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Licensee Event Report (LER) Compilation

For month of December 1987

Oak Ridge National Laboratory

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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 87-003
EMERGENCY FEEDWATER SYSTEM ACTUATION DURING POWER REDUCTION DUE TO MAIN FEEDWATER PUMPS CONTROL SYSTEM PROBLEMS.
EVENT DATE: 080887 REPORT DATE: 090887 NSSS: BW TYPE: PWR
VENDOR: LOVE CONTROLS CORP.
WESTINGHOUSE ELECTRIC CORP.

(NSIC 206314) ON 8/8/87, DURING A POWER REDUCTION WITH THE UNIT AT APPROXIMATELY 20 PERCENT POWER, THE 'B' MAIN FEEDWATER (MFW) MAIN BLOCK VALVE CLOSED AS EXPECTED SHIFTING 'B' MFW PUMP FROM FEEDWATER SPEED CONTROL TO DIFFERENTIAL PRESSURE CONTROL. AT THIS TIME AN INTEGRATED CONTROL SYSTEM (ICS) HIGH FEEDWATER DEMAND SIGNAL CAUSED 'B' MFW PUMP TURBINE CONTROLS TO SHIFT TO A 'TRACK AND HOLD' MODE. SATISFACTORY MANUAL CONTROL OF 'B' MFW PUMP COULD NOT BE REGAINED SO 'B' MFW PUMP WAS REMOVED FROM SERVICE. WHEN 'B' MFW PUMP WAS REMOVED FROM SERVICE, 'A' MFW PUMP FAILED TO ADEQUATELY SUPPLY REQUIRED FEEDWATER FLOW. THIS RESULTED IN AN EMERGENCY FEEDWATER INITIATION AND CONTROL (EFIC) SYSTEM ACTUATION ON LOW LEVEL IN 'B' STEAM GENERATOR. POWER WAS PROMPTLY REDUCED AND THE MAIN TURBINE WAS TRIPPED BY CONTROL ROOM OPERATIONS PERSONNEL. THE UNIT WAS STABLIZED AT APPROXIMATELY 3 PERCENT POWER. A REACTOR TRIP DID NOT OCCUR. THE ROOT CAUSE OF THE ICS HIGH FEEDWATER DEMAND SIGNAL COULD NOT BE DETERMINED. TROUBLESHOOTING OF THE ICS REVEALED NO PROBLEMS. THE CAUSE OF THE INADEQUATE MFW SUPPLY FROM 'A' MFW PUMP WAS THE RESULT OF AN MFW PUMP TURBINE GOVERNOR VALVE FULCRUM PIN THAT HAD COME OUT OF POSITION RESTRICTING VALVE MOVEMENT. THIS LIMITED MAIN STEAM SUPPLY TO THE 'A' MFW PUMP TURBINE WHICH RESULTED IN REDUCED PUMP CAPACITY.

[2] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 87-004
REACTOR TRIP AND EMERGENCY FEEDWATER ACTUATION DURING POWER ASCENSION DUE TO MAIN FEEDWATER PUMPS AND CONTROL SYSTEM PROBLEMS.
EVENT DATE: 081587 REPORT DATE: 091487 NSSS: BW TYPE: PWR
VENDOR: LOVE CONTROLS CORP.

(NSIC 206316) ON 08/15/87 DURING POWER ASCENSION OPERATIONS AT APPROX. 12%, 'B' MAIN FEEDWATER (MFW) PUMP CONTROL WAS SHIFTED FROM MANUAL TO AUTOMATIC. FEEDWATER OSCILLATIONS ENSUED RESULTING IN THE MFW PUMP TURBINE CONTROLS SHIFTING TO A 'TRACK AND HOLD' MODE. 'B' MFW PUMP TURBINE SPEED UNEXPECTELY DECREASED TO A SPEED LOWER THAN THAT REQUIRED TO SUPPLY FEEDWATER TO THE STEAM GENERATORS. EMERGENCY FEEDWATER (EFW) SUBSEQUENTLY ACTUATED ON LOW STEAM GENERATOR WATER LEVEL. A REACTOR TRIP OCCURRED DUE TO HIGH REACTOR COOLANT SYSTEM PRESSURE. AFTER THE TRIP, CONTROL OF THE 'B' MFW PUMP WAS REGAINED AND EFW WAS SECURED. POST TRIP PLANT RESPONSE WAS SATISFACTORY. THE PLANT STARTUP PROCEDURE WAS MODIFIED TO PROVIDE GUIDANCE TO PLANT OPERATORS TO KEEP MFW PUMPS IN MANUAL CONTROL UNTIL AT LEAST 20% POWER BEFORE SWITCHING TO AUTOMATIC CONTROL. THIS PROVIDES FOR MORE STABLE OPERATING CONDITIONS FOR TRANSFER FROM MANUAL TO AUTOMATIC. TROUBLESHOOTING REVEALED THAT THE INTEGRATED CONTROL SYSTEM (ICS) WAS SUPPLYING A NEGATIVE VOLTAGE TO THE MFW PUMP TURBINE LOVEJOY CONTROL SYSTEM. THIS IS BELIEVED TO HAVE CAUSE ABNORMAL PUMP OPERATION. A DAMAGED CHIP IN THE 'TRACK AND HOLD' CIRCUIT WAS REPLACED AND MODIFICATIONS TO THE ICS WERE MADE TO PREVENT THE INPUT OF A NEGATIVE VOLTAGE TO THE MFW PUMP TURBINE CONTROL SYSTEM.

[3] ARNOLD DOCKET 50-331 LER 87-026
LIMITING CONDITION FOR OPERATION NOT IMMEDIATELY RECOGNIZED DURING ISOLATION VALVE REPAIR.
EVENT DATE: 090387 REPORT DATE: 092887 NSSS: GE TYPE: BWR

(NSIC 206581) ON SEPTEMBER 3, 1987, AT 0751 HOURS, THE PLANT WAS OPERATING AT 99% OF RATED THERMAL POWER WHEN THE REACTOR WATER CLEANUP (RWCU) SYSTEM WAS TAKEN OUT OF SERVICE TO ALLOW MAINTENANCE ON THE RWCU OUTLET ISOLATION VALVE (MO2740). DUE TO A MISINTERPRETATION OF THE REQUIREMENTS FOR MAINTENANCE, THE INVOLVED OPERATOR DE-ENERGIZED THE ISOLATION VALVE IN THE OPEN POSITION. THIS CONDITION IS ALLOWED

BY DAEC TECH SPECS (3.7.D.3) PROVIDED A 24 HOUR LIMITING CONDITION FOR OPERATION (LCO) IS ENTERED. AT THE TIME THE VALVE WAS DE-ENERGIZED OPEN THE LCO WAS INADVERTENTLY NOT ENTERED. AT 1313 HOURS THIS SITUATION WAS IDENTIFIED AS A RESULT OF A PANEL CHECK FOLLOWING AN EXPECTED RWCU ISOLATION. RATHER THAN ENTERING AN LCO AT THIS TIME, THE VALVE WAS MANUALLY CLOSED ALLOWING THE REQUIREMENTS OF TECH SPEC 3.7.D.2 TO BE MET. THIS EVENT HAD NO AFFECT ON THE SAFE OPERATION OF THE PLANT. THE ROOT CAUSE OF THIS EVENT IS PERSONNEL ERROR ON THE PART OF UTILITY LICENSED OPERATORS WHO DE-ENERGIZED THE VALVE IN THE OPEN POSITION WITHOUT ENTERING AN LCO. AS CORRECTIVE ACTION, OPERATIONS PERSONNEL WERE COUNSELED, VIA DAEC OPERATIONS DEPARTMENT INSTRUCTION PROCEDURE 00, SECTION 6.2, USING THIS EVENT AS A SPECIFIC EXAMPLE OF THE IMPORTANCE OF COMPLYING WITH TECH SPECS WHEN PERFORMING MAINTENANCE. THIS EVENT IS BEING REPORTED FOR INFORMATION ONLY.

[4] BEAVER VALLEY 2 DOCKET 50-412 LER 87-011 REV 01
UPDATE ON INADVERTENT SAFETY INJECTION DUE TO PERSONNEL ERROR.
EVENT DATE: 073087 REPORT DATE: 092887 NSSS: WE TYPE: PWR

(NSIC 206527) ON 7/30/87, WITH THE PLANT IN HOT STANDBY, TWO SURVEILLANCE PROCEDURES WERE IN PROGRESS WHICH REQUIRED THE CHANNEL IV STEAM PRESSURE BISTABLES FOR THE 21C STEAM GENERATOR TO BE IN A TRIPPED CONDITION. AN ADDITIONAL SURVEILLANCE PROCEDURE WHICH REQUIRED THE TRIPPING OF CHANNEL III STEAM PRESSURE BISTABLES FOR THE 21 C STEAM GENERATOR, WAS ALSO INITIATED. WHEN THE CHANNEL III BISTABLES WERE TRIPPED, AT 1433 HOURS, A SAFETY INJECTION OCCURRED DUE TO A LOW STEAMLINE PRESSURE IN THE 21C STEAMLINE. THE OPERATORS FOLLOWED EMERGENCY OPERATING PROCEDURES E-0 AND ES-1.1 TO STABILIZE THE PLANT. THE CAUSE FOR THIS EVENT WAS ATTRIBUTED TO PERSONNEL ERROR DURING THE PERFORMANCE OF THE SURVEILLANCE PROCEDURE WHICH TRIPPED THE CHANNEL III STEAM PRESSURE BISTABLES. THE TEST ENGINEER DID NOT MAINTAIN THE REQUIRED COGNIZANCE OF THE INTENT OF THE SURVEILLANCE PROCEDURE AND SUBSEQUENTLY DID NOT SATISFY THE INITIAL CONDITIONS WITHIN THE PROCEDURE. INADEQUATE COMMUNICATIONS BETWEEN ALL INVOLVED GROUPS AND AN INADEQUATE REVIEW OF THE TEST PROCEDURE BY THE OPERATIONS SHIFT ALSO CONTRIBUTED TO THIS EVENT. THE TEST ENGINEER HAS BEEN REASSIGNED AND THIS INCIDENT WILL BE REVIEWED AT DEPARTMENTAL SAFETY MEETINGS WITHIN THE OPERATIONS, TESTING AND MAINTENANCE GROUPS. THERE WERE NO SAFETY IMPLICATIONS TO THE PUBLIC. THE ENGINEERED SAFETY FEATURES SYSTEM FUNCTIONED AS DESIGNED.

[5] BEAVER VALLEY 2 DOCKET 50-412 LER 87-013
INADVERTENT REALIGNMENT OF MAIN FILTER BANK DAMPERS DUE TO PERSONNEL ERROR.
EVENT DATE: 080687 REPORT DATE: 090287 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: BEAVER VALLEY 1 (PWR)

(NSIC 206217) 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-303, 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.

[6] BEAVER VALLEY 2 DOCKET 50-412 LER 87-012
MANUAL REACTOR TRIP DUE TO DROPPED RODS.
EVENT DATE: 080787 REPORT DATE: 090287 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 206272) ON 8/7/87, DURING LOW POWER PHYSICS TESTING, FOUR (4) CONTROL RODS FELL INTO THE CORE. THE OPERATORS, AS PER THE PROCEDURE FOR A DROPPED ROD, MANUALLY TRIPPED THE REACTOR AND STABILIZED THE PLANT. AN INVESTIGATION BASED ON UNIT 1 EXPERIENCE AND SOME TROUBLESHOOTING IDENTIFIED A FAULT IN THE MOVABLE FIRING CIRCUIT CARD IN THE 2BD POWER CABINET AS THE CAUSE OF THE FAILURE. THIS CARD WAS REPLACED. RODS WERE WITHDRAWN TO CONTINUE LOW POWER PHYSICS TESTING. ON 8/10/87 THE SAME FOUR (4) RODS AGAIN FELL INTO THE CORE. AGAIN, THE OPERATOR MANUALLY TRIPPED THE REACTOR. FURTHER INVESTIGATION DISCOVERED A FAILED THYRISTOR IN THE "B" MOVABLE GRIPPER CIRCUIT CARD (WESTINGHOUSE MODEL #6050D12G01) IN THE 2BD POWER CABINET. THIS FAILED THYRISTOR WAS IDENTIFIED TO BE THE CAUSE OF BOTH EVENTS. THERE WERE NO SAFETY IMPLICATIONS DUE TO THESE EVENTS. TRIPPING THE REACTOR WAS A CONSERVATIVE ACTION FOR THESE EVENTS, AS THE PLANT'S SAFETY ANALYSIS ALLOWS SUCH ROD MISALIGNMENT AT LOW POWER LEVELS.

[7] BEAVER VALLEY 2 DOCKET 50-412 LER 87-014
REACTOR TRIP ON LOW-LOW STEAM GENERATOR LEVEL DUE TO PERSONNEL ERROR.
EVENT DATE: 081587 REPORT DATE: 090987 NSSS: WE TYPE: PWR

(NSIC 206082) ON 8/15/87, WITH THE UNIT IN POWER OPERATION, REACTOR POWER WAS BEING INCREASED TO 8.0 E-5 AMPS TO SUPPORT NUCLEAR INSTRUMENTATION CALIBRATION. THE STEAM DUMP SYSTEM WAS BEING USED TO COMPENSATE FOR REACTOR POWER CHANGES. DURING THE POWER INCREASE, THE OPERATORS NOTICED THAT THE STEAM DUMP SYSTEM DID NOT APPEAR TO RESPOND PROPERLY WHILE IN AUTOMATIC. THE OPERATOR THEN PLACED THE STEAM DUMP SYSTEM IN MANUAL, INPUT A 3% DEMAND AND PLACED THE SYSTEM BACK TO AUTOMATIC. THE DEMAND INCREASED TO APPROXIMATELY 16% DUE TO THE UPWARD ROD MOTION, TERMINATING THE PRIMARY SYSTEM TEMPERATURE RISE. THE STEAM GENERATOR LEVELS ALL SWELLED IN RESPONSE TO THE INCREASED STEAM DEMAND, THEN RAPIDLY DROPPED. FEEDWATER WAS BEING CONTROLLED IN MANUAL AND COULD NOT RESPOND TO THE INCREASED STEAM DEMAND. THE OPERATOR ATTEMPTED TO INCREASE FEEDWATER FLOW, BUT THE LEVELS CONTINUED TO DROP. A REACTOR TRIP ON LOW-LOW LEVEL IN THE 21A STEAM GENERATOR OCCURRED AT 0237 HOURS. THE OPERATORS USED THE EMERGENCY PROCEDURES TO STABILIZE THE PLANT. THE CAUSE OF THIS EVENT WAS ATTRIBUTED TO THE INEXPERIENCE OF THE OPERATORS IN ROD MOTION WITH RODS OF HIGH DIFFERENTIAL WORTH. ADMINISTRATIVE GUIDANCE HAS BEEN PROVIDED TO THE OPERATORS TO PREVENT RECURRENCE. THERE WERE NO SAFETY IMPLICATIONS. THE REACTOR PROTECTION SYSTEM FUNCTIONED AS DESIGNED TO PLACE THE PLANT IN A STABLE CONDITION.

[8] BEAVER VALLEY 2 DOCKET 50-412 LER 87-015
REACTOR TRIP DUE TO LOW-LOW STEAM GENERATOR LEVEL DUE TO OPERATOR ERROR.
EVENT DATE: 081587 REPORT DATE: 090487 NSSS: WE TYPE: PWR

(NSIC 206083) ON 8/15/87, REACTOR POWER LEVEL WAS BEING INCREASED TO SUPPORT NUCLEAR INSTRUMENTATION CALIBRATION. AS POWER APPROACHED 20%, THE LEVELS OF ALL THREE (3) STEAM GENERATORS BEGAN TO DECREASE RAPIDLY. OPERATORS INCREASED FEED FLOW AND ISOLATED STEAM GENERATOR BLOWDOWN, BUT WERE UNABLE TO PREVENT LEVEL FROM DROPPING TO THE LO-LO STEAM GENERATOR LEVEL SETPOINT. THIS INITIATED AN AUTOMATIC REACTOR TRIP, AS PER DESIGN. THE OPERATORS STABILIZED THE PLANT USING THE REACTOR TRIP RESPONSE PROCEDURE. POST-TRIP EVALUATION DETERMINED THE EVENT HAD OCCURRED DUE TO OPERATOR ERROR, RESULTING FROM INITIAL CORE PARAMETER RESPONSES. DURING THE POWER INCREASE, OPERATORS HAD BEEN WITHDRAWING RODS AS REQUIRED. HOWEVER, DUE TO HIGH ROD WORTHS AND A NEAR ZERO MODERATOR TEMPERATURE COEFFICIENT, THIS RESULTED IN A GREATER THAN EXPECTED RCS HEATUP, WHICH CAUSED THE STEAM GENERATOR LEVELS TO SWELL. CORRECTIVE ACTION THROUGH ROD INSERTION AND FEEDWATER ADDITION RESULTED IN A COOLDOWN AND STEAM GENERATOR SHRINK. THE SHRINK

RESULTED IN A LO-LO STEAM GENERATOR REACTOR TRIP. OPERATIONS HAS BEEN PROVIDED WITH ADMINISTRATIVE GUIDANCE CONCERNING PROPER OPERATING TECHNIQUES UNDER THESE CONDITIONS. THESE CONDITIONS ARE THE RESULT OF THE INITIAL CORE, AND WILL IMPROVE WITH CORE AGE. THERE WERE NO SAFETY IMPLICATIONS DUE TO THIS EVENT AS ALL ESF SYSTEMS/COMPONENTS FUNCTIONED AS DESIGNED.

[9] BEAVER VALLEY 2 DOCKET 50-412 LER 87-016
INADVERTENT START OF AUXILIARY FEEDWATER PUMP CAUSED BY AN ERRONEOUSLY LOW DISCHARGE PRESSURE SIGNAL.
EVENT DATE: 081687 REPORT DATE: 091587 NSSS: WE TYPE: PWR

(NSIC 206084) ON 8/16/87, WITH THE UNIT IN THE STARTUP MODE AT 1% REACTOR POWER, TURBINE DRIVEN AUXILIARY FEEDWATER PUMP 2FWE-P22 WAS STARTED FOR A SURVEILLANCE TEST. FOLLOWING A PRESET TIME DELAY, MOTOR DRIVEN AUXILIARY FEEDWATER PUMP 2FWE-P23B AUTOMATICALLY STARTED. THE CAUSE OF THE INADVERTENT START WAS DETERMINED TO BE THE ISOLATION OF PRESSURE SWITCH 2FWE-PS158B, WHICH PROVIDED AN ERRONEOUSLY LOW 2FWE-P22 DISCHARGE PRESSURE SIGNAL TO THE STARTING CIRCUITRY OF 2FWE-P23B. THE PUMP RAN FOR TEN SECONDS BEFORE BEING SHUT DOWN. NO SAFETY IMPLICATIONS RESULTED AS THE PUMP STARTED PROPERLY IN RESPONSE TO A VALID SIGNAL. THE ISOLATION VALVE FOR 2FWE-PS158B WAS REOPENED AND THE TEST RE-PERFORMED WITHOUT FURTHER INCIDENT. AVAILABLE EVIDENCE INDICATES THAT THE SWITCH WAS PROPERLY ALIGNED ON 7/4/87; HOWEVER, IT IS JUDGED THAT IT WAS INADVERTENTLY ISOLATED BY PERSONNEL IN THE AREA DURING THE VARIOUS PHASES OF TRANSITION TESTING. A SIMILAR AUTOSTART ON 7/18/87 WAS NOT REPORTED AT THAT TIME BECAUSE IT WAS BELIEVED TO BE A CONSEQUENCE OF TESTING. THE AUTOMATIC STARTS OF 2FWE-P23B WERE REPORTED AT 1258 HOURS AND 1348 HOURS ON 8/16/87 IN ACCORDANCE WITH 10 CFR 50.73.B.2.II. TO PREVENT FURTHER PROBLEMS, ALIGNMENTS FOR SAFETY RELATED SYSTEMS WERE CHECKED AND VERIFIED TO BE SATISFACTORY ON 8/18/87.

[10] BEAVER VALLEY 2 DOCKET 50-412 LER 87-017
INADVERTENT FEEDWATER ISOLATION DUE TO A PROCEDURAL DEFICIENCY.
EVENT DATE: 081687 REPORT DATE: 091487 NSSS: WE TYPE: PWR

(NSIC 206309) ON 8/16/87, WITH THE UNIT IN THE STARTUP MODE, A SURVEILLANCE TEST ON THE STEAM DUMP SYSTEM WAS IN PROGRESS. THIS TEST REQUIRED PLACING THE STEAM DUMP SYSTEM IN THE STEAM PRESSURE MODE, THE REMOVAL OF THE TURBINE FIRST STAGE PRESSURE - TREF SIGNAL (CARD C8-552), AND THE INSERTION OF AN ANALOG SIGNAL OF 1.3 VOLTS FOR THE TREF SIGNAL. THE STEAM DUMP SYSTEM WAS THEN PLACED IN THE TAVE MODE. THIS INPUT A 4 DEGREE ERROR SIGNAL REQUIRED FOR THE TEST. AFTER COMPLETING THE TEST AND IN THE PROCESS OF RESTORING THE TREF SIGNAL, THE ANALOG SIGNAL WAS REMOVED PRIOR TO PLACING THE STEAM DUMP SYSTEM IN THE STEAM PRESSURE MODE. THIS RESULTED IN A LARGE ENOUGH TEMPERATURE ERROR TO OPEN ALL THE STEAM DUMPS. THE STEAM GENERATORS "SWELLED" TO THE HIGH LEVEL SETPOINT CAUSING A TURBINE TRIP AND A FEEDWATER ISOLATION SIGNAL AT 1340 HOURS. THE STEAM DUMP SYSTEM WAS TAKEN TO OFF AND CLOSED. THE FEEDWATER ISOLATION SIGNAL WAS RESET, THE BYPASS AND CONTAINMENT ISOLATION FEEDWATER VALVES WERE OPENED AND THE 21A MAIN FEEDWATER PUMP WAS RESTARTED. THE CAUSE FOR THIS INCIDENT WAS A DEFICIENT PROCEDURE. THIS PROCEDURE HAS BEEN REVISED; HOWEVER, THIS TEST IS NOT EXPECTED TO BE PERFORMED AGAIN. THERE WERE NO SAFETY IMPLICATIONS TO THE PUBLIC AS A RESULT OF THIS EVENT.

[11] BEAVER VALLEY 2 DOCKET 50-412 LER 87-018 REV 01
UPDATE ON REACTOR TRIP DUE TO DE-ENERGIZED ROD CONTROL POWER CABINET.
EVENT DATE: 081887 REPORT DATE: 092887 NSSS: WE TYPE: PWR
VENDOR: LAMBDA ELECTRONICS

(NSIC 206569) ON 8/18/87, MAINTENANCE INSTRUMENT TECHNICIANS WERE REPLACING THE #1 POWER SUPPLY OVERVOLTAGE LIMITER IN THE ROD CONTROL SYSTEM 1AC POWER CABINET. (THE POWER CABINETS USE TWO POWER SUPPLIES IN AN AUCTIONEERED ARRANGEMENT,

ALLOWING EITHER, BUT NOT BOTH, OF THE SUPPLIES TO BE REMOVED FROM SERVICE FOR MAINTENANCE WITHOUT AFFECTING THE OPERATION OF THE POWER CABINET.) WHILE MOUNTING THE OVERVOLTAGE PROTECTOR MODULE IN THE CABINET, THE MODULE WAS INADVERTENTLY ALLOWED TO COME IN CONTACT WITH THE ADJACENT TERMINAL BOARD. THIS SHORTED OUT THE #2 POWER SUPPLY FOR THE 1AC POWER CABINET. WITH THE #1 POWER SUPPLY ALREADY REMOVED FOR MAINTENANCE, THIS RESULTED IN DE-ENERGIZING THE 1AC POWER CABINET, CAUSING THE GROUP 1 "A" SHUTDOWN BANK AND THE GROUP 1 "A" AND "C" CONTROL BANKS OF RODS TO FALL INTO THE CORE. THIS, IN TURN CAUSED A NEGATIVE RATE REACTOR TRIP. OPERATORS STABILIZED THE PLANT USING THE REACTOR TRIP RESPONSE PROCEDURE. BOTH POWER SUPPLIES WERE REPAIRED AND RETURNED TO SERVICE. ALL MAINTENANCE INSTRUMENT TECHNICIANS ARE RECEIVING TRAINING ON THIS EVENT AND THE NECESSITY FOR MORE IN-DEPTH JOB PREPLANNING AND ATTENTION TO DETAIL. THERE WERE NO SAFETY IMPLICATIONS DUE TO THIS EVENT AS ALL ESF/RPS EQUIPMENT FUNCTIONED AS PER DESIGN.

[12] BEAVER VALLEY 2 DOCKET 50-412 LER 87-019
REACTOR TRIP/TURBINE TRIP DUE TO A SPURIOUS TURBINE OVERSPEED TRIP SIGNAL.
EVENT DATE: 082587 REPORT DATE: 092387 NSSS: WE TYPE: PWR
VENDOR: IPAC GROUP, INC.
WESTINGHOUSE ELEC CORP.-NUCLEAR ENERGY SYS

(NSIC 206514) ON 8/25/87, WITH THE UNIT IN POWER OPERATION AT 30% REACTOR POWER, TROUBLESHOOTING OF THE TURBINE ELECTRONIC OVERSPEED TRANSDUCER WAS IN PROGRESS. FEEDWATER FLOW TRANSMITTER 2FWS*FT496 WAS OUT-OF-SERVICE (ASSOCIATED BISTABLES WERE IN THE TRIPPED CONDITION). AT 1357 HOURS, A SPURIOUS TURBINE OVERSPEED SIGNAL CAUSED A TURBINE TRIP. A REACTOR TRIP DID NOT IMMEDIATELY OCCUR BECAUSE REACTOR POWER WAS BELOW THE TURBINE TRIP REACTOR TRIP (P-9) SETPOINT. THE STEAM GENERATOR (SG) LEVELS ALL SHRANK BELOW THE LOW LEVEL TRIP SETPOINT. FOLLOWING THE TURBINE TRIP, THE REACTOR TRIPPED ON 21C SG LOW LEVEL COINCIDENT WITH A STEAM FLOW/FEED FLOW MISMATCH SINCE THE 2FWS*FT496 BISTABLES WERE ALREADY TRIPPED. THE OPERATORS INITIATED EMERGENCY OPERATING PROCEDURE E-0 TO STABLIIZE THE PLANT. DURING THE 4KV AUTO BUS TRANSFER, THE REACTOR COOLANT PUMPS (RCP) TRIPPED ON A 2/3 RCP BUS UNDERFREQUENCY SIGNAL. THE PLANT WAS STABILIZED ON NATURAL CIRCULATION. THE RCPS WERE RESTARTED AT 1447 HOURS. THE CAUSE FOR THIS EVENT WAS A SPURIOUS TURBINE OVERSPEED SIGNAL. THE RCP TRIPS WERE DUE TO MALFUNCTIONING RELAYS. THE TURBINE OVERSPEED TRANSDUCER WAS REPLACED WITH AN INSTALLED SPARE AND THE UNDERFREQUENCY RELAYS WERE REPLACED WITH A DIFFERENT STYLE RELAY. THERE WERE NO SAFETY IMPLICATIONS TO THE PUBLIC.

[13] BIG ROCK POINT DOCKET 50-155 LER 87-010
REACTOR TRIP DUE TO SPURIOUS SHORT PERIOD.
EVENT DATE: 091087 REPORT DATE: 092987 NSSS: GE TYPE: BWR

(NSIC 206579) ON SEPTEMBER 10, 1987, WHILE TESTING CONTROL ROD DRIVES (AA) DURING A MAINTENANCE OUTAGE, A REACTOR PROTECTION SYSTEM (JC) ACTUATION OCCURRED. POWER LEVEL AT THE TIME OF THE TRIP WAS LESS THAN .0001 PERCENT AND THE ONE CONTROL ROD WITHDRAWN DURING TESTING SUCCESSFULLY INSERTED. ELECTRICAL NOISE INHERENT AT LOW POWER LEVELS CAUSED THE TRIP.

[14] BRAIDWOOD 1 DOCKET 50-456 LER 87-041
TECHNICAL SPECIFICATION SURVEILLANCE NOT PERFORMED WHEN UNIT 1 COMPUTER WAS INOPERABLE.
EVENT DATE: 080887 REPORT DATE: 090387 NSSS: WE TYPE: PWR

(NSIC 206304) ON AUGUST 8, 1987, A LICENSED CONTROL ROOM OPERATOR RECOGNIZED THAT A SURVEILLANCE FOR TECH SPEC 4.2.1.1.B MAY NOT HAVE BEEN PERFORMED WHEN REQUIRED DURING THE TIMES THE UNIT HAD BEEN AT OR ABOVE 15% REACTOR POWER. AT VARIOUS TIMES THE UNIT 1 COMPUTER WAS INOPERABLE, CAUSING THE AXIAL FLUX DIFFERENCE (AFD)

MONITOR ALARM TO BE INOPERABLE. THIS REQUIRED AN OPERATING SURVEILLANCE TO BE PERFORMED HOURLY FOR THE NEXT 24 HOURS AFTER THE ALARM HAD BEEN RETURNED TO OPERABLE STATUS. IN REVIEW OF THE PLANT HISTORY, THERE HAVE BEEN 5 OCCURRENCES WHERE THE SURVEILLANCE SHOULD HAVE BEEN PERFORMED FOR THIS REASON, BUT WAS NOT. THE ROOT CAUSE OF THE EVENT WAS A PROGRAMMATIC DEFICIENCY IN THAT LICENSED PERSONNEL WERE NOT TRAINED TO RECOGNIZE THAT THE SURVEILLANCE WAS REQUIRED. THE DAILY ORDERS ON AUGUST 10, 1987 ALERTED OTHER SHIFTS TO THIS PROBLEM. AN OPERATOR AID WILL BE DEVELOPED TO INFORM CONTROL ROOM PERSONNEL WHICH TECH SPECS ARE ACTIVE WHEN THE UNIT 1 COMPUTER IS CONSIDERED INOPERABLE. THERE HAVE BEEN NO PREVIOUS OCCURRENCES.

[15] BRAIDWOOD 1 DOCKET 50-456 LER 87-044
CLOSURE OF HEAT REMOVAL DISCHARGE ISOLATION VALVE DUE TO INADEQUATE OPERATING PROCEDURE.
EVENT DATE: 080887 REPORT DATE: 090487 NSSS: WE TYPE: PWR

(NSIC 206306) ON 08/07/87, LCOAR 5.2-1A WAS ENTERED TO CONDUCT REPAIRS ON 1B RESIDUAL HEAT REMOVAL (RH) VALVE 1RH8733B. REPAIRS COMPLETED ON 08/08/87. A POST-MAINTENANCE TEST WAS ALSO PERFORMED ON 08/08/87, PER PROCEDURE BWOP RH-5. AS PART OF THE VALVE LINE UP FOR BWOP RH-5, RHR DISCHARGE CROSSTIE VALVES MUST BE OPEN IN MODES 1, 2 AND 3. HOWEVER, AS PART OF THE SYSTEM LINEUP, AS DIRECTED BY THE STATION CONTROL ROOM ENGINEER (SCRE), THE CROSSTIE VALVE WAS CLOSED. REVIEW OF THIS BY THE INCOMING SHIFT, AND DISCUSSIONS HELD ON 08/09/87, CONSERVATIVELY DETERMINED THAT BOTH RW TRAINS COULD HAVE BEEN CONSIDERED INOPERABLE FOR THE 64 MINUTES REQUIRED FOR THE VAVLE TESTING. THE ROOT CAUSE WAS INADEQUATE TECH SPEC REVIEW BY THE SCREW. CORRECTIVE ACTIONS WERE: CONTROL SWITCHES FOR THE CROSSTIE VALVES HAVE BEEN TAGGED WITH CAUTION CARDS STATING THAT CLOSING THE VALVE(S) DURING MODES 1, 2, 3 OR 4 COULD RENDER BOTH TRAINS OF RHR INOPERABLE. THE INCIDENT HAS BEEN FORMALLY REVIEWED WITH THE INVOLVED PARTIES. THE SUBJECT OF PROCEDURE COMPLIANCE HAS BEEN REITERATED WITH ALL SHIFT MANAGEMENT PERSONNEL. BRAIDWOOD STATION IS CONTINUING TO REVIEW THE REPORTABILITY OF THIS EVENT. DISCUSSIONS WILL BE INITIATED WITH THE NRC. THIS EVENT IS BEING REPORTED AS A COURTESY NOTIFICATION PENDING COMPLETION OF THESE DISCUSSIONS AND FINAL DETERMINATION OF THIS EVENT'S REPORTABILITY.

[16] BRAIDWOOD 1 DOCKET 50-456 LER 87-042
CONTROL ROOM VENTILATION ENGINEERED SAFETY FEATURE ACTUATION SIGNAL DUE TO SPURIOUS ELECTRICAL NOISE FROM CONDENSATION IN DETECTOR CHAMBERS.
EVENT DATE: 080987 REPORT DATE: 090387 NSSS: WE TYPE: PWR

(NSIC 206305) ON AUGUST 9, 1987 AT 0040, MAIN CONTROL ROOM OUTSIDE AIR INTAKE TRAIN A RADIATION MONITOR (OPR31J WENT INTO HIGH ALARM AND INTERLOCK CAUSING A MAIN CONTROL ROOM VENTILATION ENGINEERED SAFETY FEATURES ACTUATION SIGNAL. AN INVESTIGATION INTO THE EVENT BY RADIATION CHEMISTRY PERSONNEL REVEALED A SMALL AMOUNT OF WATER IN THE SAMPLE LINE WITH NO RADIOACTIVITY PRESENT. THE SINGLE OCCURRENCE WAS DETERMINED TO BE SPURIOUS. THE SYSTEM WAS RESET TO NORMAL. THE ROOT CAUSE OF THE EVENT WAS WARM HUMID INTAKE AIR CONDENSING IN THE SAMPLE LINES. WATER ACCUMULATED AND WAS INTRODUCED INTO THE DETECTOR CHAMBER. THIS IS BELIEVED TO HAVE CAUSED AN ELECTRICAL PERTURBATION SEEN BY THE ELECTRICAL CIRCUITRY AS AN INCREASE IN RADIOACTIVITY. THE RADIATION CHEMISTRY TECHNICIAN REMOVED THE WATER FROM THE LINES AND CHANGED OUT THE FILTER. AN INSPECTION BY THE SYSTEM TEST ENGINEER WAS CONDUCTED AFTER THE FILTER WAS CHANGED. NO WATER WAS FOUND IN THE MONITOR OR IN THE INTAKE DUCT. THERE HAVE BEEN NO PREVIOUS OCCURRENCES OF THIS NATURE. PREVIOUS NOISE PROBLEMS HAVE BEEN DETERMINED TO BE CAUSED BY RADIO TRANSMISSIONS AND FAULTY EQUIPMENT.

[17]   BROWNS FERRY 1   DOCKET 50-259   LER 87-008 REV 01
UPDATE ON FAILURE OF POTENTIAL TRANSFORMER FUSE CONTACTS CAUSE ELECTRICAL FAULT AND ENGINEERING SAFETY FEATURE ACTUATION.
EVENT DATE: 042087   REPORT DATE: 090487   NSSS: GE   TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
                      BROWNS FERRY 3 (BWR)
VENDOR: ALLIS CHALMERS

(NSIC 206570) ON APRIL 20, 1987, DURING PERFORMANCE OF THE MONTHLY SURVEILLANCE TEST A PHASE TO PHASE SHORT OCCURRED BETWEEN CONTACTS IN THE DIESEL GENERATOR CONTROL CABINET FOR THE 3ED DIESEL GENERATOR (DG). THIS FAULT CAUSED A REFUELING ZONE ISOLATION, INITIATION OF STANDBY GAS TREATMENT AND CONTROL ROOM EMERGENCY VENTILATION, AND ON UNIT 3, A HALF SCRAM AND PRIMARY CONTAINMENT ISOLATIONS. THE CAUSE OF THE FAULT WAS A FAILURE OF THE POTENTIAL TRANSFORMER FUSE CONTACTS. ALL 4 KV POTENTIAL TRANSFORMER FUSE CONTACTS THROUGHOUT THE PLANT WILL BE INSPECTED AND MAINTENANCE PROCEDURES WILL BE REVISED TO INSPECT THE CONTACTS ON A REGULAR BASIS. AN ENGINEERING EVALUATION DETERMINED THE FUSE IN THE DG EXCITER POTENTIAL TRANSFORMER CIRCUITRY WAS UNNECESSARY, THEREFORE, THE FUSE AND SPRING FINGER CONTACTS WILL BE BYPASSED ON ALL EIGHT DGS.

[18]   BROWNS FERRY 1   DOCKET 50-259   LER 87-021
LACK OF DESIGN CONTROL RESULTS IN MAIN CONTROL ROOM PANEL INSTALLATION NOT PER FINAL SAFETY ANALYSIS REPORT.
EVENT DATE: 053187   REPORT DATE: 090487   NSSS: GE   TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
                      BROWNS FERRY 3 (BWR)

(NSIC 206041) AN INSPECTION CONDUCTED BY THE NRC ON MAY 1-31, 1987, FOUND CERTAIN MAIN CONTROL ROOM VERTICAL PANELS MOUNTED IN A MANNER INCONSISTENT WITH THE BROWNS FERRY FINAL SAFETY ANALYSIS REPORT FOR SEISMIC QUALIFICATION. THE PANELS WERE SUPPOSED TO BE MOUNTED USING BOLTS IN A MANNER SIMILAR TO THEIR EARLY SHAKER TABLE QUALIFICATION DEMONSTRATIONS. INSTEAD, THE PANEL SKIRTS ARE WELDED TO THE MOUNTING BEAMS AT IRREGULAR INTERVALS. DESIGN DRAWINGS ISSUED IN MAY 1968 FAILED TO INCORPORATE DESIGN ASSUMPTIONS MADE DURING PANEL SEISMIC QUALIFICATION AND THEREFORE LACKED SUFFICIENT DETAIL FOR THE ANCHORAGE. TVA WILL CONDUCT A SEISMIC REVIEW OF THE CONTROL ROOM VERTICAL PANELS AND WILL MODIFY THE ANCHORAGES AS NECESSARY. TIGHTER CONTROLS ON DESIGN AND INSPECTION SERVE TO PREVENT RECURRENCE. THIS REPORT IS BEING SUBMITTED FOR INFORMATION ONLY.

[19]   BROWNS FERRY 1   DOCKET 50-259   LER 87-019
INCOMPLETE SURVEILLANCE RESULTS IN A CONDITION PROHIBITED BY TECHNICAL SPECIFICATIONS.
EVENT DATE: 080687   REPORT DATE: 090187   NSSS: GE   TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
                      BROWNS FERRY 3 (BWR)
VENDOR: AMERICAN FOUNDRY & FURNACE

(NSIC 206289) ON AUGUST 6, 1987, A FIRE PROTECTION FLOW TEST ON THE RAW WATER YARD LOOP COULD NOT BE COMPLETED DUE TO VIBRATION PROBLEMS WITH ONE OF THE TEST HYDRANTS. TECH SPECS CITE THE PARTICULAR HYDRANTS TO BE CAPABLE OF DELIVERING SPECIFIC FLOW AND PRESSURE REQUIREMENTS. ALL REQUIRED TESTING COULD NOT BE COMPLETED BY THE END OF THE SURVEILLANCE INTERVAL. SINCE A SUCCESSFUL FLOW TEST HAD BEEN PERFORMED ON THE HEADER AT ANOTHER NEARBY HYDRANT THE YARD LOOP WAS CONSIDERED CAPABLE OF DELIVERING THE REQUIRED FLOW. THE DEFECTIVE HYDRANT WAS REPLACED, AND THE SURVEILLANCE WAS COMPLETED ON AUGUST 20, 1987. OTHER SYSTEM MAINTENANCE WHICH HAD BEEN IN PROGRESS WAS RESPONSIBLE FOR SCHEDULING OF THE SURVEILLANCE CLOSE TO THE EXPIRATION DATE.

[20] BROWNS FERRY 1 DOCKET 50-259 LER 87-022
ENGINEERED SAFETY FEATURE ACTUATION DUE TO PERSONNEL ERROR DURING SWITCH CALIBRATION.
EVENT DATE: 081187 REPORT DATE: 091187 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 206425) ON AUGUST 11, 1987, AT 0120, WITH ALL THREE UNITS DEFUELED, TWO EMERGENCY EQUIPMENT COOLING WATER (EECW) PUMPS WERE INADVERTENTLY STARTED DUE TO A PERSONNEL ERROR DURING CALIBRATION OF A RAW COOLING WATER PRESSURE SWITCH. A LIFTED WIRE WAS ALLOWED TO CONTACT THE TERMINAL BLOCK. THIS COMPLETED THE START LOGIC FOR THE EECW PUMPS. THIS WAS AN UNPLANNED ACTUATION OF AN ENGINEERED SAFETY FEATURE. THE INSTRUMENT MECHANICS IMMEDIATELY MOVED THE WIRE. THE OPERATOR VERIFIED THE ACTUATION SIGNAL AND SECURED THE PUMPS, RETURNING THEM TO STANDBY READINESS FIVE MINUTES AFTER INITIATION. THE INSTRUMENT MECHANICS RESET THE SWITCH AND PROPERLY TERMINATED THE WIRE. THE INSTRUMENT MECHANICS INVOLVED HAVE BEEN COUNSELED ON THE NEED FOR INCREASED CAUTION WHEN WORKING WITH ENERGIZED EQUIPMENT. A CHANGE HAS BEEN INITIATED TO THE CALIBRATION PROCEDURE TO IMPROVE THE METHOD OF ISOLATING THE SWITCH DURING CALIBRATION.

[21] BROWNS FERRY 1 DOCKET 50-259 LER 87-023
PERSONNEL ERROR RESULTS IN UNREPRESENTATIVE RADIOLOGICAL RELEASE ASSESSMENT DATA.
EVENT DATE: 082687 REPORT DATE: 092287 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 206485) ON AUGUST 26, 1987, FOLLOWING THE COMPLETION OF A MONTHLY SURVEILLANCE ON RADIATION MONITOR FILTER ACTIVITY, THE SAMPLES, NORMALLY RETAINED FOR A QUARTERLY STRONTIUM CALCULATION, WERE MISTAKENLY DISCARDSD. THE SAMPLES, TAKEN FROM REACTOR AND TURBINE BUILDING VENTILATION EXHAUST AND OFF GAS STACK MONITORS WERE LEFT UNATTENDED IN UNLABELED FLASKS BY A RADIOCHEMICAL LABORATORY TRAINEE. BROWNS FERRY TECHNICAL SPECIFICATION (TS) 4.8.B.3 REQUIRES THAT CUMULATIVE QUARTARLY AND YEARLY DOSE CONTRIBUTIONS FROM ALL GASEOUS RELEASES BE DETERMINED AT LEAST ONCE EVERY 31 DAYS. THE COMPOSITE PORTION FOR THE MONTH OF AUGUST WAS INADVERTENTLY DISPOSED WHICH WILL RESULT IN AN UNREPRESENTATIVE QUARTERLY SAMPLE TO CHARACTERIZE THE THIRD QUARTER OF 1987.

[22] BROWNS FERRY 2 DOCKET 50-260 LER 86-010 REV 02
UPDATE ON RECIRCULATION INLET NOZZLE SAFE END CRACKS.
EVENT DATE: 070286 REPORT DATE: 091587 NSSS: GE TYPE: BWR

(NSIC 206477) ON JULY 2, 1986, SENSITIVE ULTRASONIC EXAMINATION REVEALED CRACK INDICATIONS IN ALL 10 OF THE UNIT 2 RECIRCULATION SYSTEM REACTOR VESSEL INLET NOZZLE SAFE ENDS. THE CRACK INDICATIONS ARE LOCATED ADJACENT TO THE THERMAL SLEEVE CREVICE ON THE INSIDE SURFACE. CRACK SIZING PROCEDURES DETERMINED SIGNIFICANT DEGRADATION OF THE PRESSURE BOUNDARY IN BOTH DEPTH AND CIRCUMFERENCE OF THE FLAW. A SAMPLE OF THE UNIT 2 SAFE ENDS WERE INSPECTED EARLIER IN THE OUTAGE, USING LESS SPECIALIZED METHODS, AND NO CRACKING WAS DETECTED. TVA UNDERTOOK A PROGRAM TO REPLACE THE INLET SAFE ENDS AND A PORTION OF THE RISER PIPING INCLUDING THE ELBOW ON THE UNIT 2 RECIRCULATION SYSTEM. A TUNING FORK DESIGN WAS USED IN THE NEW REPLACEMENT SAFE ENDS TO ELIMINATE THE CREVICE REGION WHICH IS THOUGHT TO BE THE CAUSE OF THE CRACKING. PRIOR TO FITUP OF THE NEW REPLACEMENT SAFE ENDS, CRACKS WERE DISCOVERED IN SOME OF THE THERMAL SLEEVES. THE CRACKS DID NOT EXTEND THROUGH-WALL AND MOST WERE LOCATED ADJACENT TO THE SAFE END ATTACHMENT WELD. ALL REPAIRS ARE COMPLETE AND THE REMOVED SAFE ENDS ARE BEING EXAMINED TO QUANTIFY THE EXTENT OF CRACKING IN BOTH THE SAFE END AND THE PORTION OF ATTACHED THERMAL SLEEVE.

[23] BROWNS FERRY 2 DOCKET 50-260 LER 87-006
FIRE WATCH FUNCTION WAS NOT FULFILLED WHILE GRINDING ACTIVITIES WERE IN PROGRESS DUE TO PERSONNEL ERROR.
EVENT DATE: 082187 REPORT DATE: 091887 NSSS: GE TYPE: BWR

(NSIC 206429) ON AUGUST 20, 1987, AND AUGUST 29, 1987, WITH ALL THREE UNITS DEFUELED, SIMILAR INCIDENTS OCCURRED INVOLVING GRINDING WITHOUT FIRE WATCHES AS REQUIRED BY TECH SPEC 3.11.H. THE FIRST EVENT WAS DISCOVERED AT 1440, ON AUGUST 21, 1987, DURING A WEEKLY FIRE PROTECTION TEAM INSPECTION. WITH GRINDING WORK IN PROGRESS THE TEAM DISCOVERED THE FIRE WATCH ASLEEP. THE FIRE WATCH WAS IMMEDIATELY AWAKENED. DISCIPLINARY ACTION HAS BEEN INITIATED. THE SECOND EVENT WAS DISCOVERED AT 1015, ON AUGUST 29, 1987, WHEN A FIRE PROTECTION ENGINEER ON ROUTINE ROUNDS NOTICED THAT GRINDING WORK WAS IN PROGRESS WITHOUT THE FIRE WATCH PRESENT. THE FIRE PROTECTION ENGINEER STOPPED THE GRINDING WORK AND THE SHIFT ENGINEER VOIDED THE WELDING PERMIT. THE FIRE WATCH HAD BEEN PRESENT AT THE WORK LOCATION DURING A LONG PREPARATION PERIOD, PRIOR TO THE BEGINNING OF THE GRINDING WORK. POOR COMMUNICATIONS BETWEEN THE FIRE WATCH, THE CRAFT PERSONNEL AND THE CRAFT FOREMAN IS ATTRIBUTED TO THE ERROR. CRAFT PERSONNEL AND THEIR SUPERVISORS HAVE BEEN COUNSELED ON THE REQUIREMENTS AND IMPORTANCE OF THE FIRE WATCH FUNCTION. A DESCRIPTION OF THESE EVENTS WILL BE PROVIDED TO PERSONNEL WHO MAY BE INVOLVED IN WORK REQUIRING FIRE WATCHES.

[24] BROWNS FERRY 2 DOCKET 50-260 LER 87-008
PERSONNEL ERROR CAUSES ENGINEERED SAFETY FEATURE ACTUATION.
EVENT DATE: 082687 REPORT DATE: 092587 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 1 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 206487) ON AUGUST 26, 1987, AT 0945, WITH ALL THREE UNITS DEFUELED, AN UNPLANNED ACTUATION OF STANDBY GAS TREATMENT, CONTROL ROOM EMERGENCY VENTILATION AND REFUEL ZONE ISOLATION OCCURRED DURING THE PERFORMANCE OF A RESTART TEST PROCEDURE. PERSONNEL ERROR DURING INSTALLATION OF JUMPERS CAUSED A RELAY TO DEENERGIZE AND INITIATE THE ENGINEERED SAFETY FEATURES. THE PERSONNEL INVOLVED WERE COUNSELED AND ALL RESTART TEST PERSONNEL RECEIVED TRAINING ON THE EVENT.

[25] BRUNSWICK 1 DOCKET 50-325 LER 87-005 REV 02
UPDATE ON PRIMARY CONTAINMENT ISOLATION VALVE PROBLEMS REVEALED THROUGH LOCAL LEAK RATE TESTING.
EVENT DATE: 022887 REPORT DATE: 092287 NSSS: GE TYPE: BWR
VENDOR: ANCHOR/DARLING INDUSTRIES

(NSIC 206493) DURING THE UNIT 1 1987 REFUEL/MAINTENANCE OUTAGE, LOCAL LEAK RATE TESTING (LLRT) OF PRIMARY CONTAINMENT ISOLATION VALVES (PCIVS) IDENTIFIED NONQUANTIFIABLE LEAKAGE RATES ON FOUR VALVES, THEREBY RESULTING IN A CALCULATED PRIMARY CONTAINMENT LEAKAGE RATE OF > 0.60 LA BASED ON THE MAXIMUM PATHWAY ANALYSIS METHOD FOR ANALYZING CONTAINMENT LEAKAGE. CONSEQUENTLY, TECHNICAL SPECIFICATION 3.6.1.2B WAS EXCEEDED. THESE VALVES ARE LOCATED ON THE REACTOR FEEDWATER (B21) SYSTEM A LINE (B21-F032A AND B21.F010A) AND ON THE DRYWELL RETURN LINE OF CONTAINMENT ATMOSPHERE CONTROL (CAC) SYSTEM ANALYSIS QUANTIFIER HIGH (AQH) MONITOR CAC-AQH-1262 (CAC-SOLENOID VALVE (SV)-1211E AND CAC-SV-3439). THE PROBLEMS IDENTIFIED INVOLVED THE VALVES SEAT GASKETS, SEAT DISCS, AND A CRACKED WELD ON A VALVE BODY-TO-LEAK-OFF LINE. THE VALVES WERE REPAIRED AND RETURNED TO SERVICE. ON 6/5/87, THE LLRTS WERE COMPLETED WITH A CALCULATED PRIMARY CONTAINMENT LEAKAGE RATE OF <0.60 LA.

[26]   BRUNSWICK 1   DOCKET 50-325   LER 87-021
UNIT 1 PRIMARY CONTAINMENT GROUPS 2, 3, AND 6 ISOLATIONS AND B LOGIC SCRAM SIGNAL DUE TO INADVERTENT DEENERGIZATION OF UNITS 1 AND 2 COMMON EMERGENCY AC BUS E2.
EVENT DATE: 082187   REPORT DATE: 091887   NSSS: GE   TYPE: BWR
OTHER UNITS INVOLVED: BRUNSWICK 2 (BWR)

(NSIC 206062) AT 1510 HOURS ON 8/21/87, THE UNITS 1 AND 2 COMMON 4160 VOLT (V) ALTERNATING CURRENT (AC) EMERGENCY BUS E2 AUTOMATICALLY DEENERGIZED WHEN THE BUS B PHASE POTENTIAL TRANSFORMER OUTPUT FUSE BLEW. UNIT 1 B LOGIC SCRAM SIGNAL AND UNIT 1 GROUPS 2, 3, AND 6 ISOLATIONS OCCURRED. THIS OCCURRED WHILE BOTH UNITS WERE AT 100% POWER AND THE E2 EMERGENCY DIESEL GENERATOR (DG) WAS UNDER EQUIPMENT CLEARANCE FOR ROUTINE SCHEDULED MAINTENANCE ACTIVITIES. CONTROL OPERATORS OF BOTH UNITS BECAME AWARE OF THIS EVENT THROUGH CONTROL ROOM INDICATION AND ALARM ANNUNCIATION. FOLLOWING REPLACEMENT OF THE BLOWN FUSE, E2 WAS REENERGIZED AT 1548 HOURS. AT 1549 HOURS, THE ISOLATION SIGNALS WERE RESET AND THE B LOGIC SCRAM SIGNAL WAS RESET. THIS EVENT RESULTED FROM ACCIDENTAL ELECTRICAL GROUNDING OF ONE OF THE BUS B PHASE VOLTAGE SENSING TRANSDUCER LEADS WHILE RECONNECTING THE LEAD FOLLOWING PLANT MODIFICATION ACCEPTANCE TESTING AS PART OF THE EMERGENCY RESPONSE FACILITY INFORMATION SYSTEM (ERFIS) INSTALLATION. REVIEWS OF ERFIS RELATED PLANT MODIFICATION STEPS WILL BE CONDUCTED TO IDENTIFY AND RESOLVE CONCERNS RELATIVE TO FUTURE MANIPULATION OF ENERGIZED LEADS. APPROPRIATE PROCEDURES WILL BE IMPLEMENTED TO INSTRUCT ELECTRICIANS/TECHNICIANS CONCERNING THE LIFTING OF ENERGIZED LEADS.

[27]   BRUNSWICK 2   DOCKET 50-324   LER 87-009
PRIMARY CONTAINMENT GROUP 3 ISOLATION RESULTING FROM STICKING DIFFERENTIAL TEMPERATURE "READ-SET" SWITCH.
EVENT DATE: 081787   REPORT DATE: 091687   NSSS: GE   TYPE: BWR
VENDOR: GENERAL ELECTRIC CORP. (NUCLEAR ENG DIV)
        RILEY COMPANY, THE - PANALARM DIVISION

(NSIC 206061) AT 1455 HOURS ON 8/17/87, A PRIMARY CONTAINMENT GROUP 3 ISOLATION OF THE REACTOR WATER CLEANUP (RWCU) SYSTEM (G31) INLET OUTBOARD PRIMARY CONTAINMENT ISOLATION VALVE, G31.F004, OCCURRED WHILE THE SHIFT FOREMAN WAS PERFORMING A CONTROL ROOM BACK PANELS CHECK OF RWCU ROOM TEMPERATURES. AT THE TIME, UNIT 2 WAS AT 100%. THE CONTROL OPERATOR WAS MADE AWARE OF THIS EVENT THROUGH CONTROL ROOM INDICATION AND ALARM ANNUNCIATION. THE GROUP 3 SIGNAL WAS RESET AT APPROXIMATELY 2300 HOURS, AND THE RWCU SYSTEM WAS RETURNED TO SERVICE AT 2330 HOURS ON 8/17/87. THIS EVENT IS ATTRIBUTED TO A STICKING "READ-SET" SWITCH IN RWCU ROOM TEMPERATURE DIFFERENTIAL SWITCH (TDS), G31.TDS-N602D, WHICH ALLOWED RESIDUAL VOLTAGE TO REMAIN IN THE CIRCUITRY OF THE SUBJECT BACK PANEL TEMPERATURE INDICATOR (TI), G31.TI.7004. WHEN RWCU ROOM TDS G31.TDS.N602B WAS SUBSEQUENTLY READ, THE RESULTING ADDITIVE SIGNAL WAS SUCH THAT THE ALARM TRIP SETPOINT OF G31-TDS.N602B WAS REACHED AND THE ISOLATION OCCURRED. THE PROBLEM AFFECTING N602D WILL BE RESOLVED FOLLOWING RECEIPT OF A REPLACEMENT MODULE FOR THE INSTRUMENT. CAUTION TAGS RELATIVE TO THE OPERATION OF N602D READ-SET SWITCH AND TESTING OF N602B ARE BEING USED TO PREVENT FUTURE SIMILAR OCCURRENCES UNTIL REPAIR CAN BE AFFECTED.

[28]   BYRON 1   DOCKET 50-454   LER 87-018
REACTOR TRIP CAUSED BY MAIN FEEDWATER PUMP TRIP DUE TO A BROKEN WIRE IN THE THRUST BEARING WEAR CIRCUITRY.
EVENT DATE: 081187   REPORT DATE: 090387   NSSS: WE   TYPE: PWR
OTHER UNITS INVOLVED: BYRON 2 (PWR)
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 206322) ON AUGUST 11 BYRON UNIT ONE WAS AT 97% POWER WHEN AN ALARM FOR THE 1B FEEDWATER PUMP THRUST BEARING WEAR ANNUNCIATED. OPERATORS WENT OUT LOCALLY AND CHECKED THE AFFECTED PUMP. A HIGH FREQUENCY NOISE INDICATED THE ALARM WAS

VALID. THE UNIT LOAD WAS REDUCED TO FACILITATE TAKING THE FEEDWATER PUMP OFF LINE. THEN THE PUMP TRIPPED. A LOAD REDUCTION TO 50% POWER WAS INITIATED, BUT NOT IN TIME AS THE REACTOR TRIPPED ON LOW STEAM GENERATOR LEVEL. ALL PLANT SAFETY SYSTEMS RESPONDED AS EXPECTED. TAVG WAS STABILIZED 45 MINUTES FOLLOWING THE TRIP. A FEEDWATER ISOLATION SIGNAL WAS GENERATED BY A HIGH-2 LEVEL IN THE 1C STEAM GENERATOR LEVEL. THE ACTUATION ALERTED THE OPERATORS TO THE PROBLEM, AND THE ISOLATED FLOW TO THE AFFECTED STEAM GENERATOR AND RESTORED THE LEVEL TO NORMAL. THE THRUST BEARING WEAR CIRCUITRY WAS CHECKED AND A BROKEN WIRE TO ONE OF THE PROXIMITY PROBES WAS FOUND. THE WIRE WAS REPAIRED AND THE INSTRUMENT CALIBRATED. TO ENSURE THE PUMP WAS OPERABLE, A CHECK OF THE SHAFT PLAY WAS DONE. NO EXCESSIVE WEAR WAS DETECTED. DURING THE STARTUP THE FEEDWATER PUMP WAS MONITORED AND FOUND TO HAVE NO OPERATIONAL PROBLEMS.

[29] BYRON 1 DOCKET 50-454 LER 87-019
SAFETY INJECTION AND REACTOR TRIP FROM LOW STEAM LINE PRESSURE DUE TO FAILED MAIN TURBINE THROTTLE VALVE DURING THE THROTTLE VALVE TO GOVERNOR VALVE TRANSFER.
EVENT DATE: 081287 REPORT DATE: 091087 NSSS: WE TYPE: PWR
VENDOR: LIMITORQUE CORP.
MOOG INC.

(NSIC 206323) ON 08/12/87 BYRON UNIT 1 WAS AT 6.5 PERCENT POWER RETURNING TO POWER OPERATIONS. DURING THE THROTTLE TO GOVERNOR VALVE TRANSFER ON THE MAIN TURBINE, A THROTTLE VALVE FAILURE INITIATED EVENTS LEADING TO A TURBINE OVERSPEED. MANUAL CONTROL OF THE TURBINE WAS TAKEN IN ORDER TO REDUCE TURBINE SPEED. ONCE THE SPEED WAS REDUCED THE OPERATOR RETURNED THE TURBINE CONTROLLER BACK IN THE AUTO MODE OF OPERATION. WHEN AUTO WAS SELECTED THE GOVERNOR VALVES WENT OPEN RESULTING IN A REACTOR TRIP AND SAFETY INJECTION. THE INTERMEDIATE CAUSE FOR THE THROTTLE VALVE FAILING CLOSED WAS DETERMINED TO BE A FAILED MOOG SERVO VALVE. THE ROOT CAUSE FOR THE VALVE'S FAILURE MODE IS STILL UNDER INVESTIGATION AND WILL BE REPORTED IN A SUPPLEMENTAL REPORT. THE DEFECTIVE MOOG SERVO VALVE WAS REPLACED AND A THROTTLE VALVE TO GOVERNOR VALVE TRANSFER WAS SIMULATED REPEATEDLY TO ENSURE THAT THE CORRECTIVE ACTIONS WERE SUCCESSFUL. PLANT STARTUP PROCEDURES WILL BE REVISED TO INSTRUCT THE OPERATOR TO TRIP THE TURBINE IF MANUAL CONTROL HAS TO BE TAKEN PRIOR TO SYNCHRONIZATION IN ORDER TO REINITIALIZE THE TURBINE CONTROLLER. THERE HAS BEEN NO PREVIOUS REPORTABLE EVENTS.

[30] BYRON 1 DOCKET 50-454 LER 87-020
TEMPORARY LACK OF CONTINUOUS FIRE WATCH DUE TO COGNITIVE PERSONNEL ERROR.
EVENT DATE: 090287 REPORT DATE: 092387 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: BYRON 2 (PWR)

(NSIC 206613) ON SEPTEMBER 2, 1987, AT 1235, WITH UNIT 1 AT 86% POWER AND UNIT 2 IN MODE 4, A CONTINUOUS FIRE WATCH IN THE UNIT 2 LOWER CABLE SPREADING ROOM (LCSR) LEFT HIS ASSIGNED AREA TO COMPLETE HIS WHOLE BODY COUNT PROCESSING. THE ACTION WAS TAKEN WITH THE CO(2) SYSTEM FOR THE UNIT 2 LCSR INOPERABLE. THE CONTRACTOR ASSIGNED AS THE FIRE WATCH WAS SCHEDULED FOR TERMINATION ON THE SAME DAY AS THE EVENT. AFTER HE RECEIVED THE EMPLOYMENT TERMINATION PROCESSING PAPER FROM HIS FOREMAN, HE LEFT HIS ASSIGNED POST TO COMPLETE THE PROCESSING, THINKING HE WAS IMMEDIATELY RELEASED FROM HIS JOB RESPONSIBILITIES. THE STATION FIRE MARSHAL REPORTED THE CONDITION TO THE SHIFT ENGINEER WHERE THE UNIT 2 LCSR WAS IMMEDIATELY POSTED WITH A QUALIFIED FIRE WATCH. THE AREA WAS WITHOUT AN OPERABLE CO(2) SUPPRESSION SYSTEM AND WITHOUT A CONTINUOUS FIRE WATCH FOR 1 HOUR AND 7 MINUTES. THE CAUSE OF THE EVENT WAS A COGNITIVIE PERSONNEL ERROR. THERE HAVE BEEN 3 PREVIOUS OCCURRENCES OF THIS NATURE.

[31] BYRON 2 DOCKET 50-455 LER 87-011
REACTOR TRIP DURING 30% LOAD REJECTION TEST ON OVERTEMPERATURE DELTA T DUE TO STEAM DUMPS FAILURE TO FULLY OPEN.
EVENT DATE: 072587 REPORT DATE: 081987 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 205904) AT 1216 ON 7/25/87, UNIT 2 TRIPPED ON AN OVERTEMPERATURE CONDITION FOLLOWING A 30 PERCENT LOAD REJECTION TEST. THE CAUSE OF THE REACTOR TRIP WAS PARTIALLY DUE TO THE DIGITAL ELECTRO-HYDRAULIC (DEH) CONTROLLER BEING IN THE SEQUENTIAL VALVE MODE PROGRAM WHICH CAUSED THE GOVERNOR CONTROL VALVES TO BE IN THE MOST EFFICIENT POWER OPERATION, BUT DURING THE RUNBACK CAUSED THE VALVES TO BE IN VARYING POSITIONS CAUSING T(REF) TO WIDELY VARY. IN ADDITION, THE STEAM DUMPS DID NOT STROKE FULLY OPEN RESULTING IN DECREASED SECONDARY SIDE HEAT REMOVAL CAPABILITY. TO CORRECT THE CONDITIONS WHICH LED TO THE REACTOR TRIP, THE DEH COMPUTER IS BEING EXAMINED BY WESTINGHOUSE AND CHANGES WILL BE IMPLEMENTED AS REQUIRED. THE STEAM DUMPS WERE REPAIRED AND TESTED TO VERIFY POWER OPERATION, AND THE FEEDWATER SYSTEM WAS FINE TUNED FOR BETTER STEAM/FEED CONTROL. LOAD REJECTION TESTS OF 30 AND 50 PERCENT HAVE SINCE BEEN SUCCESSFULLY RUN AS PART OF THE STARTUP PROGRAM.

[32] BYRON 2 DOCKET 50-455 LER 87-013
INCORE FLUX MAPPING TEST PERFORMED IN MODES OTHER THAN SPECIFIED IN FSAR.
EVENT DATE: 081987 REPORT DATE: 091787 NSSS: WE TYPE: PWR

(NSIC 206324) STARTUP TEST 2.45.80, INCORE FLUX MAPPING SYSTEM CHECKOUT WAS PERFORMED IN MODES 3, 4, AND 5. FSAR TABLE 14.2-67 WHICH DESCRIBES THIS TEST INDICATES THAT THE TEST WILL BE PERFORMED IN MODE 4. THIS DEVIATION FROM THE FSAR DESCRIPTION WAS NOT REPORTED TO THE COMMISSION WITHIN ONE MONTH AS REQUIRED BY BYRON UNIT 2 OPERATING LICENSE NPF-66 CONDITION 2.C.3. THE CAUSES OF THIS EVENT WERE COGNITIVE PERSONNEL ERRORS BY THE CORPORATE ENGINEER RESPONSIBLE FOR REVIEWING THE TEST AND THE STATION ENGINEER PERFORMING THE 10CFR 50.59 EVALUATION. A MEMORANDUM WAS ISSUED TO THE CORPORATE ENGINEERING GROUP RESPONSIBLE FOR TEST REVIEWS TO EMPHASIZE THE NEED FOR IMMEDIATE NOTIFICATION OF THE LICENSING GROUPS WHEN ANY ISSUE REGARDING FSAR ITEMS NOT SATISFIED. THERE HAVE BEEN NO PREVIOUS OCCURRENCES.

[33] BYRON 2 DOCKET 50-455 LER 87-014
INOPERABLE CONTAINMENT AIRLOCK DUE TO MISSED SURVEILLANCE.
EVENT DATE: 082487 REPORT DATE: 091787 NSSS: WE TYPE: PWR

(NSIC 206416) ON 8/24/87 IT WAS DISCOVERED THAT 2BVS 6.1.3.A-1, PRIMARY CONTAINMENT TYPE B LOCAL LEAKAGE RATE TESTS OF THE EQUIPMENT HATCH AIRLOCK DOOR GASKET INTERSPACES, WAS PERFORMED 16 1/2 HOURS BEYOND THE TECH SPEC REQUIRED INTERVAL. THE CAUSE OF THE EVENT WAS A COGNITIVE PERSONNEL ERROR AND LACK OF PROPER PROCEDURES. CORRECTIVE ACTIONS INCLUDE INCREASED TESTING FREQUENCY, DEVELOPMENT OF AN ADMINISTRATIVE PROGRAM AND ULTIMATELY THE INSTALLATION OF AUTOMATIC LEAK DETECTION SYSTEMS.

[34] BYRON 2 DOCKET 50-455 LER 87-015
FEEDWATER ISOLATION ACTUATION ON HIGH-2 STEAM GENERATOR LEVEL DUE TO PERSONNEL ERROR.
EVENT DATE: 082987 REPORT DATE: 092387 NSSS: WE TYPE: PWR

(NSIC 206556) ON AUGUST 29, 1987, AT 0156, BYRON UNIT 2 WAS IN MODE 4 AT ZERO PERCENT POWER. INSTRUMENT AND CONTROL TECHNICIANS (NON-LICENSED) WERE CALIBRATING ONE OF THE 2D STEAM GENERATOR LEVEL TRANSMITTERS. THE TECHNICIAN IN CONTAINMENT ISOLATED THE WRONG LEVEL TRANSITTER. THIS ACTION SATISFIED A 2 OF 4 COINCIDENCE CAUSING A FEEDWATER ISOLATION ON STEAM GENERATOR HIGH-2 LEVEL, P-14. THE

FEEDWATER ISOLATION ACTUATED AS DESIGNED. THE LEVEL TRANSMITTER WAS UNISOLATED AND THE FEEDWATER ISOLATION SIGNAL WAS RESET. THE CAUSE OF THE ACTUATION WAS A COGNITIVE PERSONNEL ERROR ON THE PART OF THE INSTRUMENT AND CONTROL TECHNICIAN. THE TECHNICIAN WAS DISCIPLINED. THERE HAVE BEEN NO PREVIOUS OCCURRENCES OF THIS NATURE.

[35] BYRON 2 DOCKET 50-455 LER 87-017
MAIN STEAM ISOLATION BYPASS VALVE RETURNED TO SERVICE WITHOUT COMPLETING THE REQUIRED POST MAINTENANCE TESTING.
EVENT DATE: 090387 REPORT DATE: 092587 NSSS: WE TYPE: PWR

(NSIC 206614) ON SEPTEMBER 3, 1987, AT 0800, DURING THE QUALITY CONTROL REVIEW OF THE COMPLETED NUCLEAR WORK REQUEST PACKAGE, IT WAS DETERMINED THAT THE MAIN STEAM ISOLATION BYPASS VALVE, 2MS101C, WAS RETURNED TO SERVICE WITHOUT PERFORMING THE REQUIRED POST MAINTENANCE TESTING. THE OPERATING DEPARTMENT STROKED THE VALVE; BUT, THE STROKE TIME WAS NOT OBTAINED PRIOR TO RETURNING THE VALVE TO SERVICE. THE ROOT CAUSE OF THE EVENT WAS A COGNITIVE PERSONNEL ERROR. THE STATION CONTROL ROOM ENGINEER FAILED TO COMPLETE THE POST MAINTENANCE CHECKLIST PRIOR TO RETURNING THE VALVE TO SERVICE. THE CORRECTIVE ACTIONS INCLUDE A REVISION TO THE CHECKLIST WHICH WILL REQUIRE AN SRO SIGNATURE. TRAINING ON THE PROPER COMPLETION OF THE CHECKLIST WILL BE GIVEN TO THE OPERATING DEPARTMENT. THERE HAVE BEEN FIVE PREVIOUS OCCURRENCES OF THIS NATURE.

[36] CALLAWAY 1 DOCKET 50-483 LER 87-019
TECHNICAL SPECIFICATION CONTAINMENT SHUTDOWN PURGE ISOLATION VALVE NOT SURVEILLED DUE TO PROGRAMMATIC INADEQUACIES.
EVENT DATE: 050887 REPORT DATE: 091687 NSSS: WE TYPE: PWR

(NSIC 206197) ON 5/7/87 BLANK FLANGES WERE REINSTALLED ON THE CONTAINMENT SHUTDOWN PURGE ISOLATION VALVES PRIOR TO ASCENDING TO MODE 4, HOT SHUTDOWN ON 5/8/87. CONTRARY TO THE REQUIREMENTS OF TECHNICAL SPECIFICATION (T/S) 4.6.1.7.2, A LOCAL LEAK RATE TEST (LLRT) WAS NOT PERFORMED WHEN THE FLANGES WERE REINSTALLED FOLLOWING AN INTEGRATED LEAK RATE TEST (ILRT) AND THUS THE PLANT SHOULD NOT HAVE ENTERED MODE 4 AS PER T/S 4.0.4. ON 8/17/87, PLANT ENGINEERING PERSONNEL DISCOVERED THE MISSED SURVEILLANCE. THIS INCIDENT OCCURRED BECAUSE THE CONDITIONAL PART OF T/S 4.6.1.7.2 HAD NOT BEEN PROPERLY IDENTIFIED IN THE SURVEILLANCE TRACKING SYSTEM. ON 8/18/87, A LLRT WAS SUCCESSFULLY PERFORMED ON EACH CONTAINMENT SHUTDOWN PURGE ISOLATION VALVE AND BLANK FLANGE. TO PREVENT RECURRENCE, THE SURVEILLANCE TASK SHEETS, SURVEILLANCE TRACKING SCHEDULE, AND APPROPRIATE PROCEDURES WERE UPDATED TO REFLECT THIS CONDITIONAL REQUIREMENT ON 8/25/87. A CONDITIONAL SURVEILLANCE TASK FORCE HAS BEEN FORMED TO VERIFY THAT CONDITIONAL T/S SURVEILLANCE REQUIREMENTS ARE IDENTIFIED AND HAVE MECHANISMS IN PLACE TO ENSURE THEY ARE PERFORMED. BECAUSE THE LLRT WAS SUCCESSFULLY PERFORMED ON 8/18/87 AND THE FLANGES HAD NOT BEEN CYCLED SINCE 5/7/87, THE VALVES AND FLANGES PROVIDED PROPER ISOLATION OF THE CONTAINMENT SHUTDOWN PURGE SYSTEM. THUS, THERE WAS NO DANGER TO THE PUBLIC HEALTH AND SAFETY.

[37] CALLAWAY 1 DOCKET 50-483 LER 87-012 REV 01
UPDATE ON SCALING ERROR IN THE DELTA I INPUT TO THE OT DELTA T REACTOR TRIP SETPOINT.
EVENT DATE: 062587 REPORT DATE: 072987 NSSS: WE TYPE: PWR

(NSIC 205654) ON 6/25/87 WHILE AT 99% POWER, IT WAS DETERMINED THAT THE CIRCUITRY, WHICH PROVIDES A CORRECTION TO THE OVERTEMPERATURE-DELTA TEMPERATURE (OT DELTA T) REACTOR TRIP SETPOINT BASED ON CHANGES IN AXIAL FLUX DIFFERENCE (DELTA I), HAD BEEN SCALED INCORRECTLY. THE SCALING HAD BEEN INCORRECT SINCE INITIAL STARTUP IN NOVEMBER OF 1984 AND WAS NONCONSERVATIVE IN THAT THE OT DELTA T SETPOINT WOULD NOT HAVE BEEN DECREASED UNTIL VALUES OF DELTA I WERE REACHED

GREATER THAN THAT REQUIRED BY THE TECHNICAL SPECIFICATIONS. THE SCALING WAS CORRECTED ON 6/25/87 BY A TEMPORARY MODIFICATION AND A DESIGN CHANGE WAS INITIATED TO PERMANENTLY IMPLEMENT THE CHANGE. THE APPROPRIATE I&C SURVEILLANCE PROCEDURES WERE REVISED AND ARE BEING INDEPENDENTLY REVIEWED FOR TECHNICAL ACCURACY. THE APPLICABLE ENGINEERING SURVEILLANCE PROCEDURE WILL BE REVISED PRIOR TO ITS NEXT PERFORMANCE. THE INCORRECT SCALING WAS THE RESULT OF INTER-DEPARTMENTAL MISUNDERSTANDING OF THE ADJUSTMENTS REQUIRED TO ACCOUNT FOR INCORE-EXCORE CALIBRATIONS. A CONTRIBUTING FACTOR WAS THAT THE INSTRUMENT SETPOINT INDEX SPECIFIED FIXED GAIN VALUES RATHER THAN VARIABLE FOR VARIOUS COMPONENTS. THIS EVENT HAS BEEN DISCUSSED WITH THE APPROPRIATE PERSONNEL. I&C SURVEILLANCE PROCEDURES REQUIRING INPUT FROM OTHER DEPARTMENTS WHICH IMPACT REACTOR PROTECTION SCALING WERE REVIEWED AND NO ADDITIONAL DEFICIENCIES WERE IDENTIFIED.

[38]   CALLAWAY 1   DOCKET 50-483   LER 87-016
TECHNICAL SPECIFICATION VIOLATION TRITIUM SAMPLE MISSED DUE TO PERSONNEL ERROR.
EVENT DATE: 062587   REPORT DATE: 081887   NSSS: WE   TYPE: PWR

(NSIC 206280) ON 6/25/87 AT 0800 CDT, THE MONTHLY SAMPLING AND ANALYSIS OF UNIT VENT GASEOUS EFFLUENTS WAS NOT PERFORMED AS REQUIRED BY TECHNICAL SPECIFICATIONS (T/S). HEALTH PHYSICS PERSONNEL FAILED TO CONSULT THE APPROPRIATE PROCEDURE PRIOR TO PERFORMANCE OF THE SURVEILLANCE AND SUBSEQUENTLY FAILED TO OBTAIN THE REQUIRED TRITIUM SAMPLE. AN INCOMPLETE TASK DESCRIPTION SUMMARY (TDS) ON THE SURVEILLANCE TASK SHEET WAS A CONTRIBUTING FACTOR TO THIS ERROR. THE MISSED TRITIUM SAMPLE WAS DISCOVERED 7/20/87 DURING A REVIEW OF RELEASE PERMIT DATA. THE SAMPLE WAS THEN IMMEDIATELY TAKEN AND ANALYZED. THE CONCENTRATION WAS WITHIN T/S LIMITS. THE PLANT WAS IN MODE 1, POWER OPERATIONS, AT 100 PERCENT POWER DURING THIS TIME PERIOD. TIMING OF THIS LER IS BASED UPON THE DATE OF DISCOVERY OF THIS INCIDENT. TO PREVENT RECURRENCE, THE REQUIREMENT TO UTILIZE APPROPRIATE SURVEILLANCE PROCEDURES WAS EMPHASIZED TO HEALTH PHYSICS TECHNICAL SUPPORT TECHNICIANS, AND THE HEALTH PHYSICS PERSONNEL INVOLVED HAVE RECEIVED DISCIPLINARY COUNSELLING. THE MISSED SAMPLE HAD NO DETRIMENTAL EFFECTS ON THE CONTROL OF RADIOACTIVE EFFLUENTS AND THERE WERE NO OPERATIONAL ABNORMALITIES DURING THIS PERIOD WHICH WOULD HAVE CAUSED TRITIUM CONCENTRATIONS TO INCREASE. THEREFORE, SINCE THE CONCENTRATIONS MEASURED WERE SIGNIFICANTLY LESS THAN THE MAXIMUM VALUE ALLOWED BY T/S, THIS EVENT POSED NO THREAT TO THE PUBLIC HEALTH AND SAFETY.

[39]   CALLAWAY 1   DOCKET 50-483   LER 87-018
INOPERABLE ESSENTIAL SERVICE WATER SYSTEM AND TECHNICAL SPECIFICATION 3.0.3 UNKNOWINGLY ENTERED DUE TO PERSONNEL ERRORS.
EVENT DATE: 081587   REPORT DATE: 091487   NSSS: WE   TYPE: PWR
VENDOR: JAMES BURY CORP.

(NSIC 206196) ON 8/15/87 AT 0510 CDT, DURING A CONTAINMENT COOLING FAN TEST, UTILITY OPERATORS DISCOVERED ESSENTIAL SERVICE WATER (ESW) TRAIN 'B' TO THE ULTIMATE HEAT SINK ISOLATION VALVE, EF-V-0117, PARTIALLY SHUT. TRAIN 'B' WAS DECLARED INOPERABLE, THE VALVE WAS OPENED, AND TRAIN 'B' WAS DECLARED OPERABLE AT 1431. AN EVALUATION CONCLUDED THAT TOTAL TRAIN 'B' FLOW WITH THIS RESTRICTION WAS LESS THAN SPECIFIED BY DESIGN. THIS CONDITION HAD EXISTED SINCE 5/11/84 AND ITS POSITION INDICATORS HAD CONFLICTED SINCE NOTED ON A WORK REQUEST (WR) ON 5/14/84. SINCE TRAIN 'A' HAS BEEN REMOVED FROM SERVICE FOR TESTING, BOTH ESW TRAINS HAVE BEEN SIMULTANEOUSLY INOPERABLE AND TECH SPEC 3.0.3 UNKNOWINGLY ENTERED. THIS EVENT WAS DUE TO FAILURE OF UTILITY PERSONNEL TO RECOGNIZE THE EFFECT THE INDICATION PROBLEM HAD ON ESW OPERABILITY RESULTING IN LOW WORK PRIORITY PLACED ON REPAIRING THE PROBLEM. THE CAUSE OF THE DELAY IN DISCOVERING THE FLOW PROBLEM WAS FAILURE OF UTILITY PERSONNEL TO COMPARE TOTAL FLOW TO PRE-OP FLOWS WHEN BASELINING THE PUMPS IN 1984 AND FEBRUARY 1987. THE VALVE ACTUATOR HAS BEEN CAUTION TAGGED AND WILL BE REPAIRED DURING REFUEL II. VOIDED/OPEN WR'S ON SELECTED SYSTEMS WILL BE REVIEWED AND THE WR VOIDING PROCESS WILL BE REVISED.

PERSONNEL INVOLVED WERE COUNSELED. THIS EVENT WILL BE REVIEWED WITH SYSTEM ENGINEERING AND PLANNERS.

[40] CALLAWAY 1 DOCKET 50-483 LER 87-020
TECHNICAL SPECIFICATION VIOLATION WHEN TRANSPOSITION ERROR RESULTS IN MISSED CONTINUOUS FIREWATCH FOR INOPERABLE FIRE SPRINKLER SYSTEM.
EVENT DATE: 081987 REPORT DATE: 091887 NSSS: WE TYPE: PWR

(NSIC 206198) AT 0635 CDT ON 8/19/87, UTILITY OPERATIONS PERSONNEL ISOLATED THE SPRINKLER SYSTEMS TO THE TURBINE DRIVEN AUXILIARY (AUX) FEEDWATER PUMP ROOM, NORTH AND SOUTH ELECTRICAL CABLE CHASES AND AUX BUILDING (BLDG) CABLE TRAYS ELEVATIONS 1974', 2000', AND 2026', TO PERFORM MAINTENANCE ON THE AUX BLDG CABLE TRAYS ELEVATION 2026' FIRE PROTECTION HEADER ISOLATION VALVE, KC-V-0445. PER TECH SPEC (T/S) 3.7.10.2 ACTION (A), THE PUMP ROOM AND CHASES WERE PATROLLED HOURLY BY A PREVIOUSLY ESTABLISHED FIREWATCH (FW) PATROL. A CONTINUOUS FW SHOULD HAVE BEEN ESTABLISHED AT 0600 ON 8/19/87 FOR AUX BLDG ELEVATIONS 1974', 2000', AND 2026'. HOWEVER, DUE TO PERSONNEL ERROR, THE WATCH FOR THE 1974' LEVEL WAS STATIONED ON THE 2047' LEVEL. A UTILITY EQUIPMENT OPERATOR DISCOVERED THE ERROR AT 1700 AND A CONTINUOUS FW WAS ESTABLISHED ON 1974' AT 1745. FAILURE TO ESTABLISH A CONTINUOUS FW BY 0735 FOR 1974' IS A CONDITION PROHIBITED BY T/S'S. THE EQUIPMENT WAS RETURNED TO SERVICE ON 8/19/87 AT 2200. THE PLANT WAS IN MODE 1 - POWER OPERATION AT 100% POWER. THE ROOT CAUSE OF THIS EVENT IS ATTRIBUTABLE TO A UTILITY LICENSED SHIFT SUPERVISOR TRANSPOSING 1974' TO 2047' WHEN VERBALLY COMMUNICATING WITH THE LEAD FW TO DISPATCH THE WATCHES.

[41] CALLAWAY 1 DOCKET 50-483 LER 87-021
STEAM GENERATOR BLOWDOWN CONTAINMENT ISOLATION VALVE NOT SURVEYED DUE TO PERSONNEL ERROR.
EVENT DATE: 082087 REPORT DATE: 092187 NSSS: WE TYPE: PWR

(NSIC 206200) AT 1900 CDT ON 8/20/87, MAINTENANCE WAS COMPLETED ON STEAM GENERATOR BLOWDOWN ISOLATION VALVE BM-HV-0002 TO CORRECT A PACKING LEAK. TECHNICAL SPECIFICATION (T/S) 4.6.3.1 REQUIRES THAT PRIOR TO RETURNING THE VALVE TO SERVICE FOLLOWING MAINTENANCE, THE VALVE MUST BE DEMONSTRATED OPERABLE BY PERFORMANCE OF A CYCLING TEST AND VERIFICATION OF ISOLATION TIME. THIS REQUIREMENT WAS NOT MET WITHIN THE T/S REQUIRED 4 HOUR TIME PERIOD. THE PLANT WAS IN MODE 1 - POWER OPERATION, 100% REACTOR POWER AT THE TIME OF THE EVENT. ON 8/21/87, DURING A UTILITY PLANNER'S REVIEW OF THE WORK PACKAGE USED TO CORRECT THE PACKING LEAK, IT WAS IDENTIFIED THAT A CYCLING TEST HAD NOT BEEN PERFORMED. VERBAL COMMUNICATION BETWEEN THE RESPONSIBLE MAINTENANCE FOREMAN AND THE CONTROL ROOM, TO AUTHORIZE WORK COMMENCEMENT AND COMPLETION, DID NOT PROVIDE ADEQUATE INFORMATION TO INITIATE THE T/S CYCLE TEST REQUIREMENTS FOR THE VALVE. AT 1554 ON 8/21/87, BM-HV-0002 WAS SUCCESSFULLY CYCLE TESTED. THE APPLICABLE PLANT PROCEDURE, WHICH SPECIFIES THE REQUIREMENTS FOR MAINTENANCE PACKAGES, WILL BE REVISED TO MORE CLEARLY DEFINE THE CASES IN WHICH VERBAL AUTHORIZATIONS ARE PERMITTED. PROCEDURAL REQUIREMENTS WILL BE REVIEWED BY ALL UTILITY PERSONNEL AFFECTED BY THIS EVENT.

[42] CALVERT CLIFFS 1 DOCKET 50-317 LER 87-007 REV 03
UPDATE ON ENVIRONMENTAL QUALIFICATION DISCREPANCIES FOR SOLENOID VALVES RESULT IN SHUTDOWN.
EVENT DATE: 040187 REPORT DATE: 092987 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: CALVERT CLIFFS 2 (PWR)

(NSIC 206550) DURING AN NRC INSPECTION OF OUR ENVIRONMENTAL QUALIFICATION (EQ) PROGRAM IN THE WEEK OF MARCH 23, 1987, TWO SOLENOID VALVES ON EACH CALVERT CLIFFS UNIT WERE FOUND TO HAVE TAPED SPLICES WHICH HAD NOT BEEN QUALIFIED. IMMEDIATE CORRECTIVE ACTION WAS TAKEN TO REPLACE THE TAPED SPLICES WITH QUALIFIED HEAT

SHRINK TUBING ON THE DATE OF DISCOVERY (MARCH 27, 1987). AS A RESULT OF FURTHER INSPECTIONS BY BALTIMORE GAS & ELECTRIC (BG&E) ON OTHER UNIT 2 10 CFR 50.49 EQUIPMENT, BG&E MANAGEMENT DECIDED TO SHUT DOWN UNIT 1 FROM 100% POWER, MODE 1 OPERATION, TO MODE 5 (LESS THAN 200 DEGREES FAHRENHEIT) ON APRIL 1, 1987. UNIT 2 WAS IN A REFUELING OUTAGE AT THE TIME. AN EXTENSIVE INSPECTION PROGRAM WAS UNDERTAKEN TO LOOK IN DETAIL AT EQUIPMENT WITHIN THE SCOPE OF 10 CFR 50.49. ALL 10 CFR 50.49 EQUIPMENT FOR UNITS 1 AND 2 WAS DETERMINED TO BE OPERABLE AND WOULD HAVE REMAINED OPERABLE DURING ALL POSTULATED DESIGN BASIS EVENTS. ALL IDENTIFIED DISCREPANCIES WERE RESOLVED OR CORRECTED. AS A RESULT OF THESE DISCREPANCIES, INTERIUM UPGRADED EQ PROGRAM CONTROLS HAVE BEEN ESTABLISHED TO PREVENT RECURRENCE. THESE INCLUDE INTERIM METHODS TO PROVIDE CLEAR COMMUNICATIONS OF EQ MAINTENANCE REQUIREMENTS TO FIELD CRAFT PERSONNEL AND ADDITIONAL REVIEWS OF PLANNED MAINTENANCE BY DESIGN ENGINEERING PERSONNEL OR QUALIFIED EQUIPMENT QUALIFICATION REVIEWERS.

[43] CALVERT CLIFFS 2 DOCKET 50-318 LER 87-002 REV 01
UPDATE ON FAILURE OF LEAD/LAG CIRCUIT IN FEEDWATER REGULATING VALVE CONTROL SYSTEM LEADS TO LOW STEAM GENERATOR WATER LEVEL REACTOR TRIP.
EVENT DATE: 022887 REPORT DATE: 091787 NSSS: CE TYPE: PWR
VENDOR: ROCHESTER INSTRUMENT SYSTEMS, INC.

(NSIC 206315) ON FEBRUARY 28, 1987, AT 2356, THE STEAM GENERATOR DOWNCOMER LEVEL LEAD/LAG UNIT OF THE #21 FEEDWATER REGULATING VALVE (FRV) CONTROL SYSTEM FAILED TO AN OUTPUT OF +9.5 INCHES STEAM GENERATOR LEVEL. THE OUTPUT SHOULD HAVE BEEN +.5 INCHES. THE LEVEL DOMINANT THREE ELEMENT FEEDWATER CONTROL SYSTEM AUTOMATICALLY RESPONDED, IN A PROPER MANNER, BY CLOSING #21 FRV. THE CLOSED FRV CAUSED #21 STEAM GENERATOR LEVEL TO DECREASE AT A RATE OF APPROXIMATELY 0.8 INCHES PER SECOND. THE CLOSING OF #21 FRV ALSO CAUSED A HIGH DIFFERENTIAL PRESSURE ACROSS THE VALVE. THE RAPID INCREASE IN FRV DIFFERENTIAL PRESSURE CAUSED #22 STEAM GENERATOR FEED PUMP (SGFP) TO AUTOMATICALLY REDUCE SPEED (#21 SGFP WAS IN MANUAL CONTROL). CONTROL ROOM OPERATORS WERE ALERTED WHEN #22 SGFP TURBINE SPEED HOLD OR SYSTEM TROUBLE ALARM SOUNDED. OPERATORS TOOK ACTION TO FEED THE STEAM GENERATOR BUT, APPROXIMATELY 15 SECONDS AFTER THE ALARM, THE REACTOR TRIPPED ON LOW STEAM GENERATOR WATER LEVEL. ALL REACTOR SAFETY SYSTEMS FUNCUNCTIONED AS EXPECTED. BOTH STEAM GENERATOR FEED PUMPS TRIPPED SHORTLY AFTER THE REACTOR TRIP ON HIGH DISCHARGE PRESSURE. AUXILIARY FEEDWATER ACTUATION OCCURRED. EMERGENCY OPERATING PROCEDURES WERE CARRIED OUT AND THE PLANT WAS RETURNED TO A STABLE CONDITION.

[44] CATAWBA 1 DOCKET 50-413 LER 86-048 REV 01
UPDATE ON THREE CONTAINMENT PURGE SYSTEM ISOLATIONS DUE TO PROCEDURE DEFICIENCY.
EVENT DATE: 082486 REPORT DATE: 082887 NSSS: WE TYPE: PWR

(NSIC 206027) THE CONTAINMENT PURGE (VP) SYSTEM AUTOMATICALLY ISOLATED ON THREE SEPARATE OCCASIONS, DUE TO CONTAINMENT GASEOUS RADIOACTIVITY LEVELS EXCEEDING THE TRIP SETPOINT OF THE CONTAINMENT GAS MONITOR (1EMF39L). VP WAS ISOLATED AT 1620 HOURS, ON AUGUST 24, 1986, 1320 HOURS, ON AUGUST 25, 1986, AND AT 1250 HOURS, ON AUGUST 26, 1986. EACH VP ISOLATION OCCURRED JUST AFTER A STEAM GENERATOR (S/G) MANWAY DIAPHRAGM HAD BEEN REMOVED FROM THE S/GS FOR AN EDDY CURRENT INSPECTION OF THE PRIMARY SIDE. AFTER EACH VP ISOLATION, THE TRIP SETPOINT FOR 1EMF39L WAS RAISED IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS AND CONTAINMENT PURGE WAS REINITIATED. THE UNIT WAS IN MODE 6, REFUELING, AT THE TIME OF THESE INCIDENTS. THESE INCIDENTS ARE ASSIGNED CAUSE CODE D, DEFECTIVE PROCEDURE. THE HEALTH PHYSICS (HP) PROCEDURE, CONTROL OF WORK IN CONTAINMENT, DID NOT ADDRESS THE RELEASE OF RADIOACTIVE GASES DURING S/G DIAPHRAGM REMOVAL, AND THE EFFECT THIS WOULD HAVE ON 1EMF39L AND VP. THE HEALTH AND SAFETY OF THE PUBLIC WERE UNAFFECTED BY THIS INCIDENT.

[45] CATAWBA 1 DOCKET 50-413 LER 87-005 REV 02
UPDATE ON UNIT SHUTDOWNS DUE TO DESIGN DEFICIENCY WITH CONTAINMENT AIR RETURN FANS.
EVENT DATE: 013087 REPORT DATE: 090987 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: CATAWBA 2 (PWR)

(NSIC 206085) ON JANUARY 30, 1987, UNITS 1 AND 2 ENTERED TECHNICAL SPECIFICATION 3.0.3 FOLLOWING THE DETERMINATION THAT THE CONTAINMENT AIR RETURN AND HYDROGEN SKIMMER SYSTEMS WERE INOPERABLE. IT WAS DETERMINED THAT THE SYSTEMS WERE INOPERABLE DUE TO THE POSSIBILITY THAT THE CONTAINMENT AIR RETURN FANS (CARF'S) COULD BE RENDERED INOPERABLE IN THE EVENT OF A CONTAINMENT SPRAY ACTUATION WHERE THE CARF PITS WOULD BE FLOODED WITH COLLECTED SPRAY. ON BOTH UNITS, DEFLECTIVE CURBS WERE INSTALLED ON THE REACTOR BUILDING OPERATING DECK TO PREVENT COLLECTED SPRAY FROM FUNNELING INTO THE FAN PITS. UNIT 1 WAS OPERATING AT 100% WHEN TECHNICAL SPECIFICATION 3.0.3 WAS ENTERED. UNIT 2 WAS IN MODE 3, HOT STANDBY, WHEN TECHNICAL SPECIFICATION 3.0.3 WAS ENTERED. THE SYSTEMS HAD BEEN UNKNOWINGLY INOPERABLE SINCE INITIAL FUEL LOAD ON BOTH UNITS. THIS INCIDENT IS ASSIGNED CAUSE CODE B, DESIGN MANUFACTURING, CONSTRUCTION/ INSTALLATION DEFICIENCY. NO DEVICES TO PREVENT FUNNELING OF COLLECTED CONTAINMENT SPRAY INTO CARF PITS WERE SPECIFIED ON DESIGN DRAWINGS. THE CARF INOPERABILITY DID NOT IN ANY WAY INCREASE THE PROBABILITY OF A DESIGN BASIS ACCIDENT (DBA), NOR WOULD IT HAVE AFFECTED ACTUAL CORE CONDITIONS OR CREATED AN EVENT THAT LEAD TO DEGRADED CORE CONDITIONS.

[46] CATAWBA 1 DOCKET 50-413 LER 87-033
FAILURE TO VERIFY OPERABILITY OF DIESEL GENERATOR 1B DUE TO A PERSONNEL ERROR.
EVENT DATE: 040687 REPORT DATE: 090287 NSSS: WE TYPE: PWR

(NSIC 206219) ON APRIL 6, 1987, AT 1708 HOURS, THE OPERABILITY OF DIESEL GENERATOR (D/G) 1B WAS NOT VERIFIED WITHIN THE TIME LIMIT REQUIRED BY TECHNICAL SPECIFICATIONS. THIS INCIDENT WAS DISCOVERED ON AUGUST 3, 1987. THE REQUIRED WEEKLY SURVEILLANCE WAS MISSED AGAIN ON SEVEN SEPARATE OCCASIONS BEFORE DISCOVERY OF THE INCIDENT. THE MISSED SURVEILLANCES RESULTED WHEN A D/G START ATTEMPT WAS RECLASSIFIED FROM AN INVALID TEST TO A VALID FAILURE (SEE IIR C86.12-1) AND NOT PROPERLY ACCOUNTED FOR. AFTER DISCOVERY OF THE INCIDENT, THE WEEKLY OPERABILITY SURVEILLANCES WERE BEGUN AT 1525 HOURS ON AUGUST 3, 1987. THE UNIT WAS AT 100% POWER AT THE TIME OF DISCOVERY OF THIS INCIDENT. THIS INCIDENT IS CLASSIFIED AS EVENT CAUSE CODE A, PERSONNEL ERROR. ONE ERROR OCCURRED WHEN A LETTER WAS WRITTEN WHICH ASSIGNED A FAILURE TO THE WRONG D/G. ANOTHER ERROR OCCURRED WHEN THE USE OF AN UNCONTROLLED COPY OF A LOGBOOK RESULTED IN A D/G FAILURE BEING UNACCOUNTED FOR, VIOLATING TECHNICAL SPECIFICATIONS. THE THIRD ERROR OCCURRED DUE TO THE FAILURE TO RECOGNIZE THE NEED TO ADJUST THE D/G TEST FREQUENCY AS REQUIRED BY TECHNICAL SPECIFICATIONS. THIS INCIDENT AND THE IMPORTANCE OF ACCURATELY TRACKING THE D/G TESTS WERE DISCUSSED WITH THE APPROPRIATE PERSONNEL WHO WERE REMINDED TO USE ONLY THE CONTROLLED COPY OF THE D/G LOGBOOKS WHEN TRACKING D/G TEST RESULTS. THE HEALTH AND SAFETY OF THE PUBLIC WERE UNAFFECTED BY THIS INCIDENT.

[47] CATAWBA 1 DOCKET 50-413 LER 87-026
MANUAL REACTOR TRIP FOLLOWING MAIN FEEDWATER PUMP SPEED DECREASE DUE TO A MANUFACTURING DEFICIENCY.
EVENT DATE: 070687 REPORT DATE: 080587 NSSS: WE TYPE: PWR

(NSIC 205716) ON JULY 6, 1987, AT 1513:31 HOURS, WITH THE UNIT AT 100% POWER, MAIN FEEDWATER PUMP TURBINE (CFPT) 1B SUDDENLY DECREASED SPEED, WHICH STABILIZED AT APPROXIMATELY 3800 RPM. CFPT 1A SPEED AUTOMATICALLY INCREASED TO ITS MAXIMUM ATTEMPTING TO COMPENSATE FOR THE CFPT 1B RUNBACK. THE COMBINED CFPT SPEEDS COULD NOT SUPPLY ENOUGH FEEDWATER AT 100% POWER, AND STEAM GENERATOR (S/G) WATER LEVELS BEGAN TO DECREASE RAPIDLY. CONTROL ROOM PERSONNEL WERE ALERTED BY THE FEED FLOW/STEAM FLOW MISMATCH ALARM AND BEGAN A MANUAL TURBINE GENERATOR LOAD

REDUCTION, ENSURED ALL MAIN FEEDWATER (CF) CONTROL VALVES WERE OPENING, AND OPENED THE CF CONTROL BYPASS VALVES. AT 1514:00 HOURS, CFPT 1B TRIPPED ON LOW SUCTION FLOW AND MOTOR DRIVEN AUXILIARY FEEDWATER (CA) PUMP LB WAS STARTED MANUALLY TO INCREASE FEEDWATER FLOW TO THE S/GS. FOLLOWING RECEIPT OF THE S/G C LO LO LEVEL ALERT ANNUNCIATOR, THE DECISION WAS MADE TO MANUALLY TRIP THE REACTOR. THE UNIT WAS STABILIZED IN MODE 3, HOT STANDBY, AND RETURNED TO MODE 1, POWER OPERATION, ON JULY 8, 1987. THIS INCIDENT IS CLASSIFIED AS EVENT CAUSE CODE B, DESIGN, MANUFACTURING, CONSTRUCTION/INSTALLATION DEFICIENCY. A PIECE OF UNINSULATED SCRAP WIRE WAS DISCOVERED BETWEEN THE FREQUENCY TO VOLTAGE (F/V) CONVERTOR CHIP AND THE CIRCUIT BOARD ON WHICH IT WAS MOUNTED.

[48] CATAWBA 1 DOCKET 50-413 LER 87-028
REACTOR TRIP ON INADVERTENTLY SIMULATED TURBINE TRIP DUE TO FAILURE OF INTERMEDIATE STOP VALVE DURING TESTING.
EVENT DATE: 071187 REPORT DATE: 081087 NSSS: WE TYPE: PWR

(NSIC 205794) ON JULY 11, 1987, AT 0254:21:691 HOURS, WITH UNIT 1 AT APPROXIMATELY 94% POWER, THE REACTOR AUTOMATICALLY TRIPPED ON AN INADVERTENTLY SIMULATED TURBINE TRIP. DUKE POWER PERSONNEL WERE IN THE PROCESS OF CONDUCTING THE WEEKLY MAIN TURBINE VALVE MOVEMENT PROCEDURE WHEN THE DISK DUMP VALVE FOR INTERMEDIATE STOP VALVE NUMBER 6 IS SUSPECTED TO HAVE NOT RESEATED. THIS CAUSED EMERGENCY TRIP SYSTEM (ETS) HEADER PRESSURE TO DECREASE (INDICATING TURBINE TRIP), WHICH GENERATED AN AUTOMATIC REACTOR TRIP SIGNAL FROM THE SOLID STATE PROTECTION SYSTEM. THE REACTOR TRIPPED, AND THE TURBINE AUTOMATICALLY TRIPPED IMMEDIATELY FOLLOWING THE REACTOR TRIP. THIS INCIDENT HAS BEEN CLASSIFIED EVENT CAUSE CODE X, OTHER. THE FAILURE OF THE DISK DUMP VALVE FOR INTERMEDIATE STOP VALVE NO. 6 CAUSED A FALSE INDICATION OF TURBINE TRIP TO BE GENERATED, WHICH RESULTED IN THE REACTOR TRIP. DUKE POWER HAS INSTALLED A TEMPORARY MODIFICATION TO THE MAIN TURBINE VALVES TO ALLOW TESTING TO BE PERFORMED WITHOUT UNSEATING THE DISK DUMP VALVES. DURING THE NEXT REFUELING OUTAGE FOR EACH UNIT, ALL DISK DUMP VALVES WILL BE INSPECTED AND MODIFIED IF NECESSARY. THE HEALTH AND SAFETY OF THE PUBLIC WERE UNAFFECTED BY THIS INCIDENT.

[49] CATAWBA 1 DOCKET 50-413 LER 87-030
INADEQUATE RETEST PERFORMED RESULTING IN A TECHNICAL SPECIFICATION VIOLATION DUE TO INSUFFICIENT RETEST POLICY.
EVENT DATE: 072687 REPORT DATE: 082687 NSSS: WE TYPE: PWR

(NSIC 206086) ON JULY 26, 1987, AT 1630 HOURS, THE SAFETY INJECTION PUMPS TO COLD LEG ACCUMULATOR FILL VALVE WAS DECLARED OPERABLE FOLLOWING MAINTENANCE ACTIVITY WITHOUT THE PROPER PRETEST HAVING BEEN PERFORMED. THE VALVE HAD BEEN DECLARED INOPERABLE BECAUSE IT WOULD NOT CLOSE FROM THE CONTROL ROOM. DUKE POWER TECHNICIANS INCREASED THE TORQUE SETTING ON THE VALVE ACTUATOR AND LATER REPLACED THE TORQUE SWITCH SETTING IT TO THE MAXIMUM VALUE. FOLLOWING THE MAINTENANCE ACTIVITY THE VALVE WAS STROKE TIME TESTED, AND IT WAS DECLARED OPERABLE. THE FOLLOWING DAY, IT WAS DETERMINED THAT A TYPE C LEAK RATE TEST WAS REQUIRED FOLLOWING REPLACEMENT OF THE TORQUE SWITCH. THE UNIT WAS AT 100% POWER AT THE TIME OF THIS INCIDENT. THIS INCIDENT HAS BEEN CLASSIFIED EVENT CAUSE CODE E, MANAGEMENT/QUALITY ASSURANCE DEFICIENCY. PRIOR TO THIS INCIDENT, DUKE POWER DID NOT HAVE A POLICY FOR PRETEST REQUIREMENTS FOLLOWING REPLACEMENT OF TORQUE SWITCHES. THE PREVIOUS POLICY ONLY ADDRESSED CHANGES TO THE TORQUE SETTING. FOLLOWING THE DETERMINATION THAT A TYPE C LEAK RATE TEST WAS REQUIRED, THE VALVE WAS DECLARED INOPERABLE, AND THE APPROPRIATE LEAK RATE TEST WAS PERFORMED. THE HEALTH AND SAFETY OF THE PUBLIC WERE UNAFFECTED BY THIS INCIDENT.

[50]   CATAWBA 1   DOCKET 50-413   LER 87-031
VENT RADIATION MONITOR UNKNOWINGLY INOPERABLE DUE TO INADEQUATE LOGGING POLICY.
EVENT DATE: 072887   REPORT DATE: 082787   NSSS: WE   TYPE: PWR

(NSIC 206087) ON JULY 29, 1987, AT APPROXIMATELY 1341 HOURS, THE UNIT 1 VENT GAS RADIATION MONITOR WAS DISCOVERED TO BE INOPERABLE DUE TO A NON-CONSERVATIVE HIGH RADIATION ALARM SETPOINT. THE SETPOINT HAD BEEN CHANGED THREE DAYS EARLIER WHEN THE RADIATION MONITOR (EMF) HAD BEEN DECLARED INOPERABLE DUE TO A MALFUNCTIONING GAS SAMPLE PUMP. WHEN THE EMF WAS DECLARED OPERABLE ON JULY 27, 1987, THE SETPOINT WAS NOT RETURNED TO ITS NORMAL VALUE. CONSEQUENTLY, A TECHNICAL SPECIFICATION VIOLATION OCCURRED AT 0032 HOURS ON JULY 28 WHEN THE REQUIRED GRAB SAMPLES WERE NOT TAKEN WHILE THE EMF WAS UNKNOWINGLY INOPERABLE. THE UNIT WAS AT 100% POWER DURING THIS INCIDENT. THIS INCIDENT IS CLASSIFIED AS EVENT CAUSE CODE E, MANAGEMENT/QUALITY ASSURANCE DEFICIENCY. THE SHIFT SUPERVISOR AND UNIT SUPERVISOR ON DUTY WHEN THE SETPOINT WAS FIRST CHANGED DID NOT DOCUMENT THE CHANGE IN THE TECHNICAL SPECIFICATION ACTION ITEMS LOG (TSAIL) REMARKS COLUMN, BUT THEY DID DOCUMENT IT IN THE CONTROL ROOM, UNIT SUPERVISOR, AND EMF SETPOINT LOG BOOKS. THE SHIFT SUPERVISOR ON DUTY WHEN THE EMF WAS DECLARED OPERABLE CHECKED ONLY THE TSAIL, AND HE DID NOT REALIZE THAT THE SETPOINT HAD BEEN CHANGED. THE SETPOINT WAS RESTORED TO ITS ORIGINAL VALUE UPON DISCOVERY TWO DAYS LATER. THE INCIDENT WILL BE DISCUSSED WITH ALL SHIFT SUPERVISORS AND POLICY WILL BE CLARIFIED. THE HEALTH AND SAFETY OF THE PUBLIC WERE UNAFFECTED BY THIS EVENT.

[51]   CATAWBA 1   DOCKET 50-413   LER 87-032
BOTH TRAINS OF VITAL BATTERIES INOPERABLE DUE TO MISSED RETEST REQUIREMENTS.
EVENT DATE: 072987   REPORT DATE: 082887   NSSS: WE   TYPE: PWR

(NSIC 206088) ON JULY 29, 1987, AT 1220 HOURS, A DUKE POWER ENGINEER DETERMINED THAT BATTERY BANKS 1EBC AND 1EBD IN THE 125 VDC VITAL INSTRUMENTATION AND CONTROL POWER (EPL) SYSTEM WERE INOPERABLE. WHILE REVIEWING COMPLETED WORK REQUESTS ORIGINATED TO CLEAN OR REPLACE THE BATTERIES' INTERCELL CONNECTORS (STRAPS), THE ENGINEER DISCOVERED THAT 1EBC AND 1EBD HAD BEEN DECLARED OPERABLE WITHOUT THE REQUIRED INTERCELL RESISTANCE READINGS BEING TAKEN ON JULY 21, 1987, AT 2245 HOURS, AND ON JULY 24, 1987, AT 1700 HOURS, RESPECTIVELY. THIS RESULTED IN AN INOPERABLE BATTERY BANK IN BOTH TRAIN A AND TRAIN B AND ENTRY INTO TECHNICAL SPECIFICATION 3.0.3. THE UNIT WAS AT 100% POWER AT THE TIME OF THIS INCIDENT. THIS INCIDENT HAS BEEN CLASSIFIED AS EVENT CAUSE CODE E, MANAGEMENT/QUALITY ASSURANCE DEFICIENCY. SUFFICIENT INFORMATION AND GUIDELINES DID NOT EXIST IN PLANNING FOR DETERMINING PRETEST REQUIREMENTS OR TO DETERMINE THE PROPER PROCEDURES NECESSARY TO SATISFY PRETEST REQUIREMENTS. BATTERY BANKS 1EBC AND 1EBD WERE IMMEDIATELY DECLARED INOPERABLE AND THE INTERCELL RESISTANCE READINGS WERE SATISFACTORILY TAKEN. A PROCEDURE CHANGE WAS INCORPORATED IN THE BATTERY PREVENTATIVE MAINTENANCE PROCEDURE ADDING A STEP TO PERFORM THE INTERCELL RESISTANCE MEASUREMENTS WHENEVER ANY STRAPS ARE REMOVED. THE HEALTH AND SAFETY OF THE PUBLIC WERE UNAFFECTED BY THIS EVENT.

[52]   CATAWBA 2   DOCKET 50-414   LER 87-014
UNEXPECTED STEAM GENERATOR LOW LOW LEVEL SIGNAL ACTUATED DURING MAIN STEAM VALVE TESTING DUE TO UNKNOWN CAUSE.
EVENT DATE: 032987   REPORT DATE: 042887   NSSS: WE   TYPE: PWR
VENDOR: DRAGON VALVE, INC.

(NSIC 206586) ON MARCH 29, 1987, AT 0049:58 HOURS, AN UNEXPECTED STEAM GENERATOR (S/G) 2A LOW LOW LEVEL SIGNAL OCCURRED DURING STROKE TESTING OF S/G 2A MAIN STEAM ISOLATION VALVE. THE LOW LOW LEVEL SIGNAL OCCURRED IMMEDIATELY AFTER THE VALVE WAS OPENED BY THE CONTROL ROOM OPERATOR. PLANT RESPONSE WAS MINIMAL AS THE UNIT WAS IN MODE 5, COLD SHUTDOWN. CONTROL ROOM OPERATORS SUBSEQUENTLY REALIGNED AFFECTED BLOWDOWN AND AUXILIARY FEEDWATER VALVES. AN UNSUCCESSFUL ATTEMPT WAS MADE TO RECREATE THE INCIDENT ON THE NEXT SHIFT. A MALFUNCTIONING EXCESS FLOW

CHECK VALVE WAS DISCOVERED WHICH WOULD HAVE PRODUCED A FALSE LOW LOW LEVEL ON ONE CHANNEL OF NARROW RANGE LEVEL INSTRUMENTATION. THE CHECK VALVE WAS SUBSEQUENTLY REPLACED. THIS INCIDENT IS CLASSIFIED AS EVENT CAUSE CODE X, OTHER, DUE TO THE UNKNOWN CAUSE OF THE SECOND CHANNEL OF NARROW RANGE LEVEL BEING ALARMED. A CONTRIBUTING EVENT CAUSE CODE X, OTHER, IS ALSO ASSIGNED DUE TO THE MALFUNCTION OF THE EXCESS FLOW CHECK VALVE PLACING ONE CHANNEL IN A CONDITION TO PROVIDE A FALSE LEVEL SIGNAL. PLANT CONDITIONS PRESENT DURING THE INCIDENT ARE NOT ENCOUNTERED DURING POWER OPERATION, AND SUCH INCIDENT WOULD NOT BE EXPECTED. THE HEALTH AND SAFETY OF THE PUBLIC WERE UNAFFECTED BY THIS INCIDENT.

[53] CATAWBA 2 DOCKET 50-414 LER 87-015 REV 02
UPDATE ON CONTAINMENT AIR RETURN ISOLATION DAMPERS ACTUATED DUE TO DEFECTIVE PROCEDURE.
EVENT DATE: 040687 REPORT DATE: 101987 NSSS: WE TYPE: PWR

(NSIC 206546) ON APRIL 6, 1987, AT 1439.41 HOURS, THE CONTAINMENT AIR RETURN FAN B ISOLATION DAMPER UNEXPECTEDLY OPENED, WHICH CONSTITUTED AN ENGINEERED SAFEGUARD FEATURE ACTUATION. DUKE POWER STATION PERSONNEL WERE IN THE PROCESS OF CONDUCTING THE CONTAINMENT AIR RETURN FAN 2B AND HYDROGEN SKIMMER FAN 2B PERFORMANCE TEST PROCEDURE. THE UNIT WAS IN MODE 3, HOT STANDBY, AT THE TIME OF THIS INCIDENT. THIS INCIDENT IS CLASSIFIED AS EVENT CAUSE CODE D, DEFECTIVE PROCEDURE. THE CONTAINMENT AIR RETURN FAN 2B AND HYDROGEN SKIMMER FAN 2B PERFORMANCE TEST PROCEDURE DID NOT CONTAIN INSTRUCTIONS TO RESET THE ACTUATION SIGNAL FOR THE ISOLATION DAMPER. THIS ALLOWED THE ISOLATION DAMPER TO OPEN WHEN THE REQUIRED LOGIC WAS COMPLETED DURING THE PERFORMANCE OF ANOTHER SECTION OF THE PROCEDURE. CONTROL ROOM PERSONNEL CLOSED THE ISOLATION DAMPER. A PROCEDURE CHANGE WAS INCORPORATED INTO THE CONTAINMENT AIR RETURN FANS AND HYDROGEN SKIMMER FANS PERFORMANCE TEST PROCEDURES FOR BOTH UNITS TO RESET THE ACTUATION SIGNAL FOR THE ISOLATION DAMPERS. THE PLANT WAS ALWAYS WITHIN ITS DESIGN BASIS AND THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED BY THIS INCIDENT.

[54] CATAWBA 2 DOCKET 50-414 LER 87-021
REACTOR TRIP RESULTS FROM CONDENSATE TRANSIENT DUE TO UNKNOWN CAUSE.
EVENT DATE: 072787 REPORT DATE: 082787 NSSS: WE TYPE: PWR

(NSIC 206220) ON JULY 27, 1987, UNIT 2 WAS OPERATING AT 90% POWER WHEN A CONDENSATE TRANSIENT OCCURRED WHICH RESULTED IN A REACTOR TRIP. DURING THE TRANSIENT, CONDENSATE BOOSTER PUMP SUCTION PRESSURE DECREASED SIGNIFICANTLY CAUSING THE THIRD HOTWELL PUMP TO START AUTOMATICALLY. MAIN FEEDWATER PUMP SUCTION PRESSURE DECREASED, AND THE THIRD CONDENSATE BOOSTER PUMP STARTED AUTOMATICALLY. ALL CONDENSATE BOOSTER PUMPS TRIPPED. SUBSEQUENTLY, BOTH MAIN FEEDWATER PUMPS TRIPPED CAUSING A TURBINE TRIP FOLLOWED BY THE REACTOR TRIP. THE UNIT WAS STABILIZED IN MODE 3, HOT STANDBY, FOLLOWING THE REACTOR TRIP. CONDENSATE FLOW WAS REESTABLISHED BY OPENING THE CONDENSATE POLISHER BYPASS VALVES. THIS INCIDENT IS CLASSIFIED AS EVENT CAUSE CODE X, OTHER. THE INITIATING EVENT FOR THE CONDENSATE TRANSIENT AND LOSS OF CONDENSATE FLOW IS UNKNOWN. THE UNAVAILABILITY OF SECONDARY SYSTEM TRANSIENT MONITOR DATA HAS PREVENTED DUKE POWER FROM DETERMINING THE INITIATING CIRCUMSTANCES FOR THE INCIDENT. AT THE PRESENT TIME, THE EVENT IS UNDERGOING CONTINUED INVESTIGATION, AND DUKE POWER WILL REPORT ANY ADDITIONAL FINDING UPON COMPLETION OF THE INVESTIGATION IF APPROPRIATE. THE HEALTH AND SAFETY OF THE PUBLIC WERE UNAFFECTED BY THIS EVENT.

[55] CATAWBA 2 DOCKET 50-414 LER 87-022
MAIN FEEDWATER ISOLATION AND AUXILIARY FEEDWATER AUTO START DUE TO DEBRIS IN A FEEDWATER CONTROL VALVE POSITIONER.
EVENT DATE: 072887 REPORT DATE: 082887 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: CATAWBA 1 (PWR)

VENDOR: FAIRCHILD IND. (PRODUCTS DIV)
FISHER CONTROLS CO.
MOORE PRODUCTS COMPANY

(NSIC 206089) ON JULY 28, 1987, AT APPROXIMATELY 0059 HOURS, WITH THE UNIT IN MODE 3, HOT STANDBY, AN AUTOMATIC MAIN FEEDWATER (CF) ISOLATION OCCURRED DUE TO A HIGH HIGH STEAM GENERATOR (S/G) WATER LEVEL. THE CF ISOLATION CAUSED THE CF PUMP TURBINE (CFPT) TO TRIP WHICH RESULTED IN THE MOTOR DRIVEN AUXILIARY FEEDWATER (CA) PUMPS STARTING AUTOMATICALLY TO SUPPLY FEEDWATER TO THE S/GS. AT THE TIME OF THE INCIDENT DUKE POWER PERSONNEL WERE IN THE PROCESS OF CHECKING THE CALIBRATION OF THE ELECTRICAL TO PNEUMATIC (E/P) TRANSDUCER FOR 2CF37, S/G 2B CF CONTROL VALVE. WHEN PERSONNEL CLOSED THE E/P TRANSDUCER'S OUTPUT ISOLATION VALVE, 2CF37 OPENED CAUSING OVERFEED AND THE SUBSEQUENT S/G 2B HIGH HIGH LEVEL. CONTROL ROOM PERSONNEL ISOLATED 2CF37, RESTORED PROPER S/G LEVELS, REALIGNED THE CF SYSTEM, RESTARTED CFPT 2A AND SECURED THE MOTOR DRIVEN CA PUMPS. THIS INCIDENT IS CLASSIFIED AS EVENT CAUSE CODE E, MANAGEMENT/QUALITY ASSURANCE DEFICIENCY. A PREVENTIVE MAINTENANCE PROGRAM HAD NOT BEEN ESTABLISHED TO PREVENT CONTAMINANTS IN THE INSTRUMENT AIR (VI) SYSTEM FROM BUILDING UP AND CAUSING MALFUNCTIONS OF THE CF CONTROL VALVES PNEUMATIC POSITIONING SYSTEM. DEBRIS WAS DISCOVERED TO HAVE BUILT UP ON A PILOT VALVE SEAT INSIDE THE POSITIONER FOR 2CF37 WHICH CAUSED THE VALVE TO OPEN DURING THE CALIBRATION CHECK.

[56] CATAWBA 2 DOCKET 50-414 LER 87-024
MAIN FEEDWATER ISOLATION AND AUXILIARY FEEDWATER PUMP AUTO-START DUE TO A STEAM GENERATOR OVERFILL CAUSED BY A MANAGEMENT DEFICIENCY.
EVENT DATE: 080787 REPORT DATE: 090487 NSSS: WE TYPE: PWR

(NSIC 206090) ON AUGUST 7, 1987, AT APPROXIMATELY 1626 HOURS, WITH THE UNIT IN MODE 3, HOT STANDBY, AN AUTOMATIC MAIN FEEDWATER (CF) ISOLATION OCCURRED DUE TO A HIGH HIGH STEAM GENERATOR (S/G) WATER LEVEL. THE CF ISOLATION CAUSED THE CF PUMP TURBINE (CFPT) TO TRIP WHICH RESULTED IN A MOTOR DRIVEN AUXILIARY FEEDWATER (CA) PUMP STARTING AUTOMATICALLY TO SUPPLY FEEDWATER TO THE S/GS. CONTROL ROOM PERSONNEL HAD INTENTIONALLY RAISED S/G WATER LEVELS TO AID IN REDUCING REACTOR COOLANT (NC) SYSTEM TEMPERATURE TO ACHIEVE MODE 4, HOT SHUTDOWN, BY 1649 HOURS, AS REQUIRED BY TECHNICAL SPECIFICATIONS DUE TO AN INOPERABLE TURBINE DRIVEN (T/D) CA PUMP. THE UNIT ENTERED MODE 4 AT 1642 HOURS, SEVEN MINUTES PRIOR TO EXCEEDING THE TECHNICAL SPECIFICATION TIME LIMIT. S/G LEVELS WERE RESTORED TO NORMAL AND THE CA SYSTEM RETURNED TO STANDBY READINESS. THIS INCIDENT IS ATTRIBUTED TO A MANAGEMENT DEFICIENCY. STATION SUPERVISORY PERSONNEL ON SHIFT DELAYED INITIATION OF UNIT COOL DOWN. ALSO, THE INITIAL RATE OF UNIT COOL DOWN TO MODE 4 WAS LIMITED DUE TO THE BELIEF THAT THE CONDITION REQUIRING UNIT COOL DOWN WOULD BE CORRECTED. TO PREVENT FUTURE OCCURRENCES, DUKE POWER PERSONNEL WILL DISCUSS DEVELOPING CONSERVATIVE COOLDOWN TIME ESTIMATES WHICH CONSIDER EXISTING EQUIPMENT LIMITATIONS. THE HEALTH AND SAFETY OF THE PUBLIC WERE UNAFFECTED BY THIS INCIDENT.

[57] CLINTON 1 DOCKET 50-461 LER 87-035
LAND USE CENSUS PERFORMED IN 1986 FAILS TO MEET REQUIREMENTS DUE TO DEFICIENT SURVEILLANCE PROCEDURE.
EVENT DATE: 072287 REPORT DATE: 081787 NSSS: GE TYPE: BWR

(NSIC 205890) CAUSE - DEFICIENT SURVEILLANCE PROCEDURES. ON 7/22/87, AT 1150 HOURS, WITH THE PLANT IN MODE 1 (POWER OPERATION) AT APPROXIMATELY 50% REACTOR POWER, WHILE PREPARING TO PERFORM THE ANNUAL ENVIRONMENTAL LAND USE CENSUS AS REQUIRED BY TECHNICAL SPECIFICATION SURVEILLANCE 4.12.2, IT WAS DISCOVERED THAT THE PREVIOUS LAND USE CENSUS PERFORMED IN JULY 1986, DID NOT MEET THE REQUIREMENTS OF TECHNICAL SPECIFICATION 3.12.2. THE LAND USE CENSUS COMPLETED IN JULY 1986, DID NOT PERFORM A GARDEN CENSUS WITHIN THE REQUIRED THREE (3) MILE RADIUS OF THE CLINTON POWER STATION. TECHNICAL SPECIFICATION 3.12.2 REQUIRES THAT

ALL GARDENS WITHIN A THREE MILE RADIUS BE IDENTIFIED DURING THE ANNUAL LAND USE CENSUS FOR PLANTS WITH (REGULATORY GUIDE 1.111) MIXED-MODE RELEASES. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO UTILITY PERSONNEL ERROR DUE TO A DEFICIENT SURVEILLANCE PROCEDURE. CORRECTIVE ACTION INCLUDED PERFORMANCE OF THE LAND USE CENSUS TO TECHNICAL SPECIFICATION 3.12.2 REQUIREMENTS. AN ASSESSMENT OF THE SAFETY CONSEQUENCES AND IMPLICATIONS OF THIS EVENT DETERMINED THAT FAILURE TO IDENTIFY LOCATIONS WHERE BROADLEAF VEGETATION IS GROWN DOES NOT ADVERSELY IMPACT THE SAFETY OF OPERATIONS OF THE CLINTON POWER STATION. THE EVENT IS REPORTABLE UNDER THE PROVISIONS OF 10CFR50.73(A)(2)(I)(B) DUE TO AN OPERATION OR CONDITION PROHIBITED BY THE PLANT'S TECH SPECS.

[58] CLINTON 1 DOCKET 50-461 LER 87-049
VIOLATION OF THE PLANT'S TECHNICAL SPECIFICATIONS DUE TO UTILITY PERSONNEL ERROR RESULTING FROM EXCEEDING THE DAILY SURVEILLANCE INTERVAL.
EVENT DATE: 081987 REPORT DATE: 091087 NSSS: GE TYPE: BWR

(NSIC 206191) ON AUGUST 19, 1987, AT 0300 HOURS, WITH THE PLANT IN MODE 1 (POWER OPERATION), AT APPROXIMATELY 30% REACTOR POWER, A CONTROL ROOM OPERATOR IDENTIFIED THAT TECHNICAL SPECIFICATIONS WERE VIOLATED IN THAT TWO CHANNEL TESTS WERE NOT PERFORMED WITHIN THE REQUIRED TIME INTERVAL OF 24 HOURS. THE OPERATOR DETERMINED DURING PERFORMANCE OF THE SHIFT CHECKOFF LOG THAT SURVEILLANCE 9031.11, AVERAGE POWER RANGE MONITORS FLOW BIASED POWER - PERCENT FLOW CHANNEL CHECK HAD NOT BEEN PERFORMED IN THE PREVIOUS 34 HOURS AND 33 MINUTES, AND SURVEILLANCE 9030.01C021, ROD PATTERN CONTROLLER LOW POWER SETPOINT C11-N654A(B) CHANNEL FUNCTIONAL CHECKLIST HAD NOT BEEN PERFORMED IN THE PREVIOUS 35 HOURS AND 39 MINUTES. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO UTILITY PERSONNEL ERROR AS A RESULT OF CONFUSION DURING THE TRANSITION FROM AN 8 HOUR SHIFT CHECKOFF LOG FREQUENCY TO A 12 HOUR SHIFT CHECKOFF LOG FREQUENCY. THE TWO SURVEILLANCES WERE IMMEDIATELY PERFORMED SATISFACTORILY. AN ASSESSMENT OF THE SAFETY IMPLICATION AND SIGNIFICANCE DETERMINED THAT THE EVENT WAS NOT SIGNIFICANT TO THE SAFETY OF OPERATIONS OF THE CLINTON POWER STATION. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(I)(B) DUE TO AN OPERATION PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATIONS.

[59] CLINTON 1 DOCKET 50-461 LER 87-050
AUTOMATIC ACTUATION OF THE REACTOR PROTECTION SYSTEM DUE TO UTILITY NON-LICENSED OPERATOR ERROR RESULTING FROM A DEFICIENT PROCEDURE.
EVENT DATE: 082587 REPORT DATE: 091887 NSSS: GE TYPE: BWR

(NSIC 206192) ON AUGUST 25, 1987 AT 0736 HOURS WITH THE PLANT IN MODE 1 (POWER OPERATION) AT APPROXIMATELY 20% REACTOR POWER AND NORMAL OPERATING TEMPERATURE AND PRESSURE, THE TURBINE GENERATOR TRIPPED ON LOW CONDENSER VACUUM RESULTING IN A REACTOR SCRAM. OPERATIONS PERSONNEL WERE IN THE PROCESS OF SHIFTING FROM THE "B" TRAIN STEAM JET AIR EJECTORS (SJAE) TO THE "A" TRAIN SJAE. THIS WAS THE FIRST ATTEMPT TO CHANGE TRAINS DURING OPERATION. WITH BOTH TRAINS IN PARALLEL SERVICE, THE OPERATORS CLOSED THE "B" SJAE SUCTION VALVE AND WERE SECURING DRIVING STEAM WHEN CONDENSER VACUUM BEGAN DECREASING. A POSSIBLE BACKFLOW PATH WAS BELIEVED TO BE CAUSING THE VACUUM LOSS. THE "B" TRAIN RECOMBINER MANUAL OUTLET VALVE WAS THEN CLOSED BUT VACUUM CONTINUED TO DECREASE. THE DECISION WAS MADE TO RESTORE THE "B" SJAE TO SERVICE; HOWEVER, THE TURBINE GENERATOR TRIPPED. REACTOR SCRAM PROCEDURES WERE IMPLEMENTED. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO UTILITY NON-LICENSED OPERATOR ERROR RESULTING FROM A DEFICIENT VALVE LINEUP PROCEDURE. THE PROCEDURE HAS BEEN REVISED. AN ASSESSMENT OF THE SAFETY IMPLICATION AND SIGNIFICANCE DETERMINED THAT THE EVENT WAS NOT SIGNIFICANT TO THE SAFETY OF OPERATIONS OF THE CLINTON POWER STATION.

[60] CLINTON 1 DOCKET 50-461 LER 87-051
VIOLATION OF THE PLANT'S TECHNICAL SPECIFICATIONS DUE TO UTILITY PERSONNEL ERROR RESULTING FROM A PROCEDURAL DEFICIENCY.
EVENT DATE: 082887 REPORT DATE: 091787 NSSS: GE TYPE: BWR

(NSIC 206193) ON AUGUST 28, 1987, AT APPROXIMATELY 0200 HOURS WITH THE PLANT IN MODE 1 (POWER OPERATION), AT APPROXIMATELY 10% POWER, IT WAS IDENTIFIED THAT THE ROD PATTERN CONTROL SYSTEM, ROD WITHDRAWAL LIMITER HIGH POWER SETPOINT CHANNEL FUNCTIONAL TEST WAS NOT PERFORMED WITHIN ONE HOUR PRIOR TO CONTROL ROD MOVEMENT IN MODE 1 AS REQUIRED BY TECHNICAL SPECIFICATIONS. IT WAS SUBSEQUENTLY DETERMINED THAT THE SAME EVENT HAD OCCURRED ON AUGUST 17, 1987 AND MAY HAVE OCCURRED ON JULY 18, 1987. THE CAUSE OF THE EVENT IS ATTRIBUTED TO UTILITY LICENSED OPERATOR ERROR RESULTING FROM A PROCEDURAL DEFICIENCY. THE CHANNEL FUNCTIONAL TEST WAS COMPLETED SATISFACTORILY AND THE APPROPRIATE PROCEDURE WAS REVISED. THE SAFETY IMPACT OF THE EVENT IS ASSESSED AS INSIGNIFICANT SINCE THE SURVEILLANCE TEST WAS SATISFACTORILY COMPLETED SUBSEQUENT TO THE EVENT. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(I)(B) DUE TO AN OPERATION PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATIONS.

[61] CLINTON 1 DOCKET 50-461 LER 87-052
REACTOR CORE ISOLATION COOLING ISOLATION RESULTING FROM FAILURE TO RECOGNIZE THE HIGH POTENTIAL FOR ACTUATION PRIOR TO TROUBLESHOOTING.
EVENT DATE: 090287 REPORT DATE: 092487 NSSS: GE TYPE: BWR

(NSIC 206465) ON 9/2/87, AT 1840 HOURS WITH THE PLANT IN MODE 1 (POWER OPERATION) AT NORMAL OPERATING TEMPERATURE AND PRESSURE AND 64% REACTOR POWER, THE REACTOR CORE ISOLATION COOLING (RCIC) STEAM INBOARD ISOLATION VALVE AUTOMATICALLY ISOLATED. THE ISOLATION OCCURRED DURING TROUBLESHOOTING TO ISOLATE ERRATIC DIVISION 2 RCIC STEAM FLOW INDICATION THAT WAS CAUSING SPURIOUS HIGH FLOW SPIKES. CONTROL AND INSTRUMENTATION (C&I) TECHNICIANS WERE THROTTLING THE TRANSMITTER ISOLATION VALVES AS A POSSIBLE SOLUTION, WHEN THE INDICATION INCREASED AND THE THREE SECOND DELAY FOR THE RCIC INBOARD ISOLATION FUNCTION AND RCIC TURBINE TRIP SIGNAL TIMED OUT. THE TURBINE WAS IN STANDBY AND NOT RUNNING. OPERATIONS IMPLEMENTED THE AUTOMATIC ISOLATION PROCEDURE FOR RCIC AND DECLARED RCIC INOPERABLE. THE CAUSE OF THE EVENT IS ATTRIBUTED TO UTILITY PERSONNEL ERROR RESULTING FROM A FAILURE TO RECOGNIZE THE HIGH POTENTIAL FOR ACTUATION PRIOR TO MANIPULATING THE INSTRUMENT ISOLATION VALVES. C&I TECHNICIANS AND OPERATIONS SUPERVISION HAVE BEEN TRAINED ON THE REQUIREMENT TO INCLUDE POTENTIAL ENGINEERED SAFETY FEATURE (ESF) ACTUATIONS IN THE IMPACT MATRIX OF THE WORK DOCUMENT. THE SAFETY SIGNIFICANCE OF THIS EVENT WAS ASSESSED AS INSIGNIFICANT SINCE THE HIGH PRESSURE CORE SPRAY SYSTEM WOULD FUNCTION AS AN ADEQUATE BACKUP SYSTEM.

[62] CLINTON 1 DOCKET 50-461 LER 87-053
VIOLATION OF THE PLANT'S TECHNICAL SPECIFICATIONS DUE TO INCOMPLETE SURVEILLANCE PERFORMANCE RESULTING IN FAILURE TO BYPASS VALVE THERMAL OVERLOAD PROTECTION.
EVENT DATE: 091287 REPORT DATE: 100187 NSSS: GE TYPE: BWR

(NSIC 206601) ON SEPTEMBER 12, 1987, AT 0545 HOURS, WITH THE PLANT IN MODE 1 (POWER OPERATION) AT NORMAL OPERATING TEMPERATURE AND PRESSURE AND APPROXIMATELY 73% REACTOR POWER, A CONTROL ROOM OPERATOR IDENTIFIED A TECHNICAL SPECIFICATION VIOLATION. THERMAL OVERLOAD PROTECTION FOR THE FIRE PROTECTION SYSTEM DIVISION 2 CONTAINMENT ISOLATION VALVES WAS NOT IN THE BYPASS (NORMAL) STATUS SINCE APPROXIMATELY 1930 HOURS ON SEPTEMBER 11, 1987, WHEN THE OVERLOAD PROTECTION WAS TAKEN OUT OF BYPASS FOR THE THREE MONTH VALVE OPERABILITY SURVEILLANCE OF THE FIRE PROTECTION DRYWELL INBOARD ISOLATION VALVE. TECHNICAL SPECIFICATION 3.8.4.2 PERMITS THERMAL OVERLOAD PROTECTION NOT TO BE BYPASSED FOR A MAXIMUM PERIOD OF EIGHT HOURS. THE TECHNICAL SPECIFICATION WAS VIOLATED FROM 0330 HOURS UNTIL 0545 HOURS ON SEPTEMBER 12, 1987. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO UTILITY LICENSED OPERATOR ERROR RESULTING FROM A CONFUSING PROCEDURE. THE OPERATOR

PLACED THE THERMAL OVERLOAD PROTECTION IN THE BYPASS STATUS IMMEDIATELY. THE PROCEDURE WILL BE REVISED TO PREVENT FUTURE CONFUSION. THE SAFETY SIGNIFICANCE OF THIS EVENT WAS ASSESSED AS INSIGNIFICANT SINCE THE DIVISION 1 FIRE PROTECTION SYSTEM ISOLATION VALVES WERE OPERABLE AND PROVIDE ADEQUATE REDUNDANCY.

[63] CONNECTICUT YANKEE DOCKET 50-213 LER 87-014
INADEQUATE CONTRACTOR TRAINING/AWARENESS RESULTS IN INOPERABLE FIRE BARRIERS.
EVENT DATE: 081187 REPORT DATE: 091087 NSSS: WE TYPE: PWR

(NSIC 206030) DURING THE PERIOD BETWEEN AUGUST 11, 1987 AND AUGUST 14, 1987, WITH THE PLANT SHUT DOWN IN MODE 5, THREE FIRE BARRIER BREACHES OCCURRED DURING OUTAGE CONSTRUCTION ACTIVITIES IN THE SWITCHGEAR ROOM. EACH INCIDENT RESULTED IN THE AFFECTED FIRE BARRIER BEING DECLARED INOPERABLE. UPON DISCOVERY, EACH BREACH WAS TEMPORARILY RESEALED AND THE CONTINUOUS FIRE WATCH WAS SPECIFICALLY ADVISED OF THE EXISTENCE OF THE BREACH(ES). DURING THIS PERIOD, A CONTINUOUS FIRE WATCH WAS STATIONED IN THE SWITCHGEAR ROOM BECAUSE THE HALON FIRE SUPPRESSION SYSTEM WAS OUT OF SERVICE. IN ADDITION, A 20 MINUTE FIRE PATROL WAS COVERING THE SWITCHGEAR ROOM AND ADJACENT FIRE ZONES. THE CAUSE OF THESE EVENTS WAS INADEQUATE CONTRACTOR AWARENESS OF PLANT FIRE BARRIER REQUIREMENTS AND INSUFFICIENT REVIEW BY THE JOB SUPERVISOR OF THE POTENTIAL EFFECTS OF THE WORK ACTIVITIES ON FIRE BARRIERS AND PENETRATION SEALS. LONG-TERM CORRECTIVE ACTIONS INVOLVE PERMANENTLY RESEALING THE OPENINGS AND THE ENHANCEMENT OF EMPLOYEE/CONTRACTOR TRAINING PROGRAMS TO MORE CLEARLY INDICATE THE REQUIREMENTS FOR MAINTAINING FIRE BARRIER INTEGRITY. THIS EVENT IS REPORTABLE PER 10CFR50.73(A)(2)(I) SINCE IT INVOLVES A CONDITION PROHIBITED BY TECHNICAL SPECIFICATIONS.

[64] CONNECTICUT YANKEE DOCKET 50-213 LER 87-015
PERSONNEL ERROR CAUSES TEMPORARY LOSS OF SPENT FUEL PIT COOLING PUMPS POWER.
EVENT DATE: 081487 REPORT DATE: 091187 NSSS: WE TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 206031) ON AUGUST 14, 1985 AT 1415 HOURS WITH THE PLANT SHUTDOWN IN MODE 6, A MAINTENANCE ELECTRICIAN INADVERTENTLY CAUSED A PARTIAL LOSS OF NON-VITAL 480 VOLT POWER. THIS EVENT RESULTED IN A COMPLETE LOSS OF NORMAL POWER TO THE SPENT FUEL PIT COOLING SYSTEM AND THE SPENT FUEL BUILDING CRANE WITH A FUEL ASSEMBLY SUSPENDED FROM THE CRANE. THE EVENT ALSO RESULTED IN OVERHEATING AND FAILURE OF TWO UNDERVOLTAGE RELAYS. OPERATORS RECOGNIZED THE LOSS OF SPENT FUEL PIT COOLING, AND THE FACT THAT ADEQUATE TIME WAS AVAILABLE TO RESTORE POWER TO THE SYSTEM USING A PROPERLY REVIEWED AND APPROVED JUMPER CONTROL SHEET. THE JUMPER CONTROL SHEET WAS APPROVED BY THE PLANT OPERATIONS REVIEW COMMITTEE AND SPENT FUEL PIT COOLING WAS RESTORED AT 1545 HOURS. THE LOSS OF COOLING EXISTED FOR 80 MINUTES AND RESULTED IN AN INCREASE IN SPENT FUEL PIT TEMPERATURE OF 6 DEGREES FAHRENHEIT. NO TEMPERATURE LIMITS WERE EXCEEDED. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR. CORRECTIVE MEASURES INCLUDE BETTER TRAINING, AND INCREASED SUPERVISORY INVOLVEMENT. PROBLEMS EXPERIENCED WITH THE GENERAL ELECTRIC HEA RELAYS INVOLVED IN THIS EVENT ARE CURRENTLY UNDER EVALUATION AND THE RESULTS WILL BE FORWARDED IN A SUPPLEMENTAL REPORT. THIS EVENT HAS BEEN CLASSIFIED AS REPORTABLE UNDER 10CFR50.73(A)(2)(V)(B) WITH THE ASSISTANCE OF NRC STAFF.

[65] COOK 1 DOCKET 50-315 LER 87-012 REV 01
UPDATE ON TYPE B & C CONTAINMENT LEAK RATE TEST FAILURE DUE TO A RIPPED DIAPHRAGM AND AN UNDETERMINED CAUSE.
EVENT DATE: 070287 REPORT DATE: 101687 NSSS: WE TYPE: PWR
VENDOR: ITT GRINNELL

(NSIC 206519) THIS IS A SUPPLEMENTAL REPORT SUBMITTED TO PROVIDE ADDITIONAL INFORMATION REGARDING THE SUBJECT VIOLATION REPORTED ON JULY 24, 1987. ON JULY 2, 1987, THE AS-FOUND RESULTS OF TYPE C TESTING PERFORMED ON TWO MANUAL CONTAINMENT

ISOLATION DIAPHRAGM VALVES BETWEEN THE REACTOR COOLANT DRAIN TANK AND REFUELING WATER PURIFICATION PUMPS REFLECTED AN EXCESSIVE LEAK RATE. THIS RESULTED IN THE ACCUMULATED LEAKAGE FOR ALL PENETRATIONS AND VALVES SUBJECT TO TYPE B&C TESTS EXCEEDING THE ALLOWED LIMIT OF 0.60 LA OF TECH SPEC 3.6.1.2B. THE CAUSE OF ONE OF THE VALVE FAILURES WAS DETERMINED TO BE A RIPPED DIAPHRAGM. NO DEFINITIVE CAUSE FOR THE OTHER VALVE FAILURE CAN BE DETERMINED. SUBSEQUENT REPAIRS TO BOTH VALVES CORRECTED THE LEAKAGE PROBLEM. THESE VALVES WILL BE TESTED ON AN INCREASED FREQUENCY IN ORDER TO MONITOR THEIR PERFORMANCE. THE RESULTS WILL BE USED TO DETERMINE FUTURE REPAIR/REPLACEMENT ACTIONS. THE AS-LEFT LEAKAGE RATE FOR THE VALVES WAS BELOW THE APPLICABLE ISI GUIDELINE LEAKAGE LIMITS AND THE COMBINED AS-LEFT LEAKAGE FOR ALL PENETRATIONS AND VALVES SUBJECT TO TYPE B AND C TESTING WAS CALCULATED TO BE 0.046 LA, WHICH IS WELL BELOW THE TECH SPEC LIMIT OF 0.60 LA.

[66]   COOK 1   DOCKET 50-315   LER 87-014
FAILURE TO COMPLY WITH TECHNICAL SPECIFICATION REQUIREMENTS DUE TO PROCEDURAL DEFICIENCIES.
EVENT DATE: 081287   REPORT DATE: 091187   NSSS: WE   TYPE: PWR
OTHER UNITS INVOLVED: COOK 2 (PWR)

(NSIC 206053) DURING A PLANT INITIATED SAFETY SYSTEM FUNCTIONAL INSPECTION (SSFI) OF THE AUXILIARY FEEDWATER SYSTEM (AFW), (AUGUST 1987) SEVERAL DISCREPANCIES WERE NOTED CONCERNING THE TIME RESPONSE TESTING OF CERTAIN AFW SYSTEM COMPONENTS. THE PROCEDURES DID NOT LITERALLY COMPLY WITH THE TECHNICAL SPECIFICATION (T/S) REQUIREMENTS FOR THE TESTING OF THE TURBINE AND MOTOR DRIVEN AUXILIARY FEED PUMPS AND THE EMERGENCY DIESEL GENERATORS. SUBSEQUENT TESTING AND REVIEWS OF PAST DATA FOR UNIT 2 PROVED THAT ALL COMPONENTS FUNCTIONED AS DESIGNED AND WITHIN THE OVERALL TIME RESPONSE REQUIRED BY T/S. UNIT 1 TESTING WILL BE COMPLETED PRIOR TO ENTERING MODE 3 (HOT STANDBY) AFTER THE CURRENT REFUELING OUTAGE. TO PREVENT RECCURRENCE, THE APPLICABLE SURVEILLANCE PROCEDURES WILL BE REVISED PRIOR TO THE NEXT SCHEDULED SURVEILLANCE. A REEVALUATION WILL BE PERFORMED OF THE EXTENSIVE PROCEDURAL REVIEW CONDUCTED IN 1986 TO IDENTIFY SUCH DISCREPANCIES. ANY NON-COMPLYING CONDITIONS WILL BE REPORTED AS REQUIRED BY 10 CFR 50.73. A REVIEW OF THE DATA INDICATES THAT A SIGNIFICANT SAFETY PROBLEM AS DEFINED IN 10 CFR 50.59 DID NOT EXIST. A SUPPLEMENTAL LER WILL BE SUBMITTED IF THE UNIT 1 DATA INDICATES THAT A SAFETY PROBLEM OCCURRED; HOWEVER, THE RESULTS ARE EXPECTED TO BE COMPARABLE.

[67]   COOK 1   DOCKET 50-315   LER 87-015
REFUELING MANIPULATOR CRANE LIMITING CONDITIONS FOR OPERATION NOT VERIFIED DUE TO USE OF AN INADEQUATELY CALIBRATED INSTRUMENT.
EVENT DATE: 081387   REPORT DATE: 091187   NSSS: WE   TYPE: PWR
OTHER UNITS INVOLVED: COOK 2 (PWR)
VENDOR: DILLON, W.C. & CO., INC.

(NSIC 206054) ON AUGUST 9, 1987, DURING FUEL MOVEMENT IN UNIT ONE, A MALFUNCTION OF THE REFUELING MANIPULATOR CRANE LOAD CELL LED TO THE DISCOVERY THAT THE LOAD CELL HAD NOT BEEN ADEQUATELY CALIBRATED. FURTHER INVESTIGATION DETERMINED THAT NEITHER THE UNIT ONE OR UNIT TWO LOAD CELL HAD BEEN ADEQUATELY CALIBRATED SINCE THE INITIAL CALIBRATION. THIS EVENT WAS DETERMINED TO BE REPORTABLE ON AUGUST 13, 1987. TECHNICAL SPECIFICATION 3.9.6 REQUIRES A MINIMUM CRANE CAPACITY OF 3250 POUNDS AND AN OVERLOAD CUTOFF LIMIT OF 2850 POUNDS OR LESS. SINCE THE INSTRUMENTS USED TO PROVE OPERABILITY WERE NOT ADEQUATELY CALIBRATED, COMPLIANCE WITH THE LIMITING CONDITIONS FOR OPERATION WAS NOT ADEQUATELY DEMONSTRATED. THE LOAD CELLS WERE NOT ADEQUATELY CALIBRATED DUE TO NOT PERFORMING WHAT IS NOW CONSIDERED TO BE AN ACCEPTABLE CALIBRATION. FUEL MOVEMENT WAS RE-STARTED AFTER THE UNIT TWO LOAD CELL WAS PROPERLY CALIBRATED AND INSTALLED IN UNIT ONE FOLLOWING A SATISFACTORY MANIPULATOR CRANE SURVEILLANCE TEST. PROCEDURE CHANGES WILL BE MADE TO REQUIRE AN ACCEPTABLE CALIBRATION OF THE LOAD CELLS PRIOR TO EACH

REFUELING. PROCEDURE CHANGES WILL ALSO BE MADE TO ESTABLISH CONSERVATIVE SURVEILLANCE ACCEPTANCE CRITERIA TO ALLOW FOR INSTRUMENT TOLERANCES.

[68] COOK 1 DOCKET 50-315 LER 87-016
ECCS FLOW IMBALANCE CAUSED BY NORMAL SYSTEM FLUCTUATIONS.
EVENT DATE: 081587 REPORT DATE: 091487 NSSS: WE TYPE: PWR

(NSIC 206055) ON AUGUST 15, 1987, AT 1430 HOURS, WITH UNIT 1 IN MODE 6 (REFUELING) AS- FOUND DATA OBTAINED WHILE PERFORMING ROUTINE SURVEILLANCE ON THE BORIC ACID INJECTION SYSTEM, REFLECTED AN UNACCEPTABLE FLOW DISTRIBUTION AS COMPARED TO TECHNICAL SPECIFICATION (T/S) 4.5.2H. NO CAUSE OTHER THAN NORMAL SYSTEM FLUCTUATION, COMBINED WITH STANDARD INSTRUMENTATION/MEASUREMENT ERROR COULD BE IDENTIFIED AS THE REASON THE TECHNICAL SPECIFICATION LIMITS COULD NOT BE MET. THE SYSTEM WAS SUBSEQUENTLY BALANCED ON SEPTEMBER 3, 1987, IN ACCORDANCE WITH T/S 4.5.2H. THE ISSUE WILL BE INTEGRATED INTO OUR TECHNICAL SPECIFICATION UPGRADE PROGRAM WHICH WILL ASSURE THAT IT IS APPROPRIATELY CONSIDERED FOR A PROPOSED AMENDMENT TO THE TECHNICAL SPECIFICATIONS. AN ENGINEERING ANALYSIS INDICATES THAT THE BORIC ACID INJECTION SYSTEM WOULD HAVE FUNCTIONED AS DESIGNED DURING AN ACCIDENT. ADEQUATE COOLING WAS AVAILABLE TO COOL THE CORE. CAVITATION (AND THE ATTENDANT LOSS OF EFFICIENCY) WHICH WOULD BE EXPECTED TO OCCUR ONLY DURING A LARGE BREAK LOSS OF COOLANT ACCIDENT, WOULD NOT BE A CONCERN BECAUSE THE PRIMARY SOURCE OF COOLING FOR THE LARGE BREAK LOCA IS ACTUALLY THE RESIDUAL HEAT REMOVAL SYSTEM. IT WAS CONCLUDED THAT THIS EVENT DID NOT RESULT IN A CONDITION THAT POSED A SIGNIFICANT SAFETY PROBLEM AS DEFINED IN 10 CFR 50.59.

[69] COOK 1 DOCKET 50-315 LER 87-017
FAILURE TO INCORPORATE CHANGES TO PRESSURIZER LEVEL PROTECTION SET VALUES INTO PROCEDURES.
EVENT DATE: 081587 REPORT DATE: 091687 NSSS: WE TYPE: PWR

(NSIC 206056) ON AUGUST 7, 1987 IT WAS DISCOVERED THAT THE VALUES FOR THE TRANSMITTER SPAN FOR THE PRESSURIZER LEVEL WERE INCORRECT IN CALIBRATION PROCEDURES 1 THP 6030 IMP.108, 109 AND 110 "PRESSURIZER LEVEL PROTECTION SET FOR CHANNELS I, II AND III," RESPECTIVELY. THE CORRECT VALUES WERE CONTAINED IN ENGINEERING CONTROL PROCEDURE (ECP) 12-NI-01 APPROVED AND ISSUED ON JANUARY 28, 1977. THESE VALUES HAD NEVER BEEN INCORPORATED INTO UNIT ONE PROCEDURES. IT WAS DETERMINED ON AUGUST 15, 1987 THAT THE UNIT ONE HIGH PRESSURIZER LEVEL REACTOR TRIP SETPOINT OF 91 PERCENT WAS THE EQUIVALENT OF 93.0027 PERCENT OF INDICATED SPAN AS DETERMINED BY THE ECP VALUES. THIS IS OUTSIDE THE TECHNICAL SPECIFICATION ALLOWABLE VALUE OF LESS THAN OR EQUAL TO 93 PERCENT. THE REASON FOR THE ERROR WAS DEFECTIVE PROCEDURES ALTHOUGH THE ACTUAL CAUSE FOR THE CONDITION IS UNKNOWN DUE TO THE TIME FRAME INVOLVED. THE UNIT ONE PROCEDURES WERE CHANGED TO REFLECT THE ECP VALUES AND THE PRESSURIZER LEVEL TRANSMITTERS WERE RECALIBRATED WITH IMP.109 AND 110 BEING COMPLETED ON AUGUST 10, 1987 AND IMP.108 BEING COMPLETED ON AUGUST 20, 1987.

[70] COOK 1 DOCKET 50-315 LER 87-019
REACTOR COOLANT SYSTEM FLOW LESS THAN REQUIRED DURING REFUELING DUE TO PROCEDURAL INADEQUACY.
EVENT DATE: 081887 REPORT DATE: 091787 NSSS: WE TYPE: PWR

(NSIC 206057) ON AUGUST 18, 1987, AT 1120 HOURS, WITH THE UNIT IN THE REFUELING MODE AND THE REACTOR VESSEL HEAD IN PLACE, BUT NOT BOLTED, THE RESIDUAL HEAT REMOVAL SYSTEM (RHRS) FLOW (REACTOR COOLANT SYSTEM (RCS) COOLING LOOP) WAS REDUCED TO LESS THAN 3000 GALLONS PER MINUTE (GPM) AS REQUIRED BY PROCEDURES TO DRAIN THE RCS TO HALF-LOOP LEVEL FOR REQUIRED MAINTENANCE. THIS IS DONE TO PREVENT AIR BINDING OF THE RHRS PUMP. ALTHOUGH MEETING THE INTENT OF TECHNICAL SPECIFICATION 3.9.8.1 FOR REFUELING MODE RHRS, THE SURVEILLANCE REQUIREMENT OF

3000 GPM WAS NOT MET. THE ACTION STATEMENT FOR T.S. 3.9.8.1 REQUIRES THAT CONTAINMENT INTEGRITY BE ESTABLISHED WITHIN FOUR HOURS IF THERE IS NO OPERATING (GREATER THAN OR EQUAL TO 3000 GPM) RHRS LOOP. THIS WAS NOT DONE DUE TO THE REQUIREMENT NOT BEING RECOGNIZED. THIS EVENT WAS CAUSED BY THE PROCEDURAL INADEQUACY WHICH ALLOWED FOR REDUCTION OF THE RCS FLOW BELOW 3000 GPM. THE REQUIREMENT WAS RECOGNIZED AT 2030 HOURS WHILE TAKING SURVEILLANCE READINGS. THE RCS LEVEL WAS THEN INCREASED TO ALLOW GREATER THAN 3000 GPM FLOW. THE REQUIRED FLOW WAS ESTABLISHED AT 2050 HOURS. THE APPROPRIATE PROCEDURES WILL BE CHANGED AS NEEDED PRIOR TO THE NEXT REFUELING IN ORDER TO ENSURE THE REQUIREMENTS OF T.S. 3.9.8.1 ARE ADDRESSED.

[71] COOK 1 DOCKET 50-315 LER 87-018
FIRE RATED ASSEMBLIES AND DAMPERS INOPERABLE DUE TO INCORRECT APPLICATION OF SURVEILLANCE REQUIREMENT FOR INSPECTING FIRE SEALS.
EVENT DATE: 082687 REPORT DATE: 092587 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: COOK 2 (PWR)
VENDOR: AIRSTREAM PROD DIV

(NSIC 206518) ON AUGUST 26, 1987, AT 1615 HOURS, WITH UNIT 1 IN MODE 5 (COLD SHUTDOWN) AND UNIT 2 IN MODE 1 (POWER OPERATION) AT 80 PERCENT REACTOR THERMAL POWER, CERTAIN FIRE RATED ASSEMBLIES AND FIRE DAMPERS WERE DECLARED INOPERABLE. IT WAS DETERMINED THAT THE SURVEILLANCE REQUIREMENTS REQUIRED BY TECH SPEC 4.7.10.1.A AND 4.7.10.1.B WERE PERFORMED ON 10 PERCENT OF THE FIRE RATED ASSEMBLIES (WALLS, FLOORS, CEILINGS, CABLE TRAY ENCLOSURES) AND DAMPERS. THE UNTESTED FIRE RATED ASSEMBLIES WERE TECHNICALLY INOPERABLE FROM 03/29/87 UNTIL 08/29/87, WHEN THE SURVEILLANCE ACTIVITY WAS COMPLETED. THE UNTESTED FIRE DAMPERS WERE TECHNICALLY INOPERABLE FROM 07/01/87 UNTIL 08/29/87 WHEN THE SURVEILLANCE ACTIVITY WAS COMPLETED ON ALL BUT ONE DAMPER. THE FINAL DAMPER WAS TESTED ON 09/14/87. THIS EVENT WAS CAUSED BY PERSONNEL ERROR (NON-LICENSED SUPERVISOR) IN THAT THE 10 PERCENT TEST REQUIREMENT OF SURVEILLANCE 4.7.10.1.C FOR SEALED PENETRATIONS WAS INCORRECTLY APPLIED TO SURVEILLANCES 4.7.10.1.A AND 4.7.10.1.B. SURVEILLANCE PROCEDURES CURRENTLY IN PLACE HAVE BEEN REVIEWED AND DETERMINED TO BE ADEQUATE TO CORRECTLY DIRECT PERFORMANCE OF THE REQUIRED INSPECTIONS. FUTURE SURVEILLANCES PERFORMED FOR THESE ACTIVITIES WILL INCLUDE ALL FIRE RATED ASSEMBLIES AND DAMPERS.

[72] COOK 2 DOCKET 50-316 LER 87-009
RADIATION MONITOR INOPERABLE WITHOUT TECH SPEC REQUIRED COMPENSATORY SAMPLE DUE TO PERSONNEL ERROR.
EVENT DATE: 082587 REPORT DATE: 092487 NSSS: WE TYPE: PWR
VENDOR: EBERLINE INSTRUMENT CORP.

(NSIC 206454) ON AUGUST 25, 1987, AT 1225 HOURS, DURING PERFORMANCE OF SHIFT SURVEILLANCES, THE GLAND SEAL LEAKOFF (GSLO) CONDENSER EFFLUENT RADIATION MONITOR, SRA-2800, WAS FOUND TO BE INOPERABLE DUE TO EXCESSIVE CONDENSATION BUILD-UP IN THE SAMPLE LINE. A REVIEW OF SRA-2800 HISTORY FILE INDICATES THAT IT FAILED AT 0152 HOURS ON AUGUST 25, 1987. AN ADMINISTRATIVELY REQUIRED SHIFT SURVEILLANCE PERFORMED AT APPROXIMATELY 0230 HOURS DID NOT DISCOVER THE INOPERABILITY DUE TO MISCOMMUNICATION BETWEEN THE TWO LICENSED OPERATORS PERFORMING THE SURVEILLANCES (PERSONNEL ERROR). THIS MISUNDERSTANDING RESULTED IN THE EIGHT HOUR GRAB SAMPLE, REQUIRED BY TECHNICAL SPECIFICATION 3.3.3.10, NOT BEING OBTAINED UNTIL 1306 HOURS. SRA-2800 WAS RESTORED TO OPERABLE STATUS AT 1317 HOURS. CONTRIBUTING TO THIS EVENT WAS AN EQUIPMENT FAILURE ON THE UNIT 2 RADIATION MONITORING SYSTEM (RMS) WHICH REQUIRED THAT RMS DATA BE OBTAINED IN THE UNIT 1 CONTROL ROOM. THIS FAILURE PREVENTED A PERIODIC OBSERVATION OF RMS STATUS IN THE UNIT 2 CONTROL ROOM AND DISRUPTED THE NORMAL SURVEILLANCE ROUTINE. A PREVIOUSLY APPROVED DESIGN CHANGE WILL IMPROVE THE CONTROL ROOM RMS INDICATION CAPABILITY. THE INVOLVED PERSONNEL WERE COUNSELED ON THE IMPORTANCE OF CLEAR AND CONCISE COMMUNICATIONS.

[73] COOK 2 DOCKET 50-316 LER 87-010
ICE BUILDUP IN ICE CONDENSER FLOW PASSAGES DUE TO SUBLIMIATION.
EVENT DATE: 090287 REPORT DATE: 092587 NSSS: WE TYPE: PWR

(NSIC 206455) ON SEPTEMBER 2, 1987, WITH UNIT 2 IN MODE 5 (COLD SHUTDOWN), FLOW PASSAGE INSPECTIONS OF THE ICE CONDENSER REVEALED FROST AND ICE BUILDUP ON THE LATTICE FRAMES OF GREATER THAN 3/8 INCH IN A TOTAL OF FORTY-SIX FLOW PASSAGES IN SIX OF THE TWENTY-FOUR ICE CONDENSER BAYS. A SUBSEQUENT INSPECTION INDICATED THAT THERE WAS ALSO FROST AND ICE FORMATION BETWEEN THE WALLS AND ICE BASKETS ADJACENT TO THE WALLS. TECHNICAL SPECIFICATION (T/S) 4.6.5.1.B.3 LIMITS FROST OR ICE BUILDUP IN FLOW PASSAGES TO A NOMINAL THICKNESS OF 3/8 INCH. ACCORDING TO THIS T/S, BUILDUP EXCEEDING THIS LIMIT IN TWO OR MORE FLOW PASSAGES PER BAY IS EVIDENCE OF ABNORMAL DEGRADATION. THOUGH THE EVALUATION HAS CONCLUDED THAT THE DEGRADATION IS NOT SERIOUS, IT IS BELIEVED THAT ISSUANCE OF THIS VOLUNTARY LER IS APPROPRIATE SINCE SOME DEGRADATION HAS BEEN IDENTIFIED. ACTIONS TAKEN TO CORRECT THE ABNORMAL DEGRADATION INCLUDED A DEFROST OF THE ICE CONDENSER AND AN INTERNAL INVESTIGATION OF THE EVENT. THE INTERNAL INVESTIGATION, AIDED BY A PREVIOUS WESTINGHOUSE EVALUATION, INDICATED THAT THERE WERE NO SAFETY CONCERNS, THAT IS, THAT THE ICE CONDENSER REMAINED IN A CONFIGURATION IN WHICH IT WOULD HAVE PERFORMED ITS INTENDED SAFETY FUNCTION.

[74] COOPER DOCKET 50-298 LER 87-009
UNANTICIPATED REACTOR SCRAM AND GROUP ISOLATIONS DUE TO LOW REACTOR VESSEL WATER LEVEL CAUSED BY INADVERTENT MANUAL TRIP OF THE OPERATING REACTOR FEEDWATER PUMP.
EVENT DATE: 021887 REPORT DATE: 031287 NSSS: GE TYPE: BWR

(NSIC 206571) AT 1350, A REACTOR SCRAM OCCURRED ACCOMPANIED BY GROUP II, III, AND VI ISOLATIONS WHEN THE OPERATING REACTOR FEED PUMP WAS MISTAKENLY TRIPPED BY A STATION OPERATOR FROM THE LOCAL CONTROL STATION. THE STATION OPERATOR (UNLICENSED OPERATOR) HAD BEEN DISPATCHED, SUBSEQUENT TO DISCUSSIONS AND REVIEW WITH LICENSED CONTROL ROOM OPERATORS OF THE PROCEDURAL REQUIREMENTS, TO CONDUCT A LOCAL TRIP TEST OF THE STANDBY REACTOR FEED PUMP TURBINE WHICH WAS IN THE INITIAL STAGES OF BEING PLACED IN SERVICE. UPON ENTERING THE FEED PUMP ROOM, HE APPARENTLY BECAME DISORIENTED WITH RESPECT TO NORTH-SOUTH DIRECTION AND PUMP LOCATION AND TRIPPED THE OPERATING PUMP. AT THE TIME OF THIS EVENT, POWER LEVEL WAS APPROXIMATELY 50 PERCENT AND PREPARATIONS WERE BEING MADE TO INCREASE POWER TO 100 PERCENT. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR ON THE PART OF THE STATION OPERATOR DISPATCHED TO CONDUCT A LOCAL TRIP TEST OF THE REACTOR FEED PUMP TURBINE BEING PLACED IN SERVICE, IN ACCORDANCE WITH THE OPERATING PROCEDURE. A CONTRIBUTING CAUSE IS CONSIDERED TO BE THE LACK OF HUMAN FACTORS IN COMPONENT/EQUIPMENT MARKINGS. CORRECTIVE ACTION TAKEN INCLUDED ESTABLISHING STABLE PLANT CONDITIONS AND IMPLEMENTING NORMAL SCRAM RECOVERY PROCEDURES.

[75] COOPER DOCKET 50-298 LER 87-019
INITIATION OF PLANT SHUTDOWN DUE TO MALFUNCTIONING PRESSURE SUPPRESSION CHAMBER AS A RESULT OF REACTOR BUILDING VACUUM BREAKER VALVES FAILING TO TEST FULLY.
EVENT DATE: 080887 REPORT DATE: 091587 NSSS: GE TYPE: BWR
VENDOR: MILLER, RALPH A., CO.

(NSIC 206048) ON AUGUST 8, 1987, AT 2:30 AM, ONE OF THE TWO PRESSURE SUPPRESSION CHAMBER - REACTOR BUILDING VACUUM BREAKER VALVES (PC-243AV, A 20" AIR OPERATED BUTTERFLY VALVE) REPOSITIONED FROM ITS NORMALLY CLOSED TO A PARTIALLY OPEN POSITION WHEN A GASKET ON THE IN-LINE AIR FILTER/PILOT AIR REGULATOR FAILED. SUBSEQUENTLY, A SLOW DECREASE IN PRIMARY CONTAINMENT PRESSURE WAS OBSERVED WHICH, UPON FURTHER INVESTIGATION, WAS DETERMINED TO BE DUE TO THE FACT THAT THE CORRESPONDING PRESSURE SUPPRESSION CHAMBER REACTOR BUILDING VACUUM RELIEF VALVE (CV-13, WHICH IS A 20" WAFER CHECK VALVE) WAS NOT FULLY SEATED. A PLANT SHUTDOWN WAS INITIATED AND A NOTIFICATION OF UNUSUAL EVENT (NOUE) WAS DECLARED BASED UPON AN APPARENT LOSS OF PRIMARY CONTAINMENT INTEGRITY. THE CAUSE OF THE EVENT WAS

DETERMINED TO BE DUE TO VACUUM BREAKER RELIEF VALVE CV-13 NOT BEING FULLY SEATED WHICH, IN TURN, WAS DUE TO LACK OF A PRESCRIBED TECHNIQUE FOR TEST PERFORMANCE IN THE SURVEILLANCE TEST PROCEDURE. THE CAUSE OF THE IN-LINE AIR FILTER/PILOT AIR REGULATOR GASKET FAILURE IS NOT KNOWN. IMMEDIATE CORRECTIVE ACTIONS TAKEN INCLUDED DISPATCHING OPERATIONS PERSONNEL TO THE VICINITY OF THE MALFUNCTIONING VALVES AND MANUALLY CAUSING THE VACUUM RELIEF WATER CHECK VALVE TO SEAT.

[76] CRYSTAL RIVER 3 DOCKET 50-302 LER 87-016
PERSONNEL COGNITIVE ERROR RESULTS IN SP SPECIFYING RPS SETPOINTS OUTSIDE REQUIRED TECHNICAL SPECIFICATION VALUES.
EVENT DATE: 080387 REPORT DATE: 090287 NSSS: BW TYPE: PWR

(NSIC 206302) ON AUGUST 3, 1987, CRYSTAL RIVER UNIT 3 WAS IN HOT STANDBY (MODE 3). DURING A REVIEW OF STATION NON-CONFORMING OPERATIONS REPORTS (NCORS) ON INSTRUMENT DRIFT, IT WAS DISCOVERED THAT THE SURVEILLANCE PROCEDURE (SP) FOR CALIBRATION OF THE REACTOR PROTECTION SYSTEM (RPS) DID NOT CONTAIN PROPER SETPOINTS FOR ANTICIPATORY REACTOR TRIP UPON LOSS OF THE MAIN TURBINE (MT) OR BOTH MAIN FEEDWATER PUMPS (MFP). THE SIGNALS UPON WHICH THE TRIPPED INDICATION OF THE MT AND MFPS IS BASED, ARE THE OIL PRESSURES WITHIN EACH TURBINE'S CONTROL SYSTEM. THE VALUES FOR CONTROL OIL PRESSURES WERE PLACED INTO TECHNICAL SPECIFICATIONS BY A CHANGE WHICH WAS APPROVED IN JANUARY 1987. THIS EVENT WAS CAUSED BY UTILITY ENGINEERING PERSONNEL FAILING TO ADEQUATELY PERFORM THE SPECIFIED PROCEDURE REVIEW/REVISION NECESSITATED BY THE TECH SPEC CHANGE. THOSE PROCEDURES USED FOR SETTING THE ANTICIPATORY REACTOR TRIP SETPOINTS AND FOR SURVEILLANCE OF THOSE SETPOINTS HAVE BEEN REVISED TO REFLECT THE TECH SPEC REQUIRED VALUES. PLANT PERSONNEL HAVE BEEN REMINDED OF THE NECESSITY TO PAY PARTICULAR ATTENTION TO TECH SPEC CHANGES AND THEIR EFECTS.

[77] CRYSTAL RIVER 3 DOCKET 50-302 LER 87-017
ENTRY INTO HOT SHUTDOWN TO REPAIR CONTROL ROD DRIVE MECHANISM.
EVENT DATE: 080887 REPORT DATE: 090487 NSSS: BW TYPE: PWR

(NSIC 206283) ON AUGUST 8, 1987, CRYSTAL RIVER UNIT 3 WAS SHUT DOWN TO HOT STANDBY CONDITIONS TO ALLOW MAINTENANCE ON A FAILED CONTROL ROD DRIVE MECHANISM (CRDM). OPERATION HAD SECURED THE STEAM SUPPLY TO THE TURBINE DRIVEN EMERGENCY FEEDWATER PUMP (EFP) DURING COOLDOWN IN ACCORDANCE WITH PLANT PROCEDURES. PLANT TECHNICAL SPECIFICATIONS REQUIRE TWO OPERABLE EFPS DURING POWER OPERATION, STARTUP, AND HOT STANDBY OPERATING MODES. PROBLEMS ENCOUNTERED DURING CRDM REPAIRS DELAYED PLANT HEATUP. THIS PREVENTED RESTORATION OF THE PUMP TO OPERABLE STATUS WITHIN THE TIME ALLOWED BY TECHNICAL SPECIFICATIONS. THEREFORE, OPERATORS COOLED THE PLANT TO HOT SHUTDOWN CONDITIONS IN ORDER TO COMPLY WITH APPLICABLE TECHNICAL SPECIFICATION ACTION STATEMENTS. ANALYSIS WILL BE PERFORMED TO CLARIFY EMERGENCY FEEDWATER FLOW REQUIREMENTS AND EFP FLOW CAPABILITIES AT LOW REACTOR COOLANT SYSTEM TEMPERATURES AND LOW EFP STEAM SUPPLY PRESSURES. THE RESULTS OF THIS ANALYSIS MAY ALLOW THE TURBINE DRIVEN EFW PUMP TO REMAIN OPERABLE DURING SIMILAR SITUATIONS IN THE FUTURE.

[78] DAVIS-BESSE 1 DOCKET 50-346 LER 87-010
UNIT TRIP FROM FULL POWER DURING A SEVERE THUNDERSTORM.
EVENT DATE: 082187 REPORT DATE: 091887 NSSS: BW TYPE: PWR

(NSIC 206069) ON AUGUST 21, 1987 AT 2258 HOURS, DURING A SEVERE THUNDERSTORM, THE UNIT EXPERIENCED A TRIP FROM 100 PERCENT REACTOR THERMAL POWER. THE POST-TRIP RESPONSE WAS NORMAL WITH EXCEPTION OF A FASTER THAN NORMAL REACTOR COOLANT SYSTEM (RCS) TEMPERATURE AND PRESSURE REDUCTION. THE MAIN CONTRIBUTOR WAS THE EXCESSIVE STEAM LOAD WHEN THE 2ND STAGE REHEAT STEAM SUPPLY VALVE TO THE #1 MOISTURE SEPARATOR REHEATER FAILED TO AUTOMATICALLY CLOSE AFTER THE TURBINE TRIP. THE CAUSE OF THIS VALVE FAILURE WAS TRACED TO A PRESSURE SWITCH, PS9806 WHICH FAILED

TO ACTUATE. THE CAUSE OF THE UNIT TRIP WAS A PARTIAL LOSS OF THE A AND C PHASES IN THE POWER GRID WHICH CAUSED A SUDDEN SHIFT IN THE TURBINE-GENERATOR SHAFT POSITION AND RESULTANT VIBRATION SPIKES AT THE BEARINGS. THE #9 BEARING EXCEEDED ITS VIBRATION TRIP SETPOINT WHICH CAUSED A TURBINE-GENERATOR TRIP WHICH INITIATED AN ANTICIPATORY REACTOR TRIP SYSTEM (ARTS) TRIP OF THE REACTOR. THE UNIT WAS BACK ON LINE ON AUGUST 23, 1987 AND RETURNED TO FULL POWER OPERATION ON AUGUST 24, 1987. THE PRESSURE SWITCH WAS REPAIRED AND SELECTED TURBINE-GENERATOR BEARING TRIP SETPOINTS WERE INCREASED SUCH THAT ALL TRIPS NOW OCCUR AT 12 MILS.

[79] DIABLO CANYON 1 DOCKET 50-275 LER 87-012
CONTAINMENT FAN COOLER UNIT 1-3 REQUIRED COOLING WATER FLOWRATE WAS NOT MET DUE TO AN OUT OF TOLERANCE FLOW INSTRUMENT.
EVENT DATE: 121284 REPORT DATE: 091487 NSSS: WE TYPE: PWR

(NSIC 206042) ON DECEMBER 12, 1984, WITH UNIT 1 IN MODE 1 (POWER OPERATION), THE ALLOWED OUTAGE TIME OF TECHNICAL SPECIFICATION 3.6.2.3 WAS EXCEEDED WHEN THE COMPONENT COOLING WATER FLOWRATE TO CONTAINMENT FAN COOLER UNIT 1-3 WAS LESS THAN THE 2000 GPM REQUIRED BY TECHNICAL SPECIFICATION 4.6.2.3.A.2. THIS EVENT OCCURRED AGAIN ON FEBRUARY 12, 1985. ON BOTH OF THESE OCCASIONS THE FLOW INSTRUMENT (FI-36) WAS INDICATING 2000 GPM BUT THE ACTUAL FLOWRATE WAS SUBSEQUENTLY DETERMINED TO BE APPROXIMATELY 1950 GPM. THE INCORRECT INDICATION WAS DUE TO A FLOW INSTRUMENT BEING OUT-OF-TOLERANCE. THE CAUSE OF THIS EVENT WAS ATTRIBUTED TO INSTRUMENT DRIFT RESULTING IN THE FLOW INSTRUMENT (FI-36) INDICATING HIGHER THAN ACTUAL AND A FAILURE TO PERFORM AN EVALUATION OF THE RELATED TEST RESULTS WHEN THE OUT-OF-TOLERANCE CONDITION WAS IDENTIFIED. THIS EVENT WAS DISCOVERED AS THE RESULT OF A NONCONFORMANCE RESOLUTION. THE RESOLUTION REQUIRED THE REVIEW OF PREVIOUSLY PERFORMED SURVEILLANCE TESTS TO DETERMINE THE IMPACT ON TEST RESULTS OF INSTRUMENTS THAT WERE FOUND TO BE OUT-OF-TOLERANCE. AT THE TIME OF DISCOVERY OF THIS EVENT, FLOW INSTRUMENT FI-36 WAS READING WITHIN TOLERANCE AND THE SURVEILLANCE REQUIREMENT WAS BEING MET.

[80] DIABLO CANYON 1 DOCKET 50-275 LER 87-013
CONTAINMENT VENTILATION ISOLATION INITIATION DUE TO PERSONNEL ERROR.
EVENT DATE: 082487 REPORT DATE: 092387 NSSS: WE TYPE: PWR

(NSIC 206449) ON AUGUST 24, 1987, AT 1347 PDT, WITH THE UNIT IN MODE 1 (POWER OPERATION) AN AUTOMATIC INITIATION OF THE CONTAINMENT VENTILATION ISOLATION SYSTEM CVIS OCCURRED. THE SAMPLE LINE ISOLATION VALVES FOR GASEOUS 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 NON-EMERGENCY EVENT REPORT WAS MADE AT 1425 PDT, AUGUST 24, 1987. THIS EVENT WAS CAUSED BY A LICENSED SENIOR OPERATOR PERFORMING A SOURCE CHECK ON THE INCORRECT RADIATION MONITOR PRIOR TO A LIQUID RADWASTE DISCHARGE. THE CVIS WAS RESET AT 1347 PDT, AUGUST 24, 1987. THE OPERATOR WAS COUNSELED CONCERNING HIS IMPROPER ACTION. AN INCIDENT REPORT WAS ISSUED ON THIS EVENT AND WILL BE REVIEWED WITH OPERATORS TO EMPHASIZE THE IMPORTANCE OF VERIFYING EQUIPMENT PRIOR TO ACTUATING ANY FUNCTIONS ASSOCIATED WITH TESTING.

[81] DIABLO CANYON 1 DOCKET 50-275 LER 87-014
INADVERTENT START OF DIESEL GENERATOR 1-3 WHEN OPERATOR PULLED WRONG FUSE WHILE RETURNING A VITAL BUS FEEDER BREAKER TO OPERABILITY FOLLOWING PLANNED MAINTENANCE.
EVENT DATE: 082587 REPORT DATE: 092487 NSSS: WE TYPE: PWR

(NSIC 206450) ON AUGUST 25, 1987, AT 2121 PDT, WITH THE UNIT IN MODE 1 (POWER OPERATION), DIESEL GENERATOR 1-3 AUTOSTARTED AND LOADED ONTO 4 KV BUS F. THE AUTOSEQUENCED LOADS DID NOT LOAD ONTO THE BUS. THE CONTROL ROOM OPERATOR, WHILE PERFORMING HIS VERIFICATION OF OPERATING EQUIPMENT, OBSERVED THAT THE CHARGING

PUMP HAD STOPPED. THE CONTROL ROOM OPERATOR IMMEDIATELY STARTED A CHARGING PUMP TO PROVIDE REACTOR COOLANT PUMP SEAL INJECTION. AFTER THE BUS WAS TRANSFERRED BACK TO AUXILIARY POWER. THE DIESEL GENERATOR WAS SECURED AND RETURNED TO NORMAL STANDBY MODE. THE 4-HOUR NONEMERGENCY REPORT REQUIRED BY 10 CFR 50.72 WAS MADE AT 2335 PDT, AUGUST 25, 1987. THIS EVENT WAS CAUSED BY PERSONNEL ERROR WHEN AN UNLICENSED OPERATOR, IN THE PROCESS OF RETURNING A VITAL BUS FEEDER BREAKER TO OPERABILITY FOLLOWING PREVENTIVE MAINTENANCE, INADVERTENTLY PULLED THE WRONG FUSE. THE AUTOSEQUENCING LOADS DID NOT LOAD ONTO THE BUS DUE TO THE APPARENT UNDERVOLTAGE CONDITION SENSED ON THE BUS AND A DESIGN FEATURE IN THE BREAKER WHICH LOCKS IN THE TRIP CONDITION TO PREVENT THE BREAKER FROM CYCLING. REINITIATING THE BREAKER CLOSE SIGNAL ALLOWED THE BREAKERS TO CLOSE. TO PREVENT RECURRENCE, AN INCIDENT REPORT WAS ISSUED AND WILL BE REVIEWED WITH ALL OPERATORS.

[82] DIABLO CANYON 2 DOCKET 50-323 LER 87-019
AUTOSTART OF DIESEL GENERATOR 2-2 DUE TO A WIRE BEING BROKEN DURING REINSTALLATION OF A 4KV BREAKER.
EVENT DATE: 081487 REPORT DATE: 091487 NSSS: WE TYPE: PWR

(NSIC 206059) ON AUGUST 14, 1987, AT 2127 PDT, WITH UNIT 2 IN MODE 1 (POWER OPERATION), DIESEL GENERATOR (DG) 2-2 AUTOMATICALLY STARTED AND LOADED ONTO THE VITAL 4 KV BUS H. THE 4 KV BREAKER FOR COMPONENT COOLING WATER PUMP 2-3 WAS BEING REINSTALLED INTO ITS CUBICLE ON VITAL BUS H AFTER MAINTENANCE. A WIRE ASSOCIATED WITH THE VITAL BUS UNDERVOLTAGE CIRCUITRY WAS SHORT-CIRCUITED AND SEVERED BY THE MECHANICAL ACTION OF INSTALLING THE BREAKER. THE SHORT CIRCUITED WIRE RESULTED IN AN OPENED BUS POTENTIAL FUSE AND FIRST- AND SECOND-LEVEL UNDERVOLTAGE RELAY ACTUATION. THESE (SECOND-LEVEL) RELAY ACTUATIONS RESULTED IN THE AUTOMATIC START AND LOADING OF DG 2-2, VITAL BUS H STRIPPING ACTION AND LOADING OF DG 2-2 ONTO BUS H, POWERING ITS 480 V LOADS. THE EFFECTS OF THE OPENED BUS POTENTIAL FUSE AND THE SEVERED WIRE PREVENTED THE BUS STRIPPING SIGNAL FROM RESETTING. THIS PREVENTED AUTOMATIC OR MANUAL LOADING OF THE REMAINING 4 KV LOADS ONTO BUS H. AT 2307 PDT, THE BUS POTENTIAL FUSE WAS REPLACED, RESTORING OPERABILITY TO THE REMAINING EQUIPMENT ON THE BUS. ON AUGUST 15, 1987, AT 0743 PDT THE SEVERED WIRE WAS REPLACED, THE VITAL BUS ALIGNED TO OFFSITE POWER, AND DG 2-2 RETURNED TO ITS STANDBY CONDITION. THE NOTIFICATIONS REQUIRED BY 10 CFR 50.72 WERE COMPLETED ON AUGUST 15, 1987, AT 0033 PDT.

[83] DIABLO CANYON 2 DOCKET 50-323 LER 87-020
ENTRY INTO TECHNICAL SPECIFICATION 3.0.3 DUE TO BOTH TRAINS OF AUXILIARY BUILDING VENTILATION BEING INOPERABLE.
EVENT DATE: 081887 REPORT DATE: 091687 NSSS: WE TYPE: PWR

(NSIC 206060) ON AUGUST 18, 1987, AT 1320 PDT, AND AGAIN ON SEPTEMBER 1, 1987, AT 1910 PDT, WITH THE UNIT IN MODE 1 (POWER OPERATION) AT 100 PERCENT POWER, BOTH TRAINS OF THE AUXILIARY BUILDING VENTILATION SYSTEM WERE INOPERABLE, RESULTING IN VIOLATION OF TECHNICAL SPECIFICATION (TS) 3.7.6.1 AND ENTRY INTO TS 3.0.3. IN BOTH CASES, SUPPLY FAN S-34 HAD BEEN MANUALLY SECURED, AND THE FLOW SENSOR FOR THE OPERATING FAN S-33 SENSED A "NO-FLOW" CONDITION, TRIPPING FAN S-33. THE MOST PROBABLE CAUSE IS THAT A FLOW PATH FROM THE DISCHARGE PLENUM TO THE INTAKE ROOM ALLOWED PRESSURE TO EQUALIZE ENOUGH TO GIVE AN APPARENT "NO-FLOW" CONDITION. THE FLOW PATH FOR THE AUGUST 18 EVENT INVOLVED BACKFLOW THROUGH THE PARALLEL FAN S-34'S DAMPERS, AND THE SEPTEMBER 1 EVENT INVOLVED BACKFLOW THROUGH THE DOOR BETWEEN THE DISCHARGE PLENUM AND THE INTAKE ROOM. ON AUGUST 18, 1987, AT 1355 PDT, TS 3.0.3 WAS EXITED WHEN BOTH SUPPLY FANS WERE PLACED IN OPERATION. ON SEPTEMBER 1, 1987, AT 1930 PDT, TS 3.0.3 WAS EXITED WHEN ONE TRAIN OF AUXILIARY BUILDING VENTILATION WAS PLACED IN OPERATION. PG AND E IS CONTINUING THE INVESTIGATION INTO THESE EVENTS TO DETERMINE THE ROOT CAUSE AND APPROPRIATE CORRECTIVE ACTIONS. A SUPPLEMENTAL REPORT WILL BE SUBMITTED WHEN THIS INVESTIGATION IS COMPLETE.

[84] DRESDEN 2 DOCKET 50-237 LER 87-022
NUMBER OF OPERABLE APRM DOWNSCALE TRIP CHANNELS LESS THAN ALLOWABLE DUE TO BYPASSING OF APRM 4 WHILE IRM 16 WAS IN BYPASS.
EVENT DATE: 081887 REPORT DATE: 091587 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 206428) ON AUGUST 18, 1987 AT 0019 HOURS WITH UNIT 2 IN THE RUN MODE AT 94% POWER, AVERAGE POWER RANGE MONITOR (APRM) 4 GENERATED A SPURIOUS HIGH HIGH SIGNAL WHICH GENERATED A TRIP OF THE "B" REACTOR PROTECTION SYSTEM (RPS) CHANNEL. THE APRM HIGH HIGH ALARM WAS CLEARED, THE RPS TRIP RESET AND APRM 4 WAS PLACED IN BYPASS. IRM 16 HAD PREVIOUSLY BEEN BYPASSED DUE TO ERRATIC READINGS. HAVING APRM 4 AND IRM 16 SIMULTANEOUSLY IN BYPASS REDUCED THE NUMBER OF OPERABLE APRM DOWNSCALE TRIP FUNCTIONS IN THE "B" RPS CHANNEL TO BELOW THE NUMBER REQUIRED BY THE TECH SPECS. THIS VIOLATION CONDITION WAS BROUGHT TO THE ATTENTION OF MANAGEMENT PERSONNEL BY AN NRC INSPECTOR. THE VIOLATION CONDITION TERMINATED AT AUGUST 18, 1987 AT 0922 HOURS WHEN IRM 16 WAS REMOVED FROM BYPASS. THE ROOT CAUSE OF THIS EVENT WAS AN ADMINISTRATIVE DEFICIENCY AS THERE WERE NO ADMINISTRATIVE CONTROLS RESTRICTING THE COMBINATIONS OF APRMS AND IRMS THAT COULD BE BYPASSED. IMMEDIATE CORRECTIVE ACTIONS ENTAILED ISSUANCE OF AN OPERATING ORDER AND PROCEDURE CHANGES ALERTING THE OPERATORS AS TO WHICH COMBINATIONS OF APRMS AND IRMS COULD BE BYPASSED. LONG TERM CORRECTIVE ACTION WILL INCLUDE OPERATOR TRAINING AND PURSUIT OF TECH SPEC RELIEF FROM THIS REQUIREMENT.

[85] DRESDEN 2 DOCKET 50-237 LER 87-024
REACTOR SCRAM ON LOW LEVEL DUE TO 2A FEEDWATER REGULATING VALVE FAILURE.
EVENT DATE: 082187 REPORT DATE: 092187 NSSS: GE TYPE: BWR
VENDOR: COPES-VULCAN, INC.

(NSIC 206326) ON AUGUST 21, 1987 AT 1755 HOURS WITH UNIT 2 AT APPROXIMATELY 2350 MW THERMAL (93% POWER) THE REACTOR SCRAMMED ON A LOW REACTOR WATER LEVEL SIGNAL OF +8 INCHES. THE ROOT CAUSE OF THE EVENT WAS THE FAILURE OF THE 2A FEEDWATER REGULATING VALVE (FWRV). THE STEM AND PLUG ASSEMBLY OF THE 2A FWRV SEPARATED DUE TO A FATIGUE CRACK. CORRECTIVE ACTIONS ENTAILED REPLACEMENT OF THE STEM/PLUG ASSEMBLY WITH A NEW WELDED STEM/PLUG. ALSO TO PREVENT RECURRENCE A NEW TRIM PACKAGE INCORPORATING A LIGHTER PLUG THAT IS LESS SUSCEPTIBLE TO FATIGUE CRACKING WILL BE CONSIDERED AS REPLACEMENT PARTS IN THE FUTURE. THIS DECISION WILL BE MADE FOLLOWING THE TESTING BEING PERFORMED ON UNIT 3 AS A RESULT OF THE 8/7/87 UNIT 3 SCRAM DUE TO FEEDWATER SYSTEM OSCILLATING. THE SAFETY SIGNIFICANCE WAS MINIMAL SINCE ALL EMERGENCY CORE COOLING SYSTEMS WERE AVAILABLE, HOWEVER, NO ACTUATION WAS NECESSARY, THE FEEDWATER LEVEL CONTROL SYSTEM REMAINED CAPABLE OF MANUAL CONTROL OF REACTOR LEVEL AT ALL TIMES AND THE REACTOR SCRAMMED AT THE SPECIFIED CONSERVATIVE SETPOINT. FIVE (5) PREVIOUS OCCURRENCES WERE REPORTED BY LICENSEE EVENT REPORTS #87-12 AND #84-10 ON DOCKET 050249 AND #84-23, #87-16, #84-9 ON DOCKET 050237

[86] DRESDEN 2 DOCKET 50-237 LER 87-027
FUNCTIONAL TEST OF RPS MOTOR GENERATOR SET.
EVENT DATE: 091687 REPORT DATE: 092887 NSSS: GE TYPE: BWR

(NSIC 206560) ON SEPTEMBER 16, 1987 AT 0800 HOURS, WITH THE REACTOR IN THE RUN MODE AT 93% POWER, IT WAS FOUND THAT THE SIX (6) MONTH FUNCTIONAL TEST OF THE REACTOR PROTECTION SYSTEM (RPS) MOTOR-GENERATOR (MG) SET ELECTRICAL PROTECTION ASSEMBLIES (EPAS) HAD EXCEEDED THE CRITICAL DATE. THIS SURVEILLANCE, DRESDEN TECHNICAL SURVEILLANCE (DTS) 500-2, IS REQUIRED BY TECH SPEC 4.1.A.3.A. THE ROOT CAUSE OF THE EVENT WAS DETERMINED TO BE PERSONNEL ERROR ON THE PART OF THE COGNIZANT TECH STAFF SYSTEMS ENGINEER. DTS 500-2 WAS PERFORMED UPON DISCOVERY OF THE VIOLATION. BECAUSE THE RPS SYSTEM FUNCTIONED SATISFACTORILY THE EVENT WAS DEEMED TO BE OF MINIMAL SAFETY SIGNIFICANCE. CORRECTIVE ACTION INCLUDED DISCUSSION OF THIS EVENT WITH THE COGNIZANT SYSTEMS ENGINEER AND THE MECHANICAL

AND ELECTRICAL SYSTEMS GROUP LEADERS, IN ORDER TO EMPHASIZE THEIR RESPONSIBILITY TO COMPLETE SURVEILLANCES IN A TIMELY MANNER. IN ADDITION, A CRITICAL DATE SURVEILLANCE LIST WILL BE ISSUED WEEKLY TO APPLICABLE DEPARTMENT HEADS BY THE SURVEILLANCE COORDINATOR. A LETTER DESCRIBING THIS EVENT AND EMPHASIZING THE RESPONSIBILITIES OF TECH STAFF ENGINEERS AND GROUP LEADERS WAS ALSO ISSUED TO ALL TECH STAFF PERSONNEL.

[87] DRESDEN 3 DOCKET 50-249 LER 87-013
MANUAL REACTOR SCRAM DUE TO REACTOR FEEDWATER SYSTEM OSCILLATIONS DURING UNIT SHUTDOWN DUE TO FAILURE OF AIR OPERATED CONTAINMENT ISOLATION VALVE AO-3-1601-63 TO CLOSE DURING SURVEILLANCE TESTING.
EVENT DATE: 080787 REPORT DATE: 090487 NSSS: GE TYPE: BWR
VENDOR: ASCO VALVES
MILLER FLUID POWER CO.
PRATT ENGINEERING

(NSIC 206300) ON AUGUST 7, 1987 AT 0808 HOURS WITH REACTOR POWER AT 30% THERMAL AND A UNIT SHUTDOWN IN PROGRESS, SIGNIFICANT OSCILLATIONS IN FEEDWATER AND CONDENSATE SYSTEM PRESSURES WERE OBSERVED. A 3A FEEDWATER REGULATING VALVE (FWRV) LOCKOUT WAS RECEIVED IN ADDITION TO HIGH VIBRATION ALARMS FOR THE REACTOR FEEDWATER PUMPS (RFP) AND FEEDWATER REGULATING STATION. THE UNIT SHUTDOWN WAS CAUSED BY FAILURE OF CONTAINMENT PURGE VALVE AOE-1601-63 TO STANDBY GAS TREATMENT SYSTEM. THE PIPING VIBRATIONS CAUSED AN INSTRUMENT LINE BREAK ON THE 3C RFP PIPING AND A DRAIN LINE BREAK IN THE REACTOR WATER CLEANUP (RWCU) SYSTEM. THE REACTOR WAS MANUALLY SCRAMMED TO ALLOW ISOLATION OF THE LEAKS. DURING COOLDOWN, A GROUP V PRIMARY CONTAINMENT ISOLATION OCCURRED WHICH RESULTED IN A REACTOR SCRAM ON LOW WATER LEVEL AT 1029 HOURS. THE PURGE VALLEY FAILURE WAS ATTRIBUTED FWRV AIR SUPPLY SOLENOID VALVE. FURTHER TESTING IS EXPECTED TO DETERMINE THE CAUSE OF THE FEEDWATER OSCILLATIONS AND GROUP V ISOLATION. THE PIPING AND AIR OPERATOR WERE REPAIRED AND THE SOLENOID VALVE WAS REPLACED. THE SAFETY SIGNIFICANCE WAS MINIMAL SINCE PURGING OF THE CONTAINMENT CAN BE ACCOMPLISHED BY VENTING TO THE REACTOR BUILDING VENTILATION SYSTEM AND THE EMERGENCY CORE COOLING SYSTEMS WERE AVAILABLE. PREVIOUS OCCURRENCE 050-249/87-12.

[88] FERMI 2 DOCKET 50-341 LER 87-041
ACCIDENT RANGE MONITOR IS INOPERABLE BECAUSE OF DELETED SYSTEM PARAMETERS.
EVENT DATE: 032987 REPORT DATE: 092687 NSSS: GE TYPE: BWR
VENDOR: EBERLINE INSTRUMENT CORP.

(NSIC 206460) A REVIEW OF THE STANDBY GAS TREATMENT SYSTEM (SGTS) DIVISION II ACCIDENT RANGE MONITOR (AXM) CHANNEL PARAMETERS WAS CONDUCTED. IT WAS REVEALED THAT THE CHANNELS PARAMETERS WERE SET AT DEFAULT VALUES FOR CALIBRATION CONSTANTS AND THE SET POINTS. THIS RESULTED IN THE MONITOR BEING INOPERABLE WHICH IS BEING REPORTED AS A VOLUNTARY LICENSEE EVENT REPORT. THE CAUSE OF THIS EVENT WAS INADEQUATE SYSTEM TURN OVER FOLLOWING PREOPERATIONAL TESTING. THIS RESULTED IN THE IMPLEMENTATION OF INADEQUATE SURVEILLANCE AND TEST PREVENTATIVE MAINTENANCE PROCEDURES. A DAILY CHANNEL CHECK IS BEING PERFORMED FOR THE SYSTEM PARTICULATE, IODINE AND NOBLE GAS (SPING) AND AXM UNITS. THE PRACTICE WILL BE CONTINUED UNTIL APPROPRIATE CHANGES ARE MADE TO THE PROCEDURES. THE SPING AND AXM PROCEDURES WILL BE CHANGED TO CONTROL AND VERIFY MONITOR PARAMETER EDITING. A TECHNICAL SPECIFICATION INTERPRETATION WAS MADE TO CLARIFY SGTS OPERABILITY VERSUS AXM RADIATION MONITOR AVAILABILITY.

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

(NSIC 206523) ON JULY 30, 1987, IT WAS DISCOVERED THAT FIRE BRIGADE PRACTICE SESSIONS WERE NOT BEING HELD AT INTERVALS OF ONE YEAR AS REQUIRED BY THE FIRE BRIGADE PROCEDURE. INSTEAD FIRE BRIGADE PRACTICE SESSIONS WERE BEING HELD ON A ONCE PER CALENDAR YEAR BASIS FROM APRIL TO OCTOBER. THE CAUSE OF THIS EVENT WAS A MISINTERPRETATION OF THE REQUIREMENTS FOR FIRE BRIGADE TRAINING SCHEDULE. TO PREVENT RECURRENCE, THE FIRE PROTECTION PROGRAM WILL BE REVISED TO INCLUDE A TWENTY FIVE PERCENT EXTENSION INTERVAL. THIS REVISION HAS BEEN SUBMITTED TO THE NUCLEAR REGULATORY COMMISSION FOR REVIEW AND APPROVAL.

[90] FERMI 2 DOCKET 50-341 LER 87-035
REACTOR SHUTDOWN FOR EXCESSIVE VALVE LEAKAGE AND SUBSEQUENT SCRAM DUE TO INTERMEDIATE RANGE MONITORS TRIPPING.
EVENT DATE: 073187 REPORT DATE: 083087 NSSS: GE TYPE: BWR
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 206293) ON JULY 31, 1987 AT 0220 HOURS, AN UNUSUAL EVENT WAS DECLARED DUE TO EXCESSIVE PRIMARY CONTAINMENT LEAKAGE THROUGH A CHECK VALVE, AND REACTOR SHUTDOWN WAS INITIATED. AS THE OPERATOR WAS DEPRESSURIZING THE REACTOR WHILE STILL AT POWER, AN INJECTION OF COLD FEEDWATER OCCURRED. THIS COLD FEEDWATER CAUSED AN INCREASE IN POSITIVE REACTIVITY. THE CHANGE IN REACTIVITY CAUSED AN INCREASE IN REACTOR POWER LEVEL WHICH RESULTED IN AN UPSCALE TRIP OF THE INTERMEDIATE RANGE MONITORS (IRMS). AS A RESULT, A REACTOR SCRAM OCCURRED AT 0311 HOURS. EXCESSIVE VALVE LEAKAGE WAS CAUSED BY RANDOM COMPONENT FAILURE. THE SUBSEQUENT SCRAM WAS DUE TO THE REACTOR POWER LEVEL INCREASE IN THE IRM RANGE DUE TO CHANGES IN FEEDWATER FLOW. THE CHECK VALVE WAS REPAIRED AND TESTED. THE OPERATING PROCEDURES WILL BE REVIEWED AND REVISED TO CLARIFY USE OF THE TURBINE BYPASS VALVES. A DISCUSSION OF THE LESSONS LEARNED ABOUT THE REACTOR POWER LEVEL SENSITIVITY TO CHANGES IN FEEDWATER FLOW WILL BE GIVEN TO THE OPERATORS. ADDITIONAL TRAINING ON DEPRESSURIZATION WITH THE BYPASS VALVES WILL BE GIVEN.

[91] FERMI 2 DOCKET 50-341 LER 87-036
REACTOR SCRAM DUE TO LOW REACTOR PRESSURE VESSEL WATER LEVEL AS A RESULT OF MISALIGNED RESIDUAL HEAT REMOVAL SYSTEM VALVES CAUSED BY OPERATOR ERROR.
EVENT DATE: 080287 REPORT DATE: 090187 NSSS: GE TYPE: BWR

(NSIC 206247) ON AUGUST 2, 1987, AT 1107 HOURS, A REACTOR SCRAM OCCURRED ON LOW REACTOR VESSEL WATER LEVEL. DURING THE PERFORMANCE OF A FILL AND VENT OPERATION OF THE RESIDUAL HEAT REMOVAL (RHR) SYSTEM DIVISION I, AN IMPROPER VALVE LINE UP CAUSED REACTOR WATER TO BE DIVERTED FROM THE REACTOR VESSEL TO THE SUPPRESSION POOL. ISOLATION OF RHR SHUTDOWN COOLING TERMINATED THE EVENT WHICH IS AN ENGINEERED SAFETY FEATURE. TO PREVENT RECURRANCE, PLANT PERSONNEL HAVE BEEN INSTRUCTED ON LESSONS LEARNED FROM THIS EVENT. PERSONNEL INVOLVED IN THIS EVENT WERE GIVEN APPROPRIATE LEVELS OF DISCIPLINE.

[92] FERMI 2 DOCKET 50-341 LER 87-037
INADVERTENT ACTUATION OF THE INBOARD MAIN STEAM ISOLATION VALVES DUE TO PROCEDURAL INADEQUACY.
EVENT DATE: 080487 REPORT DATE: 090387 NSSS: GE TYPE: BWR

(NSIC 206248) ON AUGUST 4, 1987 AT 0516 HOURS, AN INADVERTANT CLOSURE OF THE INBOARD MAIN STEAM ISOLATION VALVES (MSIVS) OCCURRED DURING A TRANSFER OF THE REACTOR PROTECTION SYSTEM (RPS) BUS POWER SOURCES. THIS CLOSURE WAS CAUSED BY AN INADEQUACY IN A PROCEDURE WHICH DID NOT INSTRUCT THE OPERATOR TO ENERGIZE THE INBOARD MSIVS' AC SOLENOIDS AFTER TRANSFERRING BUS POWER SOURCES. WHEN THE DC SOLENOIDS WERE DE-ENERGIZED DURING TRANSFER OF THE OTHER RPS BUSES THE CLOSURE OCCURRED. WHEN THE OPERATOR RESET THE ISOLATION LOGIC, THE INBOARD MSIVS OPENED SINCE THE OPERATOR DID NOT FIRST DEPRESS THE MANUAL "CLOSE" PUSH BUTTON. IN

ORDER TO PREVENT RECURRENCE OF THIS EVENT THE OPERATING PROCEDURE WAS REVISED TO REQUIRE ENERGIZATION OF AC SOLENOIDS AFTER EACH TRANSFER OF THEIR POWER SUPPLY.

[93] FERMI 2 DOCKET 50-341 LER 87-039
FAILURE OF THE BACK-UP MANUAL SCRAM BREAKER TO TRIP DUE TO POOR WORKMANSHIP DURING ON SITE ASSEMBLY.
EVENT DATE: 080787 REPORT DATE: 090687 NSSS: GE TYPE: BWR
VENDOR: I-T-E CIRCUIT BREAKER

(NSIC 206249) ON AUGUST 7, 1987 AT 2025 HOURS, A TRANSFER OF THE REACTOR PROTECTION SYSTEM BUS A FROM ITS ALTERNATE POWER SUPPLY TO ITS NORMAL POWER SUPPLY WAS PERFORMED. DURING THIS TRANSFER THE "A" BACK UP MANUAL SCRAM BREAKER FAILED TO TRIP. INVESTIGATION DETERMINED THAT THE CRADLE ACTUATOR BLADE ON THE UNDERVOLTAGE DEVICE WAS BINDING AGAINST THE BREAKER CASE. THIS PREVENTED THE TRIP LATCH BLADE FROM TRIPPING THE BREAKER. THE SLOT WHERE THE CRADLE BLADE ENTERS THE BREAKER CASING HAD NOT BEEN PROPERLY PREPARED TO PREVENT BINDING. THIS WAS CAUSED BY POOR WORKMANSHIP DURING FIELD ASSEMBLY OF THE BREAKER. BOTH THE "A" AND "B" BACK UP MANUAL SCRAM BREAKERS WERE REPLACED WITH BREAKERS THAT WERE PROPERLY PREPARED. TESTING WAS PERFORMED TO VERIFY THEIR OPERABILITY. CORRECTIVE ACTIONS INCLUDE REVISING THE VENDOR'S TECHNICAL MANUAL TO INCLUDE A CAUTION ABOUT PROPERLY PREPARING THE SLOT IN THE BREAKER CASE.

[94] FERMI 2 DOCKET 50-341 LER 87-038
INITIATION OF DIVISION I EMERGENCY EQUIPMENT COOLING WATER DUE TO INADEQUATE SYSTEM DESIGN FOR PRESSURE COMPENSATION.
EVENT DATE: 081587 REPORT DATE: 091487 NSSS: GE TYPE: BWR

(NSIC 206418) ON AUGUST 15, 1987 AT 1921 HOURS, DURING POST MAINTENANCE TESTING OF THE RETURN HEADER ISOLATION VALVE FOR DIVISION II EMERGENCY EQUIPMENT COOLING WATER (EECW), AN AUTOMATIC INITIATION OF DIVISION I EECW OCCURRED. DIVISION I INITIATED BECAUSE THE DIVISION II EECW MAKEUP TANK WAS SUPPLYING THE REACTOR BUILDING CLOSED COOLING WATER (RBCCW) PUMP SUCTION HEAD. CLOSING THE ISOLATION VALVE ISOLATED THE MAKEUP TANK THAT WAS IN SERVICE. THIS TRIPPED THE RBCCW PUMP AND CAUSED DIVISION I EECW TO AUTOMATICALLY INITIATE. THE CONTROL ROOM OPERATOR RE-OPENED THE ISOLATION VALVE AND RESTORED RBCCW. THE CAUSE OF THIS EVENT IS A RBCCW/EECW SYSTEM DESIGN THAT DID NOT RESPOND QUICKLY ENOUGH TO SYSTEM PRESSURE CHANGES. TO PREVENT RECURRENCE, MOTOR OPERATED VALVES WERE INSTALLED TO THE DIVISION I AND II EECW MAKEUP TANKS. A POST MODIFICATION TEST WILL BE DONE PRIOR TO PLANT RESTART TO DEMONSTRATE THE EFFECTIVENESS OF THESE MOTOR OPERATED VALVES. THE APPROPRIATE PROCEDURES WERE REVISED TO MAINTAIN A MAKE UP TANK IN SERVICE WHEN OPERATING THE SYSTEM. A DESCRIPTION OF THIS EVENT WILL BE ISSUED AS REQUIRED READING TO CONTROL ROOM PERSONNEL.

[95] FERMI 2 DOCKET 50-341 LER 87-046
DEFICIENCIES IN RAYCHEM INSTALLATIONS POTENTIALLY IMPACT ACCIDENT SCENARIOS.
EVENT DATE: 082087 REPORT DATE: 092087 NSSS: GE TYPE: BWR
VENDOR: RAYCHEM CORP.

(NSIC 206067) IN RESPONSE TO INFORMATION NOTICE 86-53, "IMPROPER INSTALLATION OF HEAT SHRINKABLE TUBING", A WALKDOWN OF A SAMPLE OF RAYCHEM HEAT SHRINK INSTALLATIONS WAS PERFORMED. THE WALKDOWN IDENTIFIED DEFICIENCIES ON SEVEN COMPONENTS BASED UPON THEIR TERMINATIONS' CONFIGURATIONS. THE POTENTIAL FOR DEGRADED ENVIRONMENTAL QUALIFICATION OF THE ASSOCIATED SAFETY RELATED EQUIPMENT WAS EVALUATED. ALL OF THE DEFICIENT INSTALLATIONS WERE REPAIRED AND/OR REPLACED. THIS CONDITION WAS CAUSED BY INADEQUACIES IN THE TRAINING OF RAYCHEM INSTALLERS, LACK OF INSPECTION REQUIREMENTS AND PREVIOUS SPECIFICATION DEFICIENCIES. THE TRAINING PROGRAM WILL BE UPGRADED TO PROVIDE ADEQUATE INSTRUCTION AND INSPECTION REQUIREMENTS IN THE FUTURE.

[96] FERMI 2 DOCKET 50-341 LER 87-040 REV 01
UPDATE ON TRIP OF THE REACTOR PROTECTION SYSTEM MOTOR-GENERATOR SET FROM A POSTULATED OVERVOLTAGE CONDITION.
EVENT DATE: 082787 REPORT DATE: 102987 NSSS: GE TYPE: BWR

(NSIC 206582) ON AUGUST 27, 1987 AT 0028 HOURS, THE REACTOR PROTECTION SYSTEM (RPS) MOTOR-GENERATOR (MG) SET A TRIPPED. THIS RESULTED IN A LOSS OF POWER TO RPS BUS A AND A DIVISION I ISOLATION SIGNAL. THE MAIN STEAM ISOLATION VALVES AND MAIN STEAM LINE DRAIN VALVES CLOSED SINCE A SURVEILLANCE UNDERWAY ON DIVISION II ISOLATION LOGIC HAD THE CHANNEL D LOGIC TRIPPED AT THE TIME. FURTHER TESTING HAS NOT PROVIDED A CONCLUSIVE CAUSE FOR THIS EVENT. A POSSIBLE CAUSE IS THE PRESENCE OF CONTAMINANTS IN THE POTENTIOMETER FOR THE VOLTAGE REGULATOR. PREVENTATIVE MAINTENANCE WILL BE PERFORMED TO VERIFY THAT THE VOLTAGE REGULATOR'S POTENTIOMETER IS CLEANED REGULARLY. WHILE TESTING WAS BEING CONDUCTED, THE BREAKER WAS SHORTED TO GROUND WHICH CAUSED ANOTHER LOSS OF POWER TO RPS BUS A ON SEPTEMBER 29 AT 1800 HOURS.

[97] FERMI 2 DOCKET 50-341 LER 87-043 REV 01
UPDATE ON CONTROL CENTER HEATING VENTILATION AND AIR CONDITIONING SYSTEM ACTUATES TO RECIRCULATION MODE.
EVENT DATE: 090287 REPORT DATE: 110387 NSSS: GE TYPE: BWR

(NSIC 206576) ON SEPTEMBER 2, 1987 AT 1442 HOURS, THE CONTROL CENTER HEATING, VENTILATING AND AIR CONDITIONING DIVISION I SHIFTED FROM NORMAL TO RECIRCULATION MODE. ALSO THE DIVISION I STANDBY GAS TREATMENT SYSTEM AUTOMATICALLY INITIATED. PRIOR TO THESE ACTUATIONS, THE SEQUENCE OF EVENTS RECORDER INDICATED THAT EITHER THE CHANNEL A OR C FUEL POOL EXHAUST VENTILATION RADIATION MONITOR HAD TRIPPED DOWNSCALE. A REVIEW OF THE RECORDER STRIP CHARTS FOR THE MONITORS SHOWS NO DOWNSCALE TRIP ACTUALLY OCCURRED. THE INVESTIGATION IS CONTINUING TO DETERMINE THE CAUSE OF THE EVENT. ANY CORRECTIVE ACTIONS NECESSARY TO BE TAKEN WILL BE BASED UPON THE RESULTS OF THE INVESTIGATION.

[98] FITZPATRICK DOCKET 50-333 LER 87-004 REV 01
UPDATE ON MAIN STEAM SAFETY VALVES FOUND OUT OF TOLERANCE DURING TESTING.
EVENT DATE: 020487 REPORT DATE: 091887 NSSS: GE TYPE: BWR
VENDOR: TARGET ROCK CORP.

(NSIC 206064) DURING THE 1987 SCHEDULED REFUEL OUTAGE, 6 TARGET ROCK SAFETY/RELIEF VALVES (EIIS CODE AD) WERE REMOVED FOR TESTING IN ACCORDANCE WITH TECHNICAL SPECIFICATION 2.2.1B. THREE VALVES HAD SETPOINT VALUES OUTSIDE THE ALLOWABLE TOLERANCE. MANUAL AND AUTOMATIC DEPRESSURIZATION SYSTEM (ADS) FUNCTIONS ARE NOT AFFECTED BY SETPOINT DRIFT. IN ADDITION, 1 VALVE FAILED A PILOT DISC-TO-SEAT STICKING TEST. AS A RESULT, THE MANUAL OR ADS MODE OF THAT VALVE MAY HAVE BEEN AFFECTED. ALL THE FAILED VALVES HAVE BEEN OVERHAULED AND RETESTED. CORRECTIVE ACTIONS INCLUDE: PARTICIPATION IN THE BWR OWNERS' GROUP TO RESOLVE SETPOINT DRIFT PROBLEMS, FREQUENCY OF TESTING WILL BE INCREASED TO TEST ALL VALVES EACH CYCLE UNTIL SETPOINT DRIFT PROBLEM IS RESOLVED, AND A CHANGE TO THE FITZPATRICK TECHNICAL SPECIFICATIONS WILL BE PURSUED TO EXPAND ALLOWED SETPOINT TOLERANCE. LER 85-009 AND 85-013 ARE SIMILAR EVENTS.

[99] FT. CALHOUN 1 DOCKET 50-285 LER 87-022
DEFECTS IN EFW STORAGE TANK MANUFACTURED BY EATON METAL PRODUCTS CORPORATION.
EVENT DATE: 041587 REPORT DATE: 092587 NSSS: CE TYPE: PWR
VENDOR: EATON METAL PRODUCTS CO.

(NSIC 206453) DURING THE 1987 REFUELING OUTAGE, THE INTERIOR OF THE EMERGENCY FEEDWATER STORAGE TANK (EFWST) WAS RECOATED. THE EFWST IS A VESSEL OF APPROX. 60,000 GAL CAPACITY WITH A NOMINAL THICKNESS OF 1 INCH, AND DESIGN PRESSURE OF 50

PSIG. THE TANK WAS HYDROSTATICALLY TESTED TO 75 PSIG FOLLOWING FABRICATION. THE WELD CONTOURS ON THE INTERIOR WERE TOO ROUGH TO ENSURE GOOD COATING ADHESION, SO OPPD GROUND THE WELD FLUSH. MAGNETIC PARTICLE EXAMINATIONS DETECTED SURFACE CRACKING AND RADIOGRAPHS REVEALED EXTENSIVE SUB-SURFACE DEFECTS, INCLUDING SLAG INCLUSIONS AND LACK OF FUSION. THE PROBABLE CAUSES OF THE DEFECTS ARE; USE OF SHIELD ARC 85 ROD WITHOUT PREHEAT RESULTED IN SHRINKAGE; IMPROPER CLEANING OF COPPER DEPOSITS CAUSED LACK OF FUSION; AND POOR WORKMANSHIP. A METALLURGICAL ANALYSIS CONFIRMED THE MATERIALS OF CONSTRUCTION AND SUPPORTED THE PROBABLE CAUSES OF FAILURE, IT ALSO SHOWED THAT THE CRACKS WERE BRITTLE FRACTURES, WITH NO SIGNS OF FATIGUE, INDICATING THAT THE DEFECTS WERE NOT SERVICE INDUCED. A REPAIR PROCEDURE WAS DEVELOPED IN ACCORDANCE WITH THE REQUIREMENTS OF ASME AND WAS APPROVED BY AN AUTHORIZED NUCLEAR INSPECTOR. REPAIRS HAVE BEEN COMPLETED AND THE TANK HAS BEEN RETURNED TO SERVICE. OTHER TANKS BUILT BY THE SAME MANUFACTURER WERE REVIEWED AND IT WAS CONCLUDED THAT CATASTROPHIC FAILURES ARE UNLIKELY. DOCUMENTATION OF THE REVIEW IS IN PROGRESS AND A LONG TERM PLAN TO EVALUATE OTHER TANKS IS BEING FORMULATED.

[100] FT. CALHOUN 1 DOCKET 50-285 LER 87-024
VIAS ACTUATION DURING CALIBRATION OF RADIATION MONITOR DUE TO DEFICIENT PROCEDURES.
EVENT DATE: 081387 REPORT DATE: 091187 NSSS: CE TYPE: PWR

(NSIC 206047) AT 100% POWER, AN UNPLANNED ACTUATION OF THE VENTILATION ISOLATION ACTUATION SYSTEM (VIAS) OCCURRED AT 0826 HOURS (CDT) ON AUGUST 13, 1987, AT THE FORT CALHOUN NUCLEAR POWER STATION. WHILE CALIBRATING RADIATION MONITOR RM-062, AN INSTRUMENT AND CONTROL TECHNICIAN CONNECTED TEST EQUIPMENT TO THE RADIATION MONITOR TO TEST ITS HIGH ALARM. THE TEST EQUIPMENT WAS NOT RESET (THIS WAS NOT PROCEDURALLY REQUIRED), SO WHEN THE TECHNICIAN ENERGIZED THE TEST EQUIPMENT, AN IMMEDIATE SIGNAL WAS INTRODUCED WHICH TRIPPED VIAS INADVERTENTLY. ALL PLANT SYSTEMS INVOLVED IN THIS INCIDENT OPERATED WITHIN THEIR DESIGN BASIS WITH NO EQUIPMENT DAMAGE OR FAILURES. THE VIAS ENGINEERED SAFEGUARDS WERE IMMEDIATELY RESET. NO VIOLATION OF TECHNICAL SPECIFICATIONS OR OPERATOR ERRORS OCCURRED. TO PREVENT FUTURE UNPLANNED VIAS ACTUATIONS OF THIS NATURE, AN ADDITIONAL STEP WILL BE ADDED TO CALIBRATION PROCEDURES CP-050, CP-051, CP-060, CP-061 AND CP-062, "ELECTRONIC CALIBRATION PROCEDURE", PRE-CALIBRATION SETUP, TO ENSURE THAT THE TEST EQUIPMENT IS RESET PRIOR TO CONNECTING IT TO A RADIATION MONITOR TO PREVENT A HIGH SIGNAL FROM GENERATING A VIAS DURING EQUIPMENT SETUP.

[101] FT. ST. VRAIN DOCKET 50-267 LER 87-022
LOOP I SHUTDOWN DURING SURVEILLANCE TESTING DUE TO COMPONENT FAILURE.
EVENT DATE: 092987 REPORT DATE: 102987 NSSS: GA TYPE: HTGR

(NSIC 206632) CAUSE - FAILURE OF LOGIC CHIP. ON 9/29/87, AT 1102 HOURS, WITH THE REACTOR SHUTDOWN, LOOP I AND LOOP II STEAM GENERATORS SUPPLIED BY EMERGENCY CONDENSATE, AND ALL FOUR HELIUM CIRCULATORS WARMING ON STEAM, A LOOP I SHUTDOWN ACTUATION WAS INITIATED BY THE PLANT PROTECTIVE SYSTEM (PPS). THE ACTUATION OCCURRED DURING RESULTS SURVEILLANCE TESTING OF THE LOOP I STEAM GENERATOR PENETRATION HIGH PRESSURE TRIP SWITCHES PER TECH SPEC SURVEILLANCE SR 5.4.1.2.5.A-M/1.2.5.C-R. THE CAUSE OF THE LOOP I SHUTDOWN WAS FAILURE OF A Z40 LOGIC CHIP IN PPS MODULE CS-1A1. THE FAILURE OF THE Z40 LOGIC CHIP PLACED ONE CHANNEL OF THE LOGIC CIRCUIT IN A TRIPPED CONDITION. WHEN THE I&C TECHNICIAN TRIPPED A PRESSURE SWITCH AS PART OF THE SURVEILLANCE, THE TWO OUT OF THREE SIGNAL LOGIC NECESSARY TO ACTUATE THE PPS LOOP SHUTDOWN TRIP CIRCUITRY WAS COMPLETED. THE LOOP SHUTDOWN ISOLATED SECONDARY COOLANT FLOW TO THE LOOP I STEAM GENERATOR AND INITIATED TRIPS OF "A" AND "B" HELIUM CIRCULATORS. THE PPS MODULE CONTAINING THE FAILED Z40 LOGIC CHIP WAS REPLACED WITH AN OPERABLE SPARE MODULE AND THE SURVEILLANCE WAS SUCCESSFULLY COMPLETED. THE FAULTY PPS MODULE WAS SUBSEQUENTLY REPAIRED BY REPLACING THE FAILED Z40 LOGIC CHIP.

[102] GRAND GULF 1 DOCKET 50-416 LER 87-005 REV 02
UPDATE ON THREE MAIN STEAM LINE RADIATION MONITORS EXCEED TECHNICAL SPECIFICATION TRIP LIMIT.
EVENT DATE: 041587 REPORT DATE: 093087 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 206577) ON APRIL 14, 1987, MAIN STEAM LINE (MSL) RADIATION MONITOR K610B WAS FOUND WITH A TRIP SETPOINT THAT EXCEEDED THE ALLOWABLE LIMIT OF TECHNICAL SPECIFICATION 3.3.2-2.2.B. ON APRIL 15, MSL RADIATION MONITORS K610C AND K610D WERE FOUND WITH SETPOINTS THAT ALSO EXCEEDED THE ALLOWABLE LIMIT. TECHNICAL SPECIFICATION 3.3.2-2.2.B REQUIRES THE MSL HIGH RADIATION TRIP SETPOINTS TO BE SET LESS THAN 3 TIMES FULL POWER BACKGROUND WITH AN ALLOWABLE VALUE OF 3.6 TIMES FULL POWER BACKGROUND. THE INSTRUMENTS PROVIDE A REACTOR SCRAM SIGNAL, A MECHANICAL VACUUM PUMP TRIP SIGNAL, AND AN MSL ISOLATION SIGNAL IF THE HIGH RADIATION TRIP SETPOINT IS REACHED ON EITHER THE A OR C CHANNEL AND CONCURRENTLY ON EITHER THE B OR D CHANNEL. AN OUTBOARD MSL DRAIN LINE ISOLATION AND AN OUTBOARD REACTOR SAMPLE LINE ISOLATION OCCUR IF THE HIGH RADIATION TRIP SETPOINT IS REACHED ON THE A AND D CHANNELS. AN INBOARD MSL DRAIN LINE ISOLATION AND AN INBOARD REACTOR SAMPLE LINE ISOLATION OCCUR IF THE HIGH RADIATION TRIP SETPOINT IS REACHED ON THE B AND C CHANNELS. AN INVESTIGATION BY SERI HAS DETERMINED THAT INSTRUMENT REPLACEMENT WILL CORRECT THE DRIFT CONCERNS. THE SERI DESIGN ENGINEERING GROUP HAS EVALUATED THE SETPOINT DRIFT CONCERNS AND DETERMINED THAT INCREASED MONITORING OF THE INSTRUMENTS IS SUFFICIENT.

[103] GRAND GULF 1 DOCKET 50-416 LER 87-011
REDUNDANT ISOLATION VALVES SHARE COMMON POWER SOURCE DUE TO DESIGN ERROR.
EVENT DATE: 072987 REPORT DATE: 082887 NSSS: GE TYPE: BWR

(NSIC 206273) ON JULY 29, 1987, SERI ENGINEERS DISCOVERED THAT TWO MOTOR OPERATED CONTAINMENT ISOLATION VALVES USED IN SERIES AT CONTAINMENT PENETRATION 87 SHARED THE SAME DIVISIONAL POWER SOURCE. THIS IS CONTRARY TO UFSAR SECTION 6.2.4.1.H WHICH STATES THAT POWER OPERATED VALVES USED IN SERIES ARE TO BE DESIGNED SUCH THAT NO SINGLE EVENT CAN INTERRUPT MOTIVE POWER TO BOTH CLOSURE DEVICES. THE REACTOR WATER CLEANUP SYSTEM (RWCU) WAS BEING OPERATED IN THE "POST- PUMP" MODE AT THE TIME OF DISCOVERY WHICH REQUIRES BOTH ISOLATION VALVES TO BE OPEN. THE POST-PUMP MODE IS NORMALLY USED DURING REACTOR OPERATION. IN THIS MODE REACTOR WATER IS COOLED PRIOR TO ENTERING THE RWCU PUMP TO EXTEND PUMP SEAL LIFE. THE RWCU PUMPS WERE SECURED AFTER THE DISCOVERY AND THE INBOARD ISOLATION VALVE WAS CLOSED AND DEACTIVATED IN ACCORDANCE WITH THE TECHNICAL SPECIFICATION LCO. THE RWCU SYSTEM IS NOW BEING OPERATED IN THE "PRE-PUMP" MODE WHERE THE REACTOR WATER IS DRAWN DIRECTLY TO THE RWCU PUMPS PRIOR TO ENTERING THE HEAT EXCHANGERS. SERI MONITORED RWCU CONTAINMENT PENETRATION TEMPERATURES AND RWCU PERFORMANCE WHILE OPERATING IN THIS MODE. ALL MONITORED PARAMETERS WERE FOUND ACCEPTABLE. LONG TERM CORRECTIVE ACTION WILL BE TO CHANGE THE DIVISIONAL POWER SUPPLY FOR THE INBOARD ISOLATION VALVE.

[104] GRAND GULF 1 DOCKET 50-416 LER 87-014
NEW FUEL CONTAINER DROPPED FROM TRANSFER CART DUE TO PERSONNEL ERROR.
EVENT DATE: 073187 REPORT DATE: 083187 NSSS: GE TYPE: BWR

(NSIC 206091) ON JULY 31, 1987 A CONTAINER OF TWO NEW FUEL BUNDLES TIPPED OFF A TRANSFER CART AND FELL APPROXIMATELY 2 TO 2.5 FEET TO THE TURBINE DECK. THIS OCCURRED AS THE CRANE OPERATOR WAS RAISING THE DISCONNECTED SLINGS AWAY FROM THE CONTAINER. AT LEAST ONE OF THE SLINGS HOOKED THE UNDERSIDE OF THE TRANSFER CART RAISING ONE SIDE OF IT AND FLIPPING IT OVER. THE FUEL HANDLING CREW IMMEDIATELY NOTIFIED THEIR SUPERVISOR. THE HEALTH PHYSICIST WAS ON THE SCENE AND CONTINUOUSLY MONITORED THE CONTAINER. NO INCREASE IN DOSE RATE OR CONTAMINATION ABOVE PREVIOUSLY RECORDED DATA WAS DETECTED. THE CAUSE OF THE EVENT WAS DETERMINED TO BE INADEQUATE ATTENTION BY THE CRANE OPERATOR AND THE SIGNALMAN.

THE CRANE OPERATOR AND SIGNALMAN WERE REMOVED FROM THE FUEL HANDLING CREW AND THEIR CERTIFICATIONS REMOVED. IN ADDITION, DISCIPLINARY ACTION WAS TAKEN AGAINST THE CRANE OPERATOR AND SIGNALMAN. THIS EVENT WAS DISCUSSED WITH PERSONNEL ASSIGNED TO FUEL HANDLING ACTIVITIES. AS AN ADDITIONAL PRECAUTIONARY MEASURE, A SUPERVISOR WAS PLACED ON THE TURBINE DECK FOR THE REMAINDER OF THE FUEL RECEIPT. THE FUEL BUNDLES WERE INSPECTED AT THE FUEL VENDOR'S FACILITY. BOTH FUEL BUNDLES EXHIBITED SOME EXTERNAL DAMAGE DUE TO DEFORMATION OF THE SPACER GRIDS. CERAMIC FUEL PELLETS WERE INSPECTED FROM SELECTED FUEL RODS. SOME PELLET CRACKING WAS OBSERVED. DUE TO THE DAMAGE IDENTIFIED, THE TWO BUNDLES WILL BE REPLACED.

[105] GRAND GULF 1 DOCKET 50-416 LER 87-012
REACTOR SCRAM DUE TO TURBINE CONTROL VALVE FAST CLOSURE.
EVENT DATE: 080687 REPORT DATE: 090387 NSSS: GE TYPE: BWR

(NSIC 206274) ON AUGUST 6, 1987 AT APPROXIMATELY 0625, THE REACTOR SCRAMMED ON A TURBINE CONTROL VALVE FAST CLOSURE SIGNAL. THE EVENT WAS INITIATED BY A BACKUP PROTECTION SCHEME RELAY LOCATED IN THE SWITCHYARD CONTROL HOUSE. THE RELAY IS DESIGNED TO OPEN THE GENERATOR OUTPUT BREAKERS ONLY WHEN THE GENERATOR IS OFF-LINE. AN INVESTIGATION REVEALED THAT MOISTURE HAD COLLECTED IN THE TERMINAL CABINETS AND APPARENTLY SHORTED AN INTERLOCK SWITCH WHICH CAUSED THE RELAY TO ACTUATE AND TRIP THE TWO MAIN GENERATOR OUTPUT CIRCUIT BREAKERS. THIS RESULTED IN A TURBINE CONTROL VALVE FAST CLOSURE SCRAM INITIATION. THE SAFETY RELIEF VALVE LOW-LOW SET LOGIC FUNCTIONED PROPERLY TO CONTROL REACTOR PRESSURE, LIFTING VALVES B21-F051D, F051B, F047G, F051A, F047D, AND F051F. REACTOR PRESSURE REACHED A MAXIMUM OF 1107 PSIG. SINCE THE BACKUP PROTECTION SCHEME RELAY IS USED ONLY WHEN THE GENERATOR IS OFF-LINE, IT WAS REMOVED FROM SERVICE AND WILL REMAIN OUT OF SERVICE UNTIL THE NEXT REFUELING OUTAGE. AT THAT TIME IT WILL BE THOROUGHLY CHECKED AND ANY PROBLEMS WILL BE CORRECTED AS REQUIRED. PLANT RESTART WAS DELAYED DUE TO UNRELATED PROBLEMS DESCRIBED IN LER 87-013-0. THE PLANT REACHED CRITICALITY ON AUGUST 9, 1987, AT APPROXIMATELY 1324. SYNCHRONIZATION OCCURRED ON AUGUST 10, 1987, AT APPROXIMATELY 1225.

[106] GRAND GULF 1 DOCKET 50-416 LER 87-013 REV 01
UPDATE ON VENTILATION DUCT SECTIONS NOT DESIGNED TO WITHSTAND A DESIGN BASIS TORNADO CAUSED BY INADEQUATE DUCTWORK STIFFENERS.
EVENT DATE: 080787 REPORT DATE: 101687 NSSS: GE TYPE: BWR

(NSIC 206524) ON AUGUST 7, 1987, SERI ENGINEERS DETERMINED THAT THE HVAC DUCTWORK SECTIONS BETWEEN THREE TORNADO BACK DRAFT DAMPERS AND THE CONTROL BUILDING PERIMETER WALL APPARENTLY WERE NOT DESIGNED TO WITHSTAND THE PRESSURE DIFFERENTIALS THAT COULD BE EXPERIENCED DUE TO THE EFFECTS OF A DESIGN BASIS TORNADO. THIS FINDING WAS THE RESULT OF A SPECIAL EVALUATION PERFORMED ON 23 DUCTWORK SECTIONS AT WALL PENETRATIONS THAT REQUIRED PRESSURIZATION OR DEPRESSURIZATION PROTECTION. THE SPECIAL EVALUATION WAS INITIATED WHEN ENGINEERS WERE UNABLE TO LOCATE DOCUMENTATION TO SUPPORT THE ACCEPTABILITY OF CERTAIN DUCTWORK SECTIONS TO MEET TORNADO LOADING OR HIGH ENERGY LINE BREAK LOADING CONCERNS. THE PLANT WAS IN HOT SHUTDOWN AT THE TIME OF THE FINDING. SERI DELAYED PLANT STARTUP UNTIL THE ISSUE WAS RESOLVED. ENGINEERS DETERMINED THAT ALL BUT THREE OF THE DUCTWORK SECTIONS WERE ACCEPTABLE. THESE THREE SECTIONS WERE MADE ACCEPTABLE BY INSTALLING INTERNAL DUCTWORK STIFFENERS. THE MODIFICATIONS WERE COMPLETED ALLOWING THE PLANT TO RESTART ON AUGUST 9.

[107] HATCH 1 DOCKET 50-321 LER 87-013
FEEDWATER CONTROLLER FAILS CAUSING FEEDWATER DECREASE RESULTING IN REACTOR SCRAM.
EVENT DATE: 080387 REPORT DATE: 090287 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: HATCH 2 (BWR)
VENDOR: GENERAL ELECTRIC CO.

(NSIC 206245) ON 8/3/87 AT APPROXIMATELY 2007 CDT, UNIT 1 WAS IN THE RUN MODE AT AN APPROXIMATE POWER LEVEL OF 2433 MWT (APPROXIMATELY 100 PERCENT OF RATED THERMAL POWER). AT THAT TIME, THE MASTER FEEDWATER CONTROLLER CIRCUITRY (EIIS CODE JK) FAILED. THIS FAILURE CAUSED A DECREASE IN THE REACTOR FEEDPUMPS (RFPS EIIS CODE SJ) FLOW AND A DECREASE IN REACTOR WATER LEVEL. THE REACTOR WATER LEVEL DECREASED TO THE REACTOR PROTECTION SYSTEM (RPS EIIS CODE JC) ACTUATION SETPOINT AND A REACTOR SCRAM OCCURRED. THE ROOT CAUSE OF THIS EVENT IS EQUIPMENT FAILURE. SPECIFICALLY, TWO CAPACITORS IN THE MASTER FEEDWATER CONTROLLER AMPLIFIER SHORT CIRCUITED. WHEN THESE TWO CAPACITORS SHORTED, THE VOLTAGE OUTPUT SIGNAL TO THE INDIVIDUAL FEEDPUMP CONTROLLERS WAS LOST. THIS CAUSED THE FEEDPUMPS TO RAMP DOWN TO THEIR LOW SPEED STOPS. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDED: 1) REPLACING THE CAPACITORS AND RETURNING THE CIRCUIT BOARD TO SERVICE, AND 2) INVESTIGATING SIMILAR PROBLEMS ON UNIT 2 AND REPLACING COMPONENTS.

[108] HATCH 1 DOCKET 50-321 LER 87-014
EQUIPMENT FAILURE AND INSTRUMENT DRIFT CAUSE MONITOR ACTIVATION AND ESF ACTUATION.
EVENT DATE: 080887 REPORT DATE: 090887 NSSS: GE TYPE: BWR
VENDOR: AMPEREX ELECTRONIC CORP.
GENERAL ELECTRIC CO.

(NSIC 206058) ON 8/8/87, 8/10/87, AND 8/13/87, UNIT 1 WAS IN THE RUN MODE AT AN APPROXIMATE POWER LEVEL OF 2436 MWT (APPROXIMATELY 100 PERCENT OF RATED THERMAL POWER). AT THESE TIMES, THE MAIN CONTROL ROOM ENVIRONMENTAL CONTROL (MCREC EIIS CODE VI) SYSTEM WENT INTO THE PRESSURIZATION MODE OF OPERATION. MCREC ENTERED THIS MODE ON A RADIATION SIGNAL FROM A REFUELING FLOOR AREA RADIATION MONITOR (ARM EIIS CODE IL). THE ROOT CAUSE OF THE FIRST TWO EVENTS IS EQUIPMENT FAILURE: THE GEIGER-MUELLER (GM) TUBE'S OUTPUT SIGNAL BECAME ERRATIC DUE TO LEAKAGE OF THE IONIZING GAS. THE CAUSE OF THE THIRD EVENT IS INSTRUMENT DRIFT IN THE GM TUBE OUTPUT. CORRECTIVE ACTIONS FOR THESE EVENTS INCLUDED: 1) REPLACING THE FAILED GM TUBE; VERIFYING CORRECT INSTRUMENT SETPOINTS AND; OBSERVING THE INSTRUMENT FOR FIVE HOURS, 2) CALIBRATING THE ARM AND CHECKING WIRING AND CABLE CONNECTIONS, 3) RE-CALIBRATING AND INSTRUMENTING THE ARM AND MONITORING THE INSTRUMENTATION, 4) CONTACTING THE GM MANUFACTURER FOR ADDITIONAL TESTING INFORMATION, AND 5) DETERMINING THE FEASIBILITY OF RAISING THE CURRENT ARM SETPOINTS.

[109] HATCH 1 DOCKET 50-321 LER 87-015
EQUIPMENT FAILURE CAUSES MONITOR ACTIVATION RESULTING IN ESF ACTUATION.
EVENT DATE: 082787 REPORT DATE: 092887 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 206492) ON 8/27/87 AND 8/28/87, UNIT 1 WAS IN THE RUN MODE AT AN APPROXIMATE POWER OF 2436 MWT (APPROXIMATELY 100 PERCENT OF RATED THERMAL POWER). AT THESE TIMES, THE MAIN CONTROL ROOM ENVIRONMENTAL CONTROL (MCREC EIIS CODE VI) SYSTEM LOGIC ACTUATED. THE MCREC LOGIC ACTUATED ON A RADIATION SIGNAL FROM A REFUELING FLOOR AREA RADIATION MONITOR (ARM EIIS CODE IL) ON 8/27/87, AS A RESULT OF THE LOGIC ACTUATION, AN ACTUAL PRESSURIZATION OCCURRED. THE MCREC SYSTEM WAS LEFT IN THE PRESSURIZATION MODE AND WAS IN THIS CONDITION WHEN THE 8/28/87 EVENT OCCURRED. THE ROOT CAUSE FOR THESE EVENTS IS EQUIPMENT FAILURE: THE GEIGER-MUELLER (GM) TUBE IN THE ARM FAILED. THE FAILURE IS ATTRIBUTED TO MIS-HANDLING DURING SHIPMENT OR STORAGE. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDED: 1) CHECKING THE ARM LOOP AND PERFORMING CALIBRATIONS, 2) REPLACING THE FAILED GM TUBE, 3) INSTALLING A RECORDER AND MONITORING INSTRUMENT RESPONSE, 4) RETURNING THE MCREC SYSTEM TO ITS NORMAL MODE OF OPERATION, 5) PERFORMING ENGINEERING INVESTIGATIONS, AND 6) DETERMINING THE FEASIBILITY OF RAISING THE CURRENT ARM SETPOINTS.

[110] HATCH 2 DOCKET 50-366 LER 87-009
CIRCUIT BREAKER FAILS CAUSING POWER FAILURE RESULTING IN REACTOR SCRAM.
EVENT DATE: 080387 REPORT DATE: 090287 NSSS: GE TYPE: BWR
VENDOR: HEINEMANN ELECTRIC CO.

(NSIC 206260) ON 8/3/87 AT APPROXIMATELY 1152 CDT, UNIT 2 WAS IN THE RUN MODE AT AN APPROXIMATE POWER LEVEL OF 2193 MWT (APPROXIMATELY 90 PERCENT OF RATED THERMAL POWER). AT THAT TIME, VITAL AC (EIIS CODE EE) POWER WAS LOST. THIS RESULTED IN A DECREASE IN THE REACTOR FEEDWATER PUMPS FLOW AND A DECREASE IN REACTOR WATER LEVEL. THE REACTOR WATER LEVEL DECREASED TO THE REACTOR PROTECTION SYSTEM (RPS EIIS CODE JC) ACTUATION SETPOINT AND A REACTOR SCRAM OCCURRED. THE ROOT CAUSE OF THIS EVENT IS ELECTRICAL EQUIPMENT FAILURE. SPECIFICALLY, CIRCUIT BREAKER CB-4 WOULD OPEN UNDER UNDULY LOW FORCE CONDITIONS. IT WAS CONCLUDED AFTER FIELD TESTING AND CONSULTATION WITH THE MANUFACTURER THAT THE TRIPPING MECHANISM WAS WEAK. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDED: 1) INSTALLING JUMPERS AND REMOVING EQUIPMENT FROM SERVICE, 2) DESIGNING AND INSTALLING BARRIER BOXES, 3) VERIFYING TRIP INSTRUMENTATION AND LEVEL TRANSMITTERS IN CALIBRATION, 4) VENTING INSTRUMENT LINES AND TRANSMITTERS, 5) PERFORMING EVALUATIONS TO PRECLUDE AIR POCKETS AND SPIKING IN INSTRUMENT LINES, AND 6) VERIFYING CERTAIN OTHER SYSTEMS DO NOT HAVE LOW SUCTION TRIPS. THE RESULTS OF THE THE EVALUATIONS WILL BE PRESENTED IN AN UPDATE TO THIS LER THAT IS SCHEDULED FOR SUBMISSION BY APPROXIMATELY 12/9/87.

[111] HATCH 2 DOCKET 50-366 LER 87-010
FAILED INSTRUMENT LINE LEAKAGE EXCEEDS ALLOWABLE LIMITS RESULTING IN REACTOR SHUTDOWN.
EVENT DATE: 081987 REPORT DATE: 091787 NSSS: GE TYPE: BWR

(NSIC 206072) ON 8/18/87 AT APPROXIMATELY 2300 CDT, WITH THE UNIT IN STARTUP, PLANT PERSONNEL WERE COMPLETING A DRYWELL CLOSEOUT AND A WALKDOWN OF RECIRCULATION PIPING AND VISUALLY OBSERVED A LEAK IN A 1" INSTRUMENT LINE. THE INSTRUMENT LINE CONNECTS THE 28" RECIRCULATION DISCHARGE PIPING (EIIS CODE AD) FLOW ELEMENT WITH A FLOW TRANSMITTER. THE LEAKAGE WAS A NON-ISOLABLE FAULT IN A REACTOR COOLANT SYSTEM COMPONENT PIPE WALL. WITH ANY LEAKAGE OF THIS TYPE, THE TECHNICAL SPECIFICATIONS REQUIRES THE PLANT TO BE IN HOT SHUTDOWN WITHIN 12 HOURS AND IN COLD SHUTDOWN WITHIN THE NEXT 24 HOURS. PLANT OPERATIONS PERSONNEL IMPLEMENTED THE SHUTDOWN REQUIREMENT. THE INTERMEDIATE CAUSE OF THIS EVENT IS A FAILURE IN THE INSTRUMENT LINE IN THE HEAT AFFECTED ZONE (HAZ) ADJACENT TO A SOCKET WELD. THE ROOT CAUSE FOR THE FAILURE WILL BE EVALUATED ONCE THE FAILED AREA IS ANALYZED. CORRECTIVE ACTION FOR THIS EVENT INCLUDED: 1) SHUTTING THE PLANT DOWN, 2) DISCUSSING THE EVENT WITH NRC PERSONNEL, 3) DISCUSSING CORRECTIVE ACTIONS WITH GENERAL ELECTRIC, 4) DESIGNING, INSTALLING, AND TESTING A WELD OVERLAY, AND 5) SCHEDULING REMOVAL AND METALLURGICAL ANALYSIS OF THE FAILED AREA. THE ANALYSIS RESULTS WILL BE PRESENTED IN AN UPDATE TO THIS LER WHICH WILL BE DEVELOPED BY APPROXIMATELY 4/25/88.

[112] HATCH 2 DOCKET 50-366 LER 87-011
PROCEDURE NON-COMPLIANCE RESULTS IN FAILURE TO REVISE PROCEDURE AND MISSED SURVEILLANCE.
EVENT DATE: 090287 REPORT DATE: 100287 NSSS: GE TYPE: BWR

(NSIC 206599) ON 9/2/87 AT APPROXIMATELY 1430 CDT, UNIT 2 WAS IN THE RUN MODE AT A POWER LEVEL OF APPROXIMATELY 2205 MWT (APPROXIMATELY 90 PERCENT OF RATED THERMAL POWER). AT THAT TIME, PERSONNEL IN THE PROCEDURES UPGRADE PROGRAM (PUP) DETERMINED THAT NOT ALL OF THE SURVEILLANCE REQUIREMENTS OF TECHNICAL SPECIFICATIONS AMENDMENT 71 HAD BEEN INCORPORATED INTO APPLICABLE SURVEILLANCE PROCEDURES. AS SUCH, SOME SURVEILLANCE REQUIREMENTS FOR THE MAIN CONTROL ROOM ENVIRONMENTAL CONTROL (MCREC EIIS CODE VI) SYSTEM WERE MISSED AND THIS IS A CONDITION PROHIBITED BY THE TECHNICAL SPECIFICATIONS. THE CAUSE OF THIS EVENT IS

FAILURE ON THE PART OF MAINTENANCE SUPERVISORY PERSONNEL TO COMPLETELY FOLLOW THE ADMINISTRATIVE CONTROL REQUIREMENTS GOVERNING INCORPORATION OF TECHNICAL SPECIFICATIONS AMENDMENT REQUIREMENTS INTO APPLICABLE PROCEDURES. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDED: 1) TEMPORARILY REVISING THE SURVEILLANCE PROCEDURE AND SATISFACTORILY PERFORMING THE REVISED PROCEDURE, 2) COUNSELING INVOLVED SUPERVISORY PERSONNEL, 3) REVIEWING EXISTING ADMINISTRATIVE CONTROL PROCEDURES, 4) SUBMITTING THE SURVEILLANCE PROCEDURE FOR PERMANENT REVISION, AND 5) REVIEWING OTHER SURVEILLANCE PROCEDURES.

[113] HOPE CREEK 1 DOCKET 50-354 LER 87-015 REV 01
UPDATE ON AUTO-ISOLATION OF THE CONTROL ROOM VENTILATION SYSTEM CAUSED BY SPURIOUS SIGNAL FROM RADIATION MONITORING SYSTEM.
EVENT DATE: 021387 REPORT DATE: 093087 NSSS: GE TYPE: BWR
VENDOR: GENERAL ATOMIC CO.

(NSIC 206510) AN AUTO-ISOLATION OF THE CONTROL ROOM VENTILATION (CRV) SYSTEM OCCURRED WHEN A SPURIOUS CHANNEL D HIGH RADIATION SIGNAL WAS RECEIVED. WHEN CRV ISOLATED, THE "B" CONTROL ROOM EMERGENCY FILTRATION (CREF) UNIT AUTO-STARTED. A CHECK OF THE RADIATION MONITORING SYSTEM (RMS) DETERMINED THAT RADIATION LEVELS WERE NORMAL AT THE TIME THE SIGNAL WAS GENERATED. AS A RESULT THE SIGNAL WAS CONSIDERED SPURIOUS AND THE CRV SYSTEM WAS RETURNED TO SERVICE AND "B" CREF UNIT WAS STOPPED. SUBSEQUENT INVESTIGATION REVEALED THAT AN I&C FUNCTIONAL TEST (FT) WAS BEING PERFORMED ON THE DRYWELL LEAK DETECTION MONITORING SYSTEM AT THE TIME THE ISOLATION OCCURRED AND THAT THE ISOLATION PROBABLY RESULTED FROM A VOLTAGE SPIKE GENERATED DURING THE PERFORMANCE OF THE FT. ATTEMPTS TO REPEAT THE CIRCUMSTANCES WHICH CAUSED THE CRV RADIATION MONITOR TO ALARM HAVE PROVEN UNSUCCESSFUL, AND AS A RESULT, NO DEFINITE CORRECTIVE ACTIONS CAN BE ASSIGNED.

[114] HOPE CREEK 1 DOCKET 50-354 LER 87-033
UNANTICIPATED INITIATION OF "E" FRVS RECIRCULATION FAN DUE TO MALFUNCTIONING SWITCHES.
EVENT DATE: 073087 REPORT DATE: 083187 NSSS: GE TYPE: BWR

(NSIC 206285) ON JULY 30, 1987 AT 0716 HOURS THE PLANT WAS IN OPERATIONAL CONDITION 1 (POWER OPERATION) AT 100% POWER, GENERATING 1071 MWE WHEN AN UNANTICIPATED START OF THE "E" FILTRATION, RECIRCULATION AND VENTILATION SYSTEM (FRVS) OCCURRED. THE ROOT CAUSE OF THIS OCCURRENCE WAS DETERMINED TO BE DIRTY CONTACTS IN LOW FLOW SWITCHES IN THE "A" AND "B" FRVS TRAINS. THE CONTACTS WERE CLEANED AND THE SWITCHES WERE RESTORED TO SERVICE.

[115] HOPE CREEK 1 DOCKET 50-354 LER 87-034
REACTOR SCRAM DUE TO INADVERTENT DE-ENERGIZING OF 120VAC INVERTER.
EVENT DATE: 073087 REPORT DATE: 083187 NSSS: GE TYPE: BWR
VENDOR: CYBEREX INC.

(NSIC 206294) A REACTOR SCRAM OCCURRED WHEN A 120 VAC UNINTERRUPTABLE POWER SUPPLY (UPS) INVERTER WHICH POWERS THE FEEDWATER CONTROL CABINET BECAME DE-ENERGIZED DURING THE COURSE OF ROUTINE MAINTENANCE. THE RESULTANT LOSS OF FEEDWATER CONTROL CAUSED THE REACTOR TO SCRAM ON LOW LEVEL AT +12.5" DECREASING. SUBSEQUENT INVESTIGATION DETERMINED THAT THE INVERTER BECAME DE-ENERGIZED WHEN IT WAS BEING RETURNED TO A NORMAL CONFIGURATION. A SWITCHING ERROR CAUSED THE INVERTER MAIN POWER SUPPLY FUSE TO BLOW. SEVERAL FACTORS CONTRIBUTED TO A LACK OF AWARENESS ON THE PART OF THE OPERATOR INVOLVED THAT THE INVERTER MAIN FUSE HAD BLOWN. IN AN ATTEMPT TO CORRECT THE SWITCHING ERROR, THE BACKUP INFEED SOURCE TO THE INVERTER WAS ALSO DE-ENERGIZED. THESE ACTIONS COMBINED TO REMOVE THE ENTIRE INVERTER FROM SERVICE. CORRECTIVE ACTIONS INCLUDED VERIFYING FAULT INDICATING LAMPS ON ALL INVERTERS WERE INSTALLED, REVISING THE INVERTER OPERATING PROCEDURE,

COUNSELLING THE OPERATOR INVOLVED, AND ENHANCING OPERATOR TRAINING ON INVERTER OPERATIONS.

[116] HOPE CREEK 1 DOCKET 50-354 LER 87-035
UNANTICIPATED START OF "B" SLCS PUMP CAUSE UNDETERMINED.
EVENT DATE: 080487 REPORT DATE: 090387 NSSS: GE TYPE: BWR
VENDOR: STRUTHERS DUNN, INC.

(NSIC 206255) ON AUGUST 4, 1987 AT 2144 HOURS THE PLANT WAS IN OPERATIONAL CONDITION 1 (POWER OPERATION) AT 100% POWER PRODUCING 1065 MWE WHEN THE "B" STANDBY LIQUID CONTROL SYSTEM (SLCS) PUMP STARTED, THE SQUIB VALVE FIRED AND THE REACTOR WATER CLEANUP (RWCU) ISOLATED. THE OPERATOR SECURED THE SLCS PUMP. AT 2210 HOURS THE "B" SLCS PUMP STARTED AGAIN AND WAS AGAIN STOPPED. THE RWCU ISOLATED A SECOND TIME. THE ROOT CAUSE OF THIS OCCURRENCE HAS NOT BEEN DETERMINED, HOWEVER A RELAY IN THE SLCS PUMP START CIRCUIT WAS OBSERVED TO EXHIBIT RESONANT IMPEDANCE WHICH WAS HYPOTHESIZED TO HAVE ORIGINATED IN THE REDUNDANT REACTOR CONTROL SYSTEM (RRCS) SELF-TEST PULSES. THIS RESONANT IMPEDANCE MAY HAVE CONTRIBUTED TO THE SPURIOUS ACTUATION OF THE SYSTEM. THE SLCS PIPING WAS FLUSHED AND THE SPENT SQUIB WAS REPLACED IN THE SLCS PUMP START VALVE. THE RRCS PUMP START RELAYS WHICH HAD EXHIBITED RESONANT IMPEDANCE WERE REPLACED WITH RELAYS OF ANOTHER MANUFACTURER WHICH DO NOT EXHIBIT THIS CHARACTERISTIC.

[117] HOPE CREEK 1 DOCKET 50-354 LER 87-036
LOSS OF CONTROL POWER TO HPCI, RHR AND CORE SPRAY LOGIC CIRCUITS - ROOT CAUSE UNDER INVESTIGATION.
EVENT DATE: 080487 REPORT DATE: 090387 NSSS: GE TYPE: BWR
VENDOR: TOPAZ ELECTRONICS

(NSIC 206256) ON AUGUST 4, 1987 AT 2030 HOURS, THE PLANT WAS IN OPERATIONAL CONDITION 1 (POWER OPERATION) AT 100% POWER GENERATING 1067 MWE, WHEN SEVERAL "A" CHANNEL CLASS 1E 125 VDC LOADS FROM THE "A" DISTRIBUTION PANEL WERE DE-ENERGIZED. THE BREAKERS FOR THE AFFECTED SYSTEMS WERE THEN OPENED AND RESET. THESE ACTIONS RESTORED POWER TO ALL SYSTEMS THAT HAD LOST 125 VDC. THE ROOT CAUSE OF THIS OCCURRENCE IS STILL UNDER INVESTIGATION. CORRECTIVE ACTIONS WILL INCLUDE PROCEDURE REVISIONS AS NECESSARY AND FURTHER INVESTIGATION OF THE CHARACTERISTICS OF THE BATTERY CHARTERS AND THE TOPAZ INVERTERS AT THE NEXT PLANNED OUTAGE.

[118] HOPE CREEK 1 DOCKET 50-354 LER 87-037
REACTOR SCRAM AND HIGH PRESSURE COOLANT INJECTION DUE TO PROCEDURAL DEFICIENCY.
EVENT DATE: 081687 REPORT DATE: 091587 NSSS: GE TYPE: BWR

(NSIC 206417) ON AUGUST 16, 1987 AT 0207 HOURS THE PLANT WAS IN OPERATIONAL CONDITION 1 (POWER OPERATION) AT 85% POWER GENERATING 905 MWE. WHILE IN THE PROCESS OF RETURNING THE "C" REACTOR FEED PUMP (RFP) TO SERVICE FOLLOWING MAINTENANCE, ALL FEEDWATER PUMPS TRIPPED DUE TO A LOSS OF CONDENSER VACUUM, AND A LOW VESSEL WATER LEVEL SCRAM LEVEL 3, +12.5" DECREASING AND SUBSEQUENT HIGH PRESSURE COOLANT INJECTION (HPCI) INJECTION OCCURRED. THE LOSS OF CONDENSER VACUUM WAS CAUSED BY OPENING THE "C" REACTOR FEED PUMP TURBINE (RFPT) EXHAUST VALVE WITH A BLOWN RUPTURE DISK IN THE RFPT EXHAUST LINE. INVESTIGATION SUBSEQUENT TO THE SCRAM DETERMINED THAT THE PRIMARY CAUSE OF THE EVENT WAS A DEFICIENT FEEDWATER SYSTEM OPERATING PROCEDURE. THE PROCEDURE DID NOT CONTAIN ADEQUATE GUIDANCE FOR REMOVING AND RETURNING AN RFP TO SERVICE DURING POWER OPERATION. IMMEDIATE CORRECTIVE ACTIONS CONSISTED OF REVISING THE AFFECTED PROCEDURE. LONGER TERM CORRECTIVE ACTIONS INCLUDE EVALUATING CONDENSER VACUUM SETPOINTS AND PERFORMING A DYNAMIC RESPONSE TEST OF THE RFP AND MAIN TURBINE TO ENSURE PROPER RESPONSE OF THE RFP'S AND MAIN TURBINE TO A LOSS OF CONDENSER VACUUM.

[119] HOPE CREEK 1 DOCKET 50-354 LER 87-038
FAILURE TO PERFORM A REACTOR LEVEL INSTRUMENTATION SURVEILLANCE WITHIN THE REQUIRED PERIOD - PERSONNEL ERROR.
EVENT DATE: 081887 REPORT DATE: 091787 NSSS: GE TYPE: BWR

(NSIC 206071) ON AUGUST 18, 1987 AT 1520 HOURS THE PLANT WAS IN OPERATIONAL CONDITION 1 (POWER OPERATION) AT 100 POWER GENERATING 1072 MWE. AT THIS TIME IT WAS DETERMINED THAT A SURVEILLANCE OF THE REACTOR LEVEL INSTRUMENTATION REQUIRED BY THE PLANT TECHNICAL SPECIFICATIONS WAS OVERDUE. THE SYSTEMS WHICH ARE INITIATED BY THE LEVEL INSTRUMENTATION WERE DECLARED INOPERABLE AND THE APPLICABLE TECHNICAL SPECIFICATION ACTION STATEMENTS WERE ENTERED. THE MISSED SURVEILLANCE WAS COMPLETED SATISFACTORILY AT 1640 HOURS AND THE TECHNICAL SPECIFICATION ACTION STATEMENTS WERE CLEARED. THE ROOT CAUSES OF THIS OCCURRENCE ARE (1) THE FAILURE TO ISSUE THE WORK ORDER FOR THE SURVEILLANCE TO THE DEPARTMENT RESPONSIBLE FOR PERFORMING THE SURVEILLANCE TEST IN A TIMELY MANNER AND (2) THE FAILURE TO CROSS CHECK WORK PACKAGES AGAINST THE WEEKLY LOOK AHEAD SURVEILLANCES LIST - PERSONNEL ERRORS. CORRECTIVE ACTIONS INCLUDE COUNSELLING OF THE INDIVIDUALS RESPONSIBLE FOR THE PERSONNEL ERRORS.

[120] HOPE CREEK 1 DOCKET 50-354 LER 87-039
REACTOR SCRAM WHILE PERFORMING TURBINE OVERSPEED OPERABILITY TEST DUE TO PRESSURE TRANSIENT IN TURBINE ELECTROHYDRAULIC CONTROL SYSTEM.
EVENT DATE: 082987 REPORT DATE: 092887 NSSS: GE TYPE: BWR

(NSIC 206496) A REACTOR SCRAM OCCURRED WHILE PERFORMING A TURBINE OVERSPEED PROTECTION OPERABILITY TEST. COMBINED INTERMEDIATE VALVE (CIV) #5 WAS BEING STROKED CLOSED, AND A TRANSIENT OF UNDETERMINED ORIGIN IN THE EMERGENCY TRIP SYSTEM (ETS) CAUSED THE TURBINE CONTROL VALVES TO FASTCLOSE, RESULTING IN THE SCRAM. THE SUBJECT TEST WAS REPEATED A NUMBER OF TIMES SUBSEQUENT TO THE EVENT IN AN ATTEMPT TO DETERMINE THE FAILURE MECHANISM THAT CAUSED THE TRANSIENT IN THE ETS HEADER. BECAUSE THE TRANSIENT WAS NOT REPEATABLE, CORRECTIVE ACTIONS WERE DEVELOPED BASED ON INDUSTRY EXPERIENCE WITH THE TURBINE EHC SYSTEM/ETS HEADER. THESE ACTIONS INCLUDED INITIATING A DESIGN CHANGE TO INSTALL RESTRICTING ORIFICES IN THE FAST-ACTING SOLENOIDS FOR ALL 20 TURBINE VALVES, AND, AS AN INTERIM MEASURE PRIOR TO DCP IMPLEMENTATION, REDUCING REACTOR POWER TO LESS THAT 30% POWER WHEN PERFORMING THE TURBINE OVERSPEED PROTECTION OPERABILITY TEST.

[121] LA SALLE 1 DOCKET 50-373 LER 87-028
SPURIOUS AMMONIA DETECTION TRIP DUE TO DESIGN DEFICIENCY IN THE CHEMCASSETTE TAPE MECHANISM.
EVENT DATE: 081387 REPORT DATE: 091087 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LA SALLE 2 (BWR)
VENDOR: M D A SCIENTIFIC, INC.

(NSIC 206320) AT 0513 HOURS ON AUGUST 13, 1987, WITH UNIT 1 IN OPERATIONAL CONDITION 5 (REFUEL) AND UNIT 2 IN OPERATIONAL CONDITION 1 (RUN) AT 99% POWER, THE "B" CONTROL ROOM HVAC SYSTEM (VC) "B" AMMONIA DETECTOR (OXY-VC165YB) TRIPPED. PER DESIGN, AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION OCCURRED WHICH RESULTED IN THE FOLLOWING: THE "B" VC "ODOR EATER" (CHARCOAL ADSORBER) WAS PLACED INTO OPERATION AND THE MINIMUM OUTSIDE AIR DAMPERS CLOSED. THE CHEMCASSETTE TAPE WAS FOUND BROKEN IN THE TAKEUP SPOOL DUE TO A DESIGN DEFICIENCY IN THE CHEMCASSETTE TAPE MACHANISM. THE CHEMCASSETTE TAPE WAS READJUSTED AND THE DETECTOR WAS RESET AT 1036 HOURS THE SAME DAY. THIS EVENT IS REPORTABLE PURSUANT TO THE REQUIREMENTS OF 10CFR50.73(A)(2)(IV) DUE TO THE ACTUATION OF AN ESF SYSTEM.

[122] LA SALLE 1 DOCKET 50-373 LER 87-029
REACTOR WATER CLEANUP OUTBOARD ISOLATION VALVE CLOSURE DUE TO PROCEDURAL DEFICIENCY.
EVENT DATE: 082987 REPORT DATE: 091787 NSSS: GE TYPE: BWR

(NSIC 206321) AT 0630 HOURS ON AUGUST 29, 1987, WITH UNIT 1 IN COLD SHUTDOWN, A JUMPER, WHICH DEFEATED THE REACTOR WATER CLEANUP (RWCU) FILTER/DEMINERALIZER HIGH INLET TEMPERATURE ISOLATION, WAS BEING REMOVED IN ACCORDANCE WITH LOP-NB-01, REACTOR VESSEL LEAKAGE TEST. THE JUMPER WAS UNINTENTIONALLY GROUNDED, CAUSING A FUSE IN THE RWCU OUTBOARD ISOLATION VALVE LOGIC CIRCUITRY TO BLOW. THIS CAUSED THE RWCU OUTBOARD ISOLATION VALVE, 1G33-F004, TO CLOSE. THE FUSE WAS REPLACED AND THE RWCU SYSTEM WAS RESTARTED AT 0650 HOURS ON THE SAME DAY. THE ROOT CAUSE OF THE BLOWN FUSE AND SUBSEQUENT ISOLATION VALVE CLOSURE WAS A PROCEDURAL DEFICIENCY WHICH DIRECTED THE INSTALLATION OF A JUMPER IN A DIFFICULT TO REACH LOCATION. THE SAFETY CONSEQUENCES OF THIS EVENT WERE MINIMAL SINCE THE RWCU OUTBOARD ISOLATION VALVE CLOSED PER DESIGN UPON DE-ENERGIZATION OF THE OUTBOARD ISOLATION LOGIC. TO PREVENT FUTURE GROUNDING EVENTS DURING LOP-NB-01, THE HIGH INLET TEMPERATURE ISOLATION WILL BE DEFEATED IN A DIFFERENT PANEL LOCATION. THE NEW PANEL LOCATION WILL MAKE JUMPER INSTALLATION AND REMOVAL EASIER. IN ADDITION, A JUMPER WITH AN INSULATED BOOT WILL BE USED IN FUTURE PERFORMANCES OF LOP-NB-01. THIS REPORT IS BEING SUBMITTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(IV) DUE TO AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION.

[123] LA SALLE 1 DOCKET 50-373 LER 87-030
REACTOR SCRAM WHILE SHUTDOWN DURING SURVEILLANCE DUE TO COMMUNICATION ERROR.
EVENT DATE: 090287 REPORT DATE: 092887 NSSS: GE TYPE: BWR

(NSIC 206612) AT 0814 HOURS ON SEPTEMBER 2, 1987, WITH UNIT 1 IN OPERATIONAL CONDITION 4 (COLD SHUTDOWN), ELECTRICAL MAINTENANCE (EM) PERSONNEL WERE PERFORMING STATION SURVEILLANCE (LES-RP-107, "CONTROL ROD DRIVEN (CRD) CHARGING WATER HEADER PRESSURE TIME DELAY RELAY CALIBRATION AND FUNCTIONAL TEST." DURING THE COURSE OF THE SURVEILLANCE, HALF SCRAMS ARE GENERATED AS EACH TIME DELAY RELAY IS TESTED. A FULL SCRAM OCCURRED WHEN A HALF SCRAM CONDITION HAD NOT BEEN RESET PRIOR TO PERFORMING THE TEST ON A RELAY IN THE REMAINING (ENERGIZED) REACTOR PROTECTION SYSTEM (RPS) CHANNEL. THE CAUSE OF THIS EVENT WAS INADEQUATE COMMUNICATION BETWEEN THE ELECTRICIAN PERFORMING THE SURVEILLANCE AND THE UNIT OPERATOR. THE SAFETY CONSEQUENCES OF THIS EVENT WERE MINIMAL SINCE RPS RESPONDED PER DESIGN TO A LOW CRD CHARGING WATER HEADER PRESSURE SIGNAL. SINCE ALL CONTROL RODS WERE FULLY INSERTED AT THE TIME OF THIS EVENT, NO ROD MOTION OCCURRED AS A RESULT OF THE SCRAM. THIS EVENT IS REPORTABLE PURSUANT TO THE REQUIREMENT OF 10CFR50.73(A)(2)(IV) DUE TO THE AUTOMATIC ACTUATION OF THE REACTOR PROTECTION SYSTEM.

[124] LA SALLE 2 DOCKET 50-374 LER 87-017
SPECIAL REPORT NOT SUBMITTED WITHIN REQUIRED TIME FRAME DUE TO PERSONNEL ERROR.
EVENT DATE: 082787 REPORT DATE: 092587 NSSS: GE TYPE: BWR

(NSIC 206558) ON JUNE 29, 1987 AT 1400 HOURS WITH UNIT 2 IN OPERATIONAL CONDITION 1 (RUN) AT 64% POWER, THE DRYWELL TEMPERATURE MONITORING PROGRAM PROCEDURES, LTP-300-17 AND LTP-300-17T, WERE PERFORMED PER THE STATION SURVEILLANCE SCHEDULE. THE TEMPERATURE DATA IS GATHERED AND REVIEWED BY STATION PERSONNEL, AND ON A QUARTERLY BASIS IS TRANSMITTED OFFSITE TO THE BOILING WATER REACTOR (BWR) ENGINEERING GROUP FOR REVIEW. AN OFFSITE REVIEW DETECTED THAT THE SET OF UNIT 2 DRYWELL TEMPERATURE READINGS ON JUNE 29, 1987 WERE HIGHER THAN PREVIOUS READINGS. AN ADDITIONAL REVIEW BY STATION PERSONNEL ON AUGUST 27, 1987 DISCOVERED THAT SENSOR 2TE-VP204, LOCATED IN THE VICINITY OF SAFETY RELIEF VALVES D AND J, READ AMBIENT AIR TEMPERATURE OF 153 DEGREES F WHICH EXCEEDED TECH SPEC TABLE 3.7.7-1 LIMIT OF 150 DEGREES F AND NO SPECIAL REPORT HAS BEEN SUBMITTED AS REQUIRED BY TECH SPEC 3.7.7.A. THE FAILURE TO SUBMIT A SPECIAL REPORT WAS CAUSED BY

PERSONNEL ERROR IN THAT THE PERSONNEL PERFORMING THE PROCEDURE BELIEVED A REPORT ALREADY EXISTED FOR THE EVENT. PERSONNEL WHO PERFORM AND REVIEW THE DRYWELL MONITORING PROCEDURES WERE TAILGATED ON THIS EVENT.

[125] LIMERICK 1 DOCKET 50-352 LER 87-039
NONCOMPLIANCE WITH TECHNICAL SPECIFICATION ACTION STATEMENT BECAUSE OF LATE GRAB SAMPLE CAUSED BY PERSONNEL ERROR.
EVENT DATE: 073187 REPORT DATE: 083187 NSSS: GE TYPE: BWR

(NSIC 206252) TECHNICAL SPECIFICATION 3.3.7.1, TABLE 3.3.7.1-1 REQUIRES TAKING A GRAB SAMPLE OF THE REACTOR ENCLOSURE COOLING WATER (RECW) SYSTEM EVERY 24 HOURS WHENEVER THE RECW RADIATION MONITOR IS INOPERABLE. A UTILITY EMPLOYED CHEMISTRY TECHNICIAN, THROUGH A PERSONAL OVERSIGHT, FAILED TO RECOGNIZE THAT A GRAB SAMPLE WAS REQUIRED TO BE TAKEN FROM THE RECW SYSTEM AT 1110 HOURS ON JULY 31, 1987. ON AUGUST 1, 1987, THE SAME TECHNICIAN NOTICED THAT THE GRAB SAMPLE WHICH WAS REQUIRED EVERY 24 HOURS FOR THE RECW SYSTEM HAD NOT BEEN TAKEN ON THE PREVIOUS DAY. A SAMPLE WAS TAKEN AT 1110 HOURS ON AUGUST 1, 1987. THERE WERE NO ADVERSE CONSEQUENCES AS A RESULT OF THIS EVENT. THERE WAS NO RELEASE OF RADIOACTIVE MATERIAL AS A RESULT OF THIS EVENT. THE SAMPLE TAKEN ON AUGUST 1, 1987, WAS ANALYZED AND RADIOACTIVITY LEVELS WERE FOUND TO BE LESS THAN THE LOWER LIMIT OF DETECTION. THE RESPONSIBLE INDIVIDUAL WAS COUNSELED AND ALL SHIFT CHEMISTRY TECHNICIANS WERE INSTRUCTED BY CHEMISTRY SUPERVISION AS TO THE IMPORTANCE OF SAMPLING ON TIME AND OF PROPER COMMUNICATION DURING SHIFT TURNOVER.

[126] LIMERICK 1 DOCKET 50-352 LER 87-038
NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM INBOARD ISOLATION DUE TO A PERSONNEL ERROR.
EVENT DATE: 080287 REPORT DATE: 090187 NSSS: GE TYPE: BWR

(NSIC 206251) ON AUGUST 2, 1987 AT 1223 HOURS, DURING A REFUEL OUTAGE VARIOUS NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NSSSS) ISOLATIONS, AN ENGINEERED SAFETY FEATURE, OCCURRED WHEN FUSE B21-F15A, BLEW DURING THE INSTALLATION OF A TEMPORARY CIRCUIT ALTERATION. VALVES HV-87-128 AND HV-87-129, OF THE DRYWELL CHILLED WATER (DWCW) SYSTEM CLOSED AS A RESULT OF AN ISOLATION SIGNAL, STOPPING THE FLOW OF COOLING WATER TO THE RECIRCULATION PUMP MOTOR AIR COOLERS AND DRYWELL AIR COOLERS. OTHER SYSTEMS RECEIVED ISOLATION SIGNALS; HOWEVER, THE VALVES AFFECTED BY THESE ISOLATION SIGNALS HAD BEEN PREVIOUSLY CLOSED FOR OUTAGE ACTIVITIES. AT 1227 HOURS THE DWCW ISOLATION SIGNAL WAS BYPASSED AND COOLING WATER WAS RESTORED. AT 1255 THE BLOWN FUSE WAS REPLACED, ALL ISOLATION SIGNALS WERE RESET AND THE TEMPORARY CIRCUIT ALTERATION (TCA) WAS RE-APPLIED WITHOUT FURTHER INCIDENT. THERE WERE NO ADVERSE CONSEQUENCES AND NO RELEASE OF RADIOACTIVE MATERIAL AS A RESULT OF THIS EVENT. THE CAUSE OF THIS EVENT WAS A PERSONNEL ERROR WHEN A UTILITY EMPLOYED ENGINEER INADVERTENTLY CAUSED A SHORT TO GROUND DURING THE TCA INSTALLATION. THE ENGINEER WHO APPLIED THE TCA WAS COUNSELED ON THE IMPORTANCE OF THE CAREFUL INSTALLATION OF OF A TCA.

[127] LIMERICK 1 DOCKET 50-352 LER 87-040
NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM GROUP I ISOLATION DUE TO FAULTY WIRE CONNECTION.
EVENT DATE: 080487 REPORT DATE: 090387 NSSS: GE TYPE: BWR

(NSIC 206253) ON AUGUST 4, 1987 AT 1454 HOURS, AN ACTUATION OF THE GROUP I NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NSSSS) OCCURRED. IN ORDER TO SUPPORT OUTAGE TURBINE TESTING, THE MAIN TURBINE (MT) WAS RESET, PRESSURIZING THE EMERGENCY TRIP SYSTEM (ETS) HYDRAULICS. WHEN THE MAIN TURBINE RESET OCCURRED THE MAIN STOP VALVES (MSVS) UNEXPECTEDLY OPENED, RESULTING IN A "LOW CONDENSER VACUUM" GROUP I ISOLATION. A FAULTY CONNECTION TO AN ELECTRO HYDRAULIC CONTROL (EHC) LOGIC BOARD WAS FOUND TO HAVE CAUSED THE MSVS TO OPEN. THE ISOLATION WAS RESET, AND THE FAULTY CONNECTION WAS IDENTIFIED AND REPAIRED. ALL GROUP I ISOLATION VALVES WERE

CLOSED PRIOR TO THE ISOLATION SIGNAL THEREFORE THERE WAS NO MOVEMENT OF THE VALVES, AND NO RELEASE OF RADIATION OCCURRED. TURBINE TESTING WAS THEN COMPLETED SATISFACTORILY.

[128] LIMERICK 1 DOCKET 50-352 LER 87-042
ACTUATION OF ENGINEERED SAFETY FEATURES DUE TO A PERSONNEL ERROR.
EVENT DATE: 081887 REPORT DATE: 091787 NSSS: GE TYPE: BWR

(NSIC 206070) ON AUGUST 18, 1987 AT 0341 HOURS A DIVISION 4 LOSS OF COOLANT ACCIDENT (LOCA) SIGNAL OCCURRED. A SPURIOUS HIGH DRYWELL PRESSURE SIGNAL COMBINED WITH AN EXISTING LOW REACTOR PRESSURE RESULTED IN THE LOCA SIGNAL. THE SIGNAL CAUSED A NUMBER OF AUTOMATIC ENGINEERED SAFETY FEATURES TO OCCUR. ALL SYSTEMS WERE VERIFIED TO HAVE FUNCTIONED AS DESIGNED AND WERE RETURNED TO NORMAL BY 0432 HOURS. THERE WERE NO ADVERSE CONSEQUENCES AND NO RELEASE OF RADIATION AS A RESULT OF THIS EVENT. THE SPURIOUS HIGH DRYWELL PRESSURE SIGNAL WAS THE RESULT OF A VALVING ERROR DURING A SURVEILLANCE TEST. A FAILURE TO FOLLOW APPROVED PROCEDURES WAS THE CAUSE OF THE EVENT. THE VENDOR TECHNICIANS INVOLVED WERE COUNSELED REGARDING THE IMPORTANCE OF FOLLOWING PROCEDURES. IN ADDITION, THIS EVENT WAS REVIEWED WITH INSTRUMENT AND CONTROL TECHNICIANS AT AN "ALL HANDS" MEETING TO EMPHASIZE THE SERIOUSNESS OF THE EVENT.

[129] LIMERICK 1 DOCKET 50-352 LER 87-044
FAILURE TO COMPLY WITH TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENT DUE TO PROCEDURE DEFICIENCIES.
EVENT DATE: 082287 REPORT DATE: 092187 NSSS: GE TYPE: BWR

(NSIC 206461) ON AUGUST 22, 1987 IT WAS DISCOVERED THAT THE REMOTE SHUTDOWN PANEL (RSP) OPERABILITY TEST ST-1-088-320-1 DID NOT COMPLY WITH TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENT 4.3.7.4.2. THE EIGHTEEN MONTH SURVEILLANCE TEST FAILED TO INCLUDE A CHECK OF THE OPERABILITY OF THE REACTOR CORE ISOLATION COOLING (RCIC) FLOW CONTROLLER FROM THE RSP. THE SURVEILLANCE TEST (ST) ALSO FAILED TO TEST A DESIGN FUNCTION OF THE TRANSFER SWITCHES IN THAT IT DID NOT INCLUDE A NEGATIVE CHECK OF THE TRANSFER SWITCHES' SECONDARY FUNCTION OF ISOLATING THE RSP CIRCUITRY FROM THE MAIN CONTROL ROOM CIRCUITRY. DURING THE INVESTIGATION, THE RSP OPERABILITY CHECK OF THE RCIC CONDENSATE VACUUM PUMP WAS ALSO DETERMINED TO BE INCOMPLETE DUE TO OMISSION OF A SET OF RELAY CONTACTS FROM THE ST. THE CAUSE FOR THE OMISSION OF THE RCIC FLOW CONTROLLER OPERABILITY CHECK WAS A COMMUNICATION FAILURE BETWEEN TWO STATION ENGINEERING GROUPS WHEN THE ST WAS WRITTEN. THE NEGATIVE CHECK FOR THE TRANSFER SWITCHES WAS NOT INCLUDED IN THE ST AS A RESULT OF AN INTERPRETATION OF THE TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENT. THE CAUSE FOR THE OMISSION OF THE RCIC CONDENSATE VACUUM PUMP CONTACTS WAS A FAILURE TO UPDATE THE TEST PROCEDURE WHEN THE SYSTEM LOGIC PRINT WAS UPDATED IN 1984. THE DEFICIENCIES HAVE BEEN CORRECTED BY TEMPORARY PROCEDURE CHANGES WHICH WILL BE MADE PERMANENT BY OCTOBER 31, 1987.

[130] LIMERICK 1 DOCKET 50-352 LER 87-043
MISSED SNUBBER SURVEILLANCE REQUIREMENT DUE TO A PERSONNEL ERROR WHICH RESULTED IN A PROCEDURE DEFICIENCY.
EVENT DATE: 082487 REPORT DATE: 092887 NSSS: GE TYPE: BWR

(NSIC 206495) ON AUGUST 24, 1987 AT 1500 HOURS IT WAS DETERMINED THAT A SNUBBER ON THE HIGH PRESSURE COOLANT INJECTION SYSTEM WAS NOT VISUALLY INSPECTED WITHIN TEN MONTHS OF COMMENCING POWER OPERATION AS REQUIRED BY TECHNICAL SPECIFICATION SURVEILLANCE REQUIREMENT 4.7.4. DURING A VISUAL INSPECTION OF ALL SNUBBERS IN THE MAY 1986 SURVEILLANCE TEST OUTAGE SNUBBER EBD-106-E6-H31 (SNUBBER H31) WAS NOT INSPECTED. SNUBBER EBD-105-E6-H12 (SNUBBER H12) WAS VISUALLY INSPECTED TWICE, ONCE MISTAKENLY FOR SNUBBER H31. THE CAUSE OF THIS OVERSIGHT WAS A PERSONNEL ERROR WHICH RESULTED IN A PROCEDURAL DEFICIENCY IN THAT THE SERIAL

NUMBER LISTED IN THE SURVEILLANCE TEST FOR SNUBBER H31 WAS INCORRECTLY LISTED AS THE SERIAL NUMBER FOR SNUBBER H12. THIS NUMBER WAS INCORRECTLY ASSIGNED TO SNUBBER H31 PRIOR TO COMPLETION OF THE STARTUP TEST PROGRAM AT LIMERICK GENERATING STATION. SNUBBER H31 WAS VISUALLY INSPECTED AND FUNCTIONALLY TESTED DURING JULY OF THE 1987 REFUEL OUTAGE AND WAS DETERMINED TO BE OPERABLE. SURVEILLANCE TEST ST-4-103-098-1 WHICH PROVIDES FOR VISUAL INSPECTION OF SNUBBER H31 HAS BEEN REVISED TO INCLUDE THE CORRECT SERIAL NUMBER FOR THIS SNUBBER. THERE WERE NO ADVERSE CONSEQUENCES AS A RESULT OF THIS EVENT. THERE WAS NO RELEASE OF RADIOACTIVE MATERIAL AS A RESULT OF THIS EVENT.

[131] MAINE YANKEE DOCKET 50-309 LER 87-006 REV 01
UPDATE ON MANUAL REACTOR TRIP AFTER LOSS OF LEVEL CONTROL IN STEAM GENERATOR CAUSED BY PERSONNEL ERROR.
EVENT DATE: 062787 REPORT DATE: 090987 NSSS: CE TYPE: PWR
VENDOR: ROCHESTER INSTRUMENT SYSTEMS, INC.

(NSIC 206313) ON JUNE 27, 1987, A MANUAL REACTOR TRIP WAS INITIATED DUE TO AN UNCONTROLLABLE INCREASE IN STEAM GENERATOR #3 LEVEL. THE INCREASE IN STEAM GENERATOR LEVEL OCCURRED WHEN AN INSTRUMENT AND CONTROLS TECHNICIAN WAS REPLACING A COMPONENT IN THE FEEDWATER CONTROL CABINET. THE TECHNICIAN REMOVED OUTPUT LEADS FROM THE FAILED TIME LAG UNIT AND INTERRUPTED THE CONTROL SIGNAL TO THE #3 MAIN FEEDWATER REGULATING VALVE. THE VALVE FAILED OPEN CAUSING A RAPID RISE IN STEAM GENERATOR LEVEL. THE REACTOR WAS MANUALLY TRIPPED AS STEAM GENERATOR LEVEL APPROACHED THE AUTOMATIC TURBINE TRIP SETPOINT. ALL SYSTEMS RESPONDED PROPERLY FOLLOWING THE PLANT TRIP. THE ROOT CAUSE OF THIS EVENT IS PERSONNEL ERROR. THE IMPORTANCE OF PERFORMING THOROUGH CIRCUIT INVESTIGATIONS BEFORE CONDUCTING TROUBLESHOOTING OR MAINTENANCE WILL BE EMPHASIZED TO ALL I&C PERSONNEL. A GOOD WORK PRACTICES GUIDE WILL BE DEVELOPED TO PROVIDE GUIDANCE FOR INVESTIGATING DISCREPANCIES BETWEEN DRAWINGS AND AS-BUILT CONDITIONS AND DOCUMENTATION OF THE STATUS OF ONGOING JOBS.

[132] MCGUIRE 1 DOCKET 50-369 LER 87-014
DIESEL GENERATOR 1A CONTROL POWER CIRCUIT BREAKER WAS INADVERTENTLY LEFT OPEN DUE TO PERSONNEL ERROR.
EVENT DATE: 072887 REPORT DATE: 083187 NSSS: WE TYPE: PWR

(NSIC 206261) ON JULY 30, 1987, AT 0530, OPERATIONS DISCOVERED THE CONTROL POWER CIRCUIT BREAKER FOR DIESEL GENERATOR (D/G) 1A IN THE OPEN POSITION WHICH RENDERS D/G 1A INOPERABLE. THIS RESULTED IN A TECHNICAL SPECIFICATION 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 PERIODIC MAINTENANCE. D/G 1A WAS RETURNED TO OPERABLE STATUS AT 0645 ON JULY 31, 1987, UPON THE COMPLETION OF THE PERIODIC MAINTENANCE ACTIVITY. UNIT 1 WAS IN MODE 1, POWER OPERATION, AT 100% POWER, AT THE TIME OF THIS EVENT. 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 A PERIODIC MAINTENANCE ACTIVITY. A CONTRIBUTORY CAUSE CLASSIFICATION OF MANAGEMENT DEFICIENCY HAS ALSO BEEN ASSIGNED TO THIS EVENT BECAUSE THE INSTRUCTIONS GIVEN TO THE OPERATIONS PERSONNEL WERE NOT ADEQUATE TO ASSURE THE WORK WOULD BE DONE CORRECTLY. THIS EVENT IS CONSIDERED TO BE OF NO SIGNIFICANCE WITH RESPECT TO THE HEALTH AND SAFETY OF THE PUBLIC.

[133] MCGUIRE 1 DOCKET 50-369 LER 87-015
REQUIRED COMPENSATORY MEASURES WERE NOT TAKEN WHEN A DIESEL GENERATOR ROOM FIRE/FLOOD BARRIER WAS BREACHED DUE TO PERSONNEL ERROR.
EVENT DATE: 080587 REPORT DATE: 090887 NSSS: WE TYPE: PWR

(NSIC 206073) ON JULY 27, 1987 WITH UNIT 1 IN MODE 1, POWER OPERATION AT 100% POWER, DUKE CONSTRUCTION AND MAINTENANCE (CMD) PERSONNEL BREACHED THE FIRE/FLOOD BARRIER BETWEEN DIESEL GENERATOR ROOM 1A AND THE DIESEL LUBE OIL HOLDING TANKS ROOM. ON AUGUST 6, 1987, AT 1140, STATION PROJECTS PERSONNEL WERE ASSISTING CMD IN PREPARING TO FOAM FIRE/FLOOD PENETRATION NO. 1-733-3.1-1 AND DISCOVERED THAT CMD HAD ALREADY CUT OFF THE 2 INCH PIPE ON BOTH SIDES OF THE PENETRATION. PROJECTS NOTIFIED THE CONTROL ROOM AND THE PENETRATION WAS DECLARED INOPERABLE AT 1140. A FIRE WATCH WAS ESTABLISHED AS REQUIRED BY TECHNICAL SPECIFICATION 3.7.11. ON AUGUST 7, 1987, AT APPROXIMATELY 1530, CMD SEALED THE FIRE/FLOOD BARRIER PENETRATION AND AT 1550, OPERATIONS DECLARED THE PENETRATION OPERABLE. ON AUGUST 18, 1987, THE CAPPED PIPE WAS REMOVED FROM THE FIRE/FLOOD BARRIER AND THE BARRIER PENETRATION WAS FOAMED PER STATION PROCEDURE. THIS EVENT HAS BEEN CLASSIFIED AS PERSONNEL ERROR BECAUSE A CMD SUPERVISOR FAILED TO NOTIFY CONTROL ROOM PERSONNEL PER STATION DIRECTIVE 2.11.5, THAT A FIRE/FLOOD BARRIER WAS BREACHED SO THAT PROPER COMPENSATORY MEASURES COULD BE TAKEN. DUKE WILL PROVIDE TRAINING TO ALL CMD-NORTH PERSONNEL ON MCGUIRE STATION DIRECTIVE 2.11.5, FIRE PENETRATIONS.

[134] MCGUIRE 2 DOCKET 50-370 LER 87-010
FAILURE TO TEST A CONTAINMENT AIR LOCK SEAL OCCURRED DUE TO A DESIGN DEFICIENCY.
EVENT DATE: 073187 REPORT DATE: 090287 NSSS: WE TYPE: PWR

(NSIC 206262) ON JULY 31, 1987, INSTRUMENT AND ELECTRICAL PERSONNEL REPLACING THE PNEUMATIC MODULE IN THE UNIT 2 UPPER PERSONNEL AIR LOCK AUTOMATIC LEAK RATE MONITOR DISCOVERED A DEFICIENCY IN THE CONNECTIONS OF THE EQUIPMENT BEING REPLACED. INTERFACE TUBING BETWEEN THE PNEUMATIC MODULE AND PERMANENT TUBING WAS INCORRECTLY CONNECTED RESULTING IN THE AUTOMATIC LEAK RATE MONITOR ONLY TESTING THE INNER DOOR SEALS. WHEN THE OUTER DOOR TEST WAS SELECTED, THE TEST WOULD RETURN A POSITIVE RESULT, BUT IT WAS NOT ACTUALLY CONNECTED TO THE OUTER DOOR SEALS. UNIT 2 WAS IN MODE 1, POWER OPERATION, AT 100% POWER WHEN THIS CONDITION WAS DISCOVERED; HOWEVER, THE UNIT HAD APPARENTLY OPERATED IN ALL MODES WITH THIS CONDITION. THIS EVENT HAS BEEN CLASSIFIED AS A DESIGN DEFICIENCY DUE TO THE INCOMPLETENESS OF THE DESIGN DOCUMENTS. THE DESIGN DOCUMENTS DESCRIBING THE NECESSARY TUBING ARRANGEMENT LACKED CLARITY AND ALLOWED ANYONE WORKING WITH THE EQUIPMENT TO FALSELY BELIEVE THAT IT WAS CORRECTLY INSTALLED. THE DESIGN DOCUMENTS WERE CORRECTED AND TEMPORARY TUBING CHANGES WERE APPROVED FOR IMPLEMENTATION TO ENABLE THE AUTOMATIC LEAK RATE MONITOR TO TEST BOTH DOORS. THIS EVENT IS CONSIDERED TO BE OF NO SIGNIFICANCE WITH RESPECT TO THE HEALTH AND SAFETY OF THE PUBLIC.

[135] MCGUIRE 2 DOCKET 50-370 LER 87-013
A FIRE BARRIER WAS BREACHED WITHOUT COMPENSATORY ACTION DUE TO MANAGEMENT DEFICIENCY AND INADEQUATE STATION DIRECTIVE AND PROCEDURE.
EVENT DATE: 081087 REPORT DATE: 092887 NSSS: WE TYPE: PWR

(NSIC 206498) ON AUGUST 5, 1987 AT 1900, MCGUIRE MAINTENANCE (MNT) AND INTEGRATED SCHEDULING (IS) BECAME AWARE OF CUTS IN INSULATED BLANKETS SURROUNDING TWO AUXILIARY FEEDWATER VALVES. MNT WAS UNAWARE THE BLANKETS WERE FIRE BARRIERS AND IS ASSUMED THE FIRE BLANKETS WOULD BE REPAIRED BEFORE THE VALVES WERE DECLARED OPERABLE. DURING SHIFT TURNOVER, AT 1930, IS DISCUSSED THE FACT THAT THE FIRE BLANKETS NEEDED REPAIR BUT IS DID NOT RECOGNIZE THE BLANKETS WERE FIRE BARRIERS. NEITHER LOGGING NOR INITIATION OF A FIRE WATCH WAS CONSIDERED. A WORK REQUEST WAS WRITTEN TO REPAIR THE FIRE BLANKETS AND THE REPAIRS WERE COMPLETED ON AUGUST 12, 1987. TAGS WERE PLACED ON OTHER FIRE BARRIERS OF THIS TYPE IDENTIFYING THEM AS TECH SPECS BARRIERS. A CLASSIFICATION OF MANAGEMENT DEFICIENCY WAS ASSIGNED DUE TO SHORTCOMINGS IN STATION DIRECTIVES, PROCEDURES, AND TRAINING REGARDING INSULATED BLANKET FIRE BARRIERS. THE FIRE BARRIER INSPECTION PROCEDURE WILL BE REVISED, THE STATION DIRECTIVE WILL BE REVISED, AND EXISTING TRAINING WILL BE

ENHANCED. THE NEXT STATION SAFETY MEETING WILL INCLUDE INFORMATION TO INCREASE AWARENESS OF INSULATED BLANKET FIRE BARRIERS.

[136] MCGUIRE 2 DOCKET 50-370 LER 87-011
A HOLD DOWN BOLT WAS DISCOVERED MISSING FROM MISSILE SHIELD BLOCKS OVER THE REACTOR CAVITY DUE TO DEFECTIVE PROCEDURE AND INADQUATE INSTRUCTIONS.
EVENT DATE: 081487 REPORT DATE: 091487 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 1 (PWR)

(NSIC 206074) ON AUGUST 13, 1987, WITH UNIT 2 AT 100% POWER, MAINTENANCE (MNT) DISCOVERED FIVE MISSILE SHIELD HOLD-DOWN BOLTS IN THE DECONTAMINATION AREA. AN INVESTIGATION REVEALED A HOLD-DOWN BOLT WAS MISSING FROM ONE OF THE REACTOR CAVITY MISSILE SHIELD BLOCKS. MNT NOTIFIED OPERATIONS AT 1006 ON AUGUST 14, 1987. DESIGN ENGINEERING (DE) WAS CONTACTED TO EVALUATE THE CONDITIONS AND PROVIDE GUIDANCE ON THE OPERABILITY OF THE SHIELD BLOCKS. AT 1730, DE ADVISED MCGUIRE THAT THE SHIELD BLOCKS WERE INOPERABLE. OPERATIONS DECLARED THE SHIELD BLOCKS INOPERABLE AT THAT TIME. UNIT 2 WAS SHUT DOWN TO MODE 3, HOT STANDBY. MNT DISCOVERED FOUR OTHER BOLTS NOT INSTALLED. MNT REDRILLED THE BOLT HOLE AND INSTALLED THE MISSING BOLT ALONG WITH THE OTHER BOLTS AND VERIFIED ALL TEN BOLTS WERE INSTALLED. OPERATIONS DECLARED THE SHIELD BLOCKS OPERABLE AGAIN AT 0407 ON AUGUST 16, 1987, AND UNIT 2 WAS BACK ON LINE AT 2345. MNT WILL MODIFY THE EXISTING PROCEDURE OR WRITE A NEW PROCEDURE TO PROVIDE DETAILED REMOVAL/INSTALLATION INSTRUCTIONS FOR THE SHIELD BLOCKS. IF ADDITIONAL ACTIONS ARE PLANNED, A SUPPLEMENTAL REPORT WILL BE SUBMITTED. A CLASSIFICATION OF DEFECTIVE PROCEDURE HAS BEEN ASSIGNED TO THIS EVENT BECAUSE PROCEDURES WHICH RELATE TO THE MISSILE SHIELDS DID NOT CONTAIN ANY DETAILED INSTRUCTIONS ON THE PROPER INSTALLATION OF THE MISSILE SHIELD BLOCKS.

[137] MCGUIRE 2 DOCKET 50-370 LER 87-012
AN AUTOSTART OF AUXILIARY FEEDWATER SYSTEM DUE TO PROCEDURE DEFICIENCY - NOT REQUIRING RESET OF BOTH MAIN FEEDWATER PUMPS PRIOR TO TESTING.
EVENT DATE: 081687 REPORT DATE: 091587 NSSS: WE TYPE: PWR

(NSIC 206075) ON AUGUST 16, 1987 AT 2028, WHEN OPERATIONS PERSONNEL WERE PERFORMING STARTUP TESTING OF MAIN FEEDWATER (CF) PUMP 2B, AN UNPLANNED ENGINEERING SAFEGUARD FEATURES (ESF) ACTUATION OCCURRED. AN AUTOSTART SIGNAL TO THE AUXILIARY FEEDWATER (CA) MOTOR DRIVEN (M/D) PUMPS OCCURRED AS A RESULT OF THE INTENTIONAL TRIP OF CF PUMP 2B FOR TESTING WHEN CF PUMP 2A WAS IN THE TRIPPED CONDITION. BOTH M/D CA PUMPS WERE RUNNING BECAUSE OF OPERATIONAL CONSIDERATIONS PRIOR TO THE RECEIPT OF THE "LOSS OF BOTH MAIN FEEDWATER PUMPS" SIGNAL. OPERATIONS IMMEDIATELY RESET CF PUMP 2B AND THE CA FLOW CONTROL VALVES, 23 SECONDS AFTER CF PUMP 2B HAD BEEN TRIPPED. S/G BLOWDOWN AND SAMPLE FLOW WERE RESTORED AND CF PUMP 2A WAS RESET. NO SIGNIFICANT CHANGES IN STEAM GENERATOR (S/G) LEVEL WERE OBSERVED. UNIT 2 WAS IN MODE 2, STARTUP, AT 1% POWER AT THE TIME OF THE EVENT. THE CAUSE OF THE EVENT HAS BEEN CLASSIFIED AS DEFECTIVE PROCEDURE, BECAUSE THE CONDENSATE AND FEEDWATER SYSTEMS OPERATING PROCEDURE USED FOR THE CF PUMP STARTUP TESTING DID NOT INCLUDE RESETTING BOTH CF PUMPS PRIOR TO TRIPPING ONE OF THE CF PUMPS. PROCEDURES WERE CHANGED TO PREVENT THE REOCCURRENCE OF THIS EVENT.

[138] MILLSTONE 1 DOCKET 50-245 LER 86-018 REV 01
UPDATE ON REACTOR PROTECTION SYSTEM INITIATION FROM IRM NOISE SPIKE.
EVENT DATE: 052486 REPORT DATE: 093087 NSSS: GE TYPE: BWR

(NSIC 206574) ON MAY 24, 1986, AT 1853 HOURS, WHILE THE UNIT WAS SHUTDOWN (0 PSIG, 164 DEGREES FAHRENHEIT), A SOURCE RANGE MONITOR (SRM) WAS BEING WITHDRAWN WHEN AN AUTOMATIC ACTUATION OF THE REACTOR PROTECTION SYSTEM (RPS) OCCURRED. SRM 21 WAS BEING WITHDRAWN WHEN A NOISE SPIKE ON THE INTERMEDIATE POWER RANGE MONITOR

(IRM) 12 AND 16, CAUSED BY THE SRM DRIVE RELAYS CHATTERING, RESULTED IN AN RPS ACTUATION. THE SRM RELAYS WERE CYCLED IN AN EFFORT TO DUPLICATE THE CONDITIONS THAT OCCURRED DURING SRM 21 WITHDRAWAL, BUT NO RELAY CHATTER WAS OBSERVED AND IRM RESPONSE DID NOT INCREASE. THE SRM/IRM DRIVE RELAYS WERE PROPOSED TO BE REPLACED DURING THE 1987 REFUELING OUTAGE AND A FOLLOW UP REPORT BE SUBMITTED. THERE WERE NO SAFETY CONSEQUENCES RESULTING FROM THE RPS ACTUATION SINCE THE UNIT WAS SHUTDOWN. APPROXIMATELY 50% OF ALL SRM/IRM DRIVE RELAYS HAVE BEEN REPLACED DURING THE 1987 REFUELING OUTAGE. THE REMAINING RELAY REPLACEMENT WILL BE POSTPONED TO THE NEXT AVAILABLE OPPORTUNITY WHEN THE PLANT IS SHUTDOWN OR IN A REFUELING OUTAGE. THE POWER SUPPLY TO THE SRM/IRM DRIVE RELAYS HAS TO BE DE-ENERGIZED TO REPLACE THE RELAYS. DE-ENERGIZING THE POWER SUPPLY WILL REMOVE THE CAPABILITY TO DRIVE THE SRM/IRM IN THE CORE TO MONITOR NEUTRON FLUX LEVELS AT LOW POWER OR SHUTDOWN CONDITIONS.

[139] MILLSTONE 1 DOCKET 50-245 LER 87-001 REV 01
UPDATE ON 4160V DISTRIBUTION LOAD CENTER INSULATION REPLACED DUE TO A MANUFACTURING DEFECT.
EVENT DATE: 011387 REPORT DATE: 090987 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 206033) ON JANUARY 13, 1987 AT 1100 HOURS, VISUAL INSPECTIONS PERFORMED ON ALL 4160 VAC LOAD CENTERS AND THEIR ASSOCIATED BUS TIES REVEALED CRACKING ALONG THE HORIZONTAL NORYL INSULATORS. THE PROBLEM WAS ATTRIBUTED TO A GENERAL ELECTRIC CO. MANUFACTURING DEFECT. THERE WERE NO SAFETY CONSEQUENCES RESULTING FROM THIS OCCURRENCE. IN AUGUST OF 1987, THE BUS INSULATION ON 6 OUT OF 8 4160 VAC BUSES, WAS REPLACED WITH BAYBLEND INSULATION WHILE THE UNIT WAS IN A SCHEDULED REFUELING OUTAGE. BAYBLEND INSULATION WAS RECOMMENDED BY GENERAL ELECTRIC CO. TO BE THE SUITABLE REPLACEMENT FOR NORYL INSULATION. TWO 4160 VAC LOAD CENTERS AND ALL 4160 VAC BUS TIES WERE NOT REPLACED. IT WAS DETERMINED THAT THE INSULATION CRACKING ON THESE BUSES WAS ACCEPTABLE TO MAINTAIN THE BUSES ENERGIZED TO SUPPORT CRITICAL OUTAGE PLANT MODIFICATIONS AND POSTPONED TO THE NEXT REFUELING OUTAGE FOR REPLACEMENT.

[140] MILLSTONE 1 DOCKET 50-245 LER 87-012 REV 01
UPDATE ON DIESEL GENERATOR CEILING FIRE COATING DEGRADATION.
EVENT DATE: 051987 REPORT DATE: 092987 NSSS: GE TYPE: BWR

(NSIC 206481) ON MAY 19, 1987, AT 1130 HOURS, WITH THE PLANT AT 100% POWER, THE PLANT WAS IN THE PROCESS OF INSTALLING A MONORAIL AND PADEYE LIFTING ASSEMBLY IN THE EMERGENCY DIESEL ENCLOSURE WHEN IT WAS DISCOVERED THE OVERHEAD FIREPROOFING BLANKET MATERIAL WAS INADEQUATE TO PROVIDE THE 3 HOUR FIRE RESISTANCE RATING. THE PRIMARY CAUSE FOR THIS INADEQUACY CAN NOT BE DETERMINED THROUGH AVAILABLE DOCUMENTATION. SINCE THE DISCOVERY, A CONTINUOUS FIRE WATCH HAS BEEN ESTABLISHED IN THE ENCLOSURE AND STEPS HAVE BEEN TAKEN TO BEGIN INSTALLING A REPLACEMENT LAYER OF FIREPROOFING MATERIAL BEFORE THE END OF THE UPCOMING REFUELING OUTAGE. INSTALLATION OF THE REPLACEMENT LAYER OF FIRECOATING WAS STARTED IN AUGUST, 1987, AFTER THE 1987 REFUELING OUTAGE. THE REPLACEMENT FIRECOATING INSTALLATION WAS COMPLETED ON 9/18/87. AFTER A POST-INSTALLATION INSPECTION VERIFIED THE COATING WAS SATISFACTORY, THE CONTINUOUS FIRE WATCH WAS DISCONTINUED AND NORMAL FIRE DETECTION AND DELUGE SYSTEMS WERE DECLARED OPERATIONAL.

[141] MILLSTONE 1 DOCKET 50-245 LER 87-015 REV 01
UPDATE ON LOCAL LEAK RATE TEST FAILURES.
EVENT DATE: 060687 REPORT DATE: 092987 NSSS: GE TYPE: BWR
VENDOR: ALLIS CHALMERS
CHAPMAN VALVE & MFG
CRANE COMPANY
MASONEILAN INTERNATIONAL, INC.

TARGET ROCK CORP.
VELAN VALVE CORP.

(NSIC 206482) 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.

[142] MILLSTONE 1 DOCKET 50-245 LER 87-026
HYDRAULIC SNUBBER FAILURES CAUSED BY VARIOUS MECHANICAL AND ADJUSTMENT DEFICIENCIES.
EVENT DATE: 080387 REPORT DATE: 090187 NSSS: GE TYPE: BWR
VENDOR: BERGEN-PATTERSON PIPE SUPPORT CORPORATION
ITT GRINNELL

(NSIC 206237) DURING THE 1987 REFUEL OUTAGE, 10% OF THE TOTAL NUMBER OF SAFETY RELATED HYDRAULIC AND MECHANICAL SNUBBERS IN USE AT THE PLANT WERE FUNCTIONALLY TESTED. FOR EVERY FAILURE IDENTIFIED IN THIS REPRESENTATIVE SAMPLE, AN ADDITIONAL 10% WERE FUNCTIONALLY TESTED. THIS TESTING CONTINUED UNTIL NO ADDITIONAL INOPERABLE SNUBBERS WERE DISCOVERED. OF THE TOTAL NUMBER OF SNUBBERS TESTED DURING THE 1987 REFUEL OUTAGE, THERE WERE NO FAILURES FOR MECHANICAL SNUBBERS AND A TOTAL OF NINE (9) FAILURES FOR HYDRAULIC SNUBBERS. AS A RESULT OF THE NINE FAILED HYDRAULIC SNUBBERS, 100% OF THE HYDRAULIC SNUBBERS WERE FUNCTIONALLY TESTED DURING THE 1987 REFUEL OUTAGE. ALL SAFETY RELATED HYDRAULIC AND MECHANICAL SNUBBERS WERE VISUALLY INSPECTED AND FOUND ACCEPTABLE. THE NINE (9) FAILED HYDRAULIC SNUBBERS WERE FURTHER EXAMINED AND FOUND TO HAVE VARIOUS MECHANICAL AND ADJUSTMENT DEFICIENCIES. THE DEFICIENCIES WERE CORRECTED AND THE SNUBBERS WERE RETESTED FOR FUNCTIONAL ACCEPTABILITY.

[143] MILLSTONE 1 DOCKET 50-245 LER 87-032
TURBINE FIRST STAGE PRESSURE BYPASS SWITCHES FAILED DUE TO SET POINT DRIFT.
EVENT DATE: 081187 REPORT DATE: 091087 NSSS: GE TYPE: BWR
VENDOR: BARKSDALE COMPANY

(NSIC 206035) ON AUGUST 11, 1987, AT 1940 HOURS, WHILE THE UNIT WAS IN A COLD SHUTDOWN CONDITION (0 PSIG, 160 DEGREES FAHRENHEIT) DURING A REFUEL OUTAGE, A ROUTINE SURVEILLANCE WAS PERFORMED ON THE TURBINE FIRST STAGE PRESSURE BYPASS SWITCHES. THESE SWITCHES PROVIDE INPUT TO THE REACTOR PROTECTION SYSTEM (SYSTEM CODE JC) TO BYPASS THE LOAD REJECT SCRAM FUNCTION AT POWER LEVELS LESS THAN 45%. ALL FOUR OF THE SUBJECT SWITCHES (MANUFACTURER: BARKSDALE, MODEL: B2T-12SS-CS, RANGE: 50-1200 PSIG) FAILED TO MEET TECHNICAL SPECIFICATION SETPOINT REQUIREMENTS. THE SETPOINT DRIFT WAS ATTRIBUTED TO RELAXATION OF THE BOURDON TUBE DURING THE EXTENDED PERIOD OF DEPRESSURIZATION ASSOCIATED WITH THE REFUEL OUTAGE. (REVIEW OF PAST HISTORY HAS REVEALED THAT SWITCHES OF THIS TYPE ARE PRONE TO SETPOINT DRIFT PROBLEMS.) AT THE TIME THE EVENT WAS IDENTIFIED, THE PRESSURE SWITCHES WERE RE-CALIBRATED WITH ADDITIONAL MARGIN TO PRECLUDE SETPOINT DRIFT FROM EXCEEDING TECHNICAL SPECIFICATION SETPOINT REQUIREMENTS.

[144] MILLSTONE 1 DOCKET 50-245 LER 87-033
INADVERTENT ACTUATION OF "A" LOW PRESSURE COOLANT INJECTION SUBSYSTEM DUE TO PERSONNEL ERROR.
EVENT DATE: 081287 REPORT DATE: 091187 NSSS: GE TYPE: BWR

(NSIC 206036) ON AUGUST 12, 1987, AT 1030 HOURS, WITH THE REACTOR IN A COLD SHUTDOWN CONDITION (0 PSIG, 160 DEGREES FAHRENHEIT), A ROUTINE SURVEILLANCE WAS PERFORMED ON THE AUTOMATIC PRESSURE RELIEF (APR) SYSTEM (SYSTEM CODE BS). DURING PERFORMANCE OF THIS TEST, THE TEST TECHNICIAN ERRONEOUSLY CONNECTED A TEST PLUG TO THE WRONG JACK. THIS RESULTED IN AN INADVERTENT INITIATION OF THE "A" LOW PRESSURE COOLANT INJECTION (LPCI) SUBSYSTEM (SYSTEM CODE BO), CONSISTING OF THE "A" AND "C" LPCI PUMPS AND ASSOCIATED "A" LOOP VALVES. INJECTION OF WATER TO THE VESSEL RESULTED IN AN INCREASE IN LEVEL OF APPROXIMATELY 50", AT WHICH TIME THE AFFECTED PUMPS WERE SECURED.

[145] MILLSTONE 1 DOCKET 50-245 LER 87-028
REACTOR SCRAM WHILE PERFORMING SURVEILLANCE.
EVENT DATE: 081387 REPORT DATE: 091187 NSSS: GE TYPE: BWR

(NSIC 206034) ON AUGUST 13, 1987 AT 1220 HOURS, WHILE PERFORMING THE MAIN STEAM ISOLATION VALVE CLOSURE FUNCTIONAL TEST DURING A PLANNED REFUELING OUTAGE (137 DEGREES FAHRENHEIT, 0 PSIG), A REACTOR SCRAM OCCURRED. THE SCRAM WAS THE RESULT OF THE PERFORMANCE OF A PROCEDURE STEP BEFORE THE INITIAL CONDITIONS FOR THE PROCEDURE WERE SATISFIED. ALL CONTROL RODS WERE FULLY INSERTED PRIOR TO THIS EVENT AND NO SAFETY CONSEQUENCES RESULTED FROM THIS EVENT.

[146] MILLSTONE 1 DOCKET 50-245 LER 87-035
MISSED SURVEILLANCE ON FIRE DETECTION SYSTEM DUE TO PERSONNEL ERROR.
EVENT DATE: 082187 REPORT DATE: 091887 NSSS: GE TYPE: BWR

(NSIC 206037) ON AUGUST 21, 1987, AT 1130 HOURS, WHILE OPERATING AT 97% POWER (530 DEGREES F AND 1020 PSIG), IT WAS DISCOVERED THAT SIX TECHNICAL SPECIFICATION FIRE DETECTION SYSTEMS (IC) WERE NOT COMPLETELY ELECTRICALLY SUPERVISED. TECHNICAL SPECIFICATION SECTION 4.12.E.2 REQUIRES THAT NON-SUPERVISED CIRCUITS BETWEEN THE REQUIRED DETECTION INSTRUMENTS AND THE CONTROL ROOM BE DEMONSTRATED OPERABLE AT LEAST ONCE PER 31 DAYS. THESE FIRE DETECTION SYSTEMS WERE BEING SURVEILLED ON A SIX MONTH FREQUENCY UNDER THE TECHNICAL SPECIFICATION REQUIREMENTS FOR ELECTRICALLY SUPERVISED FIRE DETECTION SYSTEMS, THEREFORE, SIX FIRE DETECTION SYSTEMS WERE FOUND TO BE OVERDUE FOR THEIR REQUIRED SURVEILLANCES. UPON OBSERVATION OF THE OVERDUE SURVEILLANCE, THE SIX FIRE DETECTION SYSTEMS WERE TESTED AND DEMONSTRATED OPERABLE.

[147] MILLSTONE 1 DOCKET 50-245 LER 87-036
REACTOR SCRAM DURING APRM SURVEILLANCE TESTING.
EVENT DATE: 082687 REPORT DATE: 092587 NSSS: GE TYPE: BWR

(NSIC 206447) ON AUGUST 26, 1987 AT 1059 HOURS, INSTRUMENT TECHNICIAN ERROR DURING PERFORMANCE OF SURVEILLANCE PROCEDURE SP 404C, APRM CALIBRATION AND FUNCTIONAL TEST, RESULTED IN A REACTOR SCRAM FROM FULL POWER. IN NORMAL RESPONSE TO THE SCRAM, THE STANDBY GAS TREATMENT SYSTEM INITIATED DUE TO A GROUP II ISOLATION ON REACTOR VESSEL LOW WATER LEVEL. THERE WERE NO SAFETY CONSEQUENCES.

[148] MILLSTONE 1 DOCKET 50-245 LER 87-038
REACTOR SCRAM ON LOW AIR HEADER PRESSURE.
EVENT DATE: 090387 REPORT DATE: 100287 NSSS: GE TYPE: BWR
VENDOR: INGERSOLL-RAND CO.

(NSIC 206580) ON SEPTEMBER 3, 1987 AT 1421 HOURS, WHILE OPERATING AT 100% POWER, A REACTOR SCRAM OCCURRED DUE TO LOW SCRAM AIR HEADER PRESSURE THAT RESULTED FROM A LOSS OF INSTRUMENT AIR (LD) PRESSURE. AS A RESULT OF THE SCRAM, A LOW REACTOR WATER LEVEL EXISTED WHICH CAUSED A GROUP II ISOLATION SIGNAL, AND HENCE THE INITIATION OF THE STANDBY GAS TREATMENT SYSTEM (SBGT) (BH). THE REACTOR WAS

BROUGHT TO A SAFE SHUTDOWN AND THERE WERE NO SAFETY CONSEQUENCES AS A RESULT OF THIS EVENT.

[149] MILLSTONE 3 DOCKET 50-423 LER 86-050 REV 02
UPDATE ON AREA TEMPERATURE ELEMENT IN MAIN STEAM VALVE BUILDING EXCEEDING LIMITS.
EVENT DATE: 090286 REPORT DATE: 092587 NSSS: WE TYPE: PWR

(NSIC 206478) THIS SPECIAL REPORT REVISION IS BEING SUBMITTED TO PROVIDE AN UPDATED STATUS TO SPECIAL REPORT LER 86-050-01. SPECIAL REPORT LER 86-050-01 WAS SUBMITTED PURSUANT TO PLANT TECHNICAL SPECIFICATIONS 3.7.14B AND 6.9.2 TO REPORT AREA TEMPERATURE EXCURSIONS IN THE MAIN STEAM VALVE BUILDING. PLANT TECHNICAL SPECIFICATION 3.7.14B REQUIRES THAT A SPECIAL REPORT BE SUBMITTED TO THE NRC IF ONE OR MORE AREAS EXCEED THE SPECIFIED TEMPERATURE LIMIT BY LESS THAN 20 DEGREES FAHRENHEIT FOR MORE THAN 8 HOURS. LER 86-050-01 REPORTED THAT MAIN STEAM VALVE BUILDING AREA MS-01 HAD EXCEEDED THE 120 DEGREE FAHRENHEIT SPECIFIED LIMIT ON NUMEROUS OCCASIONS. AREA TEMPERATURE ELEMENT 3ECS-TE119, LOCATED ON THE TOP FLOOR ELEVATION 71'-2" IS THE ONLY ELEMENT WITHIN THE BUILDING EXCEEDING THE LIMITS. AT 0139 ON 9/2/86 3ECS-TE119 REACHED A TEMPERATURE OF 120.5 DEGREES FAHRENHEIT. THE PLANT ENTERED THE ACTION STATEMENT AT 0939. ALL ENVIRONMENTALLY QUALIFIED EQUIPMENT WAS VERIFIED TO BE OPERABLE. DUE TO NUMEROUS TEMPERATURE EXCURSIONS IN THIS AREA, A PERMANENT MODIFICATION IS BEING INSTALLED. A SUPPLEMENT TO THIS REPORT WILL BE ISSUED BY 10/01/88. THIS SPECIAL REPORT IS BEING SUBMITTED PURSUANT TO PLANT TECHNICAL SPECIFICATIONS 3.7.14B AND 6.9.2.

[150] MILLSTONE 3 DOCKET 50-423 LER 87-033
REFUELING WATER STORAGE TANK LEVEL BELOW PLANT TECHNICAL SPECIFICATIONS DUE TO INCORRECT LEVEL TRANSMITTER CALIBRATION AND PERSONNEL ERROR.
EVENT DATE: 081487 REPORT DATE: 091487 NSSS: WE TYPE: PWR

(NSIC 206092) ON AUGUST 14, 1987 AT 1400 HOURS, 100% POWER, IT WAS CONFIRMED THAT THERE WAS A CALIBRATION ERROR OF APPROXIMATELY 17,000 GALLONS IN THE NON-CONSERVATIVE DIRECTION FOR THE REFUELING WATER STORAGE TANK (RWST) LEVEL TRANSMITTERS. THE DISCOVERY WAS MADE A DAY EARLIER WHILE INVESTIGATING THE RWST LOW LEVEL ALARM. THE ERROR WAS CAUSED BY AN INCORRECT ASSUMPTION MADE DURING INITIAL CALIBRATION. IMMEDIATE OPERATOR ACTION WAS TO VERIFY THAT LEVELS IN THE RWST WERE ABOVE THE MINIMUM REQUIREMENT. AS CORRECTIVE ACTION THE LICENSEE HAS RECALIBRATED THE LEVEL TRANSMITTERS AND HAS VERIFIED THAT SIMILAR CALIBRATION ERRORS DO NOT EXIST IN ITS OTHER SAFETY RELATED TANKS. ON AUGUST 22, 1987 AT 0705 HOURS, 100% POWER, A SHIFT SUPERVISOR DISCOVERED THAT RWST LEVEL WAS 33,000 GALLONS BELOW THE MINIMUM REQUIREMENT. IMMEDIATE OPERATOR ACTION WAS TO RESTORE RWST LEVEL ABOVE THE REQUIREMENTS OF THE TECHNICAL SPECIFICATIONS. THE LOW LEVEL WAS ATTRIBUTED TO THE BORATION OF THE SPENT FUEL POOL COOLING AND PURIFICATION DEMINERALIZER ON AUGUST 20, 1987 USING THE RWST AS THE WATER SOURCE. AS CORRECTIVE ACTION IMPROVED GUIDANCE AND COMMUNICATION HAS BEEN PROVIDED IN THE PROCEDURE FOR BORATING THE SFC DEMINERALIZER.

[151] NINE MILE POINT 1 DOCKET 50-220 LER 87-013
REACTOR BUILDING EMERGENCY VENTILATION INITIATION CAUSED BY RELAY FAILURE.
EVENT DATE: 080287 REPORT DATE: 090187 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 206236) ON AUGUST 2, 1987, NINE MILE POINT UNIT 1 (NMP1) WAS OPERATING AT 89% POWER WITH THE MODE SWITCH IN THE "RUN" POSITION. AT 1300 HOURS THE UNIT EXPERIENCED A TRIP OF REACTOR PROTECTION SYSTEM (RPS) MOTOR GENERATOR (MG) SET 162. SUBSEQUENTLY, RPS BUS 11 WAS DEENERGIZED. THIS RESULTED IN A ONE-HALF SCRAM, REACTOR BUILDING EMERGENCY VENTILATION (RBEV) INITIATION, CONTROL ROOM EMERGENCY VENTILATION (CREV) INITIATION, AND AN ISOLATION OF THE REACTOR WATER CLEANUP SYSTEM (RWCU). UPON INVESTIGATION, IT WAS FOUND THAT THE COIL OF THE PNUEMATIC

TIMING RELAY IN THE PRIMARY PROTECTIVE RELAYING CIRCUIT OF MG SET 162 HAD FAILED. THIS CAUSED THE FUSE IN THAT CIRCUIT TO BLOW, DEENERGIZING ONE OF TWO SERIES CONTACTORS AND ISOLATING RPS BUS 11 FROM ITS POWER SUPPLY. IMMEDIATE CORRECTIVE ACTION INCLUDED TRANSFERRING RPS BUS 11 TO THE MAINTENANCE BUS, RESETTING THE ONE-HALF SCRAM, RESTORING THE NORMAL REACTOR BUILDING AND CONTROL ROOM VENTILATION, AND RETURNING THE RWCU TO SERVICE. SUBSEQUENT CORRECTIVE ACTION CONSISTED OF REPLACING THE BLOWN FUSE AND THE PNUEMATIC TIMING RELAY UNDER A STATION WORK REQUEST, AND RETURNING RPS BUS 11 TO MG SET 162. ADDITIONAL CORRECTIVE ACTION IS NOT DEEMED NECESSARY AT THIS TIME.

[152] NINE MILE POINT 2 DOCKET 50-410 LER 87-032 REV 01
UPDATE ON REACTOR CLEANUP SYSTEM ISOLATION ON HIGH DIFFERENTIAL FLOW OSCILLATIONS DUE TO DESIGN AND CONSTRUCTION DEFICIENCIES.
EVENT DATE: 061287 REPORT DATE: 091187 NSSS: GE TYPE: BWR

(NSIC 206081) ON JUNE 12, 1987 AT 2121 HOURS, NINE MILE POINT UNIT 2 (NMP2) EXPERIENCED ACTUATION OF AN ENGINEERED SAFETY FEATURE (ESF), SPECIFICALLY, ISOLATION OF THE REACTOR WATER CLEANUP (RWCU) SYSTEM. AT THE TIME OF THE EVENT, THE PLANT WAS IN HOT SHUTDOWN CONDITION WITH THE REACTOR MODE SWITCH IN "SHUTDOWN". REACTOR PRESSURE AND TEMPERATURE WERE APPROXIMATELY 583 POUNDS PER SQUARE INCH GAUGE (PSIG) AND 482F, RESPECTIVELY. ALTHOUGH AT THE TIME OF THE EVENT NO CONCLUSIONS COULD BE MADE, THE ROOT CAUSE OF THIS EVENT HAS BEEN DETERMINED TO BE A COMBINATION OF DESIGN AND CONSTRUCTION DEFICIENCIES. THE CORRECTIVE ACTIONS FOR THIS EVENT ARE: 1. THE RWCU SUCTION FLOW ELEMENT HAS BEEN PROPERLY ORIENTED. 2. THE FLEX HOSES WILL BE REMOVED AND THE BLOCKING GLOBE VALVES RE-ORIENTED. 3. A SPECIAL TASK FORCE HAS BEEN ASSIGNED TO EVALUATE AND TROUBLESHOOT THE RWCU FLOW TRANSMITTER PROBLEMS. 4. THE REJECT FLOW TRANSMITTER ORIFICE ELEMENT WILL BE RELOCATED.

[153] NINE MILE POINT 2 DOCKET 50-410 LER 87-038
REACTOR CORE ISOLATION COOLING STEAM SUPPLY ISOLATION VALVE CLOSED DUE TO DESIGN DEFICIENCY.
EVENT DATE: 080787 REPORT DATE: 090487 NSSS: GE TYPE: BWR

(NSIC 206268) ON AUGUST 7, 1987 AT 1450 WITH THE REACTOR AT 17% POWER AND THE MODE SWITCH IN "RUN", NINE MILE POINT UNIT 2 (NMP2) EXPERIENCED AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION. THIS EVENT CONSISTED OF A REACTOR CORE ISOLATION COOLING (RCIC) PRIMARY CONTAINMENT VALVE ISOLATION. RCIC STEAM SUPPLY OUTBOARD ISOLATION VALVE 2ICS*MOV121 CLOSED DUE TO A FALSE RCIC HIGH STEAM FLOW/INSTRUMENT LINE BREAK SIGNAL. THE MOST PROBABLE CAUSE OF THE SIGNAL WAS PRESSURE OSCILLATIONS IN THE RCIC STEAM SUPPLY LINE. THE ROOT CAUSE OF THE EVENT HAS BEEN DETERMINED TO BE A DESIGN DEFICIENCY. THE SETPOINT OF THE TRIP UNIT PRODUCING THE ISOLATION HAS BEEN EVALUATED AS BEING TOO RESTRICTIVE ALLOWING PRESSURE OSCILLATIONS IN THE STEAM SUPPLY TO ISOLATE THE RCIC TURBINE. IMMEDIATE CORRECTIVE ACTION WAS TO WALK DOWN THE RCIC'S STEAM LINE TO IDENTIFY ANY LEAKS. NONE WERE FOUND. A WORK REQUEST (WR 123778) WAS THEN ISSUED TO INVESTIGATE THE EVENT AND TRACK CORRECTIVE ACTIONS. AN ENGINEERING AND DESIGN COORDINATION REPORT (E&DCR C95102) WAS LATER SUBMITTED TO REVISE THE ISOLATION TRIP UNIT SETPOINTS.

[154] NINE MILE POINT 2 DOCKET 50-410 LER 87-049
TWO STANDBY GAS TREATMENT SYSTEM INITIATIONS DUE TO A LOW FLOW CONDITION.
EVENT DATE: 082587 REPORT DATE: 092387 NSSS: GE TYPE: BWR

(NSIC 206476) ON AUGUST 25, 1987 AT 0645 WITH THE REACTOR AT APPROXIMATELY 22% POWER AND THE MODE SWITCH IN "RUN", NINE MILE POINT UNIT 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. THE CAUSE OF THE SECOND EVENT IS UNKNOWN, AND THE INVESTIGATION IS ONGOING. A SUPPLEMENT TO LER 87-49 WILL BE SUBMITTED BY NOVEMBER 30, 1987. CORRECTIVE ACTION WILL INCLUDE REACTOR BUILDING VENTILATION SYSTEM (HVR) TRAINING FOR OPERATORS AND AN INDEPENDENT EVALUATION OF THE SBGTS AUTOMATIC INITIATION LOGIC.

[155] NINE MILE POINT 2 DOCKET 50-410 LER 87-050
AUTOMATIC INITIATION OF THE STANDBY GAS TREATMENT SYSTEM DUE TO A PERSONNEL ERROR CAUSED BY IMPROPER COMMUNICATIONS.
EVENT DATE: 082987 REPORT DATE: 092587 NSSS: GE TYPE: BWR

(NSIC 206475) ON AUGUST, 29, 1987 AT 0016 WITH THE REACTOR IN RUN (OPERATIONAL CONDITION 1) AND AT A POWER LEVEL OF APPROXIMATELY 23% RATED THERMAL CAPACITY, NINE MILE POINT UNIT 2 EXPERIENCED AN AUTOMATIC INITIATION OF THE STANDBY GAS TREATMENT SYSTEM (SBGTS) WHILE PERFORMING THE MONTHLY SURVEILLANCE TEST ON THE REACTOR BUILDING VENTILATION PROCESS RADIATION MONITORS. THE SBGTS SYSTEM WAS SECURED BY 0028 THAT SAME DAY, ENDING THE EVENT. THE ROOT CAUSE FOR THIS EVENT IS PERSONNEL ERROR DUE TO IMPROPER COMMUNICATION. THE CORRECTIVE ACTIONS FOR THIS EVENT ARE: 1. A TRAINING MODIFICATION RECOMMENDATION (TMR) HAS BEEN INITIATED, REQUESTING COMMUNICATIONS TRAINING FOR OPERATORS AND TECHNICIANS. 2. A TMR HAS BEEN SUBMITTED REQUESTING DISCUSSION OF THIS EVENT IN OPERATOR AND TECHNICIAN TRAINING. 3. A SUMMARY OF THIS EVENT WILL BE INCLUDED IN THE OPERATIONS DEPARTMENT LESSONS LEARNED BOOK.

[156] NORTH ANNA 1 DOCKET 50-338 LER 85-031
CONTROL ROOM BOTTLED AIR SYSTEM INOPERABLE.
EVENT DATE: 102985 REPORT DATE: 021486 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: NORTH ANNA 2 (PWR)

(NSIC 206437) ON JANUARY 19, 1986, WITH UNIT 1 IN MODE 1 (POWER OPERATION), THE UNIT 1 CONTROL ROOM BOTTLED AIR SYSTEM WAS DISCOVERED TO BE INOPERABLE SINCE OCTOBER 22, 1985. THE SYSTEM WAS IMMEDIATELY RETURNED TO OPERABLE STATUS. THE ACTION STATEMENT WAS NOT ENTERED ON OCTOBER 22, 1985 BECAUSE OPERATIONS PERSONNEL DID NOT REALIZE A TAG OUT HAD RENDERED THE SYSTEM INOPERABLE. THE ACTION STATEMENT EXPIRED WITHIN SEVEN DAYS. SINCE CONTINUED OPERATION AND IMPROPER ENTRY INTO AN APPLICABLE MODE WAS PROHIBITED BY THE PLANTS TECH SPECS, THIS EVENT IS REPORTABLE PURSUANT TO 10 CFR 50.73(A)(2)(I)(B). THROUGHOUT THIS EVENT THE UNIT 1 TRAIN B, AND UNIT 2 BOTTLED AIR SYSTEM REMAINED OPERABLE, AND COULD ALSO BE MANUALLY ACTUATED. THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM, DESIGNED TO BACKUP THE BOTTLED AIR SYSTEM FOR LONG TERM CONTROL ROOM HABITABILITY, ALSO REMAINED OPERABLE. CORRECTIVE ACTION TO PREVENT RECURRENCE WILL INCLUDE A DISCUSSION OF THIS EVENT AS PART OF LICENSED OPERATOR RETRAINING. IN ADDITION, GUIDELINES WILL BE PROVIDED AS TO WHICH COMPONENTS, WHEN TAKEN OUT OF SERVICE, WILL RENDER THE SYSTEM INOPERABLE AND REQUIRE ENTRY INTO THE ACTION STATEMENT. ALSO, AN ONGOING INVESTIGATION, TO DETERMINE THE ROOT CAUSE OF THIS EVENT WILL CONTINUE.

[157] NORTH ANNA 1 DOCKET 50-338 LER 87-007 REV 01
UPDATE ON LOSS OF RESIDUAL HEAT REMOVAL CAPABILITY DUE TO A BLOWN FUSE.
EVENT DATE: 042287 REPORT DATE: 100987 NSSS: WE TYPE: PWR
VENDOR: SOLID STATE CONTROLS, INC.

(NSIC 206522) AT 1530 HOURS ON APRIL 22, 1987, WITH UNIT 1 IN MODE 5, THE

RESIDUAL HEAT REMOVAL (RHR) SUCTION LINE WAS ISOLATED WHEN MOV-1701 CLOSED DUE TO A LOSS OF POWER TO THE 120 VAC VITAL BUS (VB) 1-III. MOV-1701 BEING CLOSED RESULTED A TEMPORARY LOSS OF THE RHR SYSTEM. THEREFORE, THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(VII)(B). AT 1515 HOURS ON APRIL 22, 1987, DURING THE PERFORMANCE OF A PERIODIC TEST, THE 1-III INVERTER FAILED, RESULTING IN A LOSS OF POWER TO VB 1-III. VITAL BUS 1-III SUPPLIES POWER TO AN AUXILIARY RELAY FOR PRESSURE CHANNEL P-1403 WHICH PROVIDES THE LOGIC TO CLOSE MOV-1701 ON HIGH RCS PRESSURE. WHEN THIS AUXILIARY RELAY WAS DE-ENERGIZED AN AUTO CLOSURE SIGNAL WAS SENT TO MOV-1701, THEREBY CLOSING MOV-1701. POWER WAS QUICKLY RESTORED TO VB 1-III, MOV-1701 WAS REOPENED AND RHR FLOW WAS RESTORED. THE CAUSE OF THE INVERTER FAILURE WAS DETERMINED TO BE A BLOWN FUSE. TO PREVENT RECURRENCE OF THIS TYPE EVENT, NORTH ANNA IS ACTIVELY PURSUING A TECHNICAL SPECIFICATION CHANGE TO ELIMINATE THIS RHR INTERLOCK.

[158] NORTH ANNA 1 DOCKET 50-338 LER 87-009 REV 01
UPDATE ON MAIN STEAM SAFETY VALVES SETPOINTS BELOW TECHNICAL SPECIFICATION MINIMUM.
EVENT DATE: 050887 REPORT DATE: 091787 NSSS: WE TYPE: PWR
VENDOR: CROSBY VALVE & GAGE CO.

(NSIC 206457) AT 1100 HOURS ON MAY 8, 1987, WITH UNIT 1 IN MODE 6, ELEVEN (11) OF FIFTEEN (15) MAIN STEAM LINE CODE SAFETY VALVES (MSSVS) "AS FOUND" SETPOINT PRESSURES WERE BELOW THE MINIMUM VALUE ALLOWED BY TECHNICAL SPECIFICATION 3.7.1.1. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(I)(B). ON MAY 1, 1987, ALL 15 MSSVS WERE SENT TO WYLE LABS FOR TESTING. ELEVEN OF THE FIFTEEN (15) MSSVS FAILED THE SETPOINT PRESSURE TEST. ALL ELEVEN MSSVS HAD LOW SETPOINTS AND WERE SUBSEQUENTLY READJUSTED TO WITHIN THE REQUIRED SPECIFICATION. ALL OF THE MSSVS EXHIBITED DISC TO SEAT LEAKAGE. NONE OF THE VALVES LEAKED DURING THE FINAL LEAK TEST FOLLOWING INSPECTION, CLEANING, AND REFURBISHMENT. A SAFETY EVALUATION HAS BEEN PERFORMED FOR THIS EVENT, AND THE RESULTS CONCLUDE THAT THE SAFETY ANALYSIS CONTAINED IN CHAPTER 15 OF THE UFSAR IS STILL BOUNDING AND AN UNREVIEWED SAFETY QUESTION AS DEFINED IN 10CFR50.59 WAS NOT CREATED.

[159] NORTH ANNA 1 DOCKET 50-338 LER 87-016
INOPERABLE TURBINE OVERSPEED PROTECTION SYSTEM DUE TO INCORRECTLY REEASSEMBLED INTERCEPT AND REHEAT STOP VALVES.
EVENT DATE: 081487 REPORT DATE: 091487 NSSS: WE TYPE: PWR

(NSIC 206065) ON AUGUST 14, 1987, AT 1200 HOURS WITH UNIT 1 IN MODE 5, IT WAS DETERMINED THAT REDUNDANT BUTTERFLY VALVES IN THE TURBINE OVERSPEED PROTECTION SYSTEM WOULD NOT HAVE PROPERLY ISOLATED STEAM FROM THE UNIT 1 "B" MOISTURE SEPARATOR REHEATER (MSR) TO THE NO. 1 LOW PRESSURE TURBINE DUE TO THE VALVES BEING INCORRECTLY REASSEMBLIED. THE UNIT 1 "B" REHEAT STOP VALVE WAS DETERMINED TO BE INOPERABLE AT 1300 HOURS ON JULY 8, 1987. THE NUCLEAR REGULATORY COMMISSION (NRC) WAS NOTIFIED AND CONCURRED WITH CONTINUED OPERATION THROUGH JULY 10, 1987, PROVIDED STEAM FLOW WAS ISOLATED FROM THE "B" MSR TO THE LOW PRESSURE TURBINE UTILIZING THE REDUNDANT "B" INTERCEPT VALVE. IT WAS NOT KNOWN AT THAT TIME THAT THE "B" INTERCEPT VALVE WAS NOT FUNCTIONING PROPERLY. THE INOPERABILITY OF THE "B" INTERCEPT VALVE WAS DETERMINED UPON AN INSPECTION AT 1200 HOURS, ON AUGUST 14, 1987. DURING THE OPERATION FROM JUNE 29, 1987, AT 0229 HOURS TO JULY 10, 1987, AT 2400 HOURS, TURBINE OVERSPEED PROTECTION WAS NOT PROVIDED AS REQUIRED BY TECHNICAL SPECIFICATION 3.7.1.7. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73 (A)(2)(I)(B). THE ROOT CAUSE OF THE INCORRECT REASSEMBLY WAS DETERMINED TO BE INADEQUATE MAINTENANCE PROCEDURES, AND A FAILURE TO SPECIFY COMPONENTS FOR REASSEMBLY.

[160] NORTH ANNA 1 DOCKET 50-338 LER 87-018 REV 01
UPDATE ON NON-ENVIRONMENTALLY QUALIFIED LIMITORQUE MOTOR.
EVENT DATE: 081987 REPORT DATE: 100987 NSSS: WE TYPE: PWR

(NSIC 206568) AT 1106 HOURS ON AUGUST 19, 1987, WITH UNIT 1 IN MODE 5, A LIMITORQUE MOTOR ON A MOTOR OPERATED VALVE (1-SW MOV-103C) USED TO ISOLATE SERVICE WATER (SW) TO THE INLET OF A RECIRCULATION SPRAY (RS) HEAT EXCHANGER, WAS DETERMINED TO LACK ENVIRONMENTAL QUALIFICATION DOCUMENTATION. PURSUANT TO GENERIC LETTER 85-15, THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(I)(B). THE VALVE MOTOR, INSTALLED ON JUNE 15, 1987, HAD BEEN REMOVED FROM A SPARE MOV TAGGED AS ENVIRONMENTALLY QUALIFIED. THE CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO INADEQUATE TRAINING AND PROCEDURAL CONTROLS ON ENVIRONMENTAL QUALIFICATION REQUIREMENTS AT THE TIME THE SPARE MOV WAS RECEIVED IN 1984. AS AN IMMEDIATE CORRECTIVE ACTION, THE MOTOR WAS REPLACED WITH A MOTOR WHICH HAS ENVIRONMENTAL QUALIFICATION DOCUMENTATION. SINCE OCTOBER OF 1984, ADEQUATE TRAINING AND PROCEDURAL CONTROLS HAVE BEEN ESTABLISHED AND WILL PREVENT RECURRENCE OF THIS TYPE EVENT. ANY ADDITIONAL ENVIRONMENTAL QUALIFICATION TRAINING NEEDS WILL BE ASSESSED FOR PERSONNEL PERFORMING PURCHASE ORDER MATERIAL REVIEWS AND APPROVALS AND PERSONNEL PERFORMING RECEIPT INSPECTIONS AS IT APPLIES TO THEIR DISCIPLINE. NO SIGNIFICANT SAFETY CONSEQUENCES EXISTED DURING THIS EVENT BECAUSE 1-SW-MOV-103C REMAINED OPEN. ALSO, ALTERNATE METHODS WERE AVAILABLE TO ISOLATE THE RS HEAT EXCHANGER IN THE EVENT OF TUBE LEAKAGE.

[161] NORTH ANNA 2 DOCKET 50-339 LER 85-011
NOBLE GAS HIGH RANGE EFFLUENT MONITORS INOPERABLE.
EVENT DATE: 091285 REPORT DATE: 112585 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: NORTH ANNA 1 (PWR)
VENDOR: KAMAN SCIENCES CORP.

(NSIC 206438) ON NOVEMBER 9, 1985 AT 0300 HOURS, THE 'A' VENTILATION STACK HIGH RANGE RADIATION MONITOR WAS DECLARED INOPERABLE BECAUSE THE MICROPROCESSOR KEYBOARD WOULD NOT FUNCTION. THE MONITOR WAS RETURNED TO SERVICE ON NOVEMBER 13, 1985 AT 1357 HOURS FOLLOWING REPLACEMENT OF THE MICROPROCESSOR CPU BOARD. ON NOVEMBER 12, 1985 AT 1428 HOURS, THE PROCESS VENT HIGH RANGE RADIATION MONITOR WAS REMOVED FROM SERVICE TO REPLACE THE PROCESS VENT FLOW TRANSMITTER IN ORDER TO MEET THE REQUIREMENTS OF REGULATORY GUIDE 1.97. THIS MONITOR WAS RETURNED TO SERVICE ON NOVEMBER 22, 1985 AT 1424 HOURS. DURING EACH OF THESE EVENTS, UNIT 1 WAS IN MODE 5 AND UNIT 2 WAS IN MODE 1 AT 100 PERCENT POWER. WHILE THE MONITORS WERE OUT OF SERVICE, THE PRE-PLANNED ALTERNATE MONITORING METHOD WAS UTILIZED. EACH OF THESE HIGH RANGE MONITORS WERE OUT OF SERVICE FOR GREATER THAN 72 HOURS AND ARE REPORTABLE PURSUANT TO ACTION STATEMENT 35 OF TECH SPEC 3.3.3.1.

[162] NORTH ANNA 2 DOCKET 50-339 LER 87-002
CHARGING PUMP DISCHARGE CHECK VALVE HANGS OPEN DUE TO INSOLUBLE GRANULAR TYPE SUBSTANCE IN HANGER BRACKET BUSHING.
EVENT DATE: 032387 REPORT DATE: 050787 NSSS: WE TYPE: PWR
VENDOR: VELAN VALVE CORP.

(NSIC 206585) AT 1227 HOURS ON MARCH 23, 1987, WITH UNIT 2 AT 100% POWER, THE DISCHARGE CHECK VALVE (2-CH-208) ON THE "1C" CHARGING PUMP (2-CH-P-1C) FAILED TO SHUT FOLLOWING PUMP SHUTDOWN. THIS RESULTED IN THE LOSS OF CHARGING AND SEAL INJECTION FLOW DUE TO BACKFLOW THROUGH 2-CH-P-1C. THE CAUSE OF THE CHECK VALVE HANGING OPEN WAS EXCESSIVE GRIT IN THE HANGER BRACKET BUSHINGS. AS A CORRECTIVE ACTION, THE BUSHINGS WERE REPLACED. 2-CH-P-1C WAS RETURNED TO SERVICE AT 1610 HOURS ON APRIL 3, 1987. THIS REPORT IS BEING SUBMITTED AS A VOLUNTARY LER DUE TO THE POTENTIAL FOR LOSS OF HIGH HEAD SAFETY INJECTION FLOWPATH.

[163] NORTH ANNA 2 DOCKET 50-339 LER 87-006
REACTOR TRIP CAUSED BY A FAILURE OF INTERMEDIATE RANGE DETECTOR.
EVENT DATE: 082487 REPORT DATE: 092287 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 206458) ON AUGUST 24, 1987, AT 0111 HOURS, WITH UNIT 2 IN MODE 3, AND SUBCRITICAL FOLLOWING A NORMAL SHUTDOWN, A REACTOR TRIP OCCURRED FROM AN INTERMEDIATE RANGE HIGH NEUTRON FLUX LEVEL SIGNAL (1/2 CHANNELS GREATER THAN THE CURRENT EQUIVALENT OF 25 PERCENT OF RATED THERMAL POWER). UNIT 2 HAD JUST COMPLETED THE REACTOR SHUTDOWN IN ORDER TO ENTER A SCHEDULED REFUELING OUTAGE. THE CAUSE OF THE REACTOR TRIP WAS A FAILING INTERMEDIATE RANGE NEUTRON DETECTOR WHOSE OUTPUT SPIKED AND EXCEEDED THE TRIP SETPOINT. THE DETECTOR WILL BE REPLACED DURING THE PRESENT REFUELING OUTAGE. NO SAFETY CONSEQUENCES OCCURRED AS A RESULT OF THE TRIP BECAUSE THE REACTOR WAS SHUTDOWN, AND THE EQUIPMENT AND CONTROL SYSTEMS RESPONDED AS EXPECTED. THIS EVENT IS REPORTABLE PURSUANT TO THE REQUIREMENTS OF 10CFR50.73(A)(2)(IV).

[164] OYSTER CREEK DOCKET 50-219 LER 87-030
LIGHTNING ARRESTOR INSULATOR FAILURE INDUCED VOLTAGE TRANSIENT CAUSES CONTAINMENT ISOLATION AND SBGTS INITIATION DUE TO AUTOMATIC BUS TRANSFER TIME EXCEEDING RPS RELAY DROPOUT TIME.
EVENT DATE: 042287 REPORT DATE: 091187 NSSS: GE TYPE: BWR

(NSIC 206032) ON APRIL 22, 1986 AT APPROXIMATELY 0700 HOURS, PRIMARY AND SECONDARY CONTAINMENTS ISOLATED AND THE STANDBY GAS TREATMENT SYSTEM (SBGTS) INITIATED AS A RESULT OF A VOLTAGE TRANSIENT CAUSED BY A LIGHTNING ARRESTOR INSULATOR FAILURE. THE VOLTAGE TRANSIENT CAUSED VITAL AC POWER PANEL 1 (VACP-1) TO TRANSFER TO ITS ALTERNATE POWER SUPPLY. THE POWER SUPPLY TRANSFER CAUSED SEVERAL REACTOR PROTECTION SYSTEM (RPS) RELAYS TO DEENERGIZE, CAUSING THE CONTAINMENT ISOLATIONS AND SBGTS INITIATION. AT THE TIME OF THIS EVENT, THE REACTOR MODE SWITCH WAS LOCKED IN THE SHUTDOWN POSITION, THE VESSEL HEAD HAD BEEN REMOVED AND THE VESSEL AND REACTOR CAVITY HAD BEEN FLOODED IN PREPARATION FOR REFUELING. THE SHORTEST TRANSFER TIME ACHIEVABLE WITH THE AUTOMATIC TRANSFER SWITCH EXCEEDS THE DROPOUT TIME FOR THE RPS RELAYS. THE ISOLATION SIGNAL WAS RESET AND SBGTS WAS SECURED FOLLOWING THE EVENT. THE SAFETY SIGNIFICANCE OF THIS EVENT IS MINIMAL AS ONLY A CHALLENGE TO CONTAINMENT ISOLATION AND SBGTS INITIATION LOGIC CIRCUITS OCCURRED. TO PREVENT RECURRENCE OF THIS EVENT, ENGINEERING HAS PROPOSED A CONTINUOUS POWER SUPPLY BE CONNECTED TO THE CIRCUIT WHICH CONTAINS THE RELAYS THAT CAUSE CONTAINMENT ISOLATIONS AND SBGTS INITIATIONS WHEN CERTAIN POWER INTERRUPTIONS OCCUR.

[165] OYSTER CREEK DOCKET 50-219 LER 87-029
HIGH REACTOR PRESSURE SCRAM DUE TO AIR LEAK FROM DISLODGED AIR TEST PILOT VALVE CAUSED BY INCORRECT MOUNTING CAP SCREW LENGTH.
EVENT DATE: 073087 REPORT DATE: 090387 NSSS: GE TYPE: BWR
VENDOR: AUTOMATIC VALVE COMPANY

(NSIC 206235) ON JULY 30, 1987 AT 0500 HOURS A HIGH REACTOR PRESSURE SCRAM OCCURRED WHEN AN AIR LEAK CAUSED A MAIN STEAM ISOLATION VALVE (MSIV) TO CLOSE WITH REACTOR POWER GREATER THAN THE LEVEL WHERE ONE MAIN STEAM LINE ALONE IS SUFFICIENT TO MAINTAIN REACTOR PRESSURE BELOW THE SCRAM SETPOINT. THE PLANT WAS OPERATING AT APPROXIMATELY 97% RATED POWER PRIOR TO THIS EVENT WITH REACTOR PRESSURE AT 1020 PSIG AND REACTOR TEMPERATURE AT 524F. DURING A 5% CLOSURE TEST OF MSIV NS04A, THE AIR TEST PILOT VALVE PARTIALLY DISLODGED FROM ITS MOUNTING CAUSING THE MSIV TO CLOSE BEYOND 5% AND STABILIZE AT AN INTERMEDIATE POSITION. THIS PRODUCED A LOW CONTROL AIR PRESSURE ALARM. THE OPERATORS QUICKLY DIAGNOSED THE PROBLEM AND BEGAN REDUCING POWER AND STARTED TWO ADDITIONAL AIR COMPRESSORS. FIFTEEN MINUTES HAD ELAPSED AND REACTOR POWER HAD BEEN REDUCED TO APPROXIMATELY 70% WHEN THE AIR TEST PILOT VALVE DISLODGED COMPLETELY CLOSING THE MSIV. THIS

CAUSED A HIGH REACTOR PRESSURE SCRAM. THE CAUSE OF THIS EVENT IS THE AIR TEST PILOT VALVE MOUNTING CAP SCREWS WERE 1/4 INCH SHORTER THAN VENDOR SPECIFICATIONS. THE DISLODGED AIR TEST PILOT VALVE AND ALL OTHER MSIV AIR PILOT VALVES WERE REINSTALLED WITH PROPER LENGTH CAP SCREWS. THE MSIV AIR PILOT VALVE INSTALLATION PROCEDURE WILL BE REVISED TO IDENTIFY THE PROPER LENGTH SCREWS FOR FUTURE MAINTENANCE.

[166] OYSTER CREEK DOCKET 50-219 LER 87-027
ELECTRICAL STORM INDUCED CONTAINMENT ISOLATION AND STANDBY GAS TREATMENT SYSTEM INITIATION DUE TO AUTOMATIC BUS TRANSFER TIME EXCEEDING RPS RELAY DROPOUT TIME.
EVENT DATE: 073187 REPORT DATE: 082887 NSSS: GE TYPE: BWR

(NSIC 206234) ON JULY 30, 1987 AT APPROXIMATELY 0648 HOURS, AND ON JULY 31, 1987 AT APPROXIMATELY 0355 AND 0820 HOURS PRIMARY AND SECONDARY CONTAINMENT ISOLATIONS WITH STANDBY GAS TREATMENT SYSTEM (SBGTS) INITIATIONS OCCURRED AS A RESULT OF AN AUTOMATIC ELECTRICAL BUS TRANSFER CAUSED BY A LIGHTNING STRIKE OF THE 34.5 KV OFFSITE POWER SUPPLY. AT THE TIME THE REACTOR WAS IN THE REFUEL MODE AND WAS IN THE PROCESS OF A CONTROLLED COOLDOWN FOLLOWING A SCRAM WHICH OCCURRED JULY 30, 1987 AT 0459 HOURS. THE EVENTS OCCURRED WHEN LIGHTNING STRUCK THE 34.5 KV LINES OUTSIDE THE PLANT CAUSING VOLTAGE TRANSIENTS IN THE LINES. THE VOLTAGE TRANSIENTS CAUSED VITAL AC POWER PANEL 1 (VACP-1) TO TRANSFER TO ITS ALTERNATE POWER SUPPLY. THE POWER SUPPLY TRANSFERS CAUSED SEVERAL REACTOR PROTECTION SYSTEM (RPS) RELAYS TO DE-ENERGIZE, CAUSING THE CONTAINMENT ISOLATIONS AND SBGTS INITIATIONS. THE ISOLATION SIGNAL WAS RESET MANUALLY EACH TIME AND SBGTS WAS SECURED. THE SAFETY SIGNIFICANCE OF THIS EVENT IS MINIMAL AS ONLY A CHALLENGE TO THE CONTAINMENT ISOLATIONS AND SBGTS OCCURRED. A PREVIOUS EVALUATION DISCOVERED THAT THE SHORTEST TRANSFER TIME ACHIEVABLE WITH THE INSTALLED ALTERNATE POWER AUTOMATIC TRANSFER SWITCH EXCEEDS THE DROPOUT TIME FOR THE ASSOCIATED RPS RELAYS.

[167] OYSTER CREEK DOCKET 50-219 LER 87-028
MAIN STEAM ISOLATION VALVE CLOSURE CAUSED BY DESIGN DEFICIENCY DURING SURVEILLANCE TEST.
EVENT DATE: 080287 REPORT DATE: 090487 NSSS: GE TYPE: BWR

(NSIC 206427) ON 8/2/87 AT 0340 HRS A MAIN STEAM ISOLATION VALVE (MSIV) CLOSURE WAS INADVERTENTLY INITIATED DURING AN INTERMEDIATE RANGE NEUTRON MONITOR (IRM) FRONT PANEL SURVEILLANCE. AT THE TIME THE PLANT WAS IN THE REFUEL MODE WITH REACTOR COOLANT TEMPERATURE AT 130F. TO TEST THE OPERATION OF THE IRM RANGE 10 INTERLOCK WITH THE MAIN STEAM LINE LOW PRESSURE MSIV AUTOMATIC CLOSURE SIGNAL, ONE CHANNEL OF THE REACTOR PROTECTION SYSTEM (RPS) IS JUMPERED TO PREVENT MSIV CLOSURE WHILE A TRIP SIGNAL IS INTRODUCED. WHEN TECHNICIANS INSTALLED THE REQUIRED JUMPERS AND TESTED THEM FOR TIGHTNESS, ONE END OF ONE JUMPER FELL OFF AND SHORTED TO GROUND, BLOWING AN RPS FUSE. THIS CAUSED AN MSIV ISOLATION SIGNAL IN RPS CHANNEL 2. WHEN ANOTHER ISOLATION SIGNAL WAS INTENTIONALLY INTRODUCED LATER IN THE SURVEILLANCE AN MSIV CLOSURE OCCURRED. INVESTIGATION UNCOVERING THE BLOWN FUSE, WHICH WAS REPLACED, AND THE SURVEILLANCE WAS COMPLETED AT 0418 HRS. THE ROOT CAUSE OF THE EVENT IS THAT THE CIRCUIT BEING TESTED IS NOT CONFIGURED TO FACILITATE TESTING. CONTRIBUTING CAUSES WERE THE JUMPERING TECHNIQUE OF THE TECHNICIAN AND FAILURE OF THE CONTROL ROOM OPERATORS TO FULLY INVESTIGATE A SUSPECTED ACTUATION. THIS EVOLUTION IS NOT PERFORMED DURING POWER OPERATIONS AND AN MSIV CLOSURE WHILE SHUT DOWN HAS MINIMAL EFFECT ON THE REACTOR. THEREFORE THE SAFETY SIGNIFICANCE OF THIS EVENT IS MINIMAL.

[168] OYSTER CREEK DOCKET 50-219 LER 87-032
HYDROGEN ANALYZER NOT CALIBRATED IN ACCORDANCE WITH TECH SPEC REQUIREMENTS DUE TO INADEQUATE REVIEW OF TECH SPEC AMENDMENTYSTER CREEK.
EVENT DATE: 082487 REPORT DATE: 092587 NSSS: GE TYPE: BWR

(NSIC 206480) ON AUGUST 24, 1987 AT APPROXIMATELY 1615 HOURS, IT WAS DISCOVERED THAT THE AUGMENTED OFFGAS SYSTEM H2 ANALYZER HAD BEEN CALIBRATED USING A STANDARD GAS SAMPLE OF A KNOWN VOLUME OF H2 IN AIR RATHER THAN IN N2 AS REQUIRED BY TECHNICAL SPECIFICATIONS (TS). THE PLANT HAD BEEN OPERATING IN VARIOUS POWER LEVELS AND MODES DURING THE TIME PERIOD THE H2 ANALYZER WAS NOT CALIBRATED IN ACCORDANCE WITH TS REQUIREMENTS. THIS QUARTERLY TS REQUIREMENT WAS PART OF THE RADIOLOGICAL EFFLUENT TECHNICAL SPECIFICATIONS (RETS) AMENDMENT WHICH BECAME EFFECTIVE NOVEMBER 20, 1986. ON SEPTEMBER 5, 1987, THE AOG SYSTEM H2 ANALYZER WAS CALIBRATED IN ACCORDANCE WITH TS REQUIREMENTS USING A NEWLY DEVELOPED PROCEDURE. THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR IN THAT THE PLANT ENGINEERING REVIEW OF THE RETS AMENDMENT DID NOT IDENTIFY THE RETS INCONSISTENCY WITH THE EXISTING H2 ANALYZER CALIBRATION METHOD NOR DID THEY INITIALLY IDENTIFY THE NEED FOR A PROCEDURE TO BE DEVELOPED FOR THIS CALIBRATION. FOR CORRECTIVE ACTION THIS LER WILL BE REQUIRED READING FOR ALL INVOLVED DEPARTMENTS TO STRESS THE IMPORTANCE OF A THOROUGH REVIEW OF PLANT PROCEDURES AND PRACTICES FOR COMPLIANCE WITH TS CHANGES. SINCE THIS EVENT, OYSTER CREEK LICENSING MANAGEMENT HAS ALTERED ITS POLICY AND IS ISSUING LICENSING ACTIONS ITEMS FOR TRACKING REQUISITE PROCEDURE CHANGES RESULTING FROM TS CHANGES.

[169] OYSTER CREEK DOCKET 50-219 LER 87-031
VIOLATION OF HIGH RADIATION AREA TECHNICAL SPECIFICATIONS CAUSED BY PERSONNEL ERROR DURING RESPONSE TO FIRE ALARM.
EVENT DATE: 082787 REPORT DATE: 092587 NSSS: GE TYPE: BWR

(NSIC 206479) ON 8/27/87 AT 2250 HOURS PLANT PERSONNEL VIOLATED HIGH RADIATION AREA TECH SPECS AND PROCEDURES. AT THE TIME, THE PLANT WAS OPERATING AT 99% POWER. AT 2239 HOURS A MAIN TRANSFORMER/TURBINE AREA FIRE ALARM WAS RECEIVED IN THE CONTROL ROOM. OPERATORS INVESTIGATED AND FOUND NO FIRE AT THE MAIN TRANSFORMER. AT 2242 HOURS A MOISTURE SEPARATOR HIGH WATER LEVEL ALARM WAS RECEIVED AND A SUPERVISOR (SENIOR REACTOR OPERATOR LICENSED) JOINED AN INVESTIGATING CONTROL ROOM OPERATOR AT THE CONDENSER BAY ENTRANCE. THE SPRINKLER SYSTEM TO THE CONDENSER BAY WAS MANUALLY ISOLATED BECAUSE THE SUPERVISOR SUSPECTED THAT THE FIRE WATER WAS SPRAYING ON INSTRUMENTATION AND CAUSING IT TO BEHAVE ERRATICALLY, THREATENING A TURBINE TRIP. BEFORE THE REQUESTED RADIOLOGICAL CONTROLS TECHNICIAN ARRIVED THE SUPERVISOR DIRECTED THE CONTROL ROOM OPERATOR TO CLIMB OVER THE LOCKED HIGH RADIATION AREA DOOR AND OPEN IT FROM THE INSIDE. THE SUPERVISOR BELIEVED THE THREAT OF A POSSIBLE FIRE CONDITION OUTWEIGHED THE HIGH RADIATION AREA ACCESS REQUIREMENTS. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR, AND THE SAFETY SIGNIFICANCE WAS DETERMINED TO BE MINIMAL. NO OVEREXPOSURES OF PERSONNEL RESULTED. CORRECTIVE ACTIONS INCLUDE OPERATOR TRAINING, PROVISION OF MONITORING EQUIPMENT TO OPERATIONS PERSONNEL FOR EMERGENCY USE, AND REVIEW OF THIS EVENT WITH ALL OPERATORS AND THEIR SUPERVISORS.

[170] PALISADES DOCKET 50-255 LER 87-025
ELECTRO-HDYRAULIC CONTROL FLUID SUPPLY LINE FAILURE RESULTS IN MANUAL REACTOR TRIP.
EVENT DATE: 081787 REPORT DATE: 091687 NSSS: CE TYPE: PWR
VENDOR: TELEFLEX, INC.

(NSIC 206038) ON AUGUST 17, 1987 AT 0404, CONTROL ROOM OPERATORS MANUALLY INITIATED A REACTOR TRIP FOLLOWING THE FAILURE OF AN ELECTRO-HYDRAULIC CONTROL (EHC) SYSTEM SUPPLY LINE (TG;PSX). THE REACTOR WAS CRITICAL WITH THE PLANT OPERATING AT 68 PERCENT OF RATED POWER AT THE TIME OF THE EVENT. SHORTLY BEFORE 0404, THE TURBINE GENERATOR GOVERNOR VALVES (TG;FVC) BEGAN TO CLOSE DUE TO A RAPID LOSS IN EHC SYSTEM FLUID PRESSURE. AS PRIMARY COOLANT SYSTEM PRESSURE BEGAN TO INCREASE IN RESPONSE TO GOVERNOR VALVE CLOSURE, THE REACTOR WAS MANUALLY TRIPPED. INVESTIGATION INTO THE LOSS OF EHC FLUID DETERMINED THAT A FLEXIBLE FLUID SUPPLY HOSE TO GOVERNOR VALVE CV-0576 HAD RUPTURED. THE CAUSE OF THE FAILURE WAS DETERMINED TO BE VIBRATION INDUCED FRETTING OF THE HOSES BRAIDED

STAINLESS STEEL SHEATH AT A POINT WHERE TWO HOSES WERE IN CONTACT. THIS FRETTING CAUSED THE STAINLESS STEEL WIRES TO BE ABRADED AND EXHIBIT A "KNIFE-EDGE" APPEARANCE, THEREBY WEAKENING THE HOSE AND INCREASING SUSCEPTIBILITY TO RUPTURE. THE FLEXIBLE HOSE WAS REMOVED FROM ALL FOUR TURBINE GENERATOR GOVERNOR VALVES. DYE PENETRANT EXAMINATIONS WERE PERFORMED ON THE FLARED ENDS OF THE RIGID STAINLESS STEEL TUBING REMOVED PRIOR TO FLEXIBLE HOSE INSTALLATION. NO POSITIVE INDICATIONS OF CRACKING WERE REVEALED. THIS RIGID TUBING WAS THEN REINSTALLED AND THE FLEXIBLE HOSE RESTRAINTS REMOVED.

[171] PALISADES DOCKET 50-255 LER 87-026
LEAKING AIRLOCK DOOR RESULTS IN CONTAINMENT INTEGRITY VIOLATION.
EVENT DATE: 082187 REPORT DATE: 092187 NSSS: CE TYPE: PWR
VENDOR: WOOLLEY, W. J. COMPANY

(NSIC 206039) ON AUGUST 21, 1987 AT APPROXIMATELY 0930, AN AUXILIARY OPERATOR (AO) ENTERING THE ESCAPE AIRLOCK (NH;AL) IDENTIFIED AUDIBLE AIR LEAKAGE FROM CONTAINMENT INTO THE AIRLOCK THROUGH THE CLOSED INNER AIRLOCK DOOR. THE LEAKAGE WAS IDENTIFIED WHEN THE OUTER AIRLOCK DOOR WAS OPENED AND THE REACTOR WAS CRITICAL WITH THE PLANT AT APPROXIMATELY 92 PERCENT OF RATED POWER. THIS CONDITION IS CONTRARY TO PALISADES TECHNICAL SPECIFICATION 3.6.1.A WHICH STATES THAT CONTAINMENT INTEGRITY SHALL NOT BE VIOLATED UNLESS THE REACTOR IS IN COLD SHUTDOWN CONDITION. THE AO WAS ENTERING THE ESCAPE AIRLOCK DURING INITIAL PREPARATIONS FOR PERFORMING SO-4B, "ESCAPE AIRLOCK PENETRATION LEAK TEST". SO-4B HAD BEEN SUCCESSFULLY PERFORMED ON AUGUST 14, 1987, HOWEVER, WAS BEING PERFORMED AGAIN DUE TO CONTAINMENT ENTRIES BEING MADE VIA THE ESCAPE AIRLOCK IN SUPPORT ON PERSONNEL AIRLOCK REPAIRS. ONCE THE LEAKAGE WAS IDENTIFIED, CONTAINMENT ENTRIES WERE MADE VIA THE PERSONNEL AIRLOCK. THE LEAKAGE WAS ELIMINATED BY ADJUSTING THE SEALING SURFACE TO ATTAIN A CONTINUOUS SEAL BEAD. THE LEAKAGE HAS BEEN ATTRIBUTED TO A PROCEDURAL DEFICIENCY. EXISTING PROCEDURES DO NOT IDENTIFY THE NEED FOR SEAL ADJUSTMENT AFTER STRONGBACK REMOVAL DURING SO-4B PERFORMANCE. PROCEDURES ARE BEING MODIFIED TO INCLUDE A POST-TESTING RESTORATION STEP.

[172] PALISADES DOCKET 50-255 LER 87-027
MAIN GENERATOR VOLTAGE REGULATOR FAILURE RESULTS IN REACTOR TRIP.
EVENT DATE: 082387 REPORT DATE: 092287 NSSS: CE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC SUPPLY COMPANY

(NSIC 206040) ON AUGUST 23, 1987 AT 0630, AN AUTOMATIC REACTOR TRIP OCCURRED DUE TO A LOSS OF FIELD IN THE MAIN GENERATOR (TL;PMG). THE REACTOR WAS CRITICAL WITH THE PLANT OPERATING AT 93 PERCENT OF RATED POWER AT THE TIME OF THE EVENT. THE LOSS OF FIELD IN THE MAIN GENERATOR INITIATED A LOSS OF LOAD SIGNAL AND AUTOMATIC REACTOR PROTECTION SYSTEM ACTUATION. THE LOSS OF FIELD WAS CAUSED BY THE FAILURE OF A BIAS SUPPLY POTENTIOMETER IN THE VOLTAGE REGULATOR TRINISTAT POWER AMPLIFIER. SUBSEQUENTLY, AMPLIFIER OUTPUT ESSENTIALLY WENT TO ZERO, CAUSING UNDER-EXCITATION OF THE MAIN GENERATOR AND TRIPPING OF THE MAIN TURBINE WHICH GENERATED THE LOSS OF LOAD SIGNAL. THE PLANT RESPONDED AS DESIGNED TO THE TRANSIENT WITH NO SIGNIFICANT ABNORMALITIES. THE BIAS SUPPLY POTENTIOMETER WAS REPLACED, SATISFACTORILY TESTED AND THE PLANT RETURNED TO SERVICE.

[173] PALISADES DOCKET 50-255 LER 87-028
MAIN FEEDWATER PUMP START RESULTS IN REACTOR CRITICAL WITH PCS LESS THAN 525 DEGREES F.
EVENT DATE: 082587 REPORT DATE: 092487 NSSS: CE TYPE: PWR

(NSIC 206448) ON AUGUST 25, 1987 AT 1315, WITH THE REACTOR CRITICAL, THE PRIMARY COOLANT SYSTEM (PCS) (AB) TEMPERATURE DROPPED BELOW 525 DEGREES F. THE PCS WAS BELOW 525 DEGREES F FOR TWENTY SECONDS WITH A MINIMUM TEMPERATURE OF 524.6 DEGREES F OBTAINED. THE REACTOR WAS TAKEN CRITICAL AT 0222. THIS OCCURRENCE IS

BEING REPORTED AS AN OPERATIONAL CONDITION PROHIBITED BY PLANT TECHNICAL SPECIFICATION 3.1.3C. THE PCS TEMPERATURE DECREASE WAS CAUSED BY THE STARTING OF MAIN FEEDWATER PUMP, P-1B (SJ;P) SHORTLY AFTER A BORON ADDITION TO THE PCS. BORON WAS ADDED TO PCS TO MITIGATE XENON BURNOUT AND MAINTAIN CRITICALITY. THE FEEDWATER PUMP WAS STARTED AS PART OF THE ROUTINE PROCEDURE FOR RETURNING THE SECONDARY SYSTEM TO SERVICE. THE PCS TEMPERATURE DROP WAS ALLEVIATED BY WITHDRAWING CONTROL RODS, THEREBY INCREASING REACTIVITY AND PCS TEMPERATURE.

[174] PALISADES DOCKET 50-255 LER 87-029
PERSONNEL ERROR RESULTS IN INADVERTENT AUXILIARY FEEDWATER SYSTEM ACTUATION.
EVENT DATE: 090187 REPORT DATE: 092887 NSSS: CE TYPE: PWR

(NSIC 206483) ON SEPTEMBER 1, 1987 AT 1310 THE AUXILIARY FEEDWATER ACTUATION SYSTEM (AFAS) WAS INADVERTENTLY ACTUATED. THIS RESULTED IN STARTING AUXILIARY FEEDWATER PUMPS, P-8A AND P-8C (SJ;P). THE ACTUATION OCCURRED DURING THE PERFORMANCE OF TECHNICAL SPECIFICATION SURVEILLANCE PROCEDURE MI-39, "AUXILIARY FEEDWATER ACTUATION LOGIC TEST". THE REACTOR WAS CRITICAL WITH THE PLANT OPERATING AT 92 PERCENT OF RATED POWER AT THE TIME OF THE EVENT. THE INSTRUMENT AND CONTROL TECHNICIAN PERFORMING MI-39 INADVERTENTLY PRESSED TWO STEAM GENERATOR SIGNAL CHANNEL SENSORS INSTEAD OF A SIGNAL CHANNEL AND ACTUATION CHANNEL SENSOR. AS THE TWO SIGNAL CHANNEL SENSORS WERE PRESSED, AFAS WAS ACTUATED AND P-8A AND P-8C STARTED. THE FEEDWATER PUMP ACTUATIONS WERE ACKNOWLEDGED BY CONTROL ROOM OPERATORS WHO SECURED THE FEEDWATER PUMPS. THE AUXILIARY FEEDWATER PUMPS RAN FOR APPROXIMATELY THIRTY SECONDS BEFORE BEING SECURED. THE SURVEILLANCE PROCEDURE IS BEING REVIEWED TO DETERMINE IF ENHANCEMENTS TO THE PROCEDURAL STEPS CAN BE MADE TO MORE CLEARLY DEFINE REQUIRED ACTIONS. THIS EVENT IS ALSO BEING REVIEWED BY THE HUMAN PERFORMANCE EVALUATION SYSTEM COORDINATOR.

[175] PALISADES DOCKET 50-255 LER 87-030
SAFETY INJECTION TANK SAMPLING NOT PERFORMED IN ACCORDANCE WITH TECH SPEC.
EVENT DATE: 090387 REPORT DATE: 093087 NSSS: CE TYPE: PWR

(NSIC 206484) DURING AN AUDIT OF THE PALISADES PLANT CHEMISTRY PROGRAM ON SEPTEMBER 3, 1987, QUALITY ASSURANCE PERSONNEL IDENTIFIED THAT SAFETY INJECTION (SI) TANK, T-82B (BP;TK) WAS NOT BEING SAMPLED FOR BORON CONCENTRATION AT THE FREQUENCY SPECIFIED IN PALISADES TECHNICAL SPECIFICATIONS (TS). A FOOTNOTE WITHIN TABLE 4.2.1 OF TS REQUIRES THAT T-82B BE SAMPLED WEEKLY FOR BORON CONCENTRATION. CONTRARY TO THIS REQUIREMENT, T-82B WAS BEING SAMPLED MONTHLY ALONG WITH SI TANKS, T-82A, C AND D. THE REACTOR WAS CRITICAL WITH THE PLANT OPERATING AT 92 PERCENT OF RATED POWER WHEN THE DISCREPANCY WAS IDENTIFIED. THE REQUIREMENT TO SAMPLE SI TANK T-82B WEEKLY WAS INSTITUTED WITH THE ISSUANCE OF LICENSE AMENDMENT 74. THIS FREQUENCY WAS TO BE MAINTAINED FOR THE REMAINDER OF CYCLE 5 OPERATION UNTIL A LEAKING CHECK VALVE COULD BE REPAIRED. THE INTENT OF THIS AMENDMENT WAS MET AND WEEKLY SAMPLING DISCONTINUED IN JULY 1984. THE WEEKLY SAMPLING REQUIREMENT WAS TO BE DELETED FROM TS WITH THE ISSUANCE OF LICENSE AMENDMENT 101, HOWEVER, DUE TO AN OVERSIGHT WHEN PREPARING THE SUPPORTING TS CHANGE REQUEST, THE REQUIREMENT WAS NOT DELETED. SI TANK T-82B WAS CONSEQUENTLY SAMPLED AFTER THE DISCREPANCY WAS IDENTIFIED.

[176] PALO VERDE 1 DOCKET 50-528 LER 87-001 REV 01
UPDATE ON ESF ACTUATION CAUSED BY A VOLTAGE SPIKE CONCURRENT WITH LOW RADIATION MONITOR SETPOINT.
EVENT DATE: 020387 REPORT DATE: 091887 NSSS: CE TYPE: PWR
VENDOR: KAMAN SCIENCES CORP.

(NSIC 206472) THIS IS A SUPPLEMENT TO LER 1-87-001-00 SUBMITTED ON MARCH 2, 1987. ON FEBRUARY 3, 1987 AT 0232 MST, PALO VERDE UNIT 1 WAS IN MODE 5 (COLD SHUTDOWN) WHEN A CONTROL ROOM ESSENTIAL FILTRATION ACTUATION SIGNAL (CREFAS) WAS ACTUATED

ON CHANNELS A AND B. THE ROOT CAUSE OF THE ACTUATION WAS A VOLTAGE SPIKE, BELIEVED TO BE CAUSED BY EQUIPMENT NOISE IN RADIATION MONITORING UNIT 29, COMPOUNDED BY AN EXCESSIVELY CONSERVATIVE SETPOINT (APPROXIMATELY 1 DECADE LESS THAN THE ESTABLISHED TECHNICAL SPECIFICATION VALUE). EQUIPMENT NOISE IS AN EXPECTED OCCURRENCE AND WOULD NOT HAVE CAUSED THE ACTUATION HAD THE PROPER SETPOINTS BEEN INPUTTED TO THE MONITOR. AS IMMEDIATE CORRECTIVE ACTION, THE SETPOINTS WERE READJUSTED TO BE CONSISTENT WITH THE TECHNICAL SPECIFICATIONS. ON GOING CORRECTIVE ACTION TO PREVENT RECURRENCE INVOLVES REVISING THE DEFAULT ALARM/TRIP SETPOINTS TO BE CONSISTENT WITH THE TECHNICAL SPECIFICATIONS. THERE WERE NO COMPONENT OR SYSTEM FAILURES THAT CONTRIBUTED TO THE EVENT. SIMILAR EVENTS WERE REPORTED IN LERS 1-85-011-02 AND 1-85-031-01.

[177] PALO VERDE 1 DOCKET 50-528 LER 87-023
CHANNEL CHECK NOT PERFORMED DUE TO PERSONNEL ERROR.
EVENT DATE: 080287 REPORT DATE: 083187 NSSS: CE TYPE: PWR

(NSIC 206281) ON AUGUST 10, 1987, IT WAS DISCOVERED THAT AT 2238 MST ON AUGUST 2, 1987, PALO VERDE UNIT 1 WAS IN MODE 1 (POWER OPERATION) AT APPROXIMATELY 100 PERCENT POWER WHEN A UTILITY LICENSED CONTROL ROOM OPERATOR DID NOT PERFORM THE CHANNEL CHECK FOR THE EXCORE LINEAR POWER CHANNEL "C" DURING THE PERFORMANCE OF 41ST-1ZZ33 (MODE 1 SURVEILLANCE LOGS). THE OPERATOR BELIEVED THAT THE LINEAR POWER CHANNEL WAS NOT OPERABLE DUE TO THE LOG POWER CHANNEL "C" BEING INOPERABLE AND THEREFORE DID NOT PERFORM SURVEILLANCE TESTING SINCE SURVEILLANCE TESTING IS NOT REQUIRED TO BE PERFORMED ON INOPERABLE EQUIPMENT. THE ROOT CAUSE OF THE EVENT WAS COGNITIVE PERSONNEL ERROR BY THE UTILITY LICENSED CONTROL ROOM OPERATOR IN THAT HE DID NOT REALIZE THAT THE EXCORE LINEAR POWER CHANNEL WOULD BE OPERABLE WITH THE LOG POWER CHANNEL INOPERABLE. AS CORRECTIVE ACTION, THE CONTROL ROOM OPERATOR WAS COUNSELED ON THE IMPORTANCE OF PROPER SURVEILLANCE TEST PERFORMANCE. SIMILAR EVENTS WERE PREVIOUSLY REPORTED IN LER'S 1-86-038-00 AND 1-86-041-00.

[178] PALO VERDE 1 DOCKET 50-528 LER 87-018
REACTOR TRIP OCCURS DURING SHUTDOWN DUE TO PRESSURE BOUNDARY LEAKAGE.
EVENT DATE: 082787 REPORT DATE: 092587 NSSS: CE TYPE: PWR

(NSIC 206508) AT APPROXIMATELY 2037 ON AUGUST 27, 1987 PALO VERDE UNIT 1 WAS IN MODE 1 (POWER OPERATION) AT APPROXIMATELY 8 PERCENT REACTOR POWER WHEN A REACTOR TRIP OCCURRED DUE TO THE AXIAL SHAPE INDEX OF THE REACTOR CORE. THE TRIP OCCURRED AS THE UNIT WAS SHUTTING DOWN TO EVALUATE A POSSIBLE REACTOR COOLANT SYSTEM (RCS) PRESSURE BOUNDARY LEAK. THE PRESENCE OF A PRESSURE BOUNDARY LEAK WAS CONFIRMED AFTER THE UNIT WAS STABILIZED IN MODE 3 (HOT STANDBY). THE ROOT CAUSE OF THE REACTOR TRIP WAS A DEFICIENT PROCEDURE. TO PREVENT RECURRENCE PROCEDURAL ENHANCEMENTS HAVE BEEN IMPLEMENTED. THE CAUSE OF THE LEAK WAS A CRACKED WELD ON THE UPSTREAM SIDE OF THE ISOLATION VALVE FOR THE FLANGED REFUELING WATER LEVEL INDICATION. THE ROOT CAUSE FOR THE CRACKED WELD APPEARS TO BE A FATIGUE FAILURE, HOWEVER FURTHER ANALYSIS IS STILL BEING CONDUCTED. AS CORRECTIVE ACTION THE VALVE WAS CUT OUT AND REPLACED. FURTHER CORRECTIVE ACTIONS WILL DEPEND ON THE RESULTS OF THE ROOT CAUSE EVALUATION AND WILL BE ADDRESSED IN A SUPPLEMENT TO THIS REPORT.

[179] PALO VERDE 1 DOCKET 50-528 LER 87-024
CORE OPERATING LIMIT SUPERVISORY SYSTEM RENDERED INOPERABLE DUE TO REVERSED CIRCUIT CARDS.
EVENT DATE: 090287 REPORT DATE: 100187 NSSS: CE TYPE: PWR
VENDOR: HONEYWELL CORP.

(NSIC 206525) ON SEPTEMBER 2, 1987 AT 0815 MST, WITH UNIT 1 IN MODE 1 (POWER OPERATION) OPERATING AT APPROXIMATELY 59 PERCENT POWER, IT WAS IDENTIFIED THAT TWO DMA INTERFACE CARDS PROVIDING INPUT TO THE CORE OPERATING LIMIT SUPERVISORY

SYSTEM (COLSS) HAD BEEN INCORRECTLY INSTALLED. THIS RENDERED THE COLSS INOPERABLE. TECH SPECS 4.2.1.2 AND 4.2.4.2 WHICH REQUIRE MONITORING OF CERTAIN REACTOR CORE OPERATING CHARACTERISTICS EVERY 2 HOURS WHEN COLSS IS INOPERABLE ABOVE 20 PERCENT RATED THERMAL POWER WERE NOT MET BEGINNING AT 0106 THROUGH 0815 ON SEPTEMBER 2, 1987. THE ROOT CAUSE OF THE EVENT HAS BEEN DETERMINED TO BE A COGNITIVE PERSONNEL ERROR BY A COMPUTER TECHNICIAN WHO INADVERTENTLY REVERSED THE TWO CARDS DURING TROUBLESHOOTING UNDER APPROVED WORK ORDER DOCUMENTS. THE TECHNICIAN DID NOT OBTAIN INDEPENDENT VERIFICATION OF THE SERIAL NUMBERS AND ADDRESSES FOR THE CARDS WHICH WERE BEING CHANGED DURING THE TROUBLESHOOTING EFFORTS. THIS IS CONTRARY TO AN APPROVED PROCEDURE (WORK ORDER). AS IMMEDIATE CORRECTIVE ACTION, THE CARDS WERE RETURNED TO THEIR CORRECT LOCATIONS AND COLSS WAS RESTORED TO AN OPERABLE STATUS. THE COMPUTER TECHNICIAN HAS BEEN COUNSELLED ON THE IMPORTANCE OF THE ACCURACY OF HIS WORK AND WHEN IT IS NECESSARY TO HAVE INDEPENDENT VERIFICATIONS CONDUCTED.

[180] PALO VERDE 2 DOCKET 50-529 LER 87-015
ESF ACTUATION CAUSED BY SPURIOUS ALARM/TRIP SIGNAL ON RADIATION MONITOR.
EVENT DATE: 081687 REPORT DATE: 090387 NSSS: CE TYPE: PWR

(NSIC 206312) ON AUGUST 16, 1987 AT 0609 MST, 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 IS CURRENTLY IN PROGRESS TO DETERMINE THE CAUSE OF THE EVENT, WHICH WILL INCLUDE AN EVALUATION TO DETERMIN IF UTILIZATION OF THE PIC WAS A CONTRIBUTING FACTOR. THE RESULTS OF THE INVESTIGATION WILL BE PROVIDED IN A SUPPLEMENT TO THIS REPORT. 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.

[181] PALO VERDE 3 DOCKET 50-530 LER 87-001 REV 01
UPDATE ON ENGINEERED SAFETY FEATURE ACTUATION.
EVENT DATE: 0?2887 REPORT DATE: 091887 NSSS: CE TYPE: PWR
VENDOR: ELGAR, CORP.

(NSIC 206473) THIS IS A SUPPLEMENT TO LER 3-87-001-00. AT APPROXIMATELY 1301 MST ON MARCH 28, 1987, PALO VERDE UNIT 3 WAS PREPARING TO ENTER MODE 6 (REFUELING) WHEN CONTROL ROOM ESSENTIAL FILTRATION ACTUATION SIGNALS (CREFAS) AND CONTAINMENT PURGE ISOLATION ACTUATION SIGNALS (CPIAS) WERE RECEIVED ON BOTH CHANNELS OF THE BALANCE OF PLANT ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (BOP ESFAS). THESE BOP ESFAS ACTUATIONS WERE ANNUNCIATED IN THE CONTROL ROOM ON THE MAIN CONTROL BOARD. ALL ASSOCIATED EQUIPMENT RESPONDED AS DESIGNED. CONTROL ROOM PERSONNEL SECURED ALL ACTUATED EQUIPMENT IN ACCORDANCE WITH APPROVED PROCEDURES BY 1413 MST ON MARCH 28, 1987. BOTH CHANNELS OF CREFAS AND CPIAS WERE UNAVAILABLE FOR 1 HOUR AND 12 MINUTES. AT APPROXIMATELY 1301 MST, IMMEDIATELY PRIOR TO THE BOP ESFAS ACTUATION, A "PNL 26 AC UNDV/GND" TRIP WAS RECEIVED ON DISTRIBUTION PANEL 26 AND "120VDC INV B AC/DC STATUS" TROUBLE ALARM WAS RECEIVED FOR INVERTER "B". THE STATIC TRANSFER SWITCH AUTOMATICALLY TRANSFERRED FROM THE "B" INVERTER TO THE 120 VAC VOLTAGE REGULATOR. IMMEDIATELY FOLLOWING THE STATIC TRANSFER SWITCH OPERATION, BOTH CHANNELS OF CREFAS AND CPIAS WERE ACTUATED ON THE BOP ESFAS SYSTEM. THE ROOT CAUSE OF THIS EVENT WAS ATTRIBUTED TO THE SLOW STATIC TRANSFER SWITCH OPERATION AND A FAULTY DC TO DC CONVERTER BOARD. AS A CORRECTIVE ACTION THE STATIC TRANSFER SWITCH WAS RECALIBRATED AND THE DC TO DC CONVERTER BOARD WAS REPLACED. A SIMILAR EV

[182] PALO VERDE 3 DOCKET 50-530 LER 87-002 REV 01
UPDATE ON ENGINEERED SAFETY FEATURE ACTUATION CAUSED BY IMPROPER CALIBRATION OF A STATIC TRANSFER SWITCH AND FAULTY DC CONVERTER BOARD.
EVENT DATE: 061587 REPORT DATE: 091887 NSSS: CE TYPE: PWR
VENDOR: ELGAR, CORP.

(NSIC 206474) THIS IS A SUPPLEMENT TO LER 3-87-002. AT APPROXIMATELY 0611 MST ON JUNE 15, 1987, PALO VERDE UNIT 3 WAS IN MODE 5 (COLD SHUTDOWN) WHEN CONTROL ROOM ESSENTIAL FILTRATION ACTUATION SIGNALS (CREFAS) AND CONTAINMENT PURGE ISOLATION ACTUATION SIGNALS (CPIAS) WERE RECEIVED ON BOTH CHANNELS OF THE BALANCE OF PLANT ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (BOP ESFAS). THESE BOP ESFAS ACTUATIONS WERE ANNUNCIATED IN THE CONTROL ROOM ON THE MAIN CONTROL BOARD. ALL ASSOCIATED EQUIPMENT RESPONDED AS DESIGNED. CONTROL ROOM PERSONNEL SECURED ALL ACTUATED EQUIPMENT IN ACCORDANCE WITH APPROVED PROCEDURES BY 0657 MST ON JUNE 15, 1987. THE DURATION OF THIS EVENT WAS APPROXIMATELY 46 MINUTES. AT APPROXIMATELY 0611 MST, THE DISTRIBUTION PANEL D26 MOMENTARILY LOST POWER AND A "120VAC INV B AC/DC STATUS" TROUBLE ALARM WAS RECEIVED FOR INVERTER "B". AS DESIGNED, THE STATIC TRANSFER SWITCH AUTOMATICALLY TRANSFERRED FROM THE "B" INVERTER TO THE 120 V AC VOLTAGE REGULATOR. IMMEDIATELY FOLLOWING THE STATIC TRANSFER SWITCH OPERATION, BOTH CHANNELS OF CREFAS AND CPIAS WERE ACTUATED ON THE BOP ESFAS SYSTEM. THE ROOT CAUSE OF THIS EVENT WAS ATTRIBUTED TO THE STATIC TRANSFER SWITCH CALIBRATION AT AN UNNECESSARILY HIGH VALUE AND A FAULTY DC TO DC CONVERTER BOARD.

[183] PALO VERDE 3 DOCKET 50-530 LER 87-003
VALVE BOLTING NONCONFORMANCES COULD POTENTIALLY RESULT IN THE INABILITY OF TWO SHUTDOWN COOLING SYSTEM VALVES TO PERFORM THEIR FUNCTIONS.
EVENT DATE: 092987 REPORT DATE: 100287 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: PALO VERDE 1 (PWR)
PALO VERDE 2 (PWR)
VENDOR: BORG-WARNER CORP.
LIMITORQUE CORP.

(NSIC 206526) ON 9/29/87, WITH UNIT 3 IN MODE 5 (COLD SHUTDOWN), A CONDITION WAS IDENTIFIED THAT IF LEFT UNCORRECTED, COULD HAVE RESULTED IN THE IMPROPER OPERATION OF TWO UNIT 3 SHUTDOWN COOLING ISOLATION VALVES IN REDUNDANT TRAINS. THIS DETERMINATION RESULTED FROM AN ENGINEERING EVALUATION REQUEST WHICH HAD BEEN DISPOSITIONED TO RESOLVE IDENTIFIED VALVE YOKE TO MOTOR OPERATOR BOLTING NONCONFORMANCES. THE ROOT CAUSE OF THIS EVENT HAS BEEN DETERMINED TO BE VENDOR ERROR. TWO VALVES SUPPLIED TO THE ARIZONA NUCLEAR POWER PROJECT (ANPP) DID NOT HAVE THE REQUIRED DESIGN CHANGES IMPLEMENTED PRIOR TO SHIPMENT NOR DID THE VENDOR PROVIDE NOTIFICATION OF THESE CHANGES TO ANPP. THE CAUSE FOR THE VENDOR ERROR HAS NOT BEEN DETERMINED AT THIS TIME. AS CORRECTIVE ACTION, THE "AS-INSTALLED" BOLTING HAS BEEN REPLACED WITH ALTERNATIVE BOLTING MATERIAL. AN INSPECTION WAS CONDUCTED TO ENSURE THAT THE SIMILAR SHUTDOWN COOLING ISOLATION VALVES IN UNITS 1 AND 2 HAD ADEQUATE DESIGN MARGIN AND WERE ACCEPTABLE FOR CONTINUED OPERATION. THE RESULTS OF THE INSPECTION INDICATE THAT NO MODIFICATIONS ARE REQUIRED AT THIS TIME. IN ORDER TO PREVENT RECURRENCE, A QUALITY ASSURANCE AND AN ENGINEERING REPRESENTATIVE WILL CONDUCT AN EVALUATION AT THE VENDOR'S FACILITIES TO DETERMINE THE EXTENT OF THESE DEFICIENCIES AND THE POTENTIAL FOR TRANSPORTABILITY TO OTHER VALVES SUPPLIED BY VENDOR.

[184] PEACH BOTTOM 2 DOCKET 50-277 LER 87-015
PRIMARY CONTAINMENT ISOLATION DUE TO PARTIAL LOSS OF OFFSITE POWER.
EVENT DATE: 081687 REPORT DATE: 091587 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: PEACH BOTTOM 3 (BWR)

(NSIC 206043) ON AUGUST 16, 1987, AT 0501 HOURS WITH BOTH UNITS IN COLD SHUTDOWN, A PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) GROUP II INBOARD ISOLATION OCCURRED

ON UNIT 2 AND A PCIS GROUP II OUTBOARD ISOLATION OCCURRED ON UNIT 3. IN ADDITION, THE UNIT 2 REACTOR PROTECTION SYSTEM (RPS) GENERATED A HALF SCRAM SIGNAL. THESE ACTIONS OCCURRED WHEN TREE CONTACT CAUSED A FAULT ON THE 220-05 LINE. BECAUSE OF TEMPORARY WIRING OF THE PROTECTIVE RELAYS ON THE 220-05 LINE, THE FAULT WAS NOT ISOLATED AND RESULTED IN THE TRIPPING OF THE 220-08 LINE WHICH IS THE NO. 2 STARTUP SOURCE. THE LOSS OF THE NO. 2 STARTUP SOURCE RESULTED IN A FAST TRANSFER TO THE ALTERNATE OFFSITE SUPPLY FOR THE FOUR OF THE EIGHT 4KV EMERGENCY BUSES. THE FAST TRANSFER FUNCTIONED AS DESIGNED. THE DIESEL GENERATORS WERE AVAILABLE BUT UNCHALLENGED. THE PCIS FUNCTIONED PROPERLY IN RESPONSE TO THE FAST TRANSFER. THERE WERE NO ADVERSE CONSEQUENCES AS A RESULT OF THIS EVENT. THE NO. 2 STARTUP SOURCE WAS RESTORED WITHIN 3 MINUTES AND THE ISOLATION AND HALF SCRAM SIGNALS WERE RESET WITHIN 13 MINUTES. THE TEMPORARY WIRING OF THE PROTECTIVE RELAYS WAS REVISED AND THE TREE WHICH CAUSED THE FAULT WAS REMOVED. THIS EVENT IS REPORTABLE DUE TO THE ACTUATION OF THE PCIS, AN ENGINEERED SAFETY FEATURE.

[185] PEACH BOTTOM 2 DOCKET 50-277 LER 87-013
LOSS OF POWER TO THE 'A' RHR SYSTEM LOGIC BUS AND PCIS GROUP IIB ISOLATION DUE TO PERSONNEL ERROR.
EVENT DATE: 082087 REPORT DATE: 092187 NSSS: GE TYPE: BWR

(NSIC 206451) ON AUGUST 20, 1987, AT 1015 HOURS AND 1035 HOURS, A PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) GROUP IIB ISOLATION OCCURRED WHILE OPERATING THE RESIDUAL HEAT REMOVAL SYSTEM IN THE SHUTDOWN COOLING MODE. THESE ACTIONS OCCURRED AS THE RESULT OF INSTRUMENTATION AND CONTROLS TECHNICIANS FAILING TO RECOGNIZE THE IMPLICATIONS OF ALL WORK ACTIONS WHILE REPLACING HFA RELAYS AND RELAY COILS IN A SAFETY RELATED CABINET. THIS FAILURE RESULTED IN A REPOSITIONED ELECTRICAL CONDUIT GROUNDING AN ELECTRICAL RELAY. THIS GROUNDED RELAY BLEW FUSE (10A-F2A) AND RESULTED IN A SUBSEQUENT LOSS-OF-POWER TO THE 'A' RESIDUAL HEAT REMOVAL SYSTEM LOGIC BUS. THE LOSS-OF-POWER RESULTED IN DE- ENERGIZATION OF RELAY 10-K114A AND SUBSEQUENT CLOSURE OF THE SHUTDOWN COOLING MODE SUCTION VALVES. THERE WERE NO ADVERSE CONSEQUENCES AS A RESULT OF THIS EVENT. THERE WAS NO OBSERVABLE CHANGE IN REACTOR COOLANT TEMPERATURE. THE FUSE WAS REPLACED AND THE ALARMS AND PCIS ISOLATION SIGNAL WERE RESET. THE SHUTDOWN COOLING MODE WAS REESTABLISHED BY 1138 HOURS. THE APPROPRIATE INSTRUMENTATION AND CONTROLS TECHNICIANS WILL BE PROVIDED INSTRUCTION TO AVOID A REPEAT OF THIS EVENT.

[186] PEACH BOTTOM 2 DOCKET 50-277 LER 87-018
SHUTDOWN COOLING ISOLATION WHEN BLOCKING WAS RE-APPLIED DUE TO PROGRAMMATIC DEFICIENCY.
EVENT DATE: 082887 REPORT DATE: 093087 NSSS: GE TYPE: BWR

(NSIC 206488) ON AUGUST 28, 1987 AT 1048 HOURS THE SHUTDOWN COOLING SUCTION VALVES AUTOMATICALLY CLOSED AND THE 'D' RESIDUAL HEAT REMOVAL (RHR) PUMP TRIPPED. THE CAUSE OF THE ISOLATION WAS THE DEENERGIZATION OF THE 'A' RHR LOGIC WHEN A TEMPORARILY CLEARED SAFETY BLOCK WAS RE-APPLIED. IT WAS NOT RECOGNIZED THAT REAPPLICATION OF THE SAFETY BLOCK, WHICH HAD BEEN TEMPORARILY CLEARED SEVERAL MONTHS EARLIER, WOULD CAUSE THE ISOLATION. THERE WERE NO ADVERSE SAFETY CONSEQUENCES. SHUTDOWN COOLING WAS RETURNED TO SERVICE WITHIN EIGHT MINUTES AND THE REACTOR WATER CLEANUP SYSTEM REMAINED IN SERVICE DURING THAT PERIOD TO REMOVE DECAY HEAT. A WEAKNESS IN THE BLOCKING AND PERMITS PROGRAM ALLOWED THIS TO OCCUR. THERE ARE NO SPECIFIED PRECAUTIONS TO ASSURE THAT RE- APPLICATION OF APPROVED BLOCKING IS ACCEPTABLE GIVEN POSSIBLE CHANGES IN PLANT CONDITIONS. THE BLOCKING AND PERMITS PROGRAM IS BEING REVIEWED TO IDENTIFY IMPROVEMENTS WHICH WOULD PREVENT RECURRENCE OF THIS EVENT AND SIMILAR EVENTS.

[187] PERRY 1 DOCKET 50-440 LER 87-055
FAILED AVERAGE POWER RANGE MONITOR PAGE FLOW CARD RESULTS IN REACTOR PROTECTION SYSTEM ACTUATION.
EVENT DATE: 072987 REPORT DATE: 082887 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 206279) ON JULY 29, 1987 AT 0957, A REACTOR PROTECTION SYSTEM (RPS) ACTUATION OCCURRED DUE TO A TRIP SIGNAL GENERATED BY A NEUTRON MONITORING SYSTEM AVERAGE POWER RANGE MONITOR (APRM). PLANT INSTRUMENTATION AND CONTROL PERSONNEL WERE PERFORMING A WORK ORDER TO REPLACE THE APRM FLOW CARDS FOR ALL EIGHT APRMS WITH NEW CARDS SUPPLIED BY GENERAL ELECTRIC. ALSO DURING THIS TIME, A RPS CHANNEL A/C TRIP SIGNAL WAS PRESENT DUE TO TROUBLESHOOTING OF A MAIN STEAM LINE RADIATION MONITOR. WHEN APRM D WAS TAKEN OUT OF BYPASS AFTER REPLACEMENT OF THE APRM CARD, A RPS CHANNEL B/D TRIP SIGNAL WAS GENERATED. THE CONCURRENT RPS CHANNEL TRIP SIGNALS RESULTED IN A COMPLETE RPS ACTUATION. NO CONTROL ROD MOVEMENT OCCURRED SINCE ALL RODS WERE ALREADY FULLY INSERTED. THE CAUSE OF THIS EVENT WAS A MISPOSITIONING OF SWITCH S-3 ON AN APRM CARD DUE TO MISSING MOUNTING HARDWARE FOR THE SWITCH AS SUPPLIED BY GENERAL ELECTRIC. ALL APRM CARDS HAVE BEEN REPLACED AND SUCCESSFULLY CALIBRATED FOR THE APRMS. GENERAL ELECTRIC HAS REPLACED THE MOUNTING HARDWARE AND REPORTED THAT THE CARD TESTS SATISFACTORILY. NO FURTHER CORRECTION ACTIONS ARE PLANNED.

[188] PERRY 1 DOCKET 50-440 LER 87-057
DEFICIENT CHANGE TO SURVEILLANCE INSTRUCTION RESULTS IN REACTOR PROTECTION SYSTEM ACTUATION DURING SCRAM DISCHARGE VOLUME SENSING LINE FLUSH.
EVENT DATE: 081387 REPORT DATE: 091187 NSSS: GE TYPE: BWR

(NSIC 206185) ON AUGUST 13, 1987 AT 1125, FLUSHING OF A SCRAM DISCHARGE VOLUME (SDV) WATER LEVEL SENSING LINE CAUSED A FALSE HIGH SDV LEVEL SIGNAL AND REACTOR PROTECTION SYSTEM (RPS) ACTUATION. AT THE TIME OF THE EVENT, PLANT TECHNICIANS WERE PERFORMING A FLUSH OF THE SDV LEVEL SENSING LINES IN ACCORDANCE WITH A RECENTLY ISSUED CHANGE TO A SURVEILLANCE INSTRUCTION. THE NEW STEPS DIRECTED THAT FLUSH WATER BE INJECTED INTO A COMMON SENSING LINE FOR THE RPS CHANNEL C AND D TRANSMITTERS, CAUSING AN INDICATED HIGH SDV LEVEL AND SATISFYING THE LOGIC TO INITIATE THE RPS ACTUATION. NO ACTUAL CONTROL ROD MOVEMENT OCCURRED SINCE THE CONTROL RODS WERE ALREADY INSERTED. THE ROOT CAUSE OF THIS EVENT WAS AN INADEQUATE INSTRUCTION FOR THE FLUSHING OF THE SENSING LINES. SURVEILLANCE INSTRUCTIONS SVI-C11-T0045 A,B,C AND D WHICH CONTAINED THE FLUSHING STEPS HAVE BEEN REVISED. THE CHANGE TO THESE INSTRUCTIONS ENSURES THAT AN RPS ACTUATION WILL NOT RESULT WHEN THE LINES ARE FLUSHED. IN ADDITION, THE INDIVIDUALS RESPONSIBLE FOR THE TECHNICAL REVIEW OF THE ORIGINAL INSTRUCTION CHANGE HAVE BEEN COUNSELED TO MORE CAREFULLY CONSIDER THE IMPACT OF ESF LOGIC WHEN PERFORMING THEIR REVIEWS.

[189] PERRY 1 DOCKET 50-440 LER 87-058
CONTROL RODS WERE WITHDRAWN PRIOR TO PERFORMING SOURCE RANGE MONITORS SIGNAL TO NOISE RATIO VERIFICATION IN VIOLATION OF TECHNICAL SPECIFICATIONS.
EVENT DATE: 081587 REPORT DATE: 091187 NSSS: GE TYPE: BWR

(NSIC 206186) AT 8/15/87 AT 1245 SEVEN CONTROL RODS WERE INDIVIDUALLY WITHDRAWN PRIOR TO VERIFYING THE SIGNAL TO NOISE RATIO FOR THE SOURCE RANGE MONITORS (SRM) AS REQUIRED BY TECH SPEC 3.3.7.6. THE CONTROL RODS WERE BEING WITHDRAWN FOR VENTING THE ASSOCIATED HYDRAULIC CONTROL UNIT (HCU). VENTING WAS SUSPENDED AND SIGNAL TO NOISE RATIO CALCULATIONS WERE PERFORMED WITH SATISFACTORY RESULTS. THE EVENT WAS CAUSED BY PERSONNEL ERROR. THE OPERATORS WERE AWARE OF THE REQUIREMENT FOR SRM OPERABILITY BUT OVERLOOKED THE SIGNAL TO NOISE RATIO VERIFICATION REQUIREMENT, WHEN PERFORMING POST MAINTENANCE VENTING. ADDITIONALLY, THE SRM SIGNAL TO NOISE RATIO VERIFICATION IS PERFORMED AS PART OF THE TECH SPEC ROUNDS (TSR) WHEN IN OPERATIONAL CONDITION 5 BUT THE TSR DID NOT REQUIRE THE

VERIFICATION IN OPERATIONAL CONDITIONS 3 OR 4. THE OPERATORS INVOLVED WITH THE EVENT HAVE BEEN COUNSELED ON THE NEED FOR GREATER ATTENTION TO DETAIL AND THE SPECIFIC REQUIREMENTS FOR CONTROL ROD WITHDRAWAL. TSR WILL BE REVISED TO REQUIRE SRM SIGNAL TO NOISE RATIO VERIFICATION IN OPERATIONAL CONDITIONS 3 AND 4. A REVIEW OF THE TSR WILL BE CONDUCTED TO ENSURE ALL APPLICABLE SURVEILLANCES ARE CAPTURED. ADDITIONALLY, A NOTE WILL BE ADDED TO THE SYSTEM OPERATING INSTRUCTION FOR VENTING CONTROL ROD HCU'S TO ALERT THE OPERATORS OF THE NEED TO PERFORM THE SRM SIGNAL TO NOISE RATIO CALCULATIONS.

[190] PERRY 1 DOCKET 50-440 LER 87-059
REACTOR WATER CLEANUP CONTAINMENT ISOLATION DUE TO INDICATED HIGH DIFFERENTIAL FLOW WHILE ATTEMPTING TO BRING THE A FILTER/DEMINERALIZER IN SERVICE.
EVENT DATE: 082187 REPORT DATE: 091887 NSSS: GE TYPE: BWR

(NSIC 206187) ON AUGUST 21, 1987 AT 1033, AN UNEXPECTED REACTOR WATER CLEANUP (RWCU) CONTAINMENT ISOLATION OCCURRED DUE TO INDICATED HIGH DIFFERENTIAL FLOW. WHEN AN ATTEMPT WAS MADE TO PLACE THE A FILTER/DEMINERALIZER (F/D) IN SERVICE BY THROTTLING THE A F/D FLOW CONTROL VALVE (FCV) AND THE F/D BYPASS VALVE, THE FCV FAILED TO THE OPEN POSITION. THE VALVE FAILURE INDUCED A FLOW PERTURBATION IN THE SYSTEM CAUSING AN INDICATED HIGH DIFFERENTIAL FLOW AND A SUBSEQUENT RWCU ISOLATION. INVESTIGATION DETERMINED THAT THE CAUSE OF THIS EVENT WAS A PARTICULATE BUILDUP IN THE FCV POSITIONER AIR FLOW RESTRICTION ORIFICE BLOCKING THE INSTRUMENT CONTROL AIR AND CAUSING THE VALVE TO FAIL OPEN. THE PARTICULATE CONTAMINATION IS BELIEVED TO HAVE BEEN INTRODUCED DURING THE JUST COMPLETED MAINTENANCE OUTAGE, IN WHICH MODIFICATION WORK REQUIRED THE INSTALLATION OF A TEMPORARY AIR SUPPLY LINE. THE VALVE POSITIONER WAS REPLACED, THE AIR SUPPLY LINE WAS BLOWN DOWN, AND A PARTICLE COUNT ANALYSIS WAS PERFORMED ON THE AIR SUPPLY LINE TO THE FCV. THE RESULTS OF THE PARTICLE COUNT WERE SATISFACTORY, THE FCV WAS CALIBRATED, A LOOP CALIBRATION WAS PERFORMED FOR THE CONTROLLER TO THE FCV, AND THE A F/D WAS RETURNED TO STANDBY READINESS ON AUGUST 24.

[191] PERRY 1 DOCKET 50-440 LER 87-060
PERSONNEL ERRORS RESULT IN INOPERABLE EFFLUENT RADIATION MONITOR AND TECHNICAL SPECIFICATION VIOLATION.
EVENT DATE: 082287 REPORT DATE: 091887 NSSS: GE TYPE: BWR

(NSIC 206188) ON AUGUST 22, 1987 AT 1010, THE TURBINE BUILDING/HEATER BAY (TB/HB) VENT RADIATION MONITOR WAS DISCOVERED TO HAVE BEEN INOPERABLE FOR APPROXIMATELY 19 HOURS WITHOUT THE ACTIONS REQUIRED BY TECHNICAL SPECIFICATION 3.3.7.10 BEING PERFORMED. THE TB/HB VENT ISOKINETIC SAMPLE PUMP WAS TO BE TAGGED OUT DUE TO IDENTIFIED METAL TO METAL VIBRATION NOISE. ACTIONS WERE TAKEN TO MEET THE TECHNICAL SPECIFICATION LIMITING CONDITIONS FOR OPERATION (LCO) FOR THIS PLANT CONDITION. HOWEVER, THE PLANT OPERATORS MISTAKENLY TAGGED OUT THE TB/HB VENT RADIATION MONITOR SAMPLE PUMP. THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR DUE TO INATTENTION TO DETAIL. A PLANT OPERATOR TAGGED THE WRONG EQUIPMENT OUT-OF-SERVICE, A SECOND PLANT OPERATOR PERFORMING THE INDEPENDENT VERIFICATION FAILED TO IDENTIFY THE DISCREPANCY, AND THE CONTROL ROOM OPERATOR FAILED TO RESPOND TO THE ALARM AND ANNUNCIATION AS REQUIRED BY THE ALARM RESPONSE INSTRUCTION. THE ANNUNCIATION WAS CLEARED DUE TO THE ALARM BEING MISTAKENLY IDENTIFIED AS THE EXPECTED PUMP OUT-OF-SERVICE ALARM. UPON DISCOVERY, THE CHEMISTRY UNIT WAS IMMEDIATELY NOTIFIED, THE TAGGING ERROR WAS CORRECTED, AND A NOBLE GAS SAMPLE WAS DRAWN. THE CONTROL ROOM OPERATOR AND THE PLANT OPERATORS INVOLVED IN THIS EVENT HAVE RECEIVED DISCIPLINARY ACTIONS.

[192] PERRY 1 DOCKET 50-440 LER 87-062
CONTAINMENT ISOLATION VALVE ENERGIZED IN VIOLATION OF TECHNICAL SPECIFICATIONS.
EVENT DATE: 082887 REPORT DATE: 092587 NSSS: GE TYPE: BWR

(NSIC 206578) ON AUGUST 28, 1987 THE RESIDUAL HEAT REMOVAL (RHR) HEAD SPRAY CONTAINMENT ISOLATION VALVE WAS ENERGIZED IN VIOLATION OF TECHNICAL SPECIFICATION 3.6.4. THE HEAD SPRAY VALVE HAD BEEN TAKEN OUT OF SERVICE FOR CONTROL CIRCUITRY MODIFICATION ON AUGUST 26. AT THAT TIME, THE VALVE WAS DEACTIVATED AND SECURED IN THE CLOSED POSITION FOR CONTAINMENT PENETRATION ISOLATION. AT 0940 ON AUGUST 30, IT WAS DISCOVERED THAT PRIOR TO COMPLETION OF THE REQUIRED RETESTS, THE VALVE HAD BEEN ENERGIZED. THE VALVE WAS THEN DEENERGIZED BY OPENING THE BREAKER AND A NEW TAGOUT WAS IMPLEMENTED TO MAINTAIN TECHNICAL SPECIFICATION COMPLIANCE. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR. THE UNIT SUPERVISOR AUTHORIZING CLEARANCE OF THE TAGOUT WAS NOT AWARE OF THE IMPACT REENERGIZING THE VALVE HAD ON CONTAINMENT INTEGRITY REQUIREMENTS. THE OPERATOR INVOLVED WITH THE EVENT HAS BEEN COUNSELED REGARDING THE NEED FOR GREATER ATTENTION TO DETAIL AND A CLOSER REVIEW OF WORK PACKAGES FOR IMPACT ON TECHNICAL SPECIFICATION REQUIREMENTS. RETESTING OF THE HEAD SPRAY VALVE HAS BEEN COMPLETED.

[193] PERRY 1 DOCKET 50-440 LER 87-061
DRYWELL AIR LOCK CLOSING MECHANISM DESIGN CAUSES BUSHING FAILURE RESULTING IN AIR LOCK DOOR FAILURE AND TECHNICAL SPECIFICATION VIOLATION.
EVENT DATE: 083087 REPORT DATE: 092587 NSSS: GE TYPE: BWR
VENDOR: WOOLLEY, W. J. COMPANY

(NSIC 206464) ON AUGUST 30, 1987 AT 0415, IT WAS IDENTIFIED THAT TECHNICAL SPECIFICATION 3.6.2.3 FOR THE DRYWELL AIR LOCK WAS VIOLATED WHEN THE INNER AIR LOCK DOOR WAS NOT MAINTAINED CLOSED WITH THE OUTER AIR LOCK DOOR INOPERABLE. INSPECTION OF THE AIR LOCK DOOR IDENTIFIED THAT A BUSHING IN THE CLOSING MECHANISM HAD FALLEN OUT, ALLOWING THE DOOR OPERATOR TO INDICATE THE CLOSED POSITION WITHOUT THE CLOSING PINS ACTUALLY BEING ENGAGED. THE BUSHING WAS REPLACED WHEN THE LATCHING MECHANISM WAS REALIGNED. FURTHER OPERATION OF THE CLOSING LINKAGE WAS VERIFIED DURING EACH AIRLOCK USE UNTIL A NEW BUSHING AND RETAINING RING WAS INSTALLED. THE CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO THE DESIGN OF THE AIR LOCK DOOR CLOSING MECHANISM. DUE TO THE IMPROPER LINEUP OF THE BUSHING HOLES, EXTRA STRESS IS APPLIED TO THE BUSHINGS CAUSING THEM TO FAIL. AS THE RESULT OF THIS EVENT, AN ENGINEERING DESIGN CHANGE HAS BEEN INITIATED TO CHANGE THE APPROPRIATE COMPONENTS OF THE AIR LOCK DOOR TO PROVIDE FOR PROPER ALIGNMENT OF THE BUSHING HOLES.

[194] POINT BEACH 2 DOCKET 50-301 LER 87-002
LOSS OF LOAD REACTOR TRIP DUE TO LIGHTNING STRIKE.
EVENT DATE: 081687 REPORT DATE: 091487 NSSS: WE TYPE: PWR

(NSIC 206431) A UNIT 2 REACTOR TRIP OCCURRED FROM 100% POWER AT 1955 HOURS ON AUGUST 16, 1987. THE TRIP WAS CAUSED BY A LOSS OF ELECTRICAL LOAD. THE UNIT 2 GENERATOR BREAKER OPENED DUE TO A LIGHTNING STRIKE NEAR THE PLANT CAUSING A FLASHOVER BETWEEN THE GROUND AND "B" AND POSSIBLY THE "C" PHASES OF THE 345 KV LINE FOR THE UNIT 2 X01 TRANSFORMERS. AS A RESULT A TURBINE TRIP AND REACTOR TRIP OCCURRED. DUE TO THE NATURE OF THE TRANSIENT, AN AUTO BUS TRANSFER OF NON-VITAL BUSES TO OFFSITE POWER DID NOT OCCUR RESULTING IN THE REACTOR COOLANT PUMPS, CIRCULATING WATER PUMPS, MAIN FEED PUMPS, CONDENSATE PUMPS AND TURBINE AUXILIAIRES BEING DEENERGIZED. CONDENSER VACUUM WAS QUICKLY LOST. A FEW MINUTES AFTER THE TRIP, THE MAIN STEAM ISOLATION VALVES WERE CLOSED MANUALLY FROM THE CONTROL ROOM TO STOP THE FLOW OF STEAM TO THE TURBINE HALL. THE "B" MAIN STEAM ISOLATION VALVES WERE CLOSED MANUALLY FROM THE CONTROL ROOM TO STOP THE FLOW OF STEAM TO THE TURBINE HALL. THE "B" MAIN STEAM ISOLATION VALVE CLOSED TO WITHIN 1 INCH OF THE FULL CLOSED POSITION BUT WAS EASILY SEATED BY MANUAL OPERATOR ACTION. ALL OTHER EQUIPMENT OPERATED AS DESIGNED DURING THIS TRANSIENT. THIS INCIDENT WAS CLASSIFIED AS AN UNUSUAL EVENT. THE "B" AND "C" PHASE 19 KV/345 KV TRANSFORMERS WERE INSPECTED AND TESTED AS WELL AS THE FAST BUS TRANSFER.

[195] POINT BEACH 2 DOCKET 50-301 LER 87-003
MAIN STEAM ISOLATION VALVES OPEN WITHOUT TRIP POWER AVAILABLE.
EVENT DATE: 081887 REPORT DATE: 091787 NSSS: WE TYPE: PWR

(NSIC 206049) DURING A STARTUP OF UNIT 2 ON AUGUST 18, 1987, THE UNIT'S TWO MAIN STEAM ISOLATION VALVES (MSIVS) WERE DISCOVERED TO BE WITHOUT THE DC CONTROL POWER NEEDED TO TRIP THE VALVES CLOSED. THE VALVES WERE OPEN. THE REACTOR WAS AT 2 PERCENT POWER FOR ABOUT FIVE HOURS BEFORE THE CONDITION WAS DISCOVERED. UPON DISCOVERY, THE OPERATOR IMMEDIATELY RESTORED CONTROL POWER TO THE VALVES THEREBY RETURNING THE TRIP CIRCUITRY TO AN OPERABLE CONDITION. NO OTHER SAFETY SYSTEMS RECEIVE POWER FROM THIS POWER SUPPLY. THE CAUSE OF THE MSIVS BEING OUT OF SERVICE WAS PERSONNEL ERROR IN THAT THE POWER TO THE SOLENOID VALVES WAS NOT RESTORED CORRECTLY AFTER BEING REMOVED FROM SERVICE DURING OUTAGE MAINTENANCE. AN INCIDENT INVESTIGATION TEAM WAS APPOINTED TO INVESTIGATE THIS EVENT AND REPORT CONCLUSIONS AND RECOMMENDATIONS. THIS LER INCLUDES THE PERTINENT FINDINGS OF THE INCIDENT INVESTIGATION TEAM.

[196] PRAIRIE ISLAND 1 DOCKET 50-282 LER 87-015
SEVERE WEATHER CAUSED AUTOMATIC START OF DIESEL GENERATOR AND OTHER EQUIPMENT ACTUATIONS.
EVENT DATE: 072787 REPORT DATE: 082687 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: PRAIRIE ISLAND 2 (PWR)

(NSIC 205885) ON THE AFTERNOON OF JULY 27, 1987, BOTH UNITS WERE OPERATING AT FULL POWER. D2 DIESEL GENERATOR WAS OUT OF SERVICE FOR APPENDIX R WORK. AT 1520 SEVERE WEATHER MOVED INTO THE PLANT SITE AND AT ABOUT 1535 A TORNADO WAS SEEN BY SITE PERSONNEL. SEVERAL CHANGES IN EQUIPMENT STATUS TOOK PLACE AS A RESULT OF THE SEVERE WEATHER. AT 1545 A NOTIFICATION OF UNUSUAL EVENT WAS DECLARED. AT 1814 THE NUE WAS TERMINATED. CAUSE OF THE EVENT WAS SEVERE WEATHER ON THE PLANT SITE. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(IV). PUBLIC HEALTH AND SAFETY WERE UNAFFECTED SINCE PLANT SYSTEMS RESPONDED AS REQUIRED. RESTORATIVE EFFORTS WERE TIMELY AND PROPER THROUGHOUT THE EVENT. THE IMMEDIATE CORRECTIVE ACTIONS WERE TO RESTORE THE PLANT TO ITS PREVIOUS CONDITION. INSPECTION OF D1 DIESEL GENERATOR BEARINGS WILL BE DONE AT ITS FORTHCOMING PREVENTIVE MAINTENANCE OUTAGE. LONG TERM ACTIONS ARE BEING STUDIED. A RELATED EVENT WAS REPORTED AS RO 80-20.

[197] PRAIRIE ISLAND 1 DOCKET 50-282 LER 87-016
PERSONNEL ERROR CAUSED LOSS OF OFFSITE POWER SOURCE AND AUTOMATIC START OF DIESEL GENERATORS.
EVENT DATE: 073187 REPORT DATE: 082887 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: PRAIRIE ISLAND 2 (PWR)

(NSIC 206282) IN THE EARLY MORNING OF JULY 31, 1987, BOTH UNITS WERE OPERATING AT FULL POWER. RELAY SPECIALISTS WERE INVESTIGATING POSSIBLE CAUSES FOR A PREVIOUS FALSE TRIPPING OF NO. 10 TRANSFORMER BACKUP RELAYING. AFTER PROPERLY REMOVING THE BACKUP RELAYING FROM SERVICE FOR THE TESTS, THE TEST EQUIPMENT WAS INADVERTENTLY CONNECTED TO THE 6H2 BREAKER FAILURE RELAY. WHEN TEST CURRENT WAS APPLIED, AT ABOUT 0407, THE 6H2 BREAKER FAILURE RELAY OPERATED, RESULTING IN ISOLATION OF THE 161KV SUBSTATION, NO. 10 TRANSFORMER, AND NO. 1 RESERVE TRANSFORMER. THE LOSS OF NO. 1 RESERVE TRANSFORMER INITIATED POWER RESTORATION SCHEMES ON THE UNIT 1 SAFEGUARDS BUSES AND AUTO-START OF BOTH DIESEL GENERATORS (A DRY START). SINCE OFFSITE POWER WAS STILL AVAILABLE TO THE SAFEGUARDS BUSES, LOADING OF THE DIESEL GENERATORS WAS NOT REQUIRED. CAUSE OF THE EVENT WAS PERSONNEL ERROR: TEST EQUIPMENT WAS CONNECTED TO THE WRONG RELAY. ONCE THE CAUSE OF THE TRIPPING WAS DETERMINED TO BE INADVERTENT OPERATION INITIATED BY TEST PERSONNEL, LOCKOUTS WERE RESET AND BREAKERS WERE RECLOSED TO RESTORE THE TRANSMISSION SYSTEM AND PLANT AUXILIARY POWER SOURCES TO NORMAL. THIS EVENT IS

REPORTABLE UNDER 10CFR50.73(A)(2)(IV). PUBLIC HEALTH AND SAFETY WERE UNAFFECTED SINCE PLANT SYSTEMS RESPONDED AS REQUIRED.

[198] QUAD CITIES 1 DOCKET 50-254 LER 86-035
REACTOR CORE ISOLATION COOLING INOPERABLE DUE TO FAILED OVERSPEED METER.
EVENT DATE: 112686 REPORT DATE: 120186 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 206561) ON NOVEMBER 26, 1986, UNIT ONE WAS IN THE RUN MODE AT 95 PERCENT OF RATED CORE THERMAL POWER. AT 0200 HOURS, THE REACTOR CORE ISOLATION COOLING (RCIC) TURBINE OVERSPEED AND RCIC TURBINE TRIP ALARMS WERE RECEIVED WHILE RCIC WAS NOT IN OPERATION. RCIC WAS DECLARED INOPERABLE AND APPROPRIATE SURVEILLANCES WERE INITIATED PER TECH SPEC 4.5.E.2. THE CAUSE OF THIS EVENT WAS DETERMINED TO BE A FAILED LIGHT BULB IN THE RCIC TUBING SPEED METER ON THE 901-4 PANEL. THE BULB HAD REACHED ITS END OF LIFE. THE BURNT OUT BULB WAS REPLACED AND AT 0455 HOURS RCIC WAS AGAIN DETERMINED TO BE OPERABLE. THE RECURRING PROBLEMS WITH THE RCIC ELECTRICAL OVERSPEED TRIP RELAY HAS CAUSED GENERAL ELECTRIC TO RECOMMEND REMOVAL OF THE TRIP RELAY. ACTION ITEM RECORD 4-85-16 IS INVESTIGATING THIS RECOMMENDATION. THIS REPORT IS SUBMITTED TO COMPLY WITH THE REQUIRES OF 10 CFR 50.73(A)(2)(V)(B), WHICH REQUIRES THE REPORTING OF ANY EVENT THAT COULD HAVE PREVENTED THE FULFILLMENT OF THE SAFETY FUNCTION OF SYSTEMS NEEDED TO REMOVE RESIDUAL HEAT.

[199] QUAD CITIES 1 DOCKET 50-254 LER 86-036
GASEOUS EFFLUENT PARTICULATE SAMPLES LOST BY OFFSITE LABORATORY.
EVENT DATE: 120486 REPORT DATE: 121986 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: QUAD CITIES 2 (BWR)

(NSIC 206562) TECH SPEC 4.8.A.1 REQUIRES THAT A REPRESENTATIVE SAMPLE OF GASEOUS EFFLUENTS SHALL BE OBTAINED IN ACCORDANCE WITH THE SAMPLING AND ANALYSIS PROGRAM SPECIFIED IN TABLE 4.8-1. CONTRARY TO THE TECH SPEC REQUIREMENTS, THE JULY, 1986, MONTHLY PARTICULATE SAMPLES FOR THE MAIN CHIMNEY AND THE UNIT ONE AND UNIT TWO REACTOR VENT SYSTEMS WERE LOST. THE SAMPLES WERE COLLECTED AND SENT TO TELEDYNE ISOTOPES - MIDWEST FACILITY TO BE ANALYZED. WHEN THE RESULTS FOR THE MONTH OF JULY WERE NOT REPORTED IN A TIMELY MANNER, TELEDYNE ISOTOPES WAS CONTACTED. THE INQUIRY REVEALED THAT THEY COULD NOT LOCATE THE PARTICULATE SAMPLES. THE MISSING DATA WAS PROJECTED BY AVERAGING RECENT SAMPLE RESULTS. NUCLEAR SERVICES TECHNICAL SUPPORT GROUP, WHICH SETS UP OFFSITE ANALYSIS CONTRACTS, HAS BEEN INFORMED OF THE SITUATION AND WILL REQUIRE ANY VENDOR TO HAVE A SAMPLE TRACKING PROGRAM IMPLEMENTED. THIS REPORT IS SUBMITTED IN ACCORDANCE WITH THE REQUIREMENTS OF 10 CFR 50.73(A)(2)(I), WHICH REQUIRES THE REPORTING OF ANY OPERATION OR CONDITION PROHIBITED BY THE TECH SPECS.

[200] QUAD CITIES 1 DOCKET 50-254 LER 86-038
LOW VACUUM SCRAM DURING STARTUP DUE TO PERSONNEL ERROR.
EVENT DATE: 120986 REPORT DATE: 122386 NSSS: GE TYPE: BWR

(NSIC 206563) ON DECEMBER 9, 1986, UNIT ONE WAS IN THE PROCESS OF STARTING UP PER QGP 1-1, NORMAL UNIT STARTUP. UNIT ONE WAS AT 15 PERCENT THERMAL POWER AND 925 PSIG REACTOR PRESSURE. AT 1733 HOURS, A REACTOR SCRAM OCCURRED DUE TO LOW CONDENSER VACUUM. CONDENSER VACUUM HAD BEEN SIGNIFICANTLY LOWER THAN NORMAL SINCE VACUUM HAD BEEN ESTABLISHED. THE DECISION WAS MADE TO CONTINUE THE UNIT STARTUP EVEN THOUGH VACUUM WAS NOT ADEQUATE. THE BELIEF WAS THAT ROLLING THE TURBINE AND SYNCHRONIZING THE GENERATOR WOULD IMPROVE CONDENSER VACUUM BECAUSE THE HEAT LOAD ON THE CONDENSER WOULD BE LESS. THE ROOT CAUSE FOR THIS EVENT WAS DETERMINED TO BE PERSONNEL ERROR IN THAT IT WAS DECIDED THAT CONDENSER VACUUM WOULD IMPROVE WHEN THE MAIN TURBINE WAS ROLLED AND THE GENERATOR WAS SYNCHRONIZED. CONTRIBUTING TO THIS EVENT WAS POOR COMMUNICATION AND PROCEDURE

DEFICIENCY. CORRECTIVE ACTION FOR THIS EVENT WILL INCLUDE DISCUSSIONS WITH THE OPERATING PERSONNEL INVOLVED. STRESSED AT THESE DISCUSSIONS WILL BE THE NEED TO ADHERE TO ALL PROCEDURES AND THE NEED TO GIVE ADEQUATE INFORMATION AT SHIFT TURNOVER AND WHEN SENDING PERSONNEL OUT TO PERFORM A JOB. PROCEDURE QGP 1-1, NORMAL UNIT STARTUP, IS ALSO BEING REVISED TO CLARIFY THE STEPS NEEDED TO START UP THE OFF-GAS SYSTEM. THIS REPORT IS SUBMITTED TO COMPLY WITH 10CFR50.73(A)(2)(IV).

[201] QUAD CITIES 1 DOCKET 50-254 LER 87-003
RCIC INOPERABLE DUE TO FLOW CONTROLLER FAILURE.
EVENT DATE: 020587 REPORT DATE: 022587 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 206565) AT 1050 HOURS, ON FEBRUARY 5, 1987, UNIT ONE WAS IN THE RUN MODE AT 100 PERCENT CORE THERMAL POWER. WHILE PERFORMING QOS 1300-S2 (REACTOR CORE ISOLATION COOLING (RCIC) PUMP OPERABILITY TEST), IT WAS DISCOVERED THAT THE FLOW CONTROLLER DID NOT RESPOND TO AUTOMATIC FLOW CONTROL SIGNALS. HOWEVER, IT DID WORK SATISFACTORILY IN MANUAL. RCIC WAS DECLARED INOPERABLE AND APPROPRIATE NOTIFICATION PER 10CFR50.72 AND OPERABILITY TESTING PER TECH SPEC 3.5.E.2 WAS COMPLETED. THE EXACT CAUSE FOR THIS FLOW CONTROLLER FAILURE HAS YET TO BE DETERMINED. THE INSTRUMENT MAINTENANCE DEPARTMENT (IMD) IS TO TEAR DOWN THE CONTROLLER PARTS THAT WERE REPLACED TO DETERMINE THE CAUSE AND THIS WILL BE REPORTED IN A SUPPLEMENT. A REPLACEMENT FLOW CONTROLLER, CONTAINING A NEW POWER SUPPLY AND SETPOINT TAPE CONTROLLER, WAS INSTALLED BY THE IMD. THE AMPLIFIER PORTION OF THE ORIGINAL CONTROLLER WAS NOT REPLACED. RCIC WAS SUBSEQUENTLY DETERMINED TO BE OPERABLE THE SAME DAY, FEBRUARY 5, 1987 AT 1800 HOURS. THIS REPORT IS SUBMITTED TO SATISFY THE REQUIREMENTS OF 10CFR50.73(A)(2)(V).

[202] QUAD CITIES 1 DOCKET 50-254 LER 87-005
TURBINE TRIP/REACTOR SCRAM DUE TO MOISTURE SEPARATOR HIGH LEVEL.
EVENT DATE: 031787 REPORT DATE: 040787 NSSS: GE TYPE: BWR
VENDOR: FISHER GOVERNOR

(NSIC 206566) ON MARCH 17, 1987, QUAD CITIES UNIT ONE WAS AT 92 PERCENT CORE THERMAL POWER. AT 1102 HOURS, A REACTOR SCRAM OCCURRED DUE TO TURBINE STOP VALVE CLOSURE I.E., A TURBINE TRIP. THE TURBINE TRIP WAS CAUSED BY A HIGH LEVEL IN MOISTURE SEPARATORS 1C AND 1D. THE LEVEL CONTROL VALVE LCV 1-3508A FROM THE 1B MOISTURE SEPARATOR DRAIN TANK (MSDT) TO THE 1D1 HIGH PRESSURE FEEDWATER HEATER WAS STUCK OPEN. WHEN THIS VALVE WAS ISOLATED FOR REPAIRS, THE TWO OTHER LCVS FROM THE 1B MSDT TO THE 1D2 AND 1D3 HEATERS AND THE EMERGENCY DRAIN VALVE FAILED TO COMPENSATE FOR THIS ACTION. THIS CAUSED A HIGH LEVEL IN TWO MOISTURE SEPARATORS WHICH RESULTED IN THE TURBINE TRIP AND REACTOR SCRAM. THE CAUSE FOR THIS EVENT WAS ATTRIBUTED TO THE LEVEL CONTROLLERS BEING OUT OF ADJUSTMENT. CONTRIBUTING CAUSES WERE EQUIPMENT FAILURE (STUCK VALVE) AND INADEQUATE PREPARATION AND PLANNING PRIOR TO ISOLATING LCV 1-3508A. A PROGRAM WILL BE DEVELOPED TO TRACK SETPOINT CHANGES TO LEVEL CONTROL VALVES ON THIS AND OTHER SYSTEMS. THE STUCK LCV WAS FREED AND STROKED SUCCESSFULLY SEVERAL TIMES. DISCUSSIONS WERE HELD WITH THE INDIVIDUALS INVOLVED. THIS REPORT IS SUBMITTED TO COMPLY WITH 10CFR50.73(A)(2)(IV).

[203] QUAD CITIES 1 DOCKET 50-254 LER 87-006
HPCI INOPERABLE DUE TO LOOSE SOLENOID SOLDERED CONNECTION.
EVENT DATE: 040387 REPORT DATE: 042487 NSSS: GE TYPE: BWR
VENDOR: BARKSDALE COMPANY

(NSIC 206567) ON APRIL 3, 1987, QUAD CITIES UNIT ONE WAS IN THE RUN MODE AT 100 PERCENT THERMAL POWER. AT 1345 HOURS, WHILE PERFORMING THE QUARTERLY HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM FLOW RATE TEST, IT WAS DISCOVERED THAT

THE HPCI TURBINE COULD NOT BE RESET FROM THE CONTROL ROOM. THE HPCI SYSTEM WAS DECLARED INOPERABLE. APPROPRIATE SURVEILLANCES WERE INITIATED PER TECH SPEC 3.5.C.2 AND THE NRC WAS NOTIFIED AT 1635 HOURS. TROUBLESHOOTING BY THE ELECTRICAL MAINTENANCE DEPARTMENT REVEALED THAT A LOOSE SOLDERED CONNECTION ON THE SV-8 SOLENOID VALVE COIL HAD CAUSED THE PROBLEM. THE SV-8 SOLENOID VALVE IS ASSOCIATED WITH THE RESET MECHANISM THAT ALLOWS THE HPCI SYSTEM TO BE RESET REMOTELY FROM THE CONTROL ROOM. THE CONNECTION WAS LOOSE DUE TO THE NORMAL VIBRATION CREATED BY HPCI OPERATION. THE SOLENOID COIL WAS REPLACED AND BY TESTING IT WAS VERIFIED THAT THE HPCI TURBINE COULD BE RESET REMOTELY FROM THE CONTROL ROOM. AS A PREVENTATIVE MEASURE, THE SV 8 AND SV-12 VALVE SOLENOIDS WILL BE REPLACED ON BOTH UNIT ONE AND UNIT TWO HPCI SYSTEMS, AND A RESTRAINT WILL BE PLACED ON THE SOLENOID WIRING TO REDUCE STRAIN ON THE CONNECTIONS. THIS REPORT IS SUBMITTED TO COMPLY WITH THE REQUIREMENTS OF 10CFR50.73(A)(2)(V).

[204] QUAD CITIES 2 DOCKET 50-265 LER 86-018
ENGINEERED SAFETY FEATURE ACTUATION DUE TO RADIOGRAPHIC TESTING.
EVENT DATE: 111386 REPORT DATE: 112686 NSSS: GE TYPE: BWR

(NSIC 206439) AT 1645 HOURS ON NOVEMBER 13, 1986, WHILE UNIT TWO WAS SHUT DOWN FOR A REFUELING OUTAGE, THE REACTOR BUILDING VENTILATION (RBV) (VA) SYSTEM TRIPPED DUE TO A HIGK RADIATION LEVEL SENSED BY THE RBV EXHAUST DAMPERS CLOSED. STANDBY GAS TREATMENT (SBGT) (BH) STARTED, AND THE CONTROL ROOM VENTILATION (VI) SYSTEM WENT ON 100 PERCENT RECIRCULATION. INVESTIGATION REVEALED THAT THE CAUSE FOR THIS EVENT WAS DUE TO CONTRACTOR PERSONNEL PERFORMING RADIOGRAPHY ON THE UNIT TWO STANDBY LIQUID CONTROL (SBLC) (BR) SYSTEM PIPING. AT 1855 HOURS, RADIOGRAPHY WAS AGAIN REQUIRED ON THE UNIT TWO SBLC PIPING. THIS CAUSED THE ACTUATIONS MENTIONED ABOVE TO OCCUR AGAIN. APPROPRIATE NOTIFICATIONS WERE MADE AT 2030 HOURS TO SATISFY THE REQUIREMENTS OF 10 CFR 50.72 AND THE NRC RESIDENT INSPECTOR WAS NOTIFIED AT 2100 HOURS. SINCE THE RADIOGRAPHY WAS REQUIRED TO VERIFY WELDING PERFORMED ON THE SBLC SYSTEM, NO FURTHER CORRECTIVE ACTION IS CONSIDERED NECESSARY. THIS REPORT IS SUBMITTED TO COMPLY WITH THE REQUIREMENTS OF 10 CFR 50.72(A)(2)(IV).

[205] QUAD CITIES 2 DOCKET 50-265 LER 86-019
SINGLE CONTROL ROD SCRAM WHILE SHUTDOWN DUE TO REACTOR PROTECTION SYSTEM FUSE REMOVAL.
EVENT DATE: 112886 REPORT DATE: 121086 NSSS: GE TYPE: BWR

(NSIC 206440) ON NOVEMBER 28, 1986 UNIT TWO WAS IN THE REFUEL MODE WITH THE REACTOR CORE COMPLETELY UNLOADED. REACTOR PROTECTION SYSTEM (RPS) CHANNEL B WAS OUT OF SERVICE WITH A SCRAM SIGNAL PRESENT TO PERFORM HFA RELAY REPLACEMENT. MAINTENANCE WAS REQUIRED ON A SOLENOID VALVE (2-302-20A) IN THE SCRAM DISCHARGE VOLUME (SDV) PORTION OF THE CONTROL ROD DRIVE (CRD) SYSTEM. TO ELECTRICALLY DISARM THIS VALVE, A FUSE WAS REMOVED AT 1045 HOURS IN THE PANEL ASSOCIATED WITH THE A RPS CHANNEL. THIS GENERATED A FULL SCRAM SIGNAL TO 47 CONTROL RODS. ONE OF THESE 47 CONTROL RODS (K-3) WAS BEING WITHDRAWN AT THIS TIME TO PERFORM CRD VENTING FOLLOWING MAINTENANCE. THE SCRAM SIGNAL CAUSED CONTROL ROD K-3 TO SCRAM INTO POSITION 00. THE ROOT CAUSE FOR THIS EVENT WAS A MANAGEMENT DEFICIENCY IN THAT IT WAS NOT ANTICIPATED THAT THE RESULTS OF PERFORMING ALL THIS WORK SIMULTANEOUSLY COULD RESULT IN AN ENGINEERED SAFETY FEATURE ACTUATION. TO PREVENT RECURRENCES OF THIS NATURE, DISCUSSIONS WERE HELD WITH THOSE PEOPLE INVOLVED TO STRESS THE IMPORTANCE OF COMPLETE PLANNING AND PREPARATION FOR ALL WORK TO BE PERFORMED. THIS EVENT IS BEING REPORTED TO COMPLY WITH THE REQUIREMENTS OF 10 CFR 50.73(A)(2)(IV).

[206] QUAD CITIES 2 DOCKET 50-265 LER 86-020
SPURIOUS TRIP OF "B" REACTOR BUILDING VENTILATION RADIATION MONITOR CAUSES ISOLATION OF REACTOR BUILDING VENTILATION.
EVENT DATE: 121186 REPORT DATE: 010387 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 206441) AT 2200 HOURS ON DECEMBER 11, 1986, UNIT TWO WAS IN THE REFUEL MODE AT 0 PERCENT CORE THERMAL POWER. FOLLOWING PERFORMANCE OF QIS 34-1 (REACTOR BUILDING VENTILATION MONITORING CALIBRATION), THE UNIT TWO NUCLEAR STATION OPERATOR (NSO) RETURNED THE "B" REACTOR BUILDING VENTILATION RADIATION MONITOR TO "NORMAL" FROM "BYPASS" AT THE INSTRUMENT MECHANIC'S REQUEST. WHEN THIS WAS DONE, THE REACTOR BUILDING VENTILATION ISOLATED AND THE "B" STANDBY GAS TREATMENT SYSTEM AUTOSTARTED. ATTEMPTS WERE MADE TO DUPLICATE THE EVENT, BUT WERE UNSUCCESSFUL. IN ADDITION, THE RADIATION MONITOR BYPASS SWITCH WAS CHECKED FOR DISCONTINUITIES. NO PROBLEMS WERE FOUND. THIS IS CONSIDERED AN ISOLATED EVENT, AND NO FURTHER CORRECTIVE ACTION IS DEEMED NECESSARY. THIS REPORT IS BEING SUBMITTED TO COMPLY WITH 10 CFR 50.73 (A)(2)(IV), WHICH REQUIRES THE REPORTING OF ANY EVENT OR CONDITION THAT RESULTED IN MANUAL OR AUTOMATIC ACTUATION OF ANY ENGINEED SAFETY FEATURE.

[207] QUAD CITIES 2 DOCKET 50-265 LER 86-021
ESF ACTUATION ON LOSS OF BUS 18 DUE TO PERSONNEL ERROR.
EVENT DATE: 121986 REPORT DATE: 010887 NSSS: GE TYPE: BWR

(NSIC 206442) ON DECEMBER 19, 1986, QUAD CITIES UNIT TWO WAS IN A REFUEL AND MAINTENANCE OUTAGE WITH THE REACTOR MODE SWITCH IN SHUTDOWN. AT 1919 HOURS, AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION OCCURRED WHILE THE OPERATIONAL ANALYSIS DEPARTMENT (QAD) WAS PERFORMING TRIP CHECKS ON THE BUS 23-1 TO BUS 28 CIRCUIT BREAKER. THE ESF ACTUATION INCLUDED A REACTOR SCRAM, GROUP I, II, AND III ISOLATIONS AND A REACTOR BUILDING VENTILATION SYSTEM ISOLATION. THE ROOT CAUSE FOR THIS EVENT WAS DETERMINED TO BE PERSONNEL ERROR. QAD PERSONNEL DID NOT REALIZE THAT CIRCUIT BREAKER INTERLOCKS WERE STILL IN EFFECT EVEN THOUGH THE BREAKER WAS RACKED OUT TO ITS "TEST" POSITION. WHEN THE BUS 23-1 TO BUS 28 BREAKER WAS CLOSED LOCALLY BY QAD WHILE IN THE TEST POSITION, INTERLOCKS CAUSED THE BUS 29 TO BUS 28 CROSSTIE BREAKER TO OPEN AND THUS POWER TO BUS 28 WAS LOST. THIS RESULTED IN THE ESF ACTUATION. CORRECTIVE ACTION FOR THIS EVENT WAS TO STOP FURTHER TRIP CHECKS AND TO RESTORE POWER TO BUS 28 FROM BUS 29. POWER WAS RESTORED AND ALL SYSTEMS WERE RETURNED TO NORMAL AT 1939 HOURS. FURTHER CORRECTIVE ACTION WILL BE TO ADD A NOTE OF CAUTION TO THE QAD EVALUATION CHECKLIST (QAP 1500-S7) TO WARN OF SITUATIONS OF THIS NATURE. THIS REPORT IS SUBMITTED TO COMPLY WITH 10CFR 50.73(A)(2)(IV).

[208] RANCHO SECO DOCKET 50-312 LER 85-016 REV 01
UPDATE ON SPURIOUS CLOSURE OF DHR DROPLINE ISOLATION VALVE CAUSED BY IMPROPER ROUTING SHIELDED INSTRUMENT CABLE.
EVENT DATE: 080885 REPORT DATE: 092287 NSSS: BW TYPE: PWR

(NSIC 206445) ON AUGUST 8, AUGUST 14, AND SEPTEMBER 23, 1985, WHILE IN COLD SHUTDOWN, THE DECAY HEAT REMOVAL SYSTEM (DHS) SUCTION BLOCK VALVE (HV-20002) AUTOMATICALLY CLOSED ON A HIGH REACTOR COOLANT SYSTEM (RCS) PRESSURE SIGNAL, THUS RESULTING IN A TEMPORARY LOSS OF THE DHS SYSTEM CAPABILITY. IN ALL THREE CASES, DHS FLOW WAS RE-ESTABLISHED IN LESS THAN AN HOUR. HV-20002 IS DESIGNED TO CLOSE AUTOMATICALLY WHEN THE RCS PRESSURE EXCEEDS 255 PSIG. THE RCS PRESSURE RECORDED BY OPERATIONS PERSONNEL AT THE TIME OF THE EVENTS WAS APPROXIMATELY 230 PSIG. ALTHOUGH NO DEFINITE REASON FOR THE VALVE CLOSURE WAS DETERMINED, AN INVESTIGATION OF THE EVENTS INDICATED THAT VOLTAGE SPIKES ON PRESSURE TRANSMITTER PT-21099 CIRCUITRY CAUSED THE BLOCK VALVE TO CLOSE. PT-21099 WAS REPLACED AND CALIBRATED DURING THE CYCLE 7 REFUELING OUTAGE AND A SUCCESSFUL MAINTENANCE TEST WAS PERFORMED FOLLOWING THE EVENTS TO ENSURE THE PROPER OPERABILITY OF THE DECAY

HEAT VALVE INTERLOCK AND ASSOCIATED INSTRUMENTATION. THE SPURIOUS DECAY HEAT ISOLATION SIGNAL WAS TRACED TO IMPROPERLY ROUTING SHIELDED INSTRUMENT CABLE (1R1S04B6A) THROUGH CHANNEL B POWER TRAYS AND CONDUIT TO A PENETRATION, AS DOCUMENTED IN NCR S-5263, REVISION 3. THE VOLTAGE SPIKE ON THE PT-21099 INSTRUMENT CIRCUIT APPEARS, IN A TEST, WHEN THE POWER CIRCUIT TO MOTOR OPERATED VALVE HV-20002 WAS ENERGIZED.

[209] RANCHO SECO DOCKET 50-312 LER 87-015 REV 01
UPDATE ON CARBON DIOXIDE PROTECTED FIRE ZONES DELUGE SYSTEMS LEFT DEACTIVATED AND THE ZONES UNATTENDED.
EVENT DATE: 020787 REPORT DATE: 101587 NSSS: BW TYPE: PWR

(NSIC 206548) DURING COLD SHUTDOWN CONDITIONS, BOTH OPERATIONS PERSONNEL AND SECURITY PERSONNEL OBSERVED NUMEROUS CARDOX ZONES DEACTIVATED WHEN THERE WAS NO SAFETY CONCERN FOR PERSONNEL. THERE WERE NUMEROUS OCCURRENCES WHEN PERSONNEL DISABLED FIRE PROTECTION SUBSYSTEMS IN THE PLANT FOR PERSONNEL SAFETY REASONS, BUT FAILED TO REACTIVATE THEM FOR OVER AN HOUR AFTER THE PERSONNEL SAFETY CONCERN WAS RESOLVED. IT SHOULD BE NOTED THAT THE DETECTION CAPABILITY FOR THE ZONE WAS NOT DISABLED BY THE CO2 SYSTEM BEING TURNED OFF. THE RANCHO SECO FIRE HAZARDS ANALYSIS HYPOTHESIZES THAT A FIRE IN ANY DESIGNATED AREA WILL CAUSE THE LOSS OF ALL FUNCTIONS SUPPORTED BY EQUIPMENT AND CABLES IN THAT AREA. IMMEDIATE CORRECTIVE ACTION WAS TO REQUIRE AN HOURLY VERIFICATION THAT PERSONNEL ARE, IN FACT, IN THE DEACTIVATED CO2 ZONE WHEN A SYSTEM IS DEACTIVATED.

[210] RANCHO SECO DOCKET 50-312 LER 87-039
HIGH RADIATION AREA NOT PROPERLY POSTED.
EVENT DATE: 082687 REPORT DATE: 091587 NSSS: BW TYPE: PWR

(NSIC 206052) ON AUGUST 26, 1987, AS A RESULT OF AN INSPECTION OF THE REACTOR BUILDING, AN NRC INSPECTOR ASKED TO SEE SURVEYS OF THE REACTOR HEAD LAYDOWN AREA. AT APPROXIMATELY 1030 HOURS, ON AUGUST 26, 1987, A RADIATION PROTECTION TECHNICIAN CONDUCTED A RADIATION SURVEY INSIDE THE REACTOR HEADSTAND AREA IN THE REACTOR BUILDING. THE AREA INSIDE THE HEADSTAND WAS FOUND TO HAVE A GENERAL AREA DOSE RATE OF 300 MREM/HR; HOWEVER, THE AREA WAS NOT POSTED AS A HIGH-RADIATION AREA. RADIATION PROTECTION HAD NOT PERFORMED ROUTINE SURVEYS INSIDE THE HEADSTAND AREA BECAUSE THEY DID NOT CONSIDER IT A NORMAL OR ACCESSIBLE WORK AREA. THE REACTOR HEAD LAYDOWN AREA WAS POSTED AS: "RADIOACTIVE CONTAMINATED. DO NOT ENTER WITHOUT RP APPROVAL." FAILURE TO CONSPICUOUSLY POST THE REACTOR HEAD LAYDOWN AREA AS A HIGH-RADIATION AREA IS A VIOLATION OF TECHNICAL SPECIFICATION 6.13.1.A. THE FOLLOWING CORRECTIVE ACTIONS WERE TAKEN AS A RESULT OF THIS INCIDENT: 1) THE REACTOR HEAD LAYDOWN AREA WAS POSTED AS A HIGH-RADIATION AREA. 2) SURVEYS WERE TAKEN TO ENSURE THAT NO OTHER UNPOSTED HIGH-RADIATION AREAS EXIST WITHIN THE REACTOR BUILDING. 3) ACTIONS HAVE BEEN INITIATED TO DEFINE THE TERM "ACCESSIBLE" AND INCORPORATE THE DEFINITION INTO APPROPRIATE RADIATION PROTECTION PROCEDURES.

[211] RIVERBEND 1 DOCKET 50-458 LER 86-005 REV 01
UPDATE ON UNLABELED SWITCH CAUSES ACTIVATION OF WATER CURTAIN LEADING TO SCRAM.
EVENT DATE: 010786 REPORT DATE: 090987 NSSS: GE TYPE: BWR

(NSIC 206028) ON 1/7/86 AT 0847 WITH THE UNIT IN OPERATIONAL CONDITION 2 (STARTUP), A FIRE PROTECTION WATER CURTAIN WAS INADVERTENTLY ACTUATED BY A CONSTRUCTION EMPLOYEE. WATER FROM THIS ACTUATION RAN INTO TWO MOTOR CONTROL CENTERS AND THROUGH AN UNSEALED PENETRATION IN THE FLOOR AND EVENTUALLY INTO A LOAD CENTER ON THE NEXT LOWER ELEVATION. THE RESULTING SHORT IN THE LOAD CENTER CAUSED A TRANSFORMER TO BURN UP WHICH CAUSED THE BREAKER FEEDING THAT LOAD CENTER TO TRIP. THIS BREAKER ALSO FED TWO ADDITIONAL LOAD CENTERS, THE LOSS OF WHICH EVENTUALLY CAUSED A REACTOR TRIP ON HIGH INTERMEDIATE RANGE MONITORS.

INVESTIGATION INTO THE EVENT DETERMINED THAT THE SOLENOID ACTIVATION SWITCH WAS UNMARKED AND MISTAKENLY THOUGHT TO BE A DOOR LATCH. THE IDENTIFICATION AND PROPER LABELING OF ALL SIMILAR SWITCHES HAS BEEN COMPLETED. THERE WERE NO SAFETY CONSEQUENCES TO THE PUBLIC AS A RESULT OF THIS EVENT.

[212] RIVERBEND 1 DOCKET 50-458 LER 86-028 REV 01
UPDATE ON INCORRECT TRANSFORMER TAP SETTINGS RESULTING IN AN UNANALYZED CONDITION.
EVENT DATE: 040486 REPORT DATE: 091587 NSSS: GE TYPE: BWR

(NSIC 206029) ON 4/3/86 WITH THE UNIT IN OPERATIONAL CONDITION 4 (COLD SHUTDOWN), A REVIEW WAS CONDUCTED OF THE SAFETY RELATED SYSTEM VOLTAGE PROFILE TESTING PROGRAM (SST 6). AS A RESULT OF THIS REVIEW, IT WAS DISCOVERED THAT THE CLASS 1E TRANSFORMER TAP SETTINGS WERE NOT IN ACCORDANCE WITH THE SUPPORTING CALCULATIONS WHICH WERE INTENDED TO OPTIMIZE THE VOLTAGE LEVELS AT THE SAFETY-RELATED BUSES FOR THE MAXIMUM AND MINIMUM LOAD CONDITIONS. A LIMITING CONDITION FOR OPERATION WAS INITIATED ON 4/4/86 AT 1450 PROHIBITING PLANT OPERATIONS FROM ENTERING MODES 1, 2, AND 3 AND A DESIGN CHANGE WAS INITIATED TO CHANGE THE TRANSFORMER TAP SETTINGS IN ACCORDANCE WITH THE SUPPORTING CALCULATIONS. THE WORK WAS COMPLETED AND THE LCO WAS CLEARED AT 1320 ON APRIL 9, 1986. NO SAFETY CONSEQUENCES RESULTED FROM THE ABOVE CONDITION AND THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT ENDANGERED.

[213] RIVERBEND 1 DOCKET 50-458 LER 87-016
SPURIOUS AUTO START OF STANDBY GAS TREATMENT AND ANNULUS MIXING SYSTEMS CAUSED BY LIGHTNING.
EVENT DATE: 080587 REPORT DATE: 090387 NSSS: GE TYPE: BWR

(NSIC 206295) ON 8/5/87 AT 1831 WITH THE UNIT AT 91 PERCENT POWER (PLANT IN COAST DOWN), AUTO STARTS OF THE ANNULUS MIXING (AM) AND STANDBY GAS TREATMENT (SBGT) SYSTEMS OCCURRED. A LIGHT POLE IN THE PARKING LOT ADJACENT TO THE PLANT WAS STRUCK BY LIGHTNING AT THE SAME TIME THE AUTO STARTS OCCURRED. THE AM AND SBGT SYSTEMS WERE RETURNED TO THEIR NORMAL STANDBY LINEUP. IT IS ASSUMED THAT AN ELECTRICAL ANOMALY CAUSED BY THE LIGHTNING STRIKE CAUSED THE AUTO STARTS OF THESE SYSTEMS. THE SAFETY AND HEALTH OF THE PUBLIC WERE NOT AFFECTED BY THIS EVENT BECAUSE THESE SYSTEMS OPERATED IN THE EMERGENCY MODE, WHICH IS THE SAFEST MODE OF OPERATION.

[214] RIVERBEND 1 DOCKET 50-458 LER 87-017 REV 01
UPDATE ON MISPOSITIONED INSTRUMENT VALVE RENDERS LEAK DETECTION SYSTEM INOPERABLE.
EVENT DATE: 081187 REPORT DATE: 101487 NSSS: GE TYPE: BWR

(NSIC 206584) AT 2300 ON 8/11/87 WITH THE UNIT AT APPROXIMATELY 91 PERCENT POWER, REACTOR CORE ISOLATION COOLING (RCIC) TRANSMITTER 1E51*N055E INSTRUMENT ISOLATION VALVE WAS FOUND IN THE CLOSED POSITION DURING THE PERFORMANCE OF A SURVEILLANCE TEST PROCEDURE. THIS CAUSED THE RCIC TURBINE EXHAUST DIAPHRAGM PORTION OF THE DIVISION I RCIC LEAK DETECTION SYSTEM ISOLATION LOGIC TO BE INOPERATIVE. NO CAUSE FOR THE MISPOSITIONED VALVE HAS BEEN IDENTIFIED. AT 0810 ON 9/14/87, INVESTIGATION INTO A HALF SCRAM SIGNAL WHICH WOULD NOT RESET REVEALED THAT AN ISOLATION VALVE FOR A SCRAM DISCHARGE VOLUME (SDV) FLOAT SWITCH WAS CLOSED. AS A RESULT, THE FLOAT SWITCH WOULD NOT HAVE SENSED A HIGH SDV WATER LEVEL CONDITION UNTIL THE SDV LEVEL HAD RISEN ABOVE THE TECHNICAL SPECIFICATION ALLOWABLE TRIP SETPOINT. AS CORRECTIVE ACTION, GENERAL MAINTENANCE PROCEDURE GMP-0042 HAS BEEN REVISED TO CLARIFY THE DEFINITION OF INDEPENDENT VERIFIER. ADDITIONALLY, THE METHOD FOR VERIFYING VALVE POSITION WAS ADDED TO ENSURE POSITIVE VERIFICATION OF THE AS-LEFT VALVE POSITION. INSTRUMENT VALVE LINEUPS SERVICING SAFETY RELATED INSTRUMENTS AND ALL OTHER INSTRUMENTS CONTAINED IN TECHNICAL SPECIFICATIONS WILL BE VERIFIED PRIOR TO START-UP FOLLOWING THE CURRENT REFUELING OUTAGE. THERE WAS

NO SIGNIFICANT IMPACT ON THE HEALTH AND SAFETY OF THE PUBLIC AS A RESULT OF THESE MISPOSITIONED INSTRUMENT VALVES.

[215] RIVERBEND 1 DOCKET 50-458 LER 87-018
DEFICIENT TRACKING OF DIESEL FUEL OIL SURVEILLANCES.
EVENT DATE: 090187 REPORT DATE: 092887 NSSS: GE TYPE: BWR

(NSIC 206504) ON 9/01/87, WITH THE UNIT AT APPROXIMATELY 84 PERCENT POWER, IT WAS DISCOVERED THAT TWO CHEMICAL PROPERTIES (ASH WEIGHT AND DISTILLATION TEMPERATURE) OF A DIESEL FUEL OIL SHIPMENT MADE TO RIVER BEND STATION (RBS) ON 6/25/87 DID NOT MEET THE ACCEPTANCE CRITERIA AS SPECIFIED BY THE RBS TECHNICAL SPECIFICATIONS. ADDITIONALLY, THE RESULTS OF FUEL OIL TESTING AND ANALYSIS WERE NOT EVALUATED WITHIN 31 DAYS FOLLOWING THE DELIVERY, AS REQUIRED BY THE SAME TECHNICAL SPECIFICATION. THE DISCOVERY DATE OF 9/1/87 WAS 67 DAYS PAST THE DATE OF FUEL OIL DELIVERY. SAMPLING AND ANALYSES WERE COMPLETED ON THE FUEL OIL STORAGE TANKS WHERE THE SUSPECT FUEL OIL WAS OFF-LOADED WITH ACCEPTABLE RESULTS. AS SHOWN BY THIS ANALYSIS, ALL FUEL OIL THAT HAS BEEN USED OR IS CONTAINED WITHIN ANY DIESEL FUEL OIL STORAGE TANK, DID MEET ACCEPTANCE CRITERIA. ADEQUATE CORRECTIVE ACTION HAS BEEN TAKEN TO PRECLUDE THE POTENTIAL OF UNACCEPTABLE FUEL OIL BEING USED IN THE FUTURE. ADDITIONALLY, CORRECTIVE MEASURES HAVE BEEN ESTABLISHED TO BETTER TRACK EVENT RELATED SURVEILLANCES WITHIN THE CHEMISTRY SECTION. THERE WAS NO ADVERSE IMPACT ON THE HEALTH AND SAFETY OF THE PUBLIC AS A RESULT OF THIS EVENT.

[216] ROBINSON 2 DOCKET 50-261 LER 87-022
REACTOR TRIP DUE TO INTERMEDIATE RANGE CHANNEL TRIP SETPOINT PROCEDURAL DEFICIENCY.
EVENT DATE: 081087 REPORT DATE: 090887 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 206307) ON AUGUST 10, 1987, AT 2334 HOURS, WITH UNIT NO. 2 IN STARTUP AT ABOUT EIGHT PERCENT POWER, INTERMEDIATE RANGE CHANNEL N-35 INITIATED A HIGH FLUX REACTOR TRIP. THE NRC WAS NOTIFIED PURSUANT TO 10CFR50.72(B)(2)(II). PRIOR TO THE EVENT, THE REACTOR COOLANT AVERAGE TEMPERATURE (TAVG) WAS DECREASING AS THE UNIT WAS SYNCHORNIZED, PICKING UP AN INITIAL LOAD OF ABOUT 35 MEGAWATTS. A UTILITY LICENSED OPERATOR BEGAN WITHDRAWING RODS TO INCREASE TAVG, AND RECEIVED A ROD STOP SIGNAL. REACTOR POWER INCREASED DUE TO TAVG DECREASE, AND A HIGH FLUX REACTOR TRIP RESULTED. INTERMEDIATE RANGE DETECTOR N-35 HAD JUST BEEN REPLACED, AND THE HIGH FLUX TRIP SETPOINT HAD YET TO BE DETERMINED. THE SETPOINT IS BASED ON DETECTOR CURRENT EQUIVALENT TO 25% OF FULL REACTOR POWER AS DETERMINED BY STARTUP TESTING. THE EXISTING SETPOINT USED WAS THAT OF THE OLD DETECTOR, BUT THE NEW DETECTORS RESPONSE WAS SIGNIFICANTLY HIGHER. CHANNEL N-35 REACHED THE CURRENT EQUIVALENT TO THE OLD DETECTOR SETPOINT BEFORE THE AUTOMATIC TRIP COULD BE BYPASSED. THIS RESULTED IN THE REACTOR TRIP. FOLLOWING THE TRIP, N-35 WAS CALIBRATED, AND THE PLANT RETURNED TO POWER OPERATION AT APPROXIMATEY 0400 HOURS ON AUGUST 11, 1987.

[217] SALEM 1 DOCKET 50-272 LER 87-006 REV 01
UPDATE ON BOTH TRAINS OF HIGH HEAD SI DECLARED INOPERABLE DUE TO INOPERABLE CHARGING PUMPS.
EVENT DATE: 052587 REPORT DATE: 100687 NSSS: WE TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 206520) ON MAY 25, 1987 AT 1607 HOURS, IT WAS OBSERVED THAT THE SPRING CHARGING MOTOR, FOR NO. 11 CENTRIFUGAL CHARGING PUMP (CCP)(BQ) 4KV BREAKER (GE MAGNE BLAST TYPE AM-4.16-350-2N, S/N 0224A6261-011), WAS NOT ATTACHED TO THE BREAKER FRAMEWORK. THE PUMP WAS SUBSEQUENTLY DECLARED INOPERABLE. PRIOR TO DISCOVERY OF THE NO. 11 CCP BREAKER CONDITION, THE NO. 12 CCP EMERGENCY POWER SUPPLY (1C DIESEL GENERATOR (EK) HAD BEEN MADE UNAVAILABLE AS A RESULT OF A

DESIGN CHANGE IN PROGRESS. SINCE NONE OF THE TECH SPEC 3.5.2 ACTION STATEMENTS CAN BE MET WHEN BOTH CENTRIFUGAL CHARGING PUMPS ARE DECLARED INOPERABLE (ONE OF WHICH IS DECLARED INOPERABLE DUE TO AN INOPERABLE EMERGENCY POWER SUPPLY) TECH SPEC 3.0.5 WAS ENTERED. VISUAL EXAMINIATION OF 4KV BREAKERS FOUR (4) 1/4 INCH BOLTS REVEALED NO ADDITIONAL LOOSE BOLTS. AS A RESULT OF PRELIMINARY INVESTIGATIONS OF THIS EVENT (NPRDS, NUCLEAR NETWORK, ... ETC.), PSE&G IS AWARE OF OTHER SPRING CHARGING MOTOR MOUNTING BOLT PROBLEMS AT OTHER PLANTS. INVESTIGATION OF INPO SER 14-87 "BREAKER FAILURE DUE TO LOOSE MOUNTING BOLTS" HAS BEEN COMPLETED. THE BREAKER MANUFACTURER CONCURS THAT THIS OCCURRENCE WAS AN ISOLATED CASE. HOWEVER, THE MAINTENANCE PROCEDURE FOR INSPECTION AND TESTING OF THIS BREAKER HAS BEEN MODIFIED TO DETAIL MOTOR BOLT TIGHTENING SPECIFICATIONS AND THE APPLICATION OF A LACQUER TO INDICATE VISIBLY IF A BOLT LOOSENS.

[218] SALEM 2 DOCKET 50-311 LER 86-009 REV 01
UPDATE ON LOSS OF NO. 22 STATION POWER TRANSFORMER.
EVENT DATE: 091186 REPORT DATE: 093087 NSSS: WE TYPE: PWR
VENDOR: I-T-E CIRCUIT BREAKER
WESTINGHOUSE ELECTRIC CORP.

(NSIC 206611) ON 9/11/86, AT 1858 HOURS, DURING ROUTINE POWER OPERATION, A REACTOR TRIP OCCURRED. THE FIRST OUT ANNUNCIATOR WAS "REACTOR COOLANT LOW FLOW/REACTOR COOLANT PUMP BREAKER OPEN AND P-8 (REACTOR ABOVE 36% POWER)". THE PLANT WAS STABILIZED IN MODE 3 (HOT STANDBY). THE INITIATING CAUSE OF THE REACTOR TRIP/TURBINE TRIP WAS ISOLATION OF NO. 22 STATION POWER TRANSFORMER (SPT) 13/4KV, DUE TO PHASE A AND PHASE B DIFFERENTIAL RELAY PROTECTION OPERATION. THE LOSS OF NO. 22 SPT RESULTED IN THE LOSS OF "F" & "G" 4KV GROUP BUSSES. LOSS OF THESE BUSSES CAUSED THE LOSS OF NO. 23 AND NO. 24 REACTOR COOLANT PUMPS (RCP'S) RESULTING IN THE REACTOR TRIP/TURBINE TRIP. SIMULTANEOUSLY, AN ELECTRICAL FAULT IN 2F 4160/230V TRANSFORMER OCCURRED. THE FAULT RESULTED IN THE OPERATION OF OVERCURRENT RELAY PROTECTION WHICH OPENED THE 4KV BREAKER 2F5D SUPPLYING THE 2F 4160/230V TRANSFORMER. TESTING OF NO. 22 SPT CONFIRMED THE TRANSFORMER SUSTAINED AN INTERNAL FAULT. INVESTIGATION OF THE TWO FAILED TRANSFORMERS REVEALED DAMAGE ATTRIBUTABLE TO TURN-TO-TURN FAULTS IN BOTH TRANSFORMERS. THE DIRECT CAUSE OF THE TURN-TO-TURN FAULTS COULD NOT BE IDENTIFIED. CORRECTIVE ACTIONS INCLUDED CLARIFICATION AND EXPANSION OF REQUIRED INSPECTIONS FOR OIL FILLED AND DRY TYPE TRANSFORMERS AND PERIODIC OIL AND GAS SAMPLING OF OIL FILLED TRANSFORMERS.

[219] SALEM 2 DOCKET 50-311 LER 87-011 REV 01
UPDATE ON REACTOR TRIP OCCURRED DUE TO A HIGH HIGH STEAM GENERATOR LEVEL ATTRIBUTED TO PERSONNEL ERROR.
EVENT DATE: 080687 REPORT DATE: 090887 NSSS: WE TYPE: PWR

(NSIC 206050) ON AUGUST 6, 1987 AT 0912 HOURS, A TURBINE TRIP/REACTOR TRIP OCCURRED DUE TO NO. 24 STEAM GENERATOR (S/G) HIGH-HIGH LEVEL. AT THE TIME OF THE TRIP, A NO. 24 S/G LEVEL CHANNEL II FUNCTIONAL TEST WAS IN PROGRESS. OTHER THAN THE TRIP, NO ABNORMAL EVENTS WERE IDENTIFIED. THE ROOT CAUSE OF THE TRIP HAS BEEN ATTRIBUTED TO PERSONNEL ERROR. THE NUCLEAR CONTROL OPERATOR DURING THE FUNCTIONAL TEST DID NOT CORRECT A FEED-STEAM FLOW DEVIATION. A PERFORMANCE REVIEW OF THE NUCLEAR CONTROL OPERATOR HAS BEEN COMPLETED. APPROPRIATE CORRECTIVE DISCIPLINE HAS BEEN INITIATED. A DISCUSSION OF THIS EVENT WILL BE INCLUDED IN THE NEXT CYCLE OF OPERATOR REQUALIFICATION TRAINING AND WILL BE CONSIDERED FOR INCLUSION IN INITIAL OPERATOR TRAINING FOR THE S/G WATER LEVEL CONTROL SYSTEM. ADDITIONALLY, A COPY OF THIS LER WILL BE INCORPORATED INTO THE OPERATIONS DEPARTMENT NEWSLETTER.

[220] SAN ONOFRE 1 DOCKET 50-206 LER 87-012
CONTAINMENT AIR LOCK HATCH DRIVE FAILURE CAUSES A BRIEF BREACH OF CONTAINMENT INTEGRITY.
EVENT DATE: 082587 REPORT DATE: 092487 NSSS: WE TYPE: PWR
VENDOR: CHICAGO BRIDGE AND IRON COMPANY

(NSIC 206446) ON AUGUST 25, 1987, WITH UNIT 1 IN MODE 1 AT 92% POWER, AN ENTRY WAS MADE INTO CONTAINMENT VIA THE PERSONNEL AIR LOCK AT 0750. WHEN THE AIR LOCK CONTROLS WERE SUBSEQUENTLY OPERATED TO CLOSE THE AIR LOCK HATCH INSIDE OF CONTAINMENT, THE HATCH DID NOT FULLY CLOSE DUE TO HATCH DRIVE FAILURE. THIS CONDITION WAS NOT OBSERVABLE AT THE LOCATION FROM WHICH THE HATCH WAS BEING OPERATED. AT 0808, AS THE OUTER HATCH WAS BEING OPENED TO ADMIT ANOTHER PERSON TO CONTAINMENT, IT WAS OBSERVED THAT THE INNER HATCH WAS NOT FULLY CLOSED. THE OUTER HATCH WAS IMMEDIATELY CLOSED AND CONTAINMENT INTEGRITY WAS RESTORED IN LESS THAN 30 SECONDS. THE INNER AIR LOCK HATCH WAS MANUALLY CLOSED FROM INSIDE CONTAINMENT AT 0925. DURING THE BRIEF PERIOD THAT BOTH INNER AND OUTER AIR LOCK HATCHES WERE OPEN SIMULTANEOUSLY, THE UNIT WAS IN A CONDITION NOT PERMITTED BY TECHNICAL SPECIFICATION (TS) LIMITING CONDITION FOR OPERATION (LCO) 3.6.1.(B).(1). THE FAILED HATCH DRIVE HAS BEEN REPAIRED AND CORRECTIVE ACTIONS HAVE BEEN, OR WILL BE, IMPLEMENTED TO PREVENT RECURRENCE. THE HEALTH AND SAFETY OF PLANT PERSONNEL AND THE PUBLIC WERE NOT AFFECTED BY THIS VENT.

[221] SAN ONOFRE 2 DOCKET 50-361 LER 87-010
TOXIC GAS ISOLATION SYSTEM (TGIS) ACTUATION DURING FLOOR CLEANING.
EVENT DATE: 080687 REPORT DATE: 083187 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: SAN ONOFRE 3 (PWR)

(NSIC 206257) ON AUGUST 6, 1987, AT 1742, WITH UNITS 2 AND 3 AT 96% POWER AND 100% POWER, RESPECTIVELY, TRAIN 'A' OF THE TOXIC GAS ISOLATION SYSTEM (TGIS) (EIIS SYSTEM CODE VI) ACTUATED THE CONTROL ROOM EMERGENCY AIR CLEANUP SYSTEM (CREACUS) (EIIS SYSTEM CODE VI) ON HIGH BUTANE GAS LEVEL. CREACUS OPERATED IN THE ISOLATION MODE, AS DESIGNED, WHILE THE CAUSE FOR THE ACTUATION WAS INVESTIGATED. CONCRETE FLOOR CLEANING IN PREPARATION FOR PAINTING HAD COMMENCED IN AREAS OUTSIDE THE CONTROL ROOM PRESSURE BOUNDARY INCLUDING IN THE VICINITY OF THE TGIS SAMPLE POINT AIR INLETS. AT 1747, AFTER TGIS BUTANE ANALYZER READINGS WERE VERIFIED TO BE BELOW THE ACTUATION SETPOINT, TGIS WAS RESET, AND CREACUS WAS RETURNED TO THE STANDBY MODE. THERE WAS NO SAFETY SIGNIFICANCE TO THIS EVENT SINCE ALL TGIS AND CREACUS COMPONENTS FUNCTIONED AS DESIGNED. CHEMICAL RELEASE FROM THE CLEANING AGENT BEING USED IN THE AREA OF THE TGIS MONITOR EIIS COMPONENT CODE MON) SAMPLER AIR INLETS WAS DETERMINED TO HAVE CAUSED THE ACTUATION. GRAB SAMPLES OF VAPOR ASSOCIATED WITH THE CLEANING AGENT CONFIRMED THAT THE AGENT, CONSISTING OF ACETIC ACID, METHYLENE CHLORIDE, AND OTHER CHEMICALS, PRODUCES HYDROCARBONS SENSITIVE TO THE BUTANE ANALYZER. CLEANING WITH THE AGENT WAS DISCONTINUED IN THE AREA OF THE AIR INLETS.

[222] SAN ONOFRE 2 DOCKET 50-361 LER 87-012
CONTROL ROOM ISOLATION SYSTEM (CRIS) INADVERTENT ACTUATION DURING FUNCTIONAL TESTING.
EVENT DATE: 080887 REPORT DATE: 090487 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: SAN ONOFRE 3 (PWR)

(NSIC 206258) ON AUGUST 8, 1987, AT 1957, WITH UNIT 2 AT 95% POWER AND UNIT 3 AT 100% POWER, THE CONTROL ROOM EMERGENCY AIR CLEANUP SYSTEM (CREACUS) WAS INADVERTENTLY ACTUATED WHILE PERFORMING THE 31-DAY FUNCTIONAL TEST ON CONTROL ROOM ISOLATION SYSTEM (CRIS) TRAIN B RADIATION MONITOR. THE CAUSE OF THE CRIS ACTUATION WAS THE FAILURE OF THE TECHNICIAN TO PLACE THE MONITOR BACK IN "ALARM DEFEAT" PRIOR TO RELEASING THE "RESET/BYPASS" SWITCH. RECOGNIZING THIS ERROR, THE TECHNICIAN IMMEDIATELY RESET THE CRIS AND RETURNED THE MONITOR TO "ALARM DEFEAT". CREACUS COMPONENTS OPERATED ACCORDING TO DESIGN WITH THE EXCEPTION OF

CONTROL ROOM CABINET AREA EMERGENCY COOLING UNIT E-423, WHICH DID NOT START. E-423 WAS SUBSEQUENTLY STARTED AND STOPPED BY USING ITS CONTROL ROOM HANDSWITCH AND ALSO BY MANUALLY INITIATING CRIS TRAIN B. IT WAS CONCLUDED THAT THE FAN FAILURE WAS ASSOCIATED WITH THE IMMEDIATE RESETTING OF CRIS BY THE TECHNICIAN. THE TECHNICIAN INVOLVED WAS COUNSELED AND ALL OTHER TECHNICIANS INVOLVED WITH RADIATION MONITOR TESTING WERE BRIEFED ON THE EVENT. A SIMILAR INADVERTENT ACTUATION OF CRIS DURING PERFORMANCE OF THE SAME FUNCTIONAL TEST OCCURRED PREVIOUSLY AND WAS REPORTED IN LER 84-077. A DESIGN CHANGE WILL INSTALL A KEYLOCK SWITCH FOR THE BYPASS FUNCTION AND A SPRING-LOADED SWITCH FOR THE RESET FUNCTION.

[223] SAN ONOFRE 3 DOCKET 50-362 LER 87-003 REV 01
UPDATE ON RADIOACTIVE PARTICLES IN AN UNRESTRICTED AREA.
EVENT DATE: 020287 REPORT DATE: 090187 NSSS: CE TYPE: PWR

(NSIC 206259) ON FEBRUARY 2, 1987, A FUEL FRAGMENT (PARTICLE) WAS INADVERTENTLY REMOVED FROM THE SAN ONOFRE RESTRICTED AREA BY A WORKER. THE PARTICLE WAS RETURNED TO THE RESTRICTED AREA AND LOCATED ON FEBRUARY 3, 1987, ON A GARMENT THE WORKER HAD WORN IN THE PLANT THE DAY BEFORE. HEALTH PHYSICS (HP) TECHNICIANS HAD BEEN UNABLE, THE DAY BEFORE, TO FIND THE PARTICLE EVEN THOUGH IT APPARENTLY HAD REPEATEDLY ALARMED THE NEWLY INSTALLED AND HIGHLY BETA SENSITIVE PERSONNEL MONITORING BOOTHS. ON FEBRUARY 21, 1987, AN HP TECHNICIAN DISCOVERED A PARTICLE IN HIS HOME AS THE RESULT OF SELF-INITIATED SURVEY. THE PARTICLE WAS RETURNED TO THE SITE THE SAME DAY. THE ARTICLE MAY HAVE LEFT THE SITE ON THE INDIVIDUAL OR ON ONE OF THE SURVEY INSTRUMENTS USED BY THE TECHNICIAN TO PERFORM THE HOME SURVEY. THE CAUSE OF THIS EVENT WAS EVIDENTLY LACK OF THOROUGHNESS IN PERSONNEL OR EQUIPMENT MONITORING ON THE PART OF THE TRAINED AND KNOWLEDGEABLE TECHNICIAN. SUBSEQUENT TO THE FEBRUARY 2 OCCURRENCE, HP TECHNICIANS HAVE BEEN PROVIDED ADDITIONAL INSTRUCTION AND ADDITIONAL ADMINISTRATIVE CONTROLS HAVE BEEN DEVELOPED TO PRECLUDE FUTURE RELEASE OF POTENTIALLY CONTAMINATED INDIVIDUALS.

[224] SAN ONOFRE 3 DOCKET 50-362 LER 87-015
CONTAINMENT PURGE ISOLATION SYSTEM ACTUATION DUE TO FAILURE OF AIRBORNE MONITOR 3RT-7807.
EVENT DATE: 083087 REPORT DATE: 092987 NSSS: CE TYPE: PWR
VENDOR: NUCLEAR MEASUREMENTS CORP.

(NSIC 206598) ON 8/30/87, AT 0636, CONTAINMENT PURGE ISOLATION SYSTEM (CPIS) TRAIN "B" ACTUATED ON AN INSTRUMENT FAILURE SIGNAL FROM CONTAINMENT AIRBORNE MONITOR 3RT-7807 (IODINE CHANNEL). THERE WAS NO PURGE IN PROGRESS; THEREFORE, CPIS COMPONENTS DID NOT ACTUATE. AT 1051, THE FAILED MONITOR WAS REMOVED FROM SERVICE, AND CPIS WAS RESET. IT WAS DETERMINED THAT THE DETECTOR FAILED DUE TO A FAILURE OF THE PHOTOMULTIPLIER (PM) TUBE. CPIS TRAIN "B" WAS RETURNED TO OPERABLE STATUS ON 9/8/87, FOLLOWING COMPLETION OF PM TUBE REPLACEMENT AND SATISFACTORY FUNCTIONAL TESTING OF THE MONITOR. A CALCULATION OF THE FAILURE RATE OF THE PM TUBES AT SAN ONOFRE AND A REVIEW OF VENDOR AND INDUSTRY DATA DETERMINED THAT THE EXPERIENCED FAILURE RATE IS WELL WITHIN THE EXPECTED FAILURE RATE TYPICAL FOR THIS EQUIPMENT. THEREFORE, ADDITIONAL CORRECTIVE ACTION IS NOT WARRANTED AT THIS TIME. THERE WAS NO SAFETY SIGNIFICANCE TO THIS EVENT SINCE, HAD A CONTAINMENT PURGE BEEN IN PROGRESS, THE INSTRUMENT FAILURE WOULD HAVE RESULTED IN CPIS PERFORMING ITS SAFETY FUNCTION, AS DESIGNED.

[225] SEABROOK 1 DOCKET 50-443 LER 87-015
INADVERTENT SAFETY INJECTION DUE TO INSPECT A MAIN CONTROL BOARD.
EVENT DATE: 081387 REPORT DATE: 091187 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 206189) ON 08/13/87 AT 1:30PM 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. THE B CENTRIFUGAL CHARGING PUMP AND EMERGENCY DIESEL GENERATOR STARTED, TRAIN B SAFETY INJECTION SYSTEM HIGH AND LOW HEAD VALVES REALIGNED, A TRAIN B PHASE A CONTAINMENT ISOLATION OCCURRED, A TRAIN B CONTAINMENT VENTILATION ISOLATION SIGNAL WAS GENERATED, CONTROL BUILDING TRAIN B EMERGENCY AIR CLEANUP FAN STARTED, AND MAIN FEEDWATER ISOLATION OCCURRED. 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).

[226] SEABROOK 1 DOCKET 50-443 LER 87-016
CONTROL ROOM MAKEUP VENTILATION SYSTEM ISOLATION LOGIC DUE TO SYSTEM HARDWARE DESIGN AND PROCEDURE INADEQUACIES INOPERABLE.
EVENT DATE: 081987 REPORT DATE: 091887 NSSS: WE TYPE: PWR

(NSIC 206190) ON AUGUST 19, 1987, DURING THE PERFORMANCE OF THE MONTHLY TECHNICAL SPECIFICATION SURVEILLANCE 4.3.3.1, IT WAS IDENTIFIED THAT TRAINS A & B OF THE CONTROL ROOM MAKEUP VENTILATION SUPPLY AND EXHAUST SUBSYSTEM WOULD NOT HAVE PERFORMED THEIR INTENDED SAFETY FUNCTION. TECHNICAL SPECIFICATION SURVEILLANCE 4.3.3.1 REQUIRES THAT THE REDUNDANT RADIATION MONITORS BE CAPABLE OF ISOLATING THE CONTROL ROOM MAKEUP VENTILATION SUPPLY UPON RECEIPT OF A HIGH RADIATION SIGNAL FROM ANY OF THE RADIATION MONITORS. CONTRARY TO THIS, THE SURVEILLANCE OF THE EAST AIR INTAKE MONITORS DID NOT PROVIDE THE ISOLATION SIGNALS WHEN A TEST SIGNAL WAS INDUCED. UPON INVESTIGATION IT WAS DETERMINED THAT THE CONTROL LOGIC WAS NOT PROPERLY ALIGNED TO "ARM" THE ISOLATION CIRCUIT, THUS RENDERING THE ISOLATION FUNCTION INOPERABLE. THE ROOT CAUSE OF THE EVENT HAS BEEN ATTRIBUTED TO SYSTEM HARDWARE DESIGN AND PROCEDURE INADEQUACIES. A REVIEW WAS PERFORMED AND THERE WAS NO INDICATION THAT THE FUNCTION HAD BEEN ARMED SINCE THE PREVIOUS SURVEILLANCE PERFORMED ON JULY 20, 1987. TO PREVENT RECURRENCE, A TEMPORARY MODIFICATION HAS BEEN COMPLETED THAT ADDED AN INDICATOR LIGHT TO EACH OF THE RADIATION MONITOR ISOLATION CIRCUITS TO INDICATE THE ARMED STATUS FOR THE CONTROL LOGIC.

[227] SEQUOYAH 1 DOCKET 50-327 LER 85-031 REV 02
UPDATE ON AUXILIARY BUILDING ISOLATION OCCURRED AS A RESULT OF HIGH RADIATION LEVELS FROM A CRACKED CVCS SAMPLE LINE WELD DUE TO FATIGUE FAILURE.
EVENT DATE: 072985 REPORT DATE: 082987 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 206426) THIS LER IS BEING REVISED TO INCLUDE ADDITIONAL INFORMATION ON THE EVENT AND TO AMEND THE CORRECTIVE ACTION. THIS LER IS REVISED IN ITS ENTIRETY TO COMPLY WITH NUREG-1022. ON JULY 29, 1985, WITH UNITS 1 AND 2 IN MODE 1 AT 100 PERCENT POWER, AN AUXILIARY BUILDING ISOLATION OCCURRED AT 2225 CST. THE ISOLATION WAS AUTOMATICALLY INITIATED BY HIGH RADIATION LEVELS WHICH WERE DETECTED BY THE AUXILIARY BUILDING STACK VENTILATION MONITOR, RM-90-101. UPON THE ISOLATION, THE AUXILIARY BUILDING GAS TREATMENT SYSTEM STARTED AND PROVIDED FILTRATION OF EFFLUENTS EXITING THE BUILDING TO THE ENVIRONMENT. ALL RELEASES TO THE ENVIRONMENT WERE WELL WITHIN THE OFFSITE DOSE LIMITS. THE LEAK WAS LOCATED ON THE UNIT 2 LETDOWN LINE AT A SAMPLE LINE CONNECTION WELD UPSTREAM OF VALVE 2-62-674 IN THE CHEMICAL AND VOLUME CONTROL SYSTEM. THIS SAMPLE LINE IS DOWNSTREAM OF THE LETDOWN HEAT EXCHANGERS AND UPSTREAM OF THE VOLUME CONTROL TANK. THE LEAK WAS ISOLATED, AND CLEANUP WAS INITIATED FOR APPROXIMATELY 600

GALLONS OF REACTOR COOLANT FLUID WHICH DRAINED TO A COLLECTION TANK. THE CAUSE OF THE FAILURE OF THE METAL IN THE WELD HEAT AFFECTED ZONE WAS DETERMINED TO BE HIGH CYCLE, VIBRATION-INDUCED FATIGUE INITIATED FROM THE OUTSIDE DIAMETER OF THE PIPE.

[228] SEQUOYAH 1 DOCKET 50-327 LER 87-026 REV 01
UPDATE ON INADEQUATE RADIOLOGICAL CONTROL COVERAGE IN THE TRANSFER OF RADIOACTIVE WASTE CAUSED BY PERSONNEL ERROR RESULTS IN TECHNICAL SPECIFICATION VIOLATION.
EVENT DATE: 052687 REPORT DATE: 090987 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 206436) THIS LER IS REVISED TO CLARIFY THE DESCRIPTION OF EVENT AND CORRECTIVE ACTIONS. ON 5/26/87, WITH BOTH UNITS IN MODE 5 (COLD SHUTDOWN), A RADWASTE CASK SHIPPING LINER CONTAINING SOLIDIFIED RADWASTE WAS TRANSFERRED WITHIN THE AUX. BUILDING FROM THE RAILROAD ACCESS BAY UP TO THE REFUELING FLOOR TO BE LOWERED INTO THE CASK DECONTAMINATION ROOM. DOSE RATES FROM THE LINER EXCEEDED 1,000 MREM/HR. AT TWO TIMES IN THE TRANSFER PROCESS WHILE THE LINER WAS IN TRANSIT AND WHILE THE LINER WAS BEING LOWERED INTO THE CASK DECONTAMINATION ROOM, RADIOLOGICAL CONTROL (RAD CON) COVERAGE DID NOT PROVIDE POSITIVE CONTROL OVER THE ACTIVITIES WITHIN THESE AREAS RESULTING IN VIOLATION OF UNITS 1 & 2 TECH SPEC 6.12.2. UPON REALIZING THAT POSITION CONTROL WAS NOT EXHIBITED, A RADCON TECHNICIAN RESPONDED TO PROVIDED THE REQUIRED COVERAGE. THIS EVENT WAS THE RESULT OF PERSONNEL ERROR WHEN FIRST, THE MOVEMENT OF THE LINER FROM THE RAILROAD BAY TO THE CASK DECONTAMINATION ROOM WAS NOT PROPERLY COVERED AS REQUIRED BY TECH SPEC 6.12.2 AND RADIOLOGICAL CONTROL INSTRUCTION (RCI)-13, "ACCESS CONTROL TO HIGH RADIATION AREAS WHEN RADIATION INTENSITY IS GREATER THAN OR EQUAL TO 1,000 MREM/HR." SECOND, WHEN THE LINER WAS LOWERED INTO THE CASK DECONTAMINATION ROOM, AN INDIVIDUAL WAS PRESENT IN THE ROOM WITHOUT PROPER RAD CON COVERAGE AS REQUIRED.

[229] SEQUOYAH 1 DOCKET 50-327 LER 87-046
THE POTENTIAL EXISTS FOR UNACCEPTABLE DILUTION OF THE ECCS DURING A LARGE BREAK LOCA DUE TO POSTULATED NON-BORATED WATER ADDITION TO CONTAINMENT SUMP.
EVENT DATE: 072287 REPORT DATE: 081887 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 205899) THIS LER IS BEING PROVIDED AS A VOLUNTARY REPORT TO INFORM NRC OF A POTENTIAL ISSUE THAT HAS GENERIC IMPLICATIONS TO ALL WESTINGHOUSE PRESSURIZED WATER REACTORS. TENNESSEE VALLEY AUTHORITY RECEIVED A TECHNICAL BULLETIN FROM WESTINGHOUSE CONCERNING BORON CONCENTRATION REQUIREMENTS AFTER A LOSS OF COOLANT ACCIDENT (LOCA), ALL CONTROL RODS OUT (ARO), LONG-TERM COOLING MODE OF OPERATION. THE BULLETIN STATES THAT THE RECENT TREND TOWARDS LONGER RELOAD CYCLE LENGTHS AND THE INTRODUCTION OF A POSITIVE MODERATOR TEMPERATURE COEFFICIENT AT LOWER POWER LEVELS HAVE RESULTED IN THE NEED TO RECONFIRM THAT THE REACTOR CORE REMAINS SUBCRITICAL DURING A POST-LOCA, ARO, LONG-TERM COOLING MODE OF OPERATION. THE CONCERN IS WHETHER THERE IS AN ACCEPTABLE BORON CONCENTRATION IN THE EMERGENCY CORE COOLING SYSTEM TO MAINTAIN THE REACTOR CORE SUBCRITICAL DURING A LARGE BREAK LOCA WITH THE ADDITION OF NONBORATED WATER TO THE PRIMARY CONTAINMENT SUMP. THE ESSENTIAL RAW COOLING WATER, HIGH PRESSURE FIRE PROTECTION, PRIMARY WATER, AND COMPONENT COOLING WATER PIPING SYSTEMS ARE POTENTIAL SOURCES FOR NONBORATED WATER. AN EVALUATION IS PRESENTLY BEING PERFORMED TO DETERMINE IF THERE ARE ANY SAFETY IMPLICATIONS AND IF AN CORRECTIVE ACTIONS WILL BE NECESSARY. CURRENT INDICATIONS ARE THAT AN UNACCEPTABLE BORON DILUTION CONCERN DOES NOT EXIST.

[230] SEQUOYAH 1 DOCKET 50-327 LER 87-042
INADVERTENT STARTING OF FIRE PUMPS DURING A LOCA COULD DEGRADE THE AUXILIARY POWER SYSTEM BECAUSE OF A DESIGN ERROR.
EVENT DATE: 072387 REPORT DATE: 081887 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 205897) THIS REPORT IS BEING SUBMITTED AS A "VOLUNTARY LER" TO IDENTIFY A POTENTIAL PROBLEM WITH THE SEQUOYAH 1E AUXILIARY POWER SYSTEM AND TO KEEP NRC INFORMED OF ONGOING ACTIVITIES AT SEQUOYAH NUCLEAR PLANT. A CONDITION ADVERSE TO QUALITY WAS REPORTED ON APRIL 14, 1987, WITH BOTH UNITS 1 AND 2 IN MODE 5 (COLD SHUTDOWN) DESCRIBING A CONDITION WHERE CALCULATIONS FOR THE AUXILIARY ELECTRIC POWER SYSTEM HAVE NOT INCLUDED THE EFFECT ON SAFETY-RELATED EQUIPMENT OF THE FIRE PUMPS STARTING AND RUNNING DURING A LOSS OF COOLANT ACCIDENT (LOCA). THE FIRE PUMPS ARE SUPPLIED POWER FROM THE CLASS 1E POWER SYSTEM; HOWEVER, THE DESIGN BASIS FOR SEQUOYAH NUCLEAR PLANT DOES NOT INCLUDE A LOCA AND CONCURRENT FIRE. DURING A LOCA, THE CONTAINMENT TEMPERATURE CAN BE HIGH ENOUGH TO CAUSE THE FIRE PUMPS TO START BECAUSE OF THE ACTUATION OF THE TEMPERATURE SENSORS ON THE FIRE DETECTION SYSTEM. FIRE DETECTION SENSORS THAT DETECT IONIZED AIR CAN ALSO BE ACTUATED BY LOCA CONDITIONS AND CAUSE THE FIRE PUMPS TO START. STARTING THE FIRE PUMPS CONCURRENT WITH A LOCA COULD POTENTIALLY DEGRADE THE AUXILIARY ELECTRIC POWER SYSTEM VOLTAGE AND THEREBY PREVENT SAFETY-RELATED EQUIPMENT FROM PERFORMING ITS INTENDED FUNCTION. THE ROOT CAUSE OF THE CONDITION WAS A DESIGN ERROR WHEN THE DESIGN ENGINEER FAILED TO CONSIDER THE POSSIBILITY OF INADVERTENTLY STARTING THE FIRE PUMPS DURING A LOCA.

[231] SEQUOYAH 1 DOCKET 50-327 LER 87-052
DESIGN ERROR RESULTING IN NONREPRESENTATIVE LOAD TESTING OF DIESEL GENERATORS.
EVENT DATE: 072387 REPORT DATE: 082287 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)
VENDOR: MORRISON-KNUDSON COMPANY, INC.

(NSIC 205901) THIS REPORT IS BEING SUBMITTED AS A "VOLUNTARY LER" TO IDENTIFY POTENTIAL PROBLEMS WITH THE EMERGENCY DIESEL GENERATORS AND TO KEEP NRC INFORMED OF ONGOING ACTIVITIES AT SEQUOYAH NUCLEAR PLANT. A SIGNIFICANT CONDITION REPORT (SCR) WAS INITIATED ON MARCH 20, 1986, WITH BOTH UNITS 1 AND 2 IN MODE 5 (COLD SHUTDOWN), DESCRIBING A CONDITION WHERE THE CAPABILITY OF EMERGENCY DIESEL GENERATOR 2B-B TO RECOVER FROM THE TRANSIENT OF THE CONTAINMENT SPRAY PUMP STARTING FOLLOWING A PHASE B CONTAINMENT ISOLATION WITH OTHER RANDOM LOADS CONNECTED WAS UNCERTAIN. A REMOTE POSSIBILITY EXISTS THAT THE ELECTRIC BOARD ROOM AIR HANDLING UNIT COULD START AT PRECISELY THE SAME TIME THAT THE CONTAINMENT SPRAY PUMP STARTS AND RESULT IN THE SPEED OF THE DIESEL GENERATOR TO DROP BELOW THE FIVE PERCENT LIMITATION DESCRIBED IN FINAL SAFETY ANALYSIS REPORT (FSAR). A SECOND SCR WAS INITIATED ON JANUARY 14, 1987, WITH BOTH UNITS IN MODE 5, THAT IDENTIFIED A PROBLEM WHERE THE DIESEL GENERATORS' CAPABILITY TO PROVIDE ADEQUATE ONSITE POWER HAD NOT BEEN DEMONSTRATED IN ACCORDANCE WITH THE FSAR. THE CAUSE OF THESE CONDITIONS WAS A DESIGN ERROR. DESIG DESIGN ENGINEERS FAILED TO PROPERLY EVALUATE THE INTEGRATED OPERATION AND RESPONSE OF THE DIESEL GENERATORS, THEIR LOADING, AND TO SPECIFY THE CORRECT PREOPERATIIONAL TEST.

[232] SEQUOYAH 1 DOCKET 50-327 LER 87-044
INADEQUATE COMMUNICATION BETWEEN DESIGN ORGANIZATIONS RESULTS IN POTENTIAL PROBLEM WITH 1E ELECTRICAL EQUIPMENT DUE TO UNANALYZED FLOODING EFFECTS.
EVENT DATE: 072487 REPORT DATE: 081987 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 205898) THIS LER IS BEING PROVIDED AS A VOLUNTARY REPORT TO INFORM NRC OF A POTENTIALLY GENERIC ISSUE CONCERNING POSTULATED BREAKS IN MODERATE ENERGY LINES. FLOODING EFFECTS OF MODERATE ENERGY LINE BREAKS (MELBS) WERE NOT ADEQUATELY ANALYZED DUE TO INADEQUATE COMMUNICATION BETWEEN THE PIPE BREAK ANALYST AND THE SYSTEM ENGINEERS RESPONSIBLE FOR PERFORMING THE ANALYSIS. AS A RESULT, TVA CONTRACTED WITH SARGENT AND LUNDY TO PERFORM A STUDY OF MELBS. SARGENT AND LUNDY HAS DETERMINED THAT FLOODING ASSOCIATED WITH MODERATE ENERGY LINE BREAKS (MELBS) COULD POTENTIALLY SUBMERGE AND IMPAIR COMPONENTS REQUIRED FOR SAFE SHUTDOWN AND THREATEN INTEGRITY OF CERTAIN STRUCTURES DUE TO INCREASED LOADING. ALL IDENTIFIED CONCERNS RESULTING FROM THE MELB FLOODING STUDY HAVE BEEN ADDRESSED BY TVA AND

PREVIOUSLY REPORTED OR EVALUATED BY A REALISTIC ANALYSIS WHICH INDICATES NO FAILURES ARE LIKELY TO OCCUR. TO COMPLY WITH TVA INTERNAL DESIGN STANDARDS, CERTAIN ACTIONS (SUCH AS SEALING OF BUILDING WALLS) ARE BEING TAKEN TO MINIMIZE THE IMPACT OF POTENTIAL MELBS BEFORE RESTART OF UNIT 2. OTHER ACTIONS (SUCH AS PROTECTING CABLES FROM WATER) ARE BEING IMPLEMENTED BEFORE RESTART FROM FUEL CYCLE 5 FOR EITHER UNIT. ADDITIONAL ACTIONS HAVE BEEN TAKEN TO IMPROVE THE COMMUNICATIONS ACROSS THE ENGINEERING DISCIPLINES.

[233] SEQUOYAH 1 DOCKET 50-327 LER 87-037
THE CAPACITY OF THE ENGINEERED SAFETY FEATURES EQUIPMENT COOLERS HAS BEEN DETERMINED INADEQUATE FOR LOCA CONDITIONS DUE TO HVAC CALCULATION ERRORS.
EVENT DATE: 072887 REPORT DATE: 082787 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)
VENDOR: AMERICAN FOUNDRY & FURNACE
PORTER, H. K. COMPANY, INC.

(NSIC 206290) WITH UNITS 1 AND 2 IN MODE 5 (COLD SHUTDOWN), A CONDITION ADVERSE TO QUALITY WAS IDENTIFIED IN THAT THE CAPABILITY OF THE UNIT 1 AND UNIT 2 ENGINEERED SAFETY FEATURES (ESF) ROOM COOLERS WAS INDETERMINATE, AND THAT A REVIEW OF MECHANICAL CALCULATIONS FOR TECHNICAL ADEQUACY WAS REQUIRED. DURING THIS REVIEW, IT WAS DISCOVERED THAT WATTS BAR NUCLEAR PLANT'S HEAT LOAD DATA HAD BEEN USED AS THE BASIS FOR THE SEQUOYAH COOLING CALCULATIONS. TVA DETERMINED THAT A COOLING LOAD CALCULATION VERIFICATION PROGRAM WAS REQUIRED TO DETERMINE THE EFFECTS OF USING THE APPROPRIATE SEQUOYAH HEAT LOAD DATA. TWENTY-THREE ESSENTIAL HEATING, VENTILATION, AND AIR CONDITIONING (HVAC) CALCULATIONS WERE REGENERATED USING THE LATEST SEQUOYAH HEAT LOAD DATA BASED ON A LOSS OF COOLANT ACCIDENT (LOCA) CONDITION, WHICH ASSUMES ESF EQUIPMENT IS IN OPERATION AND THE HVAC IS OPERATING IN THE EMERGENCY MODE. THE PROGRAM CONCLUDED THAT BASED ON THE PRESENT REQUIREMENTS FOR ESSENTIAL RAW COOLING WATER (ERCW) FLOW TO THE COOLERS, THREE (PER TRAIN) AREA/ROOM COOLER CALCULATION RESULTS INDICATED THE CAPACITY OF THE COOLERS TO BE INADEQUATE. THE REMAINING 20 COOLERS HAD THE CAPACITY TO PERFORM THEIR INTENDED FUNCTION BASED ON THE REGENERATED CALCULATION RESULTS.

[234] SEQUOYAH 1 DOCKET 50-327 LER 87-041 REV 01
UPDATE ON LOSS OF RESIDUAL HEAT REMOVAL FLOW RESULTING FROM MISPOSITIONING OF A BREAKER DUE TO PERSONNEL ERROR.
EVENT DATE: 080487 REPORT DATE: 091487 NSSS: WE TYPE: PWR

(NSIC 206515) ON AUGUST 4, 1987, AT 2149 EDT, WITH UNIT 1 IN MODE 5 (0 PERCENT POWER, 2 PSIG, 133 DEGREES F), A MOMENTARY LOSS OF RESIDUAL HEAT REMOVAL (RHR) FLOW OCCURRED WHEN VALVE 1-FCV-74-2 INADVERTENTLY CLOSED, ISOLATING RHR PUMP 1A-A SUCTION FROM THE REACTOR COOLANT SYSTEM LOOP 4 HOT LEG. THE CONTROL ROOM OPERATOR WAS IMMEDIATELY ALERTED TO THIS CONDITION BY ANNUNCIATION OF THE RHR HIGH DISCHARGE, MINI-FLOW ALARM. THE OPERATOR ASSESSED THE POSITION OF RHR SUCTION VALVE 1-FCV-74-2 AND MANUALLY STOPPED RHR PUMP 1A-A. THE APPLICABLE ACTION STATEMENT OF LIMITING CONDITION FOR OPERATION (LCO) 3.4.1.4 WAS ENTERED AND ABNORMAL OPERATING INSTRUCTION (AOI)-14 WAS PLACED IN EFFECT FOR USE IN RECOVERY OPERATIONS. RHR LOOP A FLOW WAS REESTABLISHED AT 2153 EDT ON AUGUST 4, 1987, BY REOPENING VALVE 1-FCV-74-2 AND RESTARTING RHR PUMP 1A-A, AT WHICH TIME LCO 3.4.1.4 WAS EXITED. THE RHR SYSTEM ISOLATION WAS CAUSED BY PERSONNEL ERROR DURING INITIAL PLANT LINEUP FOR PERFORMANCE OF INSTRUMENT MAINTENANCE INSTRUCTION (IMI)-99 RT-699B. A STUDENT ASSISTANT UNIT OPERATOR (AUO) WAS DISPATCHED TO 120V AC VITAL INSTRUMENT POWER TO VERIFY THE POSITION OF BREAKER 2 (THE CHECKLIST CALLED FOR THE BREAKER TO BE CLOSED). THE AUO TRAINEE, FOR NO EXPLAINABLE REASON, OPENED THE BREAKER AND THEN REALIZING HIS MISTAKES CLOSED IT BACK.

[235] SEQUOYAH 1 DOCKET 50-327 LER 87-053
INADVERTENT DIESEL GENERATOR START CAUSED BY PERSONNEL INATTENTION TO DETAIL AND LACK OF PROCEDURAL CLARITY DURING INSTALLATION OF A JUMPER.
EVENT DATE: 080587 REPORT DATE: 082887 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 206292) ON 8/5/87 AT 0015 EST WITH UNITS 1 AND 2 IN MODE 5 (0 PERCENT POWER, 2 PSIG, 133 DEGREES F AND 0 PERCENT POWER, 0 PSIG, 119 DEGREES F, RESPECTIVELY, THE 1B-B DIESEL GENERATOR WAS INADVERTENTLY STARTED DURING CONNECTION OF A TEMPORARY JUMPER IN THE TRAIN B SOLID STATE PROTECTION SYSTEM (SSPS) OUTPUT RELAY CABINET FOR THE PERFORMANCE OF INSTRUMENT MAINTENANCE INSTRUCTION (IMI)-99-RT699B, "RESPONSE TIME TESTING OF ENGINEERED SAFETY INJECTION SIGNAL WITH STATION BLACKOUT UNITS 1 AND 2." THE PROCEDURE REQUIRED PLACEMENT OF THE JUMPER ACROSS A SET OF SSPS RELAY CONTACTS TO PREVENT A DIESEL START FROM THE SAFETY INJECTION SIGNAL (SIS) ONLY. THE PROCEDURE SPECIFIED NOT TO LIFT THE WIRES ON EITHER SIDE OF THE CONTACTS; HOWEVER, TO FACILITATE THE ATTACHMENT OF A SPADE LUG TYPE JUMPER, THE TERMINAL SCREWS WERE LOOSENED CAUSING A MOMENTARY LOSS OF CONTINUITY ACROSS THE CONTACTS. THIS IN TURN CAUSED A SIMULATED SIS AND STARTED THE 1B-B DIESEL GENERATOR. THE OTHER THREE DIESEL GENERATORS (1A-A, 2A-A, AND 2B-B) DID NOT START BECAUSE OF THE COMMON START CIRCUITRY BEING INHIBITED UNDER THE DIRECTION OF THIS PROCEDURE. IMMEDIATE CORRECTIVE ACTIONS WERE TO REMOVE THE SPADE LUG JUMPER AND REPLACE IT WITH AN ALLIGATOR TYPE JUMPER. THE DIESEL GENERATOR WAS THEN SHUT DOWN AT 0041 EST, INITIAL TEST CONDITIONS WERE REVERIFIED, AND THE TEST WAS CONTINUED.

[236] SEQUOYAH 1 DOCKET 50-327 LER 87-049
INADEQUATE PROCEDURE DURING CONSTRUCTION RESULTED IN IMPROPERLY SIZED MOTOR THERMAL OVERLOAD PROTECTION.
EVENT DATE: 080687 REPORT DATE: 090587 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 206435) ON DECEMBER 1, 1986, WITH BOTH UNITS 1 AND 2 IN MODE 5 (COLD SHUTDOWN), A SIGNIFICANT CONDITION REPORT (SCR) WAS INITIATED DESCRIBING A CONDITION IN WHICH THE MOTOR OVERLOADS WERE IMPROPERLY SIZED ON MOTOR-OPERATED VALVES (MOVS) AND OTHER SAFETY-RELATED LOW-VOLTAGE INDUCTION MOTORS. THERE WAS NO ASSURANCE THAT DURING A CONDITION WHERE THE VOLTAGE WAS NEAR THE DEGRADED VOLTAGE RELAY SETPOINT AND THE AMBIENT TEMPERATURE WAS AT THE UPPER LIMIT OF THE DESIGN BASIS (104 DEGREES F), THAT THE INSTALLED OVERLOADS WOULD ALLOW THE MOTOR STARTERS TO PERFORM THEIR SAFETY-RELATED ACTION WITHOUT UNCERTAINTIES. THE OVERLOADS WERE INSTALLED WITHOUT CONSIDERING MINIMUM VOLTAGE LEVEL AND WERE SIZED TO PROTECT THE VALVES IN THE EVENT THAT THEY MALFUNCTIONED RATHER THAN ENSURING THAT THE SAFETY FUNCTION WAS FULFILLED. ADDITIONALLY, THE THERMAL OVERLOAD PROTECTION DEVICES HAVE NOT YET BEEN FUNCTIONAL TESTED AT 104 DEGREES F. THE ROOT CAUSE OF THIS CONDITION WAS AN INADEQUATE PROCEDURE DURING CONSTRUCTION. THE CONSTRUCTION ENGINEERS SELECTED OVERSIZED LOADS FROM A PROCEDURE PROVIDED BY DESIGN THAT WAS NOT ADEQUATE TO ENSURE THAT THE OVERLOADS WOULD ALLOW THE MOTORS TO PERFORM THEIR SAFETY-RELATED ACTION WITHOUT UNCERTAINITIES.

[237] SEQUOYAH 1 DOCKET 50-327 LER 87-055
LOSS OF AUXILIARY FEEDWATER DUE TO INADEQUATE INSTALLATION OF INSTRUMENTATION SENSE LINES.
EVENT DATE: 080687 REPORT DATE: 090587 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 206308) THIS REPORT IS BEING SUBMITTED AS A VOLUNTARY LER TO IDENTIFY A POTENTIAL PROBLEM WITH THE SEQUOYAH AUXILIARY FEEDWATER (AFW) SYSTEM. ON FEBRUARY 17, 1987, SEQUOYAH PLANT PERSONNEL WERE NOTIFIED OF SIGNIFICANT CONDITION REPORT (SCR) SQNEEB8743, REVISION 0. AN SCR IS USED BY THE DIVISION OF NUCLEAR ENGINEERING (DNE) TO IDENTIFY CONDITIONS ADVERSE TO QUALITY. THIS SCR IS

APPLICABLE TO UNITS 1 AND 2 AND IDENTIFIED THAT A LOSS OF WATER SUPPLY TO THE AFW MAY OCCUR AS THE RESULT OF THE INADEQUATE ROUTING OF PRESSURE SWITCH SENSE LINES. THE AFW SYSTEM FUNCTION IS TO SUPPLY FEEDWATER FLOW TO THE STEAM GENERATORS SO THAT REACTOR COOLANT SYSTEM STORED HEAT AND REACTOR DECAY HEAT CAN BE DISSIPATED THROUGH THE MAIN STEAM SYSTEM STEAM RELIEVING FEATURES. THE CAUSE OF THIS CONDITION HAS BEEN DETERMINED TO BE INADEQUATE DESIGN OUTPUT SPECIFICATIONS FOR INSTALLATION OF INSTRUMENT LINES. NO IMMEDIATE CORRECTIVE ACTION IS REQUIRED BASED ON THE AFW SYSTEM NOT BEING REQUIRED IN THE PRESENT MODE OF OPERATION (MODE 5) FOR BOTH UNITS. ENGINEERING CHANGE NOTICES ARE PRESENTLY BEING IMPLEMENTED TO RESOLVE THE INSTALLATION DEFICIENCIES APPLICABLE TO UNITS 1 AND 2 PRESSURE SWITCHES SENSE LINES FOR THE MOTOR-DRIVEN AFW PUMPS.

[238] SEQUOYAH 1 DOCKET 50-327 LER 87-054
TECH SPEC ACTION REQUIREMENT NOT MET ON CONTROL ROOM ISOLATION RAD MONITORS BECAUSE EQUIPMENT WAS INOPERABLE FOR REQUIRED TESTING.
EVENT DATE: 081087 REPORT DATE: 090887 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)
VENDOR: GENERAL ATOMIC CO.

(NSIC 206433) ON 8/10/87, AT 1100 EST WITH UNITS 1 & 2 IN MODE 5 (0% POWER, 4 PSIG, 130 F AND 0% POWER, 0 PSIG, 130 F, RESPECTIVELY), IT WAS NECESSARY TO REMOVE FROM SERVICE TRAIN B CONTROL ROOM ISOLATION RADIATION MONITOR RM-90-126 TO PERFORM SURVEILLANCE INSTRUCTION (SI)-83, "CHANNEL CALIBRATION FOR RADIATION MONITORING SYSTEM." IT WAS NECESSARY TO REMOVE THIS RADIATION MONITOR FROM SERVICE BECAUSE OF THE TECH SPEC SURVEILLANCE REQUIREMENT (SR) FOR CHANNEL CALIBRATION BEING DUE. TRAIN A CONTROL ROOM ISOLATION RADIATION MONITOR RADIATION MONITOR RM-90-125 WAS INOPERABLE BECAUSE OF A HIGH VOLTAGE SPECIAL TEST PERFORMANCE ON SELECTED 1E CABLES WITH POTENTIAL FOR CABLE PULLBY DAMAGE. THE CABLES BEING TESTED WERE IN THE OUTPUT CIRCUITRY OF THE RM-90-125 RADIATION MONITOR. WITH BOTH TRAINS INOPERABLE, TECH SPEC REQUIRED THAT WITHIN 1 HOUR TO INITIATE AND MAINTAIN OPERATION OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM (CREVS) IN THE RECIRCULATION MODE OF OPERATION. THIS COULD NOT BE ACCOMPLISHED AS BOTH TRAINS OF THE CREVS WERE INOPERABLE PENDING RESOLUTION OF A DESIGN DEFICIENCY, WHICH WAS PREVIOUSLY REPORTED BY LER SQRO-50-327/87039. THE ROOT CAUSE OF THIS EVENT WAS A COMBINATION OF THE SR BEING DUE ON RM-90-126, THE REQUIRED HIGH VOLTAGE TESTING BEING PERFORMED ON RM-90-125 AND THE DESIGN DEFICIENCY ON THE CREVS ALL EXISTING CONCURRENTLY.

[239] SEQUOYAH 1 DOCKET 50-327 LER 87-056
THE COMBINED BYPASS LEAKAGE LIMIT WAS POTENTIALLY VIOLATED DUE TO AN INADEQUATE TECH SPEC AND LACK OF A SPECIFIC DESIGN CRITERIA.
EVENT DATE: 081187 REPORT DATE: 091087 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 206432) WITH UNIT 1 IN MODE 5 (0 PERCENT POWER, 3 PSIG, 130 DEGREES F) AND UNIT 2 IN MODE 5 (0 PERCENT POWER, 0 PSIG, 132 DEGREES F), THE TECH SPEC LISTING OF SECONDARY CONTAINMENT BYPASS LEAKAGE PATHS (TABLE 3.6-1) WAS DETERMINED TO BE DEFICIENT. ON AUGUST 11, 1987, THIS DEFICIENCY WAS DETERMINED TO BE POTENTIALLY REPORTABLE. THIS TABLE, WHICH SPECIFIES THE PENETRATIONS FOR CONSIDERATION IN DETERMINING THE COMBINED BYPASS LEAKAGE RATE FOR LCO 3.6.1.2.C, WAS FOUND TO BE INCOMPLETE IN THAT IT FAILED TO INCLUDE ALL OF THE POTENTIAL BYPASS LEAKAGE PATHS TO THE AUXILIARY BUILDING FOR THE CURRENT PLANT CONFIGURATION. BY USING THE INCOMPLETE LIST IN THE PAST, THE POTENTIAL EXISTED FOR EXCEEDING THE TECH SPEC ALLOWABLE LIMIT FOR COMBINED BYPASS LEAKAGE. BOTH THE UNIT 1 AND 2 TECH SPECS CONTAIN THIS DEFICIENCY. THE ERRORS IN THE TECH SPEC BYPASS LEAKAGE TABLE RESULTED FROM A LACK OF SPECIFIC DESIGN CRITERIA DEFINING SECONDARY CONTAINMENT BYPASS PENETRATIONS. WHEN A DESIGN BASIS FOR THE TECH SPEC LIST COULD NOT BE FOUND DURING A RECENT DESIGN REVIEW, A NEW BASIS WAS DEVELOPED WHICH IDENTIFIED

AN ADDITIONAL 26 PENETRATIONS WHICH MET THE DEFINITION OF BYPASS LEAKAGE PATH TO THE AUXILIARY BUILDING.

[240] SEQUOYAH 1 DOCKET 50-327 LER 87-051
SKID-MOUNTED VALVES WERE NOT CHECKED EVERY THIRTY-ONE DAYS DUE TO AN IMPROPER ADMINISTRATIVE EVALUATION.
EVENT DATE: 081287 REPORT DATE: 091087 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 206434) ON AUGUST 12, 1987, WITH UNITS 1 AND 2 IN MODE 5) 0 PERCENT POWER, 3 PSIG, 130 DEGREES F, AND 0 PERCENT POWER, 0 PSIG, 131 DEGREES F, RESPECTIVELY) DURING AN ESSENTIAL RAW COOLING WATER SYSTEM INSPECTION BY NRC, IT WAS DETERMINED THAT CERTAIN SKID-MOUNTED VALVES ASSOCIATED WITH SAFETY-RELATED EQUIPMENT WERE NOT BEING CHECKED FOR PROPER POSITION ON A 31-DAY INTERVAL AS REQUIRED BY THE TECH SPEC SURVEILLANCE REQUIREMENT (SR) 4.7.4.A. A SUBSEQUENT WALKDOWN OF SAFETY-RELATED PUMPS AND EQUIPMENT ASSOCIATED WITH THE COMPONENT COOLING WATER SYSTEM IDENTIFIED ADDITIONAL SKID-MOUNTED VALVES THAT WERE NOT CHECKED AS REQUIRED BY TECH SPEC SR 4.7.3.A. THE ROOT CAUSE OF THIS PROPLEM WAS THE ORIGINAL ADMINISTRATIVE POSITION OF CONSIDERING SKID-MOUNTED VALVES AS PART OF THE EQUIPMENT. AS A RESULT, THE VALVES WERE NOT DENOTED ON THE DRAWINGS WHICH WERE USED WHEN WRITING THE SURVEILLANCE INSTRUCTIONS AND WERE NOT CONSIDERED WITHIN THE SCOPE OF THE TS SR. SAFETY-RELATED SKID-MOUNTED EQUIPMENT IS BEING REVERIFIED TO ENSURE SKID-MOUNTED VALVES REQUIRING A 31-DAY SURVEILLANCE TO COMPLY WITH TECH SPEC 3.7.3 AND 3.7.4 ARE IDENTIFIED. SURVEILLANCE INSTRUCTIONS WILL THEN BE REVISED TO ADD THESE VALVES AND THE VALVES CHECKED FOR PROPER POSITION.

[241] SEQUOYAH 1 DOCKET 50-327 LER 87-057
DURING PERFORMANCE OF A SPECIAL TEST INSTRUCTION A CONTROL ROOM ISOLATION WAS INADVERTENTLY INITIATED DUE TO AN INDETERMINATE CAUSE.
EVENT DATE: 081587 REPORT DATE: 091587 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 206063) ON AUGUST 15, 1987, WITH UNITS 1 AND 2 IN MODE 5 (0 PERCENT POWER) 3 PSIG, 131 DEGREES F AND 0 PERCENT POWER, ATMOSPHERIC PRESSURE, 130 DEGREES F, RESPECTIVELY), A CONTROL ROOM ISOLATION OCCURRED FROM THE TRAIN A CHLORINE MONITOR WHILE SETTING UP TO PERFORM A FUNCTIONAL TEST FOR SPECIAL TEST INSTRUCTION (STI)-30, "DC HIGH VOLTAGE TEST FOR SELECTED 1E CABLE WITH POTENTIAL FOR CABLE PULLBY DAMAGE FOR CONDUIT MC 1607A." AFTER VERIFICATION THAT THE SIGNAL WAS SPURIOUS, THE CONTROL ROOM ISOLATION WAS SUBSEQUENTLY CLEARED BY THE ASSISTANT SHIFT ENGINEER BY PUSHING THE RESET BUTTON ON THE CHLORINE DETECTOR, AND THE CONTROL ROOM VENTILATION SYSTEM WAS RETURNED TO NORMAL. THE CAUSE OF THIS EVENT IS INDETERMINATE. AN ATTEMPT WAS MADE TO REPRODUCE THE CONTROL ROOM ISOLATION BY SIMULATION OF CONDITIONS AND EVENTS BEFORE THE ISOLATION AND ADDITIONAL TROUBLESHOOTING. HOWEVER, A RECURRENCE OF THE EVENT COULD NOT BE INITIATED. ALL SYSTEMS PERFORMED AS REQUIRED BY PLANT DESIGN DURING THE CONTROL ROOM ISOLATION. THIS EVENT WAS NOT CAUSED BY AN ACTUAL CHLORINE RELEASE; THUS, THERE WAS NO THREAT TO CONTROL ROOM OPERATORS. NO FURTHER CORRECTIVE ACTION IS PLANNED AS SPECIFIC FAILURES OR NONCOMPLIANCES HAVE NOT BEEN IDENTIFIED.

[242] SEQUOYAH 2 DOCKET 50-328 LER 87-007
TECHNICAL SPECIFICATION REQUIREMENT NOT MET ON A CONTAINMENT HYDROGEN ANALYZER DUE TO A COMBINATION OF DEFICIENCIES.
EVENT DATE: 081587 REPORT DATE: 091487 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 1 (PWR)
VENDOR: COMSIP DELPHI INC.

(NSIC 206456) ON AUGUST, 15 1987, WITH UNITS 1 AND 2 IN MODE 5 (COLD SHUTDOWN),

IT WAS DETERMINED THAT A POTENTIAL REPORTABLE EVENT HAD OCCURRED IN THAT, FOR UNIT 2, THE FOLLOWING CONDITIONS WERE IDENTIFIED FOR TRAIN A OF THE CONTAINMENT HYDROGEN ANALYZER SYSTEM: 1) THE SAMPLE LINE TO LOWER CONTAINMENT WAS PARTIALLY PLUGGED WITH A FOREIGN MATERIAL, 2) THE UPPER CONTAINMENT COULD NOT BE SAMPLED SINCE A SAMPLE LINE WAS NOT INSTALLED AS REQUIRED BY DESIGN DRAWINGS, AND 3) A LOW POINT IN THE FIELD ROUTED LINE RESULTING IN A 25-FOOT CONDENSATE TRAP WAS DISCOVERED WHICH COULD POTENTIALLY CAUSE THE VENDOR RECOMMENDED MAXIMUM ALLOWABLE INLET PRESSURE DROP TO BE EXCEEDED. THESE CONDITIONS WERE DISCOVERED AS A RESULT OF RECENT EFFORTS BY TVA AND THE ANALYZER MANUFACTURER TO RESOLVE CALIBRATIONAL PROBLEMS. THE CAUSE OF THE PLUGGED SAMPLE LINE (WHICH HAS BEEN UNPLUGGED) HAS NOT BEEN DETERMINED. THE EXCESSIVE CONDENSATE TRAP LENGTH WAS DUE TO INSUFFICIENT INSTALLATION INSTRUCTIONS. THE CAUSE OF THE LACK OF AN UPPER CONTAINMENT TRAIN A SAMPLE LINE IS A CONSTRUCTION INSTALLATION DEFICIENCY. UPPER CONTAINMENT SAMPLE TUBING HAS BEEN INSTALLED, AND THE EXCESSIVE TRAP LENGTH HAS BEEN REMOVED BY REROUTING OF THE SAMPLE TUBING.

[243] SHEARON HARRIS 1 DOCKET 50-400 LER 87-044
TECHNICAL SPECIFICATION SURVEILLANCE FOR AXIAL FLUX DIFFERENCE MISSED DUE TO DISTRACTION OF OPERATOR BY POWER ESCALATION.
EVENT DATE: 071387 REPORT DATE: 081287 NSSS: WE TYPE: PWR

(NSIC 205798) ON JULY 13, 1987, THE SHEARON HARRIS NUCLEAR POWER PLANT WAS OPERATING AT FULL POWER. AXIAL FLUX DIFFERENCE (AFD) MONITORING AND LOGGING WAS ONGOING IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS. THIS LOGGING WAS NECESSARY BECAUSE THE AUTOMATIC AFD COMPUTER ALARM WAS NOT OPERABLE. AT 1230, THE AFD READING WAS MISSED BECAUSE THE OPERATOR WAS DISTRACTED BY A POWER ESCALATION AFTER A 35 MW TURBINE RUNBACK. THIS ERROR WAS DISCOVERED DURING THE NEXT SHIFT'S REVIEW OF THE LOGGED DATA. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR. ACTIONS TAKEN TO PREVENT RECURRENCE WERE (1) THE OPERATOR WAS COUNSELED AND (2) REPAIRS ARE IN PROGRESS TO REPAIR THE AUTOMATIC AFD MONITOR.

[244] SHEARON HARRIS 1 DOCKET 50-400 LER 87-046
MAIN FEEDWATER RECIRCULATION FLOW CONTROL VALVE FAILED RESULTING IN A MAIN FEEDWATER PUMP 'B' TRIP ON LOW FLOW ACTUATING AUXILIARY FEEDWATER SYSTEM.
EVENT DATE: 072287 REPORT DATE: 082087 NSSS: WE TYPE: PWR
VENDOR: MASONEILAN INTERNATIONAL, INC.

(NSIC 206266) THE PLANT WAS OPERATING AT 1 PERCENT REACTOR POWER IN MODE 2, PLANT START-UP, ON JULY 22, 1987. ONLY ONE MAIN FEEDWATER PUMP, 'B', WAS IN SERVICE. 'B' MAIN FEEDWATER PUMP TRIPPED ON LOW-FLOW WHICH CAUSED THE AUTOMATIC START, AS REQUIRED, OF BOTH MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS. THE AUXILIARY FEEDWATER PUMPS STARTED AND RAN AS REQUIRED TO SUPPLY WATER TO THE STEAM GENERATORS. PLANT TEMPERATURE AND PRESSURE REMAINED STABLE AT NORMAL NO LOAD VALUES. AN OPERATOR SUBSEQUENTLY STARTED THE 'A' MAIN FEEDWATER PUMP AND SECURED THE AUXILIARY FEEDWATER PUMPS. THE CAUSE OF THE LOW FLOW WAS INVESTIGATED. THE FEEDBACK LINKAGE ON THE 'B' FEEDWATER PUMP RECIRCULATION VALVE WAS FOUND LOOSE. THIS PREVENTED THE ACTUATOR FROM RECEIVING A SIGNAL TO OPEN AND CAUSED THE LOW FLOW CONDITION FOR THE 'B' FEEDWATER PUMP. THE ROOT CAUSE OF THE FAILURE WAS VIBRATION OF THE VALVE. THE LOOSE LINKAGE ON THE FEED PUMP RECIRCULATION VALVE WAS REPAIRED. A PLANT MAINTENANCE CHECKLIST HAS BEEN DEVELOPED TO MONITOR THE RECIRCULATION VALVE FEEDBACK LINKAGES MONTHLY TO PREVENT A RECURRENCE OF THIS FAILURE. A PLANT CHANGE REQUEST WAS ALSO GENERATED TO INVESTIGATE AND RESOLVE THE HIGH VIBRATION EXPERIENCED BY THE RECIRCULATION VALVE. THE EVENT IS REPORTABLE PER 10CFR50.73 (A)(2)9IV) AS AN AUTOMATIC ACTUATION OF AN ENGINEERED SAFEGUARDS FEATURE.

[245] SHEARON HARRIS 1 DOCKET 50-400 LER 87-041
PLANT TRIP DUE TO THE LOSS OF INSTRUMENT AIR CAUSED BY AN IMPROPERLY PREPARED CLEARANCE VALVE RESTORATION LINEUP.
EVENT DATE: 080487 REPORT DATE: 090387 NSSS: WE TYPE: PWR

(NSIC 206265) THE PLANT WAS OPERATING AT 100 PERCENT REACTOR POWER IN MODE 1, POWER OPERATIONL ON AUG, 1987. INSTRUMENT AIR DRVER 1A WAS OUT OF SERVICE FOR REPAIRS. INSTRUMENT AIR DRYER AS BYPASSED AT APPROXIMATELY 1715 HOURS TO REPLACE THE DESICCANT. WORK WAS COMPLETED THE DRYER 1B AND AT APPROXIMATELY 2150 HOURS THE CLEARANCE ON THE DRYER WAS REMOVED. THE RESTORATION ALIGNMENT ON THE CLEARANCE FOR PLACING DRYER 1B BACK INTO SERVICE WAS INCORRECT. WHEN THE VALVES WERE REPOSITIONED IN ACCORDANCE WITH THE CLEARANCE RESTORATION LINEUP THE INSTRUMENT AIR COMPRESSORS WERE ISOLATED FROM THE AIR SYSTEM. THE AIR ISOLATION CAUSED AIR PRESSURE TO DECAY AND CAUSED AIR OPERATED VALVES TO GO TO "FAIL SAFE" POSITIONS. IN PARTICULAR, THE MAIN FEEDWATER REGULATING VALVES TO DRIFT SHUT AND THE HEATER DRAIN LEVEL CONTROL VALVES TO DIVERT DRAIN FLOW TO THE CONDENSER. HEATER DRAIN PUMPS 1A AND 1B TRIPPED AND A MANUAL TURBINE RUNBACK WAS INITIATED. THIS WAS FOLLOWED BY A TRIP OF MAIN FEEDWATER PUMP 1A WHICH INITIATED AN AUTOMATIC TURBINE RUNBACK. THE LOSS OF INSTRUMENT AIR RESULTED IN A DECREASE IN STEAM GENERATOR WATER LEVELS. THE TURBINE RUNBACK RESULTED IN SHRINK IN STEAM GENERATOR LEVELS AND RESULTED IN A REACTOR TRIP AT 2154 HOURS DUE TO STEAM GENERATOR FEEDWATER STEAM/FLOW MISMATCH WITH LOW STEAM GENERATOR WATER LEVELS.

[246] SHEARON HARRIS 1 DOCKET 50-400 LER 87-047
AUXILIARY FEEDWATER ACTUATION DUE TO MAIN FEEDWATER PUMP 1A TRIPS CAUSED BY DISCHARGE PRESSURE SWITCHES BEING OUT OF CALIBRATION.
EVENT DATE: 080587 REPORT DATE: 090487 NSSS: WE TYPE: PWR
VENDOR: MERCOID CORP.

(NSIC 206267) THE PLANT WAS IN MODE 3, HOT STANDBY, AT 0 PERCENT REACTOR POWER ON AUGUST 5, 1987. DURING THE PERIOD OF AUGUST 4, 1987 AT 2344 HOURS AND AUGUST 5, 1987 AT 0212 HOURS, THERE WERE FOUR AUXILIARY FEEDWATER (AFW) ACTUATIONS DUE TO TRIPS OF MAIN FEEDWATER (MFW) PUMP 1A ON HIGH DISCHARGE PRESSURE. A WORK TICKET WAS ISSUED FOR INCTRUMENT AND CONTROL (I&C) TECHNICIANS TO CHECK THE SETPOINTS OF THE PRESSURE SWITCHES ON THE DISCHARGE OF MFW PUMP 1A. THREE OUT OF THE FOUR PRESSURE SWITCHES WERE FOUND TO BE OUT OF CALIBRATION; THEY HAD DRIFTED TO A LOWER SETPOINT. THE FOURTH PRESSURE SWITCH WAS WITHIN ACCEPTABLE TOLERANCE. THE PRESSURE SWITCHES WERE RECALIBRATED AND MFW PUMP LA STARTED AT 1328 HOURS AND RAN NORMALLY. ACTION TO PREVENT RECURRENCE IS THAT DESIGN CHANGES ARE BEING PREPARED FOR THE FEEDWATER AND CONDENSATE SYSTEMS WHICH WILL REDUCE THE DISCHARGE PRESSURE OF THE MAIN FEEDWATER PUMPS AND PROVIDE MARGIN BETWEEN OPERATING CONDITIONS AND EQUIPMENT TRIP SETPOINTS. THE ACTUATION LOGIC FOR THE AUXILIARY FEEDWATER SYSTEM WILL BE REVIEWED TO DETERMINE IF THE ACTUATION ON A MAIN FEEDWATER PUMP TRIP CAN BE REDUCED. THIS EVENT IS REPORTABLE UNDER 10CFR50.73(A)(2)(IV) AS AN ACTUATION OF AN ENGINEERED SAFETY FEATURE (AUXILIARY FEEDWATER SYSTEM).

[247] SHEARON HARRIS 1 DOCKET 50-400 LER 87-048
TECHNICAL SPECIFICATION SURVEILLANCE FOR AXIAL FLUX DIFFERENCE MISSED DUE TO PERSONNEL ERROR.
EVENT DATE: 081187 REPORT DATE: 091087 NSSS: WE TYPE: PWR

(NSIC 206080) ON AUGUST 11, 1987, THE SHEARON HARRIS NUCLEAR POWER PLANT WAS OPERATING AT FULL POWER. AXIAL FLUX DIFFERENCE (AFD) MONITORING AND LOGGING WAS ONGOING IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS. THIS LOGGING WAS NECESSARY BECAUSE THE AUTOMATIC AFD COMPUTER ALARM WAS NOT OPERABLE. AT 0930, THE AFD READING WAS MISSED BECAUSE THE OPERATOR WAS DISTRACTED WHILE CONCENTRATING HIS ATTENTION ON THE COMPLETION OF THE 0900 DAILY SURVEILLANCE REQUIREMENTS AND CONTROL OPERATOR'S LOGS. THIS ERROR WAS DISCOVERED DURING THE NEXT SHIFT'S REVIEW OF THE LOGGED DATA. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR. ACTIONS

TAKEN TO PREVENT RECURRENCE WERE: (1) THE OPERATOR WAS DISCIPLINED, (2) REPAIRS ARE IN PROGRESS FOR THE AUTOMATIC AFD MONITOR, AND (3) ADDITIONAL GUIDANCE IS BEING PROVIDED TO THE OPERATORS.

[248] SHEARON HARRIS 1 DOCKET 50-400 LER 87-043
SIX OUTSIDE CONTAINMENT ISOLATION VALVES WERE OMITTED FROM MONTHLY POSITION CHECK DUE TO PERSONNEL ERROR IN PROCEDURE DEVELOPMENT.
EVENT DATE: 082087 REPORT DATE: 092187 NSSS: WE TYPE: PWR

(NSIC 206463) THE PLANT WAS OPERATING AT 100 PERCENT REACTOR POWER IN MODE 1, POWER OPERATION, ON AUGUST 20, 1987. DURING A REVIEW OF CONTAINMENT ISOLATION VALVES, IT WAS DISCOVERED AT 1400 HOURS THAT SIX MANUAL ISOLATION VALVES WERE NOT INCLUDED IN OPERATION SURVEILLANCE TEST (OST)-1029, CONTAINMENT PENETRATION OUTSIDE ISOLATION VALVE VERIFICATION, AS REQUIRED. OST-1029 SATISFIES TECHNICAL SPECIFICATIONS ITEM 4.6.1.1.A FOR A 31-DAY VERIFICATION THAT THE VALVES ARE CLOSED AND LOCKED. UPON DISCOVERY, A CHECK OF THE SIX ISOLATION VALVES WAS MADE AND IT WAS FOUND THAT ALL SIX VALVES WERE CLOSED AND LOCKED AS REQUIRED. THERE IS NO INDICATION THAT ANY OF THE VALVES HAVE BEEN OPENED SINCE THEY WERE PLACED IN THEIR PROPER CLOSED POSITION PRIOR TO INITIAL CRITICALITY OF THE PLANT. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR IN OMITTING THE SIX ISOLATION VALVES FROM OST-1029 FOR VERIFICATION OF THEIR LOCKED CLOSED POSITION AS REQUIRED. THE IMMEDIATE CORRECTIVE ACTION UPON DISCOVERY WAS TO VERIFY THAT THE SIX VALVES WERE IN THE CLOSED AND LOCKED POSITION AS REQUIRED, WHICH THEY WERE. ACTION TO PREVENT RECURRENCE IS THAT OST-1029 HAS BEEN REVISED TO INCLUDE THESE SIX VALVES IN THE MONTHLY VERIFICATION OF POSITION. THIS EVENT IS BEING REPORTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(I)(B) AS A VIOLATION OF TECHNICAL SPECIFICATIONS ACTION ITEM 4.6.1.1.A.

[249] SHOREHAM DOCKET 50-322 LER 87-026
ESF ACTUATIONS RESULTED FROM LOSS OF POWER TO THE NSST DUE TO TECHNICIAN ERROR DURING RELAY TRIP TESTING.
EVENT DATE: 080787 REPORT DATE: 090487 NSSS: GE TYPE: BWR

(NSIC 206303) ON AUGUST 7, 1987 AT 1053, A LOSS OF THE NORMAL STATION SERVICE TRANSFORMER (NSST) OCCURRED DUE TO PERSONNEL ERROR WHILE PERFORMING A RELAY CALIBRATION. THE PLANT WAS IN OPERATIONAL CONDITION 4 WITH THE MODE SWITCH IN SHUTDOWN AND ALL RODS INSERTED IN THE CORE. THE TECHNICIAN WAS PERFORMING A PREVENTIVE MAINTENANCE (PM) ACTIVITY TO CALIBRATE A RELAY IN THE NSST BACKUP PROTECTION SCHEME. HE INCORRECTLY ASSUMED THAT A RELAY THAT WAS SUPPOSED TO BE REMOVED PRIOR TO PERFORMING THE CAL., WAS ALREADY TAKEN OUT DURING A PREVIOUS PM. HE THEN ENERGIZED THE CIRCUIT SIMULATING A MAIN TRANSFORMER BREAKER FAILURE. THIS ENERGIZED THE NSST BACK-UP PROTECTION CIRCUIT, CAUSING LOSS OF THE NSST. AS A RESULT OF THIS RELAY ACTUATION, A FAST TRANSFER OF THE EMERGENCY BUSES WAS BLOCKED. THE SLOW TRANSFER TO THE RSST WAS INITIATED CAUSING NUMEROUS ESFS TO INITIATE AND THE TWO OPERABLE EDGS (102 AND 103) TO START BUT NOT LOAD. THE REACTOR BUILDING STANDBY VENTILATION SYSTEM (RBSVS)/CONTROL ROOM AIR CONDITIONING (CRAC) SYSTEM WATER CHILLERS SUBSEQUENTLY BECAME INOPERABLE DUE TO AN ELECTRICAL PROBLEM WITH THE CONDENSING WATER TEMPERATURE CONTROL VALVES. INVESTIGATION INTO THE CAUSE IS CONTINUING. OPERATORS TOOK IMMEDIATE ACTIONS TO RESTORE THE SYSTEMS TO THEIR ORIGINAL CONFIGURATION PRIOR TO THE EVENT.

[250] SHOREHAM DOCKET 50-322 LER 87-027
INCORRECT COMPOSITING OF CONTINUOUS LIQUID RELEASES REQUIRED BY TECH SPECS DUE TO PROCEDURAL ERROR.
EVENT DATE: 082187 REPORT DATE: 091187 NSSS: GE TYPE: BWR

(NSIC 206318) ON AUGUST 21, 1987 AT 0800, IT WAS DETERMINED BY PLANT MANAGEMENT THAT THE PROCEDURE USED FOR WEEKLY, MONTHLY AND QUARTERLY COMPOSITE SAMPLES OF

CONTINUOUS LIQUID RELEASES DID NOT ADEQUATELY MEET THE REQUIREMENTS FOR REPRESENTATIVE SAMPLING AS REQUIRED BY THE TECH SPECS. THE PLANT WAS IN OPERATIONAL CONDITION 4 (COLD SHUTDOWN) WITH THE MODE SWITCH IN SHUTDOWN AND ALL RODS INSERTED IN THE CORE. THE PROCEDURE SP 74.20.010 (CONTINUOUS LIQUID RELEASE - GENERAL SAMPLING, ANALYSIS AND DATA STORAGE) WAS BEING UTILIZED FROM DECEMBER 7, 1984 UNTIL SEPTEMBER 29, 1986 TO SATISFY TECH SPEC 3/4.11.1.1. THIS PROCEDURE WAS REVISED SEPT. 30, 1986 TO MEET THE REQUIREMENTS. THE PROCEDURE WAS DEFICIENT IN THAT IT ALLOWED THE LIQUID EFFLUENT SAMPLES TO BE COMPOSITED INCORRECTLY FOR THE WEEKLY, MONTHLY AND QUARTERLY RESULTS. ALTHOUGH THE DAILY COMPOSITES WERE CORRECT, THE TECH SPEC REQUIRED COMPOSITES WERE NOT AN ADEQUATE REPRESENTATION OF WHAT WAS BEING RELEASED IN RELATION TO THE SYSTEMS' FLOW. UPON DETERMINATION OF THE PROBLEM, THE PROCEDURE WAS REVISED. INITIALLY, IT WAS DETERMINED NOT TO BE REPORTABLE AND ONLY A PROCEDURAL DEFICIENCY. HOWEVER, FURTHER EVALUATION BY PLANT MANAGEMENT DETERMINED THAT THE STEPS IN THE PROCEDURE TO PREPARE COMPOSITE LIQUID EFFLUENT SAMPLES DID NOT ADEQUATELY MEET THE REQUIREMENTS OF THE TECH SPECS.

[251] SHOREHAM DOCKET 50-322 LER 87-028 REV 01
UPDATE ON METEOROLOGICAL WIND DIRECTION MONITORING INSTRUMENTATION INOPERABLE DUE TO A WORN CABLE AND CONNECTOR TO SENSOR.
EVENT DATE: 082987 REPORT DATE: 091487 NSSS: GE TYPE: BWR

(NSIC 206325) THIS REVISION PROVIDES ADDITIONAL NOTIFICATION ON THE INOPERABILITY OF MONITORING INSTRUMENTATION AT THE 400 FT OFFSITE METEOROLOGICAL TOWER. THIS REVISION IS SUBMITTED IN ACCORDANCE WITH SHOREHAM TECH SPECS 3.3.7.3 AND 6.9.2. AS ORIGINALLY REPORTED IN LER 87-028 REVISION 00, THE AIR TEMPERATURE MONITORING INSTRUMENTATION AT THE 400 FT OFFSITE METEOROLOGICAL TOWER WAS DECLARED INOPERABLE 8/27/87. SUBSEQUENT TO 8/27/87, FOR TWO PERIODS BETWEEN 8/29/87 AND 8/30/87, THE WIND DIRECTION MONITORING INSTRUMENTATION AT THE TOWER WAS NOT OPERATING PROPERLY. THIS WAS BROUGHT TO THE ATTENTION OF THE CONTROL ROOM AND SUBSEQUENTLY THE WIND DIRECTION MONITORING SYSTEM WAS DECLARED INOPERABLE. ALTHOUGH THE SYSTEM WAS OPERATING PROPERLY AFTER 8/30/87, IT WAS DECIDED NOT TO DECLARE THE SYSTEM OPERABLE UNTIL THE PROBLEM WAS CORRECTED. THE CAUSE OF THE PROBLEM HAS BEEN IDENTIFIED AS BEING A WORN CABLE AND CONNECTOR TO THE WIND DIRECTION SENSOR AT THE 150 FT ELEVATION OF THE TOWER. THE CABLE AND CONNECTORS TO THE INSTRUMENT, AS WELL AS THE INSTRUMENT'S CROSS-ARM, ARE SCHEDULED TO BE REPLACED SOMETIME DURING THE WEEK OF SEPTEMBER 21, 1987, DEPENDING ON WEATHER CONDITIONS AND THE AVAILABILITY OF THE REPLACEMENT PARTS.

[252] SOUTH TEXAS 1 DOCKET 50-498 LER 87-001
LACK OF SYSTEM KNOWLEDGE RESULTS IN INOPERABLE UNIT VENT RADIATION MONITORS.
EVENT DATE: 082487 REPORT DATE: 092387 NSSS: WE TYPE: PWR
VENDOR: GENERAL ATOMIC CO.

(NSIC 206470) AT 2040 HOURS ON AUGUST 24, 1987, A CONTROL ROOM OPERATOR DISCOVERED THAT BOTH UNIT VENT RADIATION MONITORS HAD LOST SAMPLE FLOW AND WERE INOPERABLE. THE OPERATORS SUBSEQUENTLY ESTABLISHED THAT THE MONITORS' SAMPLE PUMPS HAD BEEN SHUT DOWN AT APPROXIMATELY 1500 HOURS ON THAT DAY. THE SHUTDOWN WAS APPARENTLY CAUSED BY PERSONNEL ERROR AT A RADIATION MONITOR COMPUTER CONSOLE IN THE HEALTH PHYSICS OFFICE, WHEN THE PUMPS WERE APPARENTLY STOPPED BY AN INAPPROPRIATE RESPONSE TO A LOSS OF PROCESS FLOW ALARM FROM THE MONITORS. THE RESULTING LOSS OF SAMPLE FLOW ALARM WAS NOT NOTICED BY OPERATORS AMONG NUMEROUS AND FREQUENT LOSS OF PROCESS FLOW ALARMS CAUSED BY MAINTENANCE ACTIVITIES BY TECHNICIANS ON THE UNIT VENT MONITORS. CONTINUOUS AIR MONITORS WERE FUNCTIONING DURING THE EVENT AND, AS SUCH, NO TECHNICAL SPECIFICATIONS WERE VIOLATED. SUPPLEMENTARY TRAINING HAS BEEN PERFORMED FOR OPERATORS AND PLANT PERSONNEL ON SYSTEM OPERATION. SYSTEM MONITOR COMPUTER CONSOLE GRAPHICS HAVE BEEN IMPROVED AND ADDITIONAL CAUTIONS ARE BEING ADDED TO IMPROVE PLANT PERSONNEL AWARENESS OF MONITORS FOR WHICH TECHNICAL SPECIFICATIONS APPLY.

[253] SOUTH TEXAS 1 DOCKET 50-498 LER 87-002
LOSS OF SAMPLE FLOW TO CONTAINMENT PURGE RADIATION MONITOR CAUSES CONTAINMENT VENTILATION ISOLATION.
EVENT DATE: 082687 REPORT DATE: 092587 NSSS: WE TYPE: PWR

(NSIC 206471) ON AUGUST 26, 1987 AT 17:10 HOURS DURING INITIAL CORE LOAD, A CONTAINMENT VENTILATION ISOLATION OCCURRED AS A RESULT OF LOSS OF SAMPLE FLOW TO A REACTOR CONTAINMENT BUILDING PURGE RADIATION MONITOR. THE SUPPLEMENTARY PURGE SYSTEM WHICH WAS IN OPERATION AT THE TIME WAS ISOLATED AS DESIGNED. SUBSEQUENT INVESTIGATION OF THE CAUSE OF THE LOSS OF SAMPLE FLOW WAS NOT CONCLUSIVE. REVIEW OF OPERATOR ACTIONS IN THE CONTROL ROOM, ALARM LOGS, AND ATTEMPTED REPRODUCTION OF THE EVENT BY SIMULATED VIBRATION OF THE PUMP CONTROL RELAY WHICH COULD HAVE CAUSED THE MONITOR TO LOSE SAMPLE FLOW REVEALED THAT THE MOST PROBABLE ROOT CAUSE WAS UNAUTHORIZED LOCAL OPERATION OF THE MONITOR'S SKID-MOUNTED CONTROLS, POWER, OR VALVES OR OPERATION FROM THE CONTROL ROOM PANEL. CORRECTIVE ACTIONS TO PREVENT RECURRENCE INCLUDE CONSIDERATION OF ADDED SKID COMPONENT PROTECTION DEVICES, ADDITION OF CONTROL ROOM RADIATION MONITOR CONSOLE PROTECTIVE COVERS, ADDITIONAL TRAINING FOR PLANT PERSONNEL, AND CAUTION SIGNS AT THE MONITOR SKIDS.

[254] SOUTH TEXAS 1 DOCKET 50-498 LER 87-005
CONTROL ROOM VENTILATION AUTO-ACTUATION TO RECIRCULATION MODE DUE TO PERSONNEL ERROR AND INCORRECT OPERATOR RESPONSE.
EVENT DATE: 090487 REPORT DATE: 100287 NSSS: WE TYPE: PWR

(NSIC 206507) AT APPROXIMATELY 1500 HOURS ON SEPTEMBER 4, 1987 WITH THE UNIT 1 IN MODE 5, AN EMPLOYEE ACCIDENTLY TRIPPED AN AC BREAKER ON A NON-CLASS 1E VOLTAGE REGULATING TRANSFORMER, CAUSING THE LOSS OF BACKUP POWER TO A NON-CLASS 1E INVERTER. INCORRECT OPERATOR RESPONSE TO THE LOSS OF BACKUP POWER THEN RESULTED IN THE LOSS OF CONTROL POWER TO THE CONTROL ROOM TOXIC GAS MONITOR ACTUATION RELAYS. THIS CAUSED AN AUTO-ACTUATION OF THE CONTROL ROOM VENTILATION SYSTEM TO THE RECIRCULATION MODE. THE ROOT CAUSE OF THIS EVENT IS THE LACK OF OPERATOR KNOWLEDGE IN RESPONDING TO THE LOCAL ALARM ON THE INVERTER. THE CORRECTIVE ACTIONS WILL INCLUDE TRAINING OF OPERATORS ON THE PROPER OPERATION OF INVERTER SYSTEMS AND INSTALLING PROTECTIVE GUARDS ON THE BREAKERS IN HIGH TRAFFIC AREAS.

[255] SOUTH TEXAS 1 DOCKET 50-498 LER 87-006
CONTROL ROOM VENTILATION ACTUATION TO RECIRCULATION MODE DUE TO CONTROL ROOM VENTILATION RADIATION MONITOR LOSS OF FLOW AS A RESULT OF PERSONNEL ERROR.
EVENT DATE: 090587 REPORT DATE: 100287 NSSS: WE TYPE: PWR

(NSIC 206604) AT 1210 HOURS ON SEPTEMBER 5, 1987 WITH THE PLANT IN MODE 5, WHILE CHANGING A FILTER, A HEALTH PHYSICS (HP) TECHNICIAN ISOLATED THE SAMPLE FLOW TO THE CONTROL ROOM VENTILATION INTAKE RADIATION MONITOR CAUSING CONTROL ROOM VENTILATION TO AUTOMATICALLY SHIFT TO THE RECIRCULATION MODE. THE CONTROL ROOM VENTILATION ACTUATION IS AN ENGINEERED SAFETY FEATURE. THE CONTROL ROOM VERIFIED THE AUTO-RECIRCULATION, AND AFTER HP RESTORED THE MONITOR SAMPLE FLOW, THE CONTROL ROOM REALIGNED THE CONTROL ROOM VENTILATION TO ITS NORMAL CONFIGURATION. THE ROOT CAUSE OF THE OCCURRENCE WAS THAT THE HP TECHNICIAN CHANGING THE FILTER USED THE INCORRECT PROCEDURE. ADDITIONALLY THE TECHNICIAN WAS NOT FAMILIAR ENOUGH WITH THE RADIATION MONITORING SYSTEM TO RECOGNIZE THAT THE INCORRECT PROCEDURE WAS BEING USED. TO PREVENT RECURRENCE THE HP TECHNICIANS HAVE BEEN TRAINED ON THE RADIATION MONITORING SYSTEM PROCEDURES.

[256] ST. LUCIE 2 DOCKET 50-389 LER 86-014
SURVEILLANCE INTERVAL FOR HOT SHUTDOWN CONTROL PANEL INSTRUMENTATION EXCEEDED DUE TO PERSONNEL ERROR.
EVENT DATE: 100886 REPORT DATE: 110786 NSSS: CE TYPE: PWR

(NSIC 206443) ON OCTOBER 8, 1986, IT WAS DISCOVERED THAT THE TIME LIMIT FOR THE PERFORMANCE OF THE EIGHTEEN MONTH HOT SHUTDOWN CONTROL PANEL (HSCP) EMERGENCY DIESEL GENERATOR (EDG) WATTMETER CALIBRATION SURVEILLANCE HAD BEEN EXCEEDED. THE SURVEILLANCE WAS TO BE PERFORMED NO LATER THAN SEPTEMBER 6, 1986 AND SEPTEMBER 19, 1986 FOR THE 2A AND 2B HSCP EDG WATTMETERS, RESPECTIVELY. THE ROOT CAUSE OF THE EVENT WAS A COGNITIVE PERSONNEL ERROR BY UTILITY MAINTENANCE PERSONNEL WHO FAILED TO ADEQUATELY MONITOR THE REQUIRED SURVEILLANCE INTERVAL. FOR CORRECTIVE ACTIONS, THE MAINTENANCE PERSONNEL HAVE CALIBRATED BOTH NSCP EDG WATTMETERS AND ARE IMPLEMENTING A DEPARTMENTAL SURVEILLANCE/TESTING SCHEDULE PROCEDURE. PLANT MANAGEMENT HAS REEMPHASIZED THE IMPORTANCE OF COMPLETING ALL TECH SPEC SURVEILLANCES IN A TIMELY MANNER.

[257] SUMMER 1 DOCKET 50-395 LER 86-018 REV 01
UPDATE ON PERSONNEL OVEREXPOSURE.
EVENT DATE: 110786 REPORT DATE: 010687 NSSS: WE TYPE: PWR

(NSIC 206444) ON NOVEMBER 7, 1986, RADIOACTIVE CONTAMINATION WAS DETECTED ON A VERY SMALL AREA OF THE BACK OF THE RIGHT HAND OF AN INDIVIDUAL EXITING THE RADIATION CONTROL AREA. IMMEDIATE LICENSEE ACTIONS INCLUDED DECONTAMINATION OF THE INDIVIDUAL AND FOLLOWUP SURVEYS IN THE AREAS IN WHICH HE WORKED. SKIN DOSE CALCULATIONS WERE PERFORMED ASSUMING THE SOURCE TO BE COBALT 60. UTILIZING THE METHOD SUGGESTED BY IEN 86-23, A DOSE EXCEEDING 75 RAD WAS CALCULATED. RECOGNIZING THE HEALTH AND WELL BEING OF THE INDIVIDUAL AS BEING FIRST AND FOREMOST, AVAILABLE INFORMATION WAS COMPILED AND FORWARDED TO SEVERAL RECOGNIZED EXPERTS IN THE FIELD OF POINT SOURCE CONTAMINATION AND RESULTING BIOLOGICAL EFFECTS. IN ORDER TO PROVIDE A TIMELY REPORT, A CONSERVATIVE DOSE EQUIVALENT OF 42.8 REM TO THE EXTREMITY WAS SUBSEQUENTLY REPORTED. THE ONGOING STUDY AND INVESTIGATION OF THE EVENT HAVE DETERMINED A DOSE EQUIVALENT WHICH MORE PROPERY REFLECTS POTENTIAL BIOLOGICAL EFFECTS OF 0.43 REM. THE ORIGINAL CALCULATION WAS BASED UPON VERY CONSERVATIVE METHODOLOGY AND ASSUMPTIONS ABOUT THE EXPOSURE. THE REVISED CALCULATION REFLECTS THE INTENT OF 10CFR20.4(C). THE HEALTH OF THE INDIVIDUAL AND THE SAFETY OF THE PLANT WERE UNAFFECTED BY THIS INCIDENT.

[258] SUMMER 1 DOCKET 50-395 LER 87-018
FIRE PROTECTION SURVEILLANCE NONCOMPLIANCE.
EVENT DATE: 072987 REPORT DATE: 082787 NSSS: WE TYPE: PWR

(NSIC 205871) AT 1910 HOURS, JULY 29, 1987, THE INTEGRATED FIRE AND SECURITY (IF&S) SYSTEM FAILED DUE TO SEVERE ELECTRICAL STORMS. THE SYSTEM WAS RESTORED AT 0200 HOURS, JULY 30 WITH THE EXCEPTION OF ONE PANEL (LOOP REMOTE LR 9) WHICH WAS IN BYPASS. LR9 WAS SUBSEQUENTLY DECLARED OPERABLE AT 1830 HOURS, JULY 30. AT 1230 HOURS, JULY 30, IT WAS IDENTIFIED THAT CONTAINMENT TEMPERATURE HAD NOT BEEN MONITORED BETWEEN THE HOURS OF 1910, JULY 29 TO 0200, JULY 30 AND THAT FIRE AREAS MONITORED BY LR9 HAD NOT BEEN PATROLLED FROM 0200 UNTIL 1230 HOURS, JULY 30. THE CONTROL ROOM SUPERVISOR FAILED TO PROPERLY DOCUMENT THE FAILURE OF THE IF&S SYSTEM AND FAILED TO INITIATE THE MONITORING OF THE CONTAINMENT AIR TEMPERATURE AS A REQUIRED BY TECHNICAL SPECIFICATIONS. WHEN THE SYSTEM WAS RESTORED AT 0200 HOURS, WITH THE EXCEPTION OF LR9, CONTINUATION OF COMPENSATORY ACTION FOR THE AREAS MONITORED BY LR9 WAS NOT VERIFIED DUE TO LACK OF PROCEDURAL GUIDANCE. THE FOLLOWING CORRECTIVE ACTIONS WILL BE TAKEN AS A RESULT OF THIS EVENT: 1. STATION ADMINISTRATIVE PROCEDURE (SAP-131), "FIRE PROTECTION PROGRAM PLAN," WILL BE REVISED TO ADDRESS ALL ASPECTS OF THE FIRE PROTECTION PROGRAM. 2. A FIRE PROTECTION LEADER WILL BE ASSIGNED TO EACH SHIFT AND WILL BE THE "SINGLE POINT CONTACT" FOR ALL FIRE PROTECTION RELATED ACTIVITIES. 3. THE DUTY SHIFT SUPERVISOR ATTENDED THE OPERATION AND WILL BRIEF EACH SHIFT.

[259] SURRY 1 DOCKET 50-280 LER 87-002
CONTROL/RELAY ROOM CHILLER TRIPPED DUE TO INADEQUATE SERVICE WATER FLOW.
EVENT DATE: 010487 REPORT DATE: 091787 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)

(NSIC 206044) ON JANUARY 4, 1987 AT 0610 HOURS, ONE OF THE THREE CONTROL ROOM/RELAY ROOM CHILLERS (1-VS-E-4C) (EIIS-CHU) TRIPPED ON HIGH CONDENSER DISCHARGE PRESSURE. THE 'A' CHILLER UNIT HAD PREVIOUSLY BEEN REMOVED FROM SERVICE FOR MAINTENANCE. THIS IS CONTRARY TO TECHNICAL SPECIFICATION 3.14 WHICH REQUIRES ONE CONTROL/RELAY CHILLER TO BE OPERATING AND ANOTHER TO BE OPERABLE. THE 'C' CHILLER WAS RETURNED TO SERVICE AT 0730 HOURS. AT THE TIME OF THESE EVENTS, BOTH UNITS 1 AND 2 WERE AT COLD SHUTDOWN. THE CAUSE OF THE EVENT WAS INSUFFICIENT SERVICE WATER (SW) (EIIS-BI) FLOW TO THE CONDENSER OF THE CONTROL ROOM/RELAY ROOM CHILLER UNIT. IT IS BELIEVED THAT THE MANUAL SERVICE WATER DISCHARGE ISOLATION VALVE WAS PARTIALLY OBSTRUCTED. THE VALVE WAS CYCLED SEVERAL TIMES, COOLING FLOW WAS REESTABLISHED AND THE CHILLER WAS RETURNED TO SERVICE AT 0730 HOURS. SERVICE WATER PRESSURE TO THE CONTROL/RELAY ROOM CHILLERS IS CHECKED EVERY FOUR HOURS; AND THE SW STRAINERS ARE CLEANED WHENEVER SW PRESSURE BECOMES LOW. THE CONTROL/RELAY ROOM VENTILATION SYSTEM IS BEING EVALUATED BY ENGINEERING, AND PLANS ARE BEING DEVELOPED TO UPGRADE THE SYSTEM. THE ROTATING SW STRAINER HAS BEEN REPLACED.

[260] SURRY 1 DOCKET 50-280 LER 87-009
CONTAINMENT ISOLATION VALVE INOPERABLE DUE TO MECHANICAL BINDING.
EVENT DATE: 033087 REPORT DATE: 042987 NSSS: WE TYPE: PWR
VENDOR: CROSBY VALVE

(NSIC 206572) ON MARCH 30, 1987 AT 0250 HOURS, WITH UNIT 1 AT 100% POWER, OPERATIONS PERSONNEL PERFORMING A UNIT 1 CONTAINMENT SUMP IN-LEAKAGE TEST DISCOVERED THAT THE INSIDE CONTAINMENT ISOLATION VALVE (1-DA-TV-100A) (EIIS-ISV) FOR THE CONTAINMENT SUMP PUMPS (EIIS-P) WOULD NOT CLOSE. AT 0545 HOURS, THE MANUAL ISOLATION VALVES FOR THE TRIP VALVE WERE CLOSED AND CONTAINMENT INTEGRITY WAS SATISIFIED. AT ALL TIMES DURING THIS EVENT, THE OUTSIDE CONTAINMENT TRIP VALVE FOR THE CONTAINMENT SUMPS REMAINED OPERABLE. OPERATIONS PERSONNEL DETERMINED THAT THE VALVE IS MECHANICALLY BOUND IN THE OPEN POSITION. THE VALVE WILL BE DISASSEMBLED DURING THE NEXT OUTAGE OF SUFFICIENT DURATION TO DETERMINE THE FAILURE MECHANISM AND TO FACILITATE REPAIRS. UNTIL THAT TIME, THE MANUAL ISOLATION VALVES WILL REMAIN CLOSED TO ENSURE CONTAINMENT INTEGRITY.

[261] SURRY 1 DOCKET 50-280 LER 87-011 REV 01
UPDATE ON REACTOR TRIP ON LOW RCS FLOW DUE TO FAILURE OF LOOP STOP VALVE.
EVENT DATE: 051687 REPORT DATE: 101287 NSSS: WE TYPE: PWR
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 206521) ON MAY 16, 1987 AT 0824 HOURS, WITH UNIT 1 AT 100% POWER, A LOW FLOW REACTOR TRIP OCCURRED WHEN "A" LOOP REACTOR COOLANT SYSTEM (RCS) (EIIS-AB) FLOW DECREASED TO 47%. FOLLOWING THE REACTOR TRIP, THE SOURCE RANGE CHANNELS (EIIS-DET) DID NOT AUTOMATICALLY REINSTATE. ALL OTHER PROTECTION AND CONTROL SYSTEMS FUNCTIONED PROPERLY. OPERATORS FOLLOWED APPROPRIATE PLANT PROCEDURES AND STABILIZED THE PLANT FOLLOWING THE REACTOR TRIP. THIS EVENT OCCURRED WHEN THE "A" HOT LEG LOOP STOP VALVE (EIIS-ISV) STEM FAILED, PERMITTING THE DISC TO DROP, PARTIALLY BLOCKING LOOP FLOW. A METALLURGICAL ANALYSIS WAS PERFORMED WHICH DETERMINED THAT THE STEM FAILURE WAS ATTRIBUTED TO STRESS CORROSION CRACKING. THE STRESS CORROSION CRACKING WAS DUE TO A COMBINATION OF EXCESSIVE BACKSEATING FORCE AND A PROCESS OF THERMAL AGING. AT THE NEXT OUTAGES OF SUFFICIENT DURATION, SAMPLES FROM THE UNIT 1 AND UNIT 2 STEMS WILL UNDERGO METALLURGICAL ANALYSIS, AND THE UNIT 2 STEMS WILL UNDERGO ULTRASONIC TESTING AS WAS PREVIOUSLY PERFORMED ON UNIT 1. THE FAILURE OF SOURCE RANGE CHANNELS TO REINSTATE WAS DUE TO THE UNDER COMPENSATION OF THE INTERMEDIATE RANGE CHANNEL NI-36. THE SOURCE

RANGE CHANNELS WERE MANUALLY REINSTATED, AND TECHNICIANS READJUSTED THE INTERMEDIATE RANGE COMPENSATING VOLTAGE.

[262] SURRY 1 DOCKET 50-280 LER 87-018
CONTROL/RELAY ROOM CHILLERS INOPERABLE DUE TO INADEQUATE SERVICE WATER FLOW.
EVENT DATE: 072987 REPORT DATE: 082887 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)

(NSIC 206241) ON JULY 29, 1987 AT 2155 HOURS, THE FREON RELIEF VALVE LIFTED ON THE 'A' CONTROL/RELAY ROOM CHILLER (EIIS-CHU), AND THE CHILLER WAS DECLARED INOPERABLE. THE 'C' CHILLER HAD PREVIOUSLY BEEN DECLARED INOPERABLE FOR THE SAME REASON, LEAVING ONLY 'B' CHILLER OPERATING. THIS IS CONTRARY TO TECHNICAL SPECIFICATION 3.14 WHICH REQUIRES ONE CONTROL/RELAY ROOM CHILLER TO BE OPERATING AND ANOTHER TO BE OPERABLE. THE 'C' CHILLER WAS RETURNED TO SERVICE AT 2236 HOURS, BUT IT TRIPPED AGAIN AT 0015 HOURS ON JULY 30. BY 0140 HOURS THE 'A' AND 'C' CHILLERS WERE RETURNED TO SERVICE. ON AUGUST 5, 1987 AT 0245 HOURS, THE 'C' CONTROL/RELAY ROOM CHILLER TRIPPED ON HIGH DISCHARGE PRESSURE AND WAS DECLARED INOPERABLE. THE 'B' CHILLER HAD PREVIOUSLY BEEN DECLARED INOPERABLE FOR THE SAME REASON, LEAVING ONLY THE 'A' CHILLER OPERATING. THE 'B' AND 'C' CHILLERS WERE RETURNED TO SERVICE ON AUGUST 5 AT 0319 HOURS. AT THE TIME OF THESE EVENTS, UNITS 1 AND 2 WERE AT 100% POWER. THE CHILLERS TRIPPED AND/OR HAD FREON RELIEF VALVES LIFT DUE TO HIGH CONDENSER DISCHARGE PRESSURE RESULTING FROM LOW SERVICE WATER (SW) FLOW. THE LOW SW FLOW WAS DUE TO A BUILDUP OF SEAWEED IN THE UPSTREAM SW STRAINERS (EIIS-STR). THE SW STRAINERS WERE CLEARED OF DEBRIS.

[263] SURRY 1 DOCKET 50-280 LER 87-019
MANUAL REACTOR TRIP DUE TO FAILED RELAY IN MAIN TRANSFORMER.
EVENT DATE: 080787 REPORT DATE: 090487 NSSS: WE TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 206242) ON AUGUST 7, 1987 AT 0120 HOURS, WITH THE UNIT AT APPROXIMATELY 98% POWER, A MANUAL REACTOR TRIP WAS INITIATED, FOLLOWING A SHORT RAMPDOWN, DUE TO LOSS OF COOLING AND OIL FLOW FOR THE 'B' MAIN TRANSFORMER (EIIS- XFMR). THE TRANSFORMER LOCAL ANNUNCIATOR PANEL (EIIS-ANN) HAD FAILED TO INITIATE A TROUBLE ALARM ON THE CONTROL ROOM ANNUNCIATOR PANEL. FOLLOWING THE TRIP, ALL CONTROL AND PROTECTION SYSTEMS FUNCTIONED PROPERLY EXCEPT: - ONE CHANNEL OF SOURCE RANGE NUCLEAR INSTRUMENTATION (SR NI) (EIIS-IG) DID NOT AUTOMATICALLY REINSTATE. THE 'A' MAIN FEEDWATER PUMP (MFP) (EIIS-P) DID NOT AUTOMATICALLY TRIP AND THE PUMP'S MINIMUM FLOW RECIRC VALVE DID NOT OPEN ON A LOW FEEDWATER FLOW CONDITION. OPERATORS MANUALLY REINSTATED THE SR NI AND TRIPPED THE 'A' MFP. THE UNIT WAS QUICKLY STABILIZED AT HOT SHUTDOWN. POWER WAS LOST TO THE 'B' MAIN TRANSFORMER COOLING FANS AND OIL CIRCULATING PUMPS WHEN AN UNDERVOLTAGE RELAY (EIIS-RLY) IN THE COOLING CONTROL CIRCUIT FAILED. THE RELAY AND ANNUNCIATOR PANEL FAILURES ARE ATTRIBUTED TO AGING. ONE CHANNEL OF SR NI FAILED TO REINSTATE DUE TO UNDER-COMPENSATION OF ONE CHANNEL OF IR NI. THE 'A' MFP DID NOT TRIP AND ITS MINIMUM FLOW RECIRC VALVE DID NOT OPEN DUE TO A FAILED AIR REGULATOR DIAPHRAGM TO THE PNEUMATIC PRESSURE SWITCH WHICH SENSES MAIN FEEDWATER FLOW. THE FAILED RELAY AND THE LOCAL ANNUNCIATOR PANEL WERE REPLACED.

[264] SURRY 1 DOCKET 50-280 LER 87-020
MAIN CONTROL ROOM VENTILATION ISOLATION DUE TO HIGH VOLTAGE OUTPUT ON CHLORINE GAS DETECTOR.
EVENT DATE: 081387 REPORT DATE: 091187 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)
VENDOR: CAPITAL CONTROLS CO.

(NSIC 206045) ON AUGUST 13, 1987 AT 0117 HOURS, AND ON AUGUST 21, 1987 AT 2126 HOURS, THE MAIN CONTROL ROOM 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-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) REALIGNED. AT THE TIME OF THESE OCCURRENCES, BOTH UNIT 1 AND UNIT 2 WERE OPERATING AT 100% POWER. THE CHLORINE GAS TREATMENT SYSTEM AT THE SEWAGE TREATMENT PLANT IS SCHEDULED TO BE REPLACED WITH AN ULTRAVIOLET TREATMENT SYSTEM. AT THAT TIME, A TECHNICAL SPECIFICATION REVISION WILL BE PREPARED TO REMOVE THE CHLORINE GAS DETECTORS.

[265] SURRY 1 DOCKET 50-280 LER 87-021
CONTROL/RELAY ROOM CHILLERS TRIPPED DUE TO INADEQUATE SERVICE WATER FLOW.
EVENT DATE: 081887 REPORT DATE: 091187 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)

(NSIC 206046) ON AUGUST 18, 1987 AT 1139 HOURS, ONE OF THE THREE CONTROL/RELAY ROOM CHILLERS (1-VS-E-4B) (EIIS-CHU) TRIPPED ON HIGH CONDENSER DISCHARGE PRESSURE. THE 'A' CHILLER UNIT HAD PREVIOUSLY BEEN REMOVED FROM SERVICE IN ORDER TO CLEAN ITS SERVICE WATER (SW) SUCTION STRAINER. AT 1142 HOURS, THE 'C' CHILLER WAS DECLARED INOPERABLE WHEN ITS FREON RELIEF VALVE LIFTED. THIS IS CONTRARY TO TECHNICAL SPECIFICATION 3.14 WHICH REQUIRES ONE CONTROL/RELAY ROOM CHILLER TO BE OPERATING AND ANOTHER TO BE OPERABLE. THE 'A' CHILLER WAS RETURNED TO SERVICE AT 1143 HOURS AND THE 'B' CHILLER WAS RETURNED TO SERVICE AT 1300 HOURS. AT THE TIME OF THESE EVENTS, BOTH UNITS 1 AND 2 WERE AT 100% POWER. THE CAUSE OF THIS EVENT WAS INSUFFICIENT SW FLOW TO THE CONDENSER OF THE CHILLER UNITS DUE TO A BUILDUP OF SEAWEED IN THE Y-TYPE SUCTION STRAINERS. THE SW STRAINERS WERE CLEARED OF DEBRIS. SERVICE WATER PRESSURE TO THE CONTROL/RELAY ROOM CHILLERS IS CHECKED EVERY FOUR HOURS; AND THE SW STRAINERS ARE CLEANED WHENEVER SW PRESSURE BECOMES LOW. THE CONTROL/RELAY ROOM VENTILATION SYSTEM IS BEING EVALUATED BY ENGINEERING, AND PLANS ARE BEING DEVELOPED TO UPGRADE THE SYSTEM. THE ROTATING SW STRAINER HAS BEEN REPLACED.

[266] SURRY 1 DOCKET 50-280 LER 87-022
MAIN CONTROL ROOM VENTILATION ISOLATION DUE TO HIGH VOLTAGE OUTPUT ON CHLORINE GAS DETECTOR.
EVENT DATE: 083087 REPORT DATE: 092587 NSSS: WE TYPE: PWR
VENDOR: CAPITAL CONTROLS CO.

(NSIC 206490) ON AUGUST 30, 1987 AT 1533 HOURS, AND ON SEPTEMBER 3, 1987 AT 2314 HOURS, THE MAIN CONTROL ROOM 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 CHLORINE GAS PRESENT IN THE CONTROL ROOM. THE CHLORINE DETECTOR WAS RESET AND THE MAIN CONTROL ROOM VENTILATION EIIS-VI REALIGNED. AT THE TIME OF THESE OCCURRENCES, BOTH UNIT 1 AND UNIT 2 WERE OPERATING AT 100% POWER. 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.

[267] SURRY 1 DOCKET 50-280 LER 87-023
INADVERTENT AUTO START OF STANDBY CHARGING PUMP DUE TO PERSONNEL ERROR.
EVENT DATE: 090187 REPORT DATE: 092587 NSSS: WE TYPE: PWR

(NSIC 206491) ON SEPTEMBER 1, 1987 AT 1432 HOURS, WITH THE UNIT 1 'B' CHARGING PUMP IN SERVICE, THE STANDBY CHARGING PUMP (1-CH-P-1A) (EIIS-P) AUTOMATICALLY

STARTED ON A LOW CHARGING PUMP DISCHARGE PRESSURE SIGNAL. THIS OCCURRED WHEN AN INSTRUMENT TECHNICIAN INADVERTENTLY ISOLATED THE CHARGING PUMP DISCHARGE HEADER PRESSURE TRANSMITTER EIIS-PT. THE PRESSURE TRANSMITTER WAS UNISOLATED, AND THE 'A' CHARGING PUMP WAS RETURNED TO STANDBY CONDITION AT 1441 HOURS. AT THE TIME OF THIS EVENT, UNIT 1 WAS AT 100% POWER. A HPES INVESTIGATION IS IN PROGRESS AT THE TIME OF THIS REPORT. RECOMMENDATIONS ANTICIPATED FOR IMPLEMENTATION ARE INCREASED TRAINING ON INSTRUMENT MARK NUMBER DESIGNATIONS AND INCREASED EMPHASIS ON VERIFICATION OF MARK NUMBERS.

[268] SUSQUEHANNA 2 DOCKET 50-388 LER 87-009
REACTOR WATER CLEANUP ISOLATION FROM HIGH ROOM DIFFERENTIAL TEMPERATURE SIGNAL.
EVENT DATE: 090287 REPORT DATE: 100287 NSSS: GE TYPE: BWR

(NSIC 206583) ON SEPTEMBER 2, 1987 WITH UNIT 2 IN CONDITION 1 AT 100% POWER, A DIVISION II REACTOR WATER CLEANUP (RWCU) ISOLATION OCCURRED. THE ISOLATION WAS INITIATED FROM A RWCU FILTER/DEMINERALIZER (F/D) ROOM HIGH DIFFERENTIAL TEMPERATURE SIGNAL. THE RWCU SYSTEM ISOLATED PER DESIGN AND IT WAS VERIFIED THAT NO ACTUAL STEAM LEAKS HAD TRIGGERED THE ISOLATION. THE HVAC ZONE II SUPPLY HEATERS WERE TURNED ON TO DECREASE ROOM DELTA TEMPERATURE AND THE HIGH DELTA TEMPERATURE ALARM CLEARED SHORTLY THEREAFTER. RWCU WAS THEN RETURNED TO SERVICE. RWCU DIVISION II ROOM TEMPERATURES WERE PERIODICALLY MONITORED OVER THE NEXT TWO NIGHTS AND NO SIGNIFICANT FLUCTUATIONS WERE NOTED. AN EVALUATION/REVIEW OF THE TEMPERATURE LEAK DETECTION REQUIREMENTS FOR THE RWCU SYSTEM WILL BE MADE REGARDING TEMPERATURE INSTRUMENT SETPOINTS AND LOCATIONS. FINAL CORRECTIVE ACTIONS WILL BE DETERMINED AT THE CONCLUSION OF THE EVALUATION. THE CAUSE OF THE EVENT WAS ATTRIBUTED TO TWO FACTORS, TEMPERATURE ELEMENT LOCATION AND THE DESIGN TEMPERATURE SETPOINT SELECTION FOR THE RWCU F/D ROOM DIFFERENTIAL TEMPERATURE TRIP. THIS EVENT HAS BEEN DETERMINED TO BE REPORTABLE PER 10CFR 50.73 (A)(2)(IV), IN THAT AN UNPLANNED ENGINEERED SAFETY FEATURE (ESF) ACTUATION TOOK PLACE WHEN THE RWCU SYSTEM ISOLATED AUTOMATICALLY ON A HIGH DIFFERENTIAL TEMPERATURE SIGNAL FROM RWCU F/D ROOM TEMPERATURE INSTRUMENTATION. CONSEQUENCES RESULTIN

[269] SUSQUEHANNA 2 DOCKET 50-388 LER 87-010
AUXILIARY BOILER ARC-OVER CAUSES PRIMARY CONTAINMENT ISOLATION VALVE CLOSURE.
EVENT DATE: 090287 REPORT DATE: 100287 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: SUSQUEHANNA 1 (BWR)
VENDOR: BECKMAN INSTRUMENTS, INC.

(NSIC 206557) ON SEPTEMBER 2, 1987 AT 2118 HOURS, DURING NORMAL OPERATIONS AT 100% POWER, AUXILIARY BOILER "B" EXPERIENCED AN INTERNAL ELECTRICAL ARC-OVER WHICH CAUSED AN OVERCURRENT TRIP OF ITS 13.8KV SUPPLY BREAKER. THIS CAUSED A TRANSIENT ON STARTUP BUS WHICH RESULTED IN CLOSURE OF A UNIT 2 PRIMARY CONTAINMENT ISOLATION VALVE AND VARIOUS OTHER SYSTEM PERTURBATIONS AND ALARMS. REACTOR POWER REMAINED CONSTANT THROUGHOUT THE OCCURRENCE AND RECOVERY. ALL AFFECTED SYSTEMS WERE PROMPTLY RETURNED TO SERVICE. CAUSE OF THE ARC-OVER WAS A BROKEN CONDUCTIVITY ELEMENT COMBINED WITH THE BOILER BEING OPERATED IN HOT STANDBY WITH NO LOADS FOR AN EXTENDED PERIOD OF TIME OF SIX HOURS. THE BROKEN CONDUCTIVITY ELEMENT CAUSED UNNECESSARY CHEMICAL INJECTION INTO THE BOILER'S FEEDWATER. SINCE THERE WAS NO LOAD ON THE BOILER, ONLY INTERMITTENT INJECTIONS OF FEEDWATER WERE REQUIRED TO MAINTAIN PRESSURE IN THE BOILER. THESE PERIOD OF LOW FEEDWATER FLOW PERMITTED CONCENTRATION OF CHEMICALS IN THE FEEDWATER CREATING HIGHLY CONDUCTIVE FEEDWATER. WHEN THIS HIGHLY CONDUCTIVE FEEDWATER WAS INJECTED INTO THE BOILER, AN ARC-OVER ACROSS THE ELECTRODES OCCURRED. THE BROKEN CONDUCTIVITY ELEMENT WAS REPLACED AND THE AUXILIARY BOILER PREPARED FOR OPERATION.

[270]   THREE MILE ISLAND 2   DOCKET 50-320   LER 87-002
UNATTENDED RADIOACTIVE MATERIAL IN AN UNRESTRICTED AREA.
EVENT DATE: 022587   REPORT DATE: 032687   NSSS: BW   TYPE: PWR

(NSIC 206573) AT APPROXIMATELY 1330 HOURS ON WEDNESDAY, FEBRUARY 25, 1987, RADIOACTIVE MATERIAL WAS DISCOVERED UNATTENDED IN A PICKUP TRUCK IN AN UNRESTRICTED AREA. THE MATERIAL CONSISTED OF A 500 ML BOTTLE THAT CONTAINED RADIOACTIVE FILTERS. RADIOLOGICAL SURVEYS OF THE 500 ML BOTTLE INDICATED 46 MR/HR GAMMA AND 2.4 RAD/HR BETA CONTACT DOSE RATES. 10 CFR 20.405(A)(1)(V) REQUIRES LICENSEES TO REPORT RADIATION LEVELS IN UNRESTRICTED AREAS IN EXCESS OF TEN TIMES ANY APPLICABLE LIMIT. 10 CFR 20.105(B)(1) PROHIBITS THE POSSESSION OR TRANSFER OF LICENSED MATERIAL IN SUCH A MANNER AS TO CREATE RADIATION LEVELS IN AN UNRESTRICTED AREA THAT COULD RESULT IN A DOSE IN EXCESS OF TWO MILLIREM IN ANY ONE HOUR. THEREFORE, THIS EVENT IS REPORTABLE PURSUANT TO 10 CFR 20.405(A)(1)(V) SINCE THE RADIATION LEVELS OF THE SAMPLE BOTTLE WERE IN EXCESS OF TEN TIMES THE LIMIT SET FORTH IN 10 CFR 20.105(B)(1). BASED ON CRITIQUES, IT HAS BEEN DETERMINED THAT THE SAMPLE BOTTLE WAS UNKNOWINGLY LOADED INTO THE PICKUP TRUCK WITH OTHER RADIOACTIVE MATERIALS ON TUESDAY, FEBRUARY 24, 1987. THE SAMPLE BOTTLE WAS NOT DISCOVERED OR REMOVED FROM THE TRUCK BED WHEN THE OTHER MATERIAL WAS REMOVED.

[271]   THREE MILE ISLAND 2   DOCKET 50-320   LER 87-007
LOW PRESSURE IN THE CABLE ROOM AND TRANSFORMER ROOM HALON SYSTEM DUE TO A DEFICIENT SOLENOID PILOT VALVE.
EVENT DATE: 080787   REPORT DATE: 091787   NSSS: BW   TYPE: PWR
VENDOR: CARDOX CORP.

(NSIC 206317) DURING THE 2300-0700 SHIFT COMMENCING ON AUGUST 7, 1987 A QUALITY ASSURANCE (QA) MONITOR OBSERVED AN APPARENT ABNORMAL PRESSURE INDICATION ON ONE (1) OF 19 HALON CYLINDERS FOR THE CABLE ROOM AND TRANSFORMER ROOM HALON SYSTEM. SPECIFICALLY, HALON BOTTLE FH-165-002164 INDICATED APPROXIMATELY 340 PSIG. IT WAS LATER DETERMINED THAT THIS INDICATION WAS 200 PSIG LESS THAN THE MINIMUM LIMIT, I.E., 90% OF FULL PRESSURE, REQUIRED BY THE LIMITING CONDITION OF OPERATION (LCO) TECHNICAL SPECIFICATION (TECH SPECS) 3.7.10.3. HOWEVER, THE QA MONITOR DID NOT REALIZE THAT THE OBSERVED CONDITION CONSTITUTED NONCOMPLIANCE WITH THE REFERENCED LCO UNTIL AUGUST 18, 1987, WHEN THE EVENT WAS DISCUSSED WITH THE TMI-2 FIRE PROTECTION ENGINEER. ACCORDINGLY, AT 1408 HOURS ON 8/18/87, THE ACTION STATEMENT OF TECH SPEC 3.7.10.3 WAS ENTERED. THE REQUIRED HOURLY FIREWATCH AND BACKUP FIRE SUPPRESSION EQUIPMENT WERE ESTABLISHED WITHIN ONE (1 HOUR) AND RESTORATION OF THE SYSTEM TO OPERABLE STATUS WAS ACCOMPLISHED WITHIN 14 DAYS. A REPLACEMENT HALON BOTTLE WAS INTALLED AND THE SYSTEM WAS INITIALLY RESTORED TO OPERABLE STATUS AT 0740 HOURS ON 8/20/87. AN INVESTIGATION DETERMINED THAT THE ROOT CAUSE WAS A DEFICIENT SOLENOID PILOT VALVE. THE VALVE WAS REPLACED AND THE SYSTEM WAS RETURNED TO OPERABLE STATUS ON 9/12/87.

[272]   TROJAN   DOCKET 50-344   LER 87-020
SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE MISSED DUE TO INADVERTENT DELETION FROM SCHEDULE.
EVENT DATE: 080387   REPORT DATE: 090287   NSSS: WE   TYPE: PWR

(NSIC 206250) ON AUGUST 3, 1987, DURING A REVIEW OF SURVEILLANCE RECORDS, IT WAS DETERMINED THAT THE MONTHLY CHANNEL CHECK OF THE SEISMIC MONITORING INSTRUMENTATION REQUIRED BY TECHNICAL SPECIFICATION 4.3.3.3.1 WAS NOT PERFORMED IN JULY 1987. THE CAUSE OF THIS EVENT WAS AN INADVERTENT DELETION OF THE SURVEILLANCE FROM THE COMPUTERIZED SURVEILLANCE SCHEDULE. IT IS NOT KNOWN EXACTLY HOW THE SURVEILLANCE WAS DELETED, BUT IT MAY HAVE BEEN A PERSONNEL ERROR WHILE ENTERING DATA ON THE SCHEDULE. THE REQUIRED SURVEILLANCE WAS PERFORMED AND THE SEISMIC MONITORING INSTRUMENTATION WAS VERIFIED AS OPERABLE. THE COMPUTERIZED SURVEILLANCE SCHEDULE WILL BE REVIEWED TO IDENTIFY ADDITIONAL

SAFEGUARDS WHICH CAN BE IMPLEMENTED TO PROTECT AGAINST INADVERTENT DELETION OF ENTRIES. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[273] TROJAN DOCKET 50-344 LER 87-021
OPENING OF INCORRECT FUSE DRAWER CAUSED 12.47 KV BUS UNDERVOLTAGE - EDG STARTED.
EVENT DATE: 082187 REPORT DATE: 091887 NSSS: WE TYPE: PWR

(NSIC 206068) ON AUGUST 21, 1987, ACTIONS WERE IN PROGRESS TO MAKE 12.47 KV BUSES H1 AND H2 OPERATIONAL FROM THE UNIT AUXILIARY TRANSFORMER. AN OPERATOR HAD JUST COMPLETED INSTALLING POTENTIAL TRANSFORMER FUSES FOR BUSES H1 AND H2 WHEN HE OPENED THE STARTUP TRANSFORMER FUSE DRAWER BY MISTAKE. THIS CAUSED A SENSED UNDERVOLTAGE CONDITION, AND THE "A" EMERGENCY DIESEL GENERATOR STARTED AUTOMATICALLY. THE CAUSE WAS PERSONNEL ERROR. THE OPERATOR OPENED THE POTENTIAL TRANSFORMER FUSE DRAWER FOR THE STARTUP TRANSFORMER THINKING THAT THIS WAS THE DRAWER FOR THE MAIN GENERATOR POTENTIAL TRANSFORMER FUSES. A CONTRIBUTING CAUSE WAS A LACK OF A PROCEDURE FOR REPLACING THE POTENTIAL TRANSFORMER FUSES. THE OPERATOR WAS COUNSELED ON THE CORRECT METHOD FOR POTENTIAL TRANSFORMER FUSE REPLACEMENT. A WRITTEN PROCEDURE WILL BE PREPARED PROVIDING INSTRUCTIONS ON FUSE REPLACEMENT. THE ADMINISTRATION OF LOAD DISPATCHER SWITCHING ORDERS AND THE PRACTICES ON FUSE REPLACEMENT WILL BE REVIEWED FOR IMPROVEMENT. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[274] TROJAN DOCKET 50-344 LER 87-007 REV 01
UPDATE ON PRESSURIZER SAFETY VALVES LIFTED OUT-OF-TOLERANCE DURING SURVEILLANCE TESTING DUE TO DEFICIENCY IN BENCH TEST PROCEDURE.
EVENT DATE: 082287 REPORT DATE: 092187 NSSS: WE TYPE: PWR
VENDOR: CROSBY VALVE & GAGE CO.

(NSIC 206414) DURING PERFORMANCE OF PRESSURIZER SAFETY VALVE (PSV) TESTING ON APRIL 1, 1987, PSV-8010C LIFTED AT 2596 PSIG. ON AUGUST 22, 1987, PSV-8010A LIFTED AT 2658 PSIG AND PSV-8010B LIFTED AT 2598 PSIG. THIS EXCEEDS THE TECH SPEC ALLOWED TOLERANCE OF 2485 PSIG +/- 1% (I.E., 2460 TO 2510 PSIG). THE VALVE LIFT SETPOINT WAS ADJUSTED AND THE VALVE RETESTED SATISFACTORILY. THE CAUSE OF THIS EVENT IS A DEFICIENCY IN THE BENCH TEST PROCEDURE FOR THE PSVS. THE LIFT SETTINGS ESTABLISHED IN BENCH TESTING DO NOT CORRELATE TO THE IN-PLACE TESTING LIFT SETTINGS. BENCH-TO-IN-PLACE TESTING SETPOINT CORRELATIONS WILL BE DEVELOPED FOR EACH OF THE PSVS IF BENCH TESTING IS USED IN THE FUTURE. IN THE INTERIM, IN-PLACE TESTING WILL BE USED FOR ESTABLISHING FINAL LIFT SETTINGS. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[275] TROJAN DOCKET 50-344 LER 87-022
ACCUMULATOR WATER LEVELS INADVERTENTLY ALLOWED TO DECREASE BELOW TECH SPEC LIMIT.
EVENT DATE: 082287 REPORT DATE: 092187 NSSS: WE TYPE: PWR

(NSIC 206415) ON AUGUST 22, 1987, A LEAK TEST OF THE "A" ACCUMULATOR DISCHARGE CHECK VALVES TO THE REACTOR COOLANT SYSTEM WAS IN PROGRESS. DURING THE TEST, THE WATER LEVEL IN THE "B" AND "C" ACCUMULATORS INADVERTENTLY DECREASED BELOW THE TECH SPEC LIMIT OF 64% AS FOLLOWS: "B" ACCUMULATOR 58%, "C" ACCUMULATOR 59%. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR IN THAT THE OPERATORS HAD ALLOWED THE "A" ACCUMULATOR LEVEL TO DROP BELOW THE LOW LEVEL ALARM SETPOINT AND WERE NOT ATTENTIVE TO SUBSEQUENT LEVEL DECREASES IN THE "B" AND "C" ACCUMULATORS. THE REASON THE ACCUMULATOR LEVELS DECREASED BELOW THE TECH SPEC LIMIT IS STILL BEING INVESTIGATED. THE ACCUMULATORS WERE REFILLED AND OPERATORS WERE COUNSELED ON THE NEED TO CLOSELY MONITOR ACCUMULATOR LEVELS DURING POT-2-4. A CAUTION REQUIRING CLOSE MONITORING OF ACCUMULATOR LEVELS WILL BE ADDED TO POT-2-4. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[276] TROJAN DOCKET 50-344 LER 87-023
FEEDWATER REGULATING VALVE FAILED TO CLOSE DUE TO TURBINE TRIP ON HI-HI STEAM GENERATOR LEVEL.
EVENT DATE: 082687 REPORT DATE: 092587 NSSS: WE TYPE: PWR

(NSIC 206553) ON AUGUST 26, 1987 A PLANT STARTUP WAS IN PROGRESS, WHEN INCREASING WATER LEVEL IN THE "C" STEAM GENERATOR WAS OBSERVED. THE "C" FEEDWATER REGULATING VALVES AND BYPASS VALVES WERE CLOSED FROM THE CONTROL ROOM, HOWEVER, "C" STEAM GENERATOR LEVEL CONTINUED TO INCREASE. LOCAL CLOSURE OF THE "C" FEEDWATER BYPASS ISOLATION VALVE (8 INCH) ALSO FAILED TO STOP THE OVER-FEEDING OF THE "C" STEAM GENERATOR AND A FEEDWATER ISOLATION SIGNAL WAS RECEIVED ON HI-HI LEVEL IN THE "C" STEAM GENERATOR. THE CAUSE OF THIS EVENT WAS FAILURE OF THE "C" FEEDWATER REGULATING VALVE TO FULLY CLOSE. THE VALVE WAS FOUND TO BE OPEN ABOUT ONE TURN. THIS WAS DUE TO THE VALVE LOCAL OPERATOR NOT BEING ADEQUATELY SET IN THE NEUTRAL POSITION, WHICH PREVENTS FULL CLOSURE OF THE VALVE FROM THE CONTROL ROOM. THE IMMEDIATE CORRECTIVE ACTION WAS TO REDUCE REACTOR POWER AND STABILIZE STEAM GENERATOR LEVELS USING THE AUXILIARY FEEDWATER. THE PROCEDURES FOR POSITIONING THE FEEDWATER REGULATING VALVE LOCAL OPERATORS WILL BE REVIEWED. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[277] TROJAN DOCKET 50-344 LER 87-024
TURBINE TRIP/REACTOR TRIP DUE TO "C" STEAM GENERATOR HIGH-HIGH LEVEL.
EVENT DATE: 082887 REPORT DATE: 092587 NSSS: WE TYPE: PWR

(NSIC 206555) ON AUGUST 28, 1987, THE PLANT WAS OPERATING AT 30 PERCENT POWER AT NORMAL OPERATING TEMPERATURE AND PRESSURE. AT 0831, A TURBINE TRIP OCCURRED DUE TO "C" STEAM GENERATOR HIGH-HIGH LEVEL WHILE BRINGING THE SOUTH MAIN FEED PUMP (MFP) ON LINE. THIS RESULTED IN A REACTOR TRIP. THE CAUSE OF THE EVENT WAS COGNITIVE PERSONNEL ERROR. THE OPERATOR MISREAD THE AUTOMATIC SPEED CONTROLLER INDICATION AND PLACED THE PUMP IN SERVICE AT MAXIMUM SPEED INSTEAD OF MINIMUM SPEED. THE RESULTING TRANSIENT LED TO INADEQUATE FEED FLOW AND STEAM GENERATOR LEVEL CONTROL. SLUGGISH BEHAVIOR OF THE "C" FEEDWATER REGULATING VALVE WAS A CONTRIBUTING FACTOR. THE EVENT WAS DISCUSSED WITH ALL PLANT OPERATORS. THE SLUGGISH BEHAVIOR OF THE "C" FEEDWATER REGULATING VALVE IS BEING INVESTIGATED. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[278] TROJAN DOCKET 50-344 LER 87-025
CONTAINMENT VENTILATION ISOLATION DUE TO SPURIOUS ACTUATION OF NOBLE GAS MONITOR.
EVENT DATE: 083087 REPORT DATE: 092987 NSSS: WE TYPE: PWR
VENDOR: VICTOREEN INSTRUMENT DIVISION

(NSIC 206494) ON AUGUST 30, 1987, A CONTAINMENT VENTILATION ISOLATION OCCURRED DURING A CONTAINMENT PRESSURE REDUCTION. THE CONTAINMENT VENTILATION ISOLATION WAS DUE TO A MOMENTARY SPIKE ON THE CONTAINMENT RADIATION MONITORING SYSTEM LOW LEVEL NOBLE GAS MONITOR (PRM-1C) WHICH EXCEEDED THE SETPOINT FOR THE MONITOR. THE CAUSE OF PRM-1C SPIKING HIGH IS BELIEVED TO BE A SPURIOUS ELECTRONICS FAILURE IN THE DETECTOR ANTI-JAM CIRCUITRY. THIS CIRCUITRY IS PROVIDED TO PROTECT DETECTORS FROM SATURATION. UPON FAILURE OF THE ANTI-JAM CIRCUITRY, PRM-1C IS DESIGNED TO FAIL HIGH. THE PRINTED CIRCUIT CARD FOR THE ANTI-JAM CIRCUITRY WAS REPLACED. PRM-1C WAS RETURNED TO SERVICE. NO FURTHER SPURIOUS ACTUATION OF PRM-1C WAS OBSERVED. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[279] TROJAN DOCKET 50-344 LER 87-026
MISSED SURVEILLANCE OF POST-ACCIDENT LEAKAGE OUTSIDE CONTAINMENT.
EVENT DATE: 090187 REPORT DATE: 100187 NSSS: WE TYPE: PWR

(NSIC 206591) ON SEPTEMBER 1, 1987 DURING A REVIEW OF SURVEILLANCE RECORDS, IT WAS DETERMINED THAT PERIODIC ENGINEERING TEST (PET) 9-3, "POST-ACCIDENT LEAKAGE

OUTSIDE CONTAINMENT", REQUIRED BY TECHNICAL SPECIFICATION 6.4.?.A. WAS NOT PERFORMED WITHIN THE 18 MONTH INTERVAL. THE SYSTEMS FOR WHICH THE SURVEILLANCE WAS MISSED WERE: (L) THE "B" CENTRIFUGAL CHARGING PUMP PIPING. (2) THE SAFETY INJECTION SYSTEM. AND (3) THE RESIDUAL HEAT REMOVAL SYSTEM. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR. A SCHEDULE FOR PET-9-3 HAD NOT BEEN PREPARED AND. THEREFORE, THE SURVEILLANCE WAS MISSED. THE IMMEDIATE CORRECTIVE ACTION WAS TO PERFORM THE TECHNICAL SPECIFICATION REQUIRED SURVEILLANCE. AN UPDATED SURVEILLANCE SCHEDULE FOR PET-9-3 WAS ISSUED WHICH SHOWS THE DUE DATES FOR THE NEXT REQUIRED SURVEILLANCE. THE PET SCHEDULE WILL BE ADDED TO THE COMPUTERIZED SURVEILLANCE SCHEDULE TO ELIMINATE THE NEED TO MANUALLY SCHEDULE SURVEILLANCES. THIS WILL BE COMPLETED BY MARCH 1, 1988. THIS EVENT HAD NO EFFECT ON PUBLIC HEALTH AND SAFETY.

[280] TURKEY POINT 3 DOCKET 50-250 LER 87-022
REACTOR TRIP BREAKERS OPEN WHILE IN COLD SHUTDOWN DUE TO SPURIOUS SPIKE ACTUATING THE INTERMEDIATE RANGE LOW POWER HIGH FLUX REACTOR TRIP.
EVENT DATE: 073187 REPORT DATE: 083187 NSSS: WE TYPE: PWR
VENDOR: POWER DESIGNS INC.

(NSIC 206238) ON JULY 31, 1987, UNIT 3 WAS AT COLD SHUTDOWN (MODE 5). AT 0548, THE REACTOR TRIP BREAKERS (RTB) WERE CLOSED FOR MAINTENANCE AND TESTING BY WESTINGHOUSE AND FPL INSTRUMENT AND CONTROL PERSONNEL. A SPURIOUS SPIKING PROBLEM WAS KNOWN TO EXIST IN THE NUCLEAR INSTRUMENTATION SYSTEM (NIS) INTERMEDIATE RANGE (IR) CABINET. A PLANT WORK ORDER HAD BEEN SUBMITTED TO THE I&C DEPARTMENT TO MAKE THE NECESSARY REPAIRS TO NIS CH-36. AT 0717, THE OPERATOR ON SHIFT DEMONSTRATED TO THE ON-COMING OPERATOR THE KNOWN SPURIOUS SPIKING PROBLEM IN THE NUCLEAR INSTRUMENTATION SYSTEM INTERMEDIATE RANGE CHANNELS. THE DRAWER FOR NIS CH-32 WAS PULLED OPEN AND CAUSED NIS CH-36 TO SPIKE HIGH. THE SPURIOUS SPIKE ACTUATED. THE IR LOW POWER HIGH FLUX REACTOR TRIP ALARM AND OPENED THE RTB'S. INVESTIGATION OF THE PROBLEM REVEALED A FAULTY LOCKING MECHANISM ON THE 25 VOLT POWER SUPPLY CONNECTOR FOR NIS CH-36. THE LOCKING MECHANISM WAS REPLACED AND THE POWER SUPPLY CABLES FOR NIS CH-32 AND NIS CH-36 WERE RE-ROUTED IN THE NIS CABINET SO THEY WOULD NOT INTERFERE WITH EACH OTHER IN THE COURSE OF MAINTENANCE AND TESTING OF THEIR CHANNELS.

[281] TURKEY POINT 4 DOCKET 50-251 LER 87-018 REV 01
UPDATE ON OPERATION WITH GREATER THAN 2 PERCENT CALCULATED QUADRANT POWER TILT RATIO.
EVENT DATE: 071287 REPORT DATE: 081187 NSSS: WE TYPE: PWR
VENDOR: ELECTRO CORPORATION
WESTINGHOUSE ELECTRIC CORP.

(NSIC 206549) UNIT 4 WAS INCREASING POWER AFTER BEING RETURNED TO SERVICE ON JULY 9, 1987. POWER RANGE DETECTOR N-41 WAS REPLACED DURING THE OUTAGE AND THUS THE CHANNEL REQUIRED RECALIBRATION. UNTIL THE CALIBRATION WAS COMPLETED THE UPPER AND LOWER SECTION DEVIATION ALARMS WERE DECLARED TO BE OUT OF SERVICE AND A MANUAL CALCULATION OF QUADRANT POWER TILT RATIO (QPTR) WAS PERFORMED EACH SHIFT. AT 0023 ON JULY 11, THE POWER INCREASE WAS STOPPED AT 92% DUE TO A CALCULATED 2.84% QPTR. DUE TO OPERATING DIFFICULTIES UNRELATED TO QPTR, REACTOR POWER WAS REDUCED TO 75%. PRIOR TO INCREASING POWER FROM 75% A FLUX MAP WAS RUN AND IT VERIFIED HOT CHANNEL FACTORS TO BE ACCEPTABLE AND QPTR TO BE LESS THAN 2%. THE MANUAL CALCULATION QPTR PERFORMED CONCURRENTLY WITH THE FLUX MAP SHOWED DETECTOR CURRENTS TO USE IN THE MANUAL CALCULATIONS COULD BE ISSUED. THE 24 HOUR TIME LIMIT FOR TECH SPEC 3.2.6.I.1 HAD BEEN EXCEEDED. THE OVERPRESSURE/OVERTEMPERATURE DELTA TEMPERATURE SETPOINTS WERE NOT REDUCED AS REQUIRED BY TECH SPEC 3.2.6.1.2, BECAUSE POWER RANGE DETECTOR N-44 WAS OUT OF SERVICE. POWER RANGE LOGIC WILL NOT ALLOW TWO CHANNELS TO BE OUT OF SERVICE AT THE SAME TIME WITHOUT GENERATING A REACTOR TRIP. THE APPLICABLE PROCEDURE WILL BE REVISED TO ADDRESS THE 24 HOUR TIME LIMIT OF TECH SPECS 3.2.6.1.1.

[282] TURKEY POINT 4 DOCKET 50-251 LER 87-021
PROCESS RADIATION MONITOR SPIKE CAUSE CONTROL ROOM VENTILATION AND CONTAINMENT VENT ISOLATION.
EVENT DATE: 072987 REPORT DATE: 082887 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: TURKEY POINT 3 (PWR)
VENDOR: TRACER LAB

(NSIC 206239) ON JULY 29, 1987, AT 0935, WITH UNIT 4 AT 100% POWER, PROCESS RADIATION MONITOR (PRM) R-11, THE RADIOACTIVE PARTICULATE CONTAINMENT RADIATION MONITOR, SPIKED HIGH, ACTUATING THE CONTAINMENT VENT AND CONTROL ROOM VENTILATION ISOLATION LOGIC (EIIS:IL,JN). THE CONTROL ROOM VENTILATION ISOLATED AND SWITCHED INTO THE RECIRCULATION MODE, AND THE CONTAINMENT VENT ISOLATED, PER DESIGN. AT THE TIME OF THE SPIKE, WORK WAS BEING PERFORMED ON PRM R-15, THE CONDENSER AIR EJECTOR RADIATION MONITOR. IN THE PROCESS OF RANGING THE HIGH VOLTAGE POWER SUPPLY FROM 1000 VOLTS TO 1500 VOLTS AN ELECTRICAL TRANSIENT OCCURRED WHICH MOMENTARILY SPIKED THE R-11 DRAWER. R-11 SPIKED ABOVE ITS SETPOINT, RESULTING IN THE CONTAINMENT VENT AND CONTROL ROOM VENTILATION ISOLATION. TROUBLESHOOTING OF R-15 WAS HALTED, THE CAUSE OF THE SPIKE WAS VERIFIED, AND R-11 WAS RESET. BASED ON THE ABOVE, THE HEALTH AND SAFETY OF THE PUBLIC WERE NOT AFFECTED. THE AFFECTED PRMS DRAWERS HAVE BEEN REPLACED WITH AN UPGRADED SOLID STATE DRAWER. THE NEW DRAWERS DO NOT HAVE A SWITCHING CIRCUIT FOR HIGH VOLTAGE TESTING. THEY ARE EXPECTED TO BE LESS SUSCEPTIBLE TO ELECTRICAL TRANSIENTS AND SHOULD RESULT IN IMPROVED PERFORMANCE. ADDITIONAL DETAILS: SIMILAR OCCURRENCES: LER'S 251-86-030, 250-87-005, 250-87-008 MANUFACTURER: R-11: TRACER LAB INSTRUMENT. MODEL NUMBER: MAP-1B.

[283] VERMONT YANKEE DOCKET 50-271 LER 87-004 REV 01
UPDATE ON MISSED FIRE HOSE SURVEILLANCE DUE TO INADEQUATE PROCEDURE REVISION.
EVENT DATE: 072987 REPORT DATE: 100787 NSSS: GE TYPE: BWR

(NSIC 206552) ON 7/29/87, DURING NORMAL PLANT OPERATION AT 83% POWER, IT WAS DISCOVERED THAT SURVEILLANCE OF THE INSIDE FIRE HOSE STATIONS (EIIS=N/A) HAD NOT BEEN COMPLETED AS SPECIFIED BY TECH SPEC 4.13.C.1.B AND TECH SPEC 4.13.C.1.D. THE MISSED SURVEILLANCES ARE ATTRIBUTED TO ERRORS MADE IN REVISING THE SURVEILLANCE TEST PROGRAM AND SURVEILLANCE PROCEDURE. SUBSEQUENT INSPECTIONS AND HYDROSTATIC TESTS INITIATED ON 7/19/87 AND COMPLETED ON 8/13/87 IDENTIFIED THAT ALL INSIDE FIRE HOSE STATIONS WERE OPERABLE AT ALL TIMES.

[284] VERMONT YANKEE DOCKET 50-271 LER 87-005
TURBINE CONTROL SYSTEM MALFUNCTION RESULTS IN REACTOR SCRAM DUE TO PRESSURE TRANSIENT.
EVENT DATE: 080787 REPORT DATE: 090387 NSSS: GE TYPE: BWR
VENDOR: CAMPBELL, THOMAS J. CO
GENERAL ELECTRIC CO.
MICRO SWITCH

(NSIC 206419) ON 8/7/87 AT 2159, IN THE PROCESS OF SHUTTING DOWN FOR A SCHEDULED REFUELING OUTAGE WITH THE MAIN GENERATOR OFF-LINE AND TURBINE IN COAST-DOWN, A REACTOR SCRAM OCCURRED AS A RESULT OF REACTOR VESSEL HIGH PRESSURE. THE REACTOR VESSEL PRESSURE TRANSIENT WAS THE RESULT OF A MALFUNCTION IN THE TURBINE CONTROL SYSTEM (JJ)* THAT CAUSED THE BY-PASS VALVES TO CLOSE. PRIOR TO THIS EVENT, PRESSURE CONTROL WAS BEING PERFORMED BY THE ELECTRICAL PRESSURE REGULATOR (EPR). SOON AFTER SWITCHING TO THE MECHANICAL PRESSURE REGULATOR (MPR) FOR CONTROL (2155), THE BY-PASS VALVES SHUT RESULTING IN A REACTOR SCRAM FROM HIGH REACTOR VESSEL PRESSURE. THE INABILITY OF THE MPR TO CONTROL PRESSURE WAS CAUSED BY A CLOGGED SENSING LINE VALVE (JJ)* WHICH PREVENTED THE MPR FROM RESPONDING TO PRESSURE CHANGES. TO CORRECT THE CONTROL SYSTEM PROBLEM, THE SENSING LINE VALVE WILL BE REPLACED.

[285]   VERMONT YANKEE   DOCKET 50-271   LER 87-007
CONTAINMENT ISOLATION VALVE FAILURES DUE TO SEAT LEAKAGE.
EVENT DATE: 081287   REPORT DATE: 091187   NSSS: GE   TYPE: BWR
VENDOR: ALLIS CHALMERS
        WALWORTH COMPANY

(NSIC 206422) ON 8/12/87, 8/18/87 AND 8/23/87 WHILE PERFORMING TYPE C LEAK RATE TESTING WITH THE PLANT SHUTDOWN FOR THE 1987 REFUEL OUTAGE MAIN STEAM DRAIN VALVE MSD-77, LIQUID RADWASTE VALVES LRW-83, LRW-94, LRW-95, AND PRIMARY CONTAINMENT ATMOSPHERIC CONTROL VALVES PCAC-8,9,10,23 AND PCAC-6,7,6A,6B,7A,7B (EIIS=SB,WK,BB) WERE FOUND TO HAVE SEAT LEAKAGE ABOVE THAT PERMITTED BY TECH SPEC SECTION 3.7.A.4 (NOTE: THE PCAC VALVES ARE TESTED IN THE GROUPS LISTED AND ONE VALVE IN EACH GROUP IS SUSPECTED OF NOT MEETING THE ACCEPTANCE CRITERIA). ALSO ON 8/20/87 THE SUM TOTAL TYPE B (PENETRATIONS) AND TYPE C (VALVES) LEAKAGE EXCEEDED THAT ALLOWED BY 10CFR50 APPENDIX J. APPENDIX J LIMITS ALLOWABLE TOTAL B AND C PENETRATION LEAKAGE TO 0.60 LA. VERMONT YANKEE WILL PERFORM MAINTENANCE ON ALL OF THE ABOVE VALVES TO DETERMINE THE CAUSE OF FAILURE AND RETEST THEM TO ENSURE THAT SEAT LEAKAGE IS WITHIN ALLOWABLES PRIOR TO PLANT STARTUP FOLLOWING THE 1987 REFUELING OUTAGE.

[286]   VERMONT YANKEE   DOCKET 50-271   LER 87-008
LOSS OF NORMAL POWER DURING SHUTDOWN DUE TO ROUTING ALL OFF-SITE POWER SOURCES THROUGH ONE BREAKER.
EVENT DATE: 081787   REPORT DATE: 091487   NSSS: GE   TYPE: BWR

(NSIC 206430) AT APPROXIMATELY 1400 HOURS ON 8/17/87 WHILE THE PLANT WAS IN A REFUELING OUTAGE AND ALL OFF-SITE POWER WAS BEING ROUTED THROUGH ONE SET OF BREAKERS, AN INTERRUPTION ON THE GRID CAUSED THE PLANT TO LOSE NORMAL POWER SUPPLIES. THE EMERGENCY DIESEL GENERATORS (EDG) RESPONDED AS REQUIRED AS DID OTHER ENGINEERED SAFETY SYSTEMS. WHEN THE EDG'S STARTED, THEY WERE ABLE TO SUPPLY POWER TO ALL NECESSARY SYSTEMS. THREE PUMPS THAT STARTED IMMEDIATELY AFTER THE EDG'S WERE 2 SERVICE WATER PUMPS AND THE ELECTRIC FIRE PUMP. THE STARTING OF THESE THREE PUMPS, IN ADDITION TO THE DIESEL DRIVEN FIRE PUMP CAUSED A PRESSURE SURGE WHICH RUPTURED A TEMPORARY PIPING SYSTEM FABRICATED FROM 2" SCHEDULE 80 PVC PIPING (EIIS = KP). THE PVC PIPING WAS MADE BY ESLON. THE RUPTURED PIPE SPILLED ABOUT 2000 GALLONS OF RIVER WATER ONTO THE REFUELING FLOOR OF THE REACTOR BUILDING. AS A RESULT OF THE SPILL, THIS WATER WAS COMMUNICATED THROUGH THE FLOOR DRAIN SYSTEM WHICH RESULTED IN CONTAMINATING LOCAL AREAS OF THE REACTOR BUILDING. MINOR SEEPAGES THROUGH THE INTERFACE BETWEEN THE REACTOR BUILDING REFUEL FLOOR PANELING AND THE REACTOR BUILDING EXTERIOR WALLS WERE DETECTED. NO EQUIPMENT WAS DAMAGED AS A RESULT OF THE SPILL.

[287]   VERMONT YANKEE   DOCKET 50-271   LER 87-009
RELIEF VALVE ACCUMULATOR FAILED LEAK TEST DUE TO SOLENOID VALVE LEAKAGE.
EVENT DATE: 081887   REPORT DATE: 091787   NSSS: GE   TYPE: BWR
VENDOR: ASCO VALVES
        NUPRO COMPANY

(NSIC 206423) ON 8/18/87, WITH THE PLANT SHUTDOWN FOR THE 1987 REFUELING OUTAGE, THE "C" MAIN STEAM (EIIS=SB) RELIEF VALVE ACCUMULATOR ASSEMBLY (REFER TO ATTACHED SKETCH #1) WAS FOUND TO HAVE SEAT LEAKAGE THAT EXCEEDED THE ACCEPTANCE CRITERIA SPECIFIED IN THE SURVEILLANCE PROCEDURE. THE OTHER THREE MAIN STEAM RELIEF VALVE ACCUMULATOR ASSEMBLIES PASSED THE LEAK TESTS. INITIALLY, THE LEAKAGE WAS THOUGHT TO BE THROUGH THE CHECK VALVE IN THE ASSEMBLY. ON 9/4/87 THE CHECK VALVE WAS DISASSEMBLED AND HAD A SMALL DEPOSIT OF CORROSION PRODUCT ON THE SEAT. THE VALVE INTERNALS WERE CLEANED AND ON 9/4/87 THE ASSEMBLY WAS UNSUCCESSFULLY RETESTED. DURING THE RETEST, EXCESSIVE LEAKAGE WAS OBSERVED THROUGH THE SOLENOID VALVE'S EXHAUST PORT. THE ROOT CAUSE OF THIS LEAKAGE THROUGH THE SOLENOID VALVE IS NOT KNOWN AT THIS TIME. THE SOLENOID VALVE WILL BE INSPECTED AND THE ACCUMULATOR

ASSEMBLY WILL BE SUCCESSFULLY RETESTED PRIOR TO PLANT STARTUP. VERMONT YANKEE HAS NOT HAD A FAILURE OF AN ACCUMULATOR ASSEMBLY DUE TO A LEAKING SOLENOID VALVE IN THE LAST FIVE YEARS. BASED ON THIS, VERMONT YANKEE BELIEVES THAT NO SIGNIFICANT PROBLEM EXISTS WITH THE SYSTEM. THE CHECK VALVE APPEARED TO BE FUNCTIONING PROPERLY AND WAS NOT THE CAUSE OF THE FAILURE.

[288] VERMONT YANKEE DOCKET 50-271 LER 87-010
INADVERTENT PRIMARY CONTAINMENT ISOLATION SYSTEM ACTUATION AS A RESULT OF A DEFECTIVE PROCEDURE.
EVENT DATE: 082087 REPORT DATE: 090887 NSSS: GE TYPE: BWR

(NSIC 206424) AT APPROXIMATELY 0329 HOURS ON 8/20/87 WITH THE REACTOR IN THE COLD SHUTDOWN CONDITION, A PRIMARY CONTAINMENT ISOLATION SYSTEM (PCIS) GROUP III ISOLATION OCCURRED WITH THE SUBSEQUENT INITIATION OF THE STANDBY GAS TREATMENT SYSTEM (SBGTS). THE PCIS ISOLATION, WHICH ISOLATES THE PRIMARY CONTAINMENT ATMOSPHERE, WAS INITIATED BY A HIGH RADIATION SIGNAL FROM ONE OF THE TWO REFUEL FLOOR ZONE RADIATION DETECTORS (EIIS=DET). TWO DETECTORS, LOCATED ON THE REFUEL FLOOR, PROVIDE SIGNALS TO THE REFUEL FLOOR ZONE MONITORS: ONE ON THE EAST SIDE AND ONE ON THE WEST SIDE. THE ISOLATION OCCURRED, FOLLOWING A ROUTINE DETECTOR CALIBRATION CHECK, WHEN AN OPERATOR REMOVED THE BYPASS FROM THE TRIP FUNCTION AFTER ASSURANCE FROM THE RP TECHNICIAN THAT THE ALARM/TRIP HAD BEEN RESET WHEN, IN FACT, THE TRIP HAD NOT BEEN RESET. THE CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO A DEFECTIVE PROCEDURE. THE ISOLATION WAS PROMPTLY RESET AND SYSTEMS RETURNED TO NORMAL OPERATION. TECH SPEC DETECTOR OPERABILITY REQUIREMENTS WERE SATISFIED AT ALL TIMES.

[289] VERMONT YANKEE DOCKET 50-271 LER 87-011
PROCEDURAL INTERPRETATION LEADS TO DEGRADED CONDITION DURING REFUELING.
EVENT DATE: 082087 REPORT DATE: 091987 NSSS: GE TYPE: BWR

(NSIC 206420) ON AUGUST 18, 1987 AT 1107, WITH THE REACTOR MODE SWITCH IN REFUEL MODE, THE "A" STATION BATTERY (EIIS=EJ) WAS TAKEN OUT OF SERVICE FOR TESTING. ON AUGUST 20, 1987 AT 1640, THE CONTROL ROOM OPERATORS DETERMINED THAT WITH "A" BATTERY OUT, THE DIESEL DG-1-1B (EIIS=EK) COULD NOT BE CONSIDERED OPERABLE BECAUSE BATTERY BACKUP CONTROL POWER WAS NOT AVAILABLE. AT THIS TIME IT WAS DETERMINED THAT FROM AUGUST 18, 1987 (1107) TO AUGUST 20, 1987 (1640), THE REQUIREMENTS OF TECH SPEC SECTION 3.5.H.4 FOR MINIMUM SUBSYSTEMS REQUIRED FOR REFUELING, WERE NOT SATISFIED. ALL FUEL MOVEMENT WAS HALTED IMMEDIATELY AND THE MODE SWITCH PLACED IN SHUTDOWN. THE ROOT CAUSE OF THIS EVENT WAS PROCEDURAL ERROR IN INTERPRETATION OF THE TECH SPEC REQUIREMENTS DURING REFUEL MODE. CORRECTIVE ACTION INVOLVES FURTHER CLARIFICATION FOR REQUIRED EQUIPMENT DURING REFUELING AND OPERATION.

[290] VERMONT YANKEE DOCKET 50-271 LER 87-012
SCRAM DUE TO RADIOGRAPHERS FOLLOWING INADEQUATE PROCEDURES.
EVENT DATE: 082687 REPORT DATE: 092287 NSSS: GE TYPE: BWR

(NSIC 206511) AT APPROXIMATELY 0430 HRS ON 8/26/87, WHILE THE PLANT WAS IN A REFUELING CONDITION, AN UNEXPECTED HIGH MAIN STEAM LINE RADIATION SCRAM SIGNAL WAS RECEIVED. THE SIGNAL WAS THE RESULT OF A CONTRACTOR PERFORMING RADIOGRAPHY ON A WELD OF A PIPE NEAR THE MAIN STEAM LINE RADIATION DETECTORS. (EIIS=DET) THE SOURCE USED IN THE RADIOGRAPHY RAISED THE RADIATION LEVELS IN THE AREA OF THE DETECTORS ABOVE THE SCRAM SETPOINT. THE REACTOR PROTECTION SYSTEM (RPS) RESPONDED AS IT WAS DESIGNED. THE OPERATORS RESPONDED TO THE SCRAM AS REQUIRED BY PROCEDURE. THE CONTRACTORS PERFORMING THE INSPECTION WERE FOLLOWING AN APPROVED PROCEDURE, AND HAD APPROPRIATELY INFORMED THE CONTROL ROOM THAT THEY WERE ABOUT TO COMMENCE THE INSPECTION. THE PROCEDURE, HOWEVER, DID NOT WARN THE CONTROL

ROOM THAT A SCRAM WOULD OCCUR AS A RESULT OF THIS INSPECTION AND DID NOT ASSURE THAT THE DETECTORS WERE BYPASSED PRIOR TO PERFORMING THE INSPECTION.

[291] VOGTLE 1 DOCKET 50-424 LER 87-051
AFW FLOW TRANSMITTERS INOPERABLE DUE TO INADEQUATE INSTRUCTIONS AND PERSONNEL ERROR.
EVENT DATE: 062087 REPORT DATE: 090887 NSSS: SS TYPE: PWR
VENDOR: ROSEMOUNT ENGINEERING COMPANY

(NSIC 206277) ON MAY 4, 1987, A MAINTENANCE WORK ORDER (MWO) WAS WRITTEN TO CHECK THE AUXILIARY FEEDWATER (AFW) SYSTEM FLOW TRANSMITTER INPUTS TO THE PLANT SAFETY MONITORING SYSTEM. ON JUNE 20, 1987, PLANT TECHNICIANS DISCOVERED THAT TWO AFW SYSTEM FLOW TRANSMITTERS (1FT-15150 AND 1FT-15152) IN THE SAME TRAIN FAILED TO OPERATE PROPERLY. THE FOUR AFW FLOW TRANSMITTERS IN THE OTHER TRAIN WERE OPERABLE. A MWO ADDITION WAS WRITTEN TO REPLACE THE DEFECTIVE TRANSMITTERS. ON AUGUST 6, 1987, WORK SCHEDULERS, WHO WERE PROCESSING THE WORK ORDER TO REPLACE THE FLOW TRANSMITTERS, DISCOVERED THAT THE PLANT HAD NOT ENTERED INTO A LIMITING CONDITION OF OPERATION (LCO) AS REQUIRED BY TECHNICAL SPECIFICATIONS (T.S.). THE CONTROL ROOM WAS NOTIFIED, THE PLANT ENTERED INTO A LCO, THE FLOW TRANSMITTERS WERE REPLACED AND THE LCO WAS DISCONTINUED ON AUGUST 8, 1987. THE CAUSE OF THIS EVENT WAS INADEQUATE INSTRUCTIONS AND PERSONNEL ERROR. CONTROL ROOM AND OTHER PLANT PERSONNEL REVIEWING THE DEFICIENCY CARDS AND MWO'S INADVERTENTLY OVERLOOKED THE NEED TO INITIATE A T.S. LCO. CORRECTIVE ACTION INCLUDES GENERATING A LIST OF PLANT EQUIPMENT GOVERNED BY T.S. FOR USE BY PLANT OPERATORS IN PROMPTLY IDENTIFYING OF LCO'S AND CHANGES TO PLANT INSTRUCTIONS.

[292] VOGTLE 1 DOCKET 50-424 LER 87-049
ENTRY INTO LCO 3.0.3 DUE TO INOPERABLE ESF ROOM COOLER CHILLER TRAINS.
EVENT DATE: 072287 REPORT DATE: 082187 NSSS: SS TYPE: PWR
VENDOR: BARKSDALE CONTROLS DIV
TRANE COMPANY

(NSIC 205878) AT ./30 CDT ON JULY 22, 1987, WITH UNIT 1 IN MODE 1 AT 100% REACTOR POWER, TECHNICAL SPECIFICATION (T.S.) LIMITING CONDITION FOR OPERATION (LCO) 3.0.3 WAS ENTERED BECAUSE A TEMPERATURE SWITCH FAILED ON THE TRAIN B ENGINEERED SAFETY FEATURE (ESF) CHILLER WHILE A TRAIN A ESF ROOM COOLER UNIT WAS OUT OF SERVICE FOR PREVENTIVE MAINTENANCE. SINCE THIS RESULTED IN PORTIONS OF BOTH ESF CHILLER TRAINS BEING INOPERABLE, A CONDITION EXISTED WHICH WAS NOT PROVIDED FOR IN THE ACTION REQUIREMENTS OF T.S. LCO 3.7.11 AND ENTRY INTO LCO 3.0.3. WAS REQUIRED. THE TRAIN A COOLER UNIT WAS FUNCTIONALLY TESTED TO PROVE OPERABILITY AND WAS DECLARED OPERABLE. LCO 3.0.3 WAS EXITED AT 0815 CDT ON JULY 22, 1987. THE TEMPERATURE SWITCH ON THE TRAIN B ESF CHILLER WAS REPLACED AND THE B CHILLER WAS DECLARED OPERABLE AT 0459 CDT ON JULY 23, 1987.

[293] VOGTLE 1 DOCKET 50-424 LER 87-050
REACTOR TRIP CAUSED BY INSTRUMENT TECHNICIAN'S ERROR.
EVENT DATE: 072887 REPORT DATE: 082787 NSSS: SS TYPE: PWR

(NSIC 206276) ON JULY 28, 1987, AT APPROXIMATELY 0941 CDT WITH THE UNIT IN MODE 1 AND AT 100% OF RATED THERMAL POWER, AN AUTOMATIC REACTOR TRIP OCCURRED AS A RESULT OF THE EMERGENCY TRIP HYDRAULIC FLUID PRESSURE TRANSMITTERS SENSING A LOW PRESSURE AND GENERATING A REACTOR TRIP SIGNAL. THIS REACTOR TRIP WAS CAUSED BY PERSONNEL ERROR. AN INSTRUMENT TECHNICIAN EMPLOYED AN IMPROPER TECHNIQUE FOR MAKING TEST CONNECTIONS. CONTRIBUTING TO THIS EVENT WAS AN INADEQUATE ENGINEERING REVIEW BY THE RESPONSIBLE SYSTEM ENGINEER. CORRECTIVE ACTIONS INCLUDED THE DISCIPLINING OF THE INSTRUMENT TECHNICIAN, HIS FOREMAN AND THE SYSTEM ENGINEER. SHOP MEETINGS WERE HELD TO DISCUSS THIS EVENT WITH INSTRUMENT

TECHNICIANS. ENGINEERING SUPPORT PERSONNEL RECEIVED EXTENSIVE COUNSELING CONCERNING THIS EVENT.

[294] VOGTLE 1 DOCKET 50-424 LER 87-052
INADVERTENT CONTAINMENT VENTILATION ISOLATION OCCURRED DURING SOURCE CHECK OF RADIATION MONITOR.
EVENT DATE: 080987 REPORT DATE: 090887 NSSS: SS TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 206278) AT 1023 CDT ON AUGUST 9, 1987, WITH UNIT 1 IN MODE 1 AT 90% REACTOR POWER, A CONTAINMENT VENTILATION ISOLATION (CVI) OCCURRED DURING A MONTHLY SOURCE CHECK OF THE CONTAINMENT VENT EFFLUENT RADIOGAS MONITOR (1RE-2565C). A CHEMISTRY TECHNICIAN ATTEMPTED TO CONVERT THE DISPLAY FOR 1RE-2565C TO COUNTS PER MINUTE (CPM) AFTER FAILING TO OBTAIN AN OBSERVABLE RESPONSE WHEN THE SOURCE CHECK WAS REMOTELY PERFORMED. IN CONVERTING THE DISPLAY TO CPM, THE TECHNICIAN INCORRECTLY REVISED THE HIGH ALARM SETPOINT, RESULTING IN THE CVI. CONTROL ROOM OPERATORS VERIFIED THAT A HIGH ALARM CONDITION DID NOT EXIST AND RESET THE CVI AT 1035 CDT. SUBSEQUENTLY, IT WAS IDENTIFIED THAT THE CHECK SOURCE ACTUATOR FOR 1RE- 2565C HAD FAILED (IT HAS SINCE BEEN REPAIRED). DUE TO PROBLEMS WITH THE CHECK SOURCE ACTUATORS AT PLANT VOGTLE, WESTINGHOUSE IS REVISING THEIR ACTUATOR DESIGN AND A NEW PROTOTYPE IS EXPECTED SHORTLY. THE CVI WAS CAUSED BY PROCEDURAL AND TRAINING INADEQUACIES. TO PREVENT RECURRENCE, APPROPRIATE PROCEDURES WILL BE REVISED BY NOVEMBER 15, 1987. SEVERAL TECHNICIANS WHO WORK ON THE RADIATION MONITORS HAVE RECENTLY COMPLETED ADDITIONAL TRAINING.

[295] VOGTLE 1 DOCKET 50-424 LER 87-053
PERSONNEL ERROR LEADS TO EXCEEDING TECHNICAL SPECIFICATION SURVEILLANCE TIME LIMIT.
EVENT DATE: 081887 REPORT DATE: 091787 NSSS: SS TYPE: PWR

(NSIC 206093) DURING THE PERIOD FROM AUGUST 18, 1987, TO SEPTEMBER 2, 1987, WITH THE PLANT OPERATING IN MODE 1 AND AT 100% OF RATED THERMAL POWER, SURVEILLANCE INTERVALS FOR MEASURING THE STROKE TIMES OF THIRTEEN (13) VALVES REQUIRED TO BE TESTED BY ASME SECTION XI EXCEEDED THE REQUIRED TECHNICAL SPECIFICATION (T.S.) TIME REQUIREMENTS BY TIME INTERVALS RANGING FROM SIX (6) TO SEVENTEEN (17) DAYS. THIS EVENT WAS DISCOVERED BY ROUTINE DATA REVIEWS CONDUCTED BY AN INSERVICE TEST ENGINEER. SUBSEQUENTLY ALL MISSED SURVEILLANCES WERE PERFORMED AND ALL VALVES WERE VERIFIED TO BE WITHIN THEIR SPECIFIED MAXIMUM STROKE TIME LIMIT. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR INVOLVING TWO (2) REGULATORY COMPLIANCE SPECIALISTS. THE DUE DATES FOR THE QUARTERLY VALVE SURVEILLANCES WERE CHANGED WITHOUT REALIZING THE REQUIRED MONTHLY SURVEILLANCE TIME LIMITS WOULD BE EXCEEDED. CORRECTIVE ACTION INCLUDED COUNSELING THE INDIVIDUALS INVOLVED.

[296] VOGTLE 1 DOCKET 50-424 LER 87-054
CONTAINMENT HYDROGEN LEVEL INDICATION INOPERABLE DUE TO PERSONNEL ERROR.
EVENT DATE: 082487 REPORT DATE: 092387 NSSS: SS TYPE: PWR

(NSIC 206509) ON AUGUST 24, 1987, AT APPROXIMATELY 0945 CDT WITH THE UNIT IN MODE 1, AN INSTRUMENT AND CONTROL (I&C) TECHNICIAN DISCOVERED THAT THE CHANNEL A CONTAINMENT HYDROGEN MONITOR CONTROL ROOM INDICATION AND ALARM WAS INOPERABLE. FURTHER INVESTIGATION REVEALED THAT THE CHANNEL A CONTROL ROOM HYDROGEN INDICATION AND HIGH HYDROGEN ALARM HAD BEEN INOPERABLE FOR APPROXIMATELY 21 DAYS, WHICH EXCEEDED THE ACTION STATEMENT REQUIREMENTS. THIS DID NOT MEET THE UNIT 1 TECHNICAL SPECIFICATION REQUIREMENT FOR MODE 1. THE CONTROL ROOM CHANNEL A CONTAINMENT HYDROGEN MONITOR INDICATION AND ALARM WAS RENDERED INOPERABLE DUE TO REVERSAL OF THE SIGNAL WIRE LEADS AT THE CABINET. THIS WAS CAUSED BY PERSONNEL ERROR AND RESULTED IN NONCOMPLIANCE WITH THE SURVEILLANCE PROCEDURE FOR THE

INSTRUMENT. CORRECTIVE ACTION INCLUDED COUNSELING OF INVOLVED PERSONNEL AND CONDUCTING I&C DEPARTMENT TRAINING ON PROCEDURE COMPLIANCE.

[297] WATERFORD 3 DOCKET 50-382 LER 87-014 REV 01
UPDATE ON FIRE HOSE HOUSE AND COMPUTER HALON SYSTEM SURVEILLANCE INTERVALS EXCEEDED DUE TO PERSONNEL ERROR.
EVENT DATE: 050487 REPORT DATE: 092887 NSSS: CE TYPE: PWR

(NSIC 206547) AT 1600 ON MAY 4, 1987, WATERFORD STEAM ELECTRIC STATION UNIT 3 WAS AT 100% POWER. DURING A REVIEW OF TECHNICAL SPECIFICATION (TS) SURVEILLANCE RECORDS, THE TS COORDINATOR DISCOVERED THAT AN INSPECTION OF FIRE HYDRANT HOSE HOUSES WAS COMPLETED 4 DAYS LATER THAN TS REQUIREMENTS 4.7.10.5.A AND 4.0.2. ALLOW. ON AUGUST 28, 1987, A SIMILAR PROBLEM WAS DISCOVERED IN THAT A SURVEILLANCE OF THE COMPUTER ROOM HALON SYSTEM WAS COMPLETED 8 DAYS LATER THAN TS REQUIREMENTS ALLOW. THE ROOT CAUSE OF THESE EVENTS WAS THAT THE AUTOMATED SCHEDULING SYSTEM DID NOT ADJUST DUE DATES FOR SURVEILLANCES WHICH WERE COMPLETED EARLY, AND THE TS COORDINATOR FAILED TO MANUALLY RESCHEDULE THE SURVEILLANCE AS REQUIRED BY PLANT PROCEDURE. THIS TASK IS NOW HANDLED BY THE STATION INFORMATION MANAGEMENT SYSTEM WHICH HANDLES PLANNING AND SCHEDULING REQUIREMENTS FOR TS SURVEILLANCES. THIS PROGRAMMED METHOD OF SCHEDULING WILL RECOGNIZE AND ACCOUNT FOR EARLY COMPLETIONS, THEREBY PREVENTING RECURRENCE OF THIS SITUATION. SINCE THE FIRE HYDRANT HOSE HOUSE AND THE COMPUTER ROOM HALON SYSTEM SURVEILLANCES WERE COMPLETED AND NO DISCREPANCIES WERE FOUND, THERE IS A HIGH LEVEL OF CONFIDENCE THAT THE AFFECTED FIRE DETECTION AND SUPPRESSION SYSTEMS WOULD HAVE FUNCTIONED PROPERLY HAD A FIRE OCCURRED. THERE WAS THEREFORE NO EFFECT ON PLANT SAFETY AS A RESULT OF THIS EVENT.

[298] WATERFORD 3 DOCKET 50-382 LER 87-021
FIRE SEAL MISSING DUE TO ERROR IN CONSTRUCTION DOCUMENTATION.
EVENT DATE: 080787 REPORT DATE: 090887 NSSS: CE TYPE: PWR

(NSIC 206076) AT 1141 HOURS ON AUGUST 7, 1987, WATERFORD STEAM ELECTRIC STATION UNIT 3 WAS OPERATING AT 100% POWER WHEN OPERATIONS PERSONNEL, PERFORMING A ROUTINE PLANT INSPECTION, DISCOVERED THAT THE FIRE SEAL FOR PENETRATION VI A0126 WAS MISSING. THE MISSING FIRE SEAL IS LOCATED ON THE +46 FOOT ELEVATION OF THE REACTOR AUXILIARY BUILDING WHERE A 4 INCH VENT LINE FOR THE B EMERGENCY DIESEL GENERATOR PENETRATES THE FLOOR THROUGH AN 8 INCH SLEEVE. A FIRE WATCH WAS PROMPTLY ESTABLISHED IN ACCORDANCE WITH TECHNICAL SPECIFICATION 3.7.11. IT IS PROBABLE THAT THIS CONDITION EXISTED SINCE PLANT STARTUP, THEREFORE THE PLANT WAS IN A CONDITION PROHIBITED BY TECHNICAL SPECIFICATION 3.7.11 BETWEEN DECEMBER 18, 1984 AND AUGUST 7, 1987. FURTHER INVESTIGATION REVEALED THAT PENETRATION SEAL VI A0126 WAS INADVERTENTLY DELETED FROM THE PENETRATION SEAL LIST IN THE SPRING OF 1984, PRIOR TO THE SYSTEM TURNOVER WALKDOWNS AND DEVELOPMENT OF SURVEILLANCE PROCEDURES. THE SEAL THEREFORE WAS NEVER INDICATED AS BEING REQUIRED FOR INSPECTION IN THE CURRENT "PENETRATION TABLE" NOR CONTAINED IN PROCEDURE ME-3-006, "FIRE BARRIER PENETRATION SEALS". THE SEAL IS EXPECTED TO BE REPAIRED BY SEPTEMBER 30, 1987 AND STATION MODIFICATION 2001 HAS BEEN ISSUED TO REVISE THE APPROPRIATE DOCUMENTS.

[299] WATERFORD 3 DOCKET 50-382 LER 87-022
ESF VENTILATION ACTUATIONS DUE TO SHORTED RELAY.
EVENT DATE: 081487 REPORT DATE: 091487 NSSS: CE TYPE: PWR
VENDOR: ELECTRO SWITCH CORP.
WESTINGHOUSE ELECTRIC CORP.

(NSIC 206077) AT 1520 HOURS AUGUST 14, 1987, WATERFORD STEAM ELECTRIC STATION UNIT 3 WAS OPERATING AT APPROXIMATELY 100% POWER WHEN A 120 VAC CONTROL RELAY SHORTED TO GROUND AND TRIPPED A CONTROL POWER SUPPLY BREAKER. THIS RESULTED IN

AN AUTOMATIC START OF THE B TRAIN CONTROL ROOM AND FUEL HANDLING BUILDING EMERGENCY FILTRATION UNITS, A TRIP OF THE B ESSENTIAL CHILLER, AND ISOLATION OF THE B TRAIN SUPPLY VALVE TO THE NON-SAFETY COMPONENT COOLING WATER HEADER. THE AFFECTED EQUIPMENT WAS RETURNED TO NORMAL, APPROPRIATE ACTION REQUIREMENTS WERE ENTERED FOR THE EQUIPMENT AFFECTED, AND AN INVESTIGATION WAS BEGUN. AT 1622 HOURS A 120 VAC BREAKER WAS FOUND OPEN AND WAS RECLOSED. THE BREAKER TRIPPED IMMEDIATELY AND THE SAME EQUIPMENT ACTUATIONS OCCURRED. THE FAILED RELAY AND AN ASSOCIATED CONTROL CIRCUIT CARD WERE REPLACED. THE FAILURE WAS ATTRIBUTED TO MECHANICAL BINDING WHICH OVERHEATED A RELAY OPERATING COIL. AN ENGINEERING EVALUATION OF THE RELIABILITY OF THIS MODEL RELAY IS UNDERWAY. SINCE THE AFFECTED EMERGENCY FILTRATION UNITS ARE CLASSIFIED AS ENGINEERED SAFETY FEATURES (ESF) VENTILATION SYSTEMS, THIS EVENT IS REPORTED AS AN AUTOMATIC ESF ACTUATION. ALL AFFECTED EQUIPMENT WAS DECLARED OPERABLE AT 0633 HOURS AUGUST 16, 1987. THERE WAS NO EFFECT ON PLANT POWER OR OTHER SYSTEMS REQUIRED FOR SAFETY. THERE WERE THEREFORE NO SAFETY CONSEQUENCES AS A RESULT OF THIS EVENT.

[300] WATERFORD 3 DOCKET 50-382 LER 87-023
INVALID CONDENSER VACUUM PUMP SAMPLES DUE TO LOSS OF DEMISTER LOOP SEAL.
EVENT DATE: 081487 REPORT DATE: 091487 NSSS: CE TYPE: PWR

(NSIC 206078) AT 1340 HOURS ON AUGUST 13, 1987, WATERFORD STEAM ELECTRIC STATION UNIT 3 WAS OPERATING AT 100% POWER WHEN OPERATIONS PERSONNEL DECLARED THE CONDENSER VACUUM PUMP NOBLE GAS RADIATION MONITOR (NGRM) TO BE OUT OF SERVICE. IN ACCORDANCE WITH TECHNICAL SPECIFICATION 3.3.3.11 ACTION 37, HEALTH PHYSICS PERSONNEL BEGAN TAKING TWELVE HOUR SAMPLES FOR GROSS ACTIVITY ANALYSIS. DUE TO CONDENSATION FROM CONDENSER VACUUM PUMP (CVP) STEAM CARRYOVER BLOCKING NGRM SAMPLE LINES, THE LOOP SEAL IN THE NGRM SAMPLE DRYER DRAIN WAS DRAWN INTO THE SAMPLE PUMP. THIS RESULTED IN THE MONITOR SAMPLING TURBINE BUILDING ATMOSPHERE RATHER THAN CVP EFFLUENT. SINCE THIS RESULTED IN INVALID GRAB SAMPLES OF CVP EFFLUENT GAS, THE PLANT WAS IN A CONDITION PROHIBITED BY TECHNICAL SPECIFICATIONS FOR APPROXIMATELY 17 HOURS. THE CONDITION WAS DISCOVERED AND CORRECTED BY 2120 HOURS ON AUGUST 14, 1987. A TEMPORARY SAMPLING LOCATION AND INSTRUCTION WERE PROVIDED UPON DISCOVERY OF THE INVALID SAMPLING POINT. ON AUGUST 17, 1987, A METHOD WAS PROVIDED TO OBTAIN SAMPLES AT THE NORMAL LOCATION. HEALTH PHYSICS SAMPLING PROCEDURES ARE BEING REVISED AND HEALTH PHYSICS PERSONNEL HAVE BEEN NOTIFIED OF THE EVENT AND CORRECTIVE ACTIONS BY LETTER. SINCE HEALTH PHYSICS PERSONNEL HAVE A METHOD TO SAMPLE CVP EFFLUENT GAS, THE INOPERABILITY OF THE NGRM DOES NOT POSE A SAFETY CONCERN.

[301] WOLF CREEK 1 DOCKET 50-482 LER 87-030
POTENTIAL TRANSFORMER FAILURE CAUSES PARTIAL LOSS OF OFFSITE POWER AND REACTOR TRIP AND SUBSEQUENT SHUT DOWN SEQUENCER ACTUATION DURING RESTORATION.
EVENT DATE: 072087 REPORT DATE: 081987 NSSS: WE TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 205882) ON JULY 20, 1987, AT APPROXIMATELY 2302 CDT, A PARTIAL LOSS OF OFFSITE POWER OCCURRED WHEN A SWITCHYARD POTENTIAL TRANSFORMER FAILED. THE LOSS OF POWER RESULTED IN A LOSS OF POWER TO THE MAIN FEEDWATER PUMP 'A' (MFP) CONTROLLERS AND A SUBSEQUENT COASTDOWN OF THE MFP 'A' TURBINE. AN EMERGENCY DIESEL GENERATOR START AND SHUT DOWN SEQUENCER ACTUATION OCCURRED, WHICH REENERGIZED THE MFP CONTROLLERS AFTER A MFP COASTDOWN OF APPROXIMATELY 10 SECONDS. THE RESULTANT REENERGIZATION OF THE MFP CONTROLLERS CAUSED AN MFP TRIP DUE TO TURBINE OVERSPEED. THE PARTIAL LOSS OF FEEDWATER FLOW RESULTED IN A REACTOR TRIP ON LO-LO STEAM GENERATOR LEVEL. A PLANT MODIFICATION WILL BE IMPLEMENTED TO PROVIDE A MORE RELIABLE SOURCE OF POWER FOR THE MFP CONTROLLERS WHEN A PARTIAL LOSS OF OFFSITE POWER OCCURS. THIS WOULD PREVENT LOSS OF CONTROL POWER TO THE MFP TURBINE AFTER A PARTIAL LOSS OF OFFSITE POWER. DURING RECOVERY, ON JULY 21, 1987, AT APPROXIMATELY 1442 CDT, FOLLOWING REPLACEMENT OF THE POTENTIAL TRANSFORMER, A SHUT DOWN SEQUENCER ACTUATION OCCURRED AS A RESULT OF A

PERSONNEL ERROR WHILE TRANSFERRING LOADS BACK TO THE SWITCHYARD TRANSFORMER. THE PROCEDURE HAS BEEN ENHANCED TO PREVENT SIMILAR OCCURRENCES.

[302] WOLF CREEK 1 DOCKET 50-482 LER 87-036
INADVERTENT RELEASE OF A SECONDARY LIQUID WASTE MONITOR TANK WITHOUT PRIOR SAMPLING CAUSED BY PROCEDURE INADEQUACY.
EVENT DATE: 072687 REPORT DATE: 100287 NSSS: WE TYPE: PWR

(NSIC 206605) ON JULY 26, 1987, DURING AN AUTHORIZED RELEASE OF SECONDARY LIQUID WASTE MONITOR TANK 'B' (SLWMT), THE CONTENTS OF SLWMT 'A' WERE INADVERTENTLY RELEASED. THIS OCCURRENCE WAS DISCOVERED ON JULY 27, 1987, WHILE OBTAINING ROUTINE TANK LOGS, WHEN IT WAS NOTED THAT SLWMT 'A' LEVEL HAD DECREASED FROM SIX PERCENT TO ZERO PERCENT DURING THE TIME PERIOD OF THE SLWMT 'B' RELEASE. FOLLOWING THIS DISCOVERY, THE TANK LINEUPS WERE VERIFIED TO BE PROCEDURALLY CORRECT. CALCULATIONS PERFORMED SUBSEQUENT TO THE EVENT CONFIRMED THAT NO RELEASE LIMITS HAD BEEN EXCEEDED. DURING A REVIEW OF THIS EVENT ON SEPTEMBER 3, 1987, IT WAS DETERMINED THAT THIS EVENT CONSTITUTED A TECHNICAL SPECIFICATION VIOLATION AS SLWMT 'A' WAS NOT SAMPLED PRIOR TO THE RELEASE. THIS EVENT HAS BEEN ATTRIBUTED TO A PROCEDURAL INADEQUACY. THE APPLICABLE PROCEDURE DID NOT ADDRESS COMPLETE ISOLATION OF THE INSERVICE SLWMT WHILE THE OTHER WAS BEING RELEASED. THE PROCEDURE HAS BEEN REVISED.

[303] WOLF CREEK 1 DOCKET 50-482 LER 87-031
SPENT FUEL POOL HEAT EXCHANGER ROOM DOORS NOT THREE-HOUR FIRE RATED DUE TO INSTALLATION NOT IN ACCORDANCE WITH DESIGN REQUIREMENTS.
EVENT DATE: 081187 REPORT DATE: 091087 NSSS: WE TYPE: PWR
VENDOR: MOSLER SAFE CO

(NSIC 206194) ON AUGUST 11, 1987, IT WAS DETERMINED THAT A VIOLATION OF TECHNICAL SPECIFICATION 3.7.11, WHICH REQUIRES THAT ALL FIRE BARRIER PENETRATIONS SEPARATING SAFETY RELATED FIRE AREAS BE OPERABLE, HAD OCCURRED BECAUSE THE FUEL POOL COOLING HEAT EXCHANGER ROOM SHIELDING DOORS COULD NOT BE CERTIFIED AS THREE-HOUR FIRE RATED BARRIERS. THIS CONDITION WAS DISCOVERED BY ENGINEERING PERSONNEL DURING EVALUATION OF A PROBLEM WITH THE LATCH BOLT AND KEEPER OF ONE OF THE DOORS. AN HOURLY FIRE WATCH PATROL OF THE DOORS HAD BEEN IN EFFECT SINCE APRIL, 1987, TO FACILITATE OTHER INSPECTION ACTIVITIES. THESE TWO DOORS ARE THE ONLY SHIELDING DOORS IN THE PLANT INSTALLED IN THREE-HOUR FIRE RATED WALLS. THE ROOT CAUSE OF THIS DISCREPANCY COULD NOT BE DETERMINED. A DESIGN CHANGE WILL BE IMPLEMENTED TO PROVIDE FIRE RATED DOORS TO BE INSTALLED IN SERIES WITH THE EXISTING DOORS AND ALSO TO ENHANCE THE DOOR CLOSURE MECHANISM.

[304] WOLF CREEK 1 DOCKET 50-482 LER 87-033
INOPERABLE CONTAINMENT ISOLATION VALVE CAUSED BY INCOMPLETE RETESTING FOLLOWING MAINTENANCE ACTIVITY.
EVENT DATE: 081387 REPORT DATE: 091487 NSSS: WE TYPE: PWR

(NSIC 206195) ON JULY 12, 1987, THE POSITION INDICATION/TRAVEL LIMIT SWITCH ON CONTAINMENT ISOLATION VALVE EF HV-46 WAS ADJUSTED TO CORRECT A POSITION INDICATION ANOMALY UNDER A TROUBLESHOOTING WORK REQUEST. DURING THE NORMAL CLOSEOUT REVIEW OF THE W[illegible]EQUEST AUTHORIZING THIS ACTIVITY, ON 8/13/87, IT WAS DETERMINED THAT A LOCAL [illegible] RATE T[illegible]ST (LLRT) SHOULD HAVE BEEN PERFORMED TO DEMONSTRATE VALVE OPE[illegible]Y PRIO[illegible]STORING THE VALVE TO SERVICE. A TYPE C LLRT WAS PERFORMED [illegible] VALVE ON [illegible] AT APPROXIMATELY 1930 CDT. A QUANTIFICATION OF T[illegible]D THAT THE VALVE LEAKAGE WAS SUFFICIENT TO CAUS[illegible]ATE OF ALL TYPE B AND C LLRTS TO EXCEED THE TECHNICAL SPEC[illegible] A. THE LIMIT SWITCH ON THE VALVE WAS THEN RE-ADJUSTED. [illegible]RMED, WITH SATISFACTORY RESULTS, ON 8/14/87, AT APPROXIMAT[illegible]VIEW OF THIS EVENT CONCLUDED THAT THE

VALVE WAS INOPERABLE BETWEEN 7/12 AND 8/14, WHICH IS CONTRARY TO TECH SPEC 3.6.3, WHICH REQUIRES THAT ALL CONTAINMENT ISOLATION VALVES BE OPERABLE. THIS EVENT OCCURRED AS A RESULT OF INCOMPLETE PRETEST INSTRUCTIONS ON THE TROUBLESHOOTING WORK REQUEST. THE PROCEDURE FOR TROUBLESHOOTING VALVE OPERATORS HAS BEEN REVISED TO INCLUDE A PREREQUISITE FOR DETERMINING APPROPRIATE PRETEST REQUIREMENTS, WITH SPECIAL ATTENTION TO THE NECESSITY OF PERFORMING 'AS-FOUND' AND 'AS-LEFT' LLRT'S.

[305] WOLF CREEK 1 DOCKET 50-482 LER 87-034
FAILURE TO EFFECTIVELY COMMUNICATE ALLOWS AN OPEN DOOR CREATING A CONTROL ROOM PRESSURE BOUNDARY BREACH.
EVENT DATE: 082087 REPORT DATE: 092187 NSSS: WE TYPE: PWR

(NSIC 206199) ON AUGUST 20, 1987, AT APPROXIMATELY 1330 CDT, AN NRC RESIDENT INSPECTOR FOUND DOOR 36171, WHICH SEPARATES THE CONTROL ROOM FROM AN ELECTRICAL CHASE AREA, PROPPED OPEN FOR REWORK OF PENETRATIONS BY FACILITIES AND MODIFICATION (F&M) PERSONNEL. THE SHIFT SUPERVISOR WAS AWARE OF THE REWORK ACTIVITY, BUT WAS UNAWARE THAT IT REQUIRED THE DOOR TO BE PROPPED OPEN. THE SHIFT SUPERVISOR PROMPTLY TOOK COMPENSATORY MEASURES UPON NOTIFICATION THAT THE DOOR WAS PROPPED OPEN. WITH THIS DOOR OPEN, IT WAS CONCLUDED THAT POSITIVE PRESSURE COULD NOT BE MAINTAINED IN THE CONTROL ROOM WHICH IS CONTRARY TO TECHNICAL SPECIFICATIONS (T/S) AND SHOULD HAVE RESULTED IN T/S 3.0.3 ENTRY. THIS EVENT HAS BEEN ATTRIBUTED TO A FAILURE TO EFFECTIVELY COMMUNICATE BETWEEN THE SHIFT SUPERVISORS INVOLVED AND THE F&M PERSONNEL PERFORMING THE WORK ACTIVITY. A PERMANENT SIGN HAS BEEN POSTED ON THIS DOOR TO CLEARLY IDENTIFY ITS CONTROL ROOM PRESSURE BOUNDARY FUNCTION. F&M ENGINEERS HAVE BEEN INSTRUCTED TO ADD PRECAUTIONARY STATEMENTS INTO THEIR WORK PACKAGES TO EMPHASIZE PRESSURE BOUNDARY CONSIDERATIONS. IN ADDITION, A CHANGE HAS BEEN MADE TO THE CONTROL ROOM VENTILATION ISOLATION SIGNAL ALARM RESPONSE PROCEDURE TO CLOSE THE DOORS IF THEY ARE OPEN.

[306] WPPSS 2 DOCKET 50-397 LER 87-025
ENGINEERED SAFETY FEATURE ISOLATIONS AND ACTUATIONS CAUSED BY REACTOR PROTECTION SYSTEM EQUIPMENT FAILURE.
EVENT DATE: 080887 REPORT DATE: 090387 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 206264) 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 ASSEMBLY (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 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, 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) ISOLATION AND STANDBY GAS TREATMENT (SGT) SYSTEM AND CONTROL ROOM 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. THE ROOT CAUSE OF THE SPURIOUS TRIP HAS NOT BEEN DETERMINED. THE SUPPLY SYSTEM SENT THE BREAKER TO GENERAL ELECTRIC (BREAKER MANUFACTURER) FOR ANALYSIS. THE ROOT CAUSE INFORMATION WILL BE PROVIDED IN A SUPPLEMENTAL REPORT.

[307] WPPSS 2 DOCKET 50-397 LER 87-027
TECHNICAL SPECIFICATION SURVEILLANCE "APRM CALIBRATION FROM HEAT BALANCE" NOT PERFORMED WITHIN TIME LIMITS DUE TO PERSONNEL ERROR.
EVENT DATE: 083187 REPORT DATE: 093087 NSSS: GE TYPE: BWR

(NSIC 206499) ON AUGUST 31, 1987 IT WAS DISCOVERED THAT THE WEEKLY "AVERAGE POWER RANGE MONITOR (APRM) CALIBRATION FROM HEAT BALANCE", HAD NOT BEEN COMPLETED WITHIN THE REQUIRED TIME PLUS 25% AS REQUIRED BY PLANT TECHNICAL SPECIFICATION TABLE 4.3.1.1-1.2. THE CALIBRATION WAS DUE TO BE PERFORMED ON AUGUST 25, 1987, WAS OVERDUE ON AUGUST 27, 1987, AND WAS NOT PERFORMED UNTIL SEPTEMBER 1, 1987. THE SURVEILLANCE PROCEDURE WAS PERFORMED THE NEXT DAY (SEPTEMBER 1, 1987 WHICH COINCIDED WITH THE NORMAL SURVEILLANCE SCHEDULE. THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR. A PLANT TECHNICAL STAFF SUPERVISOR FAILED TO ROUTE THE SURVEILLANCE MONITORING SYSTEM (SMS) COMPUTER CARD TO THE PLANT SHIFT ENGINEERS WHO PERFORM THE SURVEILLANCE. THE SMS COMPUTER CARD SERVES AS A REMINDER TO THE SHIFT ENGINEER TO PERFORM THE SURVEILLANCE. CORRECTIVE ACTIONS INCLUDED A REVIEW OF PLANT HISTORICAL DATA (THE TRANSIENT DATA ACQUISITION SYSTEM-TDAS-STORES A SET OF PLANT DATA ONCE EVERY MINUTE AND STORES THIS FOR TWO WEEKS BEFORE OVERWRITING WITH NEW DATA) WHICH CONCLUDED THAT FIVE (5) OF THE SIX (6) APRM CHANNELS WERE WITHIN THE ALLOWABLE TECHNICAL SPECIFICATION TOLERANCES FROM THE TIME THE SURVEILLANCE WAS DUE TO WHEN IT WAS PERFORMED ON SEPTEMBER 1, 1987. ONLY FOUR (4) OF THE SIX (6) APRM CHANNELS (TWO (2) OF THREE (3) PER TRIP SYSTEM) ARE REQUIRED TO BE OPERABLE.

[308] WPPSS 2 DOCKET 50-397 LER 87-026
DIESEL/GENERATOR STORED DIESEL FUEL OIL LESS THAN TECHNICAL SPECIFICATION LIMIT DUE TO INADEQUATE SETPOINT PROCESS CONTROL.
EVENT DATE: 091587 REPORT DATE: 092587 NSSS: GE TYPE: BWR

(NSIC 206513) ON 8/24/87 DURING AN NRC SAFETY SYSTEM FUNCTIONAL INSPECTION (SSFI) IT WAS DETERMINED THAT THE TABLE OF DIESEL FUEL (GALLONS) VERSUS TANK DIPSTICK LEVEL FOR EACH DIESEL OIL STORAGE TANK WAS IN ERROR. THERE ARE THREE DIESEL OIL STORAGE TANKS AT WNP-2, ONE ASSOCIATED WITH EACH DIESEL/GENERANTOR SET. ON SEPTEMBER 15, 1987 USING NEW TABLES BASED ON NEW GENERATION ENGINEERING CALCULATIONS, PLANT LOGS FROM JULY 1986 TO SEPTEMBER 1987 WERE REVIEWED AND IT WAS DETERMINED THAT THERE WERE FOUR INTERVALS DURING THIS PERIOD WHEN THE AMOUNT OF FUEL IN A STORAGE TANK WAS BELOW THE PLANT TECH SPEC LIMIT. THESE OCCURRENCES WERE FOR THE DIVISION 1 OR 2 DIESEL/GENERATOR SYSTEMS ONLY. THE HIGH PRESSURE CORE SPRAY DIESEL/GENERATOR STORAGE TANK WAS NEVER, FOR THE REVIEW PERIOD, BELOW ITS TECH SPEC LIMIT. CORRECTIVE ACTIONS INCLUDE: A TASK FORCE IS REVIEWING THE SETPOINT PROCESS AT WNP-2, NEW CALCULATIONS FOR USABLE TANK CAPACITY VERSUS DIPSTICK LEVEL WERE COMPLETED, THE OPERATION'S EQUIPMENT OPERATOR DAILY LOG SHEETS WERE REVISED WITH THE NEW TANK LEVEL CRITERIA, THE REVISED TABLES HAVE BEEN INCORPORATED INTO THE PLANT PROCEDURES, AND THE TANK LOW ALARM LEVEL SETPOINTS WILL BE CHANGED. THE CAUSE OF THIS EVENT WAS INADEQUATE PROCESS CONTROL OF PLANT SETPOINTS. ORIGINAL CALCULATIONS PROVIDED FOR AN ERROR MARGIN BUT

[309] ZION 1 DOCKET 50-295 LER 87-014
ESSENTIAL HEAT TRACE DEENERGIZED ON BORIC ACID SYSTEM DUE TO A POLICY MISUNDERSTANDING AND DEFICIENT PROCEDURE.
EVENT DATE: 080487 REPORT DATE: 090287 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: ZION 2 (PWR)

(NSIC 206301) ON AUGUST 4, 1987 AT 0845 HOURS, AN OPERATING EQUIPMENT ATTENDANT (B-MAN) DISCOVERED BOTH REDUNDANT CHANNELS OF HEAT TRACE DEENERGIZED ON A PORTION OF THE FLOWPATH BETWEEN THE OPERABLE BORIC ACID TANKS (BAT) AND THE SUCTION LINES FOR THE TWO UNIT 1 BORIC ACID TRANSFER PUMPS. UNIT 1 WAS AT 94.5% POWER, AND UNIT 2 WAS IN HOT STANDBY. THE HEAT TRACE HAD BEEN DEENERGIZED BY ELECTRICIANS (EM'S) WHO WERE REPLACING A PORTION OF THE CIRCUIT. THE B-MAN NOTIFIED THE SHIFT SUPERVISOR, WHO HAD THE EM'S RESTORE THE CIRCUIT. THE EVENT WAS CAUSED BY A MISUNDERSTANDING OF POLICY REGARDING BREAKER MANIPULATION AND A DEFICIENT EM PROCEDURE. AT APPROXIMATELY 0815 HOURS, THE EM'S OBTAINED LICENSED SHIFT FOREMAN APPROVAL TO START WORK ON THIS CIRCUIT. THE SHIFT FOREMAN WAS NOT AWARE THAT THE WORK WOULD INVOLVE DEENERGIZING BOTH REDUNDANT HEAT TRACE CHANNELS, AND THE EM

PROCEDURE DID NOT REQUIRE THAT AN OUT OF SERVICE CARD BE HUNG BEFORE DEENERGIZING THE CIRCUIT. SAFETY EFFECT WAS MINIMIZED BY THE SHORT DURATION (LESS THAN THREE HOURS) OF THE EVENT AND BY THE FACT THAT A TRANSFER PUMP WAS RUNNING, PROVIDING SUFFICIENT FLOW TO PREVENT BORIC ACID SOLIDIFICATION IN THE LINE. CORRECTIVE ACTION WILL INCLUDE REVISIONS TO THE EM PROCEDURES FOR WORK ON HEAT TRACE AND A REVIEW OF THE POLICY OF EM MANIPULATION OF HEAT TRACE BREAKERS DURING TROUBLE SHOOTING/REPAIR.

[310] ZION 1 DOCKET 50-295 LER 87-015
INOPERABLE RESIDUAL HEAT REMOVAL TRAIN DUE TO A PUMP SUCTION ISOLATION VALVE OUT OF SERVICE.
EVENT DATE: 082587 REPORT DATE: 092487 NSSS: WE TYPE: PWR

(NSIC 206516) ON 8/25/87 AT 1330 HOURS, THE 1B RESIDUAL HEAT REMOVAL PUMP SUCTION ISOLATION VALVE (1MOV-RH8700B) WAS TAKEN OUT OF SERVICE, OPEN, MAKING THE ZION UNIT 1 RESIDUAL HEAT REMOVAL TRAIN B INOPERABLE. DUE TO MISINTERPRETATION OF OPERABILITY REQUIREMENTS BY THE LICENSED SHIFT SUPERVISION, THE TECH SPEC REQUIRED SURVEILLANCES FOR CONTINUED UNIT OPERATION WERE NOT INITIATED UNTIL APPROXIMATELY TWENTY FIVE HOURS LATER. WITH UNIT 1 OPERATING AT 95% POWER, THE MISSED SURVEILLANCE PUT UNIT 1 IN A CONDITION PROHIBITED BY PLANT TECH SPECS. WHEN THE OVERSIGHT WAS RECOGNIZED, THE SURVEILLANCES WERE BEGUN. FOR FIVE DAYS PRIOR TO THE EVENT, VALVE 1MOV-RH8700B HAD BEEN THE SUBJECT OF A TECHNICAL STAFF INVESTIGATION INTO THE OCCASIONAL FAILURE OF THIS VALVE TO RESPOND TO A CONFIRMATORY OPEN SIGNAL COINCIDENT WITH A PUMP START. DURING THIS TIME THE VALVE REMAINED IN SERVICE AND OPEN AS REQUIRED BY PROCEDURE, EXCEPT DURING TEST STROKING AT THE TECHNICAL STAFF REQUEST. THIS TESTING AND THE NEED TO KEEP THE VALVE OPEN FOR NORMAL OPERATION LEAD THE LICENSED SHIFT SUPERVISION TO OVERLOOK THE POST-ACCIDENT VALVE FUNCTION WHEN IT WAS TAKEN OUT OF SERVICE FOR REPAIR.

[311] ZION 2 DOCKET 50-304 LER 87-007
AUTOSTART OF 2A AUX FEEDWATER PUMP DURING TEST DUE TO MISALIGNMENT OF RELAY CONTACT.
EVENT DATE: 082687 REPORT DATE: 092587 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELEC CORP.-NUCLEAR ENERGY SYS

(NSIC 206517) ON AUGUST 26, 1987 AT 17:15 HOURS, SECTION 10 OF PT-5 (REACTOR PROTECTION LOGIC: REACTOR AT NORMAL OPER. CONDITIONS OR AT HOT SHUTDOWN) WAS BEING PERFORMED FOR TRAIN B WHEN THE 2A AUX FEEDWATER PUMP (2A AFW PP) UNEXPECTEDLY AUTOSTARTED. TECH STAFF PERFORMED A CONTINUITY CHECK ON THE CIRCUIT AND CONCLUDED THAT THE AFP3-XB RELAY HAD AN OPEN CONTACT. ELECTRICIANS REPLACED THIS RELAY WITH A NEW RELAY. OPERATIONS REPEATED THE TEST SECTION ON AUGUST 28, 1987 AT 16:15 HOURS AND THE 2A AFW PP AUTOSTARTED AGAIN. THIS TIME, TECH STAFF FOUND THAT THE AFP1-XB RELAY HAD A MOVEABLE CONTACT WHICH HAD SHIFTED DOWN AND MISALIGNED SO THAT IT COULD NOT CLOSE; THIS RELAY WAS ALSO REPLACED. THE TEST WAS RE-RUN SUCCESSFULLY. THESE TWO OCCURRENCES WERE BOTH CAUSED BY A WESTINGHOUSE TYPE NBFD-66S RELAY. THE SAFETY SIGNIFICANCE OF THESE TWO EVENTS WAS MINIMAL BECAUSE THE RELAY CONTACT FAILURES WERE CONSERVATIVE. THE 2A AFW PP AUTOSTART CIRCUITRY IS A "DE-ENERGIZE TO ACTUATE" SYSTEM. ALL OTHER NBFD-66S RELAYS NOW IN SERVICE IN REACTOR PROTECTION WERE INSPECTED FOR MISALIGNED CONTACTS AND NONE WERE FOUND. FOR FUTURE RELAY INSTALLATIONS AND REPLACEMENT, ELECTRICAL MAINTENANCE WILL INCLUDE STEPS IN THEIR PROCEDURE SO THAT THE RELAY IS INSPECTED FOR DAMAGE AND CONTACT ALIGNMENT AS A PART OF THE BENCH TEST.

COMPONENT INDEX

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

COMPONENT INDEX

COMPONENT INDEX

SYSTEM INDEX

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

SYSTEM INDEX

SYSTEM INDEX

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KEYWORD INDEX

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

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

KEYWORD INDEX

VENDOR CODE INDEX

NRC FORM 335 (2-84) NRCM 1102, 3201, 3202

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BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE

1. REPORT NUMBER (Assigned by TIDC, add Vol. No., if any)	NUREG/CR-2000, Vol. 6, No. 12 ORNL/NSIC-200
2. TITLE AND SUBTITLE	Licensee Event Report (LER) Compilation for Month of December 1987
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5. AUTHOR(S)	
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12. SUPPLEMENTARY NOTES	

13. ABSTRACT (200 words or less)

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

14. DOCUMENT ANALYSIS – a. KEYWORDS/DESCRIPTORS	Licensee event report (LER) unusual occurrences
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