PORTION OF THE FIRE WATER SYSTEM WHERE THE BREAK OCCURRED WAS ISOLATED AND AT 1140 THE FIRE WATER SYSTEM WAS REPRESSURIZED AND RETURNED TO NORMAL OPERATING STATUS WITH THE EXCEPTION OF THE ISOLATED SECTION. ALL FIRE PROTECTION EQUIPMENT WAS BELIEVED TO BE OPERABLE AT THIS TIME. AT 1720 AN AUXILIARY OPERATOR REPORTED THAT NO FIRE WATER PRESSURE WAS AVAILABLE TO THE SPRINKLER SYSTEM IN THE AREA OF THE VOLUME REDUCTION SYSTEM (WB). OPERATIONS PERSONNEL FURTHER REVIEWED SYSTEM DRAWINGS TO IDENTIFY ALL AFFECTED AREAS. AT 1745 COMPENSATORY FIRE TOURS WERE INITIATED AND BY 2320 ALL REQUIRED BACKUP FIRE SUPPRESSION EQUIPMENT HAD BEEN STAGED AND THE ISOLATED PORTIONS OF THE SYSTEM PRESSURIZED BY UTILIZING FIRE HOSES AS TEMPORARY CONNECTIONS. THE FAILED SECTION OF PIPE HAS BEEN REPLACED, THE SYSTEM

HYDROSTATICALLY TESTED IN ACCORDANCE WITH NFPA-24 AND THE FIRE WATER SUPPRESSION SYSTEM.

[185] PALISADES DOCKET 50-255 LER 87-040
TECHNICAL SPECIFICATION REQUIREMENTS NOT MET FOR DIESEL OIL STORAGE TANK LEVEL DUE TO TEMPERATURE INDUCED TRANSMITTER DRIFT.
EVENT DATE: 112287 REPORT DATE: 122287 NSSS: CE TYPE: PWR
VENDOR: ROBERTSHAW CONTROLS COMPANY

(NSIC 207420) ON 11/22/87 AT 0258 OPERATIONS PERSONNEL IDENTIFIED THAT THE LEVEL IN DIESEL OIL STORAGE TANK T-10 (DC;TK) WAS BELOW THE PLANT TECHNICAL SPECIFICATION (TS) LIMIT. TS 3.7.1 REQUIRES THAT A MINIMUM OF 16,000 GALLONS OF OIL BE IN T-10 WHEN THE PRIMARY COOLANT SYSTEM (AB) IS HEATED ABOVE 325 DEGREES. AT THE TIME OF DISCOVERY, APPROXIMATELY 12,000 GALLONS OF OIL WERE IN T-10. THE REACTOR WAS CRITICAL WITH THE PLANT OPERATING AT 99% OF RATED POWER WHEN THE CONDITION WAS DISCOVERED. THE EVENT WAS DISCOVERED WHEN A CONTROL ROOM OPERATOR NOTED A DISCONTINUITY FROM PREVIOUS DATA TRENDS IN THE DIESEL OIL STORAGE TANK LEVEL INDICATION. TANK LEVEL WAS PHYSICALLY VERIFIED TO BE AT APPROXIMATELY 12,000 GALLONS. AN UNUSUAL EVENT WAS DECLARED AND REACTOR POWER REACTION INITIATED IN ACCORDANCE WITH TS 3.0.3. BY 0732, APPROXIMATELY 4,900 GALLONS OF OIL HAD BEEN ADDED AND STORAGE TANK LEVEL VERIFIED TO BE GREATER THAN 16,000 GALLONS. THE ERRONEOUS LEVEL INDICATIONS HAVE BEEN ATTRIBUTED TO TEMPERATURE INDUCED TRANSMITTER (DC;LT) DRIFT. AN EVALUATION HAS BEEN UNDERTAKEN TO ASSESS THE POTENTIAL FOR ALTERNATIVE LEVEL INSTRUMENTATION. UNTIL POTENTIAL ACTIONS AS A RESULT OF THIS EVALUATION ARE IMPLEMENTED, DAILY LEVEL VERIFICATION WILL BE PERFORMED MANUALLY.

[186] PALO VERDE 1 DOCKET 50-528 LER 87-026
AUTOMATIC ACTUATION OF CONTAINMENT PURGE ISOLATION SYSTEM DUE TO PERSONNEL ERROR.
EVENT DATE: 102787 REPORT DATE: 112487 NSSS: CE TYPE: PWR

(NSIC 207205) ON OCTOBER 27, 1987 AT APPROXIMATELY 0530 MST, PALO VERDE UNIT 1 WAS IN MODE 6 (REFUELING) WHEN AN ACTUATION OF THE BALANCE OF PLANT ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (BOP ESFAS) OCCURRED WHICH WAS CAUSED BY AN INADVERTENT CONTAINMENT PURGE ISOLATED ACTUATION SIGNAL (CPIAS). CPIAS ALSO CROSS TRIPPED A CONTROL ROOM ESSENTIAL FILTRATION ACTUATION SIGNAL (CREFAS) BY DESIGN. THE ESF ACTUATION OCCURRED WHEN A RADIATION PROTECTION TECHNICIAN DID NOT TAKE SUFFICIENT MEASURES TO ENSURE THAT CPIAS WAS PLACED IN BYPASS PRIOR TO MODIFYING A CONVERSION FACTOR FOR RADIATION MONITORING UNIT RU-37 (POWER ACCESS PURGE AREA). THE ROOT CAUSE OF THIS EVENT WAS A COGNITIVE PERSONNEL ERROR IN THAT THE TECHNICIAN DID NOT ENSURE THAT CPIAS WAS PLACED IN BYPASS AS REQUIRED BY PROCEDURAL CONTROLS. ALSO, PROCEDURAL CONTROLS DID NOT CONTAIN SUFFICIENT INSTRUCTIONS IN A FORMAT CONDUCIVE TO OPERATOR USEABILITY. TO PREVENT RECURRENCE: THE RESPONSIBLE INDIVIDUAL WILL BE RE-INSTRUCTED AS TO THE IMPORTANCE OF ENSURING COMPLIANCE WITH PROCEDURAL REQUIREMENTS, TRAINING WILL BE CONDUCTED TO ENSURE THAT DEPARTMENT PERSONNEL ARE AWARE OF THE IMPORTANCE OF BYPASSING RADIATION MONITORS PRIOR TO CHANGING PARAMETERS, AND THE PROCEDURE WILL BE REVISED.

[187] PALO VERDE 2 DOCKET 50-529 LER 87-015 REV 01
UPDATE ON ESF ACTUATION CAUSED BY SPURIOUS ALARM/TRIP SIGNAL ON RADIATION MONITOR (RU-30).
EVENT DATE: 081687 REPORT DATE: 112487 NSSS: CE TYPE: PWR

(NSIC 207207) THIS IS A SUPPLEMENT TO LER 2-87-015-00 SUBMITTED ON SEPTEMBER 3, 1987. ON AUGUST 16, 1987 AT 0609, PALO VERDE UNIT 2 WAS IN MODE 1 (POWER OPERATION) AT 100 PERCENT POWER WHEN A CONTROL ROOM ESSENTIAL FILTRATION ACTUATION (CREFAS) WAS INITIATED ON CHANNELS A AND B OF THE ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS). THE ACTUATION WAS IN RESPONSE TO A SPURIOUS

ALARM/TRIP SIGNAL ON THE "B" CONTROL ROOM VENTILATION INTAKE NOBLE GAS MONITOR (RU-30), AND OCCURRED COINCIDENT WITH AN ATTEMPT TO RESET A LOCAL ALARM ON THE SUBJECT MONITOR WITH A PORTABLE INDICATION AND CONTROL UNIT (PIC). AN INVESTIGATION WAS CONDUCTED TO DETERMINE THE CAUSE OF THE EVENT, AND TO DETERMINE IF IMPROPER UTILIZATION OF THE PIC WAS A CONTRIBUTING FACTOR. THE RESULTS OF THE INVESTIGATION ARE PROVIDED IN THIS SUPPLEMENT. AS IMMEDIATE CORRECTIVE ACTION, THE ACTUATED EQUIPMENT WAS RETURNED TO A NORMAL OPERATIONS CONFIGURATION, AND THE TRAIN "B" CREFAS WAS PLACED IN BYPASS PENDING FURTHER EVALUATION. OTHER EVENTS INVOLVING CREFAS ACTUATIONS HAVE BEEN REPORTED, HOWEVER, THESE EVENTS DID NOT INVOLVE THE SEQUENCE OF ACTIVITIES NOTED ABOVE AND ARE NOT CONSIDERED SIMILAR EVENTS.

[188] PALO VERDE 2 DOCKET 50-529 LER 87-019
REACTOR TRIP OCCURS DURING STARTUP DUE TO AXIAL SHAPE INDEX OUT-OF-BOUNDS.
EVENT DATE: 112287 REPORT DATE: 122187 NSSS: CE TYPE: PWR
VENDOR: ELECTRO-MECHANICS

(NSIC 207509) AT APPROXIMATELY 1940 ON NOVEMBER 22, 1987 PALO VERDE UNIT 2 WAS IN MODE 1 (POWER OPERATION) AT APPROXIMATELY 7% REACTOR POWER WHEN A REACTOR TRIP OCCURRED AS A RESULT OF THE CORE'S AXIAL SHAPE INDEX (ASI) BEING OUT OF BOUNDS. THE TRIP OCCURRED AS THE UNIT WAS BEING STARTED UP AFTER BEING SHUTDOWN FOR APPROXIMATELY 40 HOURS. THE REACTOR TRIP WAS UNCOMPLICATED AND THE PLANT WAS STABILIZED VERY QUICKLY TERMINATING THE EVENT. THE ROOT CAUSE OF THE TRIP WAS A DEFICIENT PROCEDURE. THE PROCEDURE DID NOT PROVIDE ADEQUATE GUIDANCE TO ENSURE THAT ASI WAS VERIFIED TO BE WITHIN BOUNDS PRIOR TO RAISING POWER ABOVE THE POINT THAT THE CORE PROTECTION CALCULATOR (CPC) (JC) SHIFTED FROM A DEFAULT VALUE TO A CALCULATED VALUE. TO PREVENT RECURRENCE, PROCEDURAL ENHANCEMENTS HAVE BEEN IMPLEMENTED. ADDITIONALLY, A FAULT CONTACT (CNTR) IN THE REACTOR PROTECTION SYSTEM (RPS) (JC) CONTRIBUTED TO THE EVENT IN THAT A REACTOR TRIP OCCURRED ON A ONE-OF-FOUR COINCIDENCE VICE THE DESIGNED TWO-OF-FOUR COINCIDENCE. THE FAULTY CONTACT WAS REPLACED AND A ROOT CAUSE OF FAILURE WAS INITIATED TO PROVIDE FURTHER ANALYSIS. THERE HAVE BEEN NO PREVIOUS SIMILAR EVENTS.

[189] PALO VERDE 3 DOCKET 50-530 LER 87-005
MAIN STEAM ISOLATION SYSTEM ACTUATION DUE TO FAULTY LOGIC BOARD.
EVENT DATE: 103087 REPORT DATE: 112487 NSSS: CE TYPE: PWR
VENDOR: AUTOMATION INDUSTRIES INC.

(NSIC 207208) AT APPROXIMATELY 1618 MST ON OCTOBER 30, 1987, PALO VERDE UNIT 3 WAS IN MODE 3 (HOT STANDBY) WHEN THE MAIN STEAM ISOLATION SYSTEM (MSIS) WAS AUTOMATICALLY ACTUATED. THE MSIS IS PART OF THE ENGINEERED SAFETY FEATURES ACTUATION SYSTEM. DURING TROUBLESHOOTING TO LOCATE A GROUND ON A CLASS LE 125 VDC DISTRIBUTION CABINET, A MAIN STEAM AND FEEDWATER ISOLATION VALVE (MSFIV) LOGIC CABINET WAS DEENERGIZED AT 1617 MST AND REENERGIZED AT 1618 MST. WHEN REENERGIZED, MAIN STEAM ISOLATION VALVE (MSIV) SGE-UV-170 OPENED AND A STEAM GENERATOR 1 MSIS AUTOMATIC ACTUATION OCCURRED DUE TO HIGH LEVEL IN THE STEAM GENERATOR. THE ROOT CAUSE OF THE EVENT WAS A FAULTY LOGIC CARD FOR MSIV-170. A SECOND LOGIC CARD WAS TESTED AND ALSO FOUND TO BE FAULTY. A THIRD LOGIC CARD TESTED SATISFACTORILY AND WAS INSTALLED IN THE LOGIC CABINET AS CORRECTIVE ACTION. THE FIRST AND SECOND CARDS WILL BE SENT TO THE VENDOR FOR ANALYSIS AND REWORK. THERE HAVE BEEN NO PREVIOUS SIMILAR EVENTS REPORTED.

[190] PEACH BOTTOM 2 DOCKET 50-277 LER 87-007 REV 02
UPDATE ON REACTOR SCRAM DURING INTERMEDIATE RANGE MONITOR TESTING.
EVENT DATE: 061987 REPORT DATE: 112387 NSSS: GE TYPE: BWR

(NSIC 207180) ON JUNE 19, 1987, WITH UNIT 2 IN THE REFUELING MODE WITH THE CORE OFF LOADED, A FULL SCRAM SIGNAL WAS GENERATED BY THE REACTOR PROTECTION SYSTEM

(RPS) LOGIC. THE SCRAM OCCURRED DURING THE PERFORMANCE OF THE SURVEILLANCE TEST PROCEDURE FOR VERIFICATION OF INTERMEDIATE RANGE MONITOR (IRM) OPERABILITY. THE TEST INVOLVES THE VERIFICATION OF PROPER IRM INPUT TO THE RPS, AND PROPER OPERATION OF IRM ALARMS AND INDICATORS. THE SCRAM WAS CAUSED BY PROCEDURAL DEFICIENCIES, COMBINED WITH PERSONNEL ERROR. PROCEDURAL REVISIONS ARE BEING MADE AND DISCIPLINARY GUIDELINES HAVE BEEN EXERCISED AS PART OF THE EFFORTS TO PREVENT RECURRENCE. THE UNPLANNED RPS ACTUATION MAKES THIS EVENT REPORTABLE.

[191] PEACH BOTTOM 2 DOCKET 50-277 LER 87-022 REV 01
UPDATE ON LOSS OF HIGH PRESSURE COOLANT INJECTION DUE TO UNKNOWN CAUSE.
EVENT DATE: 092987 REPORT DATE: 123187 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: PEACH BOTTOM 3 (BWR)
VENDOR: BUSSMANN MFG (DIV OF MCGRAW-EDISON)
EXIDE ELECTRONICS CORP
RUNDEL ELECTRIC

(NSIC 207429) ON 9/29/87 AT 0020 HOURS WITH THE UNIT IN COLD SHUTDOWN, THE HIGH PRESSURE COOLANT INJECTION SYSTEM (HPCI) WAS DISABLED DUE TO A BLOWN POWER FUSE IN THE HPCI LOGIC PANEL. AT 0125 HOURS, THE 2D BATTERY CHARGER WAS DECLARED INOPERABLE DUE TO A FAILURE OF THE AC UNDERVOLTAGE RELAY. THE EXACT CAUSE OF THESE OCCURRENCES IS NOT CERTAIN. THE BATTERY CHARGER TROUBLE DOES NOT APPEAR TO BE THE CAUSE OF THE FUSE BLOWING. IT IS MORE LIKELY THAT THE FAULT IN THE HPCI TRIGGERED A RANDOM FAILURE IN THE BATTERY CHARGER. WEEKLY SURVEILLANCE TESTING OF A SYNCHRONIZED DIESEL GENERATOR WAS IN PROGRESS AT THE TIME OF THE EVENT; HOWEVER, THIS APPEARS TO BE COINCIDENTAL. THERE WERE NO ADVERSE CONSEQUENCES AS A RESULT OF THIS EVENT. IMMEDIATE CORRECTIVE ACTIONS WERE TO REPLACE THE FUSE WITHIN 35 MINUTES, PLACE THE STATIC INVERTER ON AC FEED AND MONITOR BATTERY TERMINAL VOLTAGE EVERY FOUR HOURS UNTIL A REPLACEMENT UTILITY BATTERY CHARGER WAS PLACED IN SERVICE. THE UNDERVOLTAGE RELAY SOCKET ON THE BATTERY CHARGER WAS FOUND TO BE UNDERRATED. WHILE THIS IS NOT BELIEVED TO HAVE CONTRIBUTED TO THE CAUSE OF THE EVENT, REPLACEMENT SOCKETS FOR BOTH THE UNIT 2 AND UNIT 3 BATTERY CHARGERS HAVE BEEN PROCURED. NO ADDITIONAL CORRECTIVE ACTIONS ARE PLANNED. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR 50.73(A)(2)(V).

[192] PERRY 1 DOCKET 50-440 LER 87-070
LOSS OF REACTOR PROTECTION SYSTEM BUS DUE TO AN OVER VOLTAGE TRIP OF THE ELECTRICAL PROTECTION ASSEMBLY RESULTS IN A DIVISION I BALANCE OF PLANT ISOLATION.
EVENT DATE: 102487 REPORT DATE: 112387 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 207283) ON OCTOBER 24, 1987 AT 0101, A REACTOR PROTECTION SYSTEM (UPS) ELECTRICAL PROTECTION ASSEMBLY (EPA) BREAKER TRIPPED DUE TO AN OVERVOLTAGE CONDITION, RESULTING IN DEENERGIZATION OF RPS BUS A AND A NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NSSSS) DIVISION BALANCE OF PLANT (BOP) ISOLATION. AFFECTED SYSTEMS AND COMPONENTS WERE RESTORED BY 0215. THE ROOT CAUSE OF THIS EVENT HAS NOT BEEN DETERMINED. INVESTIGATION HAS DETERMINED THAT THE EPA BREAKERS DID TRIP ON A HIGH RPS MOTOR-GENERATOR (MG) SET OUTPUT VOLTAGE. HOWEVER, EXTENSIVE TROUBLESHOOTING IDENTIFIED NO EVIDENT CAUSE FOR THE MG SET OUTPUT VOLTAGE DRIFTING HIGH. THE POTENTIAL CAUSE OF THIS EVENT HAS BEEN ISOLATED TO THE VOLTAGE REGULATOR AND/OR RHEOSTAT. THEREFORE, BOTH WERE REPLACED AND WILL BE SENT TO THE VENDOR FOR ADDITIONAL EVALUATION. THE EXISTING REPETITIVE TASK WHICH IS PERIODICALLY PERFORMED ON THE RPS MG SETS AND THE ASSOCIATED CONTROL CIRCUITRY WILL BE REVISED TO ENSURE THAT THE OUTPUT VOLTAGE RHEOSTAT IS PROPERLY ADJUSTED AND SECURED. IT WILL ALSO REQUIRE CLEANING OF THE RHEOSTAT WIPER AND WINDINGS AND TESTING FOR DEAD SPOTS THROUGHOUT THE FULL RANGE OF RESISTANCE. A SUPPLEMENTAL REPORT WILL BE SUBMITTED UPON COMPLETION OF THE VENDOR EVALUATION.

[193] PERRY 1 DOCKET 50-440 LER 87-071
FAULTY TEMPERATURE SWITCH MODULE RESULTS IN A REACTOR WATER CLEANUP ISOLATION DURING THE PERFORMANCE OF A ROUTINE CHANNEL CHECK.
EVENT DATE: 102787 REPORT DATE: 112387 NSSS: GE TYPE: BWR
VENDOR: RILEY-BEAIRD, INC.

(NSIC 207284) ON OCTOBER 27, 1987 AT 0900 A REACTOR WATER CLEANUP (RWCU) SYSTEM CONTAINMENT ISOLATION OCCURRED DURING THE PERFORMANCE OF CHANNEL CHECKS OF LEAK DETECTION SYSTEM TEMPERATURE INSTRUMENTS. WHEN A CONTROL ROOM OPERATOR PLACED THE READ/SET SWITCH FOR RWCU PUMP ROOM 2 DIFFERENTIAL TEMPERATURE TO THE "READ" POSITION, THE RWCU CONTAINMENT ISOLATION OCCURRED. PLANT OPERATORS VERIFIED NO ACTUAL HIGH TEMPERATURES EXISTED AND THE RWCU SYSTEM WAS RETURNED TO OPERATION AT 0905. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO A FAULT IN THE TEMPERATURE SWITCH MODULE (RILEY MODEL #86) PRODUCING A SIGNAL SPIKE. TROUBLESHOOTING OF THE SWITCH MODULE DID NOT IDENTIFY ANY ABNORMALITIES WITH THE CIRCUITRY. THIS ISOLATION HAS BEEN THE ONLY OCCURRENCE SINCE THE SWITCH UNITS WERE REPLACED WITH UPGRADED MODULES (COMPLETED DECEMBER 11, 1986). THEREFORE, NO ADDITIONAL MODIFICATIONS OR ACTIONS ARE PLANNED AT THIS TIME. HOWEVER, THE ROUTINE SYSTEM PERFORMANCE MONITORING WILL CONTINUE AS REQUIRED. ADDITIONALLY, ENGINEERING EVALUATIONS FOR POSSIBLE FUTURE SYSTEM MODIFICATIONS ARE BEING PURSUED.

[194] PERRY 1 DOCKET 50-440 LER 87-072
REACTOR SCRAM AND HPCS INJECTION CAUSED BY LOSS OF FEEDWATER FLOW DUE TO IMPROPER TRANSFER OF STATION LOADS.
EVENT DATE: 102787 REPORT DATE: 112587 NSSS: GE TYPE: BWR

(NSIC 2073[illegible]) ON OCTOBER 27, 1987 AT 0637, FOLLOWING AN IMPROPER TRANSFER OF STATION LOADS, A REACTOR SCRAM OCCURRED DUE TO A REACTOR WATER LEVEL OF LESS THAN LEVEL 3 (+177.7 INCHES ABOVE TOP OF ACTIVE FUEL (TAF)). IN ADDITION, NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM ISOLATIONS, AS WELL AS HIGH PRESSURE CORE SPRAY AND REACTOR CORE ISOLATION COOLING SYSTEM INITIATIONS OCCURRED DUE TO A REACTOR VESSEL LEVEL OF LESS THAN LEVEL 2 (+129.8 INCHES ABOVE TAF). THE CAUSE OF THE EVENT WAS PERSONNEL ERROR. WHEN TRANSFERRING STATION LOADS FROM THE STARTUP TRANSFORMER TO THE AUXILIARY TRANSFORMER, A CONTROL ROOM OPERATOR INADVERTENTLY DE-ENERGIZED THE HOT SURGE TANK LOW LEVEL TRIP LOGIC, RESULTING IN A LOSS OF FEEDWATER TO THE REACTOR VESSEL. CORRECTIVE ACTIONS TO PREVENT RECURRENCE INCLUDE DISCIPLINARY ACTION FOR THE OPERATOR INVOLVED IN THIS EVENT AND IMPLEMENTATION OF AN ENGINEERING DESIGN CHANGE TO INCREASE THE RELIABILITY OF HOT SURGE TANK LOW LEVEL TRIP LOGIC. SUBMITTAL OF THIS REPORT ALSO MEETS THE REQUIREMENTS FOR TECH SPEC 3.5.1, ACTION G, SPECIAL REPORT FOR EMERGENCY CORE COOLING SYSTEM ACTUATION AND INJECTION INTO THE REACTOR COOLANT SYSTEM.

[195] PERRY 1 DOCKET 50-440 LER 87-073
SOLENOID AIR PILOT VALVES STICK DUE TO EXCESSIVE HEAT EXPOSURE RESULTING IN MAIN STEAM ISOLATION VALVES SLOW CLOSURE AND SUBSEQUENT MANUAL REACTOR SCRAM DURING SHUTDOWN.
EVENT DATE: 102987 REPORT DATE: 112587 NSSS: GE TYPE: BWR
VENDOR: AUTOMATIC SWITCH COMPANY (ASCO)

(NSIC 207333) ON OCTOBER 29, 1987 AT 1837, 2155 AND 2216 THREE MAIN STEAM ISOLATION VALVES (MSIV) FAILED TO FAST CLOSE IN THE TIME REQUIRED BY TECH SPECS 3.4.7 AND 3.6.4. SUBSEQUENTLY, THE THREE MSIVS WERE STROKED WITH SATISFACTORY CLOSURE TIMES. ON NOVEMBER 3 AT 1157 AND 1208 TWO OF THE SAME MSIVS FAILED TO CLOSE WITHIN THE REQUIRED TIME. BASED ON REPEAT FAILURES A PLANT SHUT DOWN WAS COMMENCED AT 1330. THE REACTOR WAS MANUALLY SCRAMMED AT 1819. THE CAUSE OF THE MSIVS DELAYED CLOSURES HAS BEEN ATTRIBUTED TO THE ASSOCIATED FAST CLOSURE DUAL SOLENOID AIR PILOT VALVES STICKING IN THE NORMAL ENERGIZED POSITION WHEN DEENERGIZED. THE DUAL SOLENOID VALVES WERE STICKING DUE TO DEGRADATION OF THE ELASTOMER DISC AND CORE ASSEMBLY SEALS CAUSED BY EXPOSURE TO EXCESSIVE HEAT FROM

PREVIOUSLY EXISTING STEAM LEAKS. THE ELASTOMER DISCS AND SEALS FOR ALL MSIV DUAL SOLENOID PILOT VALVES HAVE BEEN REPLACED. OTHER EQUIPMENT HAS BEEN EVALUATED FOR ANY ADVERSE AFFECTS FROM KNOWN STEAM LEAKS. ADDITIONAL TEMPERATURE MONITORING EQUIPMENT HAS BEEN INSTALLED AND SPECIAL INSTRUCTIONS ESTABLISHED SPECIFYING ACTIONS TO BE TAKEN UPON EXCEEDING BASELINE TEMPERATURE VALUES. LABORATORY ANALYSES WILL BE PERFORMED TO CONFIRM THE FAILURE MECHANISM OF THE ELASTOMER.

[196] PERRY 1 DOCKET 50-440 LER 87-074
FLOW INDICATION INACCURACY RESULTS IN INDICATED HIGH DIFFERENTIAL FLOW AND REACTOR WATER CLEANUP ISOLATION.
EVENT DATE: 111487 REPORT DATE: 121487 NSSS: GE TYPE: BWR

(NSIC 207383) ON NOVEMBER 14, 1987 AT 0054, AN UNEXPECTED REACTOR WATER CLEANUP (RWCU) INBOARD CONTAINMENT ISOLATION OCCURRED DUE TO INDICATED HIGH DIFFERENTIAL FLOW. PLANT OPERATORS VERIFIED THAT NO ACTUAL SYSTEM LEAKAGE EXISTED AND RETURNED RWCU TO SERVICE AT 0110. THE CAUSE OF THIS EVENT IS FLOW INDICATION INACCURACIES DURING LOW RWCU FLOW CONDITIONS DUE TO DETECTOR LOW FLOW INACCURACIES AND A LOW VOLTAGE CLIP AT THE SQUARE ROOT CONVERTERS. DURING THIS EVENT, FLOW THROUGH THE FEEDWATER RETURN LINE WAS BELOW 15 PERCENT AND IT IS BELIEVED THAT THE COMBINED FLOW INDICATION ERRORS CAUSED THE ISOLATION. AS A RESULT OF THIS EVENT, AN INCREASE OF THE DIFFERENTIAL FLOW TRIP SETPOINT AND/OR TIME DELAY IS BEING EVALUATED TO ALLOW ADDITIONAL OPERATING MARGIN UNDER THESE CONDITIONS. THIS EVALUATION WILL INCLUDE DESIGN RWCU LINE BREAK ANALYSIS. ADDITIONALLY, THE FEASIBILITY OF ELIMINATING RETURN TO FEEDWATER FLOW WHEN THE FLOW RATE APPROACHES THE SQUARE ROOT CONVERTER FLOW CLIP IS BEING EVALUATED. THESE CHANGES WILL REDUCE THE PROBABILITY OF HAVING AN INDICATED HIGH DIFFERENTIAL FLOW ISOLATION.

[197] PERRY 1 DOCKET 50-440 LER 87-075
PERSONNEL ERROR RESULTS IN A VIOLATION OF TECH SPEC DUE TO THE PLANT EXCEEDING 150 PSIG WITH THE REACTOR CORE ISOLATION COOLING SYSTEM INOPERABLE.
EVENT DATE: 111487 REPORT DATE: 121487 NSSS: GE TYPE: BWR

(NSIC 207510) ON NOVEMBER 14, 1987 AT APPROXIMATELY 0245, THE PLANT OPERATED WITH THE REACTOR PRESSURE GREATER THAN 150 PSIG FOR APPROXIMATELY 45 MINUTES WITH THE REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM INOPERABLE. THIS WAS IN VIOLATION OF THE REQUIREMENTS OF TECH SPEC 3.7.3 FOR OPERABILITY OF THE RCIC SYSTEM. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR. THE SHIFT SUPERVISOR (SS) INCORRECTLY DETERMINED THAT RCIC WAS NOT REQUIRED TO BE OPERABLE FOR UP TO 12 HOURS AFTER EXCEEDING 150 PSIG REACTOR PRESSURE. THIS WAS DUE TO A MISINTERPRETATION OF THE SINGLE ASTERISK FOOTNOTE IN TECH SPEC 3.7.3 AND A FAILURE TO UTILIZE AVAILABLE RESOURCES, INCLUDING THE TECH SPECS, TO CORRECTLY ANSWER THE RCIC OPERABILITY QUESTION. PERSONNEL STATEMENTS AND INTERVIEWS WITH OTHER LICENSED OPERATORS ON SHIFT INDICATE THAT THIS MISINTERPRETATION IS NOT A WIDESPREAD CONCERN. THE SS INVOLVED IN THIS EVENT HAS RECEIVED DISCIPLINARY ACTION. IN ADDITION, ALL LICENSED OPERATORS WILL BE FORMALLY TRAINED ON THE SEQUENCE OF EVENTS WHICH LED TO THIS REPORT, WITH EMPHASIS ON THE RESPONSIBILITY TO UTILIZE ALL AVAILABLE RESOURCES TO ENSURE TECH SPEC COMPLIANCE.

[198] PERRY 1 DOCKET 50-440 LER 87-076
DEFICIENT SURVEILLANCE INSTRUCTIONS RESULT IN MAIN STEAM DRAIN LINE ISOLATIONS.
EVENT DATE: 120387 REPORT DATE: 122987 NSSS: GE TYPE: BWR

(NSIC 207487) ON DECEMBER 3, 1987 AT APPROXIMATELY 2145 AND 2305, UNEXPECTED MAIN STEAM DRAIN LINE ISOLATIONS OCCURRED DUE TO NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NSSSS) ISOLATION SIGNALS DURING THE PERFORMANCE OF TURBINE STOP VALVE SURVEILLANCE INSTRUCTIONS. THE CAUSE OF THIS EVENT WAS INADEQUATE SURVEILLANCE INSTRUCTIONS THAT DID NOT IDENTIFY THE TRIPPING OF THE NSSSS ISOLATION LOGIC

CHANNELS NOR INCLUDE STEPS TO RESET THE ISOLATION CHANNELS DURING THE SURVEILLANCE TESTS. THESE SVIS SIMULATE THE OPENING OF THE TURBINE STOP VALVES BY JUMPERING ACROSS APPROPRIATE RELAY CONTACTS, WHICH RESULTED IN A NSSSS ISOLATION SIGNAL BEING GENERATED DUE TO LOW MAIN CONDENSER VACUUM. AS A RESULT OF THIS EVENT, THE SURVEILLANCE INSTRUCTIONS HAVE BEEN REVISED TO INCLUDE APPROPRIATE STEPS TO RESET THE NSSSS ISOLATION CHANNELS DURING THEIR PERFORMANCE. ADDITIONALLY, ALL OTHER SURVEILLANCE INSTRUCTIONS INVOLVING THE TURBINE STOP VALVES HAVE BEEN REVIEWED TO ENSURE SIMILAR INADEQUACIES DID NOT EXIST AND NO FURTHER PROBLEMS WERE IDENTIFIED.

[199] PILGRIM 1 DOCKET 50-293 LER 87-010
FULL REACTOR SCRAM SIGNAL DUE TO SPURIOUS TRIP OF AVERAGE POWER RANGE MONITOR "E".
EVENT DATE: 070287 REPORT DATE: 112087 NSSS: GE TYPE: BWR

(NSIC 207195) ON JULY 2, 1987, AT 0530 HOURS, AN AUTOMATIC ACTUATION OF THE REACTOR PROTECTION SYSTEM (RPS) OCCURRED DUE TO ERRATIC OPERATION AND SUBSEQUENT SPURIOUS TRIP OF AVERAGE POWER RANGE MONITOR (APRM) "E". THE PLANT WAS IN A COLD CONDITION WITH THE REACTOR MODE SWITCH IN REFUEL AND ALL FUEL ASSEMBLIES REMOVED FROM THE REACTOR VESSEL. NO CONTROL RODS WERE WITHDRAWN AND THE REACTOR VESSEL HEAD WAS REMOVED. THE SPURIOUS TRIP OF APRM "E" INSERTED A HALF-SCRAM SIGNAL INTO CHANNEL A" OF THE RPS. THE CHANNEL "A" HALF-SCRAM SIGNAL COINCIDENT WITH AN EXISTING CHANNEL "B HALF-SCRAM RESULTED IN A FULL RPS REACTOR SCRAM TRIP SIGNAL. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO A SPURIOUS TRIP OF APRM "E". ALTHOUGH THE CAUSE OF THE SPURIOUS TRIP COULD NOT BE ESTABLISHED WITH CERTAINTY, IT IS BELIEVED TO HAVE RESULTED FROM DISTURBANCE OF THE APRM CABLING AND CONNECTORS DURING EXTENSIVE UNDERVESSEL WORK THAT WAS IN PROGRESS AT THE TIME OF THIS EVENT. FOLLOWING THIS EVENT, IMMEDIATE ACTION WAS TAKEN TO PLACE APRM "E" IN BYPASS AND RESET THE RPS SCRAM SIGNAL. TESTING OF APRM "E" DID NOT IDENTIFY COMPONENT MALFUNCTION OR FAILURE. THE UNDERVESSEL WORK HAS NOW BEEN COMPLETED AND NO FURTHER OCCURRENCES OF SPURIOUS TRIPS OF APRM "E" HAVE BEEN OBSERVED.

[200] PILGRIM 1 DOCKET 50-293 LER 87-011
FULL REACTOR SCRAM TRIP SIGNAL DUE TO FAILED LOGIC CARD.
EVENT DATE: 070787 REPORT DATE: 112487 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 207260) ON JULY 7, 1987, AT 0031 HOURS AND 0105 HOURS, AN AUTOMATIC ACTUATION OF THE REACTOR PROTECTION SYSTEM (RPS) OCCURRED WHEN THE NO. 1 AND NO. 2 ELECTRICAL PROTECTION ASSEMBLIES (EPAS) TRIPPED AND INTERRUPTED ELECTRICAL POWER FROM THE "A" RPS MOTOR-GENERATOR (MG) SET, TO CHANNEL "A" OF THE RPS. PRIOR TO THESE EVENTS, THE POWER SUPPLY BREAKER FROM THE "B" RPS MG SET HAD BEEN OPENED TO PROVIDE ELECTRICAL ISOLATION FOR MAINTENANCE ACTIVITIES TO BE PERFORMED UNDER MAINTENANCE REQUEST 87-45- 216. THIS BREAKER CONFIGURATION RESULTS IN A HALF-SCRAM SIGNAL ON CHANNEL "B" OF THE RPS. THE TRIP OF EPAS NO. 1 AND NO. 2 RESULTED IN A HALF-SCRAM SIGNAL ON CHANNEL "A" OF THE RPS. THE CHANNEL "A" HALF-SCRAM SIGNAL COINCIDENT WITH THE CHANNEL "B" HALF-SCRAM SIGNAL RESULTED IN A FULL REACTOR SCRAM TRIP SIGNAL AND SECONDARY CONTAINMENT ISOLATION SIGNAL. NO CONTROL ROD MOVEMENT OCCURRED AS A RESULT OF THESE EVENTS. THE CAUSE OF THIS EVENT WAS A FAILED LOGIC CARD IN THE CONTROL CIRCUITRY FOR EPA NO. 1. THE FAILED LOGIC CARD WAS REPLACED AND EPA NO. 1 WAS SATISFACTORILY TESTED AND RETURNED TO SERVICE. AT THE TIME OF THESE EVENTS, THE PLANT WAS IN A COLD CONDITION WITH THE REACTOR MODE SWITCH IN THE REFUEL POSITION AND ALL FUEL ASSEMBLIES REMOVED FROM THE REACTOR VESSEL.

[201] PILGRIM 1 DOCKET 50-293 LER 87-012
FULL REACTOR SCRAM TRIP SIGNAL DUE TO SPIKING OF INTERMEDIATE RANGE MONITORS.
EVENT DATE: 092887 REPORT DATE: 120787 NSSS: GE TYPE: BWR

(NSIC 207295) ON SEPTEMBER 28, 1987, THREE UNEXPECTED AUTOMATIC ACTUATIONS OF THE REACTOR PROTECTION SYSTEM (FULL SCRAM TRIP SIGNALS) OCCURRED IN CLOSE SUCCESSION (1425, 1427 AND 1428 HOURS). THE ACTUATIONS RESULTED FROM SPURIOUS SPIKING OF THE INTERMEDIATE RANGE MONITORS (IRMS) AND SUBSEQUENT HI-HI LEVEL TRIPS OF CHANNELS "A" AND "B". NO CONTROL ROD MOVEMENT RESULTED FROM THESE EVENTS. THE SPURIOUS SPIKING OF THE IRMS WAS CAUSED BY HIGH FREQUENCY WELDING BEING PERFORMED IN THE DRYWELL IN THE VICINITY OF THE MONITORS. IMMEDIATE ACTION WAS TAKEN TO SUSPEND THE WELDING ACTIVITIES AND NO FURTHER SPIKING OF THE IRMS WAS OBSERVED. ADDITIONAL GUIDANCE HAS BEEN PROVIDED REGARDING WELDING ACTIVITIES TO BE PERFORMED IN THE REACTOR BUILDING. THIS GUIDANCE WILL AID IN MINIMIZING THE EFFECTS OF WELDING ACTIVITIES IN THE VICINITY OF SENSITIVE ELECTRONIC EQUIPMENT AND COMPONENTS. AT THE TIME OF THESE EVENTS, THE PLANT WAS IN A COLD CONDITION WITH THE REACTOR MODE SWITCH IN THE REFUEL POSITION AND ALL FUEL ASSEMBLIES REMOVED FROM THE REACTOR VESSEL. NO CONTROL RODS WERE WITHDRAWN AND THE REACTOR VESSEL HEAD WAS REMOVED. CONTROL ROOM INDICATIONS FOR THE REACTOR PROTECTION SYSTEM (RPS) FUNCTIONED AS EXPECTED DURING THESE EVENTS. ALTHOUGH THESE EVENTS ACTUATED THE RPS, THEY DID NOT RESULT IN A CONDITION ADVERSE TO THE SAFE OPERATION OF THE PILGRIM NUCLEAR POWER STATION.

[202] PILGRIM 1 DOCKET 50-293 LER 87-008
UNPLANNED ISOLATION OF SHUTDOWN COOLING DUE TO PERSONNEL ERROR.
EVENT DATE: 101587 REPORT DATE: 111587 NSSS: GE TYPE: BWR

(NSIC 207193) ON OCTOBER 15, 1987, AT 2100 HOURS AN UNPLANNED ISOLATION OF THE RESIDUAL HEAT REMOVAL SYSTEM SHUTDOWN COOLING MODE (SDC) OCCURRED. THE ISOLATION WAS CAUSED WHEN AN ENERGIZED POWER SUPPLY LEAD TO RELAY 16A-K2B BECAME DISCONNECTED DURING A WORK TASK. DE-ENERGIZING THE RELAY CAUSED THE AUTOMATIC CLOSURE OF TWO ISOLATION VALVES AND THE UNPLANNED ISOLATION OF THE SDC SYSTEM. IMMEDIATE ACTION WAS TAKEN TO RECONNECT THE POWER SUPPLY LEAD TO RELAY 16A-K28 AND THE SDC SYSTEM WAS RESTORED APPROXIMATELY FIVE MINUTES AFTER THE ISOLATION. THERE WERE NO COMPONENT OR SYSTEM FAILURES OR MALFUNCTIONS THAT CAUSED THIS EVENT OR RESULTED FROM THIS EVENT. THE CAUSE OF THIS EVENT IS PERSONNEL ERROR. THE ELECTRICAL ISOLATION TAGOUT FOR MODIFICATION WORK BEING PERFORMED UNDER MAINTENANCE REQUEST (MR) 87-33-540, INADVERTENTLY FAILED TO IDENTIFY AND ISOLATE A VOLTAGE SOURCE THAT SUPPLIED POWER TO THE CIRCUIT CONTAINING RELAY 16A-K2B. ALSO CONTRIBUTING TO THIS EVENT WAS THE FAILURE OF CONTRACTOR ELECTRICAL CRAFT PERSONNEL TO VERIFY THAT THE CIRCUIT CONTAINING RELAY 16A-K28 HAD BEEN COMPLETELY DE-ENERGIZED PRIOR TO COMMENCEMENT OF WORK. THIS EVENT OCCURRED DURING AN EXTENDED OUTAGE WITH THE PLANT IN A COLD SHUTDOWN CONDITION, THE REACTOR COMPLETELY REFUELED AND THE MODE SWITCH IN THE REFUEL POSITION. ALL CONTROL RODS WERE FULLY INSERTED PRIOR TO THIS EVENT.

[203] PILGRIM 1 DOCKET 50-293 LER 87-009
CLASS I CONDUIT ROUTED THROUGH CLASS II AREA OF CIRCULATING WATER INTAKE STRUCTURE DUE TO ORIGINAL DESIGN AND CONSTRUCTION DEFICIENCY.
EVENT DATE: 102387 REPORT DATE: 112187 NSSS: GE TYPE: BWR

(NSIC 207194) ON 10/23/87, AT 1123 HOURS, FAILURE AND MALFUNCTION REPORT NO. 87-600 WAS WRITTEN TO DOCUMENT AN ORIGINAL DESIGN/CONSTRUCTION DEFICIENCY IDENTIFIED DURING A REVIEW OF CONDUIT LAYOUT DRAWING E-343. THE REVIEW IDENTIFIED THAT CLASS I CONDUITS WERE ROUTED THROUGH THE CLASS II PORTION OF THE CIRCULATING WATER INTAKE STRUCTURE. THESE CONDUITS CONTAIN ELECTRICAL CABLES THAT PROVIDE POWER TO SAFETY RELATED CIRCUITS OF THE SALT SERVICE WATER (SSW) SYSTEM. THIS ORIGINAL DESIGN/CONSTRUCTION CONFIGURATION DEVIATES FROM THE REQUIREMENTS OF SECTION 12.2.1.1 OF THE FINAL SAFETY ANALYSIS REPORT WHICH STATES IN PART, "....CLASS II DESIGNATED STRUCTURES AND/OR EQUIPMENT SHALL NOT DEGRADE THE INTEGRITY OF ANY STRUCTURES AND/OR EQUIPMENT DESIGNATED CLASS 1." AT THE TIME THIS CONDITION WAS INITIALLY IDENTIFIED, THE PLANT WAS IN AN EXTENDED OUTAGE, WITH THE REACTOR VESSEL IN A COLD SHUTDOWN CONDITION AND DEFUELED, WITH THE MODE

SWITCH IN THE REFUEL POSITION. THE CAUSE OF THIS EVENT IS A DEFICIENCY IN ORIGINAL PLANT DESIGN AND CONSTRUCTION. A PLANT DESIGN CHANGE PDC 87-64 HAS BEEN ISSUED TO REROUTE THE AFFECTED CONDUITS PRIOR TO RESTART OF THE PLANT. A SAFETY EVALUATION WAS PERFORMED TO ASSESS THE INTERIM PLANT CONDITIONS AND THE SAFETY IMPLICATIONS OF PERFORMING REFUELING OPERATIONS.

[204] POINT BEACH 1 DOCKET 50-266 LER 87-004 REV 01
UPDATE ON LOSS OF THE RED INSTRUMENT BUS.
EVENT DATE: 051587 REPORT DATE: 111687 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: POINT BEACH 2 (PWR)
VENDOR: C & D BATTERIES, DIV OF ELTRA CORP.
WESTINGHOUSE ELECTRIC CORP.

(NSIC 207319) ON MAY 15, 1987, WITH UNIT 1 AND UNIT 2 SHUT DOWN, THE STATION BATTERY (D05) BREAKER WAS OPENED TO ISOLATE THE STATION BATTERY FOR CELL REPLACEMENT. WHEN THE BATTERY (D05) BREAKER WAS OPENED, THE BATTERY CHARGER (D07) CAUSED A VOLTAGE SPIKE ON THE DC BUS (D01). THE VOLTAGE SPIKE AFFECTED THE ASSOCIATED INSTRUMENT BUS POWER SUPPLY INVERTERS (1DY01 AND 2DY01) FOR BOTH UNITS AND THE STANDBY (SWING) INVERTER DYOA. THIS INITIATED A REACTOR PROTECTION SYSTEM ACTUATION IN BOTH UNITS. THE VOLTAGE OSCILLATION WAS CLEARED WITHIN 10 SECONDS BY MANUAL RECLOSURE OF THE BATTERY BREAKER. THE LOSS OF VOLTAGE ON THE UNIT 1 RED INSTRUMENT BUS RESULTED IN A 2/4 POWER RANGE REACTOR TRIP SIGNAL FROM CHANNELS N41 AND N43. N41 TRIPPED WHEN ITS POWER SUPPLY WAS DEENERGIZED. N43 WAS IN TRIP DUE TO MODIFICATION WORK. IT ALSO RESULTED IN A 1/2 INTERMEDIATE RANGE REACTOR TRIP SIGNAL. THE VOLTAGE PERTURBATION ON THE UNIT 2 RED INSTRUMENT BUS RESULTED IN A 1/2 IR N35 REACTOR TRIP SIGNAL AS WELL AS A 1/2 SOURCE RANGE N31 REACTOR TRIP SIGNAL. ALL REACTOR PROTECTION SYSTEM CIRCUITRY FUNCTIONED AS DESIGNED DURING THIS EVENT. AN INVESTIGATION INTO THE CIRCUMSTANCES OF THIS EVENT HAS REVEALED THAT WITHOUT THE FILTERING EFFECT OF THE BATTERY, THE BATTERY CHARGES WILL PRODUCE VOLTAGE PERTURBATIONS ON THEIR RESPECTIVE LOADS (THE INSTRUMENT BUSES).

[205] POINT BEACH 1 DOCKET 50-266 LER 87-005
REACTOR TRIP WITH SAFETY INJECTION DUE TO SPRAY VALVE FAILURE.
EVENT DATE: 112187 REPORT DATE: 121587 NSSS: WE TYPE: PWR
VENDOR: BAILEY CONTROLS CO.
FOXBORO CO., THE

(NSIC 207427) ON NOVEMBER 21, 1987, AT 0305 HOURS, UNIT 1 REACTOR COOLANT PRESSURE BEGAN DECREASING AND THE REACTOR TRIPPED DUE TO LOW PRESSURIZER PRESSURE. REACTOR COOLANT PRESSURE CONTINUED TO DECREASE UNTIL SAFETY INJECTION INITIATED FLOW TO THE REACTOR COOLANT SYSTEM. PRESSURE WAS LOST WHEN A PRESSURIZER SPRAY VALVE STUCK OPEN. AN UNUSUAL EVENT WAS DECLARED AT APPROXIMATELY 0331 HOURS AND WAS TERMINATED AT 0457 HOURS. AN INSPECTION OF THE PRESSURIZER PRESSURE CONTROL EQUIPMENT REVEALED TWO FAILURES, A BROKEN SOLDER JOINT IN A PRESSURIZER SPRAY VALVE CONTROLLER AND A STUCK PILOT VALVE IN THE ASSOCIATED PRESSURIZER SPRAY VALVE POSITIONER RESULTED IN THE SPRAY VALVE BEING MAINTAINED IN THE FULL OPEN POSITION . THE MINIMUM PRESSURE REACHED WHILE THE SPRAY VALVE WAS STUCK OPEN WAS 1378 PSIG. THE MAXIMUM PRESSURE REACHED AFTER THE PRESSURIZER WAS FILLED WITH WATER WAS APPROXIMATELY 2077 PSIG. THE CONTROLLER AND PILOT STEM WERE REPAIRED AND RETURNED TO SERVICE. THE UNIT WAS THEN RETURNED TO FULL POWER OPERATION.

[206] POINT BEACH 2 DOCKET 50-301 LER 87-005
TURBINE RUNBACK CAUSED BY A DROPPED CONTROL ROD.
EVENT DATE: 112087 REPORT DATE: 121787 NSSS: WE TYPE: PWR
VENDOR: BUSSMANN MFG (DIV OF MCGRAW-EDISON)

(NSIC 207438) ON NOVEMBER 20, 1987, UNIT 2 WAS IN THE PROCESS OF INCREASING POWER FOLLOWING A REFUELING OUTAGE. REACTOR POWER WAS AT 53% WHEN CONTROL ROD C7 WAS DISCOVERED TO BE OUT OF ALIGNMENT. INVESTIGATION REVEALED THAT A FUSE IN THE MOVABLE LIFT COIL CIRCUIT FOR ROD C7 IN CONTROL BANK D WAS BLOWN. THE FUSE WAS REPLACED AND AN ATTEMPT WAS MADE TO WITHDRAW THE INDIVIDUAL ROD. DURING THIS ATTEMPT, A SECOND FUSE (A DIFFERENT ONE IN THE MOVABLE LIFT COIL CIRCUIT) BLEW ALLOWING THE ROD TO DROP FULLY INTO THE CORE. THE DROPPED ROD WAS SENSED BY NUCLEAR INSTRUMENTATION CAUSING A 20% TURBINE RUNBACK. FURTHER INVESTIGATION REVEALED THAT NO ELECTRICAL OR MECHANICAL PROBLEMS APPEARED TO EXIST WITH THE ROD DRIVE SYSTEM; THEREFORE, THE SECOND FUSE WAS REPLACED AND THE ROD WAS RECOVERED. THE CAUSE OF THE BLOWN FUSES APPEARS TO BE NORMAL FUSE FATIGUE.

[207] PRAIRIE ISLAND 1 DOCKET 50-282 LER 87-001
DIESEL GENERATOR INOPERABILITY CAUSED BY JACKET COOLANT SYSTEM AIR INLEAKAGE.
EVENT DATE: 020487 REPORT DATE: 112487 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: PRAIRIE ISLAND 2 (PWR)
VENDOR: FAIRBANKS MORSE

(NSIC 207322) ON FEBRUARY 4, 1987, UNIT 1 AND UNIT 2 WERE AT FULL POWER. A ROUTINE OPERABILITY TEST OF D2 DIESEL GENERATOR WAS IN PROGRESS. AT 1449, D2 DIESEL GENERATOR (D2) TRIPPED AND LOCKED OUT DUE TO LOW JACKET COOLANT PRESSURE (9 PSIG). D2 WAS DECLARED INOPERABLE AT THIS TIME. AT 1659, D2 WAS STARTED AND KEPT RUNNING FOR THE DURATION OF THE TEST BY PERIODICALLY VENTING AIR FROM THE JACKET COOLANT SYSTEM AND THUS MAINTAINING ADEQUATE SYSTEM PRESSURE (36 PSIG NORMAL). ACCUMULATION OF GAS IN THE ENGINE COOLING SYSTEMS HAD BEEN NOTED EARLIER, BUT HAD NOT JEOPARDIZED OPERABILITY OF THE ENGINE, AND NO CAUSE HAD YET BEEN FOUND. INVESTIGATION CONCLUDED IN LATE MARCH REVEALED THE SOURCE OF THE GAS INLEAKAGE AS NORMAL WEAR TO THE CARBON FACE OF THE MECHANICAL SEAL IN THE JACKET COOLANT PUMP. THIS ALLOWED AIR TO BE DRAWN INTO THE JACKET COOLANT SYSTEM, CAUSING THE JACKET COOLANT PUMP TO BECOME AIR BOUND, THUS CAUSING A DRASTIC DECREASE IN PRESSURE. THE MECHANICAL SEAL WAS REPLACED. SUBSEQUENT TESTING SHOWED THE AIR INLEAKAGE PROBLEM HAS BEEN SOLVED. D1 DIESEL GENERATOR WAS INSPECTED; NO SIMILAR PROBLEMS WERE FOUND.

[208] PRAIRIE ISLAND 1 DOCKET 50-282 LER 87-007
AUXILIARY FEED PUMP INOPERABILITY CAUSED BY FOREIGN MATERIAL IN SUCTION PIPING.
EVENT DATE: 051687 REPORT DATE: 112487 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: PRAIRIE ISLAND 2 (PWR)
VENDOR: PACIFIC PUMPS

(NSIC 207184) ON MAY 16, 1987, UNIT 1 WAS AT HOT SHUTDOWN FOLLOWING A REFUELING OUTAGE. DURING THE OUTAGE, NO. 11 TURBINE-DRIVEN AUXILIARY FEEDWATER PUMP (TDAFWP) MODIFICATION AND PREVENTIVE MAINTENANCE HAD BEEN DONE, AND EXTENSIVE MOTOR VALVE TESTING OF THE COOLING WATER SUPPLY VALVE MV-32025, HAD TAKEN PLACE. OPERABILITY TESTING OF THE PUMP REVEALED THE DISCHARGE PRESSURE WAS LOW. THE PUMP WAS DISASSEMBLED AND INSPECTED; BROKEN CLAMSHELLS AND SLUDGE WERE REMOVED FROM THE PUMP CASING AND A FEW CLAMSHELL PIECES WERE REMOVED FROM THE FIRST AND SECOND STAGE IMPELLERS. THE PUMP OPERATED SATISFACTORILY AFTER REASSEMBLY. ALL PUMPS WERE FLUSHED; SIMILAR AMOUNTS OF FOREIGN MATERIAL WERE REMOVED FROM THE OTHER PUMPS. PERIODIC FLUSHING IS NOW DONE. THE CAUSE OF THIS FAILURE WAS THE PRESENCE OF A MIXTURE OF SLUDGE AND BROKEN CLAM SHELLS LOCATED IN THE COOLING WATER SUPPLY LINE TO THE AFW PUMP. THIS MIXTURE WAS MOVED INTO THE SUCTION PIPING OF THE AUXILIARY FEEDWATER PUMP DURING UNUSUAL CONDITIONS REQUIRED TO TEST MOTOR OPERATED VALVE OPERATION WITH DIFFERENTIAL PRESSURE. THE PUMP WOULD HAVE REMAINED OPERABLE UNDER NORMAL SERVICE CONDITIONS.

[209] PRAIRIE ISLAND 1 DOCKET 50-282 LER 87-018
RELAY SPECIALIST INADVERTENTLY CAUSED INOPERABILITY OF ONE VOLTAGE RESTORING SCHEME.
EVENT DATE: 102887 REPORT DATE: 112487 NSSS: WE TYPE: PWR

(NSIC 207191) ON OCTOBER 28, 1987 UNIT 1 WAS AT STEADY-STATE FULL POWER. D1 DIESEL GENERATOR WAS OUT OF SERVICE FOR SCHEDULED ANNUAL PREVENTIVE MAINTENANCE. BUS 15 RELAY TESTING WAS BEING DONE IN CONJUNCTION WITH THE OUTAGE OF D1 DIESEL GENERATOR. A WIRE HAD BEEN LIFTED PER PROCEDURE TO ALLOW RELAY TESTING. IN ATTEMPTING TO RE-TERMINATE THE WIRE, THE RELAY SPECIALIST INADVERTENTLY TOUCHED THE LIFTED WIRE TO A STUD CONNECTION ON THE RELAY WHICH WAS CONNECTED TO THE NEGATIVE SIDE OF THE PANEL POWER SUPPLY, BLOWING THE PANEL POWER SUPPLY FUSE WHICH DEENERGIZED THE PANEL RELAYS, AND MAKING THE BUS 15 VOLTAGE RESTORING SCHEME INOPERABLE. DURING THE EVENT THE REDUNDANT TRAIN OF SAFEGUARDS EQUIPMENT REMAINED OPERABLE. BUS 15 REMAINED ENERGIZED FROM ITS NORMAL OFFSITE SOURCE, AND ITS ALTERNATE OFFSITE SOURCE WAS ALWAYS AVAILABLE MANUALLY. THE BLOWN FUSE FOR BUS 15 VOLTAGE RESTORING SCHEME WAS REPLACED AND THE BUS WAS DECLARED OPERABLE WITHIN 13 MINUTES.

[210] PRAIRIE ISLAND 1 DOCKET 50-282 LER 87-019
FAILURE TO MANUALLY LOG AXIAL FLUX DIFFERENCE WITH PLANT PROCESS COMPUTER OUT OF SERVICE.
EVENT DATE: 111087 REPORT DATE: 121087 NSSS: WE TYPE: PWR

(NSIC 207396) ON NOVEMBER 10, 1987, UNIT 1 WAS AT 100% POWER. AT 0800 THE PLANT PROCESS COMPUTER FAILED WITH A CORRESPONDING LOSS OF THE FLUX DIFFERENCE DEVIATION ALARM REQUIRED BY TECH SPEC 3.10.B.9. IN ACCORDANCE WITH THE TECH SPECS, THE UNIT 1 REACTOR OPERATOR INITIATED THE "FLUX DEVIATION LOG" TO MANUALLY LOG AXIAL FLUX DIFFERENCE ON AN HOURLY BASIS. HE DIRECTED A REACTOR OPERATOR TRAINEE TO RECORD THE REQUIRED AXIAL FLUX DIFFERENCES. AT 1100, THE REACTOR OPERATOR REVIEWED THE TRAINEE'S LOG AND DISCOVERED THAT THE TRAINEE HAD LOGGED THE WRONG VALUES FOR AXIAL FLUX DIFFERENCE; INSTEAD OF LOGGING THE FLUX DIFFERENCE FOR EACH EXCORE DETECTOR, HE HAD LOGGED REACTOR POWER. AXIAL FLUX DIFFERENCE WAS LOGGED PROPERLY AT 1100 AND THEREAFTER UNTIL THE PLANT PROCESS COMPUTER WAS RESTORED. THE FLUX DIFFERENCE LOG IS BEING CHANGED TO CLARIFY ITS USE. THERE WAS NO EFFECT ON PUBLIC HEALTH AND SAFETY SINCE THE UNIT 1 POWER DISTRIBUTION WAS WELL WITHIN TECH SPEC LIMITS THROUGHOUT THE EVENT.

[211] QUAD CITIES 1 DOCKET 50-254 LER 87-022
NEUTRON MONITOR SPIKED HIGH HIGH ON REACTOR SCRAM WHILE SHUT DOWN.
EVENT DATE: 110987 REPORT DATE: 112387 NSSS: GE TYPE: BWR

(NSIC 207291) IN NOVEMBER 9, 1987, QUAD CITIES UNIT ONE WAS IN THE REFUEL MODE WITH FUEL BEING LOADED INTO THE CORE. AT 1941 HOURS, INTERMEDIATE RANGE MONITOR (IRM) 14 SPIKED HIGH-HIGH RESULTING IN A HALF SCRAM ON CHANNEL A OF THE REACTOR PROTECTION SYSTEM (RPS). RPS CHANNEL B ALREADY HAD A HALF SCRAM MANUALLY INSERTED DUE TO MAINTENANCE TO IRMS ON THAT RPS CHANNEL. THEREFORE. A FULL REACTOR SCRAM OCCURRED. AT 2000 OURS. NRC WAS NOTIFIED VIA THE EMERGENCY NOTIFICATION SYSTEM OF THIS EVENT PER 10CFR50.72. THE CAUSE OF THIS EVENT HAS NOT YET BEEN DETERMINED, BUT IT IS BELIEVED THAT IRM 14 HARDWARE IS AT FAULT. NUCLEAR WORK REQUEST Q59487 HAS BEEN WRITTEN TO INVESTIGATE AND REPAIR IRM 14. THIS SHOULD BE ACCOMPLISHED PRIOR TO UNIT ONE RESTART. THE CAUSE OF THE SPIKE AND ACTIONS TAKEN TO PREVENT RECURRENCE WILL BE ADDRESSED IN A SUPPLEMENTAL REPORT. THIS REPORT IS SUPPLIED PER 10CFR50.73(A)(2)(IV).

[212] RANCHO SECO DOCKET 50-312 LER 86-016 REV 02
UPDATE ON DECAY HEAT SYSTEM TRIP FOLLOWING AN ARC IN A SUMP LEVEL INDICATOR.
EVENT DATE: 100386 REPORT DATE: 111887 NSSS: BW TYPE: PWR
VENDOR: MASTER SPECIALTIES

(NSIC 207159) WHILE IN COLD SHUTDOWN ON OCTOBER 3, 1986, DURING INSTRUMENT & CONTROL INVESTIGATION OF ABNORMAL INDICATION ON PANEL H2SFB FOR DECAY HEAT SYSTEM (DHS) "B" ROOM SUMP STACK LIGHTS, SFAS "B" BISTABLES TRIPPED CAUSING HV-20002 TO CLOSE, WHICH TRIPPED DHS "B" PUMP. THE PLANT WAS WITHOUT THE USE OF THE NORMAL DHS FOR APPROXIMATELY 13 MINUTES. DUE TO THE EXTENDED PERIOD THAT THE PLANT HAS BEEN SHUTDOWN, THERE WAS A SMALL, BUT DETECTABLE INCREASE OF REACTOR COOLANT TEMPERATURE. STEPS WERE TAKEN IMMEDIATELY TO RESTORE A DHS TRAIN TO SERVICE IN ACCORDANCE WITH THE INTENT OF TECH SPEC 3.1.1.5. THIS EVENT IS REPORTABLE ACCORDING TO 10 CFR PART 50.73(A)(2)(IV & V). THE IMMEDIATE CAUSE OF THE INCIDENT WAS AN ARC THAT OCCURRED WHEN I&C TECHNICIANS WERE TROUBLESHOOTING AN ABNORMAL INDICATION ON PANEL H2SFB FOR DHS "B" PUMP ROOM (EAST) SUMP STACK LIGHTS (18 INCH LEVEL INDICATION). THE ARC INITIATED THE TRIP OF INVERTER "B". THE PAINT ON THE LENS RETAINING CLIP OF THE INDICATOR HAD DETERIORATED, AND THE METAL OF THE CLIP PROVIDED A PATH TO GROUND. AS CORRECTIVE ACTIONS, THE LENS CLIPS FOR THE INDICATORS INSTALLED IN THE PLANT WILL BE REPAINTED OR REPLACED BY FEBRUARY 28, 1988. ADDITIONALLY, THIS EVENT WILL BE INCLUDED IN ONGOING OPERATOR TRAINING. AS A LONG TERM CORRECTIVE ACTION, THE DC VITAL POWER SUPPLIES WILL BE MODIFIED TO BE EQUIPPED WITH STATIC TRANSFER SWITCHES.

[213] RANCHO SECO DOCKET 50-312 LER 87-036 REV 01
UPDATE ON BLOCKAGE OF BEARING COOLER WATER SUPPLY AND RETURN LINES.
EVENT DATE: 060887 REPORT DATE: 111887 NSSS: BW TYPE: PWR
VENDOR: BABCOCK & WILCOX CANADA LTD.

(NSIC 207197) ON JUNE 8, 1987, DURING ROUTINE MAINTENANCE IN COLD SHUTDOWN CONDITIONS, THE DISTRICT DISCOVERED THAT THE PIPE SUPPLYING WATER FOR BEARING COOLING FOR THE REACTOR BUILDING SPRAY (RBS) PUMP "B" WAS COMPLETELY BLOCKED. PREVIOUSLY, RBS PUMP "A" WAS FOUND PARTIALLY BLOCKED ON APRIL 18, 1987. THE FUNCTION OF THE WATER IS TO MAINTAIN THE TEMPERATURE OF THE PUMP BEARINGS LESS THAN 125F WITH A MINIMUM WATER FLOW RATE OF FOUR GPM. THE NSRW SPRAY PONDS WERE CLEANED IN PREVIOUS REFUELING OUTAGES, HOWEVER, IT WAS NOT ON A DOCUMENTED REGULAR INTERVAL. THE DISTRICT HAS INSTITUTED A PREVENTIVE MAINTENANCE PROCEDURE FOR PUMP BEARING COOLING WATER SUPPLY AND RETURN LINES, TO BE PERFORMED AS A MINIMUM AT EACH REFUELING OUTAGE.

[214] RANCHO SECO DOCKET 50-312 LER 87-041
POTENTIAL TRIP OF NUCLEAR SERVICE COOLING WATER PUMPS DUE TO DESIGN DEFICIENCY.
EVENT DATE: 102687 REPORT DATE: 112487 NSSS: BW TYPE: PWR
VENDOR: ROBERTSHAW CONTROLS COMPANY

(NSIC 207336) A DESIGN REVIEW OF THE NUCLEAR SERVICE COOLING WATER SYSTEM (NSCW) REVEALED CLASS 2 LEVEL SWITCHES IN THE CONTROL CIRCUITRY FOR THE TWO REDUNDANT NSCW PUMPS. THESE PUMPS DRAW SUCTION FLOW FROM THEIR RESPECTIVE NSCW SURGE TANKS. THE SWITCHES ARE DESIGNED TO TRIP THE NSCW PUMPS ON LOW NSCW SURGE TANK LEVEL. THE LEVEL SWITCHES DO NOT HAVE SAFETY FEATURES ACTUATION SIGNAL (SFAS) OVERRIDE, THUS ALLOWING A SWITCH FAILURE TO PREVENT THE PUMP FROM STARTING ON SFAS SIGNAL. DURING A SEISMIC EVENT, IT IS POSSIBLE FOR BOTH LEVEL SWITCHES TO TRIP THEIR RESPECTIVE NSCW PUMPS. A TRIP OF THE NSCW PUMPS WOULD CONSTITUTE A CONDITION THAT ALONE COULD PREVENT THE FULFILLMENT OF THE SAFETY FUNCTION OF A SYSTEM NEEDED TO REMOVE RESIDUAL HEAT OR MITIGATE THE CONSEQUENCES OF AN ACCIDENT. THIS EVENT IS REPORTABLE TO THE NRC IN ACCORDANCE WITH 10 CFR 50.73(A)(V)(B) AND 10 CFR 50.73(A)(V)(D).

[215] RANCHO SECO DOCKET 50-312 LER 87-045
SURVEILLANCE PROCEDURE NOT PERFORMED BY TECHNICAL SPECIFICATION REQUIRED DATE DUE TO PERSONNEL ERROR.
EVENT DATE: 110587 REPORT DATE: 120487 NSSS: BW TYPE: PWR

(NSIC 207348) ON NOVEMBER 5, 1987, SURVEILLANCE PROCEDURES SP.301, SP.304, SP.307, AND SP.311 WERE COMPLETED. THESE SURVEILLANCE PROCEDURES SATISFY THE BATTERIES' MONTHLY AND WEEKLY TEST REQUIREMENTS. WEEKLY BATTERY TESTING IS NORMALLY PERFORMED USING PROCEDURES SP.300, SP.303, SP.306, AND SP.310; HOWEVER, THE MONTHLY PROCEDURES HAVE BEEN WRITTEN TO SATISFY THE WEEKLY TESTING REQUIREMENTS FOR TIMES WHEN THE MONTHLY AND WEEKLY TESTS ARE THE SAME. THE DUE DATE FOR THE WEEKLY BATTERY TESTING WAS NOVEMBER 2, 1987. IF THE TECH SPEC ALLOWED 25% EXTENSION WAS ADDED, THE DUE DATE WOULD HAVE BEEN NOVEMBER 4, 1987. SURVEILLANCE PROCEDURES SP.301, SP.304, SP.307, AND SP.311 WERE NOT PERFORMED UNTIL NOVEMBER 5, 1987, THEREBY RENDERING THE BATTERIES TECHNICALLY INOPERABLE FOR ONE DAY. THE SURVEILLANCE TEST COMPLETED ON THE FOLLOWING DAY DEMONSTRATED THAT THE BATTERIES WERE OPERABLE. A SEARCH OF PREVIOUSLY SUBMITTED LERS DISCLOSED THREE SIMILAR OCCURRENCES. LERS 85-08, 86-06, AND LER 87-12 WERE RELATED TO TECH SPEC REQUIRED SURVEILLANCES MISSED DUE TO SCHEDULING PROBLEMS.

[216] RANCHO SECO DOCKET 50-312 LER 87-044
POTENTIAL LOSS OF NUCLEAR SERVICE ELECTRICAL BUILDING ESSENTIAL HVAC DUE TO A DESIGN ERROR.
EVENT DATE: 111087 REPORT DATE: 120487 NSSS: BW TYPE: PWR
VENDOR: PACIFIC AIR PRODUCTS

(NSIC 207347) A DISTRICT REVIEW OF THE NUCLEAR SERVICE ELECTRICAL BUILDING (NSEB) ESSENTIAL HVAC RAISED CONCERNS ABOUT THE DESIGN ADEQUACY OF THE ISOLATION DAMPER BETWEEN SEISMIC I AND SEISMIC III DUCT WORK. THE CONCERN IS THAT DURING A SEISMIC EVENT, THE AIR SUPPLY TO THE DAMPERS WHICH ISOLATE SEISMIC I FROM SEISMIC III DUCTWORK COULD FAIL. THE ISOLATION DAMPERS ARE NORMAL-OPEN, FAIL-OPEN DAMPERS, AND INSTRUMENT AIR IS REQUIRED FOR THEM TO PERFORM THEIR DESIGNED SAFETY FUNCTION, TO CLOSE. IF THE ISOLATION DAMPERS FAIL TO CLOSE THE NSEB ESSENTIAL HVAC WILL NOT PERFORM ITS INTENDED SAFETY FUNCTION. THIS CONDITION EXISTS IN BOTH HSEB HVAC TRAINS. THE FAILURE OF THE ISOLATION DAMPERS TO CLOSE WOULD CONSTITUTE A CONDITION THAT ALONE COULD PREVENT THE FULFILLMENT OF THE SAFETY FUNCTION OF A SYSTEM THAT IS NEEDED TO MITIGATE THE CONSEQUENCES OF AN ACCIDENT. THIS CONDITION IS REPORTABLE IN ACCORDANCE WITH 10 CFR 50.73(A)(2)(V)(D). TO RESOLVE THIS PROBLEM THE ISOLATION DAMPERS WILL BE CHANGED TO NORMAL-OPEN FAIL-CLOSED LOGIC BY JANUARY 4, 1988.

[217] RANCHO SECO DOCKET 50-312 LER 87-047
FAILURE TO CONDUCT CONTINUOUS SAMPLE DURING REACTOR BUILDING PURGE DUE TO PERSONNEL ERROR.
EVENT DATE: 112487 REPORT DATE: 122287 NSSS: BW TYPE: PWR

(NSIC 207439) AT 1758 ON NOVEMBER 16, 1987, A REACTOR BUILDING (RB) PURGE WAS INITIATED BY OPERATIONS PERSONNEL PRIOR TO INITIATING A CONTINUOUS PURGE SAMPLE. THE RP TECHNICIAN WAS INFORMED AT APPROXIMATELY 1803 THAT THE RB PURGE HAD BEEN INITIATED. HE STARTED A CONTINUOUS RB PURGE STACK SAMPLE AT ABOUT 1805. ON NOVEMBER 24, 1987, RADIATION PROTECTION DISCOVERED THE FAILURE TO CONDUCT CONTINUOUS SAMPLING FROM 1758 TO 1805 ON NOVEMBER 16, 1987, DURING A NORMAL ADMINISTRATIVE REVIEW. TECHNICAL SPECIFICATION 3.16, TECHNICAL SPECIFICATION TABLE 3.16-1, AND TECHNICAL SPECIFICATION TABLE 4.22-1 REQUIRE CONTINUOUS SAMPLING DURING RB PURGES. PREVIOUS SIMILAR LERS: 86-03, 86-19, 86-22, 86-28, 86-29, 86-33, 87-43, AND RO-77-4.

[218] RIVERBEND 1 DOCKET 50-458 LER 86-024 REV 05
UPDATE ON ESF ACTUATIONS DUE TO AN EPA BREAKER FAILURE.
EVENT DATE: 032586 REPORT DATE: 111687 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CORP. (NUCLEAR ENG DIV)

(NSIC 207221) AT 1626 ON 3/25/86 WITH THE UNIT SUBCRITICAL, POWER TO REACTOR PROTECTION SYSTEM (RPS) BUS B WAS LOST DUE TO A TRIP OF AN ELECTRICAL PROTECTION ASSEMBLY (EPA) BREAKER. THE CAUSE WAS UNKNOWN AND UNDER INVESTIGATION WHEN ANOTHER TRIP OF THE SAME BREAKER OCCURRED AT 2211 THE SAME DAY. THE LOSS OF THE RPS BUS CAUSED RESIDUAL HEAT REMOVAL (RHR) TRAIN A TO BE ISOLATED WHICH RESULTED IN A LOSS OF SHUTDOWN COOLING. AT 1636, RHR TRAIN A WAS RESTARTED IN THE SHUTDOWN COOLING MODE AND THE ISOLATIONS WERE RESET. THE CAUSE OF THE EPA BREAKER TRIP WAS TRACED TO A POTENTIAL PROBLEM IN THE TRIP CONDITION CIRCUITRY OF THE EPA BREAKERS WHICH WERE EXERCISED DURING THE PERFORMANCE OF A SURVEILLANCE TEST PROCEDURE. THE BREAKER WAS REWORKED PER DESIGN, TESTED AND RETURNED TO SERVICE ON 4/2/86. AT 1911 ON 12/24/86 WITH THE UNIT AT FULL POWER, POWER TO THE REACTOR PROTECTION SYSTEM (RPS) BUS B WAS AGAIN LOST RESULTING IN A HALF-SCRAM AND THE DIVISION 2 ISOLATIONS OF REACTOR WATER CLEANUP AND OTHER DIVISION 2 CONTAINMENT ISOLATION VALVES. IMMEDIATE ACTION WAS TAKEN TO PLACE RPS BUS B ON ITS ALTERNATE POWER SUPPLY AND POWER WAS RESTORED IN 12 APPROXIMATELY ONE MINUTE. THE ISOLATIONS WERE RESET AND THE SYSTEM LINEUPS WERE RETURNED TO NORMAL. AN IMMEDIATE FUNCTIONAL TEST OF THE EPA WAS PERFORMED SHOWING NO APPARENT MALFUNCTION. AFTER THE TEST, THE EPA WAS RETURNED TO SERVICE.

[219] RIVERBEND 1 DOCKET 50-458 LER 87-020 REV 01
UPDATE ON MISSED POST MAINTENANCE SURVEILLANCE ON ISI VALVES.
EVENT DATE: 060686 REPORT DATE: 120287 NSSS: GE TYPE: BWR

(NSIC 207244) DURING AN INTERNAL AUDIT CONDUCTED IN SEPTEMBER 1987 ON ALL ASME WORK PACKAGES, ON SEPTEMBER 25 AND 26, 1987 RESPECTIVELY, IT WAS DISCOVERED THAT AFTER PERFORMING MAJOR REPAIR WORK ON ASME VALVES 1B21*AOVF032B WHICH WAS PLACED BACK INTO SERVICE ON NOVEMBER 29, 1986 AND ASME VALVE 1E21*AOVF006 WHICH WAS PLACED BACK INTO SERVICE ON 6/6/86, THEY WERE DECLARED OPERABLE AND PLACED BACK INTO SERVICE WITHOUT PERFORMING REQUIRED ASME XI VALVE EXERCISE TESTS. THE OMISSION OF THESE TESTS ON MAINTENANCE WORK DOCUMENTS WAS DUE TO PROGRAM INADEQUACIES IN ENGINEERING CONTROL AND MAINTENANCE PLANNING. CORRECTIVE ACTION INCLUDED ENGINEERING REORGANIZATION IN JANUARY, 1987, AND RETRAINING OF THE ENGINEERING, OPERATIONS, AND MAINTENANCE DEPARTMENTS ON THE CIRCUMSTANCES OF THIS REPORTABLE CONDITION. ALSO, REVISIONS OF PLANT AND ENGINEERING PROCEDURES WHICH COVER ASME XI REPAIR & REPLACEMENT AND THE MAINTENANCE WORK ORDER WERE REVISED TO PRECLUDE THESE OCCURRENCES. SINCE ALL VALVES IN QUESTION WERE VERIFIED OPERABLE AND COULD HAVE PERFORMED THEIR SAFETY FUNCTIONS, THERE WAS NO ADVERSE IMPACT ON THE HEALTH AND SAFETY OF THE PUBLIC, AS A RESULT OF THIS CONDITION.

[220] RIVERBEND 1 DOCKET 50-458 LER 86-050 REV 01
UPDATE ON RWCU FAILURE TO ISOLATE DUE TO A FAULTY BYPASS SWITCH.
EVENT DATE: 080786 REPORT DATE: 101987 NSSS: GE TYPE: BWR
VENDOR: RILEY-BEAIRD, INC.

(NSIC 206861) AT 0633 ON 8/7/86 WITH THE PLANT IN MODE 1 AND OPERATING AT 100 PERCENT RATED REACTOR POWER, THE DIVISION II LEAKAGE DETECTION SYSTEM DIFFERENTIAL TEMPERATURE SWITCH BEGAN ACTUATING ERRATICALLY. THE REACTOR WATER CLEANUP (RWCU) HEAT EXCHANGER ROOM HIGH TEMPERATURE ALARM ANNUNCIATED ONCE AND STAYED ON A SECOND TIME DUE TO THE DIFFERENTIAL TEMPERATURE SWITCH ACTUATIONS, BUT THE EXPECTED RWCU ISOLATION DID NOT OCCUR. THE RWCU ISOLATION LOGIC BYPASS SWITCH WAS TAKEN TO THE BYPASS POSITION TO FACILITATE REMOVING THE DEMINERALIZER FILTER BEDS FROM SERVICE. WHEN THIS SWITCH WAS RETURNED TO THE NORMAL POSITION, THE DIVISION II RWCU VALVES ISOLATED. THE DIFFERENTIAL TEMPERATURE AND LOGIC BYPASS SWITCHES WERE REPLACED, AFTER WHICH THE RWCU SYSTEM WAS RESTORED TO

OPERATION. SINCE THE ISOLATION SIGNAL GENERATED BY THE FAILED DIFFERENTIAL TEMPERATURE SWITCH WAS NOT VALID, THE HEALTH AND SAFETY OF THE PUBLIC WAS NOT AFFECTED BY THIS EVENT.

[221] RIVERBEND 1 DOCKET 50-458 LER 87-009 REV 01
UPDATE ON LOW PRESSURE CORE SPRAY/INJECTION INOPERABLE DUE TO IMPROPER TRANSMITTER CALIBRATION.
EVENT DATE: 060287 REPORT DATE: 121687 NSSS: GE TYPE: BWR
VENDOR: CROSBY VALVE
ROSEMOUNT, INC.

(NSIC 207489) AT 1220 ON 6/2/87, WITH THE UNIT AT APPROXIMATELY 70 PERCENT POWER, IT WAS DETERMINED THAT THE SETPOINTS FOR THE REACTOR PRESSURE VESSEL SAFETY RELIEF VALVES WERE OUTSIDE THE TECHNICAL SPECIFICATION ALLOWABLE VALUES. A REVIEW OF THE LOOP CALIBRATION REPORTS REVEALED THAT THE LOW PRESSURE ECCS INJECTION PERMISSIVE FUNCTIONS WERE ALSO CALIBRATED OUTSIDE THE TECHNICAL SPECIFICATION LIMITS. INVESTIGATION INDICATED THAT THERE WAS NO HEAD CORRECTION TAKEN INTO ACCOUNT IN THE CALIBRATION FOR THESE TRANSMITTERS. THE AFFECTED SYSTEMS WERE DECLARED INOPERABLE AND THE ASSOCIATED TECHNICAL SPECIFICATION ACTION STATEMENTS WERE ENTERED. AS IMMEDIATE CORRECTIVE ACTION, THE TRIP UNITS ASSOCIATED WITH THESE TRANSMITTERS WERE ADJUSTED TO COMPENSATE FOR THE LACK OF HEAD CORRECTION IN THE TRANSMITTER CALIBRATION. THE ASSOCIATED SYSTEMS WERE SUBSEQUENTLY DECLARED OPERABLE WITHIN THE 12 HOURS ALLOWED BY THE RIVER BEND STATION TECHNICAL SPECIFICATIONS. PERMANENT CORRECTIVE ACTION WAS IMPLEMENTED DURING AN OUTAGE BEGINNING 6/18/87 WHEN THE TRANSMITTERS WERE RECALIBRATED AND THE TRIP UNITS RETURNED TO THEIR ORIGINAL SETTINGS. ALL CALIBRATIONS WERE COMPLETED BY 6/23/87. THERE WAS NO IMPACT ON THE SAFE OPERATION OF THE PLANT OR TO THE HEALTH AND SAFETY OF THE PUBLIC AS A RESULT OF THIS EVENT. THE CONTINUED OPERATION OF THE PLANT HAD NO IMPACT ON SAFETY.

[222] RIVERBEND 1 DOCKET 50-458 LER 87-023
RADIATION MONITOR HEAT EXCHANGERS PLUGGED WITH CORROSION DUE TO SERVICE WATER CHEMISTRY.
EVENT DATE: 101987 REPORT DATE: 112587 NSSS: GE TYPE: BWR
VENDOR: GENERAL ATOMIC CO.

(NSIC 207285) ON 10/19/87 DURING SHUTDOWN (MODE 5), THE COOLING WATER LINES TO HEAT EXCHANGER 1RMS*HX11A SERVING ONE OF THE TWO REDUNDANT ANNULUS EXHAUST RADIATION MONITORS (1RMS*RE11A) WERE FOUND TO BE PLUGGED DUE TO A BUILDUP OF CORROSION. THIS DISCOVERY WAS MADE DURING ROUTINE INSPECTION OF THE HEAT EXCHANGERS. ON 11/18/87, THE COOLING WATER LINES TO THE HEAT EXCHANGER IN THE REDUNDANT TRAIN WERE ALSO FOUND TO BE PLUGGED DUE TO CORROSION BUILDUP. THE ROOT CAUSE THAT LED TO PLUGGING OF THESE LINES WAS THE PAST CONDITION OF SERVICE WATER CHEMISTRY. CORRECTIVE ACTION HAS BEEN INITIATED TO CLEAN THE CORROSION FROM THE COOLING WATER SUPPLY AND RETURN LINES OR REPLACE THE CARBON STEEL LINES WITH STAINLESS STEEL FOR BOTH RADIATION MONITOR HEAT EXCHANGERS. DEMONSTRATION OF ADEQUATE FLOW TO THESE RADIATION MONITORS WILL BE PERFORMED ON A SIX MONTH SCHEDULE UNTIL A CORROSION INHIBITOR IS INTRODUCED INTO THE SERVICE WATER SYSTEM. THE INTRODUCTION OF A CORROSION INHIBITOR ALONG WITH A CORROSION MONITORING AND CONTROL PROGRAM AND ROUTINE INSPECTIONS, WILL PREVENT THIS CONDITION FROM DEVELOPING IN THE FUTURE. FAILURE OF THESE RADIATION MONITORS WOULD NOT HAVE RESULTED IN A SIGNIFICANT RISK TO THE PUBLIC. POTENTIAL UNFILTERED RELEASES WOULD HAVE BEEN PREVENTED BY REDUNDANT, DIVERSE RADIATION INDICATION.

[223] RIVERBEND 1 DOCKET 50-458 LER 87-024
REACTOR PROTECTION SYSTEM INITIATION DUE TO PERSONNEL MISCOMMUNICATION.
EVENT DATE: 102087 REPORT DATE: 111987 NSSS: GE TYPE: BWR

(NSIC 207220) AT APPROXIMATELY 0449 HOURS ON 10/20/87 WITH THE UNIT IN MODE 5 (REFUELING), INSTRUMENT AND CONTROLS (I&C) TECHNICIANS INITIATED A FULL REACTOR SCRAM SIGNAL ON THE REACTOR PROTECTION SYSTEM (RPS) WHILE PERFORMING THE INSTALLATION OF MODIFICATION REQUEST (MR) 85-0770. WHILE I&C TECHNICIANS WERE COMMUNICATING WITH ON SHIFT OPERATIONS PERSONNEL DURING THE PERFORMANCE OF THIS MR, A BREAKDOWN IN COMMUNICATIONS INADVERTENTLY CAUSED THE OPERATIONS PERSONNEL TO FAIL TO REALIZE THAT THE I&C TECHNICIANS WERE PROCEDING TO PERFORM WORK ON THE OTHER DIVISION OF THE RPS. WITH A HALF SCRAM SIGNAL ALREADY PRESENT RESULTING FROM A FUSE THAT HAD BLOWN UNKNOWN TO THE I&C TECHNICIANS, THE COMMENCEMENT OF WORK ON THE OTHER DIVISION COMPLETED THE LOGIC FOR A FULL SCRAM SIGNAL. FOR OPEN JOB PLANS THAT REQUIRE WORK ON MORE THAN ONE LOGIC CHANNEL, A PROCEDURAL STEP REQUIRING INITIAL AND DATE HAS BEEN IMPLEMENTED TO ENSURE WORK IS COMPLETE AND LOGIC RESET ON ONE CHANNEL PRIOR TO PROCEDING TO ANOTHER CHANNEL. NO CONTROL ROD MOVEMENT WAS EXPERIENCED IN THIS EVENT SINCE ALL RODS WERE ALREADY INSERTED. HAD THIS EVENT OCCURRED DURING POWER OPERATIONS, RPS WOULD HAVE PERFORMED SIMILARLY INITIATING A REACTOR SCRAM AND PLACING THE PLANT IN A SHUTDOWN CONDITION. SINCE THE RPS PERFORMED AS DESIGNED, THERE WAS NO ADVERSE IMPACT ON THE HEALTH AND SAFETY OF THE PUBLIC AS A RESULT OF THIS EVENT.

[224] RIVERBEND 1 DOCKET 50-458 LER 87-025
VALID FAILURE OF DIESEL GENERATOR DUE TO OUTPUT BREAKER FAILURE.
EVENT DATE: 102187 REPORT DATE: 112087 NSSS: GE TYPE: BWR
VENDOR: GOULD BROWN BOVERI COMPANY

(NSIC 207222) AT 1235 ON 10/21/87 A VALID FAILURE OCCURRED ON THE DIVISION II DIESEL GENERATOR (DG) 1EGS*EG1B WHILE PERFORMING THE MONTHLY SURVEILLANCE TEST. IN ADDITION, THE DIVISION I DG WAS DISASSEMBLED FOR 18 MONTH REFUELING OUTAGE INSPECTIONS. AT THE TIME OF THE DIVISION II DG FAILURE, RBS HAD NO OPERABLE ONSITE AC SOURCES. THEREFORE, THIS CONDITION COULD HAVE PREVENTED THE FULFILLMENT OF THE SAFETY FUNCTION OF REQUIRED SYSTEMS IN THE EVENT OF A LOSS OF OFFSITE POWER. THIS REPORT ALSO SATISFIES THE REPORTING REQUIREMENTS OF TECHNICAL SPECIFICATIONS 4.8.1.1.3 AND 6.9.2. AN IMMEDIATE LOCAL INSPECTION OF THE BREAKER REVEALED THAT THE BREAKER WAS RACKED IN BEYOND THE "CONNECTED" POSITION THEREBY HOLDING THE MECHANICAL RACKING RELEASE LEVER OUT AND PREVENTING THE BREAKER FROM CLOSING. THE BREAKER WAS SUBSEQUENTLY RACKED INTO ITS PROPER POSITION AND THE BREAKER WAS SUCCESSFULLY CLOSED. THE DG WAS SUBSEQUENTLY RETURNED TO OPERABLE STATUS AT 2200 ON 10/21/87. THE ROOT CAUSE OF THE BREAKER FAILURE HAS BEEN INVESTIGATED. THIS INVESTIGATION HAS CONCLUDED THAT THE CAUSE OF FAILURE IS INDETERMINATE. AT THE TIME OF THIS FAILURE, THE UNIT WAS IN OPERATIONAL MODE 5 (REFUELING) WITH REACTOR WATER LEVEL GREATER THAN 23 FEET ABOVE THE REACTOR VESSEL FLANGE.

[225] RIVERBEND 1 DOCKET 50-458 LER 87-028
RESIDUAL HEAT REMOVAL SYSTEM ISOLATION DUE TO HIGH DIFFERENTIAL TEMPERATURE.
EVENT DATE: 110687 REPORT DATE: 120787 NSSS: GE TYPE: BWR

(NSIC 207312) AT APPROXIMATELY 1900 ON 11/6/87, WITH THE UNIT IN OPERATIONAL CONDITION 5 (REFUELING) AND THE RESIDUAL HEAT REMOVAL SYSTEM (RHR) IN THE SHUTDOWN COOLING MODE OF OPERATION AND PUMPING THE UPPER CONTAINMENT FUEL POOLS TO THE RADWASTE SYSTEM, THE SHUTDOWN COOLING SUCTION VALVE (1E12*MOVF009) AND THE DISCHARGE VALVE TO RADWASTE (1E12*MOVF049) AUTOMATICALLY CLOSED. AS A RESULT, RHR PUMP A (1E12*PC002A) TRIPPED. THE CAUSE OF THESE ISOLATIONS WAS AN ACTUAL HIGH DIFFERENTIAL TEMPERATURE SIGNAL FROM THE RHR A PUMP ROOM LEAK DETECTION SYSTEM. AT THE TIME OF THE ISOLATIONS THE RHR A PUMP ROOM UNIT COOLER WAS NOT IN OPERATION. THE UNIT COOLER WAS RETURNED TO SERVICE, THE ISOLATION SIGNAL WAS RESET, AND THE RHR A LOOP WAS SUBSEQUENTLY RETURNED TO SERVICE AT 1905. THE ROOT CAUSE OF THE HIGH DIFFERENTIAL TEMPERATURE CONDITION HAS NOT YET BEEN DETERMINED. INVESTIGATION IS STILL CONTINUING, HOWEVER, TEMPERATURE STRATIFICATION DUE TO LACK OF VENTILATION IS THE MOST PROBABLE CAUSE. SINCE NO ACTUAL REACTOR COOLANT

SYSTEM LEAKAGE EXISTED AND SHUTDOWN COOLING WAS RESTORED IN APPROXIMATELY FIVE MINUTES, THERE WAS NO IMPACT ON THE HEALTH AND SAFETY OF THE PUBLIC AS A RESULT OF THIS EVENT.

[226] RIVERBEND 1 DOCKET 50-458 LER 87-026
MISSED FIRE DOOR SURVEILLANCE DUE TO AN ERROR IN THE SURVEILLANCE TEST PROCEDURE.
EVENT DATE: 111287 REPORT DATE: 121187 NSSS: GE TYPE: BWR

(NSIC 207490) AT APPROXIMATELY 1000 HOURS ON 11/12/87, WITH THE UNIT IN OPERATIONAL CONDITION 5 (REFUELING), IT WAS DISCOVERED THAT TWO FIRE DOORS WERE NOT INCLUDED IN THE SURVEILLANCE TEST PROCEDURE (STP). THESE DOORS ARE REQUIRED BY TECHNICAL SPECIFICATIONS TO BE CLOSED, AND THEIR POSITION CHECKED DAILY, TO ENSURE SEPARATION OF REDUNDANT TRAINS OF SAFETY RELATED EQUIPMENT. A REVIEW OF THE PROCEDURE HISTORY FILE REVEALED THAT THESE TWO DOORS WERE INADVERTENTLY OMITTED ON 10/13/87 WHILE MAKING THE REVISION DURING THE ANNUAL REVIEW OF THIS PROCEDURE. A REVIEW OF THE FIRE WATCH/PATROL LOG FROM 10/13/87 TO 11/12/87 INDICATED THAT NO FIRE WATCH OR PATROL HAD BEEN IN EFFECT FOR THESE AREAS DURING THIS TIME PERIOD. SINCE THE DAILY FIRE DOOR POSITION CHECK WAS NOT PERFORMED ON THESE DOORS AND NO FIRE WATCH WAS POSITIONED DURING THE TIME THESE SURVEILLANCES WERE MISSED, THIS CONDITION CONSTITUTED A CONDITION PROHIBITED BY THE RIVER BEND STATION TECHNICAL SPECIFICATIONS. AT APPROXIMATELY 1150 ON 11/12/87, THE STP WAS REVISED TO INCLUDE THE TWO FIRE DOORS IDENTIFIED. THE STP WAS SUBSEQUENTLY SATISFACTORILY PERFORMED BY 1300 ON THE SAME DAY. FURTHER CORRECTIVE ACTION HAS BEEN TAKEN TO REVIEW ALL FIRE DOOR STP'S AGAINST THE APPROPRIATE DESIGN DOCUMENTS.

[227] RIVERBEND 1 DOCKET 50-458 LER 87-030
FAILURE TO PERFORM AREA TEMPERATURE MONITORING.
EVENT DATE: 111887 REPORT DATE: 121887 NSSS: GE TYPE: BWR

(NSIC 207493) ON 11/18/87, WITH THE UNIT IN OPERATIONAL CONDITION 4 (COLD SHUTDOWN), IT WAS DISCOVERED THAT THE SURVEILLANCE REQUIREMENT FOR TECHNICAL SPECIFICATION 3.7.8 "AREA TEMPERATURE MONITORING" HAD NOT BEEN PROPERLY PERFORMED FOR THE REACTOR PLANT COMPONENT COOLING WATER (RPCCW) SYSTEM (CC) AREAS. THE CAUSE OF THE SURVEILLANCE OMISSION FROM THE PROCEDURE IS CONSIDERED TO BE RELATED TO MULTIPLE SURVEILLANCES BEING CONDUCTED VIA A SINGLE PROCEDURE. AS CORRECTIVE ACTION, THIS AND OTHER MULTIPLE SURVEILLANCE PROCEDURES WERE REVIEWED FOR OMISSIONS AND THE MISSED SURVEILLANCES HAVE BEEN ADDED TO THE APPLICABLE STPS. THE PRELIMINARY EVALUATION HAS CONCLUDED THAT IT IS VERY UNLIKELY THAT THIS TEMPERATURE LIMIT HAS EVER BEEN EXCEEDED. THEREFORE, THERE WAS NO SIGNIFICANT IMPACT ON HEALTH AND SAFETY OF THE PUBLIC AS A RESULT OF THIS PROCEDURAL OMISSION.

[228] RIVERBEND 1 DOCKET 50-458 LER 87-029
RESIDUAL HEAT REMOVAL SYSTEM ISOLATION DUE TO INADVERTENT JUMPER GROUNDING.
EVENT DATE: 111987 REPORT DATE: 122187 NSSS: GE TYPE: BWR

(NSIC 207492) AT 0248 ON 11/19/87, WITH UNIT IN OPERATIONAL CONDITION 4 (COLD SHUTDOWN) AN ERROR OCCURRED DURING THE PERFORMANCE OF SURVEILLANCE TEST PROCEDURE (STP)-207-5251. DUE TO THE INADVERTENT GROUNDING OF A JUMPER USED TO PERFORM THE STP, FUSE B21H-F76B BLEW AND DE-ENERGIZED CONTROL LOGIC CAUSING RESIDUAL HEAT REMOVAL (RHR) SHUTDOWN COOLING SUCTION VALVE LE12-MOVF009 TO ISOLATE AND, IN TURN, TRIPPED RHR PUMP A (1E12-PC002A). THE FUSE WAS REPLACED VIA MAINTENANCE WORK ORDER R055867. THE VALVE WAS RESTORED TO ITS PROPER POSITION, AND THE PUMP WAS RESTARTED TO RESTORE SHUTDOWN COOLING. THE STP WAS RESTARTED AND SUCCESSFULLY COMPLETED. LONG JUMPERS WERE CONNECTED END-TO-END FOR USE IN PERFORMING STPS REQUIRING EXTRA-LONG JUMPERS. AS A RESULT, SHOP TRAINING ON THE PROPER USE OF TOOLS AND EQUIPMENT WAS CONDUCTED FOR INSTRUMENT AND CONTROLS (I&C) PERSONNEL. SHUTDOWN COOLING WAS INOPERATIVE FOR A SHORT PERIOD OF TIME (36 MINUTES) DURING THIS EVENT. ALTERNATE MEANS OF SHUTDOWN COOLING WERE AVAILABLE. BASED ON THE

ABOVE. THERE WAS NO IMPACT ON THE SAFE OPERATION OF THE PLANT OR TO THE HEALTH AND SAFETY OF THE PUBLIC AS A RESULT OF THIS EVENT.

[229] RIVERBEND 1 DOCKET 50-458 LER 87-031
STANDBY SERVICE WATER INITIATION ON LOW PRESSURE TRANSIENT.
EVENT DATE: 112887 REPORT DATE: 122887 NSSS: GE TYPE: BWR

(NSIC 207494) ON 11/28/87 AT 1000 WITH THE UNIT IN MODE 4 (COLD SHUTDOWN), A FULL INITIATION OF THE STANDBY SERVICE WATER (SSW) SYSTEM OCCURRED DURING THE PERFORMANCE OF SURVEILLANCE TEST PROCEDURE (STP)-200-0603, "DIVISION III REMOTE SHUTDOWN SYSTEM CONTROL CIRCUIT OPERABILITY TEST". A LOW SERVICE WATER PRESSURE SIGNAL IS INITIATED BY SECURING A SERVICE WATER PUMP. SWP-P2C RESTARTED UPON RETURNING CONTROL TO THE MAIN CONTROL ROOM DUE TO THE SEALED IN LOW PRESSURE START SIGNAL. THE PUMP WAS SECURED A SECOND TIME, AND THE SYSTEM WAS SUBSEQUENTLY RETURNED TO STANDBY LINE-UP. IT WAS DETERMINED THAT PROCEDURAL INADEQUACIES EXISTED ALLOWING THE OPERATOR TO TRANSFER SYSTEM CONTROL WITHOUT TAKING ADEQUATE PRECAUTIONS TO AVOID SYSTEM INITIATION. THE PROCEDURE HAS BEEN CORRECTED VIA TEMPORARY CHANGE NOTICE. MODIFICATION REQUEST (MR) 86-1542 HAS BEEN INITIATED TO IMPLEMENT A TIME DELAY FOR AUTOMATIC INITIATION OF SSW ON A LOW PRESSURE SIGNAL TO ELIMINATE INITIATIONS ON TRANSIENT LOW PRESSURE. SSW SYSTEM INITIATED AS DESIGNED ON DETECTION OF LOW PRESSURE. THERE WAS NO IMPACT TO THE HEALTH AND SAFETY OF THE PUBLIC AS A RESULT OF THIS EVENT.

[230] ROBINSON 2 DOCKET 50-261 LER 87-026
POTENTIAL FOR RESIDUAL HEAT REMOVAL PUMP FAILURE DUE TO AN INADEQUACY IN DESIGN.
EVENT DATE: 103087 REPORT DATE: 112987 NSSS: WE TYPE: PWR
VENDOR: INGERSOLL-RAND CO.

(NSIC 207318) ON OCTOBER 30, 1987, LICENSEE REVIEW OF A SITUATION AT A PLANT OF SIMILAR VINTAGE NUCLEAR STEAM SUPPLY SYSTEM (NSSS) DESIGN AS H. B. ROBINSON UNIT NO. 2 DETERMINED THAT THE POTENTIAL FOR DEGRADED RECIRCULATION FLOW FOR THE RESIDUAL HEAT REMOVAL (RHR) PUMPS (LOW HEAD SAFETY INJECTION (SI) PUMPS) MAY EXIST. THE NSSS DESIGNER HAD IDENTIFIED TWO CONCERNS RECENTLY INVOLVING THE POTENTIAL FOR A LOSS OF DISCHARGE FLOW FOR ONE PUMP WITH TWO PUMPS IN OPERATION, AND, THE POTENTIAL FOR INSUFFICIENT MINIFLOW CAPACITY FOR ONE PUMP OPERATION. THESE CONCERNS WERE IDENTIFIED AS ALSO POTENTIAL FOR THE HIGH HEAD SI PUMPS. THE LICENSEE'S PLANT NUCLEAR SAFETY COMMITTEE HELD SESSION AND DETERMINED THAT FURTHER EVALUATION WAS REQUIRED. IN ADDITION, PLANT EMERGENCY OPERATING PROCEDURES WERE IMMEDIATELY REVISED TO PRECLUDE PUMP DAMAGE DURING CONDITIONS WITH NO DISCHARGE FLOWPATH. THE MINIFLOW CAPACITY ISSUE WAS REFERRED TO THE NSSS DESIGNER FOR EVALUATION. THE LICENSEE NOTIFIED THE NUCLEAR REGULATORY COMMISSION AT 1625 HOURS, PURSUANT TO 10CFR50.72(B)(2)(II). ON NOVEMBER 17, 1987, I&E NOTICE 87-59 WAS ISSUED REGARDING THE TWO NSSS DESIGN CONCERNS. THE FINAL NSSS EVALUATION REPORT WILL BE DISCUSSED IN A SUPPLEMENTAL REPORT TO THIS LICENSEE EVENT REPORT.

[231] ROBINSON 2 DOCKET 50-261 LER 87-027
INOPERABILITY OF REDUNDANT EQUIPMENT DUE TO INADVERTENT LOSS OF MOTOR CONTROL CENTER MCC-6.
EVENT DATE: 110487 REPORT DATE: 120387 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 207257) ON NOVEMBER 4, 1987, AT 1133 HOURS, UNIT NO. 2 EXPERIENCED A MOMENTARY LOSS OF POWER TO MOTOR CONTROL CENTER MCC-6 AND CONSEQUENTLY INSTRUMENT BUSS 4. THIS EVENT PLACED THE OPERATION OF THE PLANT IN A CONDITION CONTRARY TO THE REQUIREMENTS OF TECHNICAL SPECIFICATION 3.7.2.D DUE TO THE LOSS OF MCC-6 AND ASSOCIATED SAFEGUARDS EQUIPMENT WHILE EMERGENCY DIESEL GENERATOR "A" WAS OUT OF SERVICE FOR MAINTENANCE. PLANT OPERATIONS PERSONNEL WERE REMOVING A PROTECTIVE

COVER [illegible]OM OVER THE "OPEN/CLOSE" PUSHBUTTON ON THE FRONT DOOR OF THE BREAKER CUB[illegible] THE MCC-6 BREAKER. THE COVER CONTACTED THE PUSHBUTTON, OPENING THE BRE[illegible]R, [illegible] CAUSED THE LOSS OF POWER. THE BREAKER WAS THEN CLOSED WITHIN APP[illegible]IMATE[illegible] TEN SECONDS. THE LOSS OF MCC-6 AND INSTRUMENT BUSS 4 CAUSED A TURF[illegible] "RUNBACK" AND THE PLANT WAS STABILIZED AT ABOUT 60 PERCENT POWER. THE UNIT [illegible]S RETURNED TO 100 PERCENT POWER AT 2040 HOURS. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(VII). THERE WERE NO FAILED COMPONENTS DURING THE EVENT. THE CAUSE OF THE EVENT IS ATTRIBUTED TO THE DESIGN OF THE PUSHBUTTON PROTECTIVE COVER AND INADEQUATE CARE BEING EXERCISED IN ITS REMOVAL.

[232] ROBINSON 2 DOCKET 50-261 LER 87-028
DIESEL GENERATOR "B" AIR START FAILURE WHILE DIESEL GENERATOR "A" INOPERABLE.
EVENT DATE: 110587 REPORT DATE: 120487 NSSS: WE TYPE: PWR
VENDOR: FAIRBANKS MORSE

(NSIC 207292) ON NOVEMBER 5, 1987, WITH UNIT NO. 2 AT 100 PERCENT POWER AND DIESEL GENERATOR "A" OUT OF SERVICE FOR MAINTENANCE, DIESEL GENERATOR "B" FAILED TO START AT 0250 HOURS. OPERABILITY TESTING WAS UNDERWAY AND REQUIRED A LOCAL START OF THE DIESEL WITH ONE OF TWO PARALLEL AIR START SOLENOID VALVES ISOLATED. AT THE TIME, THERE WAS INDICATION THE AIR WAS VENTING TO ATMOSPHERE AND THE START SEQUENCE WAS STOPPED. A SECOND START ATTEMPT AT 0256 HOURS WAS SUCCESSFUL AS WAS A THIRD SOON AFTER. THE PLANT NUCLEAR SAFETY COMMITTEE DETERMINED DIESEL GENERATOR "B" WAS INOPERABLE FOR THE SIX MINUTES AND THE LICENSEE NOTIFIED THE NRC OF A FOUR-HOUR NONEMERGENCY EVENT PURSUANT TO 10CFR50.72 (B)(2)(III) AT 1847 HOURS. PRELIMINARY INSPECTION OF THE AIR START SYSTEM COMPONENTS COULD NOT DETERMINE ROOT CAUSE. THE FAILURE, HOWEVER, HAS NOT RECURRED AND THE DIESEL HAS STARTED SUCCESSFULLY DURING EACH WEEKLY OPERABILITY TEST SINCE THE EVENT. THE LICENSEE PLANS TO REPLACE THE SOLENOID VALVE INVOLVED IN THE TEST ALIGNMENT AND EXAMINE IT THOROUGHLY WITH A SUPPLEMENTAL REPORT TO FOLLOW. THIS REPORT IS SUBMITTED PURSUANT TO 10CFR50.73(A)(2)(V).

[233] SALEM 1 DOCKET 50-272 LER 87-014
LOSS OF CONTROL OF A LOCKED HIGH RADIATION AREA DOOR DUE TO PERSONNEL ERROR.
EVENT DATE: 100887 REPORT DATE: 110987 NSSS: WE TYPE: PWR

(NSIC 207128) ON OCTOBER 8, 1987 AT 1030 HOURS A RADIATION PROTECTION TECHNICIAN FOUND A PLASTIC SHOE COVER WEDGED TO BLOCK OPEN A LOCKED HIGH RADIATION AREA DOOR INTO THE UNIT 1 BIOSHIELD IN THE AREA OF NO. 14 REACTOR COOLANT PUMP (AB). UPON DISCOVERY, THE TECHNICIAN RESTORED CONTROL OF THE DOOR AND REPORTED THE EVENT. AT 1225 HOURS THE SAME DAY, THE SAME HRA DOOR WAS AGAIN FOUND WEDGED OPEN VIA A PLASTIC SHOE COVER. THE SHOE COVER WAS AGAIN REMOVED AND THE EVENT WAS REPORTED. THE ROOT CAUSE OF THESE OCCURRENCES HAS BEEN DETERMINED TO BE PERSONNEL ERROR. INVESTIGATIONS HAVE IDENTIFIED A NUMBER OF INDIVIDUALS WHO HAD ACCESS TO THIS AREA DURING THE TIME PERIOD WHEN THE HRA DOOR BLOCKAGES OCCURRED ALTHOUGH IT HAS NOT BEEN DETERMINED WHO WAS DIRECTLY INVOLVED. MEMO'S FROM THE GM-SO HAVE BEEN ISSUED INFORMING STATION PERSONNEL OF THIS EVENT AND REQUESTING ASSISTANCE IN DETERMINING ALL RELEVANT FACTS ASSOCIATED WITH THE HRA DOOR BLOCKAGE. TO DATE, INDIVIDUALS HAVE COME FORWARD TO DISCUSS THE ISSUE, ALTHOUGH, NO ONE HAS IDENTIFIED THEMSELVES OR OTHERS AS BEING RESPONSIBLE FOR THE HRA DOOR BLOCKAGE. THESE DISCUSSIONS HAVE SHOWN THAT NOT ALL RADIATION WORKERS ARE AS AWARE OF HRA KEY CONTROL PROCEDURES AS THEY SHOULD BE.

[234] SALEM 1 DOCKET 50-272 LER [illegible]15
NON-COMPLIANCE DUE TO PERSONNEL ERROR.
EVENT DATE: 102387 REPORT DATE: 112087 NSSS: WE TYPE: PWR

(NSIC 207177) ON OCTOBER 23, 1987 AT 0227 HOURS, FUEL WAS MOVED FROM THE CORE TO THE SPENT FUEL POOL, IN SUPPORT OF REFUELING ACTIVITIES, WITH TWO OF THREE DIESEL

GENERATORS (D/G) INOPERABLE. THIS IS CONTRARY TO THE REQUIREMENTS OF TECH SPEC 3.8.1.2.B. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO PERSONNEL ERROR. THE EO FAILED TO COMPLY WITH THE SPECIFIC ORDERS OF A TAGGING REQUEST RESULTING IN THE RACKING DOWN OF THE UNIT 1 1A D/G INSTEAD OF UNIT 2 2A D/G, THE UNIT 1 DESK NCO FAILED TO RECOGNIZE THAT A RETEST OF 1A D/G WAS REQUIRED TO ENSURE ITS OPERABILITY (AS PER PROCEDURE), AND A GENERAL COMMUNICATION FAILURE BY THE UNIT 1 DESK NCO TO DISCUSS THE "WRONG BREAKER RACK OUT" EVENT WITH HIS SUPERVISOR OR THE ON-COMING SHIFT. IF THE COMMUNICATION HAD TAKEN PLACE, THE 1A D/G INOPERABILITY WOULD HAVE BEEN RECOGNIZED AND CORRECTED PRIOR TO MAKING 1B D/G INOPERABLE AND FUEL MOVED. CORRECTIVE ACTION INCLUDES APPROPRIATE CORRECTIVE DISCIPLINARY ACTION TO THE UNIT 1 DESK NCO, THE UNIT 2 SHIFT SUPERVISOR, AND THE EO INVOLVED IN THIS EVENT, A REVIEW BY THE PSE&G NUCLEAR TRAINING CENTER (NTC) FOR INCLUSION IN APPLICABLE TRAINING PROGRAMS, DISCUSSION OF THIS EVENT IN THE OPERATIONS DEPARTMENT NEWSLETTER, A HUMAN FACTORS EVALUATION OF THIS EVENT, AND PREPARATION OF A LETTER TO ALL STATION PERSONNEL ADDRESSING THIS EVENT.

[235] SALEM 1 DOCKET 50-272 LER 87-016
POWER OPERATED RELIEF STOP VALVE CABLING FOUND DEGRADED.
EVENT DATE: 110287 REPORT DATE: 120787 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 2 (PWR)

(NSIC 207345) ON NOVEMBER 02, 1987, POWER LEAD CABLING FOR PRESSURIZER POWER OPERATED RELIEF STOP VALVE, 1PR6, (AB) WAS FOUND IN A DEGRADED CONDITION. THIS DISCOVERY WAS MADE DURING MOVAT TESTING WHEN THE WIRING INSULATION INSIDE THE VALVE MOTOR OPERATOR WAS FOUND DETERIORATED. SUBSEQUENT INVESTIGATION REVEALED SIMILAR WIRE DEGRADATION FOR SALEM UNIT 2 2PR6 AND 2PR7 VALVES POWER LEAD AND CONTROL CABLING. THE ROOT CAUSE OF THIS EVENT WAS INADEQUATE DESIGN REVIEW. THE CABLING (EPR INSULATED) FOR THESE VALVES IS ENVIRONMENTALLY QUALIFIED TO PROVIDE A LIFE EXPECTANCY OF 40 YEARS AT AN AVERAGE AMBIENT TEMPERATURE OF 120F, HOWEVER, THE AMBIENT TEMPERATURE WITHIN THE PRESSURIZER ENCLOSURE CAN BE AS HIGH AS 160F. AT 160F AMBIENT SERVICE TEMPERATURE, THE LIFE EXPECTANCY OF THE CABLING DECREASES TO APPROXIMATELY 6.7 YEARS BASED ON ARRHENUS CALCULATIONS. CORRECTIVE ACTION INCLUDED A PRESSURIZER ENCLOSURE INSPECTION (BOTH UNITS), REPLACEMENT OF DEGRADED CABLE, UPGRADING THE EQ PREVENTIVE MAINTENANCE REQUIREMENTS FOR LIMIT SWITCHES ON VALVES IN THE PRESSURIZER ENCLOSURE, ENGINEERING REVIEW TO ADDRESS APPROPRIATE INSPECTION AND/OR REPLACEMENT FREQUENCY CRITERIA FOR PRESSURIZER ENCLOSURE CABLING AND COMPONENTS, AND ENGINEERING INVESTIGATION TO EVALUATE OTHER PLANT AREAS WHERE THE AMBIENT ENVIRONMENTAL CONDITIONS PRECLUDE THE CURRENT ENVIRONMENTAL QUALIFICATION.

[236] SALEM 2 DOCKET 50-311 LER 87-012
RHR PUMP ROOM FLOOD CURB MISSING DUE TO PERSONNEL ERROR.
EVENT DATE: 093087 REPORT DATE: 103087 NSSS: WE TYPE: PWR

(NSIC 207131) ON SEPTEMBER 30, 1987 AT 1400 HOURS, ENGINEERS, PERFORMING A SALEM UNIT 2 WALKDOWN, DISCOVERED A SIX INCH CURB WAS NOT INSTALLED AT THE ENTRANCE OF THE AUXILIARY BUILDING SUMP TANK ROOM (ELEVATION 55'), AS PER DESIGN. THE MISSING CURB IS DESIGNED TO PREVENT FLOODING OF BOTH RHR PUMP ROOMS (BP) IN THE EVENT OF A MODERATE ENERGY LINE (MEL) BREAK. THE ROOT CAUSE COULD NOT BE POSITIVELY IDENTIFIED ALTHOUGH IT HAS BEEN ATTRIBUTED TO PERSONNEL ERROR. THE CURBE WAS RE-INSTALLED AND LABELED.

[237] SALEM 2 DOCKET 50-311 LER 87-013 REV 01
UPDATE ON TECHNICAL SPECIFICATION SURVEILLANCE 4.8.1.1.3.A MISSED DUE TO INADEQUATE PROCEDURAL CONTROL.
EVENT DATE: 100287 REPORT DATE: 120987 NSSS: WE TYPE: PWR

(NSIC 207403) ON OCTOBER 2, 1987 IT WAS DISCOVERED THAT THE PRIOR MONTH'S

REQUIRED TECH SPEC SURVEILLANCE 4.8.1.1.3.A.2 WAS PERFORMED FOR NO. 21 DIESEL GENERATOR (D/G) FUEL OIL TRANSFER PUMP (DC) BUT NOT FOR NO. 22 D/G FUEL OIL TRANSFER PUMP. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO INADEQUATE PROCEDURAL CONTROL OF THE MANAGED MAINTENANCE INFORMATION SYSTEM (MMIS) COMPUTERIZED DATA BASE. THE SURVEILLANCE TEST WORK ORDER (STWO) WAS MODIFIED PRIOR TO ITS ISSUE RESULTING IN THE COMPLETION OF THE SURVEILLANCE TEST FOR NO. 21 D/G FUEL OIL TRANSFER PUMP BUT NOT NO. 22 D/G FUEL OIL TRANSFER PUMP. A REVIEW OF OTHER SIMILAR HISTORICALLY PERFORMED SURVEILLANCES WAS CONDUCTED. NO OTHER MISSED OR INCOMPLETE SURVEILLANCES WERE FOUND. THEREFORE THIS WAS AN ISOLATED EVENT. THE SURVEILLANCE TEST WORK ORDER FORM WAS SUBSEQUENTLY MODIFIED TO PRECLUDE RECURRENCE. AN ADMINISTRATIVE PROCEDURE WILL BE DEVELOPED BY THE PLANNING DEPARTMENT TO ENSURE ADMINISTRATIVE CONTROLS FOR THE STWO PROGRAM, REQUIRED BY ADMINISTRATIVE PROCEDURE (AP) - 9, "MAINTENANCE PROGRAM", ARE COMPLIED WITH. THIS WILL ENSURE ANOTHER ADMINISTRATIVE CONTROL CONCERN SIMILAR TO THIS EVENT DOES NOT OCCUR.

[238] SALEM 2 DOCKET 50-311 LER 87-014
INCORRECT DIESEL GENERATOR INFEED BREAKER SETPOINT DUE TO INADEQUATE DOCUMENTATION CONTROL.
EVENT DATE: 102287 REPORT DATE: 111987 NSSS: WE TYPE: PWR

(NSIC 207196) ON OCTOBER 22, 1987 AT 1315 HOURS, IT WAS IDENTIFIED THAT THE SETPOINT ON THE DIESEL GENERATOR (D/G) (EK) OVERCURRENT PROTECTION RELAYS FOR SALEM UNIT 2 WERE INCORRECT. THIS RESULTED IN A LACK OF BREAKER COORDINATION BETWEEN THE SW PUMP (BI) OVERCURRENT PROTECTION RELAYS AND THE UPSTREAM D/G OVERCURRENT PROTECTION RELAYS. PSE&G ENGINEERS IDENTIFIED THESE INCORRECT SETTINGS WHILE PERFORMING A REVIEW OF BREAKER COORDINATION. ANALYSES OF THE EFFECTS OF THE INCORRECT BREAKER SETTINGS CONCLUDED THAT THE POSSIBILITY EXISTED TO ISOLATE TWO OUT OF THREE D/G'S FROM THEIR RESPECTIVE VITAL BUSSES DURING A LOSS OF COOLANT ACCIDENT (LOCA) AND LOSS OF OFFSITE POWER (BLACKOUT) DUE TO A SW BAY PIPE RUPTURE. ONE D/G IS NOT ADEQUATE TO BRING THE PLANT TO SAFE SHUTDOWN. A "FOUR HOUR" REPORT WAS MADE TO THE NRC ON OCTOBER 22, 1987 AT 1358 HOURS. ALSO, TECH SPEC 3.0.3 WAS ENTERED AT 1315 HOURS ON OCTOBER 22, 1987. IT WAS EXITED THE SAME DAY AT 1737 HOURS UPON D/G INFEED BREAKER OVERCURRENT PROTECTION SETPOINT CORRECTION. THE ROOT CAUSE OF THE INCORRECT RELAY SETTINGS IS INADEQUATE DOCUMENTATION CONTROL. THE D/G RELAY SETPOINTS HAVE BEEN CORRECTED SO THAT BREAKER COORDINATION NOW EXISTS BETWEEN THE SW PUMP AND THE D/G RELAYS. AN ELECTRICAL TASK FORCE HAS BEEN ESTABLISHED TO ADDRESS BREAKER COORDINATION CONCERNS AT SALEM GENERATING STATION.

[239] SAN ONOFRE 1 DOCKET 50-206 LER 87-005 REV 01
UPDATE ON NUCLEAR INSTRUMENTATION POWER RANGE CHANNEL DECLARED INOPERABLE DUE TO LOW OUTPUT.
EVENT DATE: 042887 REPORT DATE: 111987 NSSS: WE TYPE: PWR

(NSIC 207161) ON 3/24/87, NUCLEAR INSTRUMENTATION SYSTEM (NIS) POWER RANGE CHANNEL 1207 WAS DECLARED INOPERABLE DUE TO LOW OUTPUT. IN ACCORDANCE WITH THE ACTION REQUIREMENTS OF LCO 3.5.1, THE CHANNEL WAS PLACED IN THE TRIPPED CONDITION AT THE HIGH-RANGE OVERPOWER TRIP SETPOINT (109%). ON 4/28/87, AT 2055, WHILE REDUCING POWER FOR CLEANING OF CONDENSER WATER BOXES, NIS CHANNEL 1207 AUTOMATICALLY RESET FROM THE TRIPPED STATE TO THE INOPERABLE CONDITION WHEN POWER REACHED 70% AND THE MODE SELECTOR SWITCH WAS REPOSITIONED. THE CHANNEL SIMILARLY RESET TO THE INOPERABLE CONDITION FROM THE TRIPPED STATE ON 4/29/87, AT 0424, AS THE UNIT WAS BEING RETURNED TO 92% POWER. ALTHOUGH LCO 3.5.1 PERMITS UP TO 8 HOURS TO INITIALLY TRIP AN INOPERABLE CHANNEL, AND UP TO 2 HOURS IN THE UN-TRIPPED CONDITION FOR THE PURPOSE OF SURVEILLANCE TESTING THEREAFTER, IT DOES NOT EXPLICITLY PERMIT THE CHANNEL T[illegible]E IN AN INOPERABLE CONDITION FOR ANY OTHER REASON. ACCORDINGLY, THE CONDIT[illegible] LTED IN AN ENTRY INTO LCO 3.0.3. THE NIS OPERATIONS PROCEDURE DID NOT PROVID[illegible] FOR MAINTAINING THE CHANNEL IN A TRIPPED

CONDITION. AS CORRECTIVE ACTION, THE PROCEDURE WAS REVISED TO ENSURE THAT THE AFFECTED CHANNEL OUTPUT IS INCREASED PRIOR TO THE MODE SELECTOR SWITCH BEING REPOSITIONED.

[240] SAN ONOFRE 1 DOCKET 50-206 LER 87-014
BATTERY WEEKLY SURVEILLANCE NOT PERFORMED PRIOR TO MODE CHANGE DUE TO PERSONNEL ERROR.
EVENT DATE: 062687 REPORT DATE: 112587 NSSS: WE TYPE: PWR

(NSIC 207243) ON 10/29/87, A REVIEW OF TECHNICAL SPECIFICATION SURVEILLANCE RESULTS REVEALED THAT AT 817 ON 06/26/87, AN OPERATIONAL MODE CHANGE OCCURRED WITHOUT BATTERY NO. 1 WEEKLY SURVEILLANCE HAVING BEEN PERFORMED. THIS IS CONTRARY TO TECH SPEC 4.0.4. THERE WAS NO SAFETY SIGNIFICANCE TO THIS EVENT SINCE ALTERNATE SURVEILLANCES PERFORMED ON THE BATTERY BETWEEN 06/24/87 AND 06/27/87 DETERMINED THE BATTERY REMAINED CAPABLE OF PERFORMING ITS INTENDED SAFETY FUNCTION THROUGHOUT THE EVENT. ADDITIONALLY, THE BATTERY WEEKLY SURVEILLANCE CONDUCTED ON 07/01/87 WAS SATISFACTORILY COMPLETED. THIS EVENT OCCURRED DUE TO THE FAILURE OF MAINTENANCE PERSONNEL TO NOTIFY THE SHIFT SUPERINTENDENT (SS) REGARDING THE STATUS OF THE REQUIRED SURVEILLANCE. MAINTENANCE PERSONNEL DOCUMENTED ON THE MAINTENANCE ORDER THE FACT THAT THE SURVEILLANCE COULD NOT BE PERFORMED AS SCHEDULED. THE MAINTENANCE PROCEDURE REQUIRES IN SUCH CASES THAT THE SS BE NOTIFIED IN WRITING BY THE COGNIZANT MAINTENANCE SUPERVISOR SO THAT OPERATIONS PERSONNEL CAN ASSESS THE IMPACT OF THE INCOMPLETE SURVEILLANCE ON, AMONG OTHER THINGS, OPERABILITY AND MODE CHANGES. SINCE THE NOTIFICATION WAS NOT MADE, OPERATIONS WAS THEREFORE UNAWARE OF THE INCOMPLETE SURVEILLANCE AND ADMINISTRATIVE CONTROLS PREVENTING THE MODE CHANGE WERE NOT INITIATED.

[241] SAN ONOFRE 1 DOCKET 50-206 LER 87-016
ASCO SOLENOID VALVE FAILURES.
EVENT DATE: 111087 REPORT DATE: 121087 NSSS: WE TYPE: PWR
VENDOR: AUTOMATIC SWITCH COMPANY (ASCO)

(NSIC 207357) ON FIVE OCCASIONS FROM MARCH TO SEPTEMBER 1987, DURING ROUTINE TESTING, FOUR DIFFERENT AIR OPERATED VALVES FAILED TO STROKE. ONE VALVE IS A CONTAINMENT ISOLATION VALVE ON THE CONTAINMENT VENT LINE, ONE VALVE IS A CONTAINMENT SPRAY SYSTEM VALVE, AND THE OTHER TWO VALVES ARE LETDOWN ORIFICE ISOLATION VALVES. FOR THE CONTAINMENT SPRAY VALVE AND THE CONTAINMENT ISOLATION VALVE, APPROPRIATE ACTIONS WERE TAKEN IN ACCORDANCE WITH TECHNICAL SPECIFICATION. FOR THE LETDOWN SYSTEM VALVES, 10 CFR 50.59 EVALUATIONS WERE PERFORMED, ON APPROPRIATE ACTIONS WERE TAKEN BASED ON THESE EVALUATIONS. SUBSEQUENT EVALUATIONS ATTRIBUTED EACH OF THESE UNSUCCESSFUL STROKING OPERATIONS TO A FAILURE OF THEIR ASSOCIATED AUTOMATIC SWITCH COMPANY (ASCO) THREE-WAY SOLENOID AIR VALVE (MODEL NO. 206-380-2 OR 206-380-3). INSPECTION OF THREE OF THE FOUR SOLENOID VALVES REVEALED THAT THERE WAS A THIN HARD FILM THAT HAD FORMED BETWEEN THE TOP OF THE SLUG AND SLUG HOUSING. ON NOVEMBER 10, 1987, AN ENGINEERING EVALUATION WAS COMPLETED WHICH INCLUDED THAT THE PRESENCE OF THE FILM WAS THE CAUSE FOR THESE SOLENOID VALVE FAILURES. BASED ON THIS EVALUATION, IT WAS DETERMINED THAT THE FILM CONSTITUTES A SINGLE CONDITION WHICH HAD RENDERED INDEPENDENT TRAINS IN MULTIPLE SYSTEMS INOPERABLE.

[242] SAN ONOFRE 2 DOCKET 50-361 LER 86-027 REV 01
UPDATE ON REACTOR TRIP CAUSED BY FAILED CEA POSITION INDICATION.
EVENT DATE: 091386 REPORT DATE: 121887 NSSS: CE TYPE: PWR
VENDOR: COMBUSTION ENGINEERING, INC.

(NSIC 207410) ON SEPTEMBER 13, 1986, AT 0952, WITH UNIT 2 AT 60% POWER, A REACTOR TRIP WAS GENERATED BY ALL FOUR CORE PROTECTION CALCULATOR (CPC) CHANNELS WHEN A

DEFECTIVE CONTROL ELEMENT ASSEMBLY (CEA) POSITION SENSOR CAUSED GENERATION OF PENALTY FACTORS AFFECTING THE DEPARTURE FROM NUCLEATE BOILING RATIO (DNBR) TRIP SETPOINT. THE TRIP RECOVERY PROCEEDED NORMALLY, AND THE HEALTH AND SAFETY OF THE PUBLIC AND PLANT PERSONNEL WAS NOT AFFECTED BY THIS EVENT. DURING MOVEMENT OF PART-LENGTH GROUP 1, AN ERRATIC POSITION INDICATION SIGNAL FROM CEA 34 TO CONTROL ELEMENT ASSEMBLY CALCULATOR (CEAC) 1, GAVE A FALSE OUTWARD POSITION DEVIATION, AND WITH THE CEA'S BELOW 139.25 INCHES, CAUSED THE CEAC TO APPLY PENALTY FACTORS TO THE CPC DNBR AND LOCAL POWER DENSITY (LPD) CALCULATIONS. THE PENALTY DOES NOT APPLY FOR CEA'S ABOVE 139.25 INCHES, OR FOR INWARD POSITION DEVIATIONS. THE REED SWITCH POSITION TRANSMITTER (RSPT) FOR CEA 34 WAS FOUND TO BE DEFECTIVE AND AS REPLACED. THE MOST PROBABLE CAUSE OF THE ERRATIC SIGNAL WAS A LOOSE SOLDER JOINT. THIS IS CONSIDERED AN ISOLATED EVENT, THEREFORE, NO SPECIFIC CORRECTIVE ACTION IS CURRENTLY PLANNED FOR THE RSPT'S. SCE, IN CONCERT WITH COMBUSTION ENGINEERING, IS DEVELOPING A MEANS TO PREVENT SINGLE TRAIN CEAC OUTPUT TO THE CPC'S FROM CAUSING REACTOR TRIPS.

[243] SAN ONOFRE 2 DOCKET 50-361 LER 87-020
HEALTH PHYSICS ASPECTS OF SDCS ISOLATION VALVE 2HV-9378 EVENT.
EVENT DATE: 083187 REPORT DATE: 111187 NSSS: CE TYPE: PWR

(NSIC 207100) UNIT 2 LER 87-014 REPORTED AN UNISOLABLE REACTOR COOLANT SYSTEM (RCS) LEAK THROUGH A SHUTDOWN COOLING SYSTEM ISOLATION VALVE'S PACKING GLAND. THIS INFORMATIONAL LER IS BEING SUBMITTED TO PROVIDE INFORMATION ON THE RADIOLOGICAL CONDITIONS AND CONTROLS ASSOCIATED WITH THE EVENT, AND PRESENT THE CORRECTIVE ACTIONS INITIATED AS A RESULT OF "LESSONS LEARNED". SPECIFICALLY, SCE HAS CONCLUDED THAT DEFICIENCIES EXISTED: (1) IN THE COMMUNICATION OF INFORMATION AND INSTRUCTIONS BETWEEN OPERATIONS AND HEALTH PHYSICS PERSONNEL DURING AN EXIGENT SITUATION; (2) WITH THE CLARITY OF HP PROCEDURES; AND, (3) WITH THE UNDERSTANDING BY HEALTH PHYSICS PERSONNEL OF THEIR RESPONSIBILITIES DURING EXIGENT SITUATIONS. SCE IS REVISING THE RADIOLOGICAL EXPOSURE PERMIT PROGRAM AND PROCEDURES. ADDITIONAL TRAINING OF OPERATIONS AND HEALTH PHYSICS PERSONNEL IS BEING DEVELOPED. INTERIM INSTRUCTIONS AND GUIDANCE HAVE BEEN ISSUED.

[244] SAN ONOFRE 2 DOCKET 50-361 LER 87-021
SPURIOUS FUEL HANDLING ISOLATION SYSTEM TRAIN "B" ACTUATION DURING DESIGN CHANGE WORK.
EVENT DATE: 102687 REPORT DATE: 112587 NSSS: CE TYPE: PWR

(NSIC 207269) AT 1014 ON 10/26/87, A SPURIOUS ACTUATION OF TRAIN "B" OF THE FUEL HANDLING ISOLATION SYSTEM (FHIS) OCCURRED. THERE WAS NO INDICATION OF INCREASED RADIATION LEVELS IN THE FUEL HANDLING BUILDING (FHB). AFTER THE FHB AIRBORNE ACTIVITY LEVELS WERE CONFIRMED TO BE NORMAL, THE FHIS WAS RESET AND FHB VENTILATION RETURNED TO NORMAL AT 1345. ALL FHIS TRAIN "B" COMPONENTS FUNCTIONED AS DESIGNED. THE SPURIOUS ACTUATION OCCURRED DURING INSTALLATION OF A DESIGN CHANGE IN THE CONTAINMENT PURGE ISOLATION SYSTEM (CPIS) TRAIN "B" MONITOR, WHICH SHARES A CABINET WITH THE FHIS TRAIN "B" MONITOR. AS PREVIOUSLY REPORTED IN LER 87-010 (DOCKET NO. 50-362), THE CPIS AND FHIS WIRING IS ROUTED IN COMMON WIRE BUNDLES. DEENERGIZATION OF A CPIS ALARM RELAY, NECESSARY FOR THE DESIGN CHANGE, IS BELIEVED TO HAVE INDUCED AN ACTUATION SIGNAL IN THE FHIS CIRCUITRY. THE CABINET HOUSING THE CPIS AND FHIS WILL BE MODIFIED TO SEPARATE THE WIRING CURRENTLY ROUTED IN COMMON WIRE BUNDLES. THERE WAS NO SAFETY SIGNIFICANCE TO THIS EVENT AS ALL FHIS TRAIN "B" COMPONENTS OPERATED IN ACCORDANCE WITH DESIGN.

[245] SAN ONOFRE 2 DOCKET 50-361 LER 87-022
FUEL HANDLING ISOLATION SYSTEM (FHIS) TRAIN 'A' AND 'B' SPURIOUS ACTUATION.
EVENT DATE: 102787 REPORT DATE: 112587 NSSS: CE TYPE: PWR
VENDOR: DRESSER INDUSTRIES, INC.

(NSIC 207270) AT 1050 ON OCTOBER 27, 1987, WITH UNIT 2 IN MODE 5, FUEL HANDLING ISOLATION SYSTEM (FHIS) MONITORS 2RT-7822 AND 2RT-7823, TRAIN 'A' AND 'B' RESPECTIVELY, WERE SPURIOUSLY ACTUATED FROM AN APPARENT NOISE SPIKE. AT APPROXIMATELY THE SAME TIME, THE COMPONENT COOLING WATER MONITOR 2RT-7819, WHICH HAS NO CONTROL FUNCTIONS, AND CONTAINMENT AIRBORNE MONITOR 2RT-7804, WHICH HAD BEEN REMOVED FROM SERVICE, ALSO ALARMED. AFTER VERIFYING FUEL HANDLING BUILDING RADIATION LEVELS WERE BELOW THE ACTUATION SETPOINT, THE FHIS WAS RESET/SECURED. EVIDENT ON EACH MONITOR'S RECORDER WAS A LARGE INSTANTANEOUS RISE AND DROP OF RECORDED RADIATION LEVELS, INDICATIVE OF A NOISE SPIKE. INVESTIGATION INTO THE CAUSE OF THE ELECTRICAL NOISE, HOWEVER, HAS YET TO DETERMINE THE SOURCE. DURING VERIFICATION THAT ALL FHIS COMPONENTS FUNCTIONED AS DESIGNED, IT WAS DISCOVERED THAT THE SAMPLE PUMPS FOR BOTH TRAINS HAD TRIPPED OFF. INVESTIGATION HAS NOT REVEALED ANY ACTIVITY THAT WOULD HAVE CAUSED THE PUMPS TO TRIP AT SOME TIME BETWEEN THE TIME OF THEIR LAST SURVEILLANCE AND THE FHIS ACTUATION NOR A REASON FOR A TRIP DURING THE ACTUATION. THESE INVESTIGATIONS ARE CONTINUING AND A SUPPLEMENTAL LICENSEE EVENT REPORT WILL BE SUBMITTED UPON THEIR COMPLETION. THERE WAS NO SAFETY SIGNIFICANCE TO THIS EVENT SINCE ALL FHIS COMPONENTS OPERATED IN ACCORDANCE WITH DESIGN.

[246] SAN ONOFRE 2 DOCKET 50-361 LER 87-023
SPURIOUS TRAIN "B" TGIS CHLORINE CHANNEL ACTUATION.
EVENT DATE: 110787 REPORT DATE: 120487 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: SAN ONOFRE 3 (PWR)

(NSIC 207370) ON NOVEMBER 7, 1987, AT 1055, WITH UNIT 2 IN MODE 5 AND UNIT 3 AT 100% POWER IN MODE , TRAIN "B" TGIS INITIATED OPERATION OF BOTH TRAINS OF CREACUS ON RECEIPT OF A HIGH CHLORINE GAS SIGNAL. CREACUS OPERATED IN THE ISOLATION MODE AS DESIGNED, UNTIL IT AS DETERMINED THAT THE SIGNAL WAS SPURIOUS AND THAT NO CHLORINE GAS WAS PRESENT. FIRE DETECTION AND ACTUATION SUPERVISORY SYSTEM TROUBLE ALARMS (WHICH ARE FED FROM THE SAME ELECTRICAL SOURCE) WERE RECEIVED SIMULTANEOUSLY WITH THE TGIS AND CREACUS ALARMS. THE ROOT CAUSE INVESTIGATION HAS NOT YET DETERMINED THE CAUSE OF THE ACTUATION AND IS CONTINUING. THE INVESTIGATION RESULTS WILL BE REPORTED IN A SUPPLEMENT TO THIS LER. THERE IS NO SAFETY SIGNIFICANCE TO THIS EVENT SINCE ALL TGIS AND CREACUS COMPONENTS OPERATED AS DESIGNED.

[247] SAN ONOFRE 2 DOCKET 50-361 LER 87-024
FUEL HANDLING AND CONTAINMENT PURGE ISOLATION SPURIOUS ACTUATION DURING VITAL BUS TRANSFER.
EVENT DATE: 110987 REPORT DATE: 120987 NSSS: CE TYPE: PWR

(NSIC 207371) AT 1813 ON 11/9/87, A SPURIOUS ACTUATION OF TRAIN "A" OF BOTH THE FUEL HANDLING ISOLATION SYSTEM (FHIS) AND THE CONTAINMENT PURGE ISOLATION SYSTEM (CPIS) OCCURRED. OPERATORS VERIFIED THAT RADIATION LEVELS WERE NORMAL AND RESET FHIS AND CPIS AT 1817. ALL TRAIN "A" FHIS AND CPIS COMPONENTS FUNCTIONED AS DESIGNED. AT THE TIME OF THE ACTUATIONS, THE TRAIN "A" 1E 120 VAC BUS, WHICH PROVIDES POWER TO TRAIN "A" OF BOTH CPIS AND FHIS, WAS BEING TRANSFERRED FROM ITS NORMAL POWER SOURCE TO ITS ALTERNATE POWER SOURCE; HOWEVER, THIS EVOLUTION COULD NOT BE DEMONSTRATED TO HAVE CAUSED THE FHIS AND CPIS ACTUATIONS. THE INVESTIGATION TO DETERMINE THE ROOT CAUSE OF THIS EVENT IS CONTINUING. THE INVESTIGATION RESULTS WILL BE REPORTED IN A SUPPLEMENT TO THIS LER. THERE IS NO SAFETY SIGNIFICANCE TO THIS EVENT SINCE ALL TRAIN "A" FHIS AND CPIS COMPONENTS FUNCTIONED AS DESIGNED.

[248] SAN ONOFRE 2 DOCKET 50-361 LER 87-025
EIGHTEEN MONTH SURVEILLANCE OF SNUBBERS NOT PERFORMED.
EVENT DATE: 112387 REPORT DATE: 122287 NSSS: CE TYPE: PWR
VENDOR: PACIFIC SCIENTIFIC COMPANY

(NSIC 207452) ON NOVEMBER 23, 1987, WITH UNIT 2 IN MODE 5, IT WAS DETERMINED THAT SEVEN MECHANICAL SNUBBERS ON SAFETY RELATED PIPING WERE NOT INCLUDED IN THE SNUBBER SURVEILLANCE PROGRAM, AND THEREFORE WERE INOPERABLE BECAUSE THEY HAD NOT BEEN SUBJECTED TO THE TESTING REQUIRED TECHNICAL SPECIFICATION 4.7.6. ON DECEMBER 21, 1987, WITH UNIT 2 IN MODE 1, AN ADDITIONAL UNTESTED SNUBBER WAS IDENTIFIED. NOVEMBER 25, 1987, THE FIRST SEVEN SNUBBERS WERE VISUALLY INSPECTED AND FUNCTIONALLY LISTED SATISFACTORILY AND WERE RETURNED TO OPERABLE STATUS. ON DECEMBER 22, 1987, THE EIGHTH SNUBBER WAS VISUALLY INSPECTED AND FUNCTIONALLY TESTED SATISFACTORILY AND WAS TURNED TO OPERABLE STATUS. BECAUSE THE SNUBBERS WERE FOUND TO HAVE BEEN CAPABLE OF INFORMING THEIR DESIGN FUNCTION, THE OPERABILITY OF THE ASSOCIATED PIPING WAS NOT AFFECTED BY THIS EVENT. THE CAUSE OF THIS EVENT WAS AN ADMINISTRATIVE ERROR WHICH PREVENTED THESE SNUBBERS FROM BEING DOCUMENTED COMPLETELY DURING INSTALLATION. ALL UNIT 2 SNUBBER RECORDS HAVE BEEN VIEWED AND NO ADDITIONAL DOCUMENTATION DISCREPANCIES HAVE BEEN IDENTIFIED. THE HEALTH AND SAFETY OF PLANT PERSONNEL, AND THE PUBLIC, WAS NOT AFFECTED BY THIS EVENT.

[249] SAN ONOFRE 3 DOCKET 50-362 LER 87-017
TRIP ON LOW CONDENSER VACUUM DURING INFLUX OF SEAWEED.
EVENT DATE: 101187 REPORT DATE: 111087 NSSS: CE TYPE: PWR

(NSIC 207101) ON 10/11/87 AT 1010, WITH REACTOR POWER AT 77%, FLOW DIRECTION THROUGH THE CIRCULATING WATER INTAKE AND OUTFALL TUNNELS WAS REVERSED TO PERFORM A HEAT TREATMENT OF THE CIRCULATING WATER SYSTEM. AT APPROXIMATELY 1500, A HEAVY INFLUX OF SEAWEED OCCURRED WHICH LED TO THE STOPPAGE OF ONE CIRCULATING WATER PUMP AND A DEGRADATION OF CONDENSER VACUUM. THE OPERATORS BEGAN TO NORMALIZE (RETURN TO NORMAL ALIGNMENT) THE CIRCULATING WATER TUNNELS AT 1646 IN AN ATTEMPT TO REDUCE THE EFFECTS OF THE SEAWEED INFLUX. AT 1702, THE TURBINE TRIPPED ON LOW VACUUM, CAUSING THE REACTOR TO TRIP ON LOSS OF LOAD. THE EMERGENCY FEEDWATER ACTUATION SYSTEM FOR STEAM GENERATOR #2 (EFAS 2) ACTUATED ON LOW STEAM GENERATOR LEVEL. ALL COMPONENTS ASSOCIATED WITH EFAS 2 WERE VERIFIED TO HAVE OPERATED IN ACCORDANCE WITH DESIGN. THE TURBINE TRIP ON LOW VACUUM RESULTED FROM NORMALIZING THE CIRCULATING WATER TUNNELS WITH ONE CIRCULATING WATER PUMP STOPPED DURING AN INGRESS OF LARGE AMOUNTS OF SEAWEED. OPERATING INSTRUCTIONS FOR REALIGNING THE CIRCULATING WATER TUNNELS DID NOT INCLUDE ADEQUATE PREREQUISITES THAT WOULD HAVE PREVENTED NORMALIZING THE CIRCULATING WATER TUNNELS UNDER THESE CONDITIONS. THE CONDENSER WATER BOXES WERE CLEANED AS NECESSARY, AND THE UNIT WAS RETURNED TO POWER OPERATION AT 1530 ON 10/12/87.

[250] SEABROOK 1 DOCKET 50-443 LER 87-015 REV 01
UPDATE ON INADVERTENT SAFETY INJECTION.
EVENT DATE: 081387 REPORT DATE: 123187 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 207488) ON AUGUST 13, 1987 AT 1:30 PM EDT, MULTIPLE ENGINEERED SAFETY FEATURES (ESFS) ACTUATED UPON RECEIPT OF A LOW PRESSURIZER PRESSURE SIGNAL. THE CAUSE WAS DETERMINED TO BE THE ACTUATION OF SAFETY INJECTION RESET CAUSED BY A MOMENTARY CONTACT BOUNCE OF THE TRAIN B PRESSURIZER PRESSURE SAFETY INJECTION BLOCK/RESET SWITCH RESULTING FROM AN IMPACT TO THE MAIN CONTROL BOARD. ONCE THE TRAIN B SAFETY INJECTION SIGNAL WAS GENERATED, ALL SYSTEMS OPERATED AS DESIGNED. APPROXIMATELY 10,000 GALLONS OF BORATED WATER WERE INJECTED INTO THE REACTOR COOLANT SYSTEM. OPERATORS RESPONDED IN ACCORDANCE WITH SEABROOK STATION PROCEDURES. SAFETY INJECTION WAS PROPERLY TERMINATED, AND ALL SYSTEMS WERE RETURNED TO NORMAL STATUS. TO PREVENT RECURRENCE, THE INSTALLATION PROCEDURES FOR THE MODIFICATION IN PROGRESS WERE AMENDED TO REQUIRE THE SOLID STATE PROTECTION SYSTEM (SSPS) TO BE IN THE TEST MODE BEFORE PERFORMING SIMILAR WORK ACTIVITIES ON THE MAIN CONTROL BOARD (MCB). AN ENGINEERING EVALUATION DETERMINED THAT AN IMPACT TO A CONTROL PANEL, SUCH AS THAT CAUSED BY A PRICK PUNCH, CAN PRODUCE EXCESSIVE CONTACT CHATTER IN A PROPERLY QUALIFIED OT2 SWITCH. THEREFORE,

ADMINISTRATIVE CONTROLS WILL BE ESTABLISHED BY MARCH 31, 1988 TO ENSURE THAT SUCH IMPACTS ARE MINIMIZED BUT WHEN EMPLOYED ARE ACCOMPANIED BY SPECIFIC PRECAUTIONARY MEASURES TO PREVENT ADVERSE CONSEQUENCES.

[251] SEABROOK 1 DOCKET 50-443 LER 87-019
FAILURE OF GOULD/TELEMECANIQUE J-10 RELAYS.
EVENT DATE: 101387 REPORT DATE: 111287 NSSS: WE TYPE: PWR
VENDOR: GOULD INC.

(NSIC 207138) THREE ITE GOULD/TELEMECANIQUE CLASS J CONTROL RELAYS FAILED IN NON-SAFETY-RELATED CIRCULATING WATER (CW) SYSTEM CIRCUIT APPLICATIONS DURING A FOUR MONTH PERIOD BETWEEN APRIL 1987 AND AUGUST 1987. THE FAILED RELAYS PROVIDE CW VACUUM BREAKER VALVE POSITION INDICATION. THE IMMEDIATE CAUSE OF THE RELAY FAILURES WAS THE BREAK UP OF EMBRITTLED PLASTIC AT THE END OF THE MAGNET YOKE ASSEMBLY. THE BROKEN PIECES CAUSED THE MOVEABLE PART OF THE RELAY TO JAM. THE BINDING COULD OCCUR IN EITHER THE CLOSED OR OPEN CONDITION. THE EMBRITTLEMENT RESULTED WHEN THE PLASTIC WAS SUBJECTED TO HIGH TEMPERATURES FOR A LONGER DURATION THAN THE MATERIAL COULD WITHSTAND. THE SOURCE OF THE HIGH TEMPERATURE WAS DETERMINED TO BE EXCESSIVE HEAT GENERATED BY THE MAGNETIC COIL WHEN THE RELAY IS ENERGIZED. ON OCTOBER 9, 1987, TELEMECANIQUE NOTIFIED THE NRC PURSUANT TO 10 CFR 21 REGARDING EMBRITTLEMENT OF THE ARMATURE CARRIER IN CLASS J INDUSTRIAL CONTROL RELAYS THAT HAVE NUMBER 816 A.C. COILS. THE MAGNETIC COILS IN THESE RELAYS ARE NOT STANDARD COILS AND WERE ONLY SUPPLIED TO SEABROOK STATION. ON OCTOBER 13, 1987, NEW HAMPSHIRE YANKEE NOTIFIED THE NRC OF A POTENTIAL UNANALYZED CONDITION IN ACCORDANCE WITH 10 CFR 50.72.

[252] SEQUOYAH 1 DOCKET 50-327 LER 86-015 REV 03
UPDATE ON FAILURE TO PROPERLY FUNCTIONAL TEST RCP UV AND UF DEVICES AND ELECTRICAL BREAKERS DUE TO INADEQUATE TEST PROCEDURES.
EVENT DATE: 040786 REPORT DATE: 112587 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 207241) THIS LER IS BEING REVISED TO ADDRESS A CHANGE IN THE TVA POSITION WITH RESPECT TO TESTING THE INSTANTANEOUS TRIP FUNCTION OF MOLDED-CASE CIRCUIT BREAKERS, AND TO MAKE A CORRECTION TO INFORMATION ORIGINALLY SUPPLIED WITH RESPECT TO STATUS OF CORRECTIVE ACTIONS. DEFICIENT TESTING METHODS WERE USED ON TWO SEPARATE EVENTS THAT RESULTED IN A FAILURE TO PROPERLY FUNCTIONAL TEST THE UNDER VOLTAGE (UV) AND UNDER FREQUENCY (UF) TRIP ATTACHMENTS FOR THE REACTOR COOLANT PUMPS AND THE CIRCUIT BREAKERS USED FOR ELECTRICAL OVERCURRENT PROTECTION OF THE CONTAINMENT ELECTRICAL PENETRATIONS. THESE EVENTS WERE IDENTIFIED ON APRIL 7 AND APRIL 9, 1986, RESPECTIVELY WITH BOTH UNITS IN MODE 5 AT LESS THAN 200 DEGREES F. THE DEFICIENT TEST METHOD ON THE UV AND THE UF TRIP ATTACHMENTS WAS DUE TO A FAILURE TO INJECT A SIMULATED SIGNAL AT THE INPUT OF THE SENSOR TO PERFORM THE FUNCTIONAL VERIFICATION. THE CIRCUIT BREAKERS WERE NOT TESTED BY AN APPROPRIATE METHOD IN THAT THE POLES WERE TIED IN SERIES FOR THE THERMAL CURRENT TEST. ALSO, LISTS OF FUSES AND CIRCUIT BREAKERS USED TO SELECT THE 10 PERCENT SAMPLE FOR TESTINGS WERE NOT COMPLETE. NONTESTING OF THE INSTANTANEOUS TRIP FUNCTION FOR MOLDED CASE CIRCUIT BREAKERS HAS ALSO BEEN IDENTIFIED AS A TEST DEFICIENCY.

[253] SEQUOYAH 1 DOCKET 50-327 LER 87-030 REV 01
UPDATE ON BLOWN FUSE IN EMERGENCY START CIRCUITS RESULT IN SPURIOUS EMERGENCY DIESEL GENERATOR STARTS ON TWO OCCASIONS.
EVENT DATE: 062087 REPORT DATE: 112587 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)
VENDOR: LITTLEFUSE INC

(NSIC 207326) THIS LER IS REVISED TO PROVIDE ADDITIONAL INFORMATION ON COMPLETED

CORRECTIVE ACTIONS. ON JUNE 20, 1987, AT 0055 EST, ALL FOUR STANDBY DIESEL GENERATORS (D/GS) STARTED DUE TO A BLOWN FUSE IN A REMOTE EMERGENCY START CIRCUITRY FOR 2A-A D/G. THE BLOWN FUSE (FLAS-5, LOT NO. 3) CAUSED LOSS OF VOLTAGE TO THE 2A-A REMOTE EMERGENCY START CIRCUITRY. STANDBY D/GS 1A-A, 1B-B, 2A-A, AND 2B-B STARTED AS REQUIRED. ON JULY 4, 1987, AT 1212 EST, ALL FOUR STANDBY D/GS STARTED DUE TO A BLOWN FUSE IN 125V DC VITAL BATTERY BOARD II, CIRCUIT C-16. THE BLOWN FUSE (FLAS-5, LOT NO. 2) CAUSED LOSS OF POWER TO THE LOGIC RELAY PANEL FOR 1B-B D/G. UPON LOSS OF VOLTAGE TO THE 1B-B D/G REMOTE EMERGENCY START CIRCUITRY, STANDBY D/GS 1A-A, 1B-B, 2A-A, AND 2B-B STARTED AS REQUIRED. FOR BOTH EVENTS DESCRIBED ABOVE, NO DAMAGE OCCURRED TO THE D/GS, AND NONE OF THE D/GS LOADED BECAUSE A DEGRADED VOLTAGE CONDITION DID NOT EXIST ON THE 6.9 KV SHUTDOWN BOARDS. TVA HAS CONTRACTED WITH LITTLEFUSE INC. TO SUPPLY 15,000 FLAS-5 FUSES. APPROXIMATELY 3,200 OUT OF 3,702 (DELIVERED) FUSES WERE INSTALLED IN MARCH AND APRIL 1987, 1,683 OF WHICH WERE FROM LOTS 2 AND 3. AS OF JULY 13, 1987, 69 FLAS-5 FUSE FAILURES HAVE OCCURRED. OF THESE, 67 FAILED FUSES WERE SUPPLIED IN LOTS 2 AND 3. THE LOT NUMBER OF THE REMAINING TWO FAILED FUSES COULD NOT BE DETERMINED.

[254] SEQUOYAH 1 DOCKET 50-327 LER 87-039 REV 01
UPDATE ON CONTROL ROOM EMERGENCY VENTILATION SYSTEM SINGLE FAILURE CRITERIA VIOLATED DUE TO A DESIGN ERROR WHICH COULD RESULT IN EXCEEDING ALLOWABLE DOSE TO OPERATORS.
EVENT DATE: 071087 REPORT DATE: 112587 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 207327) ON JULY 10, 1987, AT 1630 EST WITH UNITS 1 AND 2 IN MODE 5 (COLD SHUTDOWN) DURING THE PERFORMANCE OF SURVEILLANCE INSTRUCTION (SI)-144.2, "CONTROL ROOM EMERGENCY VENTILATION TEST," IT WAS OBSERVED THAT THE POTENTIAL EXISTED FOR A SINGLE FAILURE OF THE MAIN CONTROL ROOM (MCR) NORMAL PRESSURIZATION SYSTEM, WHEN OPERATING DURING A CONTROL ROOM ISOLATION (CRI) ACTUATION, TO VIOLATE GENERAL DESIGN CRITERIA (GDC)-19 OF 10 CFR 50 APPENDIX A, "CONTROL ROOM." GDC-19 REQUIRES THE MCR TO BE DESIGNED WITH ADEQUATE RADIATION PROTECTION FOR PERSONNEL DURING AN ACCIDENT CONDITION. AT 1808 EST ON JULY 10, 1987, SUBSEQUENT TO ADDITIONAL INVESTIGATION OF THE SI PERFORMANCE, BOTH TRAINS OF CONTROL ROOM EMERGENCY VENTILATION SYSTEM (CREVS) WERE DECLARED INOPERABLE, AND ACTION STATEMENT "B" OF TECH SPEC LIMITING CONDITION FOR OPERATION 3.7.7 WAS ENTERED. ON A CRI ACTUATION, THE NORMAL PRESSURIZATION FANS CONTINUE TO SUPPLY A REDUCED AMOUNT OF UNFILTERED AIR TO THE CONTROL BUILDING (CB) LOWER FLOORS ONLY, WHILE THE CREVS SUPPLIES FILTERED AIR TO THE MCR TO MAINTAIN THE MCR HABITABILITY ZONE AT A POSITIVE PRESSURE.

[255] SEQUOYAH 1 DOCKET 50-327 LER 87-060 REV 02
UPDATE ON INADVERTENT DIESEL GENERATOR START CAUSED BY INADEQUATE PROCEDURE DURING SHUTDOWN BOARD 1B-B MAINTENACE.
EVENT DATE: 082787 REPORT DATE: 112587 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)
VENDOR: GENERAL ELECTRIC CO.
MORRISON-KNUDSON COMPANY, INC.
WOODWARD GOVERNOR COMPANY

(NSIC 207328) THIS REVISION PROVIDES ADDITIONAL INFORMATION GAINED FROM LABORATORY TESTING AND FROM VISUAL INSPECTION OF THE DIESEL GENERATOR ACTUATORS THAT HAD BEEN REBUILT AS THE RESULT OF AN ACTUATOR FAILURE. THIS REPORT ALSO FULFILLS THE SPECIAL REPORTING REQUIREMENT FOR THE INOPERABLE FIRE PUMPS. ON AUGUST 27, 1987, AT 1140 EST WITH UNITS 1 AND 2 IN MODE 5 (COLD SHUTDOWN), AN INADVERTENT START OF THREE OF THE FOUR D/G UNITS (1A-A, 2A-A, AND 2B-B) OCCURRED DURING THE PERFORMANCE OF A SPECIAL MAINTENANCE INSTRUCTION (SMI). IN PERFORMING THE INSTRUCTION, AN ELECTRICIAN'S KNIFE BLADE INADVERTENTLY CAME INTO CONTACT WITH TWO BARE ENERGIZED STUDS AND CAUSED THE NORMAL FEEDER BREAKER FOR THE 1B-B

SHUTDOWN BOARD TO TRIP. THE BREAKER TRIP CAUSED A BLACKOUT CONDITION WHICH IN TURN STARTED ALL OF THE OPERABLE D/GS. THE 1B-B D/G DID NOT START BECAUSE IT WAS OUT OF SERVICE FOR MAINTENANCE. THE 1B-B SHUTDOWN BOARD WAS REENERGIZED AT 1149 EST, AND ALL OPERATING D/GS WERE SHUT DOWN AT 1153 EST. DURING SHUTDOWN OF THE 1A-AD/G, A MALFUNCTION OF THE HYDRAULIC ACTUATOR CAUSED THE DIESEL TO TRIP ON OVERSPEED. THE ROOT CAUSE OF THE D/G START WAS AN INADEQUATE PROCEDURE. THE PROCEDURE INVOLVED AS WELL AS PROCEDURES FOR THE OTHER THREE SHUTDOWN BOARDS WERE REVISED.

[256] SEQUOYAH 1 DOCKET 50-327 LER 87-071
TWO SEPARATE CONDITIONS IN THE ESSENTIAL RAW COOLING WATER (ERCW) SYSTEM RESULT IN THE POTENTIAL DEGRADATION OF ERCW FLOW TO BOTH UNITS.
EVENT DATE: 100587 REPORT DATE: 112587 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 207340) THIS LER IS COMPRISED OF TWO SEPARATE CONDITIONS RELATED TO THE ESSENTIAL RAW COOLING WATER (ERCW) SYSTEM. THE CONDITIONS ARE BEING REPORTED AS A VOLUNTARY LER TO INFORM NRC OF ISSUES INVOLVING THE ERCW SYSTEM. ON OCTOBER 5, 1987, WITH UNITS 1 AND 2 IN MODE 5, IT WAS CONCLUDED THAT ELECTRICAL INTERLOCKS WHICH ARE DESIGNED TO PREVENT AN ERCW STRAINER FROM AUTOMATICALLY BACKWASHING IF A STRAINER IN THE SAME TRAIN IS ALREADY BACKFLUSHING HAD BEEN DISABLED AND WERE NOT INCLUDED IN THE MOST RECENT ERCW WIRING DIAGRAM. THE EVENT WAS CAUSED BY PERSONNEL NOT PERFORMING AN ADEQUATE 10 CFR 50.59 EVALUATION PERFORMED TO ENSURE THAT MANUAL OPERATION DID NOT REPRESENT AN UNREVIEWED SAFETY QUESTION. THE EVENT WAS CAUSED BY OPERATIONS PERSONNEL FAILING TO FOLLOW THE APPLICABLE ADMINISTRATIVE PROCEDURE WHEN CHANGING THE STATUS OF THE SYSTEMS THAT ARE IMPORTANT TO PLANT SAFETY. APPROPRIATE DOCUMENTATION HAS SUBSEQUENTLY BEEN ISSUED TO DEMONSTRATE THAT THE MANUAL OPERATION OF THESE SYSTEMS SUBSEQUENTLY BEEN ISSUED TO DEMONSTRATE THAT THE MANUAL OPERATION OF THESE SYSTEMS IS ACCEPTABLE. TVA HAS EVALUATED THESE CONDITIONS, BOTH INDIVIDUALLY AND IN COMBINATION, AND DETERMINED THAT THE ERCW SYSTEM'S ABILITY TO PERFORM ITS DESIGNED SAFETY FUNCTION WAS NOT COMPROMISED.

[257] SEQUOYAH 1 DOCKET 50-327 LER 87-045
AN INADEQUATE DESIGN CONTROL PROCESS RESULTED IN THE LACK OF FUSE COORDINATION POTENTIALLY RENDERING THE ESSENTIAL RAW COOLING WATER SYSTEM INOPERABLE.
EVENT DATE: 100987 REPORT DATE: 110687 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 207133) ON OCTOBER 9, 1987, WITH UNITS 1 AND 2 IN MODE 5 (0 PERCENT POWER, 3 PSIG, 120 DEGREES F AND 0 PERCENT POWER, 120 PSIG, 120 DEGREES F, RESPECTIVELY), IT WAS DETERMINED THAT A POTENTIAL FAILURE OF NON-CLASS 1E SPEED SWITCHES COULD RENDER THE SAFETY-GRADE ESSENTIAL RAW COOLING WATER (ERCW) SYSTEM INOPERABLE. THE NON-CLASS 1E SPEED SWITCHES, WHICH ARE NESTED WITHIN THE CLASS 1E CIRCUITS THAT CONTROL THE ERWC TRAVELING WATER SCREENS, DID NOT HAVE THE PROPER FUSE COORDINATION. THIS COORDINATION IS REQUIRED TO ENSURE THAT THE FAILURE OF A SPEED SWITCH WOULD NOT RESULT IN THE FAILURE OF THE TRAVELING WATER SCREEN. THE CAUSE OF THE EVENT WAS AN INADEQUATE DESIGN CONTROL PROCESS THAT WAS IN PLACE WHEN A DESIGN CHANGE WAS IMPLEMENTED TO UPGRADE THE ERCW TRAVELING WATER SCREENS TO A CLASS 1E SYSTEM. THE DESIGN CONTROL PROCESS THAT WAS IN EFFECT AT THE TIME OF THE CLASS 1E UPGRADE DID NOT REQUIRE A REVIEW OF FUSE COORDINATION FOR MODIFICATIONS. FUSE DOCUMENTATION AND CALCULATIONS WHICH WOULD ENSURE FUSE COORDINATION WERE NOT PERFORMED. TVA HAS SUBSEQUENTLY ISSUED A DESIGN CHANGE NOTICE TO DOCUMENT THE FUSE REQUIREMENTS NECESSARY TO ENSURE ADEQUATE ISOLATION OF THE NON-CLASS 1E CIRCUITS WITHIN THE ERCW SYSTEM AND WILL REPLACE THE SUBJECT FUSES BEFORE UNIT 2 RESTART.

[258] SEQUOYAH 1 DOCKET 50-327 LER 87-064
IMPROPER FIT OF EMERGENCY RAW COOLING WATER FLOOD MODE SPOOL PIECES DUE TO MINOR PIPING ORIENTATION CHANGES.
EVENT DATE: 101387 REPORT DATE: 111287 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 207227) THIS REPORT IS BEING SUBMITTED AS A "VOLUNTARY REPORT" TO IDENTIFY A POTENTIAL PROBLEM WITH FLOOD MODE SPOOL PIECES AND TO KEEP NRC INFORMED OF ONGOING ACTIVITIES AT SEQUOYAH NUCLEAR PLANT. WITH BOTH UNITS 1 AND 2 IN MODE 5 (COLD SHUTDOWN), A CONDITION WHERE FIVE EMERGENCY RAW COOLING WATER (ERCW) FLOOD MODE SPOOL PIECES DID NOT FIT PROPERLY WAS DISCOVERED DURING THE INTEGRATED DESIGN INSPECTION (IDI) FOR THE ERCW SYSTEM. A CONDITION ADVERSE TO QUALITY REPORT WAS INITIATED TO INVESTIGATE THE POTENTIAL PROBLEM. DURING THE STAGE II FLOOD PREPARATION PLAN, THESE SPOOL PIECES ARE REQUIRED TO BE INSTALLED TO REPLACE COMPONENT COOLING SYSTEM WITH ERCW AS COOLING WATER FOR SPENT FUEL POOL HEAT EXCHANGERS, REACTOR COOLANT PUMP THERMAL BARRIERS, SAMPLE SYSTEM HEAT EXCHANGERS, AND TO REPLACE RAW COOLING WATER WITH ERCW FOR THE ICE CONDENSER REFRIGERATION SYSTEM. THE ROOT CAUSE OF THE CONDITION COULD NOT BE SPECIFICALLY IDENTIFIED. THE MOST PROBABLE CAUSE IS THAT GRADUAL CHANGES IN INTERNAL PIPING STRESSES OVER THE YEARS ALONG WITH SYSTEM MODIFICATIONS HAVE RESULTED IN SLIGHT CHANGES IN THE ORIENTATION OF THE PIPING. EACH SPOOL PIECE WAS INSPECTED, AND IT WAS DETERMINED THAT ALL COULD BE INSTALLED WITH HANGER ADJUSTMENTS, GASKET ADJUSTMENTS, OR ALTERATIONS IF FLOODING CONDITIONS REQUIRED INSTALLATION.

[259] SEQUOYAH 1 DOCKET 50-327 LER 87-070
DIESEL GENERATOR VOLTAGE REGULATOR HAD A SLOWED VOLTAGE RESPONSE BECAUSE OF A COMPONENT DEFECT FOUND DURING SURVEILLANCE TESTING.
EVENT DATE: 102187 REPORT DATE: 111987 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)
VENDOR: BASLER ELECTRIC COMPANY
MORRISON-KNUDSON COMPANY, INC.

(NSIC 207231) THIS REPORT IS BEING SUBMITTED UNDER 10 CFR 21 AND AS A "VOLUNTARY LER" TO IDENTIFY A DEFECTIVE VOLTAGE REGULATOR ON AN EMERGENCY DIESEL GENERATOR. ON OCTOBER 21, 1987, AT 1100 EDT WITH BOTH UNITS 1 AND 2 IN MODE 5 (COLD SHUTDOWN) IT WAS DISCOVERED WHILE TROUBLESHOOTING A VOLTAGE REGULATOR PROBLEM ON EMERGENCY DIESEL GENERATOR (D/G) 2A-A, THAT A VOLTAGE REGULATING DIODE WAS INCORRECTLY INSTALLED. DURING THE PERFORMANCE OF SURVEILLANCE INSTRUCTION (SI)-26.2A, "LOSS OF OFFSITE POWER WITH SAFETY INJECTION - D/G 2A-A CONTAINMENT ISOLATION TEST," D/G 2A-A, CAME UP TO SPEED AND THE CIRCUIT BREAKER THAT CONNECTS THE D/G TO THE 6900 VOLT SHUTDOWN BOARD CLOSED WITH THE D/G TERMINAL VOLTAGE AT APPROXIMATELY 5200 VOLTS. THE IMMEDIATE CAUSE OF THE SLOW DEVELOPMENT OF VOLTAGE WAS THAT A DIODE WAS INSTALLED WITH THE POLARITY REVERSED. THE ROOT CAUSE OF THIS EVENT IS STILL UNDER INVESTIGATION. THE VOLTAGE REGULATOR WITH THE REVERSED DIODE WAS MANUFACTURED BY BASLER ELECTRIC COMPANY, AND WAS INSTALLED ON NOVEMBER 8, 1986 FOLLOWING TROUBLESHOOTING FOR VOLTAGE SWING PROBLEMS. THE POST MAINTENANCE TEST DID NOT DETECT THE REVERSED DIODE AT THAT TIME. THE DEFECTIVE VOLTAGE REGULATOR WITH THE REVERSED DIODE HAS BEEN REPAIRED AND SI-26.2A HAS BEEN PERFORMED WITH ACCEPTABLE RESULTS.

[260] SEQUOYAH 1 DOCKET 50-327 LER 87-066
POTENTIAL LOSS OF COMPONENT COOLING WATER INVENTORY DUE TO NON-SAFETY-RELATED DESIGN OF PUMP SEAL LEAKAGE RETURN SYSTEM.
EVENT DATE: 102287 REPORT DATE: 111987 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 207228) ON OCTOBER 22, 1987, WITH UNITS 1 AND 2 IN MODE 5 (0 PERCENT POWER, 3 PSIG, 125 DEGREES F AND 0 PERCENT POWER, 50 PSIG, 127 DEGREES F, RESPECTIVELY), IT WAS DETERMINED THAT THE COMPONENT COOLING WATER SYSTEM (CCS) PUMP SEAL LEAKAGE

COULD DEPLETE THE CCS WATER INVENTORY IN APPROXIMATELY 52 HOURS FOLLOWING A LOSS OF COOLANT ACCIDENT (LOCA) ASSUMING TWO CCS PUMPS RUNNING IN THE SAME TRAIN AND LOSS OF OFFSITE POWER. THE NORMAL MAKEUP TO THE CCS IS FROM THE NON-SAFETY-RELATED DEMINERALIZED WATER SYSTEM. MAKEUP CAPABILITY IS AVAILABLE DURING THIS POSTULATED EVENT VIA THE SAFETY GRADE ESSENTIAL RAW COOLING WATER (ERCW) SYSTEM. HOWEVER, AN ENTRY INTO THE AUXILIARY BUILDING FOR INSTALLATION OF A SPOOL PIECE IS REQUIRED TO CONFIGURE THE ERCW SYSTEM AS A SOURCE OF MAKEUP TO THE CCS SURGE TANK. AN ANALYSIS WAS PERFORMED WHICH CONFIRMED THAT PERSONNEL COULD ENTER THE AUXILIARY BUILDING OF CCS INVENTORY DUE TO PUMP SEAL LEAKAGE. A REVIEW OF EXISTING PROCEDURES RESULTED IN THE CONCLUSION THAT THERE WAS NO REFERENCE TO THE ERCW SYSTEM AS EMERGENCY MAKEUP TO THE CCS, NOR WERE THERE WRITTEN INSTRUCTIONS WHICH PROVIDED GUIDANCE FOR INSTALLATION AND ALIGNMENT OF THE SPOOL PIECE. THUS, A CONDITION WAS DISCOVERED THAT WAS NOT COVERED BY THE PLANT'S OPERATING AND EMERGENCY PROCEDURES AND IS BEING REPORTED AS REQUIRED.

[261] SEQUOYAH 1 DOCKET 50-327 LER 87-065 REV 01
UPDATE ON ERCW SCREEN WASH PUMPS WERE OMITTED FROM THE ASME SECTION XI TEST PROGRAM RESULTING IN POTENTIAL DEGRADATION OF ERCW FLOW TO BOTH UNITS.
EVENT DATE: 102387 REPORT DATE: 112587 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 207338) THIS LER IS BEING REVISED TO CLARIFY THE CORRECTIVE ACTION TAKEN BY TVA TO ENSURE THAT PUMPS WHICH ARE REQUIRED TO MITIGATE THE CONSEQUENCES OF AN ACCIDENT OR ACHIEVE A SAFE SHUTDOWN OF THE PLANT ARE INCLUDED IN THE ASME SECTION XI TEST PROGRAM. ON OCTOBER 23, 1987, WITH UNITS 1 AND 2 IN MODE 5 (0 PERCENT POWER, 3 PSIG, 125 DEGREES F AND 0 PERCENT POWER, 100 PSIG, 127 DEGREES F, RESPECTIVELY), IT WAS DETERMINED THAT THE ERCW SCREEN WASH PUMPS WERE NOT INCLUDED IN THE ASME SECTION XI TEST PROGRAM. PERIODIC TESTING OF THE ERCW SCREEN WASH PUMPS IN ACCORDANCE WITH ASME SECTION XI IS NECESSARY TO DETECT POTENTIAL DEGRADATION IN THE PUMPS. A COMPLETE FAILURE OF THE ERCW SCREEN WASH SYSTEM COULD RESULT IN THE GRADUAL CLOGGING OF THE ERCW TRAVELING SCREENS AND SUBSEQUENT DEGRADATION OF ERCW FLOW TO BOTH UNITS. HOWEVER, THERE IS A HIGH DEGREE OF CONFIDENCE THAT THESE PUMPS WOULD HAVE FUNCTIONED PROPERLY DURING ACCIDENT CONDITIONS. THE PUMPS ARE TVA CLASS C SAFETY-RELATED PUMPS, WHICH HAVE BEEN IN USE SINCE INITIAL PLANT STARTUP AND THERE HAS BEEN NO HISTORY OF CLOGGING PROBLEMS WITH THE TRAVELING SCREENS DUE TO INSUFFICIENT FLOW FROM THE SCREEN WASH PUMPS. THE EVENT RESULTED WHEN PERSONNEL DEVELOPING THE SEQUOYAH ASME SECTION XI TEST PROGRAM ASSUMED THE SCREEN WASH PUMPS WERE NOT SAFETY RELATED.

[262] SEQUOYAH 1 DOCKET 50-327 LER 87-067
MONITORING TANK DILUTION FLOW WAS NOT VERIFIED DURING A LIQUID EFFLUENT RELEASE AS THE RESULT OF PERSONNEL ERROR.
EVENT DATE: 102787 REPORT DATE: 111987 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 207229) ON OCTOBER 27, 1987, WITH UNITS 1 AND 2 IN MODE 5 (0 PERCENT POWER, 3 PSIG, 120 DEGREES F AND 0 PERCENT POWER, 5 PSIG, 120 DEGREES F, RESPECTIVELY), IT WAS DISCOVERED THAT A PORTION OF A SYSTEM OPERATING INSTRUCTION (SOI) RELATING TO THE BATCH RELEASE OF LIQUID RADIOLOGICAL EFFLUENTS FROM THE MONITOR TANK WAS INCORRECTLY PERFORMED. IN ORDER TO ENSURE PROPER DILUTION OF THE MONITOR TANK CONTENTS AND COMPLY WITH THE LIMITS IMPOSED BY TECH SPEC 3.11.1.1, THE SOI REQUIRES VERIFICATION OF A COOLING TOWER BLOWDOWN FLOW RATE OF AT LEAST 15,000 GALLONS PER MINUTE (GPM). HOWEVER, FOLLOWING A REVIEW OF THE SUBJECT SOI, IT COULD NOT BE VERIFIED THAT THE PROPER DILUTION FLOW RATE (I.E., COOLING TOWER BLOWDOWN) WAS ESTABLISHED BEFORE THE RELEASE OF THE MONITOR TANK CONTENTS. THE EVENT WAS CAUSED BY A LACK OF ATTENTION TO DETAIL WHEN PERSONNEL FAILED TO FOLLOW AN APPROVED PROCEDURE AND INCORRECTLY USED AN INOPERABLE FLOW TRANSMITTER TO ESTIMATE THE DILUTION FLOW RATE. USE OF THIS FLOW TRANSMITTER WAS PROHIBITED BY THE SOI BECAUSE THE DIFFUSER POND WAS NOT AT A REQUIRED MINIMUM LEVEL.

PERSONNEL INVOLVED IN THE VIOLATION HAVE SUBSEQUENTLY BEEN COUNSELED BY MANAGEMENT PERSONNEL WHO STRESSED THE IMPORTANCE OF ATTENTION TO DETAIL AND STRICT ADHERENCE TO APPROVED PROCEDURES.

[263] SEQUOYAH 1 DOCKET 50-327 LER 87-068
TECHNICAL SPECIFICATION SURVEILLANCE INTERVAL EXCEEDED DUE TO AN INCOMPLETE WORK PACKAGE.
EVENT DATE: 102787 REPORT DATE: 111987 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 207230) ON OCTOBER 27, 1987, WITH UNITS 1 AND 2 IN MODE 5 (0 PERCENT POWER, 3 PSIG, 120 DEGREES F AND 0 PERCENT POWER, 5 PSIG, 120 DEGREES F, RESPECTIVELY), IT WAS DISCOVERED DURING A FINAL REVIEW OF A SURVEILLANCE INSTRUCTION WORK PACKAGE THAT A DATA SHEET (ONE PAGE) WAS MISSING FROM THE COMPLETED SURVEILLANCE INSTRUCTION. AS A RESULT, RELATED FIRE ALARM AND ANNUNCIATION EQUIPMENT FOR ZONE 198 (125 VOLT BATTERY BOARD ROOM II) EXCEEDED ITS TECH SPEC SURVEILLANCE INTERVAL. THE EVENT WAS CAUSED BY A PERSONNEL ERROR IN THE ELECTRICAL MAINTENANCE GROUP IN THAT THEY DID NOT VERIFY THAT THE SURVEILLANCE REQUIREMENTS WERE SATISFIED DURING PERFORMANCE OF THE SUBJECT SURVEILLANCE INSTRUCTION (SI)-234.6. HOWEVER, ONCE IT WAS DETERMINED THAT THE SUBJECT DATA WAS NOT PERFORMED, A SPECIAL PERFORMANCE PACKAGE WAS ISSUED, AND THE FIRE ALARMS AND ANNUNCIATORS WERE SUCCESSFULLY TESTED. PERSONNEL INVOLVED AND RESPONSIBLE FOR APPROVED WORK PACKAGES WILL BE COUNSELED AS TO THE IMPORTANCE OF VERIFYING (PAGES PRESENT AND LATEST REVISION) THAT ALL WORK PACKAGES SATISFY SURVEILLANCE REQUIREMENTS. ADMINISTRATIVE INSTRUCTION (AI)-4, "PREPARATION, REVIEW, APPROVAL AND USE OF SITE PROCEDURES/INSTRUCTIONS," WILL BE REVISED TO PROVIDE REQUIREMENTS FOR VERIFICATION OF CORRECT AND COMPLETE INSTRUCTIONS.

[264] SEQUOYAH 1 DOCKET 50-327 LER 87-069
FAILURE TO INCLUDE CONTAINMENT SPRAY PUMP START CONTACTS RESPONSE TIME AND VALVE RESPONSE TIME IN THE SYSTEM TOTAL RESPONSE TIME RESULTS IN FAILURE TO SATISFY SURVEILLANCE REQUIREMENT.
EVENT DATE: 102787 REPORT DATE: 111987 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 207339) ON OCTOBER 27, 1987, WITH BOTH UNITS IN MODE 5, IT WAS DETERMINED THAT THE CONTAINMENT SPRAY PUMP START INTERLOCK CONTACTS IN THE OPEN CONTROL CIRCUIT FOR THE CONTAINMENT SPRAY DISCHARGE VALVES WERE NOT BEING INCLUDED IN THE RESPONSE TIME TESTING FOR THE CONTAINMENT SPRAY SYSTEM. THE CONTAINMENT SPRAY PUMP START INTERLOCK CONTACTS OPERATE AS PART OF THE PUMP MOTOR BREAKER AND ARE IN SERIES WITH THE OPEN COIL OF THE ASSOCIATED DISCHARGE VALVE MOTOR SUCH THAT FOR THE DISCHARGE VALVE TO OPEN AUTOMATICALLY, THE PUMP START INTERLOCK CONTACTS MUST BE CLOSED. THE PRESENT PROCEDURE USED FOR RESPONSE TIME TESTING THE VALVE (INSTRUMENT MAINTENANCE INSTRUCTION (IMI)-99, RT-644A AND RT-644B, "RESPONSE TIME TESTING, ENGINEERED SAFETY FEATURES, SLAVE RELAY K644") BYPASSES THE PUMP START INTERLOCK CONTACTS VIA A PAIR OF TEST RELAY CONTACTS TO AVOID HAVING TO OPERATE THE PUMP AND SPRAY CONTAINMENT WHEN THE VALVE OPENS. AS A RESULT, THE PUMP START INTERLOCK CONTACTS CLOSURE TIME HAS NOT BEEN INCLUDED IN THE VALVE ACTUATION TIME WHEN COMBINING THE INDIVIDUAL COMPONENT TIMES IN SURVEILLANCE INSTRUCTION (SI)-247.900, "ENGINEERED SAFETY FEATURES RESPONSE TIME VERIFICATION."

[265] SEQUOYAH 2 DOCKET 50-328 LER 86-011 REV 03
UPDATE ON CONTAINMENT VENTILATION ISOLATION FROM ELECTROMAGNETIC INTERFERENCE ON A RADIATION MONITOR DUE TO A PERSONNEL ERROR.
EVENT DATE: 120786 REPORT DATE: 121887 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 1 (PWR)
VENDOR: GENERAL ATOMIC CO.

(NSIC 207501) THIS LER IS BEING REVISED TO CHANGE THE CORRECTIVE ACTION AND THE CORRECTIVE ACTION DUE DATE AND TO CLARIFY THE ROOT CAUSE. ON DECEMBER 7, 1986, AT 0820 CST WITH UNIT 2 IN MODE 5 (0 PERCENT POWER, 330 PSIG, 122 DEGREES F), A CONTAINMENT VENTILATION ISOLATION (CVI) OCCURRED. OPERATIONS PERSONNEL WERE RESPONDING TO AN INSTRUMENT MALFUNCTION ALARM FOR LOW SAMPLE LINE FLOW THROUGH THE CONTAINMENT BUILDING LOWER COMPARTMENT AIR MONITOR 2-RM-90-106 AND WERE ADJUSTING THE SAMPLE LINE THROTTLE VALVE ON THE RADIATION MONITOR (RM) WHEN THE CVI OCCURRED. ADJUSTING THE SAMPLE LINE THROTTLE VALVE CAUSED THE SAMPLE LINE FLOW SWITCH TO CHATTER GENERATING ELECTROMAGNETIC INTERFERENCE (EMI) WHICH INITIATED A HIGH RADIATION SIGNAL. FURTHER INVESTIGATION REVEALED THAT THE LOW FLOW ALARM HAD BEEN CAUSED BY A PARTIALLY CLOGGED CHARCOAL FILTER. CHEMICAL LABORATORY PERSONNEL USE TECHNICAL INSTRUCTION (TI)-16, "SAMPLING METHODS," FOR REPLACING THE FILTERS; HOWEVER, TI-16 DOES NOT GIVE GUIDELINES TO DETERMINE WHEN TO REPLACE FILTERS. SUBSEQUENTLY, CHEMICAL LABORATORY PERSONNEL HAD FAILED TO REPLACE THE FILTERS. A CONTRIBUTING CAUSE TO THIS EVENT COULD BE THE LACK OF ARC SUPPRESSION EQUIPMENT ON THE RM.

[266] SEQUOYAH 2 DOCKET 50-328 LER 87-003 REV 01
UPDATE ON FAILURE TO PERFORM INSPECTION ON ASME SECTION XI EQUIPMENT DUE TO PROCEDURAL INADEQUACY.
EVENT DATE: 013087 REPORT DATE: 111287 NSSS: WE TYPE: PWR

(NSIC 207232) THIS REVISION IS TO PROVIDE ADDITIONAL DETAILS CONCERNING A FAILURE TO PERFORM AN INSPECTION ON ASME SECTION XI EQUIPMENT DUE TO PROCEDURAL INADEQUACY. ON JANUARY 30, 1987, WITH UNIT 2 IN MODE 5 (0 PERCENT POWER, 280 PSIG, 120 DEGREES F), FIVE AREAS WERE IDENTIFIED THAT DID NOT RECEIVE AN ASME SECTION XI PRESSURE TEST INSPECTION DURING THE FIRST PERIOD OF THE PROGRAM. THIS WAS DETERMINED TO BE IN VIOLATION OF TECH SPEC SURVEILLANCE REQUIREMENT 4.0.5. ONE OF THE FIVE AREAS (CHEMICAL AND VOLUME CONTROL HOLDUP SYSTEM) HAS SINCE BEEN DETERMINED TO NOT REQUIRE A PRESSURE TEST INSPECTION IN ACCORDANCE WITH ASME SECTION XI. ALSO, FLOW CONTROL VALVE FCV-68-22 IN THE REACTOR FLANGE LEAKOFF LINE FROM THE REACTOR VESSEL FLANGE TO THE REACTOR COOLANT DRAIN TANK WAS ORIGINALLY REPORTED AS FCV-68-24. THIS EVENT WAS CAUSED BY A PROCEDURAL ERROR IN THAT THE SECTION XI PROCEDURES WERE NOT SPECIFIC ENOUGH. THE PROCEDURE ONLY INSTRUCTED PERSONNEL TO INSPECT CERTAIN TYPES OF EQUIPMENT, IT DID NOT INFORM THE INSPECTING PERSONNEL WHERE THE BOUNDARIES OF THE EQUIPMENT WERE LOCATED. THE FAILURE TO PERFORM THE ASME SECTION XI PRESSURE TEST INSPECTION DID NOT FUNCTIONALLY AFFECT THE OPERATION OF THE COMPONENTS. THEREFORE, NO SAFETY CONSEQUENCES RESULTED FROM THIS EVENT.

[267] SHEARON HARRIS 1 DOCKET 50-400 LER 87-058 REV 01
UPDATE ON EXCESSIVE REACTOR COOLANT SYSTEM LEAKAGE DUE TO VALVE FAILURE RCS HEAD VENT SYSTEM DURING TESTING.
EVENT DATE: 100987 REPORT DATE: 120987 NSSS: WE TYPE: PWR
VENDOR: TARGET ROCK CORP.

(NSIC 207392) ON 10/9/87, WITH THE PLANT IN MODE 1 AT 91% POWER, REACTOR COOLANT SYSTEM (RCS) LEAKAGE EXCEEDING TECHNICAL SPECIFICATION LIMITS OCCURRED WHEN BOTH IN SERIES ISOLATION VALVES IN THE RCS HEAD VENT SYSTEM SPURIOUSLY OPENED DURING TESTING, CREATING AN OPEN FLOWPATH FROM THE REACTOR VESSEL HEAD INTO THE CONTAINMENT ATMOSPHERE AND INTO THE PRESSURIZER RELIEF TANK. THE DOWNSTREAM VALVES SPURIOUSLY OPENED DUE TO THE PRESSURE TRANSIENT INDUCED WHEN THE UPSTREAM VALVES WERE OPENED FOR TESTING. THIS EVENT OCCURRED FIVE SEPARATE TIMES DURING THE COURSE OF TESTING, WITH ONE EVENT RESULTING IN THE DISCHARGE OF 198 GALLONS OF PRIMARY COOLANT. PLANT OPERATIONS PERSONNEL WERE BRIEFED ON THE BEHAVIOR OF THE VALVES AND ON THE SAFETY SIGNIFICANCE OF THE EVENT. THE PLANT ENTERED A SCHEDULED OUTAGE THE NEXT DAY, DURING WHICH MODIFICATIONS WERE MADE TO THE HEAD VENT SYSTEM, AND PROCEDURE REVISIONS WERE MADE, IN ACCORDANCE WITH VENDOR RECOMMENDATIONS. THESE CHANGES WERE MADE FOLLOWING A THOROUGH INVESTIGATION OF

THE EVENT WHICH REVEALED THAT PREVIOUS SIMILAR EVENTS HAD OCCURRED IN THE INDUSTRY, AT OTHER CP&L PLANTS, AND AT THE HARRIS PLANT. THE REASONS WHY SUCH INFORMATION WAS NOT CONSIDERED IN THE HARRIS PLANT DESIGN, AND WHY PROMPT CORRECTIVE ACTIONS HAD NOT BEEN TAKEN PREVIOUSLY, WERE INVESTIGATED AND APPROPRIATE ACTION WAS TAKEN TO RESOLVE THESE PROBLEMS.

[268] SHEARON HARRIS 1 DOCKET 50-400 LER 87-059
LOSS OF OFF-SITE POWER DUE TO INCOMING LINE BREAKER OPENING CAUSED BY PERSONNEL WORKING IN SWITCHYARD RELAY CABINET.
EVENT DATE: 101187 REPORT DATE: 110987 NSSS: WE TYPE: PWR
VENDOR: ENVIREX INC (DIV OF REXNORD)

(NSIC 207106) THE PLANT WAS IN MODE 5, COLD SHUTDOWN, AT 0% REACTOR POWER ON 10/11/87. ONE OF THE TWO INCOMING POWER LINES TO 1A START-UP TRANSFORMER FOR OFF-SITE POWER WAS UNDER CLEARANCE FOR RELAY CABINET MODIFICATIONS. AT 1550 HOURS, THE REMAINING INCOMING LINE BREAKER TRIPPED INITIATING A 1A-SA SAFETY BUS BLACKOUT. THE 1A-SA TRAIN SEQUENCER ACTUATED AND 1A-SA EMERGENCY DIESEL GENERATOR (EDG) STARTED AND PROPERLY ASSUMED 1A-SA SAFETY BUS TRAIN LOADS. ALL PLANT SYSTEMS FUNCTIONED AS REQUIRED WITH THE EXCEPTION OF 1A-SA EMERGENCY SERVICE WATER TRAVELING WATER SCREEN WHICH DID NOT START. THE INCOMING BREAKER TRIPPED BECAUSE OF ACCIDENTAL JARRING OF THE PROTECTION RELAYS. THIS WAS DUE TO RELAY PERSONNEL HAMMERING AND DRILLING IN THE RELAY CABINET FOR A MODIFICATION INSTALLATION. THE CORRECTIVE ACTION WAS TO STOP ALL HAMMERING AND DRILLING IN THE RELAY CABINET. OFF-SITE POWER WAS RESTORED AT 1655 HOURS AND 1A-SA DGB SECURED AT 1700 HOURS. SITE MANAGEMENT HAS ISSUED A LETTER STRESSING THE IMPORTANCE OF CAUTIOUS WORK HABITS TO THE RELAY CREW MANAGEMENT WHEN WORKING IN SHNPP SWITCHYARD AND PROCEDURES FOR REVIEWING AND CONTROLLING WORK IN THE SWITCHYARD WILL BE STRENGTHENED. THIS EVENT IS BEING REPORTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(IV) AS AN ENGINEERED SAFEGUARDS FEATURE ACTUATION.

[269] SHEARON HARRIS 1 DOCKET 50-400 LER 87-060
ISOLATION OF RESIDUAL HEAT REMOVAL SYSTEM (RHR) DURING TESTING OF VALVE INTERLOCKS DUE TO TEST EQUIPMENT FAILURE.
EVENT DATE: 101587 REPORT DATE: 111687 NSSS: WE TYPE: PWR
VENDOR: RONAN ENGINEERING COMPANY

(NSIC 207210) ON OCTOBER 15, 1987, WITH THE PLANT IN MODE 5, 130 F AND DEPRESSURIZED, DURING THE PERFORMANCE OF OPERATIONS SURVEILLANCE TEST (OST) 1071, RESIDUAL HEAT REMOVAL (RHR) SUCTION VALVE INTERLOCK TEST, THE ISOLATION VALVES BETWEEN THE REACTOR COOLANT SYSTEM (RCS) AND THE RHR PUMP SUCTIONS CLOSED ON TWO SEPARATE OCCASIONS. THIS RESULTED IN THE UNAVAILABILITY OF THE RHR SYSTEM TO REMOVE RESIDUAL HEAT. THE STEAM GENERATORS WERE AVAILABLE TO SERVE AS THE RCS HEAT SINK. THE CAUSE OF THE FIRST ISOLATION EVENT WAS APPARENTLY DUE TO MOMENTARILY SHORTING THE TEST EQUIPMENT OUTPUT, WHILE THE TEST EQUIPMENT, A RONAN X-85 CALIBRATOR, WAS BEING INSTALLED. THE SECOND EVENT WAS CAUSED BY A BATTERY FAILURE IN THE TEST EQUIPMENT DUE TO EXCESSIVE DISCHARGE. IN BOTH CASES, THE OPERATOR RECOGNIZED THE EVENT AND WAS ABLE TO RESTORE RHR OPERATION IN FIVE MINUTES AND FIFTEEN MINUTES, RESPECTIVELY. REVISIONS TO THE TEST PROCEDURE WHICH WOULD PREVENT RECURRENCE OF THE EVENT ARE UNDER CONSIDERATION.

[270] SHEARON HARRIS 1 DOCKET 50-400 LER 87-062
PERSONNEL ERROR IN SETTING STEAM DUMP CONTROLLER RESULTED IN SAFETY INJECTION, MAIN STEAM ISOLATION, AND REACTOR TRIP WHEN MAIN STEAM ISOLATION VALVES WERE OPENED.
EVENT DATE: 110787 REPORT DATE: 120787 NSSS: WE TYPE: PWR
VENDOR: FLUID COMPONENTS, INC.

(NSIC 207305) ON NOVEMBER 7, 1987, AT 0533, THE PLANT WAS IN MODE 2 AT

APPROXIMATELY 4.5% POWER. THE SAFETY INJECTION SYSTEM WAS ACTUATED, THE MAIN STEAM LINE ISOLATION ACTUATED, AND THE REACTOR TRIPPED DUE TO LOW STEAM LINE PRESSURE ON "A" STEAM GENERATOR. THE LOW PRESSURE WAS DUE TO EXCESSIVE, RAPID, AUTOMATIC CYCLING OF THE STEAM DUMP SYSTEM AFTER THE "A" MAIN STEAM ISOLATION VALVE OPENED. THE CYCLING OF THE STEAM DUMP VALVES WAS CAUSED BY AN INCORRECT SETTING ON THE STEAM HEADER PRESSURE CONTROLLER. THE SAFETY INJECTION WAS TERMINATED, THE PLANT WAS STABILIZED AND RETURNED TO NORMAL OPERATING TEMPERATURE AND PRESSURE. DURING THE EVENT, THE B EMERGENCY DIESEL GENERATOR FUEL OIL TRANSFER PUMP ROOM EXHAUST FAN (E-85A-SB) TRIPPED AFTER STARTING DUE TO A SPURIOUS LOW FLOW SIGNAL. THE ALTERNATE FAN (E-85B-SB) THEN STARTED AND CONTINUED TO RUN AS REQUIRED. THIS FAILURE DID NOT CONTRIBUTE TO THE EVENT. THE EVENT WAS CAUSED BY PERSONNEL ERROR AND A PROCEDURAL DEFICIENCY. CORRECTIVE ACTION INCLUDED PROCEDURE REVISIONS AND ADDITIONAL OPERATOR TRAINING COVERING THIS EVENT. THE PLANT WAS RESTARTED AND ON LINE IN MODE 1 BY 1505 ON NOVEMBER 8, 1987.

[271] SHEARON HARRIS 1 DOCKET 50-400 LER 87-063
PLANT TRIP DUE TO THE LOSS OF MAIN FEEDWATER CAUSED BY A MISPOSITIONED CONDENSATE RECIRCULATION VALVE.
EVENT DATE: 110887 REPORT DATE: 120787 NSSS: WE TYPE: PWR

(NSIC 207306) THE PLANT WAS OPERATING IN MODE 1, POWER OPERATION, AT 22 PERCENT REACTOR POWER ON 11/8/87. THE PLANT WAS IN THE START-UP PROCESS FOLLOWING A SCHEDULED OUTAGE AND WAS PREPARING TO INCREASE POWER FROM 100 MWE TO 150 MWE WITH ONLY THE '1A' FEEDWATER TRAIN IN SERVICE. 1A CONDENSATE PUMP TRIPPED ON LOW DISCHARGE PRESSURE WHICH CAUSED 1A CONDENSATE BOOSTER PUMP AND 1A MAIN FEEDWATER PUMP TO TRIP, WHICH RESULTED IN A TOTAL LOSS OF MAIN FEEDWATER. THE REACTOR AND TURBINE WERE THEN MANUALLY TRIPPED AT 1625 HOURS. THE MAIN STEAM ISOLATION VALVES WERE SHUT IN ORDER TO LIMIT PLANT COOLDOWN AND THE AUXILIARY FEEDWATER SYSTEM ACTUATED TO RESTORE STEAM GENERATOR WATER LEVELS. ALL PLANT SYSTEMS RESPONDED AS REQUIRED. THE IMMEDIATE CAUSE OF THE EVENT WAS THE CONDENSATE RECIRCULATION VALVE WAS IN THE "OPEN" POSITION RATHER THAN "MODULATE" POSITION AS REQUIRED BY NORMAL PLANT OPERATION. THIS CAUSED THE CONDENSATE PUMP AND CONDENSATE BOOSTER PUMP TO BE OPERATING AT NEAR RUN OUT CONDITION AND EVENTUALLY TRIPPED THE CONDENSATE PUMP ON LOW DISCHARGE PRESSURE. THE ROOT CAUSE OF THE EVENT WAS PERSONNEL ERROR AS PLANT OPERATORS WERE NOT FULLY AWARE OF ALL PLANT CONDITIONS.

[272] SHOREHAM DOCKET 50-322 LER 87-029 REV 01
UPDATE ON RESULTS OF LLRT OF PENETRATION SHOWED LEAKAGE THAT, WHEN COMBINED WITH ALL TYPE B AND C PENETRATION LEAKAGES, EXCEEDED THE TECH SPEC LIMIT OF 0.6 LA.
EVENT DATE: 090387 REPORT DATE: 120487 NSSS: GE TYPE: BWR
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 207353) ON SEPTEMBER 8, 1987 AT 1330, IT WAS DETERMINED BY THE MAINTENANCE SECTION THAT THE RESULTS OF A LOCAL LEAK RATE TEST (LLRT) OF CHECK VALVE 1T48*01V-0016A (POST ACCIDENT SAMPLING CONTAINMENT ATMOSPHERE SAMPLE RETURN), SHOWED LEAKAGE THAT, WHEN COMBINED WITH ALL "B" AND "C" PENETRATION LEAKAGES EXCEEDED THE TECH SPEC LIMIT OF 0.6 LA. DURING THE LLRT PROGRAM, THE PLANT WAS IN OPERATIONAL CONDITION 4 (COLD SHUTDOWN) WITH THE MODE SWITCH IN SHUTDOWN AND ALL RODS INSERTED IN THE CORE. SUBSEQUENT TESTING IDENTIFIED SIX ADDITIONAL VALVES WHICH HAD EXCESSIVE LEAKAGE. MAINTENANCE WORK REQUESTS (MWRS) WERE GENERATED TO INVESTIGATE THE LEAKAGE AND TO REPAIR THE VALVES AS NEEDED. THE 1T48 VALVE WAS REPAIRED AND RETESTED WITH ACCEPTABLE RESULTS. THE REMAINING VALVES ARE SCHEDULED TO BE REPAIRED. A SUPPLEMENTAL REPORT WILL BE GENERATED UPON COMPLETION OF THE LLRT AND WILL IDENTIFY ALL VALVES WHICH SIGNIFICANTLY CONTRIBUTED TO THE COMBINED LEAKAGE ALONG WITH ANY CORRECTIVE ACTION TAKEN.

[273] SHOREHAM DOCKET 50-322 LER 87-032
OKONITE TAPE INSULATED 'BACK TO BACK' SPLICES IN PRIMARY CONTAINMENT.
EVENT DATE: 100287 REPORT DATE: 111887 NSSS: GE TYPE: BWR

(NSIC 207226) THIS VOLUNTARY REPORT IS BEING SUBMITTED TO IDENTIFY A CONDITION WHICH EXISTS AT THE SHOREHAM NUCLEAR POWER STATION. ON 9/27/87, AS PART OF LILCO'S CONTINUING INVESTIGATION RESULTING FROM IE INFORMATION NOTICE 86-53 "IMPROPER INSTALLATION OF HEAT SHRINKABLE TUBING", INSPECTIONS OF ELECTRICAL EQUIPMENT IN THE PRIMARY CONTAINMENT WERE INITIATED TO IDENTIFY THOSE INSTANCES WHICH UTILIZED OKONITE TAPE INSULATED BACK-TO-BACK SPLICES. THE PURPOSE OF INCLUDING OKONITE TAPED BACK-TO-BACK SPLICES IN THIS INSPECTION PROGRAM WAS TO DETERMINE THE EXTENT OF THEIR USE AND THEIR INSTALLATION CONDITIONS SINCE THIS SPLICE CONFIGURATION WAS NOT UNIQUELY IDENTIFIED AS QUALIFIED IN THE ENVIRONMENTAL QUALIFICATION (EQ) PACKAGE FOR OKONITE TAPED SPLICES. THE NRC WAS NOTIFIED, VIA ENS, PER LICENSE CONDITION 2.F, TO ALERT THEM OF THE POSSIBLE NON-CONFORMANCE WITH 10CFR50.49 REQUIREMENTS. THE ITEMS IN THE PRIMARY CONTAINMENT AFFECTED BY THE INVESTIGATION ARE SEVEN MOVS AND TWO SOVS. SEVERAL CABLE SPLICE CONFIGURATIONS EXIST AT SHOREHAM, THEREFORE, DURING THE QUALIFICATION PROGRAM IT WAS DETERMINED THAT REPRESENTATIVE CONFIGURATIONS WERE TO BE SELECTED, PREPARED, AND TESTED. SINCE NONE OF THE CLASS 1E SPLICES AT SHOREHAM ARE POSTULATED TO BE EXPOSED TO A SUBMERGENCE CONDITION, NO SPECIAL OR SPECIFIC CONFIGURATION WAS IDENTIFIED AND TESTING WAS DEEMED TO REPRESENT ALL CONFIGURATIONS.

[274] SHOREHAM DOCKET 50-322 LER 87-031
HOURLY FIRE WATCH PATROL LATE BY 20 MINUTES DUE TO PERSONNEL ERROR.
EVENT DATE: 102587 REPORT DATE: 112087 NSSS: GE TYPE: BWR

(NSIC 207198) ON 10/25/87 AN HOURLY FIRE WATCH PATROL WAS LATE 20 MINUTES DUE TO PERSONNEL ERROR. THE PLANT WAS IN OPERATIONAL CONDITION 4 (COLD SHUTDOWN) WITH THE MODE SWITCH IN SHUTDOWN AND ALL CONTROL RODS INSERTED IN THE CORE. SEVERAL STOPS ALONG A PATROL ROUTE WERE INVOLVED IN THE EVENT. AN HOURLY FIRE WATCH PATROL IS REQUIRED THROUGHOUT SEVERAL AREAS OF THE REACTOR BUILDING, TURBINE BUILDING AND RADWASTE BUILDING DUE TO INOPERABLE FIRE RATED ASSEMBLIES. HOURLY PATROLS THROUGH THE REACTOR BUILDING ARE REQUIRED PURSUANT TO TECH SPEC 3.7.8.A AND THE SNPS FIRE PROTECTION PROGRAM. PATROLS THROUGH THE TURBINE AND RADWASTE BUILDINGS ARE REQUIRED PURSUANT TO THE FIRE PROTECTION PROGRAM AND ARE NOT TECH SPEC RELATED. ONE HOUR AND 20 MINUTES HAD ELAPSED BETWEEN 2 CONSECUTIVE PATROLS THEREBY VIOLATING TECH SPECS AND THE FIRE PROTECTION SUPERVISION REVEALED THE DISCREPANCY. FIRE PROTECTION PERSONNEL MADE IMMEDIATE NOTIFICATION TO THE WATCH ENGINEER UPON DISCOVERY. PLANT MANAGEMENT WAS NOTIFIED OF THE EVENT AND THE NRC WAS NOTIFIED IN ACCORDANCE WITH LICENSE CONDITION NPF-36 2.F. THE INDIVIDUAL RESPONSIBLE FOR THE LATE FIRE WATCH PATROL HAS BEEN COUNSELED AND A LETTER HAS BEEN PLACED IN HIS PERSONNEL FILE.

[275] SOUTH TEXAS 1 DOCKET 50-498 LER 87-011
CONTROL ROOM VENTILATION ACTUATION TO RECIRCULATION MODE DUE TO THE FAILURE OF A TOXIC GAS MONITOR PRINTED CIRCUIT BOARD.
EVENT DATE: 101787 REPORT DATE: 111187 NSSS: WE TYPE: PWR
VENDOR: FOXBORO CO., THE

(NSIC 207206) AT APPROXIMATELY 0700 ON OCTOBER 17, 1987 WITH UNIT 1 IN MODE 5, AN AUTO-ACTUATION OF THE CONTROL ROOM VENTILATION TO RECIRCULATION MODE OCCURRED AS A RESULT OF A MALFUNCTION OF A TOXIC GAS MONITOR. THE CONTROL ROOM VENTILATION ACTUATION TO RECIRCULATION MODE IS AN ENGINEERED SAFETY FEATURE (ESF). THE CONTROL ROOM OPERATORS VERIFIED THE RECIRCULATION MODE DAMPER LINEUP AND INITIATED AN INVESTIGATION OF THE EVENT. THE INVESTIGATION DETERMINED THAT THE EVENT WAS CAUSED BY A FAILURE OF A TOXIC GAS MONITOR PRINTED CIRCUIT BOARD. ON OCTOBER 19, 1987 THE FAILED PRINTED CIRCUIT BOARD WAS REPLACED AND THE SYSTEM WAS

RESTORED TO NORMAL OPERATION. AN EVALUATION IS BEING PERFORMED TO INVESTIGATE MODIFYING THE ESF ACTUATION LOGIC TO REQUIRE BOTH TOXIC GAS MONITORS TO BE INOPERABLE (MONITOR MALFUNCTION OR LOSS OF POWER) TO INITIATE AN ESF ACTUATION.

[276] SOUTH TEXAS 1 DOCKET 50-498 LER 87-012
HIGH HEAD SAFETY INJECTION SYSTEM INOPERABLE DUE TO PERSONNEL ERROR.
EVENT DATE: 110287 REPORT DATE: 121187 NSSS: WE TYPE: PWR

(NSIC 207386) PRIOR TO INITIAL CRITICALITY ON NOVEMBER 2, 1987 AT APPROXIMATELY 0510 HOURS WITH THE UNIT IN MODE 4, OPERATING PERSONNEL RECOGNIZED THE THREE HIGH HEAD SAFETY INJECTION (HHSI) COLD LEG INJECTION VALVES WERE CLOSED. THE TECHNICAL SPECIFICATIONS REQUIRE TWO OF THE THREE VALVES TO BE OPEN IN MODE 4. THE VALVES WERE IMMEDIATELY OPENED. JUST PRIOR TO ENTRY INTO MODE 4, PLANT OPERATIONS MANAGEMENT REQUESTED AN EXTRA VERIFICATION OF CORRECT MODE 4 ALIGNMENT OF SAFETY INJECTION VALVES. THE PROCEDURE USED FOR THE VALVE ALIGNMENT WAS WRITTEN FOR MODE 5 AND PERFORMANCE OF THE PROCEDURE RESULTED IN THE VALVES BEING CLOSED. THE MISALIGNMENT WAS NOT DISCOVERED UNTIL THE PLANT WAS IN MODE 4. ADDITIONALLY, OPERATING PERSONNEL DID NOT RECOGNIZE THE ALARMS ON THE ENGINEERED SAFETY FEATURE STATUS MONITORING SYSTEM IDENTIFYING THE VALVES WERE IN THE CLOSED POSITION. CORRECTIVE ACTIONS TAKEN INCLUDED DETAILED DISCUSSIONS WITH PLANT OPERATORS AND A REVISION TO THE PLANT HEATUP PROCEDURE. PRIOR TO INITIAL CRITICALITY A LIST OF MINIMUM EQUIPMENT REQUIRED FOR EACH MODE WILL BE ADDED TO THE SHIFT RELIEF AND TURNOVER LOGS TO ASSIST THE OPERATORS IN RECOGNIZING ABNORMAL CONDITIONS. THIS EVENT WAS CAUSED BY PERSONNEL ERROR.

[277] SOUTH TEXAS 1 DOCKET 50-498 LER 87-013
CONTROL ROOM VENTILATION ACTUATION TO RECIRCULATION MODE DUE TO INADVERTENT SWITCH OPERATION.
EVENT DATE: 110287 REPORT DATE: 120287 NSSS: WE TYPE: PWR

(NSIC 207314) AT APPROXIMATELY 1022 HOURS ON NOVEMBER 2, 1987 WITH THE UNIT IN MODE 4 (INITIAL STARTUP TESTING) AN AUTO-ACTUATION OF THE CONTROL ROOM VENTILATION SYSTEM INTO RECIRCULATION MODE OCCURRED AS A RESULT OF A MOMENTARY LOSS OF POWER TO THE TOXIC GAS MONITORING SYSTEM. A CONSTRUCTION ENGINEER INADVERTENTLY TURNED OFF A DISCONNECT SWITCH WHICH PROVIDED POWER TO THE TOXIC GAS MONITORS LOCATED ADJACENT TO ONE WHICH WAS BEING INSTALLED. REALIZING THE ERROR THE ENGINEER IMMEDIATELY TURNED THE SWITCH TO ITS PREVIOUS LOCATION. THE SWITCH WHICH WAS INADVERTENTLY OPERATED HAD BEEN INSTALLED WITH ITS HANDLE IN A DIFFERENT ORIENTATION AND WAS ALMOST TOUCHING THE NEW SWITCH. CONSTRUCTION PERSONNAL HAVE BEEN REINSTRUCTED REGARDING THIS EVENT AND ADMINISTRATIVE CONTROL MEASURES TO BE USED IN THE FUTURE. A MODIFICATION TO THE SWITCH WHICH WAS INADVERTENTLY OPERATED IS BEING CONSIDERED. A PLANT OPERABILITY TASK FORCE HAS BEEN INITIATED TO EVALUATE SIMILAR PLANT CHARACTERISTICS TO CONSIDER USE OF PROTECTIVE GUARDS AND CAUTION SIGNS.

[278] ST. LUCIE 1 DOCKET 50-335 LER 85-012 REV 02
UPDATE ON ISI SNUBBER INSPECTION FAILURES.
EVENT DATE: 121985 REPORT DATE: 060987 NSSS: CE TYPE: PWR
VENDOR: ITT GRINNELL
PACIFIC SCIENTIFIC COMPANY

(NSIC 207157) WHILE THE UNIT WAS SHUTDOWN FOR A SCHEDULED REFUELING OUTAGE FROM OCTOBER 20 TO DECEMBER 24, 1985, VISUAL EXAMINATIONS AND FUNCTIONAL TESTING OF SNUBBERS WERE PERFORMED PER TECH. SPEC. 4.7.10. SOME SNUBBERS FAILED THESE TESTS. THE SNUBBERS FAILED DUE TO INSTALLATION/MAINTENANCE ERRORS AND DESIGN PROBLEMS. THIS EVENT IS CONSIDERED REPORTABLE UNDER SEVERAL CATEGORIES DUE TO THE VARIOUS SYSTEMS AFFECTED. HOWEVER, ENGINEERING EVALUATION INDICATED THAT NO PIPING OR COMPONENTS HAVE BEEN ADVERSELY AFFECTED BY THESE FAILURES. ALL FAILED

SNUBBERS WERE REPAIRED, MODIFIED OR REPLACED. THE SNUBBER MAINTENANCE PROCEDURES AND TRAINING IS BEING UPGRADED. THE PSA-0.25 (250 LB LOAD) SNUBBER INTERNALS WERE ANALYZED FOR PROPER HARDNESS AND WERE FOUND TO BE WITHIN THE TENSILE STRENGTH SPECIFICATIONS. ALL PSA-0.25 (250 LB LOAD) SNUBBERS HAVE BEEN REPLACED WITH EITHER PSA-0.25(350 LB LOAD) OR PSA-1.0 SNUBBERS.

[279] ST. LUCIE 1 DOCKET 50-335 LER 87-015
UNIT MODE CHANGE WITHOUT SHIELD BUILDING INTEGRITY DUE TO A PROCEDURAL ERROR.
EVENT DATE: 101487 REPORT DATE: 111687 NSSS: CE TYPE: PWR

(NSIC 207134) ON OCTOBER 14, 1987, UNIT 1 VIOLATED TECH SPECS BY ENTERING MODE 4 (<325 DEG.) WITHOUT SHIELD BUILDING INTEGRITY. THE ERROR WAS DISCOVERED 8.5 HOURS LATER BY A LICENSED OPERATOR PERFORMING A SURVEILLANCE ON THE SHIELD BUILDING VENTILATION SYSTEM (SBVS). THE ROOT CAUSE OF THE EVENT WAS DUE TO INADEQUATE PROCEDURAL CONTROLS OF THE SHIELD BUILDING DOORS ONCE THEY HAVE BEEN CHECKED AS BEING CLOSED. CONTRIBUTING CAUSES WERE COGNITIVE PERSONNEL ERRORS BY LICENSED AND NON-LICENSED OPERATORS WHEN A SECOND CHECK SHEET PERFORMED TO CHECK SHIELD BUILDING INTEGRITY WAS IMPROPERLY REVIEWED. OPERATING PROCEDURE 1-0030120 "PRESTART CHECK-OFF LIST" WILL BE REVISED TO INCLUDE STRICTER CONTROLS ON THE SHIELD BUILDING DOORS. OPERATING PERSONNEL HAVE BEEN COUNSELED ON PROPER PAPERWORK REVIEW. TRAINING DEPARTMENT WILL REVIEW THE EVENT FOR APPROPRIATE TRAINING REQUIREMENTS AND METHODS. THE OPENED DOOR DID NOT APPRECIABLY REDUCE THE CONTAINMENTS PROTECTION FROM EXTERNAL MISSILES OR THE BIOLOGICAL SHIELDING TO PLANT PERSONNEL SINCE THE DOOR IS PROTECTED BY VARIOUS EXTERNAL CONCRETE STRUCTURES. DUE TO CONSERVATIVE ASSUMPTIONS MADE IN THE UNIT #1 FUSAR AND THE ABILITY OF THE SBVS TO STILL PERFORM ITS REQUIRED SAFETY FUNCTION, THE OPENED DOOR WOULD NOT HAVE RESULTED IN A SIGNIFICANT INCREASE OF FISSION PRODUCTS DURING A MAJOR HYPOTHETICAL ACCIDENT.

[280] ST. LUCIE 1 DOCKET 50-335 LER 87-016
REACTOR TRIP DUE TO LOSS OF CONDENSATE PUMP DUE TO INADEQUATE PROCEDURE AND SPURIOUS HIGH STARTUP RATE RPS ACTUATION DUE TO ELECTRICAL SPIKE.
EVENT DATE: 102987 REPORT DATE: 113087 NSSS: CE TYPE: PWR

(NSIC 207265) ON OCTOBER 29, 1987, AT 0334 HOURS, ST. LUCIE UNIT ONE WAS IN MODE 1 REACTOR POWER >5%, AVERAGE COOLANT TEMPERATURE >325F), WHEN THE UNIT TRIPPED ON LOW STEAM GENERATOR LEVEL DUE TO THE LOSS OF THE 1A CONDENSATE PUMP. FEEDWATER WAS RESTORED TO THE STEAM GENERATORS BY THE MANUAL STARTING OF THE AUXILIARY FEEDWATER PUMPS. ALL SYSTEMS FUNCTIONED NORMALLY. AT 0542 HOURS, THE UNIT WAS IN MODE 3 (REACTOR NOT CRITICAL, RCS TEMPERATURE >325F) PREPARING TO RETURN TO FULL POWER OPERATION, WITH THE C CHANNEL OF WIDE RANGE NUCLEAR INSTRUMENTATION ON THE REACTOR PROTECTION SYSTEM (RPS) PLACED IN TRIP CONDITION IN ACCORDANCE WITH TECHNICAL SPECIFICATION 3.3.1.1. THE LICENSED REACTOR CONTROL OPERATOR HAD JUST BEGUN PULLING THE INITIAL GROUP OF SHUTDOWN CONTROL RODS WHEN A SPURIOUS SIGNAL WAS RECEIVED ON THE B CHANNEL WIDE RANGE NUCLEAR INSTRUMENTATION AND THE REACTOR WAS TRIPPED ON HIGH STARTUP RATE. ALL SYSTEMS FUNCTIONED NORMALLY. THE ROOT CAUSE OF THE LOSS OF THE CONDENSATE PUMP WAS DETERMINED TO BE AN INADEQUATE OPERATING PROCEDURE. THE ROOT CAUSE OF THE HIGH STARTUP RATE TRIP WAS DETERMINED TO BE ELECTRICAL NOISE.

[281] ST. LUCIE 2 DOCKET 50-389 LER 86-011 REV 01
UPDATE ON BOTH DIESEL GENERATORS SIMULTANEOUSLY OUT OF SERVICE DUE TO ONE PERSONNEL ERROR AND ONE COMPONENT FAILURE.
EVENT DATE: 070986 REPORT DATE: 121687 NSSS: CE TYPE: PWR
VENDOR: O & M MANUFACTURING COMPANY
WOODWARD GOVERNOR COMPANY

(NSIC 207411) ON JULY 9, 1986, WHILE UNIT 2 WAS AT FULL POWER, THE 2A EMERGENCY

DIESEL GENERATOR (D/G) WAS TAKEN OUT OF SERVICE DUE TO FAILURE TO MEET THE REQUIRED START TIME DURING A NORMAL SURVEILLANCE TEST RUN. THE REDUNDANT 2B D/G WAS SUBSEQUENTLY STARTED, CAME UP TO RATED VOLTAGE AND FREQUENCY WITHIN THE REQUIRED TIME, BUT WAS DECLARED OUT OF SERVICE AS ONE OF THE COOLING FANS WAS RUBBING ITS SHROUD. REPAIRS WERE MADE TO BOTH DIESELS; BOTH WERE RETURNED TO SERVICE WITHIN THE APPLICABLE TECHNICAL SPECIFICATION TIME THE FAILURE OF THE 2A D/G WAS CAUSED BY EXCESSIVE TIGHTENING OF A FRICTION CLUTCH LOCKNUT. ENGINEERING EVALUATION OF THE 2B D/G FAN EVENT REVEALED THE PRESENCE OF RESONANT FREQUENCIES IN THE 12 CYLINDER ENGINE, WHICH RESULTED IN THE FLAPPING OF THE FAN DRIVE BELTS. THE VIBRATION CAUSED BY THE FLAPPING BELTS RESULTED IN THE LOOSENING OF THE FAN HUB SET SCREWS. CORRECTIVE ACTION INCLUDED: 1. INSPECTION OF TORQUE VALUES IN ALL FRICTION CLUTCH LOCKNUTS. 2. INSTRUCTING ALL PLANT D/G MAINTENANCE PERSONNEL TO CONTACT VENDORS FOR INFORMATION REGARDING COMPONENT ADJUSTMENT WHEN NECESSARY. 3. A TORSIONALLY SOFT COUPLING HAS BEEN INSTALLED IN THE FAN DRIVE SYSTEMS ON THE 12 AND 16 CYLINDER ENGINES.

[282] ST. LUCIE 2 DOCKET 50-389 LER 87-007
REACTOR TRIP ON LOSS OF LOAD CAUSED BY MAIN GENERATOR EXCITER BEARING FAILURE DUE TO PERSONNEL ERROR.
EVENT DATE: 112587 REPORT DATE: 122187 NSSS: CE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 207466) ON 25 NOVEMBER 1987, ST. LUCIE UNIT TWO WAS OPERATING AT 50 PERCENT POWER STEADY STATE IN MODE 1. AT 2331 HOURS, THE REACTOR TRIPPED ON LOSS OF LOAD DUE TO A TURBINE TRIP. THE TURBINE TRIPPED ON A MAIN GENERATOR LOCKOUT. THE FAILURE OF THE MAIN GENERATOR EXCITER BEARING CAUSED THE ARMATURE OF THE PERMANENT MAGNET GENERATOR (PMG) TO GRIND INTO ITS STATOR. WHEN THIS OCCURRED, THE PMG DISCONTINUED SUPPLYING VOLTAGE TO THE EXCITER OF THE MAIN GENERATOR. THE TRIP WAS UNCOMPLICATED AND THE UNIT WAS QUICKLY STABILIZED IN MODE 3, HOT STANDBY. THE ROOT CAUSE OF THE EVENT WAS A PERSONNEL ERROR DURING THE PERFORMANCE OF THE WEEKLY GENERATOR EXCITER GROUND CHECK ON WHICH A PROLONGED GROUND RESULTED IN THE FAILURE OF THE GENERATOR EXCITER BEARING. THE FOLLOWING CORRECTIVE ACTIONS HAVE BEEN IMPLEMENTED: REPLACED THE PMG AND THE EXCITER BEARING, REMOVED THE EXCITER BEARING THERMOCOUPLE TO PREVENT FUTURE GROUNDING OCCURRENCE OF THIS TYPE, PROCEDURES HAVE BEEN REVISED TO ENSURE ANY FUTURE GROUNDS ARE RECOGNIZED, AND TRAINING WAS GIVEN TO ALL ELECTRICAL MAINTENANCE PERSONNEL FOR BETTER UNDERSTANDING OF THE EXCITER GROUND CHECK.

[283] ST. LUCIE 2 DOCKET 50-389 LER 87-008
OIL FIRE IN REACTOR COOLANT PUMP DUE TO OIL LEAK FROM LOOSE PACKING GLAND ON INSTRUMENT ISOLATION VALVE.
EVENT DATE: 112887 REPORT DATE: 122287 NSSS: CE TYPE: PWR
VENDOR: ALLIS CHALMERS
BYRON JACKSON PUMPS, INC.

(NSIC 207467) AT 1010 HOURS ON NOVEMBER 28, 1987, WHILE IN MODE 3 PREPARING TO RETURN TO FULL POWER OPERATION, ST. LUCIE UNIT 2 EXPERIENCED AN OIL FIRE AT THE 2A1 REACTOR COOLANT PUMP (RCP). THE FLAMES WERE INITIALLY EXTINGUISHED BY THE PLANT FIRE FIGHTING TEAM WITHIN MINUTES, BUT SOME REFLASH WAS EXPERIENCED. THE FIRE WAS COMPLETELY EXTINGUISHED BY APPROXIMATELY 1030 HOURS. AN UNUSUAL EVENT WAS DECLARED AT 1023 HOURS, AND WAS TERMINATED AT 1220 HOURS, WHEN INSULATION AT THE BASE OF THE RCP HAD BEEN REMOVED TO ENSURE ALL POSSIBILITY OF REFLASH HAD BEEN ELIMINATED. THE ROOT CAUSE OF THE FIRE WAS DETERMINED TO BE OIL LEAKAGE THROUGH A PACKING GLAND WHICH HAD APPARENTLY VIBRATED LOOSE ON A PRESSURE INSTRUMENT ISOLATION VALVE ON THE DISCHARGE HEADER OF THE 2A1 RCP OIL LIFT PUMPS. CORRECTIVE ACTIONS INCLUDED VALVE REPAIR, OIL COLLECTION SYSTEM INSPECTIONS AND REPAIR, PROCEDURE CHANGES, SNUBBER MODIFICATION, AND AN ENGINEERING EVALUATION.

[284] SUMMER 1 DOCKET 50-395 LER 87-025
ENVIRONMENTAL QUALIFICATION OF 600 VOLT TAPED WIRING SPLICES.
EVENT DATE: 101487 REPORT DATE: 112087 NSSS: WE TYPE: PWR

(NSIC 207218) IN REVIEWING THE ENVIRONMENTAL QUALIFICATION (EQ) FINDINGS AT ANOTHER NUCLEAR PLANT, PROBLEMS WITH THE CONFIGURATION OF HARSH ENVIRONMENT QUALIFIED, TAPED ELECTRICAL SPLICES WERE NOTED. CONSEQUENTLY A REVIEW OF ENGINEERING SPECIFICATIONS AND INSTALLATION PRACTICES FOR THE VIRGIL C. SUMMER NUCLEAR STATION WAS INITIATED. ON 10/14/87, IT WAS CONCLUDED THAT SUCH PROBLEMS MIGHT ALSO EXIST IN THE VIRGIL C. SUMMER NUCLEAR STATION. IN AN EVALUATION OF POTENTIAL INOPERABILITY INVOLVING THE TAPED SPLICES, IT WAS DETERMINED THAT NO SYSTEM COVERED BY TECHNICAL SPECIFICATIONS WOULD BE RENDERED INOPERABLE WHEN REQUIRED AS A RESULT OF ADVERSE EFFECTS BY DESIGN BASIS ACCIDENTS ON THE SUBJECT TAPED SPLICES. THIS EVENT WAS CAUSED BY THE INSTALLATION OF 600V ELECTRICAL SPLICES UTILIZING A KERITE TAPED BACK TO BACK AND/OR "V" CONFIGURATION NOT PREVIOUSLY TESTED FOR ENVIRONMENTAL QUALIFICATION. A JUSTIFICATION FOR CONTINUED OPERATION (JCO) WAS DEVELOPED FOR THE EXISTING TAPED SPLICE CONFIGURATION AND APPROVED ON 10/16/87. A SYSTEMATIC PLAN WAS DEVELOPED TO INSPECT SPLICES TO ASSURE THE "CROTCH" OF THE BACK TO BACK AND/OR "V "SPLICE WAS ADEQUATELY FILLED WITH INSULATING MATERIAL AND RETAPED PROPERLY WHERE REQUIRED. REACTOR POWER WAS REDUCED TO 30% TO FACILITATE INSPECTION/RETAPING WITHIN CONTAINMENT. THE RELATED PLANT PROCEDURE WAS REVISED.

[285] SUMMER 1 DOCKET 50-395 LER 87-027
REACTOR TRIP RESULTING FROM POWER LOSS TO PROCESS RACK PANEL XPN 7008.
EVENT DATE: 102987 REPORT DATE: 112587 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 207342) ON OCTOBER 29, 1987, AT 0313 HOURS WITH THE PLANT AT 100% POWER, BOTH THE PRIMARY AND BACKUP POWER SUPPLIES TO ONE OF THE WESTINGHOUSE 7300 SYSTEM PROCESS RACK PANELS FAILED. THIS RESULTED IN A LOSS OF VARIOUS INSTRUMENTATION AND CONTROL FUNCTIONS INCLUDING FEEDWATER CONTROLS, PRESSURIZER PRESSURE AND LEVEL CONTROLS, AND STEAM DUMP CONTROLS. MANUAL CONTROL OF THE FEEDWATER SYSTEM WAS ATTEMPTED, BUT DUE TO THE NATURE OF THE CONTROL FAILURES, RECOVERY WAS NOT POSSIBLE. THE PLANT TRIPPED ON LOW STEAM GENERATOR "C" LEVEL COINCIDENT WITH STEAM FLOW/FEED FLOW MISMATCH. A PRESSURIZER POWER OPERATED RELIEF VALVE LIFTED, THE STEAMLINE POWER OPERATED RELIEF VALVES LIFTED, AND THE MOTOR DRIVEN AND TURBINE DRIVEN EMERGENCY FEEDWATER PUMPS STARTED. WITHIN APPROXIMATELY 13 MINUTES I&C PERSONNEL HAD RESTORED POWER TO THE PANEL AND CONTROL SYSTEMS RETURNED TO NORMAL. THE PLANT, UTILIZING THE REESTABLISHED CONTROL FUNCTIONS, RECOVERED NORMALLY FROM THE TRIP. THE CAUSE OF THE EVENT WAS ATTRIBUTED TO A FAILED CAPACITOR ON A STEAM DUMP CONTROL SIGNAL CONVERTER CARD IN ONE OF THE PROCESS RACK PANELS. THIS CAPACITOR SHORTED TO GROUND CAUSING THE BREAKERS TO BOTH THE PRIMARY AND BACKUP POWER SUPPLIES FOR THE PROCESS RACK PANEL TO TRIP.

[286] SUMMER 1 DOCKET 50-395 LER 87-028
REACTOR TRIP DUE TO POWER LOSS TO SOURCE RANGE NUCLEAR INSTRUMENTATION.
EVENT DATE: 103087 REPORT DATE: 112587 NSSS: WE TYPE: PWR

(NSIC 207343) ON OCTOBER 30, 1987, AT 0903 HOURS WITH THE PLANT IN MODE 3 AND THE SHUTDOWN BANKS WITHDRAWN, A REACTOR TRIP OCCURRED AND BOTH SETS OF SHUTDOWN BANKS INSERTED DUE TO THE LOSS OF POWER TO A SOURCE RANGE NUCLEAR INSTRUMENTATION DRAWER. THIS LOSS OF POWER OCCURRED WHEN THE WRONG TYPE OF LIGHT BULBS WERE INSERTED IN BOTH THE INSTRUMENT AND CONTROL POWER STATUS INDICATING LIGHTS IN ONE OF THE SOURCE RANGE DRAWERS IN THE CONTROL ROOM. THE INCORRECT LIGHT BULBS CAUSED BOTH THE INSTRUMENT AND CONTROL POWER FUSES TO BLOW WHEN THE LIGHT BULB HOLDER ASSEMBLY WAS REINSERTED. SUBSEQUENT INVESTIGATIONS DETERMINED THAT THE ORIGINAL BULBS WERE NEON TYPE BULBS WITH AN INFINITE RESISTANCE, WHILE THE BULBS USED AS REPLACEMENTS - THOUGH SIMILAR LOOKING - WERE INCANDESCENT TYPE BULBS

WITH AN OHMIC VALUE OF APPROXIMATELY 176 OHMS. TO PREVENT RECURRENCE, THE LICENSEE IS DEVELOPING A PROGRAM TO ADDRESS CHANGEOUT OF LIGHT BULBS IN THE CONTROL ROOM.

[287] SUMMER 1 DOCKET 50-395 LER 87-026
MISSED SURVEILLANCE OF FIRE SERVICE DIESEL BATTERY DUE TO PERSONNEL ERROR.
EVENT DATE: 110387 REPORT DATE: 120187 NSSS: WE TYPE: PWR

(NSIC 207468) ON 11/04/87, DURING A POST TEST REVIEW IT WAS DISCOVERED THAT THE WEEKLY FIRE SERVICE DIESEL BATTERY TEST EXCEEDED THE GRACE PERIOD ALLOWED BY TECHNICAL SPECIFICATIONS SECTION 4.0.2. THE WEEKLY SURVEILLANCE TEST, STP-505.001, WAS PERFORMED ON THE MORNING OF 11/02/87 WHICH WAS BEYOND THE ALLOWABLE EXTENSION DATE OF 11/01/87. SURVEILLANCE TEST PROCEDURE 505-001 IS PERFORMED ON A WEEKLY FREQUENCY TO MEET THE REQUIREMENTS OF TECHNICAL SPECIFICATIONS SECTION 4.7.9.1.3A, " FIRE SUPPRESSION WATER SYSTEM." THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR. THERE WERE NO ADVERSE CONSEQUENCES DUE TO THE FOLLOWING: THE ELECTRICAL DRIVEN FIRE PUMP REMAINED OPERABLE, BATTERY VOLTAGES WERE RECORDED ONCE PER TWELVE (12) HOURS AND THE DIESEL FIRE PUMP WAS DEMONSTRATED OPERABLE ON 11/01/87. THIS EVENT WAS DISCUSSED DURING THE MAINTENANCE WEEKLY SAFETY MEETING ON 11/04/87, AND EMPHASIS WAS PLACED ON TIMELY COMPLETION OF SURVEILLANCE TESTS. THE FIRE SERVICE DIESEL WEEKLY BATTERY TEST WAS PERFORMED AGAIN ON THE EVENT DISCOVERY DATE TO REESTABLISH THE TEST FREQUENCY FOR THE MONDAY NIGHT SHIFT. THE ELECTRICAL SHIFT LOG FOR MONDAYS WAS REVISED UNDER MAINTENANCE SPECIAL INSTRUCTION MSI-01 PROVIDING A CHECK OFF BLOCK FOR THE BATTERY TEST, STP-505.001.

[288] SURRY 1 DOCKET 50-280 LER 87-028
EDG OUTPUT BREAKER PROTECTION CIRCUIT DESIGN DEFICIENCY.
EVENT DATE: 102187 REPORT DATE: 112087 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)

(NSIC 207183) ON OCTOBER 21, 1987, WITH BOTH UNITS OPERATING AT 100% POWER, IT WAS REPORTED THAT THE EMERGENCY DIESEL GENERATORS (EIIS-DG) (EDG) OUTPUT BREAKER (EIIS-52) PROTECTION CIRCUITS CONTAINED INSTANTANEOUS OVERCURRENT TRIPPING RELAYS (EIIS-RLY). THE RELAY SETTINGS WERE SUCH THAT A FAULT ON AN INDIVIDUAL BUS LOAD COULD CAUSE THE EDG OUTPUT BREAKER TO TRIP BEFORE THE LOAD BREAKER WOULD OPEN TO ISOLATE THE FAULT. THIS CONDITION EXISTED ON ALL FOUR (4) EDG OUTPUT BREAKERS WHICH FUNCTION AS THE EMERGENCY SUPPLY (FEEDER) BREAKERS FOR THE EMERGENCY BUSES. THE CAUSE OF THIS CONDITION HAS BEEN DETERMINED TO BE A DESIGN DEFICIENCY. THE INSTANTANEOUS OVERCURRENT TRIP RELAY SETTINGS WERE INCREASED ON ALL FOUR (4) EMERGENCY SUPPLY BREAKERS TO A SETTING THAT IS HIGHER THAN A LOAD FAULT CURRENT PLUS THE BUS CURRENT TO ENSURE AN INSTANTANEOUS OVERCURRENT TRIP WILL NOT OCCUR. THE INSTANTANEOUS OVERCURRENT TRIP RELAYS WILL BE REMOVED FROM THE EMERGENCY SUPPLY BREAKERS PROTECTION CIRCUIT DURING THE NEXT UNIT REFUELING OUTAGES.

[289] SURRY 1 DOCKET 50-280 LER 87-027
SPURIOUS ENGINEERED SAFETY FEATURE ACTUATION DUE TO WATER INTRUSION INTO CONTROL PANEL.
EVENT DATE: 102787 REPORT DATE: 112587 NSSS: WE TYPE: PWR
VENDOR: SOUTHWESTERN ENGINEERING COMPANY

(NSIC 207182) ON OCTOBER 27, 1987, UNIT 1 WAS OPERATING AT 100% POWER. AT 1657 HOURS, AN ELECTRICAL SHORT IN THE TRAIN "A" TURBINE BUILDING FLOOD CONTROL (EIIS-VK) PANEL (TBFT-A) CAUSED A SPURIOUS COMPLETION OF THE FLOOD PROTECTION CIRCUIT. THIS CAUSED TWO OF THE FOUR CONDENSER CIRCULATING WATER INLET VALVES, MOV-CW-106A AND C (EIIS-BS-V), TO BEGIN AUTOMATICALLY CLOSING AS DESIGNED. OPERATIONS PERSONNEL, RESPONDING TO A DECREASING CONDENSER VACUUM, IDENTIFIED THE AFFECTED VALVES, VERIFIED THEIR ACTUATION WAS SPURIOUS, AND FULLY OPENED THEM. A

[284] SUMMER 1 DOCKET 50-395 LER 87-025
ENVIRONMENTAL QUALIFICATION OF 600 VOLT TAPED WIRING SPLICES.
EVENT DATE: 101487 REPORT DATE: 112087 NSSS: WE TYPE: PWR

(NSIC 207218) IN REVIEWING THE ENVIRONMENTAL QUALIFICATION (EQ) FINDINGS AT ANOTHER NUCLEAR PLANT, PROBLEMS WITH THE CONFIGURATION OF HARSH ENVIRONMENT QUALIFIED, TAPED ELECTRICAL SPLICES WERE NOTED. CONSEQUENTLY A REVIEW OF ENGINEERING SPECIFICATIONS AND INSTALLATION PRACTICES FOR THE VIRGIL C. SUMMER NUCLEAR STATION WAS INITIATED. ON 10/14/87, IT WAS CONCLUDED THAT SUCH PROBLEMS MIGHT ALSO EXIST IN THE VIRGIL C. SUMMER NUCLEAR STATION. IN AN EVALUATION OF POTENTIAL INOPERABILITY INVOLVING THE TAPED SPLICES, IT WAS DETERMINED THAT NO SYSTEM COVERED BY TECHNICAL SPECIFICATIONS WOULD BE RENDERED INOPERABLE WHEN REQUIRED AS A RESULT OF ADVERSE EFFECTS BY DESIGN BASIS ACCIDENTS ON THE SUBJECT TAPED SPLICES. THIS EVENT WAS CAUSED BY THE INSTALLATION OF 600V ELECTRICAL SPLICES UTILIZING A KERITE TAPED BACK TO BACK AND/OR "V" CONFIGURATION NOT PREVIOUSLY TESTED FOR ENVIRONMENTAL QUALIFICATION. A JUSTIFICATION FOR CONTINUED OPERATION (JCO) WAS DEVELOPED FOR THE EXISTING TAPED SPLICE CONFIGURATION AND APPROVED ON 10/16/87. A SYSTEMATIC PLAN WAS DEVELOPED TO INSPECT SPLICES TO ASSURE THE "CROTCH" OF THE BACK TO BACK AND/OR "V "SPLICE WAS ADEQUATELY FILLED WITH INSULATING MATERIAL AND RETAPED PROPERLY WHERE REQUIRED. REACTOR POWER WAS REDUCED TO 30% TO FACILITATE INSPECTION/RETAPING WITHIN CONTAINMENT. THE RELATED PLANT PROCEDURE WAS REVISED.

[285] SUMMER 1 DOCKET 50-395 LER 87-027
REACTOR TRIP RESULTING FROM POWER LOSS TO PROCESS RACK PANEL XPN 7008.
EVENT DATE: 102987 REPORT DATE: 112587 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 207342) ON OCTOBER 29, 1987, AT 0313 HOURS WITH THE PLANT AT 100% POWER, BOTH THE PRIMARY AND BACKUP POWER SUPPLIES TO ONE OF THE WESTINGHOUSE 7300 SYSTEM PROCESS RACK PANELS FAILED. THIS RESULTED IN A LOSS OF VARIOUS INSTRUMENTATION AND CONTROL FUNCTIONS INCLUDING FEEDWATER CONTROLS, PRESSURIZER PRESSURE AND LEVEL CONTROLS, AND STEAM DUMP CONTROLS. MANUAL CONTROL OF THE FEEDWATER SYSTEM WAS ATTEMPTED, BUT DUE TO THE NATURE OF THE CONTROL FAILURES, RECOVERY WAS NOT POSSIBLE. THE PLANT TRIPPED ON LOW STEAM GENERATOR "C" LEVEL COINCIDENT WITH STEAM FLOW/FEED FLOW MISMATCH. A PRESSURIZER POWER OPERATED RELIEF VALVE LIFTED, THE STEAMLINE POWER OPERATED RELIEF VALVES LIFTED, AND THE MOTOR DRIVEN AND TURBINE DRIVEN EMERGENCY FEEDWATER PUMPS STARTED. WITHIN APPROXIMATELY 13 MINUTES I&C PERSONNEL HAD RESTORED POWER TO THE PANEL AND CONTROL SYSTEMS RETURNED TO NORMAL. THE PLANT, UTILIZING THE REESTABLISHED CONTROL FUNCTIONS, RECOVERED NORMALLY FROM THE TRIP. THE CAUSE OF THE EVENT WAS ATTRIBUTED TO A FAILED CAPACITOR ON A STEAM DUMP CONTROL SIGNAL CONVERTER CARD IN ONE OF THE PROCESS RACK PANELS. THIS CAPACITOR SHORTED TO GROUND CAUSING THE BREAKERS TO BOTH THE PRIMARY AND BACKUP POWER SUPPLIES FOR THE PROCESS RACK PANEL TO TRIP.

[286] SUMMER 1 DOCKET 50-395 LER 87-028
REACTOR TRIP DUE TO POWER LOSS TO SOURCE RANGE NUCLEAR INSTRUMENTATION.
EVENT DATE: 103087 REPORT DATE: 112587 NSSS: WE TYPE: PWR

(NSIC 207343) ON OCTOBER 30, 1987, AT 0903 HOURS WITH THE PLANT IN MODE 3 AND THE SHUTDOWN BANKS WITHDRAWN, A REACTOR TRIP OCCURRED AND BOTH SETS OF SHUTDOWN BANKS INSERTED DUE TO THE LOSS OF POWER TO A SOURCE RANGE NUCLEAR INSTRUMENTATION DRAWER. THIS LOSS OF POWER OCCURRED WHEN THE WRONG TYPE OF LIGHT BULBS WERE INSERTED IN BOTH THE INSTRUMENT AND CONTROL POWER STATUS INDICATING LIGHTS IN ONE OF THE SOURCE RANGE DRAWERS IN THE CONTROL ROOM. THE INCORRECT LIGHT BULBS CAUSED BOTH THE INSTRUMENT AND CONTROL POWER FUSES TO BLOW WHEN THE LIGHT BULB HOLDER ASSEMBLY WAS REINSERTED. SUBSEQUENT INVESTIGATIONS DETERMINED THAT THE ORIGINAL BULBS WERE NEON TYPE BULBS WITH AN INFINITE RESISTANCE, WHILE THE BULBS USED AS REPLACEMENTS - THOUGH SIMILAR LOOKING - WERE INCANDES CENT TYPE BULBS

WITH AN OHMIC VALUE OF APPROXIMATELY 176 OHMS. TO PREVENT RECURRENCE, THE LICENSEE IS DEVELOPING A PROGRAM TO ADDRESS CHANGEOUT OF LIGHT BULBS IN THE CONTROL ROOM.

[287] SUMMER 1 DOCKET 50-395 LER 87-026
MISSED SURVEILLANCE OF FIRE SERVICE DIESEL BATTERY DUE TO PERSONNEL ERROR.
EVENT DATE: 110387 REPORT DATE: 120187 NSSS: WE TYPE: PWR

(NSIC 207468) ON 11/04/87, DURING A POST TEST REVIEW IT WAS DISCOVERED THAT THE WEEKLY FIRE SERVICE DIESEL BATTERY TEST EXCEEDED THE GRACE PERIOD ALLOWED BY TECHNICAL SPECIFICATIONS SECTION 4.0.2. THE WEEKLY SURVEILLANCE TEST, STP-505.001, WAS PERFORMED ON THE MORNING OF 11/02/87 WHICH WAS BEYOND THE ALLOWABLE EXTENSION DATE OF 11/01/87. SURVEILLANCE TEST PROCEDURE 505-001 IS PERFORMED ON A WEEKLY FREQUENCY TO MEET THE REQUIREMENTS OF TECHNICAL SPECIFICATIONS SECTION 4.7.9.1.3A, " FIRE SUPPRESSION WATER SYSTEM." THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR. THERE WERE NO ADVERSE CONSEQUENCES DUE TO THE FOLLOWING: THE ELECTRICAL DRIVEN FIRE PUMP REMAINED OPERABLE, BATTERY VOLTAGES WERE RECORDED ONCE PER TWELVE (12) HOURS AND THE DIESEL FIRE PUMP WAS DEMONSTRATED OPERABLE ON 11/01/87. THIS EVENT WAS DISCUSSED DURING THE MAINTENANCE WEEKLY SAFETY MEETING ON 11/04/87, AND EMPHASIS WAS PLACED ON TIMELY COMPLETION OF SURVEILLANCE TESTS. THE FIRE SERVICE DIESEL WEEKLY BATTERY TEST WAS PERFORMED AGAIN ON THE EVENT DISCOVERY DATE TO REESTABLISH THE TEST FREQUENCY FOR THE MONDAY NIGHT SHIFT. THE ELECTRICAL SHIFT LOG FOR MONDAYS WAS REVISED UNDER MAINTENANCE SPECIAL INSTRUCTION MSI-01 PROVIDING A CHECK OFF BLOCK FOR THE BATTERY TEST, STP-505.001.

[288] SURRY 1 DOCKET 50-280 LER 87-028
EDG OUTPUT BREAKER PROTECTION CIRCUIT DESIGN DEFICIENCY.
EVENT DATE: 102187 REPORT DATE: 112087 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)

(NSIC 207183) ON OCTOBER 21, 1987, WITH BOTH UNITS OPERATING AT 100% POWER, IT WAS REPORTED THAT THE EMERGENCY DIESEL GENERATORS (EIIS-DG) (EDG) OUTPUT BREAKER (EIIS-52) PROTECTION CIRCUITS CONTAINED INSTANTANEOUS OVERCURRENT TRIPPING RELAYS (EIIS-RLY). THE RELAY SETTINGS WERE SUCH THAT A FAULT ON AN INDIVIDUAL BUS LOAD COULD CAUSE THE EDG OUTPUT BREAKER TO TRIP BEFORE THE LOAD BREAKER WOULD OPEN TO ISOLATE THE FAULT. THIS CONDITION EXISTED ON ALL FOUR (4) EDG OUTPUT BREAKERS WHICH FUNCTION AS THE EMERGENCY SUPPLY (FEEDER) BREAKERS FOR THE EMERGENCY BUSES. THE CAUSE OF THIS CONDITION HAS BEEN DETERMINED TO BE A DESIGN DEFICIENCY. THE INSTANTANEOUS OVERCURRENT TRIP RELAY SETTINGS WERE INCREASED ON ALL FOUR (4) EMERGENCY SUPPLY BREAKERS TO A SETTING THAT IS HIGHER THAN A LOAD FAULT CURRENT PLUS THE BUS CURRENT TO ENSURE AN INSTANTANEOUS OVERCURRENT TRIP WILL NOT OCCUR. THE INSTANTANEOUS OVERCURRENT TRIP RELAYS WILL BE REMOVED FROM THE EMERGENCY SUPPLY BREAKERS PROTECTION CIRCUIT DURING THE NEXT UNIT REFUELING OUTAGES.

[289] SURRY 1 DOCKET 50-280 LER 87-027
SPURIOUS ENGINEERED SAFETY FEATURE ACTUATION DUE TO WATER INTRUSION INTO CONTROL PANEL.
EVENT DATE: 102787 REPORT DATE: 112587 NSSS: WE TYPE: PWR
VENDOR: SOUTHWESTERN ENGINEERING COMPANY

(NSIC 207182) ON OCTOBER 27, 1987, UNIT 1 WAS OPERATING AT 100% POWER. AT 1657 HOURS, AN ELECTRICAL SHORT IN THE TRAIN "A" TURBINE BUILDING FLOOD CONTROL (EIIS-VK) PANEL (TBFT-A) CAUSED A SPURIOUS COMPLETION OF THE FLOOD PROTECTION CIRCUIT. THIS CAUSED TWO OF THE FOUR CONDENSER CIRCULATING WATER INLET VALVES, MOV-CW-106A AND C (EIIS-BS-V), TO BEGIN AUTOMATICALLY CLOSING AS DESIGNED. OPERATIONS PERSONNEL, RESPONDING TO A DECREASING CONDENSER VACUUM, IDENTIFIED THE AFFECTED VALVES, VERIFIED THEIR ACTUATION WAS SPURIOUS, AND FULLY OPENED THEM. A

LEAK HAD DEVELOPED ON AN ACCESS MANWAY OF MOISTURE SEPARATOR REHEATER (EIIS-MSR) 1-MS-E-1B. WATER DRAINED FROM THIS LEAK ONTO THE TRAIN "A" FLOOD CONTROL PANEL HOUSING AND CAUSED THE SHORT IN THE FLOOD PROTECTION CIRCUIT. THE FLOOD PANEL AND ITS WIRING WERE DRIED, THE SHORT WAS CLEARED AND THE FLOOD CONTROL SYSTEM WAS RETURNED TO NORMAL STATUS. THE ACCESS MANWAY LEAK ON THE MSR HAS BEEN REPAIRED.

[290] SURRY 1 DOCKET 50-280 LER 87-029
MAIN CONTROL ROOM VENTILATION ISOLATION DUE TO HIGH VOLTAGE OUTPUT ON CHLORINE GAS DETECTOR.
EVENT DATE: 110887 REPORT DATE: 120487 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)
VENDOR: CAPITAL CONTROLS CO.

(NSIC 207359) ON NOVEMBER 8, 1987 AT 1608 HOURS, THE MAIN CONTROL ROOM (MCR) VENTILATION WAS ISOLATED DUE TO A HIGH VOLTAGE OUTPUT ON CHLORINE GAS DETECTOR, CLA-VS-100A (EIIS-DET). AS A RESULT OF THE VOLTAGE SPIKE, THE CONTROL ROOM EXHAUST FAN, 1-VS-F-15 (EIIS-FAN) TRIPPED AND SUPPLY DAMPER, 1-MOD-VS-103A (EIIS-DMP) AND EXHAUST DAMPER, 1-MOD-VS-103D (EIIS-DMP) CLOSED. THE CHEMISTRY DEPARTMENT VERIFIED THAT THERE WAS NO DETECTABLE CHLORINE GAS PRESENT IN THE CONTROL ROOM. THE CHLORINE DETECTOR WAS RESET AND THE MAIN CONTROL ROOM VENTILATION (EIIS-VI) WAS REALIGNED. AT THE TIME OF THESE OCCURRENCES, BOTH UNIT 1 AND UNIT 2 WERE OPERATING AT 100% POWER. THE MAIN CONTROL ROOM VENTILATION ISOLATED DUE TO HIGH SENSOR VOLTAGE ON THE CHLORINE GAS DETECTOR. AN INVESTIGATION PERFORMED BY THE DETECTOR VENDOR DETERMINED THAT THE DETECTOR SENSOR OPERATION IS AIR FLOW DEPENDENT DUE TO MOUNTING OF THE SENSOR IN THE VENTILATION DUCT. THE AIR FLOW HAS A COOLING EFFECT ON THE SENSOR WHICH INCREASES THE VOLTAGE OUTPUT, THUS INDUCING A SPURIOUS SIGNAL. THE CHLORINE GAS TREATMENT SYSTEM AT THE SEWAGE TREATMENT PLANT WILL BE REPLACED BY AN ULTRAVIOLET SYSTEM. AT THAT TIME, A TECHNICAL SPECIFICATION REVISION WILL BE SUBMITTED TO REMOVE THE CHLORINE GAS DETECTORS.

[291] SURRY 1 DOCKET 50-280 LER 87-032
PROTECTION SYSTEM CHANNEL INOPERABLE DUE TO FAILED SUMMATOR IN SIGNAL CONDITIONING CIRCUIT.
EVENT DATE: 111187 REPORT DATE: 121187 NSSS: WE TYPE: PWR
VENDOR: HAGAN CONTROLS

(NSIC 207395) ON NOVEMBER 11, 1987 AT 1147 HOURS, WITH UNIT 1 AT 100% STEADY STATE POWER, THE UNIT 1 REACTOR COOLANT SYSTEM (EIIS AB) 'B' LOOP AVERAGE TEMPERATURE (TAVG) PROTECTION CHANNEL FAILED HIGH. THIS FAILURE CAUSED THE CHANNEL 2 OVER-POWER AND OVER-TEMPERATURE DELTA TEMPERATURE REACTOR TRIP AND TURBINE RUNBACK SIGNALS TO INITIATE AND THEIR CORRESPONDING ALARMS TO BE RECEIVED IN THE CONTROL ROOM. ALSO, THE 'B' LOOP "HIGH/LOW TAVG" ALARM WAS RECEIVED. THIS EVENT IS CONTRARY TO TECHNICAL SPECIFICATIONS WHICH REQUIRE A MINIMUM DEGREE OF REDUNDANCY OF ONE FOR OVER-POWER OVER-TEMPERATURE DELTA TEMPERATURE REACTOR TRIPS AND HIGH STEAM FLOW WITH LOW TAVG SAFETY INJECTION (EIIS-JE). IN RESPONSE TO THE INSTRUMENT FAILURE, OPERATORS VERIFIED THAT THE FOLLOWING INDICATIONS WERE NORMAL: 'B' LOOP CONTROL CHANNEL TAVG AND DELTA T AND, 'A' AND 'C' LOOP PROTECTION AND CONTROL TAVG AND DELTA T. IN ADDITION, THE EMERGENCY RESPONSE FACILITY COMPUTER SYSTEM INDICATED THAT THE HOT AND COLD LEG INPUTS TO THE 'B' LOOP TAVG PROTECTION CHANNEL WERE NORMAL. THE AFFECTED PROTECTION CHANNELS WERE PLACED IN THE TRIP MODE AT 1303 HOURS. THE SUMMATOR IN THE SIGNAL CONDITIONING CIRCUITRY WAS DETERMINED TO BE THE CAUSE OF THE CHANNEL FAILURE AND WAS REPLACED. THE AFFECTED PROTECTION CHANNELS WERE RETURNED TO NORMAL AT 1425 HOURS.

[292] SURRY 1 DOCKET 50-280 LER 87-031
CONTAINMENT ISOLATION VALVE INOPERABLE DUE TO MECHANICAL BINDING OF VALVE OPERATOR.
EVENT DATE: 111287 REPORT DATE: 121187 NSSS: WE TYPE: PWR
VENDOR: MASONEILAN INTERNATIONAL, INC.

(NSIC 207360) ON NOVEMBER 12, 1987 AT 1720 HOURS, UNIT 1 WAS AT 100% POWER. DURING THE PERIODIC TEST FOR CONTAINMENT TRIP VALVES, IT WAS OBSERVED THAT THE CONTAINMENT INSTRUMENT AIR (EIIS-LD) ISOLATION TRIP VALVE (1-IA-TV-100) (EIIS-ISV) WOULD NOT CLOSE FULLY ON DEMAND. FAILURE OF THE TRIP VALVE TO CLOSE IS CONTRARY TO THE TECHNICAL SPECIFICATION DEFINITION OF CONTAINMENT INTEGRITY WHICH REQUIRES THAT ALL AUTOMATIC CONTAINMENT ISOLATION VALVES ARE OPERABLE OR ARE LOCKED CLOSED UNDER ADMINISTRATIVE CONTROL. AN OPERATOR WAS DISPATCHED TO THE VALVE, AND AT 1800 HOURS WAS ABLE TO FREE THE VALVE OPERATOR. A SATISFACTORY STROKE TEST OF THE VALVE WAS PERFORMED. THE VALVE OPERATOR WAS DETERMINED TO BE MECHANICALLY BOUND BY ITS AIR SOLENOID (EIIS-SOL) ELECTRICAL CONNECTION. THE ELECTRICAL CONNECTION TO THE SOLENOID ROTATED INTO A POSITION WHICH BLOCKED THE DOWNWARD MOVEMENT OF THE EXTERNAL PART OF THE VALVE OPERATOR. AN INSPECTION WAS PERFORMED OF SIMILAR VALVES ON BOTH UNITS 1 AND 2. NO SIMILAR CONFIGURATION WAS FOUND. AN ENGINEERING WORK REQUEST HAS BEEN INITIATED TO SECURE THE AIR SOLENOID TO PREVENT MOVEMENT.

[293] SURRY 1 DOCKET 50-280 LER 87-033
CHARGING PUMP COMPONENT COOLING WATER SYSTEM INOPERABLE DUE TO INADEQUATE TEST PROCEDURE.
EVENT DATE: 112487 REPORT DATE: 122287 NSSS: WE TYPE: PWR

(NSIC 207430) ON NOVEMBER 24, 1987, AT 0820 HOURS, WITH UNIT 1 AT 100% POWER, THE CHARGING PUMP COMPONENT SYSTEM (EIIS-CC) BECAME INOPERABLE. AT THE TIME OF THIS EVENT, THE 2A CHARGING PUMP CC PUMP (EIIS-P) WAS IN SERVICE. THE REDUNDANT PUMP (2B) HAD BEEN SECURED FOR HYDROSTATIC TESTING OF ITS CORRESPONDING INTERMEDIATE SEAL COOLER. AS OPERATORS WERE COMPLETING THE RESTORATION VALVE LINEUP FOLLOWING THE HYDROSTATIC TEST, THE 2A PUMP BECAME AIR BOUND RESULTING IN ZERO DISCHARGE PRESSURE INDICATION. NOTING THIS, THE CONTROL ROOM OPERATOR MANUALLY STARTED THE 2B PUMP, HOWEVER, IT ALSO BECAME AIR BOUND AND BOTH PUMPS WERE DECLARED INOPERABLE. THIS CONDITION IS PROHIBITED BY TECHNICAL SPECIFICATION 3.3.A.7.6. THE CHARGING PUMP CC PUMPS BECAME AIR BOUND DUE TO AN INADEQUATE HYDROSTATIC TEST PROCEDURE. THE PROCEDURE DID NOT PROVIDE DETAILED INSTRUCTIONS FOR DEPRESSURIZING AND VENTING THE SYSTEM FOLLOWING THE TEST. THE SECTION OF THE SYSTEM WHICH WAS TESTED PRESENTS AN UNUSUAL PIPING CONFIGURATION IN THAT IT DOES NOT CONTAIN A HIGH POINT VENT. THE VENTING WAS PERFORMED AT THE CHARGING PUMP CC PUMP CASING, A LOW POINT IN THE SYSTEM. THE HYDROSTATIC TEST PROCEDURE FOR THIS APPLICATION WILL BE ENHANCED TO INCLUDE DETAILED INSTRUCTIONS FOR DEPRESSURIZING AND VENTING THE SYSTEM AT THE COMPLETION OF THE TEST.

[294] SURRY 1 DOCKET 50-280 LER 87-037
EMERGENCY DIESEL GENERATOR AUTO-START DUE TO FAILED RELAY AND BLOWN FUSE.
EVENT DATE: 120387 REPORT DATE: 122287 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)

(NSIC 207432) ON DECEMBER 3, 1987, AT 0928 HOURS, WITH UNIT 1 AND UNIT 2 AT 100% POWER, BUS 1J UNDERVOLTAGE ALARM (EIIS-ANN) AND BUS 1J OVERVOLTAGE ALARM (EIIS-ANN) WERE RECEIVED. OPERATIONS PERSONNEL OBSERVED THAT THE MAIN STATION BATTERY (EIIS-BTRY) GROUND LIGHTS FLICKERED. EMERGENCY DIESEL GENERATOR #3 (EDG) (EIIS-DG) AUTOMATICALLY STARTED, WHICH IS AN ENGINEERED SAFETY FEATURE ACTUATION. AT 0930 HOURS, THE ALARMS CLEARED. AT 0946 HOURS, EDG #3 WAS PLACED IN ITS COOLDOWN CYCLE. AT 0950 HOURS, IT WAS RETURNED TO THE AUTOMATIC MODE. AT 0954 HOURS, THE ALARM FOR BUS 1J UNDERVOLTAGE WAS RECEIVED, AND THE #3 EDG AUTO-STARTED AGAIN AND WAS MANUALLY LOADED ONTO 1J BUS. THE CAUSE OF THE

AUTO-START OF #3 EDG WAS A BLOWN DC FUSE (EIIS-FU) IN THE DEGRADED VOLTAGE CIRCUIT. RELAY (EIIS-59) 59 ABC-1J1, THE OVERVOLTAGE RELAY SURGE CAPACITOR, FAILED AND SHORTED TO THE RELAY CASE. THIS CAUSED THE POSITIVE DC FUSE IN THE DEGRADED VOLTAGE CIRCUIT TO BLOW. THE FIRST AUTO-START OF #3 EDG AT 0928 HOURS WAS DUE TO THE ARCING OF THE SURGE CAPACITOR WITH THE RELAY CASE. WHEN THE SURGE CAPACITOR FAILED AND FLASHED TO GROUND, DC POWER WAS LOST, CAUSING THE UNDERVOLTAGE SIGNAL AND THE AUTO-START OF THE DIESEL AT 0954 HOURS. THE UNDERVOLTAGE RELAY 59 ABC-1J1 AND THE DC CONTROL FUSE WERE REPLACED, AND THE CIRCUIT WAS RETURNED TO NORMAL.

[295] SUSQUEHANNA 1 DOCKET 50-387 LER 87-029
UNANTICIPATED ESF ACTUATION DUE TO MOMENTARY LOSS OF POWER TO RPS BUS.
EVENT DATE: 110187 REPORT DATE: 120187 NSSS: GE TYPE: BWR

(NSIC 207272) AT 1620 ON NOVEMBER 1, 1987, UNIT 1 EXPERIENCED AN UNPLANNED ENGINEERED SAFEGUARD FEATURE ACTUATION DUE TO A MOMENTARY LOSS OF POWER TO THE REACTOR PROTECTION SYSTEM (RPS) BUS. UNIT 1 WAS IN REFUELING WITH THE "A" RESIDUAL HEAT REMOVAL PUMP OPERATING IN THE SHUTDOWN COOLING MODE. THE "B" RPS SYSTEM WAS ALIGNED TO ITS ALTERNATE POWER SUPPLY, A 480 VAC BREAKER ON DOUBLE ENDED LOAD CENTER (LC) 1B250/1B260. THE EVENT WAS INITIATED BY A UTILITY NONLICENSED OPERATOR WHO WAS RESTORING THE DOUBLE ENDED LOAD CENTER (LC) 1B250/1B260 TO A NORMAL OPERATING LINE-UP. THESE ACTIONS CAUSED A MOMENTARY LOSS OF POWER TO THE LOADS FED FROM 1B260, ONE OF THE LOADS BEING THE "B" RPS SYSTEM. THIS MOMENTARY LOSS WAS LONG ENOUGH TO CAUSE THE OUTBOARD ISOLATION VALVE (F008), A PRIMARY CONTAINMENT ISOLATION VALVE ON THE SHUTDOWN COOLING SUCTION OF THE RHR PUMPS, TO CLOSE, THUS TRIPPING THE RHR PUMP. THE VALVE WAS REOPENED AND SHUTDOWN COOLING WAS RESTORED. SYSTEM OPERATING PROCEDURES AND OPERATING PERSONNEL TRAINING CONCERNING REALIGNMENTS OF 480 VAC LOAD CENTERS WILL BE REVIEWED AND REVISED AS APPROPRIATE.

[296] SUSQUEHANNA 1 DOCKET 50-387 LER 87-031
UNANTICIPATED ESF ACTUATION DUE TO INSTALLATION OF JUMPER IN WRONG PANEL.
EVENT DATE: 110587 REPORT DATE: 120787 NSSS: GE TYPE: BWR

(NSIC 207350) AT 0338 ON 11-5-87, AN UNANTICIPATED ENGINEERED SAFEGUARDS FEATURE ACTUATION OCCURRED ON UNIT ONE WHICH WAS IN REFUELING. TESTING AND OPERATING PERSONNEL WERE PERFORMING SURVEILLANCE TEST SE-159-200, "18 MONTH LOGIC SYSTEM FUNCTIONAL TEST OF THE PRIMARY AND SECONDARY CONTAINMENT ISOLATION SYSTEM," WHEN THE "B" TRAINS OF STANDBY GAS TREATMENT SYSTEM (SGTS), CONTROL ROOM EMERGENCY OUTSIDE AIR SUPPLY SYSTEM (CREOASS), REACTOR BUILDING RECIRCULATION FANS UNEXPECTEDLY ACTUATED. THIS ACTUATION WAS CAUSED BY THE INSTALLATION OF A JUMPER IN THE WRONG PANEL. THE JUMPER WAS REMOVED AND AFFECTED SYSTEMS RESTORED. CAUSES OF THE EVENT WERE COGNITIVE PERSONNEL ERROR AND INCOMPLETE VERBAL COMMUNICATIONS. FOLLOWING NORMAL PRACTICE, THE TEST DIRECTOR VERBALLY INSTRUCTED THE ELECTRICIAN TO INSTALL THE JUMPER. THE TEST DIRECTOR SPECIFIED THE CORRECT TERMINAL POINTS TO THE ELECTRICIAN BUT FAILED TO SPECIFY THE PANEL. FURTHERMORE, THE TEST DIRECTOR FAILED TO RECOGNIZE THAT THE JUMPER WAS TO BE PLACED IN A PANEL DIFFERENT FROM THE PANEL USED IN THE PRECEDING TWENTY NINE STEPS OF THE TEST PROCEDURE. THE TEST DIRECTOR INVOLVED IN THE EVENT WAS COUNSELLED. THIS EVENT WILL BE REVIEWED WITH TEST DIRECTORS ON THE PLANT STAFF TECHNICAL SECTION STAFF ALONG WITH GUIDANCE ON PROPER VERBAL COMMUNICATIONS TECHNIQUES.

[297] SUSQUEHANNA 1 DOCKET 50-387 LER 87-030
DAMAGED WIRE RESULTS IN BLOWN FUSE AND AN INADVERTENT ENGINEERED SAFEGUARD FEATURE ACTUATION.
EVENT DATE: 111387 REPORT DATE: 121187 NSSS: GE TYPE: BWR

(NSIC 207391) ON NOVEMBER 13, 1987, AT APPROXIMATELY 2017 HOURS, A FUEL BLEW IN

THE CONTROL LOGIC OF THE CONTAINMENT INSTRUMENT GAS SUCTION OUTBOARD ISOLATION VALVE. THE UNIT WAS IN CONDITION 4. ELECTRICAL MAINTENANCE PERSONNEL WERE PERFORMING MT-GE-028 "TARGET ROCK SOLENOID VALVE POSITION INDICATION MAINTENANCE" IN ACCORDANCE WITH SURVEILLANCE TEST SO-125-015 "EIGHTEEN (18) MONTH CONTAINMENT INSTRUMENT GAS REMOTE POSITION INDICATOR (RPI) CHECKS." THE PROCEDURE DIRECTED THE ELECTRICIAN TO REMOVE THE REED SWITCH HOUSING COVER TO SV-12605 IN ORDER TO VERIFY THAT THE VALVE WAS OPERATING PROPERLY. WHEN THE ELECTRICIAN PERFORMED THIS STEP, A FUSE BLEW IN THE CONTROL LOGIC TO THE VALVE, CAUSING SV-12605 TO CLOSE. AS A RESULT OF SV-12605 CLOSING, THE INSTRUMENT GAS COMPRESSOR TRIPPED. THE ELECTRICIAN INVESTIGATED AND FOUND A NICKED WIRE CONNECTED TO THE REED SWITCH. AFTER THE ELECTRICIAN TAPED THE WIRE, THE VALVE OPERATED PROPERLY. THE CAUSE OF THE BLOWN FUSE WAS APPARENTLY DUE TO A NICKED WIRE TO THE REED SWITCH. THE NICKED WIRE IS BELIEVED TO HAVE SHORTED TO GROUND CAUSING THE TO BLOW. INTERIM CORRECTIVE ACTION INCLUDED TAPING THE NICKED WIRE. ON NOVEMBER 17, 1987, ELECTRICAL MAINTENANCE PERSONNEL REPLACED THE DAMAGED WIRE.

[298] SUSQUEHANNA 1 DOCKET 50-387 LER 87-032
EMERGENCY DIESEL GENERATOR "A" UNPLANNED AUTOMATIC START.
EVENT DATE: 112287 REPORT DATE: 121087 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: SUSQUEHANNA 2 (BWR)
VENDOR: AGASTAT RELAY CO.

(NSIC 207304) ON NOVEMBER 22, 1987 AT 1105 HOURS, WITH UNIT 1 OPERATING AT 9% POWER AND UNIT 2 OPERATING AT 100% POWER, AN UNPLANNED ENGINEERED SAFETY FEATURE (ESF) ACTUATION OCCURRED WHEN THE "A" DIESEL GENERATOR AUTOMATICALLY STARTED. INVESTIGATION REVEALED THAT THE CAUSE OF THE AUTO START WAS THE FAILURE OF A COIL IN A RELAY OF THE ELECTRICAL CONTROL CIRCUIT. THE COIL WAS FOUND TO BE "BURNED OUT". FAILURE OF THE COIL (OPEN CIRCUIT) SIMULATED THE EMERGENCY START CONDITION (FAILSAFE, DEENERGIZED RELAY), ALLOWING THE ENGINE TO START. THE DIESEL GENERATOR STARTED, RAN PROPERLY AND WAS AVAILABLE TO ENERGIZE THE EMERGENCY BUS AND PERFORM ITS DESIGN FUNCTION, IF REQUIRED. AS SUCH, THIS EVENT IS NOT CLASSIFIED AS A VALID TEST FAILURE, AS DEFINED BY REG. GUIDE 1.108. THE DIESEL WAS MANUALLY SHUT DOWN, THE DEFECTIVE RELAY WAS REPLACED AND IT WAS RETURNED TO ITS NORMAL STANDBY CONDITION. BASED ON A HISTORICAL REVIEW, THE COIL FAILURE OF THIS RELAY HAS BEEN DETERMINED TO BE ATTRIBUTED TO NORMAL ENERGIZED LIFE AND THERE APPEARS TO BE NO RECURRENCE TREND. OPERATION OF BOTH UNITS CONTINUED UNINTERRUPTED.

[299] SUSQUEHANNA 2 DOCKET 50-388 LER 87-012
AUXILIARY BOILER ARC-OVER CAUSES PRIMARY CONTAINMENT ISOLATION VALVE CLOSURE.
EVENT DATE: 102887 REPORT DATE: 113087 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: SUSQUEHANNA 1 (BWR)

(NSIC 207341) AT 0830 ON OCTOBER 28, 1987, A SPURIOUS ENGINEERED SAFETY FEATURE ACTUATION OCCURRED ON UNIT 2. WITH UNIT 1 IN REFUELING AND UNIT 2 IN NORMAL POWER OPERATIONS AT 100% POWER, AUXILIARY BOILER "A" EXPERIENCED AN INTERNAL ELECTRICAL ARC-OVER WHICH CAUSED AN OVERCURRENT TRIP OF ITS 13.8KV SUPPLY BREAKER. THIS CAUSED A TRANSIENT ON THE STARTUP BUS 10 WHICH RESULTED IN CLOSURE OF UNIT 2 PRIMARY CONTAINMENT ISOLATION VALVES ASSOCIATED WITH THE CONTAINMENT ATMOSPHERE CONTROL SYSTEM AND VARIOUS OTHER MINOR SYSTEM PERTURBATIONS AND ALARMS ON UNIT 2 AND UNIT 1. UNIT 2 REACTOR POWER REMAINED CONSTANT THROUGHOUT THE OCCURRENCE AND RECOVERY. ALL AFFECTED SYSTEMS WERE PROMPTLY RETURNED TO NORMAL OPERATION. CAUSE OF THE ARC-OVER AND CORRECTIVE ACTIONS TO PREVENT RECURRENCE HAVE NOT YET BEEN DETERMINED AND WILL BE DISCUSSED IN A SUPPLEMENTAL REPORT. EFFORTS UNDERWAY INCLUDE INVESTIGATIONS INTO AUXILIARY BOILER SYSTEM OPERATING PROCEDURE ADEQUACY AND PROCEDURE IMPLEMENTATION DURING THE EVENT.

[300] TROJAN DOCKET 50-344 LER 87-032
RCS WIDE RANGE PRESSURE TRANSMITTERS OUT-OF-CALIBRATION DUE TO INSTRUMENT DRIFT.
EVENT DATE: 041787 REPORT DATE: 112587 NSSS: WE TYPE: PWR
VENDOR: ITT-BARTON

(NSIC 207329) ON APRIL 17, 1987, REACTOR COOLANT SYSTEM (RCS) WIDE RANGE PRESSURE TRANSMITTERS PT-403 AND 405 WERE FOUND OUT-OF-CALIBRATION. THIS WOULD HAVE RESULTED IN AUTOMATIC ISOLATION OF THE RESIDUAL HEAT REMOVAL (RHR) SYSTEM FROM THE RCS AT A PRESSURE HIGHER THAN THE 600 PSIG LIMIT IN THE TECH SPECS. IN ADDITION, THE RCS POWER-OPERATED RELIEF VALVES WOULD HAVE LIFTED AT PRESSURES HIGHER THAN THE VALUES ALLOWED IN THE TECH SPECS (440 PSIG AND 490 PSIG) DURING A POSTULATED COLD OVERPRESSURE TRANSIENT. THE CAUSE OF THIS EVENT WAS PRESSURE TRANSMITTER DRIFT. THESE TRANSMITTERS WERE INSTALLED IN 1986 AND THE VENDOR HAS INDICATED THAT NEW TRANSMITTERS CAN EXHIBIT GREATER DRIFT. THE IMMEDIATE CORRECTIVE ACTION WAS TO RECALIBRATE THE TRANSMITTERS. THE PRE-INSTALLATION ACCEPTANCE TEST ON NEW TRANSMITTERS OF THIS TYPE IS BEING EVALUATED FOR ADEQUACY. BECAUSE OF THE SIMILARITIES BETWEEN THIS EVENT AND THE OUT-OF-CALIBRATION DISCOVERED IN THE MAIN STEAM PRESSURE TRANSMITTERS MANUFACTURED BY THE SAME VENDOR (LER 87-33, 11-25-87), FURTHER EVALUATION OF BOTH OF THESE EVENTS IS CONTINUING. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[301] TROJAN DOCKET 50-344 LER 87-029
FIRE WATCH NOT ESTABLISHED DUE TO PERSONNEL ERROR.
EVENT DATE: 070687 REPORT DATE: 110687 NSSS: WE TYPE: PWR

(NSIC 207135) ON JULY 6, 1987, THE HALON SYSTEM FOR THE REMOTE SHUTDOWN PANEL ROOM AND DELUGE SYSTEMS 18 AND 19 FOR THE ELECTRICAL PENETRATION AREA WERE REMOVED FROM SERVICE DURING MAINTENANCE. CONTRARY TO TECH SPEC 3.7.8.2, "SPRAY, SPRINKLER, AND/OR DELUGE SYSTEMS", A CONTINUOUS FIRE WATCH WAS NOT ESTABLISHED WITHIN 1 HOUR IN THE AREAS PROTECTED BY THESE SUPPRESSION SYSTEMS. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR. ALTHOUGH THE DATE FOR INITATING THE FIRE WATCH WAS CHANGED TO BE CONSISTENT WITH A SCHEDULE CHANGE FOR THE MAINTENANCE BEING PERFORMED, THE SPECIFIED TIME FOR BEGINNING THE FIRE WATCH WAS NOT CHANGED. THIS RESULTED IN THE MAINTENANCE THAT CAUSED THE SUPPRESSION SYSTEMS TO BE INOPERABLE OCCURRING PRIOR TO ESTABLISHING A FIRE WATCH. ADMINISTRATIVE ORDER 10-2, "FIRE PROTECTION", WAS REVIEWED AND IT WAS CONFIRMED THAT SUFFICIENT PROCEDURAL GUIDANCE IS PROVIDED ON THE ADMINISTRATION OF FIRE PATROLS. THE FAILURE TO PROVIDE A FIRE WATCH WAS CONSIDERED AN ISOLATED EVENT. PERSONNEL WERE COUNSELED ON THE PROCEDURAL REQUIREMENTS OF AO-10-2. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[302] TROJAN DOCKET 50-344 LER 87-031
STEAM FLOW CHANNEL BISTABLE NOT TRIPPED PER TECH SPECS DUE TO PERSONNEL ERROR.
EVENT DATE: 102187 REPORT DATE: 112087 NSSS: WE TYPE: PWR

(NSIC 207235) ON OCTOBER 21, 1987, THE "C" STEAM LINE FLOW INDICATOR FI-532 FAILED THE CHANNEL CHECK REQUIRED BY TECH SPECS 4.3.1.1 AND 4.3.2.1. CONTRARY TO LIMITING CONDITION FOR OPERATION 3.3.2, THE INOPERABLE STEAM FLOW CHANNEL WAS NOT PLACED IN THE TRIPPED CONDITION WITHIN ONE HOUR. THE CHANNEL WAS TRIPPED WITHIN APPROXIMATELY 3.5 HOURS. THE CAUSE OF THE FAILURE TO TRIP THE BISTABLES WAS PERSONNEL ERROR. THE CAUSE OF THE INCORRECT INDICATION FROM FI-532 WAS DRIFT IN THE ASSOCIATED SQUARE ROOT EXTRACTOR. PERIODIC OPERATING TEST (POT) 24-2, "DAILY OPERATING ROUTINES", WAS REVIEWED AND IT WAS CONFIRMED THAT SUFFICIENT PROCEDURAL GUIDANCE IS PROVIDED ON TECH SPEC REQUIRED ACTIONS UPON DISCOVERY OF A NONCONFORMING CONDITION. THE FAILURE TO TRIP THE BISTABLES ASSOCIATED WITH FT-532 WAS CONSIDERED AN ISOLATED EVENT. PERSONNEL WERE COUNSELED ON THE TECH SPEC REQUIREMENTS. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[303] TROJAN DOCKET 50-344 LER 87-033
MAIN STEAM PRESSURE TRANSMITTERS OUT-OF-CALIBRATION DUE TO APPARENT PROCEDURE INADEQUACY.
EVENT DATE: 103087 REPORT DATE: 112587 NSSS: WE TYPE: PWR

(NSIC 207267) ON OCTOBER 30, 1987, IT WAS NOTED THAT ALL OF THE MAIN STEAM LINE PRESSURE TRANSMITTERS APPEARED TO BE ABOUT 35 TO 50 PSIG LOWER THAN IN PREVIOUS OPERATING CYCLES. EACH TRANSMITTER WAS CHECKED AND FOUND TO HAVE A ZERO SHIFT VARYING BETWEEN 21 TO 50 PSI. I.E., THE TRANSMITTERS HAD TO BE PRESSURIZED TO 21 TO 50 PSIG BEFORE THE CONTROL ROOM PRESSURE INDICATORS WOULD READ GREATER THAN 0 PSIG. THE EXACT CAUSE OF THIS EVENT CONTINUES TO BE INVESTIGATED, AND MAY HAVE BEEN DUE TO A COMBINATION OF (1) INADEQUATE CYCLING OF NEW TRANSMITTERS. (2) INSTRUMENT DRIFT AND (3) POTENTIAL PERSONNEL ERROR IN THE CALIBRATION PROCESS. THE PRESSURE TRANSMITTERS WERE IMMEDIATELY RECALIBRATED. THE NEED FOR A NEW PROCEDURE TO PROVIDE INSTRUCTIONS FOR CHECKOUT, TESTING AND INITIAL CALIBRATION OF NEW TRANSMITTERS WILL BE EVALUATED. IF IT IS DETERMINED THAT A NEW PROCEDURE IS NECESSARY, IT WILL BE IN PLACE BY APRIL 1, 1988. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[304] TROJAN DOCKET 50-344 LER 87-034
EDG NOT DEMONSTRATED OPERABLE PER TECHNICAL SPECIFICATIONS DUE TO PERSONNEL ERROR.
EVENT DATE: 110287 REPORT DATE: 120287 NSSS: WE TYPE: PWR

(NSIC 207330) ON NOVEMBER 2, 1987, UPON REMOVAL OF THE "A" (EDG) FROM SERVICE FOR ROUTINE MAINTENANCE, THE REDUNDANT EDG WAS NOT DEMONSTRATED OPERABLE WITHIN ONE HOUR AS REQUIRED BY TECH SPEC 3.8.1.1. THE REDUNDANT EDG WAS DEMONSTRATED OPERABLE 2.5 HOURS AFTER THE "A" EDG BECAME INOPERABLE. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR. THE PERSONNEL INVOLVED DID NOT COMPLY WITH THE TECH SPECS NOR PROCEDURES. THE IMMEDIATE CORRECTIVE ACTION WAS TO DEMONSTRATE OPERABILITY OF THE "B" EDG. PERIODIC OPERATING TEST (POT) 21-2, "ENGINEERED SAFETY FEATURES AND OFFSITE POWER AVAILABILITY", WAS REVIEWED AND IT WAS CONFIRMED THAT THIS POT SPECIFICALLY REQUIRES THAT THE REDUNDANT EDG BE DEMONSTRATED OPERABLE WHEN AN EDG IS REMOVED FROM SERVICE. PERSONNEL INVOLVED WERE COUNSELED ON THE REQUIREMENTS OF THE TECH SPECS AND PROCEDURES. A REVISION TO THE OPERATIONS MANUAL HAS BEEN PREPARED THAT WILL REQUIRE THAT OPERATORS MAINTAIN A COPY OF SCHEDULED SURVEILLANCES FOR THEIR SHIFT ON THEIR DESK. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[305] TROJAN DOCKET 50-344 LER 87-035
FIRE DOORS MADE INOPERABLE DUE TO PERSONNEL ERROR.
EVENT DATE: 111887 REPORT DATE: 121887 NSSS: WE TYPE: PWR

(NSIC 207445) ON NOVEMBER 18, 1987, AT ABOUT 1300 HOURS, ELECTRICAL CABLES AND AIR HOSES WERE TEMPORARILY ROUTED BETWEEN THE COMPONENT COOLING WATER (CCW) HEAT EXCHANGER ROOM AND THE FUEL BUILDING CRANE BAY TO SUPPLY AUXILIARY EQUIPMENT REQUIRED TO SUPPORT RADIOACTIVE SPENT RESIN TRANSFERS. NORMALLY OPEN ROLLUP FIRE DOORS 430 AND 431 EQUIPPED WITH FUSIBLE LINKS ARE PROVIDED AS FIRE BARRIERS FOR THE CCW HEAT EXCHANGER ROOM SUCH THAT IN THE EVENT OF A FIRE, THE FUSIBLE LINKS WILL MELT AND THE ROLLUP DOORS WILL CLOSE. WITH THE CABLES AND HOSES ROUTED THROUGH THE OPENINGS TO THE CCW HEAT EXCHANGER ROOM, THE ROLLUP FIRE DOORS WOULD HAVE BEEN UNABLE TO COMPLETELY CLOSE IN THE EVENT OF A FIRE. THE CAUSE OF THIS EVENT WAS PROCEDURAL INADEQUACY IN THAT PROCEDURES DID NOT REQUIRE THESE DOORS TO BE POSTED AS FIRE DOORS. THE IMMEDIATE CORRECTIVE ACTION WAS TO VERIFY THE OPERABILITY OF FIRE DETECTORS IN THE AREA AND ESTABLISH AN HOURLY FIRE PATROL AS REQUIRED BY TECHNICAL SPECIFICATION 3.7.9. FOLLOWING REMOVAL OF THE CABLES AND HOSES, THE ROLLUP DOORS WILL BE CLOSED AND LEFT CLOSED AS THEIR NORMAL POSITION. SIGNS WERE ADDED AT THESE DOORS IDENTIFYING THEM AS FIRE DOORS. PROCEDURES WILL BE REVISED TO REQUIRE PERIODIC CHECKS OF FIRE DOORS FOR ADEQUATE POSTING. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[306] TURKEY POINT 3 DOCKET 50-250 LER 87-030
DESIGN BASIS RECONSTITUTION DISCOVERS RESIDUAL HEAT REMOVAL RECIRCULATION LINE NOT DESIGNED TO ASSURE ADEQUATE FLOW FOR EACH PUMP.
EVENT DATE: 102787 REPORT DATE: 112387 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: TURKEY POINT 4 (PWR)
VENDOR: INGERSOLL-RAND CO.

(NSIC 207171) ON OCTOBER 27, 1987, WHILE UNIT 3 AND UNIT 4 WERE IN MODE 5 (COLD SHUTDOWN), IT WAS DETERMINED THAT A DESIGN DISCREPANCY EXISTED IN THE RESIDUAL HEAT REMOVAL (RHR) SYSTEM. DURING THE DESIGN BASIS RECONSTITUTION OF THE RHR SYSTEM, IT WAS DISCOVERED THAT THE EXISTING MINIMUM RECIRCULATION DESIGN CONFIGURATION WAS POTENTIALLY INADEQUATE. THE PRESENT RHR SYSTEM DESIGN HAS TWO (2) RHR PUMPS DISCHARGING FLOW THROUGH A SHARED MINI FLOW RECIRCULATION LINE. IF THE PERFORMANCE OF ONE RHR PUMP IS SLIGHTLY BETTER THAN THE OTHER, THEN IT IS POSSIBLE FOR THE RHR PUMP WITH THE HIGHER DISCHARGE PRESSURE TO DEADHEAD THE RHR PUMP WITH THE LOWER DISCHARGE PRESSURE IF THE REACTOR COOLANT SYSTEM (RCS) PRESSURE IS ABOVE RHR PUMP SHUTOFF HEAD. THE EXISTING PLANT EMERGENCY OPERATING PROCEDURES REQUIRE THE OPERATOR TO TERMINATE RHR IF THE RCS PRESSURE IS ABOVE RHR PUMP SHUTOFF HEAD. HOWEVER, IT CANNOT BE ASSURED BASED ON EXISTING PLANT PROCEDURES THAT THIS STEP WILL BE ACCOMPLISHED PRIOR TO THE POTENTIAL DAMAGE OF A RHR PUMP. THIS POTENTIAL FAILURE (BY DESIGN) COUPLED WITH A SINGLE FAILURE OF THE OTHER OPERATING RHR TRAIN COULD RESULT IN COMPLETE LOSS OF RHR PUMP CAPABILITY. THE RECIRCULATION FLOW PATH FOR THE RHR PUMPS WAS AN ORIGINAL DESIGN OF THE PLANT.

[307] TURKEY POINT 4 DOCKET 50-251 LER 87-024
BORIC ACID HEAT TRACING CIRCUIT NUMBER 6 DECREASES TO LESS THAN 145 DEGREES.
EVENT DATE: 100887 REPORT DATE: 110987 NSSS: WE TYPE: PWR

(NSIC 207123) ON 10/8/87, WITH UNIT 4 IN MODE 1 AT 100% POWER, THE TEMPERATURE FOR BORIC ACID HEAT TRACING CIRCUIT NUMBER 6 (HT-6), DECREASED 145F. THIS CIRCUIT IS LOCATED ON THE CROSSTIE PIPING AND VALVES BETWEEN THE 4A AND 4B BORIC ACID PUMPS, AND ON THE PUMP'S DISCHARGE LINE TO THE INLET OF THE UNIT 4 BORIC ACID FILTER. PREPARATIONS WERE MADE TO SHUT DOWN THE UNIT AS REQUIRED BY TECH SPEC 3.0.1. THE TEMPERATURE DECREASE OCCURRED DURING THE TIME THE 4B BORIC ACID PUMP CROSSTIE LINE WAS BEING FLUSHED TO FACILITATE MAINTENANCE WORK ON VALVE 4-374. THE FLUSHING WAS STOPPED AND THE SYSTEM WAS RESTORED TO THE NORMAL ALIGNMENT. A BLENDED MAKEUP FLOW WAS PUMPED THROUGH THE BORIC ACID FLOWPATH TO VERIFY THAT THERE WAS NO BLOCKAGE IN THE PIPE AND TO ASSIST WITH THE TEMPERATURE INCREASE IN THE CIRCUIT. AT 1915, THE TEMPERATURE STARTED INCREASING AND CONTINUED TO INCREASE STEADILY UNTIL IT WAS ABOVE 145 DEGREES AT 2007 HOURS. ALSO FOUND AS A PART OF THE INVESTIGATION OF THE HEAT TRACING TEMPERATURE WAS A SENSING BULB THAT WAS NOT ADEQUATELY ATTACHED TO THE PIPING. THIS WAS NOT A CONTRIBUTING FACTOR TO THE TEMPERATURE DROP SINCE THE SYSTEM HAD BEEN FUNCTIONING PROPERLY PRIOR TO THE FLUSHING OF THE SYSTEM. THE SENSING BULB WAS REANCHORED AND THE HEAT TRACING PERIODIC TEST WAS PERFORMED ON HT-6.

[308] TURKEY POINT 4 DOCKET 50-251 LER 87-025
CONTAINMENT AND CONTROL ROOM VENTILATION ISOLATION DUE TO HIGH LEVELS OF RUBIDIUM IN CONTAINMENT FOLLOWING UNIT SHUTDOWN.
EVENT DATE: 101187 REPORT DATE: 111287 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: TURKEY POINT 3 (PWR)
VENDOR: TRACER LAB

(NSIC 206908) ON OCTOBER 12, 1987, AT 2015, WITH UNIT 4 IN MODE 3, PROCESS RADIATION MONITOR (PRM) R-11, CONTAINMENT RADIOACTIVE PARTICULATE MONITOR, ACTUATED CONTAINMENT VENT AND THE CONTROL ROOM VENTILATION ISOLATION. R-12, THE CONTAINMENT RADIOACTIVE GASEOUS MONITOR WAS VERIFIED TO BE READING NORMAL. A PARTICULATE GRAB SAMPLE WAS TAKEN, AND A LEAK RATE CALCULATION WAS INITIATED.

THE PARTICULATE GRAB SAMPLE SHOWED A HIGHER THAN EXPECTED RADIATION LEVEL, WHICH WAS MAINLY DUE TO RB-88. THE RESULTS OF THE LEAK RATE CALCULATION INDICATED THERE WAS NO UNEXPECTED RCS LEAKAGE. THE CONTAINMENT ACTIVITY LEVEL RETURNED TO NORMAL LEVELS BY APPROXIMATELY 0100, OCTOBER 13. AN INVESTIGATION OF THE INCREASED LEVEL OF RB-88 RADIATION IDENTIFIED THAT THE TIME OF THE INCREASE COINCIDED WITH THE DEPRESSURIZATION OF THE RCS, AND WITH AN INCREASE IN IODINE ACTIVITY IN THE RCS. BASED ON THIS, THE LIKELY CAUSE OF THE HIGHER THAN NORMAL ACTIVITY WAS A RAPID UNIT SHUTDOWN AND SUBSEQUENT DEPRESSURIZATION, DUE TO A HURRICANE WARNING. THIS WAS EXACERBATED BY A KNOWN LEAK IN THE FUEL CLADDING. AS DOSE EQUIVALENT IODINE LEVEL IS CONSIDERED TO BE AN INDICATOR OF FUEL CLADDING LEAKAGE, IT WILL CONTINUE TO BE CLOSELY MONITORED WHEN THE UNIT GOES CRITICAL, AND DURING SUBSEQUENT POWER OPERATION.

[309] TURKEY POINT 4 DOCKET 50-251 LER 87-026
COMPONENT COOLING WATER PUMP 4B AUTO-START DUE TO UNANTICIPATED DROP IN HEADER PRESSURE WHEN CCW PUMP 4C WAS RETURNED TO SERVICE.
EVENT DATE: 112187 REPORT DATE: 121887 NSSS: WE TYPE: PWR

(NSIC 207419) ON NOVEMBER 21, 1987, AT 1030, WITH UNIT 4 IN MODE 5, COMPONENT COOLING WATER (CCW) PUMP 4B AUTO-STARTED. AT THE TIME OF THIS EVENT, THE 4C CCW PUMP WAS BEING RETURNED TO SERVICE, AND A PRESSURE TEST OF THE PUMP WAS TO BE PERFORMED. IN PREPARATION FOR THIS TEST THE SUCTION ISOLATION VALVE WAS OPENED, AND THE EMPTY PUMP AND PIPING BETWEEN THE ISOLATION VALVES WAS FILLED. THIS RAPID FILL RESULTED IN A 7% DROP IN THE CCW SURGE TANK LEVEL, A DROP IN HEADER PRESSURE BELOW 75 PSIG, AND THE AUTO-START OF THE 4B CCW PUMP. FOLLOWING THE COMPLETION OF THE FILLING OPERATION THE HEADER PRESSURE QUICKLY RETURNED TO NORMAL. JUST PRIOR TO THIS EVENT, ON NOVEMBER 19, 1987, TURKEY POINT CHANGED FROM 2 PUMP TO 1 PUMP OPERATION DURING SHUTDOWN CONDITIONS. THE CAUSES OF THE EVENT WERE INSUFFICIENT CONSIDERATION OF THE DECREASED MARGIN BETWEEN THE NEW OPERATING PRESSURE AND THE AUTO-START SETPOINT UNDER SINGLE PUMP OPERATION AND INSUFFICIENT INSTRUCTIONS TO THE OPERATORS VALVING IN THE 4C PUMP ON THE PROPER TECHNIQUES FOR FILLING AND VENTING EMPTY PIPING. EFFORTS TO LOWER THE CCW DISCHARGE PRESSURE SETPOINT FURTHER HAVE BEEN INITIATED. THE TRAINING DEPARTMENT WILL EVALUATE THIS EVENT FOR INCORPORATION INTO FUTURE OPERATOR TRAINING ON THE PROPER METHODS TO FILL AND VENT EMPTY PIPING.

[310] VERMONT YANKEE DOCKET 50-271 LER 87-016
HIGH PRESSURE COOLANT INJECTION SYSTEM CAPACITY DEGRADED DUE TO AIR IN FLOW CONTROLLING TRANSMITTER SENSING LINE.
EVENT DATE: 110587 REPORT DATE: 120287 NSSS: GE TYPE: BWR
VENDOR: GEN ELECTRIC SUPPLY CO

(NSIC 207355) ON 11/5/87, WITH THE PLANT OPERATING AT A STEADY STATE POWER OF 100%, A CONTROL ROOM OPERATOR NOTED THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM FLOW CONTROLLER INDICATOR (FIK*) WAS READING 400 GPM WITH THE SYSTEM NOT IN OPERATION. HPCI WAS DECLARED INOPERABLE AT 0930. INVESTIGATION REVEALED THAT THE ERRONEOUS FLOW INDICATION WAS CAUSED BY AIR TRAPPED IN THE FLOW TRANSMITTER (FT*) SENSING LINE. THE AIR IN THE SENSING LINE CAUSED A FALSE DIFFERENTIAL PRESSURE TO BE SENSED BY THE TRANSMITTER SENSING DIAPHGRAM AND THEREFORE, ERRONEOUS FLOW INDICATION. VERMONT YANKEE TECH SPEC REQUIRE HPCI TO SUPPLY 4250 GPM OUTPUT FLOW. WITH THE FLOW CONTROLLER IN AUTO, THE OUTPUT OF THE HPCI SYSTEM WOULD HAVE BEEN 400 GPM LESS THAN THAT REQUIRED BY TECH SPEC, IF THE SYSTEM HAD BEEN CALLED UPON TO MAINTAIN REACTOR WATER LEVEL. THE TRANSMITTER WAS VENTED AND FILLED WITH WATER TO REMOVE THE AIR FROM THE SENSING LINE. SUBSEQUENT OPERABILITY TESTING WAS PERFORMED SATISFACTORILY AND HPCI WAS RETURNED TO SERVICE AT 1036. THE APPARENT ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO IMPROPER VENTING OF THE FLOW TRANSMITTER (FT*) SENSING LINES DURING THE LAST CALIBRATION CYCLE. PRESENTLY, CAUTION NOTES ARE BEING ADDED TO THE APPLICABLE PROCEDURE TO STRESS THE IMPORTANCE OF PROPER VENTING TECHNQUES WHILE PERFORMING CALIBRATION.

[311] VERMONT YANKEE DOCKET 50-271 LER 87-017
MAIN TURBINE TRIP AND REACTOR SCRAM FROM FEEDWATER VALVE MALFUNCTION DUE TO PERSONNEL VALVE REPAIR ERROR.
EVENT DATE: 110887 REPORT DATE: 120487 NSSS: GE TYPE: BWR
VENDOR: FISHER CONTROLS CO.

(NSIC 207320) ON 11/8/87, AT 0124, DURING A PLANT POWER REDUCTION FROM 100% TO 80% IN PREPARATION FOR SURVEILLANCE TESTING, THE TWO FEEDWATER REGULATING VALVES (EIIS-FCV) FAILED TO THROTTLE CLOSED AS REQUIRED BY DECREASED FEEDWATER FLOW DEMAND. THIS WAS DUE TO A MECHANICAL RESTRICTION OF THE MANUAL OPERATORS ON THESE VALVES THAT WAS ADDED APPROXIMATELY THREE WEEKS EARLIER TO TEMPORARILY REPAIR THE OPERATORS. WHEN INTALLED, IT WAS NOT RECOGNIZED THAT FUTURE FULL STROKE VALVE MOTION WOULD BE RESTRICTED BY THE REPAIR. THE FAILURE OF THESE VALVES TO CLOSE AS REQUIRED CREATED A HIGH REACTOR WATER LEVEL AND SUBSEQUENT TURBINE TRIP AND REACTOR SCRAM AT 0126. ALL AUTOMATIC ACTIONS OCCURRED AS REQUIRED. ALL SYSTEMS WERE STABILIZED AND RETURNED TO NORMAL WITHIN TWO MINUTES. AN UNEXPECTED PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) ACTUATION OF THE MAIN STEAM ISOLATION VALVES (MSIV'S) OCCURRED DURING OPERATOR SCRAM RESPONSE AS A RESULT OF THE REACTOR MODE SWITCH BEING MOVED FROM RUN. THIS SWITCH MOVEMENT WAS DONE PRIOR TO MAIN STEAM FLOW INSTRUMENTATION RESETTING FROM DECREASED STEAM FLOW FROM THE SCRAM. PCIS ISOLATIONS WERE PROMPTLY RESET. VALVE OPERATOR RESTRICTIONS WERE REMOVED AND REPAIRED AND FULL STROKE OF THE VALVES WAS VERIFIED. PLANT PERSONNEL WILL BE FURTHER TRAINED IN THE PROPER OPERATION OF THESE VALVES.

[312] VERMONT YANKEE DOCKET 50-271 LER 87-018
REACTOR CORE ISOLATION COOLING SYSTEM INOPERABLE DUE TO DAMAGED TURBINE EXHAUST CHECK VALVE.
EVENT DATE: 111487 REPORT DATE: 121187 NSSS: GE TYPE: BWR
VENDOR: WALWORTH COMPANY

(NSIC 207401) ON 11/14/87, WITH THE PLANT OPERATING AT 100% POWER, A SURVEILLANCE FULL FLOW TEST OF THE REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM WAS INITIATED AT 0357. IMMEDIATELY AFTER INITIATION, A RCIC TURBINE TRIP OCCURRED DUE TO HIGH RCIC TURBINE EXHAUST PRESSURE. REPEATED START ATTEMPTS ALSO RESULTED IN THE SAME TRIP. RCIC WAS THEN DECLARED INOPERABLE AT 0415. INVESTIGATION REVEALED A FAILED CHECK VALVE (EIIS=V) IN THE RCIC TURBINE EXHAUST LINE. THE DISK OF THE CHECK VALVE HAD BROKEN AWAY FROM THE DISC ARM AND LODGED IN THE VALVE BODY, THUS RESTRICTING TURBINE EXHAUST FLOW. THE BROKEN DISC WAS REPLACED AND THE RCIC SYSTEM TESTED. THE SYSTEM WAS DECLARED OPERABLE AT 1845 ON 11/18/87. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO FATIGUE FAILURE OF THE DISC DUE TO REPEATED DISC TO VALVE BODY CONTACT DURING VALVE CYCLING. TO PREVENT RECURRENCE OF THIS FAILURE, THE SUBJECT VALVE WILL BE REGULARLY EVALUATED PER A REVISION TO THE VERMONT YANKEE INSERVICE TESTING PROGRAM.

[313] VOGTLE 1 DOCKET 50-424 LER 87-055
CLOSURE OF RHR SYSTEM VALVES CAUSES LOSS OF AVAILABILITY OF ONE RHR PUMP.
EVENT DATE: 031887 REPORT DATE: 100587 NSSS: SS TYPE: PWR

(NSIC 206748) ON MARCH 18, 1987, WHILE IN MODE 3, THE RESIDUAL HEAT REMOVAL (RHR) SYSTEM DISCHARGE VALVES (1-HV-8809 A&B) WERE TEMPORARILY CLOSED IN ORDER TO PERFORM SCHEDULED SURVEILLANCE ACTIVITIES PER THE PLANT'S TECHNICAL SPECIFICATIONS. SUBSEQUENT REVIEW IDENTIFIED THAT THE VALVE CLOSURES WERE SIMILAR TO THOSE ADDRESSED IN NRC IE INFORMATION NOTICE 87-01, "RHR VALVE MISALIGNMENT CAUSES DEGRADATION OF ECCS IN PWRS." THE DIRECT CAUSE OF THE EVENT WAS THAT PLANT SURVEILLANCE PROCEDURES IMPROPERLY DIRECTED OPERATORS TO CLOSE THESE VALVES TO CONDUCT SURVEILLANCE TESTING. A CONTRIBUTORY CAUSE WAS THE FAILURE TO MAKE A PROMPT EVALUATION OF THE INFORMATION PROVIDED BY THE NRC INFORMATION NOTICE. CORRECTIVE ACTIONS INCLUDE REVISING APPROPRIATE SURVEILLANCE

PROCEDURES, INSTITUTING ADMINISTRATIVE CONTROLS TO PREVENT CLOSURE OF THESE VALVES IN MODES 1, 2, OR 3, AND CONDUCTING REVIEW SESSIONS WITH PLANT OPERATORS AND EXEMPT PERSONNEL.

[314] VOGTLE 1 DOCKET 50-424 LER 87-036 REV 01
UPDATE ON AUXILIARY FEEDWATER ACTUATION CIRCUITRY INOPERABLE DUE TO PERSONNEL ERROR.
EVENT DATE: 061587 REPORT DATE: 110287 NSSS: SS TYPE: PWR

(NSIC 207116) ON JUNE 15, 1987 AT APPROXIMATELY 0600 CDT WITH THE UNIT IN MODE 3, AN OPERATOR DISCOVERED THAT THE AUXILIARY FEEDWATER (AFW) ACTUATION CIRCUITRY HAD BEEN DISABLED FOR APPROXIMATELY 11 HOURS. THIS WAS CONTRARY TO TECHNICAL SPECIFICATION REQUIREMENTS FOR MODE 3. A CLEARANCE, WRITTEN TO SUPPORT MAINTENANCE ACTIVITIES ON THE MAIN FEEDWATER (MFW) PUMPS, ALLOWED THE CIRCUITRY TO BE IMPROPERLY DISABLED AT 1904 CDT ON JUNE 14, 1987. THE CLEARANCE WAS MEANT TO ALLOW BLOCKING AN AFW SYSTEM ACTUATION UPON RECEIPT OF A MFW PUMP TRIP SIGNAL. ON JUNE 22, 1987, DURING EVENT INVESTIGATION, A REVIEW OF PREVIOUS AFW CLEARANCES REVEALED A SIMILAR EVENT HAD OCCURRED ON JUNE 3, 1987. THE AFW ACTUATION CIRCUITRY HAD ALSO BEEN DISABLED IN THE SAME MANNER AS THE EVENT ON JUNE 15, AND WAS DISABLED FOR A PERIOD OF APPROXIMATELY FORTY-NINE (49) HOURS. PLANT PERSONNEL DISABLED THE AFW VALVE ACTUATION CIRCUITRY BY PULLING INCORRECT FUSES DUE TO INCORRECTLY USING AND INTERPRETING A PROCEDURE. THIS WAS CAUSED BY A PERSONNEL ERROR COMPOUNDED BY PROCEDURAL INADEQUACY. THE CLEARANCE WAS RELEASED AND THE IMPROPERLY PULLED FUSES WERE REINSTALLED. PROCEDURAL CHANGES HAVE BEEN MADE TO CLEARLY DESIGNATE THE PROPER FUSES TO BE PULLED FOR BLOCKING SPECIFIC FEATURES OF THE AFW ACTUATION CIRCUITRY.

[315] VOGTLE 1 DOCKET 50-424 LER 87-057 REV 01
UPDATE ON PROCEDURE DEFICIENCY RESULTS IN FAILURE TO TRIP OVERTEMP DELTA T REACTOR TRIP BISTABLE.
EVENT DATE: 080887 REPORT DATE: 111087 NSSS: SS TYPE: PWR
VENDOR: WESTON ELECTRIC INSTRUMENT

(NSIC 207118) ON AUGUST 8, 1987, WITH UNIT 1 OPERATING IN MODE 1 AT 90% REACTOR POWER, IT WAS DISCOVERED THAT A PRESSURIZER PRESSURE CHANNEL WOULD NOT PASS CHANNEL CHECK REQUIREMENTS. A LIMITING CONDITION FOR OPERATION (LCO) WAS ENTERED AT 1920 CDT AND THE ASSOCIATED HIGH PRESSURE REACTOR TRIP, LOW PRESSURE REACTOR TRIP, AND LOW PRESSURE SAFETY INJECTION ACTUATION BISTABLES WERE SUBSEQUENTLY TRIPPED PER PROCEDURE 18001-1 "PRIMARY SYSTEMS INSTRUMENTATION MALFUNCTION". AT 0600 CDT ON AUGUST 9, 1987, THE SHIFT SUPERVISOR OBSERVED THAT THE OVERTEMPERATURE DELTA T (OTDT) REACTOR TRIP BISTABLE WAS NOT TRIPPED AND CONCLUDED THE LCO ACTION REQUIREMENTS OF TECHNICAL SPECIFICATION 3.3.1 ITEM 7 HAD NOT BEEN COMPLIED WITH. A REVIEW OF PROCEDURE 18001-1 REVEALED A PROCEDURAL DEFICIENCY HAD RESULTED IN THE FAILURE TO TRIP THE OTDT BISTABLE. THE ASSOCIATED OTDT BISTABLE WAS TRIPPED AT 0830 CDT ON AUGUST 9, 1987. THIS EVENT WAS DETERMINED TO BE REPORTABLE ON SEPTEMBER 17, 1987. THE FAILED PRESSURIZER PRESSURE TRANSMITTER WAS REPLACED AND THE LCO WAS EXITED AT 0530 CDT ON AUGUST 10, 1987. PROCEDURE 18001-1 HAS BEEN REVISED TO ADD THE OTDT REACTOR TRIP BISTABLES TO THE LIST OF BISTABLES TO BE TRIPPED FOR A PRESSURIZER PRESSURE INSTRUMENT FAILURE.

[316] VOGTLE 1 DOCKET 50-424 LER 87-060
CONTROL ROOM ISOLATION ACTUATION DUE TO AN INADEQUATE PROCEDURE.
EVENT DATE: 102687 REPORT DATE: 112487 NSSS: SS TYPE: PWR

(NSIC 207280) ON OCTOBER 26, 1987 AT APPROXIMATELY 0550 CST, WITH UNIT 1 IN MODE 5 (COLD SHUTDOWN), A CONTROL ROOM ISOLATION (CRI) OCCURRED DUE TO A HIGH LEVEL ALARM OF RADIATION MONITOR 1RE-12117, THE TRAIN "B" CONTROL ROOM GASEOUS

RADIATION MONITOR. A CHEMISTRY EMPLOYEE (CONTRACTOR) INITIATED THE HIGH LEVEL ALARM WHILE RESETTING THE FUNCTION SETPOINTS. THIS EVENT WAS CAUSED BY AN INADEQUATE PROCEDURE, WHICH RESULTED IN A CHEMISTRY TECHNICIAN INITIATING THE HIGH LEVEL ALARM WHILE RESETTING THE FUNCTION SETPOINTS,. I.E., THE GAIN, THE HIGH-LEVEL ALARM, THE ALERT ALARM, THE LOW-FAIL AND BACKGROUND SETPOINTS. CORRECTIVE ACTIONS INCLUDED ISSUING A LAB STANDING ORDER TO PROVIDE INSTRUCTIONS FOR RESETTING THE SETPOINTS WITHOUT ACTIVATING A HIGH LEVEL ALARM AND AN OPERATIONS STANDING ORDER TO PROVIDE INSTRUCTIONS TO BLOCK THE CRI PRIOR TO RE-ENERGIZING THE ELECTRICAL PANELS. THESE STANDING ORDERS WILL BE INCORPORATED INTO PLANT PROCEDURES.

[317] VOGTLE 1 DOCKET 50-424 LER 87-061
FAILURE TO RECORD LIMITING CONDITIONS OF OPERATION ALLOWS IMPROPER MODE CHANGE.
EVENT DATE: 102887 REPORT DATE: 113087 NSSS: SS TYPE: PWR

(NSIC 207281) ON OCTOBER 28, 1987, AT 1742 CST WITH REACTOR COOLANT TEMPERATURE AND PRESSURE AT APPROXIMATELY 340 DEGREES FAHRENHEIT AND 350 PSIG RESPECTIVELY, UNIT 1 WENT FROM MODE 4 (HOT SHUTDOWN) TO MODE 3 (HOT STANDBY) WITHOUT COMPLETING ALL REQUIRED ACTIONS. TWO (2) ACTIONS HAD NOT BEEN COMPLETED PRIOR TO THE CHANGE TO MODE 3 RESULTING IN A VIOLATION OF TECHNICAL SPECIFICATIONS PARAGRAPH 3.0.4. THE TWO ACTIONS CONSISTED OF A FAILURE TO PERFORM FUNCTIONAL TESTS FOR THE SAFETY INJECTION (SI) PUMP "A" AND AFW PUMP TRAIN "A" DISCHARGE CHECK VALVE. THE FAILURE TO PERFORM THE FUNCTIONAL TESTS WAS CAUSED BY THE FAILURE TO PROPERLY IMPLEMENT PROCEDURE 10008-C, "RECORDING LIMITING CONDITIONS FOR OPERATION". CORRECTIVE ACTIONS CONSIST OF COUNSELING THE PERSONNEL INVOLVED AND ASSIGNING A MODE RESTRAINT TO THE MAINTENANCE WORK ORDER (MWO) FOR EQUIPMENT REQUIRED TO BE OPERABLE FOR THAT MODE.

[318] VOGTLE 1 DOCKET 50-424 LER 87-062
INADEQUATE PROCEDURE ALLOWS IMPROPER AFW VALVE LINEUP.
EVENT DATE: 102887 REPORT DATE: 113087 NSSS: SS TYPE: PWR

(NSIC 207282) ON OCTOBER 28, 1987, AT 1742 CST, UNIT 1 WENT FROM MODE 4 (HOT SHUTDOWN) TO MODE 3 (HOT STANDBY) WITHOUT COMPLETING ALL REQUIRED ACTIONS. ONE ACTION HAD NOT BEEN COMPLETED PRIOR TO THE CHANGE TO MODE 3 RESULTING IN A VIOLATION OF TECHNICAL SPECIFICATIONS PARAGRAPH 3.0.4. THIS ACTION CONSISTED OF AN IMPROPER VALVE LINEUP OF THE AUXILIARY FEEDWATER (AFW) SYSTEM. THE IMPROPER LINE-UP OF THE AFW SYSTEM WAS CAUSED BY AN INADEQUATE PROCEDURE. CORRECTIVE ACTIONS CONSIST OF REVISING THE MODE 4 TO MODE 3 PROCEDURE, 12002-1, AND COUNSELING THE PERSONNEL INVOLVED.

[319] VOGTLE 1 DOCKET 50-424 LER 87-063
REACTOR TRIP FOLLOWING TURBINE TRIP CAUSED BY VIBRATION MONITOR CABLE MOVEMENT.
EVENT DATE: 110587 REPORT DATE: 120487 NSSS: SS TYPE: PWR

(NSIC 207309) ON NOVEMBER 5, 1987, PLANT PERSONNEL WERE WORKING ON LEVEL THREE (3) OF THE TURBINE BUILDING. AT 0940 CST WITH UNIT 1 AT 100% RATED THERMAL POWER, A TURBINE TRIP OCCURRED WHEN A TURBINE VIBRATION MONITOR ACTUATED. THE TURBINE TRIP GENERATED A REACTOR TRIP SIGNAL. THE MAIN FEEDWATER SYSTEM ISOLATED AND THE AUXILIARY FEEDWATER SYSTEM ACTUATED. PLANT EQUIPMENT RESPONDED AS DESIGNED AND PLANT OPERATORS STABILIZED THE STEAM GENERATOR WATER LEVELS BY 1001 CST. THE APPARENT CAUSE OF THIS EVENT WAS THE ACTUATION OF VIBRATION MONITOR XE6371. THE ACTUATION WAS CAUSED WHEN THE MONITOR'S CABLING WAS MOVED BY PLANT PERSONNEL PERFORMING WORK ON NEARBY COMPONENTS. CORRECTIVE ACTION INCLUDES LABELING THE CABLES TO ADVISE PLANT PERSONNEL OF THE CABLING'S POTENTIAL TO TRIP THE PLANT.

[320] VOGTLE 1 DOCKET 50-424 LER 87-064
AUXILIARY FEEDWATER PUMP ACTUATION FOLLOWING A CONDENSATE PUMP WORK ACTIVITY.
EVENT DATE: 110587 REPORT DATE: 120487 NSSS: SS TYPE: PWR

(NSIC 207310) AT 2240 CST, ON NOVEMBER 5, 1987, WITH UNIT 1 IN MODE 3 (HOT STANDBY), AN AUTO-START SIGNAL WAS INITIATED FOR THE MOTOR DRIVEN AUXILIARY FEEDWATER (MDAFW) PUMPS WHEN AN OPERATOR, PERFORMING A CLEARANCE OF A STANDBY CONDENSATE PUMP (CP), INADVERTENTLY OPERATED THE WRONG HANDSWITCH ON THE LOCAL CONDENSATE VALVE CONTROL PANEL. THE DISCHARGE VALVE FOR THE OPERATING CP CLOSED WHICH CAUSED A TRIP OF A MAIN FEEDWATER (MFW) PUMP ON LOW SUCTION PRESSURE (THE OTHER MFW PUMP WAS IN THE TRIPPED POSITION). THE MFW PUMP TRIP RESULTED IN 1) A START SIGNAL TO THE MDAFW PUMPS WHICH WERE ALREADY RUNNING, AND 2) THE AFW DISCHARGE FLOW CONTROL VALVES STROKED FULL OPEN. CONTROL ROOM OPERATORS RESPONDED QUICKLY TO THROTTLE THE AFW DISCHARGE CONTROL VALVES AND WERE ABLE TO MAINTAIN STEAM GENERATOR WATER LEVELS WITHIN THE DESIRED RANGE. THE LABELS ON THE LOCAL CONDENSATE CONTROL PANEL CONTRIBUTED TO THIS EVENT'S CAUSE. THE LABELS WILL BE MODIFIED FOR THIS LOCAL PANEL AND A REVIEW WILL BE PERFORMED OF OTHER REMOTE PANELS TO IDENTIFY IF SIMILAR LABELING MODIFICATIONS ARE NEEDED.

[321] VOGTLE 1 DOCKET 50-424 LER 87-065
CONTAINMENT VENTILATION ISOLATION DUE TO ACTUATOR FAILURE AND SOFTWARE DESIGN.
EVENT DATE: 110987 REPORT DATE: 120987 NSSS: SS TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 207311) AT 2347 CST ON NOVEMBER 9, 1987, WITH UNIT 1 IN MODE 1 (POWER OPERATION) AT 98 PERCENT RATED THERMAL POWER, A CONTAINMENT VENTILATION ISOLATION (CVI) ACTUATION OCCURRED DUE TO A HIGH LEVEL ALARM OF CONTAINMENT VENT EFFLUENT MONITOR, 1RE-2565C. THIS EVENT WAS CAUSED BY A COMBINATION OF THE FAILURE OF THE CHECK SOURCE ACTUATOR TO COMPLETELY RETRACT TO ITS STORAGE POSITION AND BY AN APPARENT DESIGN INADEQUACY OF THE SOFTWARE, I.E., THE STATISTICAL COUNTING ALGORITHM WITHIN THE DATA PROCESSING MODULE (DPM) FOR THE MONITOR. THIS ALGORITHM PROVIDES AN ANTICIPATORY HIGH LEVEL ALARM. A CONTRIBUTING CAUSE WAS THE LACK OF ACTION TAKEN, DUE TO THE MONITOR BEING IN AN LCO CONDITION, WHEN AN ALERT ALARM WAS RECEIVED IN THE CONTROL ROOM. CORRECTIVE ACTIONS TAKEN INCLUDE REPAIRING THE ACTUATOR AND PURSUING A CHANGE FOR THE SOFTWARE, I.E., THE STATISTICAL COUNTING ALGORITHM, TO PROVIDE A MORE REALISTIC ANTICIPATORY HIGH LEVEL ALARM. COUNSELING OF THE INVOLVED OPERATIONS PERSONNEL WAS CONDUCTED CONCERNING THE IMPORTANCE OF PROCEDURAL COMPLIANCE.

[322] VOGTLE 1 DOCKET 50-424 LER 87-066
INADEQUATE LABELING CAUSES A PERSONNEL ERROR WHICH CAUSES A REACTOR TRIP.
EVENT DATE: 111187 REPORT DATE: 121087 NSSS: SS TYPE: PWR

(NSIC 207483) ON NOVEMBER 11, 1987, AT 0344 CST WITH UNIT 1 IN MODE 1 (POWER OPERATION) AT 100 PERCENT RATED THERMAL POWER, AN UNPLANNED ACTUATION OF THE REACTOR PROTECTION SYSTEM OCCURRED. PLANT PERSONNEL WERE CONDUCTING A SURVEILLANCE TEST OF THE TRAIN "A" REACTOR TRIP BREAKERS AS REQUIRED BY THE UNIT'S TECHNICAL SPECIFICATIONS. AS PART OF THIS TEST, THE TRAIN "A" REACTOR TRIP BREAKERS HAD BEEN OPENED. A SHIFT TECHNICAL ADVISOR (STA) DEPRESSED A SHUNT TRIP TEST PUSHBUTTON ON THE OPPOSITE TRAIN (TRAIN "B") TRIP BREAKER PANEL CABINET AND A REACTOR TRIP OCCURRED. THE TRAIN "B" MOTOR DRIVEN AUXILIARY FEEDWATER (MDAFW) PUMP AUTOMATICALLY STARTED. THE TRAIN "A" MDAFW PUMP WAS MANUALLY STARTED BECAUSE ITS AUTOMATIC INITIATION CIRCUIT WAS BLOCKED AS PART OF THE REACTOR TRIP BREAKERS SURVEILLANCE TESTING. THE TURBINE DRIVEN AFW PUMP AUTOMATICALLY STARTED ON LOW-LOW STEAM GENERATOR WATER LEVEL. OPERATORS STABILIZED PLANT CONDITIONS IN MODE 3 FOLLOWING THE REACTOR TRIP. THIS EVENT WAS CAUSED BY PERSONNEL ERROR, IN THAT AN STA OPENED THE WRONG TRIP BREAKER CABINET AND PUSHED THE OPPOSITE TRAIN'S SHUNT TRIP TEST PUSHBUTTON. CORRECTIVE ACTION INCLUDES BETTER LABELING OF EACH CABINET, REVISING THE SURVEILLANCE PROCEDURE TO

INCLUDE SPECIFIC PANEL AND PUSHBUTTON NUMBERS, AND POSITIVE DISCIPLINE OF APPROPRIATE INDIVIDUALS.

[323] VOGTLE 1 DOCKET 50-424 LER 87-067
TECHNICAL SPECIFICATION SURVEILLANCE OF AUXILIARY BUILDING MISSED DUE TO PERSONNEL ERROR.
EVENT DATE: 111687 REPORT DATE: 121687 NSSS: SS TYPE: PWR

(NSIC 207484) ON NOVEMBER 16, 1987, AT APPROXIMATELY 0700 CST WITH UNIT 1 AT 89 PERCENT RATED THERMAL POWER, A PLANT EQUIPMENT OPERATOR (PEO) ATTEMPTED TO ENTER ROOM 110 OF THE AUXILIARY BUILDING, THROUGH ROOM 118, TO PERFORM A REQUIRED ROOM TEMPERATURE READING. HE NOTED ON THE SHIFT AND DAILY SURVEILLANCES DATA SHEET (DATA SHEET) THAT THE TEMPERATURE READING COULD NOT BE TAKEN BECAUSE OF WET PAINT ON THE FLOOR OF ROOM 118 (THE ONLY ACCESS TO ROOM 110). THE DAY-SHIFT SHIFT SUPERVISOR REVIEWED THE DATA SHEET, BUT REQUIRED NO ADDITIONAL ACTION TO BE TAKEN. AT APPROXIMATELY 1900 CST, ANOTHER PEO AGAIN NOTED ON THE DATA SHEET THAT ROOM 110 WAS INACCESSIBLE. AT 2200 CST, THE NIGHT-SHIFT SHIFT SUPERVISOR, WHO WAS ADVISED OF THIS CONDITION, IMMEDIATELY TOOK ACTION TO OPEN ROOM 118. A TEMPERATURE READING OF 76 DEGREES F WAS OBTAINED AT 2225 CST. THE CAUSE OF THIS EVENT IS THE DAY-SHIFT SHIFT SUPERVISOR'S FAILURE TO OBTAIN THE TEMPERATURE READING. CORRECTIVE ACTION INCLUDES COUNSELING OF THE PERSONNEL INVOLVED.

[324] VOGTLE 1 DOCKET 50-424 LER 87-070
INADEQUATE REVIEWS OF SPECIAL CONDITION SURVEILLANCE LOGS LEAD TO MISSED SURVEILLANCES.
EVENT DATE: 112387 REPORT DATE: 121887 NSSS: SS TYPE: PWR

(NSIC 207486) ON NOVEMBER 22, 1987 AT 1240 CST, THE QUADRANT POWER TILT RATIO (QPTR) WAS CALCULATED AND FOUND TO BE WITHIN ITS ALLOWABLE LIMIT. PER TECHNICAL SPECIFICATION (T.S.) 4.2.4.1, THE QPTR IS TO BE CALCULATED EVERY TWELVE (12) HOURS WHEN THE QPTR ALARM IS INOPERABLE. AS THE ALARM WAS INOPERABLE ON NOVEMBER 22, 1987, THE NEXT CALCULATION WAS TO BE PERFORMED AT 0040 CST ON NOVEMBER 23, 1987 (0340 CST ALLOWING A MAXIMUM 25 PERCENT TIME EXTENSION). THE NEXT CALCULATION WAS COMPLETED AT 0415 CST ON NOVEMBER 23, 1987, THIRTY-FIVE (35) MINUTES AFTER THE EXPIRATION OF THE MAXIMUM ALLOWABLE TIME EXTENSION. ON DECEMBER 2, 1987 AT 1208 CST, THE ALARM WAS STILL INOPERABLE WHEN THE QPTR WAS CALCULATED. PER T.S. 4.2.4.1, THE QPTR WAS TO BE CALCULATED BY 0008 CST ON DECEMBER 3, 1987. A QPTR WAS CALCULATED BY 0311 CST, 3 MINUTES AFTER THE EXPIRATION OF THE MAXIMUM ALLOWABLE EXTENSION OF THE SURVEILLANCE INTERVAL. THE CAUSE OF THIS EVENT IS INADEQUATE REVIEWS OF THE SPECIAL CONDITION SURVEILLANCE LOGS PER PROCEDURE 14000-1, "OPERATION SHIFT AND DAILY SURVEILLANCE LOGS" BY THE CONTROL ROOM SUPERVISORY PERSONNEL. CORRECTIVE ACTION INCLUDES COUNSELING OF THE PERSONNEL INVOLVED.

[325] WATERFORD 3 DOCKET 50-382 LER 87-024
DIESEL GENERATOR FUEL OIL STORAGE TANK VOLUME REQUIREMENTS NOT MET.
EVENT DATE: 091187 REPORT DATE: 101287 NSSS: CE TYPE: PWR

(NSIC 206662) AT 1630 HOURS ON SEPTEMBER 11, 1987, WATERFORD STEAM ELECTRIC STATION UNIT 3 WAS OPERATING AT 100% POWER WHEN IT WAS DETERMINED THAT THE DIESEL FUEL OIL STORAGE TANK (DFOST) LEVEL CURVES HAD BEEN MISINTERPRETED, CAUSING THE TECHNICAL SPECIFICATION MINIMUM VOLUME REQUIREMENT FOR THE DFOST TO NOT BE MET ON SEVERAL OCCASIONS SINCE INITIAL PLANT STARTUP; CONSEQUENTLY, THE PLANT OPERATED IN A CONDITION PROHIBITED BY THE TECHNICAL SPECIFICATIONS DURING THESE TIME INTERVALS. THIS EVENT WAS DISCOVERED IN A REVIEW OF TECHNICAL SPECIFICATION LIMITING CONDITIONS FOR OPERATION WHICH WAS CONDUCTED AS A RESULT OF LER 87-009-00. THE ROOT CAUSE OF THIS EVENT WAS PROCEDURAL IN THAT THE REQUIRED VOLUME IS CHECKED BY MEASURING TANK LEVEL, AND THE MINIMUM LEVEL SPECIFIED IN

SURVEILLANCE PROCEDURES CORRESPONDED TO A TOTAL VOLUME OF OVER 38,760 GALLONS, BUT DID NOT ACCOUNT FOR APPROXIMATELY 900 GALLONS OF UNUSABLE VOLUME WHICH IS BELOW THE TRANSFER PUMP SUCTION LINE. SURVEILLANCE REQUIREMENTS FOR TANKS WITH AVAILABLE VOLUMES SPECIFIED BY TECHNICAL SPECIFICATIONS HAVE BEEN CHECKED AND NO PROBLEMS WERE FOUND. SINCE THE SHORTFALL IN TANK VOLUME WAS LESS THAN TWO PERCENT AND EACH DFOST TRANSFER PUMP CAN TAKE SUCTION FROM EITHER DFOST AND SUPPLY EITHER EMERGENCY DIESEL GENERATOR FEED TANK, THERE WAS NO DEGRADATION IN THE LEVEL OF PLANT SAFETY AS A RESULT OF THIS EVENT.

[326] WATERFORD 3 DOCKET 50-382 LER 87-025
MISSED SAMPLES DUE TO INADEQUATE ADMINISTRATIVE CONTROLS.
EVENT DATE: 100587 REPORT DATE: 110487 NSSS: CE TYPE: PWR

(NSIC 207104) FROM OCTOBER 3 TO OCTOBER 16, 1987, THERE WERE FOUR INSTANCES WHEN TECHNICAL SPECIFICATION (TS) SAMPLING REQUIREMENTS WERE MISSED AT THE WATERFORD STEAM ELECTRIC STATION UNIT 3. AT 1030 HOURS ON OCTOBER 5, 1987, WHEN IN COLD SHUTDOWN, IT WAS DISCOVERED THAT PERIODIC 4 AND 8 HOUR GAS SAMPLES FROM THE ON-SERVICE GAS DECAY TANK HAD BEEN MISSED FOR APPROXIMATELY 38 HOURS. AT 2120 HOURS ON OCTOBER 14, 1987, WHILE AT 88% POWER, IT WAS DISCOVERED THAT ONE OF THE 12 HOUR MAIN CONDENSER NOBLE GAS SAMPLES HAD MISSED ITS PERIODICITY BY 30 MINUTES. AT 0930 HOURS ON OCTOBER 16, 1987, WHILE OPERATING AT 95% POWER, IT WAS DISCOVERED THAT A WEEKLY TRITIUM SAMPLE OF THE FUEL HANDLING BUILDING VENTILATION EXHAUST HAD BEEN MISSED ON OCTOBER 9. AT 1025 HOURS ON OCTOBER 20, 1987, WHILE OPERATING AT 100% POWER, IT WAS DISCOVERED THAT A WEEKLY SAMPLE OF GAS DECAY TANK NOBLE GASES HAD BEEN MISSED ON OCTOBER 16, 1987. IN ALL FOUR INSTANCES, TS SAMPLING TIME REQUIREMENTS WERE NOT MET, SO THE PLANT OPERATED IN A CONDITION PROHIBITED BY TECHNICAL SPECIFICATIONS. THE ROOT CAUSE OF THESE EVENTS WAS A PROGRAMMATIC BREAKDOWN IN ADMINISTRATIVE CONTROLS. METHODS OF TRACKING TS REQUIRED SAMPLING HAVE BEEN REVISED TO INCLUDE AN INDEPENDENT REVIEW TO ENSURE TS SAMPLING TIME REQUIREMENTS ARE MET. SINCE ALL SUBSEQUENT SAMPLES WERE WITHIN REQUIRED LIMITS, THERE WAS NO SAFETY SIGNIFICANCE TO THESE EVENTS.

[327] WATERFORD 3 DOCKET 50-382 LER 87-026
CONTAINMENT ELECTRICAL PENETRATION BACKUP PROTECTION INOPERABLE DUE TO INADEQUATE CONSTRUCTION DOCUMENTATION.
EVENT DATE: 112387 REPORT DATE: 122287 NSSS: CE TYPE: PWR

(NSIC 207464) AT 1500 HOURS ON NOVEMBER 23, 1987, WATERFORD STEAM ELECTRIC STATION UNIT 3 WAS OPERATING AT 100% POWER WHEN OPERATIONS PERSONNEL DISCOVERED THAT CHEMICAL AND VOLUME CONTROL (CVC) VALVES 218A AND 218B HAD FAILED TO MEET THE CONTAINMENT PENETRATION BACKUP OVERCURRENT PROTECTION OPERABILITY REQUIREMENT OF TECHNICAL SPECIFICATION (TS) 3.8.4.1 SINCE INITIAL PLANT STARTUP. WORK PERFORMED UNDER A CONDITION IDENTIFICATION WORK AUTHORIZATION (CIWA) CONSISTED OF A WALKDOWN OF POWER DISTRIBUTION PANELS AND IDENTIFIED 70 DISCREPANCIES. TWO ITEMS ON THIS LIST ADDRESSED POWER SUPPLY CABLES TO CVC-218A AND 218B WHICH WERE DIRECTLY CONNECTED TO CIRCUIT BREAKERS VICE CONNECTED TO FUSES IN SERIES WITH THE BREAKERS. THUS, THE PLANT OPERATED IN A CONDITION PROHIBITED BY TS SINCE THE FUSES WERE NOT OPERABLE. THE ROOT CAUSE OF THIS EVENT WAS COGNITIVE PERSONNEL ERROR. TERMINATION SHEETS USED TO CONNECT THESE CABLES DURING PLANT CONSTRUCTION INADVERTENTLY BYPASSED THE FUSE PAIRS. ALTHOUGH THE INITIAL WALKDOWN CIWA IDENTIFIED THE DISCREPANCY, ITS SIGNIFICANCE WAS NOT REALIZED BY PERSONNEL INVOLVED IN THE REVIEW AND DISPOSITION OF THE CIWA WALKDOWN RESULTS. THESE CABLES WERE PROPERLY CONNECTED AND THE SYSTEM DECLARED OPERABLE AT 1915 HOURS ON NOVEMBER 23, 1987. SINCE CVC 218A AND 218B HAVE ALWAYS OPERATED AS DESIGNED, THERE WAS NO HEALTH OR SAFETY SIGNIFICANCE TO THIS EVENT.

[328] WATERFORD 3 DOCKET 50-382 LER 87-027
INOPERABLE FIRE BARRIER DUE TO INADEQUATE DESIGN CHANGE CONTROLS DURING PLANT CONSTRUCTION.
EVENT DATE: 112587 REPORT DATE: 122187 NSSS: CE TYPE: PWR

(NSIC 207465) AT 1932 HOURS ON NOVEMBER 25, 1987, WATERFORD STEAM ELECTRIC STATION UNIT 3 WAS OPERATING AT 100% POWER WHEN OPERATIONS PERSONNEL DISCOVERED THAT AN UNIDENTIFIED PIPE IN THE INSTRUMENT REPAIR SHOP (IRS) ON THE -4 FOOT LEVEL OF THE REACTOR AUXILIARY BUILDING (RAB) CONSTITUTED A FIRE IMPAIRMENT PER TECHNICAL SPECIFICATION (TS) 3.7.11. AN HOURLY FIREWATCH PATROL WAS IMMEDIATELY INSTITUTED. THE PIPE HAD BEEN INSTALLED PER A DESIGN CHANGE NOTICE (DCN) TO DRAIN CONDENSATION FROM AN AIR HANDLING UNIT (AHU) SCHEDULED FOR INSTALLATION PER THE DCN IN THE MULTIPLEXER (MUX) ROOM ON THE +7 FOOT ELEVATION OF THE RAB. SUBSEQUENT DESIGN MODIFICATIONS CANCELLED INSTALLATION OF THIS AHU AFTER THE ASSOCIATED DRAIN PIPING HAD BEEN INSTALLED. SINCE THE DRAIN PIPE WAS NOT CAPPED AT EITHER END, THIS CONSTITUTED A FIRE IMPAIRMENT; THUS, THE PLANT OPERATED IN A CONDITION PROHIBITED BY TS. THE ROOT CAUSE OF THE EVENT WAS PROCEDURAL IN THAT ADEQUATE FIRE PROTECTION REVIEWS WERE NOT PERFORMED AND CONTROLS WERE NOT MAINTAINED ON THE DCN DURING PLANT CONSTRUCTION. THE PIPE WAS REMOVED AND THE FIRE IMPAIRMENT CLEARED BY 0850 HOURS ON DECEMBER 1, 1987. PROCEDURES HAVE BEEN ESTABLISHED WHICH PERFORM ADEQUATE FIRE PROTECTION REVIEWS OF STATION MODIFICATIONS.

[329] WOLF CREEK 1 DOCKET 50-482 LER 87-022 REV 01
UPDATE ON REACTOR TRIP CAUSED BY LOSS OF POWER TO MAIN TURBINE ELECTRO-HYDRAULIC CONTROL SYSTEM AND SUBSEQUENT ENGINEERED SAFETY FEATURES ACTUATION CAUSED BY PROCEDURAL OMISSION.
EVENT DATE: 052887 REPORT DATE: 120187 NSSS: WE TYPE: PWR
VENDOR: ILLINOIS BELL TELEPHONE
POTTER & BRUMFIELD

(NSIC 207335) ON MAY 28, 1987, A TURBINE BUILDING FAN BREAKER MALFUNCTION CAUSED A LOSS OF POWER TO 480 VOLT MOTOR CONTROL CENTER PG1:K. THIS CAUSED LOSS OF HOUSEPOWER TO THE MAIN TURBINE ELEC[illegible]DRAULIC CONTROL (EHC) SYSTEM, WHICH CAUSED CLOSURE OF THE MAIN TURBIN[illegible]L VALVES. THIS VALVE CLOSURE CAUSED STEAM GENERATOR LEVEL "SHRINK" T[illegible]LO LEVEL SETPOINT, RESULTING IN A REACTOR TRIP, MAIN TURBINE TRIP, [illegible]EEDWATER ISOLATION, AUXILIARY FEEDWATER ACTUATION, AND STEAM GENERATOR BLOWDOWN AND SAMPLE ISOLATION AT 1429 CDT. THIS LICENSEE EVENT REPORT REVISION PROVIDES THE RESULTS OF FURTHER EVALUATIONS CONDUCTED INTO THE FAN BREAKER CUBICLE FAILURE AND THE EHC RESPONSE TO THAT FAILURE. AT 2110 CDT, WHEN ATTEMPTING TO RECLOSE THE REACTOR TRIP BREAKERS (RTB'S) THE RTB'S REOPENED AND A MAIN FEEDWATER ISOLATION OCCURRED BECAUSE A NUCLEAR INSTRUMENTATION "NEGATIVE RATE TRIP" SIGNAL HAD NOT BEEN RESET FROM THE EARLIER TRIP. THIS EVENT OCCURRED AS A RESULT OF A PROCEDURAL OMISSION, WHICH HAS BEEN CORRECTED. THE APPARENT ROOT CAUSE OF THE FAN BREAKER CUBICLE MALFUNCTION WAS A LOOSE CONNECTION ON THE LINE SIDE OF THE BREAKER.

[330] WOLF CREEK 1 DOCKET 50-482 LER 87-050
CONTAINMENT ISOLATION VALVE FAILED DURING LOCAL LEAK RATE TEST CAUSING TOTAL PATH LEAKAGE TO BE ABOVE 0.6LA - DEGRADATION OF PRIMARY SAFETY BARRIER.
EVENT DATE: 101487 REPORT DATE: 111387 NSSS: WE TYPE: PWR
VENDOR: BORG-WARNER CORP.

(NSIC 207139) ON OCTOBER 16, 1987, AN EVALUATION OF THE DATA OBTAINED DURING A CONTAINMENT ISOLATION VALVE LOCAL LEAK RATE TEST PERFORMED ON OCTOBER 14 WAS COMPLETED. FROM THE EVALUATION IT WAS DETERMINED THAT THE TOTAL PATH LEAKAGE, AS CALCULATED BY THE MAXIMUM PATHWAY METHOD, FOR ALL TYPE B AND C TESTS EXCEED THE TECH SPEC LIMIT OF 0.6 LA. THE UNIT WAS IN MODE 6, REFUELING, AT THE TIME OF THE EVENT. THIS EXCESSIVE LEAKAGE IS BEING REPORTED PURSUANT TO 10CFR

50.73(A)(2)(II) AS A DEGRADATION OF A PRINCIPAL SAFETY BARRIER. THE CAUSE OF THIS EVENT WAS A CONTAINMENT ISOLATION CHECK VALVE STICKING OPEN. THE APPARENT CAUSE OF THE VALVE STICKING OPEN COULD NOT BE DETERMINED. THIS EVENT IS CONSIDERED TO BE AN ISOLATED CASE AND THEREFORE, NO OTHER CORRECTIVE ACTION IS PLANNED. THE CONTAINMENT ISOLATION VALVE IS THE PRESSURE RELIEF PATH CHECK VALVE IN THE REACTOR COOLANT PUMP SEAL WATER RETURN LINE TO THE SEAL WATER HEAT EXCHANGER. AS A CONSERVATIVE MEASURE, A SPRING, A SEAT SPRING, AND A DISC WITHIN THE VALVE WERE REPLACED. AFTER REPLACEMENT OF THESE PARTS, THE LEAKAGE WAS REDUCED WITHIN THE 0.6LA LIMIT.

[331] WOLF CREEK 1 DOCKET 50-482 LER 87-049
FAILURE TO SUPPLY TEMPORARY POWER SOURCE TO BATTERIES RESULTS IN BATTERY DISCHARGE CAUSING MULTIPLE ENGINEERED SAFETY FEATURES ACTUATIONS.
EVENT DATE: 101587 REPORT DATE: 111687 NSSS: WE TYPE: PWR

(NSIC 207203) ON OCTOBER 15, 1987 AT APPROXIMATELY 2100 CDT, CONTAINMENT PURGE ISOLATION SIGNAL (CPIS), CONTROL ROOM VENTILATION ISOLATION SIGNAL (CRVIS), AND FUEL BUILDING ISOLATION SIGNAL (FBIS) ACTUATIONS OCCURRED. ON OCTOBER 15, 1987, AT APPROXIMATELY 2332 CDT, LOW SUCTION PRESSURE FOR AUXILIARY FEEDWATER PUMP SUCTION SWITCHOVER TO THE ESSENTIAL SERVICE WATER SYSTEM, AND THE LOAD SHED AND EMERGENCY LOAD SEQUENCER ACTUATIONS OCCURRED, INITIATING THE AUTOMATIC START OF EMERGENCY DIESEL GENERATOR (D/G) 'A'. ON OCTOBER 16, 1987, AT APPROXIMATELY 1915 CDT, CPIS, CRVIS, AND FBIS ACTUATIONS OCCURRED. THESE EVENTS OCCURRED AS THE RESULT OF LOW BATTERY VOLTAGE, CAUSED BY AN UNFORESEEN EXTENSION OF AN ELECTRICAL BUS OUTAGE. THE ROOT CAUSE OF THE LOW BATTERY VOLTAGE HAS BEEN ATTRIBUTED TO COGNITIVE PERSONNEL ERROR BY OPERATIONS AND MAINTENANCE MANAGEMENT PERSONNEL IN FAILING TO PLAN FOR CAPABILITY TO PROVIDE TEMPORARY POWER SUPPLIES TO THE BATTERIES DURING AN EXTENDED, SAFETY-RELATED ELECTRICAL BUS OUTAGE. A PROCEDURE WILL BE WRITTEN FOR DE-ENERGIZING A SAFETY-RELATED 4.16 KILOVOLT DIVISION AND SUPPLYING TEMPORARY LOWER TO THE AFFECTED BATTERIES FOR MAJOR OUTAGES.

[332] WOLF CREEK 1 DOCKET 50-482 LER 87-054
ENGINEERED SAFETY FEATURES ACTUATION CAUSED BY MOISTURE INDUCED CORROSION OF AN ELECTRICAL CONNECTOR.
EVENT DATE: 112787 REPORT DATE: 122387 NSSS: WE TYPE: PWR
VENDOR: GENERAL ATOMIC CO.

(NSIC 207498) AT 1701 CST, ON NOVEMBER 27, 1987, A PROCESS RADIATION HI HI ANNUNCIATOR WAS RECEIVED IN THE CONTROL ROOM ALONG WITH A CONTAINMENT PURGE ISOLATION SIGNAL (CPIS) AND A CONTROL ROOM VENTILATION ISOLATION SIGNAL (CRVIS). THE CONTROL ROOM LICENSED REACTOR OPERATOR (RO) DETERMINED THE SOURCE WAS THE GASEOUS CHANNEL OF ONE CONTAINMENT ATMOSPHERE RADIATION MONITOR. THE REDUNDANT MONITOR INDICATED A NORMAL READING. THE RO VERIFIED NORMAL EQUIPMENT RESPONSE TO THE CPIS AND CRVIS. AFTER HAVING A SAMPLE TAKEN, WHICH DETERMINED THERE WERE NO ABNORMAL RADIATION LEVELS IN CONTAINMENT, THE RO RESET THE CPIS AND CRVIS AT 1722 CST AND RESTORED THE VENTILATION LINEUPS TO NORMAL BY 1753 CST ON NOVEMBER 27, 1987. THE EVENT WAS CAUSED BY MOISTURE INDUCED CORROSION OF A CONNECTOR ON THE RADIATION MONITOR SKID. THE SOURCE OF THE MOISTURE WAS NOT IDENTIFIED. THE CONNECTOR WAS CLEANED AND THE RADIATION MONITOR RETURNED TO OPERABLE AT 1634 CST ON NOVEMBER 28, 1987. TO PREVENT A RECURRENCE, A HEAT SHRINK MOISTURE SEAL WAS APPLIED TO THE CONNECTOR ON DECEMBER 9, 1987.

[333] WPPSS 2 DOCKET 50-397 LER 84-091 REV 01
UPDATE ON RHR ISOLATION AND REACTOR LOW WATER LEVEL TRIP.
EVENT DATE: 082384 REPORT DATE: 112087 NSSS: GE TYPE: BWR

(NSIC 207156) WITH THE REACTOR SHUTDOWN, AN RHR ISOLATION AND A REACTOR LOW WATER

LEVEL TRIP (+13") OCCURRED WHILE PLACING THE RHR SYSTEM IN THE SHUTDOWN COOLING MODE.

[334] WPPSS 2 DOCKET 50-397 LER 87-025 REV 01
UPDATE ON ENGINEERED SAFETY FEATURE ISOLATIONS AND ACTUATIONS CAUSED BY REACTOR PROTECTION SYSTEM EQUIPMENT FAILURE.
EVENT DATE: 080887 REPORT DATE: 112087 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 207219) ON AUGUST 8, 1987 THE PLANT WAS AT 85% POWER AND ON A GRADUAL POWER ASCENSION WHEN, AT 1855 HOURS, A SPURIOUS TRIP OF THE REACTOR PROTECTION SYSTEM (RPS) ELECTRICAL PROTECTION ASSEMDLY (EPA) 3A BREAKER CAUSED A LOSS OF POWER TO RPS BUS A. THE LOSS OF POWER ON RPS BUS A CAUSED A HALF-SCRAM IN RPS DIVISION A AND MULTIPLE ENGINEERED SAFETY FEATURE (ESF) ISOLATIONS AND ACTUATIONS. THE LOSS OF RPS A POWER CAUSES AN OUTBOARD NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NSSSS) ISOLATION OF GROUPS 1 (MAIN STEAM LINE DRAINS ONLY), 2, 4 (TWO DRAIN VALVE ONLV) 5, 6, AND 7. NSSSS GROUP 7 ISOLATES THE REACTOR WATER CLEANUP SYSTEM (RWCU). IN ADDITION, THE LOSS OF RPS A POWER CAUSES A NSSSS GROUP 3 (PRIMARY AND SECONDARY CONTAINMENT VENTILATION AND PURGE SYSTEMS) AND A PARTIAL GROUP 4 (MISCELLANEOUS BALANCE OF PLANT) ISOLATION AND STANDBY GAS TREATMENT (SGT) SYSTEM AND CONTROL ROO EMERGENCY FILTRATION SYSTEM ACTUATION. PLANT OPERATORS SWITCHED RPS BUS A TO ITS ALTERNATE POWER SUPPLY AND RESTORED ALL SYSTEMS TO THEIR PRE-EVENT LINEUP WITHIN 20 MINUTES. THE CAUSE OF THE EVENT WAS A SPURIOUS TRIP OF THE RPS-EPA-3A BREAKER.

[335] WPPSS 2 DOCKET 50-397 LER 87-030
UPGRADED PLANT TECHNICAL SPECIFICATION FIRE WALL NOT CONSTRUCTED TO QUALIFY AS A THREE-HOUR FIRE BARRIER AND WALL PENETRATION NOT SEALED-CAUSE UNKNOWN.
EVENT DATE: 110687 REPORT DATE: 120787 NSSS: GE TYPE: BWR

(NSIC 207351) ON 11/6/87 DURING VERIFICATION ACTIVITIES ASSOCIATED WITH AN ENGINEERING REVIEW OF FIRE-RATED ASSEMBLIES AND WALLS (REFERENCE LER 87-004), IT WAS DISCOVERED THAT 1) A PLANT TECH SPEC DESIGNATED FIRE WALL WAS NOT CONSTRUCTED TO QUALIFY AS A THREE-HOUR BARRIER AND 2) A PENETRATION IN THE WALL, CONSISTING OF A 12" HOLE WITH AN 8" REMOTE AIR SUPPLY PIPE, WAS NOT SEALED AS REQUIRED BY THE PLANT TECH SPECS. THE WALL PROVIDES A BARRIER BETWEEN FIRE AREAS TG-1 (TURBINE GENERATOR BUILDING - ELEVATION 471') AND RC-I (RADWASTE BUILDING - ELEVATION 487'). THE PENETRATION IS A REMOTE AIR INTAKE LINE FOR THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM. AS REQUIRED BY THE PLANT TECH SPECS, THE WALL AND PENETRATION WERE PLACED ON AN HOURLY FIRE TOUR. FURTHER CORRECTIVE ACTIONS INCLUDE THE PERFORMANCE OF A 100% REVIEW OF DOCUMENTATION, AND PHYSICAL WALKDOWNS OF ALL ACCESSIBLE TECH SPEC WALLS AND PENETRATIONS, TO DETERMINE THE ADEQUACY OF THE PLANT SEAL TRACKING SYSTEM (PSTS). IN ADDITION, ENGINEERING WILL CONTINUE EFFORTS TO VERIFY THE ACCURACY OF PSTS USING SEAL CONTROL, PENETRATION SCHEDULE AND OUTSTANDING CHANGE DOCUMENT INFORMATION. ALTHOUGH THE CAUSE OF THIS EVENT CANNOT BE POSITIVELY IDENTIFIED, A PROBABLE CAUSE WAS A LACK OF COMPLETE COMMUNICATION BETWEEN ARCHITECT/ENGINEER DISCIPLINES AND SEALING CONTRACTORS.

[336] YANKEE ROWE DOCKET 50-029 LER 87-014
PLANT SHUTDOWN BECAUSE OF NON-RETURN VALVE LOW NITROGEN PRESSURE INDICATION.
EVENT DATE: 112187 REPORT DATE: 123187 NSSS: WE TYPE: PWR
VENDOR: ROCKWELL-INTERNATIONAL
WHITMAN GENERAL CORP.

(NSIC 207512) ON 11/21/87, AT 0615 HOURS, DURING NORMAL STEADY STATE MODE 1 OPERATION, THE LOW NITROGEN PRESSURE ALARM FOR MAIN STEAM SYSTEM'S NUMBER 2 NON-RETURN VALVE (NRV) WAS RECEIVED IN THE CONTROL ROOM. AT 0710 HOURS, THE NRV WAS DECLARED INOPERABLE AND A PLANT SHUTDOWN WAS INITIATED IN ACCORDANCE WITH THE

REQUIREMENTS OF TECH SPECS 3.7.1.5. THE NRC WAS NOTIFIED OF THE LOAD REDUCTION AT 0810 HOURS. AT 0834 HOURS, AN UNUSUAL EVENT WAS DECLARED AND THE APPROPRIATE STATE NOTIFICATIONS WERE MADE. THE NRC WAS NOTIFIED OF THE UNUSUAL EVENT AT 0905 HOURS. AT 1008 HOURS, THE PLANT SHUTDOWN WAS COMPLETE AND THE NRV WAS CLOSED, WITH THE EXISTING NITROGEN PRESSURE. AT 1014 HOURS, THE PLANT SECURED FROM THE UNUSUAL EVENT. A NEW LOW ACCUMULATOR NITROGEN PRESSURE SWITCH WAS CALIBRATED AND INSTALLED. THE NITROGEN ACCUMULATOR FOR NO. 2 NRV WAS CHARGED USING TEST EQUIPMENT, FOLLOWING TWO UNSUCCESSFUL ATTEMPTS WITH THE INSTALLED EQUIPMENT, THE VALVE REOPENED AND THE LOW PRESSURE ALARM CLEARED. AFTER SATISFACTORY TEST RESULTS THE NRV WAS DECLARED OPERATIONAL AT 2140 HOURS. THE ROOT CAUSE OF THIS EVENT WAS ATTRIBUTED TO A LOW NITROGEN PRECHARGE RESULTING FROM PRESSURE AND TEMPERATURE INSTRUMENTATION INACCURACY. THE PROCEDURE USED TO CHARGE NITROGEN TO THE NRV'S ACCUMULATOR WILL BE CHANGED.

[337] YANKEE ROWE DOCKET 50-029 LER 87-015
MAIN STEAM LINE PRESSURE SWITCHES INOPERABLE.
EVENT DATE: 120387 REPORT DATE: 010488 NSSS: WE TYPE: PWR
VENDOR: AUTOMATIC SWITCH COMPANY (ASCO)

(NSIC 207513) ON DECEMBER 3, 1987, WITH THE PLANT IN MODE 1 AT 100% POWER, FIVE OF THE TWELVE MAIN STEAM LINE (EIIS-SB) ISOLATION VALVE PRESSURE SWITCHES (IEEE-PS) WERE FOUND DURING ROUTINE MONTHLY SURVEILLANCE, TO BE OUT OF TECH SPEC TOLERANCE (TECH SPEC TABLE 3.3.-2, ITEM 3.A). OF THE FIVE PRESSURE SWITCHES THAT FAILED THE SURVEILLANCE, THREE WERE ADJUSTED BACK TO THE REQUIRED TRIP SET-POINT. THE REMAINING TWO (MS-PS-13 AND MS-PS-31) WERE REPLACED IN KIND BECAUSE OF THEIR FAILURE TO OPERATE PROPERLY. THE CAUSE OF THE OUT OF TOLERANCE SET-POINT SETTINGS FOR THE THREE READJUSTED SWITCHES IS ATTRIBUTED TO SET-POINT DRIFT. THE CAUSE FOR THE REMAINING TWO FAILURES IS UNDER INVESTIGATION. A PLANT SHUTDOWN WAS INITIATED AT 1140 HOURS IN ACCORDANCE WITH TECH SPEC 3.0.3 AND TERMINATED AT 1145 UPON IMPLEMENTATION OF A BYPASS JUMPER TO ACCOMPLISH THE TECH SPEC ACTION STATEMENT BY PLACING THE SWITCH IN THE TRIP POSITION. THE PLANT RETURNED TO 100% POWER AT 1210 ON THE SAME DAY. THE NRC WAS NOTIFIED VIA ENS AT 1340 HOURS DECEMBER 3, 1987. THERE WAS NO ADVERSE EFFECT TO THE HEALTH AND SAFETY OF THE PUBLIC AS A RESULT OF THIS EVENT, SINCE THE MSIV TRIP CIRCUITRY WAS PLACED IN A MORE CONSERVATIVE CONFIGURATION.

[338] ZION 2 DOCKET 50-304 LER 85-029 REV 02
UPDATE ON PURGE ISOLATION DUE TO LOW TEMPERATURE AND HIGH RADIATION SIGNAL.
EVENT DATE: 120185 REPORT DATE: 120887 NSSS: WE TYPE: PWR

(NSIC 207397) ON DECEMBER 1, 1985 WITH UNIT 2 IN COLD SHUTDOWN, DURING A UNIT 2 CONTAINMENT PURGE, THE RUNNING 2A PURGE SUPPLY FAN AND 2A EXHAUST FAN TRIPPED, THE CONTAINMENT PURGE INLET AND OUTLET ISOLATION VALVES 2AOV-RV0001, 2AOV-RV0002, 2AOV-RV0003, 2AOV-RV004 CLOSED, AND THE "AIR EXHAUST STACK RADIATION HIGH" ANNUNCIATOR ALARMED. THE CAUSE OF THIS ISOLATION WAS A SPURIOUS HIGH RADIATION ALARM FROM CONTAINMENT PURGE EXHAUST STACK AIR PARTICULATE MONITOR 2RT-PR09C WHICH ISOLATED THE PURGE INLET AND OUTLET VALVES AND TRIPPED THE RUNNING PURGE AND EXHAUST FANS. THE ROOT CAUSE OF THIS EVENT WAS A SPURIOUS SPIKE ON MONITOR 2RT-PR09C CAUSED BY VOLTAGE SPIKING ON THE AC POWER FEED TO THE MONITOR.

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for month of February 1988

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13. ABSTRACT (200 words or less)

This monthly report contains Licensee Event Report (LER) operational information that was processed into the LER data file of the Nuclear Safety Information Center (NSIC) during the one month period identified on the cover of the document. The LERs, from which this information is derived, are submitted to the Nuclear Regulatory Commission (NRC) by nuclear power plant licensees in accordance with federal regulations. Procedures for LER reporting for revisions to those events occurring prior to 1984 are described in NRC Regulatory Guide 1.16 and NUREG-1061, Instructions for Preparation of Data Entry Sheets for Licensee Event Reports. For those events occurring on and after January 1, 1984, LERs are being submitted in accordance with the revised rule contained in Title 10 Part 50.73 of the Code of Federal Regulations (10 CFR 50.73 - Licensee Event Report System) which was published in the Federal Register (Vol. 48, No. 144) on July 26, 1983. NUREG-1022, Licensee Event Report System - Description of Systems and Guidelines for Reporting, provides supporting guidance and information on the revised LER rule. The LER summaries in this report are arranged alphabetically by facility name and then chronologically by event date for each facility. Component, system, keyword, and component vendor indexes follow the summaries. Vendors are those identified by the utility when the LER form is initiated; the keywords for the component, system, and general keyword indexes are assigned by the computer using correlation tables from the Sequence Coding and Search System.

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