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Licensee Event Report (LER) Compilation

For month of September 1989

Oak Ridge National Laboratory

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

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Abstract

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

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

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[1] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 88-030
PIPING SUPPORT DISCREPANCIES CAUSED BY USE OF AN UNACCEPTABLE MODELING TECHNIQUE IN STRESS CALCULATIONS POTENTIALLY AFFECT SEISMIC CAPABILITIES OF EMERGENCY CORE COOLING SYSTEMS.
EVENT DATE: 062088 REPORT DATE: 072489 NSSS: BW TYPE: PWR

(NSIC 214939) ON 6/20/88, AN ENGINEERING REVIEW REVEALED THAT AN UNACCEPTABLE MODEL TERMINATION HAD BEEN USED IN PIPING STRESS CALCULATIONS FOR ORIGINAL PLANT CONSTRUCTION OF THE EMERGENCY CORE COOLING SYSTEMS (ECCS) SUPPLY PIPING FROM THE BORATED WATER STORAGE TANK, ADVERSELY AFFECTING THE DESIGN OF CERTAIN PIPING SUPPORTS. A PRELIMINARY EVALUATION CONCLUDED THAT A DETAILED OPERABILITY ANALYSIS WOULD MOST LIKELY SHOW THAT THE AFFECTED SYSTEMS WERE STILL OPERABLE. IN OCTOBER 1988 DURING A REFUELING OUTAGE THREE SUPPORTS WERE MODIFIED AS NECESSARY TO ENSURE OPERABILITY OF THE SUPPORTS. IN MAY 1989 A MORE DETAILED ENGINEERING EVALUATION WAS COMPLETED WHICH ASSESSED THE OPERABILITY OF THE AFFECTED SYSTEMS CONSIDERING THE AS-FOUND CONDITION OF THE SUPPORTS. AS THREE SUPPORTS WERE JUDGED TO HAVE BEEN INOPERABLE, THE AFFECTED SYSTEMS WERE CONSERVATIVELY CONSIDERED TO HAVE BEEN INOPERABLE EVEN THOUGH NO SPECIFIC ANALYSIS DEMONSTRATED THAT THE PIPING WOULD FAIL IF THE SUPPORTS WERE REMOVED. PLANT DESIGN BASES DO NOT REQUIRE THE USE OF ECCS FOR MITIGATING A SEISMIC EVENT. USE OF THIS MODELING TECHNIQUE IS CONSIDERED TO BE AN ISOLATED CASE. FIELD WALKDOWNS FOR AN ISOMETRIC DRAWING UPDATE PROJECT IN PROGRESS SHOULD HELP IDENTIFY FURTHER DISCREPANCIES.

[2] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 88-029
SEISMIC QUALIFICATION OF REACTOR COOLANT SYSTEM LETDOWN PIPING COMPROMISED DUE TO DEFICIENT PIPING SUPPORTS RESULTING FROM INADEQUATE WORK CONTROLS DURING INITIAL CONSTRUCTION.
EVENT DATE: 092388 REPORT DATE: 072489 NSSS: BW TYPE: PWR

(NSIC 214938) ON 9/23/88, WHILE IN A REFUELING OUTAGE, SEVERAL DISCREPANCIES WERE IDENTIFIED WITH RESPECT TO 7 PIPING SUPPORTS LOCATED ON THE REACTOR COOLANT SYSTEM (RCS) LETDOWN PIPING INSIDE CONTAINMENT. THE DISCREPANCIES INCLUDED LOOSE PARTS, EXCESSIVE GAPS BETWEEN PIPING AND SUPPORT MEMBERS, AND SUPPORTS WHICH WERE NOT INSTALLED AS INDICATED ON THE DESIGN DRAWINGS. ALSO, 5 OF THE DEFICIENT SUPPORTS, WHICH THE ORIGINAL STRESS ANALYSIS CALCULATIONS SHOWED AS ONE-WAY RESTRAINTS (VERTICAL ONLY), WERE INSTALLED AS TWO-WAY RESTRAINTS (HORIZONTAL AND VERTICAL). AN ENGINEERING EVALUATION DETERMINED THAT THE RESTRICTION IN LATERAL AND AXIAL PIPE MOVEMENT CAUSED BY THESE RESTRAINTS IMPOSED THERMALLY INDUCED STRESSES ON THE PIPING IN EXCESS OF ASME CODE ALLOWABLES WHICH COULD COMPROMISE THE PIPING SEISMIC QUALIFICATION. THE AFFECTED PIPING INCLUDES INSIDE CONTAINMENT ISOLATION VALVES CV-1214 AND CV-1216. THEY WERE CONSIDERED TO HAVE BEEN INOPERABLE WHILE THE CONDITION EXISTED. SINCE THE OUTSIDE CONTAINMENT ISOLATION VALVE ON THE SAME LINE HAD BEEN PREVIOUSLY DETERMINED TO BE NOT SEISMIC (LER 50-313/88-027), THE POSSIBILITY EXISTED THAT A DESIGN BASIS SEISMIC EVENT COULD HAVE CAUSED FAILURE OF BOTH THE INSIDE AND OUTSIDE VALVES AND RESULTED IN A BREACH OF CONTAINMENT INTEGRITY AS WELL AS NON-ISOLABLE LEAK.

[3] ARKANSAS NUCLEAR 1 DOCKET 50-313 LER 89-026
FAILURE TO PERFORM SURVEILLANCE OF FIRE BARRIERS DUE TO PERSONNEL ERROR.
EVENT DATE: 062789 REPORT DATE: 072789 NSSS: BW TYPE: PWR

(NSIC 214922) ON 6/27/89, FOUR TECH SPEC FIRE BARRIERS WERE FOUND NOT TO HAVE BEEN INCLUDED IN THE FIRE BARRIER PENETRATION LOG AND SUBSEQUENTLY OMITTED FROM THE FIRE BARRIER VISUAL INSPECTION PROCEDURE DEVELOPED FROM THE LOG. THE OMISSION APPARENTLY OCCURRED WHEN THE LOG WAS INITIALLY COMPILED AS A PERMANENT PLANT DOCUMENT AFTER THE FIRE BARRIER UPGRADE TO 10CFR50, APPENDIX R CRITERIA. CONSEQUENTLY, AFTER THE INITIAL BARRIER INSPECTION IN NOVEMBER 1983, THE BARRIERS HAD NOT BEEN INSPECTED AT 18-MONTH INTERVALS AS REQUIRED BY TECH SPEC. UPON

DISCOVERY A FIRE WATCH WAS POSTED TO OBSERVE THE AFFECTED AREA UNTIL THE BARRIERS WERE INSPECTED AND VERIFIED INTACT ON 7/7/89. THE BARRIERS WERE OMITTED FROM THE LOG APPARENTLY DUE TO PERSONNEL ERROR. THE METHOD USED TO INDICATE THESE PARTICULAR BARRIERS ON THE FIRE BARRIER DRAWINGS POSSIBLY CONTRIBUTED TO THE OVERSIGHT. THIS EVENT IS NOT CONSIDERED SAFETY SIGNIFICANT AS THE BARRIERS WERE VERIFIED TO BE INTACT AND WOULD HAVE PERFORMED THEIR INTENDED FUNCTION HAD A FIRE OCCURRED IN THE AREA. THE FIRE BARRIER INSPECTION PROCEDURE AND THE PENETRATION LOG WILL BE REVISED TO INCLUDE THE BARRIERS. THE APPLICABLE DRAWING WILL ALSO BE REVISED TO INDICATE THE BARRIERS AS OTHER TS BARRIERS ARE DEPICTED. A VERIFICATION OF THE PENETRATION LOG AGAINST THE FIRE BARRIER DRAWINGS WILL ALSO BE PERFORMED.

[4] ARKANSAS NUCLEAR 2 DOCKET 50-368 LER 89-012
SAFETY INJECTION SYSTEM CHECK VALVE MALFUNCTION DUE TO MISSING INTERNAL PARTS RESULTS IN REACTOR COOLANT SYSTEM BACKLEAKAGE.
EVENT DATE: 062689 REPORT DATE: 081489 NSSS: CE TYPE: PWR
VENDOR: ATWOOD & MORRILL CO., INC.

(NSIC 215049) ON 6/26/89, DURING A PLANT HEATUP, REACTOR COOLANT SYSTEM (RCS) BACKLEAKAGE THROUGH A SAFETY INJECTION SYSTEM CHECK VALVE OCCURRED THREE TIMES. FOLLOWING EACH OCCURRENCE THE VALVE WAS RESEATED BY INJECTING WATER THROUGH THE VALVE INTO THE RCS USING A HIGH PRESSURE SAFETY INJECTION PUMP. LEAKAGE WAS RETURNED TO WITHIN ALLOWABLE LIMITS AND THE PLANT HEATUP CONTINUED. ON 6/27/89, A PLANT WAS DISASSEMBLED AND INSPECTED. TWO ROLLPINS WHICH CONNECT THE VALVE DISC TO THE VALVE DISC SHAFT, MAKING THEM ONE INTEGRAL PART, WERE FOUND MISSING. THIS ALLOWED A MISALIGNMENT OF THE SEATING SURFACES OF THE VALVE RESULTING IN LEAKAGE AS THE RCS WAS PRESSURIZED. BASED ON THESE FINDINGS, ANOTHER CHECK VALVE OF THE SAME DESIGN WAS ALSO DISASSEMBLED AND INSPECTED. BOTH ROLLPINS WERE PRESENT, HOWEVER, ONE ROLLPIN WAS FOUND CRACKED AND LOOSE. THE ROLLPINS WERE REPLACED IN BOTH VALVES AND THE VALVES REASSEMBLED. A PLANT HEATUP WAS COMMENCED AND ON 7/3/89, A SATISFACTORY LEAKAGE TEST WAS PERFORMED FOR EACH VALVE. THE CAUSE OF THE MISSING ROLLPINS COULD NOT BE DETERMINED. THE CRACKED ROLLPIN IS UNDERGOING METALLURGICAL ANALYSIS TO DETERMINE THE CAUSE OF ITS FAILURE. INSPECTIONS OF ADDITIONAL CHECK VALVES ARE PLANNED FOR THE NEXT REFUELING OUTAGE.

[5] ARNOLD DOCKET 50-331 LER 89-010
INADVERTENT REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION DUE TO PERSONNEL ERROR DURING PERFORMANCE OF STEAM LEAK DETECTION SURVEILLANCE.
EVENT DATE: 070589 REPORT DATE: 072889 NSSS: GE TYPE: BWR

(NSIC 214955) ON 7/5/89, WITH THE PLANT OPERATING AT 100% POWER, THE REACTOR CORE ISOLATION COOLING (RCIC) INBOARD ISOLATION VALVE CLOSED FOLLOWING RECEIPT OF THE "B" LOGIC ISOLATION SIGNAL. THE ISOLATION SIGNAL WAS GENERATED DURING PERFORMANCE OF THE RCIC STEAM LEAK DETECTION SYSTEM SURVEILLANCE TEST PROCEDURE (STP). THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR ON THE PART OF THE UTILITY APPRENTICE INSTRUMENT TECHNICIAN WHO LIFTED TEMPERATURE SWITCH INPUT LEADS PRIOR TO COMPLETING THE PROCEDURE STEPS WHICH PLACE THE APPROPRIATE KEYLOCK TEST SWITCH IN THE TEST POSITION. CONTRIBUTING FACTORS IN THIS EVENT WERE A COMMUNICATION ERROR BETWEEN THE JOURNEYMAN INSTRUMENT TECHNICIAN AND THE APPRENTICE AND THE FACT THAT THE APPRENTICE, ALTHOUGH TRAINED IN GENERAL ON PERFORMING STPS, HAD NOT PERFORMED THIS SPECIFIC TEST PREVIOUSLY. THIS EVENT HAD NO EFFECT ON THE SAFE OPERATION OF THE PLANT. THE ISOLATION OF RCIC WAS IMMEDIATELY IDENTIFIED AND THE APPROPRIATE ACTIONS WERE TAKEN. IMMEDIATE CORRECTIVE ACTIONS WERE TO DETERMINE THE CAUSE OF THE RCIC ISOLATION. UPON DETERMINATION, THE STP WAS STOPPED AND THE RCIC SYSTEM WAS UNISOLATED. THE STP WAS RESTARTED FOLLOWING A DISCUSSION OF THE EVENT BETWEEN THE SHIFT SUPERVISORS AND THE INSTRUMENT TECHNICIANS PERFORMING THE TEST.

[6] BEAVER VALLEY 2 DOCKET 50-412 LER 89-021
STEAM GENERATOR TUBE LEAK DUE TO A LOOSE PART CAUSING TUBE DAMAGE.
EVENT DATE: 062189 REPORT DATE: 072189 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 214829) ON 6/21/89 AT 2006 HOURS, WHILE THE UNIT WAS AT 100% POWER, THE CONDENSER AIR EJECTOR RADIATION MONITOR ALARMED DUE TO INCREASED SECONDARY SIDE ACTIVITY. STATION PERSONNEL DETERMINED THIS WAS DUE TO TUBE LEAKAGE IN THE "C" STEAM GENERATOR. ISOTOPIC ANALYSIS INITIALLY ESTIMATED THE LEAKRATE AT APPROXIMATELY 70 GALLONS PER DAY (GPD). CALCULATIONS USING A REACTOR COOLANT SYSTEM (RCS) WATER BALANCE INDICATED A CONSERVATIVELY ESTIMATED LEAKRATE OF 400 TO 500 GPD. A CONTROLLED SHUTDOWN WAS INITIATED. THE UNIT WAS PLACED IN HOT STANDBY AT 2300 HOURS. OPERATORS STABILIZED ALL PLANT PARAMETERS AT THIS TIME IN ORDER TO OBTAIN A MORE ACCURATE LEAKRATE. ADDITIONAL ISOTOPIC ANALYSIS SHOWED A LEAKRATE OF 245 GPD. AN RCS WATER BALANCE CALCULATION SHOWED A LEAKRATE OF 177 GPD. SUBSEQUENT INVESTIGATION DETERMINED THE LEAK RESULTED FROM TUBE DAMAGE DUE TO A LOOSE PART IN THE "C" STEAM GENERATOR. THE LOOSE PART WAS RETRIEVED. FOUR TUBES WERE PLUGGED. ONLY ONE TUBE WAS IDENTIFIED AS ACTUALLY LEAKING. THERE WERE NO SAFETY IMPLICATIONS DUE TO THIS EVENT. THE ACTUAL LEAKRATE DID NOT EXCEED TECHNICAL SPECIFICATIONS LIMITS. THIS EVENT WAS BOUNDED BY ANALYSIS IN UFSAR SECTION 15.6.3, "STEAM GENERATOR TUBE FAILURE."

[7] BEAVER VALLEY 2 DOCKET 50-412 LER 89-019
AUXILIARY FEEDWATER PUMP ACTUATION DUE TO A LOW-LOW STEAM GENERATOR LEVEL.
EVENT DATE: 062289 REPORT DATE: 072489 NSSS: WE TYPE: PWR

(NSIC 214896) ON 6/22/89, THE UNIT WAS IN HOT STANDBY WITH THE REACTOR TRIP BREAKERS OPEN FOLLOWING A SHUTDOWN WHICH HAD BEEN INITIATED DUE TO A TUBE LEAK IN THE "C" STEAM GENERATOR. A COOLDOWN TO COLD SHUTDOWN WAS IN PROGRESS. AT 0043 HOURS, "B" STEAM GENERATOR LEVEL DECREASED TO ITS LOW-LOW SETPOINT. THIS CAUSED THE STEAM DRIVEN AUXILIARY FEEDWATER PUMP TO AUTO-START AND SUPPLY ADDITIONAL FEEDWATER TO THE STEAM GENERATORS. THE LEVEL IN THE "B" STEAM GENERATOR INCREASED AND CLEARED ITS LOW-LOW SETPOINT WITHIN 2 MINUTES. OPERATORS THEN SHUTDOWN STEAM DRIVEN AUXILIARY FEEDWATER PUMP AND CONTINUED TO INCREASE THE STEAM GENERATOR LEVEL USING THE NORMAL FEEDWATER SYSTEM. THIS EVENT WAS THE RESULT OF THE OPERATING SHIFT PERFORMING NUMEROUS CONCURRENT TASKS WHILE MANUALLY CONTROLLING STEAM GENERATOR LEVELS TO PREVENT REACTOR COOLANT SYSTEM TEMPERATURE VARIATIONS DURING A STEAM GENERATOR LEAKAGE TEST. THIS EVENT WILL BE REVIEWED WITH ALL LICENSED OPERATORS DURING A LICENSED RETRAINING SESSION. THERE WERE NO SAFETY IMPLICATIONS DUE TO THIS EVENT. AN AUXILIARY FEEDWATER SYSTEM ACTUATION DUE TO A LOW-LOW STEAM GENERATOR LEVEL IS BOUNDED BY UFSAR ANALYSIS, SECTION 15.2.7, "LOSS OF NORMAL FEEDWATER FLOW".

[8] BEAVER VALLEY 2 DOCKET 50-412 LER 89-020
INADVERTENT START OF MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS.
EVENT DATE: 062289 REPORT DATE: 072489 NSSS: WE TYPE: PWR

(NSIC 214897) ON 6/22/89, WITH THE UNIT IN HOT STANDBY, A PLANT COOLDOWN TO COLD SHUTDOWN, DUE TO A STEAM GENERATOR TUBE LEAK, WAS IN PROGRESS. DURING THE COOLDOWN, AT 0625 HOURS, A HIGH-HIGH LEVEL FEEDWATER ISOLATION (FWI) CONDITION OCCURRED IN THE "C" STEAM GENERATOR ("C" S/G). THIS RESULTED IN A TURBINE TRIP SIGNAL, A TRIP OF THE RUNNING MAIN FEEDWATER PUMP, FEEDWATER VALVE CLOSURE AND A START OF THE MOTOR-DRIVEN AUXILIARY FEEDWATER PUMPS. THE OPERATORS STABILIZED STEAM GENERATOR LEVELS AND SHUTDOWN THE AUXILIARY FEEDWATER FLOW, WHILE REESTABLISHING MAIN FEEDWATER FLOW. THE CAUSE FOR THE FWI SIGNAL WAS THE RESULT OF THE OPERATIONS CREW PERFORMING NUMEROUS CONCURRENT TASKS WHILE MANUALLY CONTROLLING STEAM GENERATOR LEVEL. THIS EVENT WILL BE REVIEWED WITH ALL LICENSED OPERATORS DURING A LICENSED RETRAINING SESSION. THERE WERE NO SAFETY IMPLICATIONS TO THE PUBLIC AS THE PLANT WAS IN A SHUTDOWN MODE AND IN THE PROCESS OF COOLING

DOWN TO COLD SHUTDOWN, AND THIS EVENT IS BOUNDED BY UFSAR SECTION 15.1.2, "FEEDWATER SYSTEM MALFUNCTIONS CAUSING AN INCREASE IN FEEDWATER FLOW".

[9] BEAVER VALLEY 2 DOCKET 50-412 LER 89-022
INADVERTENT LETDOWN ISOLATION DUE TO ENGINEERED SAFETY FEATURES (ESF) ACTUATION.
EVENT DATE: 071389 REPORT DATE: 081489 NSSS: WE TYPE: PWR
VENDOR: KEROTEST MANUFACTURING CORP.

(NSIC 215033) ON 7/13/89, THE UNIT WAS IN HOT STANDBY PREPARING FOR A REACTOR STARTUP. FOLLOWING PLANT HEATUP/PRESSURIZATION, EIGHT GALLONS PER MINUTE (GPM) INLEAKAGE INTO THE PRIMARY DRAINS TRANSFER TANK (2DGS-TK21) WAS OBSERVED, DUE TO INCREASED PUMPOUT OF THE TANK. THE LEAKAGE WAS BELIEVED TO BE THROUGH THE PRIMARY DRAINS HEADER ISOLATION VALVE (2DGS-300). A CONTAINMENT (CNMT) ENTRY WAS PERFORMED TO ATTEMPT TO RESEAT THE VALVE. DURING THIS ATTEMPT, THE VALVE WOULD NOT RECLOSE AND THE OPERATORS NOTED DECREASING PRESSURIZER (PZR) LEVEL AND INCREASING CNMT SUMP PUMPOUT RATE. CHARGING FLOW WAS INCREASED, BUT LETDOWN ISOLATED, AT 1103 HOURS, DUE TO A LOW PZR LEVEL. LETDOWN WAS RESTORED AT 1107 HOURS, BUT RE-ISOLATED AT 1114 HOURS. AN EXCESS LETDOWN FLOWPATH WAS LINED UP TO THE CHARGING SYSTEM, REDUCING PRESSURE IN THE DRAINS HEADER AND ON 2DGS-300. AT 1125 HOURS, LETDOWN WAS RESTORED. AT 1143 HOURS, 2DGS-300 WAS RECLOSED AND 2DGS-RV101 WAS CLOSED. AT 1215 HOURS, WITH INLEAKAGE TO 2DGS-TK21 STOPPED, A REACTOR COOLANT SYSTEM INVENTORY BALANCE TEST WAS INITIATED. THE CAUSE FOR THIS EVENT WAS DUE TO SEAT LEAKAGE ON 2DGS-300. A REPLACEMENT FOR THIS VALVE IS UNDER EVALUATION. THERE WERE NO SAFETY IMPLICATIONS AS A RESULT OF THIS EVENT. ALL REACTOR COOLANT SYSTEM LEAKAGE WAS WITHIN TECHNICAL SPECIFICATION LIMITS.

[10] BEAVER VALLEY 2 DOCKET 50-412 LER 89-023
POTENTIAL GENERIC FAILURE OF WESTINGHOUSE SOLID STATE PROTECTION SYSTEM.
EVENT DATE: 080489 REPORT DATE: 080989 NSSS: WE TYPE: PWR
VENDOR: POTTER & BRUMFIELD
TARGET ROCK CORP.

(NSIC 215032) ON 8/4/89, THE STATION IDENTIFIED A POTENTIAL GENERIC PROBLEM WITH ITS WESTINGHOUSE SOLID STATE PROTECTION SYSTEM AUXILIARY RELAYS. THE NORMALLY CLOSED CONTACTS OF THE RELAYS (POTTER BRUMFIELD MDR) USED TO CONTROL 2 OF THE 4 THE CONTAINMENT VACUUM PUMP SUCTION VALVES AT UNIT 2 WERE RATED AT 0.8 AMPS DC. DUE TO THEIR APPLICATION, THE CURRENT THROUGH THE RELAY CONTACTS WAS 1.4 AMPS DC. THIS RESULTED IN ARCING WHEN THE RELAYS TRIED TO OPEN IN RESPONSE TO A CONTAINMENT ISOLATION PHASE A (CIA) SIGNAL. THE ARCING COULD HOLD THE CONTACTS CLOSED, PREVENTING THE VALVES FROM CLOSING. THE STATION HAS MODIFIED THIS CONTROL CIRCUIT TO USE 2 SETS OF CONTACTS IN SERIES FROM EACH RELAY. THIS INCREASED THE CONTACTS' RATING TO 1.5 AMPS DC. THE RELAYS WERE REPEATEDLY CYCLED UNDER WORST CASE CONDITIONS. NO ARCING OCCURRED. A REVIEW OF ALL OTHER SOLID STATE PROTECTION SYSTEM AUXILIARY RELAYS AT UNITS 1 AND 2 VERIFIED THAT THERE WERE NO OTHER PROBLEMS OF THIS TYPE. THERE WERE MINIMAL SAFETY IMPLICATIONS DUE TO THIS EVENT. EACH OF THE 2 AFFECTED VALVES HAD A FULLY OPERABLE CIA VALVE IN SERIES WITH IT THAT WAS CAPABLE OF CLOSING. ADDITIONALLY, IN THE EVENT A CIA SIGNAL OCCURRED, OPERATORS COULD HAVE MANUALLY CLOSED THE AFFECTED VALVES.

[11] BIG ROCK POINT DOCKET 50-155 LER 89-006
DISCOVERED DEFECTS IN FIRE PENETRATION SEALS RESULTING IN TECH SPEC VIOLATION.
EVENT DATE: 072089 REPORT DATE: 081889 NSSS: GE TYPE: BWR

(NSIC 215054) BIG ROCK POINT TECH SPEC 3.7.12 REQUIRES THAT PENETRATION FIRE BARRIERS BE OPERATIONAL AT ALL TIMES. CONTRARY TO THE ABOVE, DURING PERFORMANCE SURVEILLANCE TEST REQUIRED BY TECHNICAL SPECIFICATION 4.7.12, TWO OF THE SURVEILLANCE BARRIERS WERE DISCOVERED WITH SMALL HOLES ON JULY 19, 1989. UPON DISCOVERY, A FIRE WATCH/PATROL WAS ESTABLISHED PER THE REQUIREMENTS OF TECHNICAL

SPECIFICATION 3.7.12 ACTION A AND B. REPAIRS WERE COMPLETED LATER ON JULY 19, 1989 AND THE FIRE WATCH/PATROL WAS TERMINATED. IN ADDITION, ON AUGUST 14, 1989, DURING A QA FIRE PROTECTION AUDIT PLANT WALKDOWN, A FIRE BARRIER PENETRATION SEAL WAS FOUND TO BE DEFICIENT. UPON DISCOVERY, A FIRE WATCH/PATROL WAS ESTABLISHED PER TECHNICAL SPECIFICATION 3.7.12 AND WILL REMAIN IN EFFECT UNTIL REPAIRS ARE COMPLETED. CAUSE OF THE DEFICIENCIES WAS ATTRIBUTED TO THE FOLLOWING: 1) FAILURE OF THE INITIAL FIRE BARRIER DESIGN REVIEW TO IDENTIFY A BREACH WHICH WAS LOCATED BEHIND A VENTILATION FAN AND A BREACH IN THE BARRIER BETWEEN THE ELECTRICAL EQUIPMENT ROOM AND THE COMPUTER ROOM. 2) INADEQUATE MAINTENANCE CONTROLS WHICH RESULTED IN A FAILURE TO REPAIR A PENETRATION AFTER A MODIFICATION.

[12] BRAIDWOOD 1 DOCKET 50-456 LER 89-006
UNIT 1 AND UNIT 2 REACTOR TRIP AS A RESULT OF LIGHTNING INDUCED VOLTAGE TRANSIENTS AFFECTING THE ROD CONTROL SYSTEM.
EVENT DATE: 071889 REPORT DATE: 031489 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: BRAIDWOOD 2 (PWR)

(NSIC 215050) AT 2020 AND 2026 ON 7/18/89 THE STATION EXPERIENCED LIGHTNING INDUCED VOLTAGE TRANSIENTS CAUSING MULTIPLE ROD DRIVE OVERVOLTAGE PROTECTION DEVICES TO ACTUATE ON UNIT 1 AND 2, RESPECTIVELY. ON UNIT 1 TEN OUT OF TEN OVERVOLTAGE PROTECTORS ACTUATED. ON UNIT 2 SEVEN OUT OF TEN OVERVOLTAGE PROTECTORS ACTUATED. THIS REMOVED POWER TO VARIOUS ROD DRIVE CONTROL CARDS AND ALLOWED NUMEROUS CONTROL RODS TO DROP. BOTH REACTORS TRIPPED DUE TO NEGATIVE RATE TRIP ON THE POWER RANGE. IMMEDIATE CORRECTIVE ACTIONS WERE TO STABILIZE THE PLANT, RESET THE OVERVOLTAGE PROTECTORS, AND VERIFY ROD CONTROL OPERABILITY. THE EXACT LOCATION OF THE LIGHTNING STRIKES ARE UNKNOWN. THE ROOT CAUSE IS INADEQUATE PROTECTION AND ISOLATION OF THE ROD CONTROL SYSTEM FROM LIGHTNING INDUCED TRANSIENTS. THE IMMEDIATE CORRECTIVE ACTIONS WERE TO RESET THE OVERVOLTAGE PROTECTORS. NO DAMAGE OCCURRED TO THE ROD CONTROL SYSTEM. THERE HAS BEEN ONE PREVIOUS OCCURRENCE OF A LIGHTNING INDUCED VOLTAGE TRANSIENT RESULTING IN A REACTOR TRIP. CORRECTIVE ACTIONS WERE IMPLEMENTED ADDRESSING BOTH ROOT AND CONTRIBUTING CAUSES FOR THE ABOVE EVENT. THE PREVIOUS CORRECTIVE ACTIONS ARE NOT APPLICABLE TO THIS EVENT.

[13] BRAIDWOOD 2 DOCKET 50-457 LER 89-003 REV 01
UPDATE ON MISPOSITIONING OF 2B CENTRIFUGAL CHARGING PUMP MANUAL MINIFLOW ISOLATION VALVE DUE TO PERSONNEL ERROR.
EVENT DATE: 032289 REPORT DATE: 080989 NSSS: WE TYPE: PWR

(NSIC 215051) ON 3/22/89 THE DISCHARGE FROM THE RH PUMPS TO THE CHEMICAL VOLUME AND CONTROL (CV) LETDOWN HEADER WAS REALIGNED. THIS REQUIRED UNLOCKING AND OPENING THE 2RH8734A, AND CLOSING AND LOCKING THE 2RH8734B. AT 1730 THE EQUIPMENT ATTENDENT (EA) UNLOCKED AND OPENED THE 2RH8734A. THE EA SAW ANOTHER LOCKED VALVE ABOUT 15 FEET TO THIS LEFT, 2CV8479B, AND MISIDENTIFIED IT AS 2RH8734B. THE 2CV8479B VALVE WAS CLOSED AND RELOCKED. AT 0920 ON 3/23/89 A DIFFERENT EA OBSERVED THAT BOTH THE 2RH8734A AND 2H8734B WERE OPEN. THE EA CLOSED AND LOCKED THE 2RH8734B. ON 6/1/89 THE 2B CV PUMP WAS STARTED. RECIRCULATION FLOW OBSERVED TO BE READING ZERO. THE 2CV8479B MANUAL RECIRC ISOLATION VALVE WAS IMMEDIATELY OPENED AND RE-LOCKED. THE RECIRC FLOW RETURNED TO NORMAL. THE ROOT CAUSE OF THIS EVENT WAS PERSONNEL ERROR. CONTRIBUTING CAUSES WERE (1) OPERATING PROCEDURES DID NOT REQUIRE TRACKING OR INDEPENDENT VERIFICATION. (2) RUSHING COMPONENT IDENTIFICATION. (3) COMMON LOCK CORE FOR AN ENTIRE TRAIN OF COMPONENTS. THE VALVE WAS IMMEDIATELY OPENED. THE INDEPENDENT VERIFICATION PROGRAM WILL BE REVISED. THE LOCKED EQUIPMENT PROGRAM WILL BE REVISED TO INCLUDE A UNIQUE LOCK FOR EACH SAFETY RELATED COMPONENT. THERE WAS A PREVIOUS OCCURRENCE OF A MISPOSITIONING OF A LOCKED SAFETY RELATED VALVE. PREVIOUS CORRECTIVE ACTIONS WERE NOT APPLICABLE TO THIS EVENT.

[14] BROWNS FERRY 1 DOCKET 50-259 LER 89-015
MOMENTARY LOSS OF SECONDARY CONTAINMENT CAUSED BY FAILURE OF WELDS ON THE DOOR LOCK MECHANISM.
EVENT DATE: 062789 REPORT DATE: 072789 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 214870) AT 0845 ON JUNE 27, 1989, WHILE A GROUP OF SIX MODIFICATIONS PERSONNEL WERE LEAVING THE REACTOR BUILDING FROM THE REFUEL FLOOR TO THE CONTROL BUILDING ROOF, BOTH DOORS OF THE UNIT 1 AIRLOCK WERE OPENED SIMULTANEOUSLY FOR APPROXIMATELY FIVE (5) SECONDS. THIS CONDITION CONSTITUTED A BREACH OF SECONDARY CONTAINMENT. THIS EVENT WAS CAUSED BY THE FAILURE OF TWO WELDS WHICH ATTACH THE BRACKET THAT HOLDS THE LOCKSET IN THE DOOR ON THE REFUEL FLOOR SIDE OF THE AIRLOCK. THESE WELDS WERE BROKEN LOOSE DUE TO HEAVY USE OF THE AIRLOCK DOORS DURING THIS OUTAGE. THE LOCKSET BRACKET WAS REATTACHED AND OPERATION OF THE DOOR VERIFIED. SIGNS WILL BE PLACED AT AIRLOCKS ABOUT PROPER USAGE. FUTURE MAINTENANCE REQUESTS WHICH AFFECT OPERABILITY OF SECONDARY CONTAINMENT DOORS WILL BE SCHEDULED FOR IMMEDIATE WORK. THE MAINTENANCE REQUEST PROCEDURE WILL ALSO BE REVISED TO ALLOW APPROPRIATE PRIORITIES TO BE PLACED ON MAINTENANCE REQUESTS UNDER ALL PLANT CONDITIONS. PERFORMANCE OF PREVENTIVE MAINTENANCE FOR SECONDARY CONTAINMENT AIRLOCK DOORS WILL BE INCREASED IN FREQUENCY. THE FEASIBILITY OF A CHANGE TO THE CURRENT INTERLOCK SYSTEM DESIGN WILL BE EVALUATED. DURING THIS EVENT, UNIT 2 WAS IN COLD SHUTDOWN WITH IRRADIATED FUEL IN THE VESSEL. THE VESSEL WAS IN THE FLOODED UP CONDITION WITH THE FUEL POOL GATES INSTALLED. UNITS 1 AND 3 WERE DEFUELED.

[15] BROWNS FERRY 1 DOCKET 50-259 LER 89-017
UNPLANNED ENGINEERED SAFETY FEATURE ACTUATION DUE TO POTENTIAL TRANSFORMER DOOR FALLING OPEN.
EVENT DATE: 070189 REPORT DATE: 072789 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)
VENDOR: GENERAL ELECTRIC CO.

(NSIC 214871) ON JULY 1, 1989 AT 1431 HOURS, THE POTENTIAL TRANSFORMER COMPARTMENT DOOR ON 4 KV SHUTDOWN BOARD B FELL OPEN, DISCONNECTING THE POTENTIAL TRANSFORMER FUSE CONTACTS FROM THE CABINET CONTACTS. THIS CAUSED A SIMULATED UNDERVOLTAGE TO BE SENSED ON THE SHUTDOWN BOARD WHICH CAUSED THE B DIESEL GENERATOR TO START AND TIE ONTO THE BOARD. FAILURE OF THE POTENTIAL TRANSFORMER DOOR LATCHING MECHANISM AND NO REQUIREMENT TO VERIFY THE DOOR WAS LATCHED AFTER CLOSING WAS THE CAUSE OF THE EVENT. AFTER CLEANING AND LUBRICATING THE LATCHING MECHANISM, THE COMPARTMENT DOOR WAS CLOSED AND VERIFIED LATCHED. FOLLOWING THE INITIAL INVESTIGATION, THE NORMAL FEEDER BREAKER FOR THE 4 KV SHUTDOWN BOARD WAS CLOSED AND THE DIESEL GENERATOR WAS SHUT DOWN. A GENERAL OPERATING INSTRUCTION WILL BE ISSUED TO REQUIRE OPERATIONS PERSONNEL TO VERIFY COMPARTMENT DOORS ARE SECURELY FASTENED AFTER WORKING IN THE COMPARTMENT. THE POTENTIAL TRANSFORMER COMPARTMENT DOORS IN THE SAFETY RELATED 4 KV SHUTDOWN BOARDS WILL BE VERIFIED SHUT AND LATCHED PROPERLY. THE ELECTRICAL MAINTENANCE INSTRUCTION COVERING 4 KV SWITCH GEAR WILL BE UPGRADED TO INCLUDE CLEANING AND LUBRICATION OF DOOR/COMPARTMENT LATCHING MECHANISMS. DURING THIS EVENT UNIT 2 WAS IN COLD SHUTDOWN WITH IRRADIATED FUEL IN THE VESSEL. UNITS 1 AND 3 WERE DEFUELED.

[16] BROWNS FERRY 1 DOCKET 50-259 LER 89-018
LOW FLOW THROUGH AREA COOLERS DUE TO INADEQUATE SURVEILLANCE.
EVENT DATE: 071289 REPORT DATE: 081189 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 214985) ON JULY 12, 1989, IT WAS DETERMINED THAT THE AREA COOLERS SERVICING

THE RESIDUAL HEAT REMOVAL (RHR) PUMPS AND THE CORE SPRAY (CS) PUMPS MAY HAVE OPERATED WITH DEGRADED FLOW RATES DURING PREVIOUS PLANT OPERATIONS. ON MAY 25, 1989, WHILE CHECKING FLOW ON THE 2C RHR PUMP ROOM COOLER AS PART OF AN INVESTIGATION FOLLOWING AN INCORRECT CABLE TERMINATION ON THE FAN MOTOR, THE AIR FLOW RATE WAS FOUND TO BE LESS THAN THE DESIGN FLOW RATE OF 10,000 CFM +/- TEN PERCENT. SUBSEQUENTLY, THE AIR FLOW RATES ON THE REMAINING RHR PUMP ROOM COOLERS (2A, 2B AND 2D) AND BOTH UNIT 2 CS PUMP ROOM COOLERS WERE MEASURED. INITIAL MEASUREMENTS OF THE REMAINING ROOM COOLERS INDICATED AIR FLOW RATES WERE BELOW THEIR DESIGN RATES. INVESTIGATION FOUND SEVERAL ITEMS THAT CONTRIBUTED TO THE LOW FLOW RATES; HOWEVER, THE ROOT CAUSE OF THIS CONDITION WAS LACK OF A PROCEDURE/PROGRAM TO PERIODICALLY MEASURE FLOW RATES THROUGH THE ROOM COOLERS. THE LOW FLOW CONDITION IS BELIEVED TO BE THE RESULT OF COOLER DEGRADATION WHICH MAY HAVE EXISTED DURING PLANT OPERATION. A TECHNICAL INSTRUCTION SHALL BE DEVELOPED FOR MEASUREMENTS OF THE RHR AND CS ROOM COOLER AIR FLOW RATES. THE UTILITY DESIGN ORGANIZATION EVALUATED THE AS-FOUND FLOWS FOR IMMEDIATE IMPACT ON EQUIPMENT OPERABILITY IN THE RHR AND CS PUMP ROOMS AND FOUND THE FLOWS ACCEPTABLE FOR THE PRESENT CONDITIONS.

[17]   BROWNS FERRY 1   DOCKET 50-259   LER 89-019
UNPLANNED ENGINEERED SAFETY FEATURE ACTUATION CAUSED BY FAILED CONNECTION IN RADIATION DETECTOR DUE TO REPEATED MOVEMENT OF DETECTOR CABLE.
EVENT DATE: 072689   REPORT DATE: 082489   NSSS: GE   TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 2 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 215113) ON JULY 26, 1989, AT 1134 HOURS, THE UNIT 1 RADIATION MONITOR ON THE REFUEL ZONE VENTILATION EXHAUST (1-RM-90-140) RECEIVED A SPURIOUS SIGNAL RESULTING IN A PRIMARY CONTAINMENT ISOLATION SYSTEM GROUP 6 ISOLATION, REFUEL ZONE ISOLATION, AND INITIATION OF THE STANDBY GAS TREATMENT SYSTEM AND THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM. THE RADIATION MONITOR INDICATION WAS VERIFIED TO BE ERRONEOUS THROUGH COMPARISON WITH OTHER INDICATIONS AND SURVEYS. THE SPURIOUS SIGNAL OCCURRED WHEN THE DETECTOR CABLE FOR RADIATION MONITOR 1-RM-90-140 WAS REPOSITIONED TO READ THE INDICATION FOR RADIATION MONITOR 1-RM-90-1. THE CAUSE OF THIS EVENT WAS A FAILED CONNECTION AT THE RADIATION DETECTOR ASSEMBLY. CONTRIBUTING TO THIS EVENT WAS AN INADEQUATE DETECTOR CLAMP CONNECTION AND THE REPEATED MOVEMENT OF THE CABLE DUE TO POOR ROUTING. THE RADIATION DETECTOR WAS SUBSEQUENTLY REPAIRED AND DECLARED OPERABLE ON JULY 26, 1989, AT 2135 HOURS. THE DETECTOR CABLE WAS REPOSITIONED TO ALLOW AN UNOBSTRUCTED VIEW FOR THE INDICATION FOR RADIATION MONITOR 1-RM-90-1. ADDITIONALLY, A VISUAL INSPECTION OF UNITS 2 AND 3 WAS PERFORMED FOR GENERIC IMPLICATIONS. MAINTENANCE CONTROLS PROVIDED BY PRESENT PLANT INSTRUCTIONS, PROCEDURES, AND PROGRAMS ARE ADEQUATE TO PREVENT RECURRENCE OF THE CABLE CONNECTION DISCREPANCY.

[18]   BROWNS FERRY 2   DOCKET 50-260   LER 89-009 REV 01
UPDATE ON UNPLANNED SCRAM AND MAIN STEAM LINE ISOLATION DUE TO SPURIOUS SPIKES ON MAIN STEAM LINE RADIATION MONITORS.
EVENT DATE: 041189   REPORT DATE: 081189   NSSS: GE   TYPE: BWR

(NSIC 215011) ON APRIL 11, 1989, AT 1414 HOURS, BROWNS FERRY UNIT 2 RECEIVED SPURIOUS SPIKES ON INDEPENDENT MAIN STEAM LINE (MSL) RADIATION MONITORS WHICH RESULTED IN COMPLETION OF THE INITIATION LOGIC FOR A FULL SCRAM AND MAIN STEAM LINE ISOLATION VALVE (MSLV) ISOLATION. ALL THREE UNITS WERE SHUTDOWN WITH UNITS 1 AND 3 DEFUELED; ALL CONTROL RODS WERE FULLY INSERTED AND MAIN STEAM ISOLATION VALVES (MSIVS) WERE CLOSED PRIOR TO THIS EVENT. THE SPECIFIC CAUSE OF THE SPURIOUS SPIKES HAS NOT BEEN DETERMINED. THE SPURIOUS SPIKES MAY HAVE BEEN CAUSED BY VIBRATIONS GENERATED BY PERSONNEL WORKING IN THE AREA HOWEVER, ATTEMPTS TO RECREATE THE SPIKES WERE UNSUCCESSFUL. THE SUBSEQUENT INVESTIGATIONS INCLUDED A RESPONSE CHECK OF THE MONITORS, VISUAL AND SIGNAL CHECKS OF THE CABLES AND

CONNECTIONS AND GENERATION OF POTENTIAL SOURCES OF ELECTROMAGNETIC INTERFERENCE. THE SIGNAL CABLES FROM THE RADIATION MONITORS WERE TEMPORARILY DISCONNECTED, TO PREVENT SUBSEQUENT SPURIOUS TRIPS. THE SCRAM AND ISOLATION LOGIC WAS RESET AT 1835 HOURS. THE MSL RADIATION MONITOR IS NOT REQUIRED IN THE CURRENT PLANT CONFIGURATION; THE SCRAM AND ISOLATION SIGNALS HAVE BEEN DISABLED TO PREVENT SUBSEQUENT SPURIOUS TRIPS.

[19] BROWNS FERRY 2 DOCKET 50-260 LER 89-015 REV 01
UPDATE ON REVERSE ROTATION OF 2C RESIDUAL HEAT REMOVAL PUMP COOLER FAN MOTOR.
EVENT DATE: 052389 REPORT DATE: 072489 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 1 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 214872) ON MAY 23, 1989, DURING THE ROUTINE PERFORMANCE OF A UNIT 2 RESIDUAL HEAT REMOVAL (RHR) SYSTEM FLOW SURVEILLANCE TEST IT WAS DISCOVERED THERE WAS NO AIR FLOW IN THE 2C RHR PUMP ROOM COOLER FAN DUCT. THE FAN WAS VERIFIED TO BE ROTATING BACKWARDS. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR. A MODIFICATION WAS RECENTLY COMPLETED WHICH REPLACED AND UPGRADED THE CABLE FEEDING THE 2C RHR PUMP COOLER FAN MOTOR. THE MODIFICATION WORK PLAN CONTAINED A VERIFICATION OF PROPER FAN ROTATION AND WAS SIGNED OF WITH THE FAN ROTATING IN THE WRONG DIRECTION. THE IMMEDIATE CORRECTIVE ACTION INCLUDED ESTABLISHING AND VERIFYING PROPER FAN ROTATION. PROPER FAN ROTATION ON THE REMAINDER OF THE UNIT 2 PUMP AREA COOLERS WAS VERIFIED. FURTHER CORRECTIVE ACTION WILL BE TO ESTABLISH MINIMUM QUALIFICATIONS FOR PERSONNEL PERFORMING CONSTRUCTION TEST FOLLOWING EQUIPMENT MODIFICATION, REVIEW THE METHODOLOGY FOR VERIFYING PROPER EQUIPMENT ROTATION, CORRECT FAN ROTATION MARKINGS AND ESTABLISH A PREVENTIVE MAINTENANCE PROGRAM FOR PERIODICALLY MEASURING AIR FLOW RATES FOR TECHNICAL SPECIFICATION RELATED FANS NOT SPECIFIED BY SURVEILLANCE REQUIREMENTS THUS NOT COVERED BY SIS. ALL THREE UNITS AT BROWNS FERRY WERE IN COLD SHUTDOWN WITH UNITS 1 AND 3 DEFUELED WHEN THIS CONDITION WAS DISCOVERED.

[20] BROWNS FERRY 2 DOCKET 50-260 LER 89-016 REV 01
UPDATE ON TECH SPEC VIOLATION DUE TO A SEISMICALLY UNQUALIFIED 480V SHUTDOWN BOARD.
EVENT DATE: 053089 REPORT DATE: 071189 NSSS: GE TYPE: BWR

(NSIC 215012) ON MAY 30, 1989, IT WAS DETERMINED THAT 480 VOLT SHUTDOWN BOARD 2A HAD NOT BEEN MAINTAINED IN A SEISMICALLY QUALIFIED CONFIGURATION BECAUSE THE NORMAL FEEDER BREAKER FOR THE BOARD WAS IN THE DISCONNECT POSITION WITH THE BREAKER COMPARTMENT DOOR OPEN. THIS PLACED THE SHUTDOWN BOARD AND LOADS SUPPLIED BY THE BOARD, INCLUDING CORE SPRAY LOOP I IN AN INOPERABLE CONDITION. THIS CORE SPRAY LOOP WAS REQUIRED TO COMPLY WITH TECHNICAL SPECIFICATIONS. THE EVENT RESULTED FROM AN UNIDENTIFIED PERSON FAILING TO COMPLY WITH ADMINISTRATIVE CONTROL REQUIREMENTS TO CLOSE THE COMPARTMENT DOOR. BREAKER COMPARTMENT DOORS ARE REQUIRED TO BE CLOSED AND LATCHED TO MAINTAIN SEISMIC QUALIFICATION ON THE SHUTDOWN BOARD. THE IMMEDIATE CORRECTIVE ACTION WAS TO CLOSE AND LATCH THE COMPARTMENT DOOR. FURTHER CORRECTIVE ACTION INCLUDES REVISING THE CONDUCT-OF-OPERATIONS INSTRUCTION TO REQUIRE INSPECTIONS OF ALL ELECTRICAL DISTRIBUTION BOARDS FOR COMPLIANCE WITH SEISMIC REQUIREMENTS DURING PLANT TOURS. THE UTILITY WILL REANALYZE THEIR REQUIREMENTS TO DETERMINE AN ACCEPTABLE TIME FRAME FOR A DOOR TO REMAIN OPEN WITHOUT THE BOARD BEING INOPERABLE. ALL THREE UNITS AT BROWNS FERRY WERE IN COLD SHUTDOWN WITH UNITS 1 AND 3 DEFUELED WHEN THIS CONDITION WAS DISCOVERED.

[21] BROWNS FERRY 2 DOCKET 50-260 LER 89-019
TECH SPEC VIOLATION CAUSED BY MISSED SURVEILLANCE ON RESIDUAL HEAT REMOVAL CHECK VALVE.
EVENT DATE: 062389 REPORT DATE: 072089 NSSS: GE TYPE: BWR

(NSIC 214873) ON JUNE 23, 1989, AT 0850 HOURS IT WAS DETERMINED THAT LOOP II OF THE RESIDUAL HEAT REMOVAL SYSTEM (RHR) FOR UNIT 2 WAS TECHNICALLY INOPERABLE AT A TIME WHEN OPERABILITY WAS REQUIRED. THE PRIMARY CONTAINMENT INBOARD TESTABLE CHECK VALVE FOR THIS LOOP HAD NOT BEEN CYCLED WITHIN ITS REQUIRED SURVEILLANCE FREQUENCY. THIS IS REPORTABLE AS A VIOLATION OF TECHNICAL SPECIFICATION REQUIREMENTS. THIS OCCURRENCE WAS THE RESULT OF INADEQUATE TRACKING AND DISPOSITIONING OF A TEST DEFICIENCY (TD), CAUSED BY PROCEDURAL INADEQUACY. SUBSEQUENT TESTING SHOWED THE CHECK VALVE WAS OPERABLE AND RHR LOOP II WAS CAPABLE OF INJECTING COOLANT IF NEEDED DURING THE PERIOD OF TIME WHEN OPERABILITY WAS REQUIRED. SHIFT TECHNICAL ADVISOR AND SENIOR REACTOR OPERATOR SIGNATURE REQUIREMENTS WILL BE ADDED TO APPLICABLE TEST PACKAGE COVER PAGES TO ENSURE AND DOCUMENT REVIEW FOR POTENTIAL LCOS. DURING THIS EVENT, UNIT 2 WAS IN COLD SHUTDOWN WITH IRRADIATED FUEL IN THE VESSEL. THE VESSEL WAS IN THE FLOODED UP CONDITION WITH THE FUEL POOL GATES INSTALLED. UNITS 1 AND 3 WERE DEFUELED.

[22] BROWNS FERRY 2 DOCKET 50-260 LER 89-020
UNPLANNED ENGINEERED SAFETY FEATURE ACTUATION DURING FUSE REPLACEMENT CAUSED BY PERSONNEL ERROR.
EVENT DATE: 070289 REPORT DATE: 073189 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 1 (BWR)
BROWNS FERRY 3 (BWR)

(NSIC 214944) ON JULY 2, 1989, AT 1335 HOURS, THE LOGIC CIRCUIT ASSOCIATED WITH THE UNIT 2 REACTOR ZONE VENTILATION RADIATION MONITOR WAS DEENERGIZED DURING REPLACEMENT OF A FUSE. THIS RESULTED IN THE ISOLATION OF UNIT 2 REACTOR ZONE VENTILATION, AND THE REFUELING ZONE VENTILATION; AND THE INITIATION OF STANDBY GAS TREATMENT AND CONTROL ROOM EMERGENCY VENTILATION. THIS EVENT WAS CAUSED BY ELECTRICIANS INCORRECTLY JUMPERING THE FUSE PRIOR TO ITS REMOVAL. THE ELECTRICIANS WERE UNFAMILIAR WITH THE ALARM TYPE FUSE AND ASSOCIATED CIRCUIT. AFTER COMPLETION OF THE INVESTIGATION, THE JUMPER WAS PROPERLY INSTALLED AND THE FUSE REPLACED. THE VENTILATION SYSTEMS WERE RETURNED TO NORMAL AT 1525 HOURS. THE ELECTRICIANS INVOLVED WERE COUNSELED ON THE NEED TO RESEARCH UNFAMILIAR EQUIPMENT OR CONDITIONS. TRAINING WILL BE PROVIDED TO ELECTRICAL MAINTENANCE CRAFT AND PLANNERS ON ALARMING FUSES AND CIRCUITS. IN ADDITION, THE MAINTENANCE INSTRUCTION FOR FUSE REPLACEMENT WILL BE REVISED TO ADD A CAUTION ABOUT THIS TYPE OF FUSE CIRCUIT. UNIT 2 WAS IN COLD CONDITION AND UNITS 1 AND 3 WERE DEFUELED WHEN THIS EVENT OCCURRED.

[23] BROWNS FERRY 2 DOCKET 50-260 LER 89-024
CONTRACT ENGINEER ENTERED HIGH RADIATION AREA WITHOUT PROPER DOSE MONITORING EQUIPMENT DUE TO PERSONNEL ERROR.
EVENT DATE: 071889 REPORT DATE: 081889 NSSS: GE TYPE: BWR

(NSIC 215014) ON JULY 18, 1989, AT 1820 HOURS, A CONTRACT ENGINEER WAS FOUND ALONE INSIDE A HIGH RADIATION AREA POSTED AROUND THE UNIT 2 FUEL POOL COOLING HEAT EXCHANGERS. CONTRARY TO THE HIGH RADIATION AREA ENTRY REQUIREMENTS OF THE RADIATION WORK PERMIT (RWP) AND TECHNICAL SPECIFICATION 6.8.3.1, THE ENGINEER DID NOT HAVE IN IS POSSESSION A DOSE WARNING DEVICE, A DOSE RATE INSTRUMENT, NOR WAS HE ACCOMPANIED BY ANYONE WHO HAD ONE OF THESE DEVICES IN THEIR POSSESSION. THE INDIVIDUAL RECEIVED A 30 MILLIREM DOSE DURING THE ENTRY FROM 1740 TO 1830 HOURS. DISCUSSION WITH THE INDIVIDUAL INDICATED THAT HE WORKED IN THE SAME AREA THE PREVIOUS DAY AND HAD NOT OBTAINED A DOSE WARNING DEVICE FOR THAT ENTRY EITHER. SUBSEQUENTLY, THE INDIVIDUAL'S TLD WAS PROCESSED AND HE WAS ASSIGNED A TOTAL DOSE FOR THE QUARTER OF 145 MILLIREMS. UNITS 1 AND 3 WERE DEFUELED AND UNIT 2 WAS IN COLD SHUTDOWN DURING THIS EVENT. THE WORKER FAILED TO PAY PROPER ATTENTION AND COMPLY WITH THE REQUIREMENTS ON THE RWP. THE AREA WAS PROPERLY POSTED, AND THE RWP CONTAINED ALL THE NECESSARY INFORMATION FOR THE INDIVIDUAL TO PERFORM HIS JOB AND NOT VIOLATE ANY REQUIREMENTS. A NEW RWP FORM PUT INTO EFFECT ON JULY 17, 1989, MAY HAVE BEEN A CONTRIBUTING CAUSE. BASED UPON A RANDOM SAMPLING OF

WORKERS IN THE ADIOLOGICALLY CONTROLLED AREA. THERE IS NO INDICATION OF A GENERAL LACK OF UNDERSTANDING OF RWP REQUIREMENTS.

[24] BROWNS FERRY 2 DOCKET 50-260 LER 89-022
TECH SPEC VIOLATION DUE TO LOSS OF TWO TRAINS OF STANDBY GAS TREATMENT SYSTEM.
EVENT DATE: 072189 REPORT DATE: 082089 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 1 (BWR)
BROWNS FERRY 3 (BWR)
VENDOR: GENERAL ELECTRIC CO.

(NSIC 215013) ON JULY 21, 1989, AT 1345 HOURS IT WAS DISCOVERED THAT THE HUMIDITY CONTROL HEATER INDICATING LIGHTS FOR STANDBY GAS TREATMENT TRAINS A AND B WERE EXTINGUISHED. INVESTIGATION REVEALED THE BREAKERS FOR THE RELATIVE HUMIDITY CONTROL HEATERS WERE IN THE TRIPPED POSITION. THE BREAKERS WERE RESET AND SBGT TRAINS A AND B WERE DECLARED INOPERABLE. THIS EVENT WAS A VIOLATION OF TECHNICAL SPECIFICATIONS WHICH REQUIRES A MINIMUM OF TWO TRAINS OF SBGT OPERABLE WHEN SECONDARY CONTAINMENT INTEGRITY IS REQUIRED, AND IS BEING REPORTED UNDER 10CFR 50.73 (A)(2)(I). MAINTENANCE REQUESTS (MRS) WERE ISSUED AGAINST EACH BREAKER. AFTER PERFORMING THE MAINTENANCE ON EACH BREAKER, THE BREAKERS WERE REINSTALLED IN THE MOTOR CONTROL CENTER. AFTER PLACING SBGT BACK INTO SERVICE, THE "A" TRAIN HUMIDITY CONTROL HEATER BREAKER TRIPPED AGAIN. THIS BREAKER WAS THEN REPLACED AND THE MOTOR CONTROL CENTER (MCC) COMPARTMENT DOORS WERE REMOVED IN ORDER TO PROVIDE MORE AIR CIRCULATION THROUGH THE BREAKER COMPARTMENTS. A VENTILATION FLOW TEST OF THE DIESEL AUXILIARY BOARD ROOM WHICH HOUSES THE MCCS WAS PERFORMED. FURTHER CORRECTIVE ACTION WILL BE TO CONTINUE TO INVESTIGATE THE CIRCUIT BREAKER TRIPPING PROBLEM AND PROVIDE A SUPPLEMENTAL REPORT.

[25] BROWNS FERRY 2 DOCKET 50-260 LER 89-023
LOSS OF SECONDARY CONTAINMENT DUE TO LOSS OF TWO TRAINS OF STANDBY GAS TREATMENT SYSTEM.
EVENT DATE: 072389 REPORT DATE: 082289 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BROWNS FERRY 1 (BWR)
BROWNS FERRY 3 (BWR)
VENDOR: ELECTRO - MOTIVE DIV. OF GM
GENERAL ELECTRIC CO.
SQUARE D COMPANY

(NSIC 215060) ON 7/23/89, AT 0920 HOURS, STANDBY GAS TREATMENT (SGT) TRAIN C BECAME INOPERABLE BECAUSE OF THE 3D DIESEL GENERATOR (DG) BECOMING INOPERABLE. SGT TRAIN A WAS PREVIOUSLY DECLARED INOPERABLE ON 7/23/89, AT 0230 HOURS BECAUSE OF PROBLEMS WITH THE RELATIVE HUMIDITY HEATER BREAKER. THE COMBINATION OF THESE 2 EVENTS CONSTITUTE A VIOLATION OF TECH SPECS WHICH REQUIRE A MINIMUM OF 2 TRAINS OF SGT OPERABLE UNDER THE EXISTING PLAN CONDITIONS. THE 3D DG WAS DECLARED INOPERABLE DUE TO OBSERVED ARCING AND SMOKE FROM INSIDE THE ENGINE CONTROL PANEL AND THE INABILITY TO SHUT DOWN THE DG BY NORMAL MEANS. THE FAILURE TO STOP THE DG WAS ATTRIBUTED TO NORMALLY OPEN CONTACTS ON ONE OF THE PINION FAILURE RELAYS, WHICH BECAME STUCK IN THE CLOSED POSITION. THE DIODE INSTALLED ACROSS THE OPERATING COIL OF THE START FAILURE AUXILIARY (SFA) RELAY FAILED, CAUSING THE OBSERVED ARCING AND SMOKE. THIS ARCING PRODUCED A VOLTAGE TRANSIENT OF SUFFICIENT MAGNITUDE TO FUSE THE PINION FAILURE RELAY CONTACTS IN THE CLOSED POSITION. THE DIODE FAILURE WAS CONSIDERED AN END OF LIFE FAILURE. THE SFA RELAY AND THE FAULTY CONTACTS ON THE PINION FAILURE RELAY WERE REPLACED. THE OPERABILITY SI FOR THE 3D DG WAS SUCCESSFULLY COMPLETED AND SGT TRAIN C WAS RETURNED TO OPERABLE STATUS AT 1000 HOURS ON 7/24/89.

[26] BRUNSWICK 1 DOCKET 50-325 LER 89-013 REV 01
UPDATE ON RWCU ISOLATION DUE TO SUSPECTED HIGH DISCHARGE TEMPERATURE FROM NON-REGEN HEAT EXCHANGER.
EVENT DATE: 041089 REPORT DATE: 081589 NSSS: GE TYPE: BWR
VENDOR: MERCOID CORP.

(NSIC 215019) ON APRIL 10, 1989, AT 1806 HOURS AND ON APRIL 18, 1989, AT 1350 HOURS, THE REACTOR WATER CLEANUP (RWCU) SYSTEM OUTBOARD ISOLATION VALVE CLOSED ON AN ISOLATION SIGNAL. IT IS BELIEVED THAT BOTH ISOLATIONS WERE THE RESULT OF TWO INDEPENDENT CONDITIONS AN INOPERABLE TEMPERATURE SWITCH AND A HIGH DISCHARGE TEMPERATURE FROM THE NONREGENERATIVE HEAT EXCHANGER (WHICH IS A NONENGINEERED SAFETY FEATURE) THE HIGH TEMPERATURE WAS CAUSED BY LOW COOLING WATER FLOW TO THE HEAT EXCHANGER. THE HIGH TEMPERATURE WAS NOT DETECTED PRIOR TO THE ISOLATION BECAUSE A WARNING ANNUNCIATION WAS NOT INITIATED DUE TO AN INOPERABLE TEMPERATURE SWITCH. THE SWITCH FAILED DUE TO AGE, HAS BEEN REPLACED, WILL BE ADDED TO THE PREVENTATIVE MAINTENANCE PROGRAM AND IS NOT Q-LIST. COOLING WATER FLOW IS TEMPERATURE MONITORED INDIRECTLY BY A COMBINATION OF ALARMS, RECORDERS AND DAILY CHECKS. NO FURTHER ACTIONS ARE PLANNED. THESE EVENTS HAD NO SAFETY SIGNIFICANCE. ON APRIL 10, 1989, THE RWCU SYSTEM WAS BEING UTILIZED AS A MEANS OF CONTROLLING VESSEL LEVEL AND ITS ISOLATION RESULTED IN THE AMOUNT OF WATER COVERING THE CORE BEING INCREASED. ON APRIL 18, 1989, THE SYSTEM ACTUATED AS PER DESIGN.

[27] BRUNSWICK 1 DOCKET 50-325 LER 89-016
FAILURE TO PERFORM TECH SPEC REQUIRED STROKE TIME TESTING ON PCIS VALVES.
EVENT DATE: 061489 REPORT DATE: 071489 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: BRUNSWICK 2 (BWR)

(NSIC 214820) ON JUNE 14, 1989, IT WAS DETERMINED THAT THE VALVES LISTED IN TABLE 1 OF THIS LER WERE NOT BEING TESTED IN ACCORDANCE WITH TECHNICAL SPECIFICATIONS; SPECIFICALLY, STROKE TIME TESTING. EER 89-0195 WAS WRITTEN TO DETERMINE THE ADEQUACY OF CURRENT TESTING AND CORRECTIVE ACTIONS NECESSARY FOR ANY DEFICIENCIES. AS A RESULT OF THIS LER, IT WAS DETERMINED THAT ALTHOUGH THE BASIS EXISTS TO DELETE THE VALVES FROM THE TECHNICAL SPECIFICATIONS, THE VALVES WILL BE STROKE TIME TESTED UNTIL REMOVED FROM THE TECHNICAL SPECIFICATIONS. IN ADDITION, IT WAS NOTED THAT THE 1-E11-F049 HAD NOT ALWAYS MET STROKE TIME REQUIREMENTS. THE ROOT CAUSE FOR THIS EVENT WAS THE FAILURE TO ESTABLISH PROPER PROCEDURAL CONTROLS FOR THE TECHNICAL SPECIFICATION REQUIRED TESTING FOR THESE VALVES. CORRECTIVE ACTIONS INCLUDE ENSURING THE F049 MEETS STROKE TIME REQUIREMENTS AND REVISING PROCEDURES TO IMPLEMENT APPROPRIATE STROKE TIME TESTING FOR THE AFFECTED VALVES. THE SAFETY SIGNIFICANCE OF THIS EVENT IS CONSIDERED MINIMAL.

[28] BRUNSWICK 1 DOCKET 50-325 LER 89-017
SPURIOUS ISOLATION OF HIGH PRESSURE COOLANT INJECTION CHANNEL A CAUSED BY SUSPECTED FAILURE OF ROSEMOUNT 510 DU TRIP UNIT.
EVENT DATE: 062489 REPORT DATE: 072189 NSSS: GE TYPE: BWR
VENDOR: ROSEMOUNT, INC.

(NSIC 214885) ON JUNE 24, 1989, AT 2105 HOURS, A HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM, STEAM LINE BREAK DELTA PRESSURE HIGH SIGNAL, A CHANNEL, ISOLATION OCCURRED ON UNIT 1. AT THE TIME OF THE EVENT THE UNIT WAS IN COLD SHUTDOWN. NO OTHER ACTIVITIES WERE IN PROGRESS, AND NO PERSONNEL WERE FOUND IN THE AREA OF THE INSTRUMENTATION WHICH INITIATES THE REFERENCED SIGNAL. IT IS BELIEVED THE ISOLATION WAS CAUSED BY A SPURIOUS TRIP OF REDUNDANT DIVISION 1 ECCS TRIP UNIT 1-E41-PDIM-N004-1. THE REFERENCED TRIP UNIT IS A ROSEMOUNT 510 DU, AND IT HAS BEEN REPLACED. A SIMILAR EVENT WAS REPORTED IN LER 2-89-008. A ROOT CAUSE INVESTIGATION OF THIS EVENT IS CONTINUING. LIMERICK IS ALSO INVESTIGATING SIMILAR PROBLEMS WITH ROSEMOUNT 510 DU TRIP UNITS AND CONTACTS HAVE BEEN ESTABLISHED WITH BOTH LIMERICK AND ROSEMOUNT. A SUPPLEMENT TO THIS REPORT WILL

BE PROVIDED BY NOVEMBER 10, 1989. THIS EVENT WOULD NOT HAVE BEEN MORE SEVERE UNDER REASONABLE AND CREDIBLE ALTERNATIVE CONDITIONS AS THE FAILURE MODE CAUSED THE SYSTEMS TO ACTUATE AS DESIGNED.

[29] BRUNSWICK 2 DOCKET 50-324 LER 88-001 REV 04
UPDATE ON MANUAL REACTOR SCRAM DUE TO DECREASING MAIN CONDENSER VACUUM AND FAILURE OF PRIMARY CONTAINMENT GROUP 2 VALVES G16-F003, F004, F019, AND F020 TO CLOSE ON ISOLATION SIGNAL.
EVENT DATE: 010288 REPORT DATE: 080989 NSSS: GE TYPE: BWR
VENDOR: ASCO VALVES
GENERAL ELECTRIC CO.
GENERAL ELECTRIC CORP. (NUCLEAR ENG DIV)

(NSIC 215044) WHILE PERFORMING A ROUTINE REACTOR SHUTDOWN IN PREPARATION FOR THE UNIT 2 1988 REFUELING/MAINTENANCE OUTAGE, A MANUAL REACTOR PROTECTION SYSTEM TRIP (SCRAM) WAS INITIATED AT 0017 HOURS ON 1/2/88, DUE TO A DECREASING CONDENSER VACUUM. REACTOR POWER WAS APPROX. 55% AND VACUUM HAD DECREASED TO APPROX. -22 INCHES MERCURY. DURING THE EXPECTED VESSEL LEVEL SHRINK FOLLOWING THE SCRAM, VESSEL LEVEL DECREASED TO APPROX. 153 INCHES, THUS INITIATING PRIMARY CONTAINMENT ISOLATION VALVE GROUPS 2, 6, AND 8 AT LOW LEVEL 1 (> 162.5"). OPERATOR VERIFICATION OF THESE VALVE CLOSURES DETERMINED THAT THE GROUP 2 VALVES 2-G16-F003, -F004, -F019, AND -F020 FAILED TO CLOSE. THESE ARE THE INBOARD AND OUTBOARD ISOLATION VALVES FOR THE DRYWELL FLOOR DRAIN SUMP (F003, F004) AND THE DRYWELL EQUIPMENT DRAIN SUMP (F010, F020). THE REMAINING SAFETY SYSTEMS OPERATED AS DESIGNED DURING THIS EVENT. INVESTIGATION OF THE DECREASING VACUUM CONDITION DETERMINED IT RESULTED FROM NUMEROUS LEAKS ON THE MAIN TURBINE AND MAIN STEAM REHEAT INTERCONNECTING PIPING TO THE MAIN TURBINE, WHICH WERE REPAIRED DURING THE UNIT OUTAGE. TO DATE, THE CAUSE OF THE GROUP 2 PCIVS FAILURE TO CLOSE HAS NOT BEEN DETERMINED. BY 2/23/90, A SUPPLEMENT TO THIS REPORT WILL BE ISSUED TO UPDATE THE ROOT CAUSE DETERMINATION OF THE FAILURE OF VALVES TO CLOSE.

[30] BRUNSWICK 2 DOCKET 50-324 LER 89-011
DIVISION I PRIMARY CONTAINMENT GROUP V ISOLATION OF THE REACTOR CORE ISOLATION COOLING SYSTEM AS RESULT OF PERSONNEL ERRORS DURING SURVEILLANCE TESTING.
EVENT DATE: 070389 REPORT DATE: 072889 NSSS: GE TYPE: BWR
VENDOR: SIMPSON COMPANY

(NSIC 214884) AT 0307 HOURS ON 7/3/89 THE UNIT 2 REACTOR CORE ISOLATION COOLING (RCIC) (E51) TURBINE RECEIVED A TRIP SIGNAL AND THE RCIC TURBINE INBOARD STEAM SUPPLY ISOLATION VALVE, E51-F007, CLOSED DUE TO A FALSE PRIMARY CONTAINMENT DIVISION I GROUP 5 ISOLATION SIGNAL DURING THE PERFORMANCE OF MAINTENANCE SURVEILLANCE TEST (MST) RHR21M. UNIT 2 WAS OPERATING AT 61%. THIS EVENT HAD MINIMAL EFFECT UPON PLANT SAFETY. AS THE RESULT OF PERSONNEL ERROR THE INVOLVED TECHNICIANS SELECTED THE WRONG RCIC ISOLATION LOGIC TERMINAL POINT FOR TESTING WITH THE TEST METER (VOLTAGE OHMMETER) NOT SELECTED TO MEASURE VOLTAGE, AS REQUIRED BY THE MST. CONSEQUENTLY, RCIC ISOLATION LOGIC ACTUATION RELAY K33 ENERGIZED TO INITIATE THE EVENT. THE MST WAS THEN SATISFACTORILY COMPLETED AND THE RCIC SYSTEM WAS RETURNED TO STANDBY READINESS AT 0324 HOURS. INVOLVED PERSONNEL WERE COUNSELED ON THE IMPORTANCE OF PROPER IDENTIFICATION OF PLANT EQUIPMENT AND ADHERENCE TO PROCEDURE INSTRUCTIONS. APPROPRIATE REAL-TIME TRAINING CONCERNING THIS EVENT WILL BE COMPLETED BY 9/29/89.

[31] BYRON 1 DOCKET 50-454 LER 89-007
CONTROL ROOM VENTILATION ACTUATION DUE TO VOLTAGE TRANSIENT CAUSED BY LIGHTNING.
EVENT DATE: 080389 REPORT DATE: 082889 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: BYRON 2 (PWR)

(NSIC 215155) ON 8/3/89, AT 2350, LIGHTNING CAUSED TRANSMISSION LINE 0622 TO TRIP

AND INITIATED A VOLTAGE TRANSIENT ON THE ELECTRICAL SYSTEM AT BYRON. PROCESS RADIATION MONITORS OPR31J AND OPR32J SENSED THIS VOLTAGE TRANSIENT AND TRANSFERRED TO INTERLOCK MODE. THE INTERLOCK SIGNAL AUTOMATICALLY TRANSFERRED THE MAIN CONTROL ROOM VENTILATION SYSTEM RECIRCULATION CHARCOAL ADSORBER DAMPERS TO THEIR ENGINEERED SAFETY FEATURE (ESF) POSITIONS. BOTH UNITS MAINTAINED POWER OPERATION (UNIT 1 AT 99% AND UNIT 2 AT 81%) WITHOUT INCIDENT. THE TWO MONITORS RETURNED TO NORMAL OPERATING CONDITION IMMEDIATELY AFTER THE VOLTAGE TRANSIENT PASSED. THE TRANSMISSION LINE WAS REPAIRED BY COMMONWEALTH EDISON'S ROCK RIVER DIVISION. PREVIOUSLY INSTALLED MODIFICATIONS HAVE DECREASED RADIATION MONITOR SENSITIVITY TO DISTRIBUTION SYSTEM TRANSIENTS. THE VOLTAGE DISTURBANCE CAUSED BY LIGHTNING IS AN ACKNOWLEDGED RISK OF TRANSMISSION LINE OPERATION AND NO FURTHER CORRECTIVE ACTIONS ARE WARRANTED. THERE HAVE BEEN A NUMBER OF ESF ACTUATIONS CAUSED BY SYSTEM VOLTAGE TRANSIENTS. THE ROOT CAUSES OF THESE EVENTS HAVE VARIED AND NO TRENDS HAVE BEEN IDENTIFIED. THIS EVENT IS REPORTABLE PER 10CFR50.73(A)(2)(IV) DUE TO THE ACTUATION OF AN ENGINEERED SAFETY FEATURE.

[32]   BYRON 2   DOCKET 50-455   LER 89-001 REV 01
UPDATE ON INADVERTENT SAFETY INJECTION DURING DIESEL GENERATOR OPERABILITY SURVEILLANCE DUE TO PROCEDURAL INADEQUACY.
EVENT DATE: 021189   REPORT DATE: 082289   NSSS: WE   TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 215083) ON 2/11/89 AT APPROXIMATELY 1240, WITH BYRON UNIT 2 IN MODE 6, A SAFETY INJECTION (SI) ACTUATION OF TRAIN A AND B EQUIPMENT OCCURRED DURING THE PERFORMANCE OF A DIESEL GENERATOR OPERABILITY SURVEILLANCE. ACTUATION OF B TRAIN EQUIPMENT WAS EXPECTED DURING THE SURVEILLANCE, BUT THE A TRAIN ACTUATION WAS UNANTICIPATED. THE INADVERTENT SI SIGNAL WAS GENERATED WHEN THE REQUIRED 2 OF 3 COINCIDENCE WAS SATISIFED FOR CONTAINMENT PRESSURE HIGH. ONE HIGH PRESSURE SIGNAL WAS GENERATED BY INSTRUMENT MAINTENANCE PERSONNEL, WHO WERE TESTING PRESSURE CHANNEL 935. CHANNEL 934 GENERATED A HIGH PRESSURE SIGNAL WHEN ITS INSTRUMENT POWER BUS WAS DEENERGIZED DURING THE SURVEILLANCE TEST. THIS OCCURRED BECAUSE THE INSTRUMENT BUS WAS SUPPLIED BY A CONSTANT VOLTAGE TRANSFORMER FROM A BUS THAT WAS INTENTIONALLY DEENERGIZED DURING THE SURVEILLANCE PROCEDURE. THE ONLY UNEXPECTED AUTOMATIC EQUIPMENT ACTUATIONS THAT OCCURRED WERE THE 2A CENTRIFUGAL CHARGING PUMP AND THE 2A DIESEL GENERATOR. ALL OTHER EQUIPMENT WAS EITHER RUNNING OR IN PULL-TO-LOCK. THE SURVEILLANCE WAS SUCCESSFULLY COMPLETED LATER ON 2/11/89. THE ROOT CAUSE OF THE SI WAS INADEQUATE PRECAUTIONS IN THE SURVEILLANCE PROCEDURE. BECAUSE OF THIS DEFICIENCY, OPERATING PERSONNEL INVOLVED IN THIS EVENT DID NOT RECOGNIZE THAT THE INSTRUMENT BUS WOULD BE DEENERGIZED DURING THE SURVEILLANCE TEST.

[33]   CALLAWAY 1   DOCKET 50-483   LER 89-001 REV 01
UPDATE ON AN EMERGENCY DIESEL GENERATOR WAS DISCOVERED TO HAVE BEEN INOPERABLE DUE TO A FAILED MAIN CONTROL BOARD SWITCH.
EVENT DATE: 020789   REPORT DATE: 072189   NSSS: WE   TYPE: PWR
VENDOR: ELECTRO SWITCH CORP.

(NSIC 214907) AT 0400 CST ON 2/15/89, DIESEL GENERATOR (D/G) TRAIN 'B' WAS DECLARED INOPERABLE WHEN THE FUEL OIL TRANSFER PUMP FAILED TO AUTOSTART DURING A SURVEILLANCE TEST. TROUBLESHOOTING DETERMINED THE CAUSE TO BE A FAULTY MAIN CONTROL BOARD (MCB) SWITCH. THE MCB SWITCH HAD DISABLED THE FUEL OIL DAY TANK LEVEL SWITCHES AND THE LOCAL HANDSWITCH. CONSEQUENTLY, THE TRANSFER PUMP WAS NOT CAPABLE OF AUTOMATICALLY SUPPLYING FUEL TO THE DAY TANK. THIS CONDITION HAD EXISTED SINCE 2/7/89. TECHNICAL SPECIFICATION ACTIONS WERE NOT INITIATED UNTIL DISCOVERY AT 0400 ON 2/15/89 AND, THEREFORE, EXCEEDED THE ALLOWED TIME LIMITS. THE PLANT WAS IN MODE 1 - POWER OPERATION AT 100% REACTOR POWER. THE ROOT CAUSE OF THIS EVENT WAS THE FAILURE OF THE MCB SWITCH CONTACTS. THE CAUSE OF THE OPEN SWITCH CONTACTS WAS MOST LIKELY DUE TO MECHANICAL BINDING. THE MCB SWITCH WAS REPLACED. THE PROBLEM WAS NOT DETECTED ON 2/7/89 BECAUSE THE DAY TANK LEVEL HAD

BEEN MANUALLY MAINTAINED ABOVE THE LOW-LEVEL SWITCH SETPOINT. THE EVENT WILL BE REVIEWED BY OPERATIONS PERSONNEL AND THE LOG SHEETS WILL BE REVISED TO ALLOW THE PUMP TO AUTOMATICALLY MAINTAIN THE FUEL LEVEL.

[34] CALLAWAY 1 DOCKET 50-483 LER 89-009
A CONTAINMENT ISOLATION VALVE FAILED TO FULLY CLOSE AGAINST THE SYSTEM DIFFERENTIAL PRESSURE WHEN TESTED.
EVENT DATE: 051189 REPORT DATE: 080889 NSSS: WE TYPE: PWR

(NSIC 215008) AT 1900 CDT ON 5/11/89, CONTAINMENT ISOLATION VALVE, EG-HV-0060, FAILED TO CLOSE COMPLETELY AGAINST THE SYSTEM DIFFERENTIAL PRESSURE (DP) WHEN TESTED AS PART OF THE MOTOR-OPERATED VALVE (MOV) TESTING PROGRAM. THE CLOSE TORQUE SWITCH DEENERGIZED THE MOTOR WITH THE VALVE APPROXIMATELY 22% FROM ITS FULL CLOSED POSITION. THE TORQUE SWITCH SETTING (TSS) WAS INCREASED AND VALVE WAS SATISFACTORILY RETESTED ON 5/12/89. PLANT WAS IN MODE 5 AT 0% REACTOR POWER. ON 7/21/89 IT WAS DETERMINED THAT EG-HV-0060 HAD BEEN INOPERABLE SINCE 3/6/86 WHEN THE VALVE WAS REPACKED. ROOT CAUSE OF THIS EVENT WAS ADDITIONAL PACKING LOADS DEVELOPED ON 3/6/86. REPACKING CAUSED INCREASED STEM FRICTION WHICH PREVENTED THE FULL CLOSURE OF THE VALVE. ON 3/6/86 THERE WERE NO REQUIREMENTS TO VERIFY VALVE OPERABILITY UNDER DP CONDITIONS. THE REQUIRED RETESTS, STROKE TIME TEST AND LOCAL LEAK RATE TEST, HAD BEEN PERFORMED WITH SATISFACTORY RESULTS. THE PROCEDURE WHICH ADMINISTRATIVELY CONTROLS THE TSS'S HAS BEEN REVISED TO REFLECT THE NEW TSS FOR EG-HV-0060. THE UTILITY'S RESPONSE TO GENERIC LETTER 89-10 "SAFETY-RELATED MOTOR-OPERATED VALVE TESTING AND SURVEILLANCE," ISSUED 6/28/89 WILL ADDRESS ADDITIONAL ACTIONS TO BE TAKEN TO PREVENT RECURRENCE OF THIS TYPE OF EVENT. THE UTILITY HAD PREVIOUSLY EXPANDED THE MOV TESTING SCOPE PRIOR TO ISSUANCE OF THE GENERIC LETTER AND AS A RESULT DETECTED THIS DEFICIENCY.

[35] CALLAWAY 1 DOCKET 50-483 LER 89-008
PLANT SHUTDOWN REQUIRED BY THE PLANT'S TECH SPEC, AN ENGINEERED SAFETY FEATURE ACTUATION, AND THE LATE COMPLETION OF A T/S ACTION.
EVENT DATE: 062389 REPORT DATE: 072489 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 214908) AT 0840 CDT ON 6/23/89, CONTROL ROD BANK 'B' FAILED TO MOVE AND WAS DECLARED INOPERABLE IN MODE 1 - POWER OPERATION AT 100% REACTOR POWER. AT 1102, A PLANT SHUTDOWN WAS COMMENCED PER TECHNICAL SPECIFICATION (T/S) 3.1.3.1 ACTION B. AT 1418 AT APPROXIMATELY 3% REACTOR POWER, THE MAIN TURBINE WAS TRIPPED. A PROTECTIVE RELAY FOR GENERATOR OUTPUT BREAKER ERRONEOUSLY SENSED A FLASHOVER AND CLEARED THE 345 KV SWITCHYARD BUS 'B'. THIS RESULTED IN A COMPLETE LOSS OF POWER TO THE 4.16 KV SAFEGUARDS BUS NB01. THIS RESULTED IN AN ENGINEERED SAFETY FEATURE ACTUATION OF THE TURBINE DRIVEN AUXILIARY FEEDWATER PUMP. AT 1419, THE REACTOR WAS MANUALLY TRIPPED AT 2% REACTOR POWER. THE PLANT WAS STABILIZED IN MODE 3 - HOT STANDBY. THE LOSS OF NB01 HAD CAUSED RADIATION MONITORS, GK-RE-05 AND GG-RE-27, TO BECOME INOPERABLE. T/S 3.3.3.1 ACTIONS 27 AND 30 WERE NOT COMPLETED WITHIN THE 1 HOUR TIME LIMIT. THE PLANT WAS RESTARTED AT 0210 ON 6/24/89 AND REACHED 100% REACTOR POWER AT ABOUT 1500 ON 6/25/89. TWO CIRCUIT BOARDS IN THE ROD CONTROL LOGIC CABINET WERE REPLACED. THE CAUSE OF THE LOSS OF NB01 WAS THE FAILURE OF A FLASHOVER RELAY DUE TO A LOOSE SET SCREW. THE RELAY WAS REPAIRED. THE CAUSE OF THE LATE COMPLETION OF T/S 3.3.3.1 ACTIONS 27 AND 30 WAS ATTRIBUTABLE TO COGNITIVE PERSONNEL ERRORS. THESE WERE IDENTIFIED BUT WERE NOT COMPLETED WITHIN THE TIME REMAINING.

[36] CALVERT CLIFFS 1 DOCKET 50-317 LER 89-011
MISSING FIRE PROTECTION DAMPER.
EVENT DATE: 063089 REPORT DATE: 073189 NSSS: CE TYPE: PWR

(NSIC 214883) ON 6/30/89, WHILE PERFORMING AN INSPECTION OF FIRE BARRIER

PENETRATIONS. WITH THE UNIT IN MODE 5 AT 0% POWER, THE FIRE PROTECTION DESIGN ENGINEER DISCOVERED A FIRE DAMPER WAS PHYSICALLY MISSING IN THE SPENT FUEL POOL VENTILATION SYSTEM. THE LOCATION OF THE DAMPER WAS IDENTIFIED ON MECHANICAL PRINTS PRIOR TO ATTEMPTING TO INSPECT IT PHYSICALLY. THE DAMPER SHOULD HAVE BEEN INSTALLED IN THE BARRIER ON ELEVATION 69FT BETWEEN THE ELECTRICAL EQUIPMENT ROOM AND THE SPENT FUEL POOL AREA. THE CAUSE OF THE EVENT WAS AN ERROR IN THE FACILITIES CHANGE REQUEST (FCR) THAT INSTALLED FIRE PROTECTION DAMPERS. AS PART OF CALVERT CLIFF'S COMPLIANCE WITH APPENDIX A TO BRANCH TECHNICAL POSITION APCSB 9.5.1, THE DAMPER SHOULD HAVE BEEN IDENTIFIED AND INSTALLED. IMMEDIATE CORRECTIVE ACTION FOR THIS EVENT INCLUDED ENTERING THE ACTION STATEMENT FOR PENETRATION FIRE BARRIERS AND MAKING A ONE HOUR REPORT UNDER 10CFR72. LONG-TERM CORRECTIVE ACTION INCLUDES INSTALLING THE MISSING DAMPER, EVALUATING THE FIRE PROTECTION PROGRAM, AND IMPROVING THE FCR PROCESS.

[37] CALVERT CLIFFS 1 DOCKET 50-317 LER 89-012
DISCOVERY OF A DISCONNECTED ACTUATOR SOLENOID PARTIALLY DISABLING THE HALON SYSTEM FOR THE 27-FOOT AND 45-FOOT SWITCHGEAR ROOMS.
EVENT DATE: 072089 REPORT DATE: 082189 NSSS: CE TYPE: PWR

(NSIC 215068) ON 7/20/89, WITH UNIT 1 SHUTDOWN, A TECHNICIAN DISCOVERED A SOLENOID DISCONNECTED FROM ITS ASSOCIATED HALON CYLINDER. THIS SOLENOID ACTUATES THE DISCHARGE OF HALON INTO THE 27-FOOT AND 45-FOOT LEVEL OF THE UNIT 1 SWITCHGEAR ROOMS. UPON DISCOVERY OF THE EVENT, THE HALON SYSTEM WAS DECLARED INOPERABLE AND AN HOURLY FIRE WATCH WAS ESTABLISHED. THE SOLENOID WAS RECONNECTED AND VERIFIED OPERABLE BY A FUNCTIONAL TEST. IT CANNOT BE ESTABLISHED HOW THE SOLENOID CAME TO BE DISCONNECTED. NO PHYSICAL EVIDENCE WAS FOUND TO SUGGEST THAT THE SOLENOID WAS INTENTIONALLY TAMPERED WITH. IT CAN ONLY BE THEORIZED THAT SOMEONE PERFORMING AN OPERATIONS OR MAINTENANCE RELATED ACTIVITY IMPROPERLY REMOVED THE SOLENOID FROM SERVICE, INADVERTENTLY LEFT THE SOLENOID DISCONNECTED, AND DID NOT CONTACT A FIRE AND SAFETY TECHNICIAN TO CONDUCT THE REQUIRED TAGOUT. THE DISCONNECTED SOLENOID ONLY PARTIALLY DISABLED THE HALON SYSTEM. SUFFICIENT HALON REMAINED IN TWO ADDITIONAL BANKS TO ADEQUATELY CONTAIN A FIRE UNTIL THE FIRE BRIGADE COULD ARRIVE. CORRECTIVE ACTIONS INCLUDE: INSTALLING IDENTIFICATION TAGS ON THE SOLENOIDS; PLACING WARNING SIGNS ON THE HALON BANKS; REVISING PROCEDURES TO ENSURE ADEQUATE TAGGING AND INDEPENDENT VERIFICATION; AND REQUIRING A MONTHLY SURVEILLANCE OF THE SOLENOID PLACEMENT.

[38] CALVERT CLIFFS 2 DOCKET 50-318 LER 89-009
CLOSED FIRE DAMPER IN PENETRATION VENTILATION EXHAUST SYSTEM.
EVENT DATE: 052289 REPORT DATE: 072189 NSSS: CE TYPE: PWR

(NSIC 214818) ON MAY 22, 1989, WHILE CALVERT CLIFFS UNIT 2 WAS IN MODE 6 AT 0% POWER, THE FIRE PROTECTION DESIGN ENGINEER DISCOVERED A FIRE DAMPER IN THE CLOSED POSITION WHILE PERFORMING A WALKDOWN OF THE PENETRATION VENTILATION EXHAUST SYSTEM FIRE DAMPERS. THE VENTILATION SYSTEM WAS NOT IN OPERATION AND IS ONLY REQUIRED TO BE OPERABLE IN MODES 1-3. THE EVENT IS REPORTABLE BECAUSE THE FAILURE OF THE DAMPER ALONE COULD HAVE PREVENTED THE PENETRATION VENTILATION EXHAUST SYSTEM FROM BEING OPERABLE. THE DAMPER WENT TO THE CLOSED POSITION DUE TO THE FAILURE OF ITS FUSIBLE LINK. CORRECTIVE ACTION INCLUDES: (1) REPLACING THE LINK AND OPENING THE DAMPER; (2) VERIFYING THE DAMPER INSPECTION IS BEING PERFORMED PROPERLY AND ON TIME; (3) IMPROVE AND CLARIFY THE DAMPER INSPECTION PROCEDURE; (4) DETERMINE IF FUSIBLE LINKS SHOULD BE REPLACED ON A CERTAIN FREQUENCY; (5) VERIFY AIR FLOW TEST PROCEDURE ENSURES THE PENETRATION VENTILATION EXHAUST SYSTEM IS PERFORMING ITS DESIGNED FUNCTION; (6) VERIFY OTHER VENTILATION SYSTEM PROCEDURES ENSURE DESIGN INTENT; (7) REVIEW SURVEILLANCE TESTING PROGRAM FOR DESIGN CONCERNS.

[39]   CALVERT CLIFFS 2   DOCKET 50-318   LER 89-011
FRACTURED NUT CAUSED BY INADEQUATE NUT MATERIAL AND DIMENSIONAL PROPERTIES RESULTS IN INOPERABLE STEAM GENERATOR SNUBBERS.
EVENT DATE: 071089   REPORT DATE: 081089   NSSS: CE   TYPE: PWR
OTHER UNITS INVOLVED: CALVERT CLIFFS 1 (PWR)
VENDOR: GRINNELL CORP.

(NSIC 214990) ON JULY 10, 1989 A CONDITION WAS DISCOVERED AT CALVERT CLIFFS WHICH COULD HAVE PREVENTED THE FULFILLMENT OF THE SAFETY FUNCTION OF STRUCTURES OR SYSTEMS NEEDED TO MITIGATE THE CONSEQUENCES OF AN ACCIDENT. WE DISCOVERED THAT THE TIE ROD NUTS HOLDING THE PISTON/CYLINDER ASSEMBLY OF OUR STEAM GENERATOR SNUBBERS INTACT WERE OF INADEQUATE MATERIAL AND DIMENSIONAL PROPERTIES. ONE NUT FRACTURED UPON BEING TORQUED TO ITS SPECIFIED VALUE. AT THE TIME OF THE DISCOVERY BOTH UNITS WERE SHUTDOWN. UNIT 1 WAS IN COLD SHUTDOWN AND UNIT 2 WAS IN REFUELING WITH THE CORE OFF-LOADED. THE CAUSE OF THE NUT FAILURE WAS DETERMINED TO BE INADEQUATE MATERIAL PROPERTIES AND PHYSICAL DIMENSIONS. THE NUTS HAD BEEN INSTALLED WITH A MARGINAL SAFETY FACTOR, WHEN TORQUED TO THEIR SPECIFIED VALUES. ALL OF OUR S/G SNUBBERS AND STEAM GENERATORS WERE DECLARED INOPERABLE. PRIOR TO RESTARTING EITHER UNIT, ALL INADEQUATE S/G SNUBBER NUTS FOR THAT UNIT WILL BE REPLACED WITH NUTS MELTING OR EXCEEDING CURRENT MATERIAL AND DIMENSIONAL STANDARDS FOR SUCH NUTS. WE HAVE ALSO VISUALLY INSPECTED OTHER SNUBBERS FROM THE SAME MANUFACTURER AND HAVE FOUND NO SIMILAR NUTS. THE SNUBBER VENDOR HAS BEEN REQUESTED TO SUPPLY US WITH OTHER INCIDENCES WHERE THIS MATERIAL IS SPECIFIED BY THE PARTICULAR SUPPLIER OF THE S/G SNUBBERS ADDITIONAL CORRECTIVE ACTION WILL BE IMPLEMENTED AS REQUIRED.

[40]   CATAWBA 1   DOCKET 50-413   LER 89-001 REV 02
UPDATE ON TRAIN A BLACKOUT DUE TO INAPPROPRIATELY INSTALLED PROTECTIVE RELAY.
EVENT DATE: 010789   REPORT DATE: 072889   NSSS: WE   TYPE: PWR

(NSIC 214898) ON 1/7/89 AT APPROX. 0302 HRS, 6900V TIE BREAKER ITC-7 TRIPPED AFTER REACTOR COOLANT (NC) PUMP 1C HAD BEEN STARTED AND POWER TO 4160 ESSENTIAL BUS 1ETA WAS LOST. THE LOSS OF POWER CAUSED AN ISOLATION OF THE BUS WITH NO BACK-UP POWER AVAILABLE DUE TO DIESEL GENERATOR 1A BEING OUT OF SERVICE. THE BLACKOUT RESULTED IN A LOSS OF POWER TO RESIDUAL HEAT REMOVAL PUMP 1A, FUEL POOL COOLING PUMP IA, AND COMPONENT COOLING PUMP 1A2. SUBSEQUENTLY, DUE TO A CHARGING FLOW CONTROL VALVE FAILING OPEN, NC SYSTEM PRESSURE INCREASED AND CAUSED A PRESSURIZER PORV TO LIFT 7 TIMES. THE 6900V SWITCHGEAR 1TC IS SEPARATED INTO TWO SIDES WHICH ARE CONNECTED BY NORMALLY OPEN TIE BREAKER 1TC-7. THE TIE BREAKER WAS CLOSED BECAUSE 1T2A WAS OUT OF SERVICE. THE TIE BREAKER TRIPPED OPEN ON OVERCURRENT BECAUSE A GROUND OVERCURRENT RELAY WAS INSTALLED IN THE TIME DELAY OVERCURRENT RELAY LOCATION. THE GROUND OVERCURRENT RELAY WAS NOT DESIGNED FOR THE INRUSH CURRENT CAUSED BY STARTING THE NC PUMP. THE UNIT WAS IN MODE 5, COLD SHUTDOWN, WHEN THE INCIDENT OCCURRED AND HAD OPERATED IN ALL MODES OF OPERATION. THIS INCIDENT IS ATTRIBUTED TO AN INAPPROPRIATE ACTION. IT APPEARS THE RELAYS WERE SWAPPED DURING THE INITIAL INSTALLATION DURING 1978. TIME DELAY AND GROUND OVERCURRENT RELAYS WERE PLACED INTO THE CORRECT LOCATIONS. AN INSPECTION WAS PERFORMED TO VERIFY THE CORRECT RELAYS WERE INSTALLED IN ALL OTHER SIMILAR APPLICATIONS.

[41]   CATAWBA 1   DOCKET 50-413   LER 89-017
MANUAL REACTOR TRIP DUE TO TORN GASKET OF MAIN FEEDWATER VALVE POSITIONER CONTROL AIR MANIFOLD.
EVENT DATE: 062689   REPORT DATE: 072689   NSSS: WE   TYPE: PWR
VENDOR: MOORE PRODUCTS COMPANY
WESTINGHOUSE ELECTRIC CORP.

(NSIC 214899) ON JUNE 26, 1989, AT APPROXIMATELY 0635 HOURS, WITH UNIT 1 IN MODE 1, POWER OPERATION, AT 100% POWER, 1CF28, STEAM GENERATOR (S/G) 1A MAIN FEEDWATER

(CF) CONTROL VALVE, SLOWLY BEGAN CLOSING, CAUSING S/G 1A LEVEL TO DECREASE. THE S/G 1A LEVEL DEVIATION ALARM WAS RECEIVED AND 1CF30 S/G 1A CONTROL BYPASS VALVE, WAS OPENED TO SUPPLY ADDITIONAL CF FLOW. AN OPERATOR WAS DISPATCHED AND DISCOVERED THAT AIR WAS LEAKING FROM THE 1CF28 CONTROL AIR MANIFOLD. A WORK REQUEST WAS ISSUED TO INVESTIGATE AND REPAIR THE AIR LEAK AS S/G 1A LEVEL CONTINUED TO DECREASE. 1CF28 EVENTUALLY CLOSED AND REACTOR POWER WAS REDUCED IN AN ATTEMPT TO MATCH CF FLOW WITH DEMAND. A MANUAL REACTOR TRIP WAS INITIATED AT 0718 HOURS, FROM 86% REACTOR POWER, JUST PRIOR TO A S/G LOW LOW LEVEL REACTOR TRIP SIGNAL. EMERGENCY PROCEDURE EP/1/A/5000/01, REACTOR TRIP OR SAFETY INJECTION, WAS ENTERED. THE TURBINE TRIPPED DUE TO THE REACTOR TRIP AND AUXILIARY FEEDWATER ACTUATION OCCURRED. A GASKET IN THE VALVE'S CONTROL AIR MANIFOLD WAS FOUND TO BE TORN, WHICH APPARENTLY OCCURRED DURING INSTALLATION OF THE GASKET, DUE TO INAPPROPRIATE ACTIONS. THE TORN GASKET WAS REPLACED AND ALL OTHER UNIT 1 S/G CF CONTROL VALVES WERE INSPECTED FOR AIR LEAKS.

[42] CATAWBA 1 DOCKET 50-413 LER 89-018
TECH SPEC VIOLATION AS A RESULT OF A MISSING UNIT VENT CONTINUOUS SAMPLE DUE TO INAPPROPRIATE ACTION.
EVENT DATE: 062889 REPORT DATE: 072689 NSSS: WE TYPE: PWR

(NSIC 214900) ON JUNE 28, 1989, AT 0007 HOURS, A HEALTH PHYSICS (HP) SHIFT TECHNICIAN DISCOVERED, UPON CHANGEOUT OF THE UNIT 1 VENT CONTINUOUS PARTICULATE AND CHARCOAL (P&C) SAMPLES, THAT THE REQUIRED CHARCOAL CARTRIDGE WAS MISSING. THIS RESULTED IN THE FAILURE TO COLLECT A DAILY SAMPLE, AND TO HAVE THE COMPLETE SAMPLE ISOTOPICALLY ANALYZED BY HP, THEREBY VIOLATING TECHNICAL SPECIFICATIONS. FOLLOWING THE DISCOVERY, THE HP TECHNICIAN PROCEEDED TO REMOVE THE PARTICULATE FILTER SAMPLE WHICH WAS FOUND CORRECTLY IN PLACE, AND REPLACED IT WITH A NEW FILTER. IN ADDITION, A NEW CHARCOAL CARTRIDGE WAS PLACED IN THE UNIT 1 VENT SAMPLE HOLDER. THE SAMPLE VALVES WERE OPENED ON THE UNIT VENT RADIATION MONITOR AND AT 0010 HOURS, THE TECHNICIAN RESTARTED THE UNIT 1 VENT SAMPLING PUMP. IT WAS UNCLEAR FOLLOWING THIS INVESTIGATION AS TO WHETHER THE CHARCOAL CARTRIDGE WAS INADVERTENTLY NOT PLACED IN THE SAMPLE HOLDER OR POSSIBLY LATER INAPPROPRIATELY REMOVED. THIS EVENT IS THE RESULT OF AN INAPPROPRIATE ACTION. UNIT 1 WAS IN MODE 1, POWER OPERATION, BEFORE AND DURING THE PERIOD COVERED BY THIS INCIDENT.

[43] CATAWBA 1 DOCKET 50-413 LER 89-019 REV 01
UPDATE ON TWO UNPLANNED AUTOMATIC ALIGNMENTS OF THE NUCLEAR SERVICE WATER SYSTEM TO THE STANDBY NUCLEAR SERVICE WATER POND DUE TO DEFECTIVE PROCEDURES.
EVENT DATE: 062889 REPORT DATE: 081889 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: CATAWBA 2 (PWR)

(NSIC 215081) ON 6/28/89, AT 2137 HRS, WHILE INSTRUMENTATION AND ELECTRICAL (IAE) PERSONNEL WERE IMPLEMENTING A NUCLEAR STATION MODIFICATION (NSM) TO REVISE RN PIT SWAPOVER LOGIC, AN AUTOMATIC ALIGNMENT OF THE NUCLEAR SERVICE WATER (RN) SYSTEM TO THE STANDBY NUCLEAR SERVICE WATER POND (SNSWP) OCCURRED. THE SWAP OF THE RN SYSTEM OCCURRED WHEN AN IAE TECHNICIAN LIFTED A LEAD PER A CALIBRATION PROCEDURE DEVELOPED FOR THE NEW LOGIC, AND ACTUATED THE LOGIC REQUIRED TO REALIGN THE RN SYSTEM TO THE SNSWP. THIS INCIDENT HAS BEEN ATTRIBUTED TO A DEFECTIVE PROCEDURE DUE TO AN ERRONEOUS CIRCUIT ISOLATION. THE APPROPRIATE CALIBRATION PROCEDURES HAVE BEEN REVISED TO IDENTIFY THE PROPER ISOLATION POINTS. ON 7/11/89, AT 1722 HRS, WHILE CONSTRUCTION MAINTENANCE DEPARTMENT (CMD) PERSONNEL WERE CONTINUING TO IMPLEMENT THE SAME NSM, ANOTHER AUTOMATIC ALIGNMENT OF THE RN SYSTEM OCCURRED. THE SWAP OCCURRED WHEN CMD PERSONNEL WERE RESTORING A PORTION OF THE RN SWAP LOGIC CIRCUITRY BACK TO SERVICE PER THE IMPLEMENTATION PROCEDURE. THIS INCIDENT WILL BE REVIEWED WITH THE APPROPRIATE PERSONNEL TO IMPROVE THE DETAIL OF STEPS IN IMPLEMENTATION PROCEDURES. THIS SECOND INCIDENT HAS BEEN ATTRIBUTED TO A DEFECTIVE PROCEDURE DUE TO THE IMPLEMENTATION PROCEDURE NOT REQUIRING THAT THE RESTORATION STEPS BE PERFORMED IN A SPECIFIC ORDER.

[44] CATAWBA 2 DOCKET 50-414 LER 89-007 REV 01
UPDATE ON CONTAINMENT AIR RETURN FAN START DUE TO POSSIBLE INADEQUATE POLICY INVOLVING THE CONTROL OF SLIDING LINKS.
EVENT DATE: 031689 REPORT DATE: 081889 NSSS: WE TYPE: PWR

(NSIC 215082) ON 3/16/89, AT 0433 HRS, UNIT 2 CONTAINMENT AIR RETURN FAN 2A, CARF-2A, STARTED IN RESPONSE TO A CONTAINMENT PRESSURE CONTROL SYSTEM (CPCS) PERMISSIVE FOR HIGH CONTAINMENT PRESSURE. AT THE TIME OF THE INCIDENT, CONTAINMENT PRESSURE WAS INCREASING DUE TO OPERATIONS PERSONNEL PREPARING FOR THE CONTAINMENT INTEGRATED LEAK RATE TEST (ILRT). HOWEVER, THE HIGH-HIGH CONTAINMENT PRESSURE (SP) SETPOINT, WHICH IS ALSO NEEDED TO START THE FAN, SHOULD NOT HAVE BEEN ACTUATED. THREE DAYS PRIOR TO THE INCIDENT, A CALIBRATION WAS PERFORMED ON A TIMER IN THE CARF-2A CONTROL CIRCUITRY. IT IS BELIEVED THAT, DURING THIS CALIBRATION, THE TIMER WAS NOT PROPERLY ISOLATED BY USE OF SLIDING LINKS WHICH SUBSEQUENTLY CAUSED AN SP SIGNAL TO BE UNKNOWINGLY SEALED-IN IN THE CARF-2A CIRCUITRY. THIS INCIDENT HAS BEEN ATTRIBUTED TO A POSSIBLE INADEQUATE POLICY INVOLVING CONTROL OF SLIDING LINKS. SLIDING LINKS WERE NOT PREVIOUSLY REQUIRED TO BE SECURED IN THE OPEN POSITION. IF THE ISOLATION SLIDING LINK UNKNOWINGLY RECLOSED DURING THE CALIBRATION, THE CIRCUIT SEAL-IN WOULD HAVE OCCURRED. UNIT 2 WAS IN MODE 5, COLD SHUTDOWN, WHEN THE SEAL-IN AND THE FAN START OCCURRED. AN IMPROVED METHOD OF PROPER ISOLATION USING SLIDING LINKS HAS BEEN EVALUATED.

[45] CATAWBA 2 DOCKET 50-414 LER 89-016 REV 01
UPDATE ON BOTH CHANNELS OF UPPER RANGE REACTOR VESSEL LEVEL INSTRUMENTATION INOPERABLE DUE TO ASSIGNMENT TO UNQUALIFIED TECHNICIAN.
EVENT DATE: 052889 REPORT DATE: 072089 NSSS: WE TYPE: PWR
VENDOR: AUTOCLAVE ENGINEERS, INC.

(NSIC 214931) ON 6/16/89, WHILE UNIT 2 WAS IN MODE 1, POWER OPERATION, AT 100% POWER, A MONTHLY INSPECTION OF THE UNIT 2 REACTOR VESSEL LEVEL INDICATION SYSTEM (RVLIS) WAS PERFORMED BY INSTRUMENTATION AND ELECTRICAL (IAE) PERSONNEL. INSPECTION OF THE UPPER RANGE LEVEL INDICATIONS REVEALED THAT BOTH CHANNELS WERE READING GREATER THAN 90%, WHEN BOTH SHOULD HAVE BEEN READING 60%. INVESTIGATION INDICATED THAT TWO ISOLATION VALVES FOR A PORTION OF BOTH UPPER RANGE INSTRUMENT LOOPS WERE CLOSED. SUBSEQUENTLY, BOTH TRAINS OF RVLIS WERE DECLARED INOPERABLE. HOWEVER, REVIEW INDICATES RVLIS COULD STILL PERFORM ITS FUNCTIONS REQUIRED FOR MITIGATING INADEQUATE CORE COOLING EVENTS UNDER NUREG-0737, SUPPLEMENT I. THE ISOLATION VALVES WERE SUBSEQUENTLY OPENED. THIS INCIDENT HAS BEEN ATTRIBUTED TO SUPERVISION ASSIGNING AN UNQULAIFIED TECHNICIAN THE TASK OF RETURNING THE UPPER RANGE LEVEL INDICATION TO SERVICE ON 5/28/89. THE TECHNICIAN DID NOT OPEN ALL REQUIRED ISOLATION VALVES. UNIT 2 HAS BEEN IN MODE 5, COLD SHUTDOWN, THROUGH MODE 1 WITH THE VALVES CLOSED. THE SIMILAR UNIT 1 VALVES WERE INSPECTED AND CONFIRMED TO BE OPEN. THIS INCIDENT HAS BEEN REVIEWED WITH ALL IAE SUPERVISION TO EMPHASIZE USE OF QUALIFIED PERSONNEL. THE APPROPRIATE STANDING WORK REQUESTS WILL BE REVISED TO BETTER DESCRIBE THE EXPECTED ACTIONS.

[46] CATAWBA 2 DOCKET 50-414 LER 89-012
OVERTEMPERATURE DELTA TEMPERATURE TECH SPEC VIOLATION DUE TO INAPPROPRIATE ACTIONS AND UNCLEAR ACCEPTANCE CRITERIA.
EVENT DATE: 070389 REPORT DATE: 080189 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 214964) ON JULY 3, 1989, AT APPROXIMATELY 2100 HOURS, WITH UNIT 2 IN MODE 1, POWER OPERATION, AT 100% POWER, THE LOOP C OVERTEMPERATURE DELTA TEMPERATURE (OTDT) COMPUTER ALARM WAS RECEIVED DURING PERFORMANCE OF THE MODE 1 PERIODIC SURVEILLANCE ITEMS PROCEDURE. A MANUAL OTDT CALCULATION WAS PERFORMED TO DETERMINE IF THE CHANNEL WAS OPERAB1E. THE ACCEPTANCE CRITERIA OF THE SURVEILLANCE PROCEDURE WERE INCORRECTLY APPLIED AND THE OTDT CHANNEL WAS DETERMINED TO BE OPERABLE. ON JULY 4, 1989, AT APPROXIMATELY 1050 HOURS, THE

NEXT SHIFT PERFORMED THE SAME MANUAL OTDT CALCULATION, AFTER OBSERVING THE COMPUTER ALARM, AND DETERMINED THE CHANNEL TO BE INOPERABLE. THE CHANNEL WAS PLACED IN THE TRIPPED POSITION, AS REQUIRED BY TECHNICAL SPECIFICATIONS, AND A WORK REQUEST WAS ISSUED TO REPAIR THE CHANNEL. THE INAPPROPRIATE ACTION OF NOT DECLARING THE CHANNEL INOPERABLE DURING THE FIRST OTDT CALCULATION IS ATTRIBUTED TO IMPROPERLY FOLLOWING THE ACCEPTANCE CRITERIA OF THE PROCEDURES DUE TO MISINTERPRETATION. CONTRIBUTING CAUSES WERE A FAILED CARD IN THE PROCESS CONTROL CABINET AND UNCLEAR ACCEPTANCE CRITERIA IN THE PROCEDURE. A PROCEDURE REVISION IS PLANNED TO CLARIFY THE ACCEPTANCE CRITERIA. THE DEFECTIVE CONTROL CARD HAS BEEN REPLACED. UNIT 2 WAS IN MODE 1 DURING THIS INCIDENT.

[47] CLINTON 1 DOCKET 50-461 LER 88-025 REV 01
UPDATE ON LOSS OF FEEDWATER HEATING SYSTEM TRANSIENT OUTSIDE DESIGN BASIS DUE TO INADEQUATE COMMUNICATION BETWEEN THE ARCHITECT ENGINEER AND THE NUCLEAR STEAM SUPPLY SYSTEM SUPPLIER.
EVENT DATE: 072888 REPORT DATE: 083089 NSSS: GE TYPE: BWR
VENDOR: BAILEY CONTROLS CO.

(NSIC 215100) ON JULY 28, 1988, CLINTON POWER STATION (CPS) EXPERIENCED A PARTIAL LOSS OF FEEDWATER (FW) HEATING. THE FW TEMPERATURE DROP, EXCLUDING THE CHANGE CAUSED BY A REDUCTION IN POWER, WAS GREATER THAN 102, BUT LESS THAN 112, DEGREES FAHRENHEIT (F). THE DESIGN BASIS LOSS OF FW HEATING TRANSIENT FOR CPS IS BASED ON A MAXIMUM TEMPERATURE TRANSIENT OF 100 DEGREES F. THE EXPECTED OCCURRENCE FREQUENCY OF THIS TRANSIENT IS ONCE PER TWENTY YEARS. THE CAUSE OF THE LOSS OF FW HEATING WAS THE INAPPROPRIATE SETTING OF THE FW HEATER LEVEL CONTROLLERS. THE CAUSE OF EXCEEDING THE DESIGN BASIS IS ATTRIBUTED TO THE FAILURE OF THE FW HEATING SYSTEM DESIGN TO MEET DESIGN REQUIREMENTS. THIS WAS CAUSED BY A LACK OF ADEQUATE COMMUNICATION BETWEEN THE NUCLEAR STEAM SUPPLY SYSTEM (NSSS) SUPPLIER AND THE ARCHITECT ENGINEER REGARDING THE NSSS DESIGN REQUIREMENTS FOR THE FW HEATING SYSTEM. FEEDWATER HEATING SYSTEM DESIGN CHANGES, INCLUDING CHANGES TO THE LEVEL TRIP SETPOINT FOR CLOSING THE EXTRACTION STEAM VALVES AND REPLACING POWER SUPPLY FUSES, HAVE BEEN MADE TO ENSURE THAT THE DESIGN BASIS IS MET. THIS REPORT ALSO INCLUDES INFORMATION REQUIRED UNDER THE PROVISIONS OF 10CFR21 RELATING TO THE POTENTIAL FOR EXCEEDING A TECHNICAL SPECIFICATION SAFETY LIMIT. THIS EVENT POSED NO UNDUE RISK TO PUBLIC HEALTH AND SAFETY.

[48] CLINTON 1 DOCKET 50-461 LER 89-027
LACK OF UNDERSTANDING OF THE EFFECT OF A MISSING SCREW ON SEISMIC QUALIFICATION RESULTS IN INOPERABILITY OF CONTROL ROOM HEATING, VENTILATING, AND AIR CONDITIONING SYSTEM.
EVENT DATE: 041189 REPORT DATE: 072889 NSSS: GE TYPE: BWR

(NSIC 214967) ON 6/28/89, ILLINOIS POWER COMPANY IDENTIFIED THAT BETWEEN 4/11/89 AND 5/18/89, THE PLANT HAD OPERATED IN A CONDITION PROHIBITED BY TECH SPECS (TS). ON 4/3/89, A SCREW SECURING CONTROL ROOM HEATING, VENTILATING AND AIR CONDITIONING SYSTEM (VC) TIME DELAY RELAYS 0KY-VC623B AND 0KY-VC624B WAS FOUND TO BE MISSING. REVIEW IDENTIFIED THAT, BECAUSE THE SCREW WAS MISSING, THE RELAYS WERE NOT SEISMICALLY QUALIFIED AND THEREFORE VC TRAIN "B" WAS INOPERABLE. TS 3.7.2 REQUIRES THAT WHEN ONE TRAIN OF VC IS INOPERABLE FOR GREATER THAN 7 DAYS, THE OPERABLE TRAIN BE RUN CONTINUOUSLY IN THE HIGH RADIATION MODE. THE REQUIREMENTS OF THIS TS WERE NOT MET FROM 4/11/89, WHEN VC TRAIN "A" WAS SHIFTED FROM THE HIGH RADIATION TO THE NORMAL MODE OF OPERATION, UNTIL 5/18/89, WHEN THE SCREW WAS REPLACED. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO THE SHIFT SUPERVISORS LACK OF UNDERSTANDING OF THE EFFECT THE MISSING SCREW HAD ON THE SEISMIC QUALIFICATION OF VC TRAIN "B". INCONSISTENT REQUIREMENTS FOR REPORTING DISCREPANT CONDITIONS SPECIFIED IN 2 PROCEDURES CONTRIBUTED TO THE EVENT. A REVIEW WILL BE PERFORMED TO DETERMINE IF OTHER DISCREPANT CONDITIONS EXIST WHICH AFFECT SYSTEM OPERABILITY. TRAINING WILL BE PROVIDED TO APPROPRIATE PERSONNEL ON SEISMIC AND ENVIRONMENTAL QUALIFICATION AND THE 2 PROCEDURES WILL BE REVISED.

[49] CLINTON 1 DOCKET 50-461 LER 89-021 REV 01
UPDATE ON FAILURE OF SHIFT SUPERVISOR TO IDENTIFY THAT SYSTEM ISOLATION WOULD REMOVE THE REACTOR WATER CLEANUP SYSTEM CONDUCTIVITY MONITOR FROM SERVICE RESULTS IN A MISSED SAMPLE.
EVENT DATE: 052189 REPORT DATE: 072889 NSSS: GE TYPE: BWR

(NSIC 214940) ON 5/21/89, PREPARATIONS WERE UNDERWAY TO PLACE THE PLANT IN MODE 2 (STARTUP). THE REACTOR WATER CLEANUP SYSTEM (RWCU) DIFFERENTIAL FLOW INSTRUMENTS WERE INOPERABLE, AS REQUIRED BY TECH SPECS (TS) FOR MODE 2, RWCU WAS SHUT DOWN AND ISOLATED. PRIOR TO THIS EVOLUTION THE SHIFT SUPERVISOR (SS) NOTIFIED CHEMISTRY THAT RWCU WAS BEING SHUT DOWN. ISOLATING RWCU REMOVED THE ONLY AVAILABLE CONTINUOUS CONDUCTIVITY MONITOR FROM SERVICE. THE TS REQUIRE CONDUCTIVITY SAMPLES TO BE TAKEN AT LEAST ONCE PER 4 HRS IN MODE 2 WHEN NO CONTINUOUS CONDUCTIVITY MONITORS ARE AVAILABLE; THIS WAS NOT DONE. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO THE FAILURE OF THE SS TO IDENTIFY THAT NO CONTINUOUS CONDUCTIVITY MONITOR WAS IN SERVICE, TO A BREAKDOWN IN COMMUNICATIONS BETWEEN CHEMISTRY AND THE SS, AND TO THE FAILURE TO PROMPTLY REPAIR THE INOPERABLE REACTOR RECIRCULATION SYSTEM CONTINUOUS CONDUCTIVITY MONITOR. CORRECTIVE ACTIONS INCLUDE BRIEFING SSS AND CHEMISTRY SUPERVISORY PERSONNEL ON THIS EVENT, REVIEWING THE SYSTEM USED TO TRACK EQUIPMENT OUT-OF-SERVICE, ESTABLISHING A PROGRAM FOR ON-LINE CHEMISTRY MONITORS AND RECORDERS THAT WILL IMPROVE THE AVAILABILITY FACTOR FOR THIS EQUIPMENT, AND ESTABLISHING AN INDEPENDENT GROUP TO REVIEW, SCHEDULE AND ENSURE COMPLETION OF WORK ACTIVITIES IN A MANNER THAT WILL BEST SUPPORT THE CLINTON POWER STATION.

[50] CLINTON 1 DOCKET 50-461 LER 89-024
INADEQUATE PROCEDURE LEADS TO MISCALIBRATION OF REACTOR WATER CLEANUP LEAK DETECTION MODULES RESULTING IN OPERATION PROHIBITED BY TECH SPECS.
EVENT DATE: 061689 REPORT DATE: 071889 NSSS: GE TYPE: BWR
VENDOR: RILEY COMPANY, THE

(NSIC 214832) ON JUNE 19, 1989, RILEY POINT MODULES 1E31-N621A, THE DIVISION 1 REACTOR WATER CLEANUP SYSTEM (RWCU) PUMP ROOM A AREA TEMPERATURE MODULE AND 1E31-N620A, THE WEST RWCU HEAT EXCHANGER ROOM TEMPERATURE MODULE FAILED THEIR CHANNEL CHECKS. INVESTIGATION INTO THE FAILURES REVEALED THAT THE MODULES HAD BEEN MISCALIBRATED ON JUNE 16, 1989. THE CONTROL AND INSTRUMENTATION (C&I) TECHNICIAN WHO PERFORMED THE CALIBRATION ON JUNE 16, 1989, USED AN INCORRECT MODE SETTING ON THE POTENTIOMETER USED TO PERFORM THE CALIBRATION. TECHNICAL SPECIFICATION 3.3.2.C.1 REQUIRES THAT WHEN EITHER OF THESE MODULES ARE INOPERABLE THEY BE RESTORED TO OPERABLE STATUS WITHIN TWO HOURS OR THAT THE RWCU SYSTEM BE ISOLATED. THIS TECHNICAL SPECIFICATION WAS NOT MET FOR APPROXIMATELY SEVENTY-TWO HOURS. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO AN INADEQUATE PROCEDURE. CONTRIBUTING TO THE EVENT WAS A PERSONNEL ERROR AND THE LACK OF A CRITERIA FOR DETERMINING IF THE RESULTS OF CHANNEL CHECKS ARE SATISFACTORY. CORRECTIVE ACTIONS INCLUDE BRIEFING C&I TECHNICIANS ON THE NEED TO REFER TO OPERATION MANUALS WHEN UNFAMILIAR WITH EQUIPMENT AND TO CONTACT SUPERVISION WHEN PROCEDURES ARE INADEQUATE. GUIDANCE ON ACCEPTANCE CRITERIA FOR CHANNEL CHECKS OF TEMPERATURE MODULES WAS PROVIDED TO OPERATIONS.

[51] CLINTON 1 DOCKET 50-461 LER 89-025
INADEQUATE CORRECTIVE ACTION FOR A PREVIOUS SIMILAR EVENT RESULTS IN A MISSED SURVEILLANCE OF ROD PATTERN CONTROL SYSTEM HIGH POWER SETPOINT.
EVENT DATE: 062189 REPORT DATE: 071989 NSSS: GE TYPE: BWR

(NSIC 214833) ON 6/22/89, ILLINOIS POWER DETERMINED THAT A CHANNEL FUNCTIONAL TEST (CFT) OF THE INSTRUMENTATION WHICH PROVIDES THE ROD PATTERN CONTROL SYSTEM (RPCS) HIGH POWER SETPOINT (HPSP) TRIP FUNCTION HAD NOT BEEN PERFORMED AS REQUIRED WHEN THE POWER RANGE ABOVE THE RPCS LOW POWER SETPOINT WAS ENTERED ON 6/21/89. THIS CFT WAS NOT PERFORMED BECAUSE THE INTEGRATED OPERATING PROCEDURE

IN USE AT THE TIME INCORRECTLY DID NOT REQUIRE PERFORMANCE OF THE CFT. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO INADEQUATE CORRECTIVE ACTION FOR A PREVIOUS SIMILAR EVENT. THE CORRECTIVE ACTIONS FOR THE PREVIOUS EVENT DID NOT REQUIRE AN INVESTIGATION OF OTHER PROCEDURES FOR SIMILAR DEFICIENCIES AND DID NOT PROVIDE A CLEAR INTERPRETATION OF A CONFUSING TECH SPEC AS AN INTERIM ACTION UNTIL A PROPOSED TECH SPEC AMENDMENT COULD BE APPROVED. CORRECTIVE ACTIONS FOR LER 89-025-00 INCLUDE BRIEFING LICENSED OPERATIONS SHIFT PERSONNEL ON THE TRIGGERS ASSOCIATED WITH THE HPSP CFT, REVIEWING PROCEDURES TO ENSURE TRIGGERS ASSOCIATED WITH THE HPSP ARE CORRECTLY ADDRESSED, REVIEWING SIMILAR LERS TO VERIFY THAT THE ROOT CAUSES/CORRECTIVE ACTIONS WERE ADEQUATE, REVISING PROCEDURES TO REQUIRE AN ADDITIONAL REVIEW OF CHANGES TO STATION PROCEDURE SECTIONS IF THOSE SECTIONS IMPLEMENT TECH SPEC REQUIREMENTS, AND REVIEWING PROCEDURES WHICH IMPLEMENT TECH SPECS.

[52]   CLINTON 1   DOCKET 50-461   LER 89-026
DEFICIENT INTEGRATED OPERATING PROCEDURE RESULTS IN MISSED VERIFICATION OF TRIP SETPOINTS FOR AVERAGE POWER RANGE MONITORS.
EVENT DATE: 062189   REPORT DATE: 072689   NSSS: GE   TYPE: BWR

(NSIC 214905) ON JUNE 26, 1989, ILLINOIS POWER (IP) DETERMINED THAT THE TECHNICAL SPECIFICATION REQUIREMENTS ASSOCIATED WITH THE SETPOINT VERIFICATIONS FOR THE AVERAGE POWER RANGE MONITOR (APRM) FLOW-BIASED SIMULATED THERMAL POWER-HIGH AND NEUTRON FLUX-HIGH REACTOR PROTECTION SYSTEM TRIP FUNCTIONS HAD NOT BEEN FULLY MET SINCE INITIAL PLANT OPERATION. THE SETPOINT VERIFICATIONS WERE NOT BEING PERFORMED AS REQUIRED PRIOR TO PLANT ENTRY INTO MODE 1 (POWER OPERATION). THE CAUSE OF THIS EVENT IS ATTRIBUTED TO A DEFICIENT PROCEDURE, THE MODE 1 CHECKLIST, WHICH MAY HAVE RESULTED FROM CONFUSION OF TECHNICAL SPECIFICATION REQUIREMENTS DURING PROCEDURE DEVELOPMENT. THE MODE 1 CHECKLIST WAS INCORRECT BECAUSE IT DID NOT REQUIRE THE SETPOINT VERIFICATIONS TO BE PERFORMED PRIOR TO THE PLANT ENTERING MODE 1. CORRECTIVE ACTIONS INCLUDE REVISING THE MODE 1 CHECKLIST TO REQUIRE THE APRM SETPOINT VERIFICATIONS PRIOR TO ENTRY OF THE PLANT INTO MODE 1, BRIEFING OPERATIONS SHIFT SUPERVISORS ON THE LESSONS LEARNED FROM THIS EVENT, REVISING ADMINISTRATIVE PROCEDURES TO REQUIRE AN ADDITIONAL REVIEW OF CHANGES TO STATION PROCEDURE SECTIONS IF THOSE SECTIONS IMPLEMENT TECHNICAL SPECIFICATION REQUIREMENTS, AND REVIEWING APPROPRIATE PROCEDURES WHICH IMPLEMENT TECHNICAL SPECIFICATIONS.

[53]   CLINTON 1   DOCKET 50-461   LER 89-028
WATER INTRUSION INTO MAIN POWER TRANSFORMER SUDDEN PRESSURE SENSOR RELAY CAUSES CORROSION AND RESULTS IN RELAY FAILURE, TURBINE GENERATOR TRIP AND REACTOR SCRAM.
EVENT DATE: 062889   REPORT DATE: 072789   NSSS: GE   TYPE: BWR
VENDOR: QUALITROL CORP.

(NSIC 214906) ON JUNE 28, 1989, WITH THE PLANT IN MODE 1 (POWER OPERATION), THE "C" PHASE MAIN POWER TRANSFORMER (MPT) SUDDEN PRESSURE SENSOR RELAY MALFUNCTIONED CAUSING A TRIP OF THE MAIN GENERATOR. THE TRIP OF THE MAIN GENERATOR RESULTED IN A TURBINE TRIP AND AN AUTOMATIC REACTOR SCRAM BECAUSE OF THE TURBINE CONTROL VALVE FAST CLOSURE SIGNAL. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO A SPURIOUS SIGNAL FROM THE MALFUNCTIONING SUDDEN PRESSURE SENSOR RELAY. THE SUDDEN PRESSURE SENSOR RELAY MALFUNCTIONED BECAUSE OF INTERNAL CORROSION RESULTING FROM WATER INTRUSION INTO THE RELAY. THE SUDDEN PRESSURE SENSOR RELAY WAS REPLACED WITH A SENSOR RELAY WHICH HAS AN AIR VENT TO PREVENT MOISTURE BUILDUP INSIDE THE RELAY. THE SUDDEN PRESSURE SENSOR RELAYS WERE REPLACED IN THE OTHER TWO MAIN POWER TRANSFORMERS AND ALSO IN THE RESERVE AUXILIARY TRANSFORMER AND THE EMERGENCY RESERVE AUXILIARY TRANSFORMER.

[54] CLINTON 1 DOCKET 50-461 LER 89-029
MECHANICAL FAILURE OF RUBBER EXPANSION JOINT BETWEEN THE "A" LOW PRESSURE TURBINE AND THE MAIN CONDENSER RESULTS IN LOSS OF CONDENSER VACUUM AND MANUAL REACTOR SCRAM.
EVENT DATE: 071489 REPORT DATE: 080989 NSSS: GE TYPE: BWR
VENDOR: WELLMAN THERMAL SYSS CORP

(NSIC 215007) ON JULY 14, 1989, WITH THE PLANT AT THIRTY-NINE PERCENT REACTOR POWER, A MANUAL REACTOR SCRAM WAS INITIATED IN ANTICIPATION OF AN AUTOMATIC SCRAM THAT WAS IMMINENT BECAUSE MAIN CONDENSER VACUUM WAS DECREASING. (A LOSS OF MAIN CONDENSER VACUUM CAUSES A TURBINE TRIP AND RESULTS IN AN AUTOMATIC REACTOR SCRAM.) THE REACTOR HAD BEEN OPERATING AT 100 PERCENT REACTOR POWER WHEN CONDENSER VACUUM BEGAN DECREASING. AFTER THE REACTOR SCRAM, GROUPS 2, 3 AND 20 AUTOMATIC CONTAINMENT ISOLATIONS OCCURRED BECAUSE OF LOW REACTOR VESSEL WATER LEVEL. GROUP 1 CONTAINMENT ISOLATION VALVES WERE MANUALLY CLOSED IN ANTICIPATION OF THE AUTOMATIC ISOLATION THAT WOULD OCCUR BECAUSE OF LOW CONDENSER VACUUM. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO A MECHANICAL FAILURE OF THE RUBBER EXPANSION JOINT LOCATED BETWEEN THE "A" LOW PRESSURE TURBINE AND THE MAIN CONDENSER. THE EXPANSION JOINT FAILED BECAUSE OF AGE, OVERTORQUING OF THE ATTACHMENT NUTS OF THE EXPANSION JOINT CLAMP ASSEMBLY, AND STEAM EXPOSURE THAT RESULTED FROM A DETACHED PROTECTIVE COVER. CORRECTIVE ACTION FOR THIS EVENT INCLUDED REPLACING THE RUBBER EXPANSION JOINTS BETWEEN BOTH THE "A" AND "B" LOW PRESSURE TURBINES AND THE MAIN CONDENSER, TORQUING THE ATTACHMENT NUTS OF THE CLAMP ASSEMBLY TO VENDOR RECOMMENDED VALUES, AND REINFORCING THE WELDS OF THE PROTECTIVE COVER.

[55] CLINTON 1 DOCKET 50-461 LER 89-030
FAILURE TO PROMPTLY REPAIR FLUSH WATER SUPPLY VALVE CAUSES LOW REACTOR WATER LEVEL AND RESULTS IN A REACTOR PROTECTION SYSTEM ACTUATION WITH THE REACTOR SHUTDOWN.
EVENT DATE: 071589 REPORT DATE: 081489 NSSS: GE TYPE: BWR

(NSIC 215035) ON JULY 15, 1989, THE PLANT WAS IN MODE 3 (HOT SHUTDOWN). AT 1940 HOURS, OPERATORS OPENED THE RESIDUAL HEAT REMOVAL SYSTEM (RHR) SHUTDOWN COOLING ISOLATION VALVES TO PLACE LOOP A OF RHR IN THE SHUTDOWN COOLING MODE. REACTOR WATER LEVEL RAPIDLY DECREASED FROM THIRTY-EIGHT INCHES TO ONE AND ONE-HALF INCHES. A SCRAM SIGNAL WAS AUTOMATICALLY INITIATED. COINCIDENTALLY, A SPURIOUS HIGH DIFFERENTIAL FLOW SIGNAL CAUSED DIVISION II REACTOR WATER CLEANUP SYSTEM (RWCU) ISOLATION VALVES TO SHUT. INVESTIGATION INTO THIS EVENT REVEALED THAT, BECAUSE THE RHR PUMP A DISCHARGE HEADER FLUSH WATER SUPPLY VALVE 1E12-063A WAS LEAKING PAST ITS SEAT, THE RHR FLUSH SUPPLY ISOLATION VALVE 1CY045 WAS SHUT. SINCE VALVE 1CY045 WAS SHUT, NO WATER WAS SUPPLIED TO THE RHR SYSTEM DURING A PORTION OF THE FLUSH AND WARMUP PERFORMED PRIOR TO PLACING RHR IN THE SHUTDOWN COOLING MODE. THEREFORE, LOOP A OF RHR WAS PARTIALLY DRAINED. WHEN THE RHR LOOP A ISOLATION VALVES WERE OPENED, THE REACTOR WATER ENTERED THE RHR SYSTEM AND REACTOR WATER LEVEL DECREASED. THE CAUSE OF THIS EVENT IS ATTRIBUTED TO THE FAILURE TO PROMPTLY REPAIR VALVE 1E12-063A. CORRECTIVE ACTIONS INCLUDE REPAIRING VALVE 1E12-063A AND EVALUATING SYSTEM ENHANCEMENTS TO REDUCE SPURIOUS RWCU TRIPS.

[56] CONNECTICUT YANKEE DOCKET 50-213 LER 89-011
SURVEILLANCE FREQUENCY EXCEEDED FOR AUXILIARY FEEDWATER INITIATION TEST.
EVENT DATE: 071789 REPORT DATE: 081189 NSSS: WE TYPE: PWR

(NSIC 214974) ON JULY 17, 1989 AT 0800 HOURS WITH THE PLANT OPERATING IN MODE 1 AT 100 PERCENT POWER, A SUPERVISORY REVIEW REVEALED THAT SURVEILLANCE TEST PROCEDURE SUR 5.2-65 "SAFETY GRADE AUTOMATIC INITIATION OF FEEDWATER CHECK" HAD NOT BEEN PERFORMED WITHIN 31 DAYS PLUS 25 PERCENT AS REQUIRED BY TECHNICAL SPECIFICATION 4.8.1.B. THE SURVEILLANCE WAS OVERDUE BY 8 DAYS. THE SURVEILLANCE WAS PERFORMED IMMEDIATELY (JULY 17, 1989) AND THE "AS FOUND" CRITERIA WAS WITHIN THE ACCEPTANCE CRITERIA INDICATING THAT THE AFFECTED EQUIPMENT WAS OPERABLE

THROUGHOUT THE PERIOD IN QUESTION. THE ROOT CAUSE OF THIS EVENT WAS FAILURE TO FOLLOW ADMINISTRATIVE PROCEDURES, AGGRAVATED BY AN ADMINISTRATIVE CONTROL THAT WAS NOT USER FRIENDLY. CORRECTIVE ACTION CONSISTS OF DISCIPLINARY MEASURES AND HUMAN FACTORS UPGRADES WHICH WILL BE COMPLETED BY OCTOBER 1, 1989. THIS EVENT IS BEING REPORTED UNDER 10CFR50.73(A)(2)(I)(B) SINCE IT INVOLVES A CONDITION PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATIONS.

[57] CONNECTICUT YANKEE DOCKET 50-213 LER 89-012
SURVEILLANCE FREQUENCY EXCEEDED FOR CO(2) FIRE SUPPRESSION SYSTEMS.
EVENT DATE: 072889 REPORT DATE: 082489 NSSS: WE TYPE: PWR

(NSIC 215106) ON JULY 28, 1989 WITH THE PLANT IN MODE 1 AND OPERATING AT 100% POWER, AN INTERNAL AUDIT DISCLOSED THAT SURVEILLANCE PROCEDURES SUR 5.5-19 AND SUR 5.5-20 DEALING WITH WEIGHING OF THE CO2 BOTTLES FOR THE PRIMARY AUXILIARY BUILDING VENTILATION SYSTEM CHARCOAL FILTER FIRE PROTECTION SYSTEM AND CABLE VAULT FIRE PROTECTION SYSTEM, RESPECTIVELY, HAD NOT BEEN PERFORMED WITHIN THE PRESCRIBED SURVEILLANCE INTERVAL IN SEPTEMBER 1987. THE SURVEILLANCES WERE PERFORMED IN JANUARY 1988 AT WHICH TIME THE "AS FOUND" WEIGHTS WERE WITHIN THE SURVEILLANCE PROCEDURE ACCEPTANCE CRITERIA. THEREFORE, THE AFFECTED FIRE PROTECTION SYSTEMS REMAINED OPERABLE. THE APPARENT ROOT CAUSE OF THE EVENT WAS AN ADMINISTRATIVE BREAKDOWN OF THE SURVEILLANCE TRACKING SYSTEM WITHIN THE PLANT MAINTENANCE DEPARTMENT. CORRECTIVE ACTION WILL CONSIST OF IMPLEMENTATION OF A MAINTENANCE DEPARTMENT PROCEDURE REQUIRING A MONTHLY REVIEW OF SURVEILLANCE PERFORMANCE STATUS. THIS WILL BE IN ADDITION TO THE EXISTING DEPARTMENTAL SURVEILLANCE TRACKING SYSTEM. THE NEW PROCEDURE WILL BE IMPLEMENTED BY OCTOBER 1, 1989. THIS EVENT IS BEING REPORTED UNDER 10CFR50.73(A)(2)(I)(B) SINCE IT IS A CONDITION PROHIBITED BY THE PLANT'S TECHNICAL SPECIFICATIONS.

[58] COOK 1 DOCKET 50-315 LER 89-009
REQUIRED POST-MAINTENANCE TESTING NOT PERFORMED DUE TO PERSONNEL ERROR PRIOR TO ENTRY INTO A MODE FOR WHICH THE EQUIPMENT WAS REQUIRED TO BE OPERABLE.
EVENT DATE: 062089 REPORT DATE: 080489 NSSS: WE TYPE: PWR

(NSIC 214952) THIS IS AN INTERIM REPORT PENDING COMPLETION OF ANALYSIS OF EVENT AND DETERMINATION OF PREVENTIVE ACTION. ON JULY 5, WITH UNIT 1 OPERATING AT 32% REACTOR POWER IT WAS DISCOVERED DURING A REVIEW OF JOB ORDER PACKAGES THAT POST-MAINTENANCE TESTING DOCUMENTATION FOR A JOB ORDER ON VALVE NO. 1-CS-442-2, UNIT 1 REACTOR COOLANT PUMP NO. 2 SEAL WATER INJECTION CONTAINMENT ISOLATION CHECK VALVE HAD NOT BEEN SIGNED OFF AS COMPLETED. A RECORD SEARCH WAS CONDUCTED AND NO EVIDENCE COULD BE FOUND TO INDICATE THAT THE POST MAINTENANCE TESTING HAD BEEN PERFORMED. THIS PLACED THE UNIT IN A CONDITION OUTSIDE OF THE OPERABILITY REQUIREMENTS OF TECH SPEC 3.6.1.2 LIMITING CONDITION FOR OPERATION AND A SHUTDOWN WAS INITIATED PER THE REQUIREMENTS OF TECH SPEC 3.0.3 AT 1545 HOURS ON 7/5/89. AN UNUSUAL EVENT WAS DECLARED AT 1605 HOURS AND THE NRC WAS NOTIFIED VIA THE ENS AT 1613 HOURS. THE RCS WAS PLACED IN HOT STANDBY AT 2132 HOURS, ENTERED HOT SHUTDOWN AT 0305 ON 7/6/89 AND COLD SHUTDOWN AT 0909 HOURS THE SAME DAY. UPON DISCOVERY OF THIS CONDITION THE UNIT WAS PLACED IN COLD SHUTDOWN AND THE VALVE WAS TESTED. THE TEST INDICATED THAT THE VALVE WAS FUNCTIONING SATISFACTORILY; THEREFORE, CONTAINMENT INTEGRITY HAD NOT BEEN COMPROMISED AND NO OTHER SAFETY CONCERNS WERE NOTED. ADDITIONAL DETAILS WILL BE INCLUDED IN A SUPPLEMENTAL REPORT.

[59] COOK 1 DOCKET 50-315 LER 89-010
SURVEILLANCE OF SEISMIC/EXPANSION GAP SEALS NOT PERFORMED DUE TO FAILURE TO RECOGNIZE THESE SEALS AS REQUIRING SURVEILLANCE PER TECH SPECS.
EVENT DATE: 070589 REPORT DATE: 080489 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: COOK 2 (PWR)

(NSIC 214989) ON JULY 5, 1989 DURING THE REVIEW OF INSTALLATION SPECIFICATION/SURVEILLANCE REQUIREMENTS, IT WAS DETERMINED THAT 414 SEISMIC/EXPANSION GAP SEALS, WHICH HAD NOT PREVIOUSLY BEEN CONSIDERED TO HAVE REQUIRED TECHNICAL SPECIFICATION SURVEILLANCE, SHOULD HAVE BEEN INCLUDED IN THE FORMAL FIRE SEAL INSPECTION PROGRAM AS REQUIRED BY TECHNICAL SPECIFICATION SURVEILLANCE 4.7.10.1.C. PRIOR TO JULY 5, 1989, THE FIRE BARRIER SILICONE TYPE FOAM PENETRATION SEALS INSTALLED UNDER SPECIFICATION DCC FP 101 QCN WERE CONSIDERED THE ONLY SEALS REQUIRING INSPECTION PER TECHNICAL SPECIFICATION SURVEILLANCE 4.7.10.1.C. AS A RESULT OF THIS DETERMINATION, THE SEISMIC/EXPANSION GAP SEALS INSTALLED UNDER SPECIFICATION DCC FP 102 QCF FOR SEISMIC DEFLECTION, EXPANSION AND NON-LOAD BEARING GAPS WERE NOT RECOGNIZED AS REQUIRING TECHNICAL SPECIFICATION SURVEILLANCE. BASED ON VISUAL AND ENGINEERING REVIEWS ALL APPLICABLE SEISMIC/EXPANSION GAP SEALS WERE CONFIRMED OPERABLE. THE PLANT WILL APPLY THE PROVISIONS OF TECHNICAL SPECIFICATIONS TO THOSE SEISMIC/EXPANSION GAP SEALS SPECIFIED IN SPECIFICATION DCC FP 102 QCF (I.E., FOUND INOPERABLE BETWEEN SURVEILLANCES, WILL REQUIRE COMPENSATORY FIREWATCHES, ETC.). ALSO, A FORMAL SURVEILLANCE PROGRAM PER THE REQUIREMENTS OF TECHNICAL SPECIFICATION 4.7.10.1.C WILL BE IN PLACE BY SEPTEMBER 15, 1989 FOR SEISMIC/EXPANSION GAP SEALS.

[60] COOPER DOCKET 50-298 LER 89-023
VALVE BODY WALL THINNING IN SAFETY RELATED THROTTLE VALVES DUE TO EROSION.
EVENT DATE: 052289 REPORT DATE: 081789 NSSS: GE TYPE: BWR
VENDOR: ANCHOR/DARLING INDUSTRIES

(NSIC 215066) DURING THE INSPECTIONS OF SAFETY RELATED THROTTLING VALVES USING ULTRASONIC TESTING (UT) TECHNIQUES, APPARENT WALL THICKNESSES SUBSTANTIALLY BELOW THE REQUIRED DESIGN MINIMUM WALL THICKNESS WERE OBSERVED ON THE RESIDUAL HEAT REMOVAL (RHR) LOOP "A" INJECTION THROTTLE VALVE. WHEN THE VALVE WAS DISASSEMBLED FOR REPAIR, NO SUBSTANTIAL WALL THICKNESS DEGRADATION COULD BE FOUND IN THE SUSPECT AREA. ALSO, THE INSPECTIONS IDENTIFIED 4 OTHER VALVES AS HAVING AS-FOUND WALL THICKNESSES BELOW THAT REQUIRED FOR THEIR ORIGINAL DESIGN PRESSURE CLASS RATING. THE AS-FOUND WALL THICKNESSES WERE SUBSEQUENTLY DETERMINED TO BE ACCEPTABLE WHEN NEW MINIMUM REQUIRED PRESSURE RATINGS WERE CALCULATED, BASED ON CONSERVATIVE SYSTEM DESIGN PRESSURES AND TEMPERATURES. THE CAUSE FOR THE ERRONEOUS UT INDICATION IN THE RHR VALVE WAS DUE TO A CASTING IRREGULARITY WITHIN CODE LIMITS WHICH OCCURRED DURING MANUFACTURE. THE CAUSE FOR THE ACTUAL WALL THINNING OBSERVED IN ALL VALVES WAS ATTRIBUTED TO EROSION DUE TO THROTTLING. TO ENSURE THAT THE MINIMUM ALLOWED WALL THICKNESSES REQUIRED BY SYSTEM DESIGN ARE MAINTAINED, WALL THICKNESS MEASUREMENTS WILL BE PERFORMED PERIODICALLY ON THE AFFECTED VALVES. VALVE REPAIR OR REPLACEMENT WILL BE PERFORMED IF FOUND TO BE NECESSARY. INVESTIGATION OF POSSIBLE VALVE MODIFICATIONS WHICH WOULD ELIMINATE OR MINIMIZE THE POTENTIAL FOR EROSION WHEN BEING THROTTLED, WILL BE PERFORMED.

[61] COOPER DOCKET 50-298 LER 89-022
IDENTIFICATION OF A CONDITION WHICH COULD HAVE RENDERED BOTH TRAINS OF STANDBY GAS TREATMENT INOPERABLE.
EVENT DATE: 061389 REPORT DATE: 071389 NSSS: GE TYPE: BWR

(NSIC 214815) ON JUNE 13, 1989, A CONDITION WAS IDENTIFIED WHICH COULD HAVE RENDERED BOTH TRAINS OF THE STANDBY GAS TREATMENT (SGT) SYSTEM INOPERABLE DUE TO A SINGLE COMPONENT FAILURE. THE DIFFERENTIAL PRESSURE CONTROL VALVES (DPCV) LOCATED ON THE DISCHARGE OF EACH SGT TRAIN ARE FAIL OPEN, AIR TO CLOSE, AND CONTROLLED VIA A SINGLE ELECTRIC TO PNEUMATIC (E/P) SIGNAL CONVERTER THROUGH THREE-WAY SOLENOID PILOT VALVES (SPV). AN ENGINEERING EVALUATION DETERMINED THAT A POTENTIAL FAILURE MODE OF THE E/P CONVERTER EXISTS, WHICH COULD RESULT IN A HIGH AIR PRESSURE OUTPUT SIGNAL. IF THIS SITUATION OCCURRED DURING SGT OPERATION, THE HIGH AIR PRESSURE OUTPUT WOULD DRIVE THE DPCV FOR THE OPERATING TRAIN CLOSED. WITH THE EXISTING INSTRUMENT AIR TUBING ARRANGEMENT TO THE SPVS, IT

WOULD NOT BE POSSIBLE TO VENT THE DPCV ACTUATOR TO REOPEN THE VALVE. ANY ATTEMPT TO OPERATE THE OTHER SGT TRAIN WOULD PROVE UNSUCCESSFUL BECAUSE BOTH DPCVS RECEIVE CONTROL SIGNALS FROM THE SAME E/P CONVERTER. THE SPV TUBING CONFIGURATION WAS APPARENTLY REVISED DURING PLANT CONSTRUCTION AS PART OF A MODIFICATION RESULTING FROM A FINAL SAFETY ANALYSIS REPORT QUESTION. THE ORIGINAL SGT DESIGN UTILIZED ONE DPCV ON THE COMMON DISCHARGE LINE. THE MODIFICATION REPLACED THE SINGLE DPCV WITH A CONTROL VALVE ON THE DISCHARGE OF EACH TRAIN, BUT MAINTAINED THE SINGLE DIFFERENTIAL PRESSURE CONTROL ARRANGEMENT.

[62] CRYSTAL RIVER 3 DOCKET 50-302 LER 89-016 REV 01
UPDATE ON ADMINISTRATIVE PROBLEMS CAUSE DEFICIENCIES IN THE ENVIRONMENTAL QUALIFICATION PROGRAM RESULTING IN PLANT EQUIPMENT NOT PROPERLY QUALIFIED.
EVENT DATE: 042689 REPORT DATE: 073189 NSSS: BW TYPE: PWR

(NSIC 214949) CRYSTAL RIVER UNIT 3 WAS IN MODE 5 (COLD SHUTDOWN) 4/26/89. ON THIS DATE, NRC INSPECTORS DISCOVERED DEFICIENCIES RELATED TO ENVIRONMENTAL QUALIFICATION OF CONTAINMENT ISOLATION VALVES LOCATED IN THE REACTOR BUILDING. DEFICIENCIES WERE THE RESULT OF DEFICIENCIES IN DETAILED DEVELOPMENT AND IMPLEMENTATION OF THE ENVIRONMENTAL QUALIFICATION PROGRAM. UTILITY PERSONNEL HAVE REPAIRED IDENTIFIED ENVIRONMENTAL QUALIFICATION DEFICIENCIES, OR HAVE JUSTIFIED CONTINUED OPERATION WITH THE DEFICIENCIES. ENVIRONMENTAL QUALIFICATION FILES CONTAIN NECESSARY JUSTIFICATION FOR CONTINUED OPERATION UNTIL REPAIRS ARE COMPLETED. THE UTILITY HAS EMBARKED ON A MAJOR VOLUNTARY EFFORT TO REVIEW PAST ACTIVITIES AND TO CORRECT ADDITIONAL ENVIRONMENTAL QUALIFICATION DEFICIENCIES THAT MAY BE DISCOVERED.

[63] CRYSTAL RIVER 3 DOCKET 50-302 LER 89-021 REV 01
UPDATE ON UNKNOWN PROBLEM IN STATIC TRANSFER SWITCH LEADS TO REACTOR PROTECTION SYSTEM ACTUATION.
EVENT DATE: 061489 REPORT DATE: 072789 NSSS: BW TYPE: PWR

(NSIC 214881) ON JUNE 14, 1989 CRYSTAL RIVER UNIT 3 WAS IN MODE 3 WITH THE REACTOR COOLANT AT 550 DEGREES F AND 2150 PSIG. AT 1201 HOURS, A REACTOR PROTECTION SYSTEM (RPS) ACTUATION WAS RECEIVED. AT THE TIME OF THE EVENT, PLANT PERSONNEL WERE PERFORMING A SURVEILLANCE TEST. DURING THE PERFORMANCE OF THE TEST, AN UNKNOWN PROBLEM IN VITAL BUS TRANSFER SWITCH (VBXS)-1A CAUSED IT TO TRANSFER TO A DE-ENERGIZED BUS. THIS INTERRUPTED POWER TO THE REACTOR COOLANT PUMP POWER MONITOR (RCPPM) SUPERVISORY CIRCUIT. THE LOSS OF THE PUMP RUNNING INDICATION INITIATED AN RPS ACTUATION SIGNAL ON MORE THAN ONE RCP SHUT DOWN. THE ROOT CAUSE OF THIS EVENT IS UNKNOWN. AFTER THE RPS ACTUATION WAS RECEIVED, A TESTING PROGRAM WAS DEVELOPED AND IMPLEMENTED. IT WAS NOT SUCCESSFUL IN PINPOINTING THE ORIGINAL MALFUNCTION, OR EVENT CAUSING THE SYSTEM TO REPEAT THE ANOMALOUS BEHAVIOR. THIS EVENT DID NOT EFFECT CORE COOLING OR REACTIVITY, THUS IT HAD NO IMPACT ON NUCLEAR SAFETY. A MONITORING PROGRAM WILL BE INSTITUTED TO LOOK AT THIS SWITCH AFTER ANY RPS ACTUATION FOR THE DURATION OF THIS FUEL CYCLE. THIS REPORT IS BEING SUBMITTED UNDER THE REQUIREMENTS OF 10 CFR 50.73.A.IV.

[64] CRYSTAL RIVER 3 DOCKET 50-302 LER 89-025
FAILED CURRENT TRANSFORMER CAUSED SEPARATION FROM NORMAL OFFSITE POWER SOURCE.
EVENT DATE: 062989 REPORT DATE: 072789 NSSS: BW TYPE: PWR

(NSIC 214918) ON 6/29/89, CRYSTAL RIVER UNIT 3 WAS IN HOT SHUTDOWN (MODE 3), COOLING DOWN TO PERFORM MAINTENANCE ON ONE OF THE TWO EMERGENCY DIESEL GENERATORS (EDG). AN ELECTRICAL STORM WAS IN PROGRESS IN THE AREA. AT 2015, THE PLANT SEPARATED FROM ITS NORMAL OFFSITE POWER SOURCE. THE OPERABLE EDG AUTOMATICALLY STARTED AND REENERGIZED THE "A" 4160V ENGINEERED SAFEGUARDS (ES) BUS. TWO MINUTES AFTER SEPARATION FROM THE NORMAL OFFSITE POWER SOURCE, OPERATORS ALIGNED POWER FROM A SECOND OFFSITE SOURCE TO THE "B" 4160V ES BUS, AND ENERGIZED THE

BUS. OPERATORS RESTORED THE NORMAL OFFSITE POWER SOURCE AT 2137. THE REACTOR WAS COOLED BY NATURAL CIRCULATION FLOW FOR APPROX. TWO HOURS DURING THE EVENT. SEPARATION FROM THE NORMAL OFFSITE POWER SOURCE OCCURRED DUE TO AN ELECTRIC FAULT IN THE 230KV SWITCHYARD. THERE WAS INSUFFICIENT DATA AVAILABLE TO DETERMINE THE EXACT CAUSE OF THE ELECTRIC FAULT. HOWEVER, IT IS BELIEVED THAT THE EVENT WAS CAUSED BY LIGHTNING.

[65] CRYSTAL RIVER 3 DOCKET 50-302 LER 89-026
INABILITY TO REPAIR EMERGENCY DIESEL GENERATOR WITHIN TIME ALLOWED BY ACTION STATEMENT RESULTS IN SHUTDOWN REQUIRED BY TECH SPECS.
EVENT DATE: 062989 REPORT DATE: 073189 NSSS: BW TYPE: PWR
VENDOR: FAIRBANKS MORSE

(NSIC 214950) ON 6/29/89 CRYSTAL RIVER UNIT THREE WAS IN OPERATIONAL MODE ONE (POWER OPERATION) AT 97% POWER. EMERGENCY DIESEL GENERATOR 1B HAD BEEN OUT OF SERVICE FOR MAINTENANCE SINCE 6/26 AT 0650. TECH SPEC 3.8.1.1 ACTION B WAS INITIATED AT THAT TIME. AT 0300 PLANT SHUTDOWN WAS INITIATED BECAUSE EGDG-1B COULD NOT BE REPAIRED IN THE TIME REMAINING IN THE ACTION STATEMENT. AN UNUSUAL EVENT WAS DECLARED BECAUSE THE PLANT WAS CONDUCTING A SHUTDOWN REQUIRED BY TECH SPECS. OPERATIONAL MODE THREE (HOT STANDBY) WAS ENTERED AT 1142 ON 6/29/89. COOLDOWN WAS INITIATED AND OPERATIONAL MODE FIVE (COLD SHUTDOWN) WAS ENTERED AT 1722. THE UNUSUAL EVENT WAS TERMINATED AT 1730. THE DIESEL GENERATOR WAS TAKEN OUT OF SERVICE DUE TO LOW CRANKCASE VACUUM. THE CRANKCASE VAPOR EJECTOR AND ITS ASSOCIATED TUBING AND OIL SEPARATOR HAVE BEEN CLEANED AND TESTED SATISFACTORILY. THE SUPERCHARGER BLOWER AND THE TURBOCHARGER WERE REPLACED. THE CAUSE OF THE INADEQUATE VACUUM HAS NOT BEEN CONCLUSIVELY DETERMINED. INVESTIGATION INTO THE CAUSE OF THE LOW VACUUM IS CONTINUING. CORRECTIVE ACTIONS WILL BE IMPLEMENTED BASED ON THE RESULTS OF THE INVESTIGATION.

[66] CRYSTAL RIVER 3 DOCKET 50-302 LER 89-027
PERSONNEL ERROR IN FAILURE TO IMPLEMENT SURVEILLANCE REQUIREMENTS OF TECH SPEC AMENDMENT RESULTS IN FAILURE TO PERFORM SURVEILLANCE IN REQUIRED INTERVAL.
EVENT DATE: 071389 REPORT DATE: 081189 NSSS: BW TYPE: PWR

(NSIC 215046) ON 7/13/89 AT 2018, CRYSTAL RIVER UNIT 3 WAS IN OPERATIONAL MODE ONE (POWER OPERATION) WHEN IT WAS DISCOVERED THAT THE CONTAINMENT HYDROGEN MONITOR MONTHLY FUNCTIONAL TESTS WERE OVERDUE ON BOTH TRAINS. TESTING OF THESE MONITORS WAS REQUIRED WHEN TECH SPEC AMENDMENT 113 BECAME EFFECTIVE 6/8/89. PROCEDURES TO PERFORM THIS SURVEILLANCE REQUIREMENT HAD NOT BEEN IMPLEMENTED. THIS EVENT WAS CAUSED BY COGNITIVE PERSONNEL ERROR ON THE PART OF DEPARTMENTS RESPONSIBLE FOR WRITING, ISSUING, SCHEDULING, AND PERFORMING NEW SURVEILLANCE PROCEDURES REQUIRED PER TECH SPEC AMENDMENTS. AN INITIAL REVIEW OF PROCEDURES WHICH CONTROL IMPLEMENTATION OF TECH SPEC AMENDMENTS INDICATES THE INTENT OF THE INSTRUCTIONS ARE ADEQUATE; HOWEVER, CLARIFICATION AS TO SPECIFIC DEPARTMENTAL RESPONSIBILITIES AND AUTHORITY IS NEEDED. FURTHER CLARIFICATION WILL BE PROVIDED IN THE FORM OF A LETTER TO APPLICABLE DEPARTMENT MANAGERS CLARIFYING DEPARTMENTAL RESPONSIBILITIES AND INDIVIDUAL AUTHORITY FOR COORDINATING AND IMPLEMENTING TECH SPEC AMENDMENTS. IN ADDITION, CHANGES TO THE PROCEDURES WILL BE CONSIDERED AND IMPLEMENTED AS NECESSARY.

[67] CRYSTAL RIVER 3 DOCKET 50-302 LER 89-028
PERSONNEL ERRORS DURING DEVELOPMENT OF REVISION TO SURVEILLANCE PROCEDURE RESULT IN CONTAINMENT INTEGRITY VALVES NOT BEING PROPERLY SURVEILLED.
EVENT DATE: 072589 REPORT DATE: 082489 NSSS: BW TYPE: PWR

(NSIC 215127) ON 7/25/89, CRYSTAL RIVER UNIT 3 WAS OPERATING IN MODE ONE (POWER OPERATION) AT 100% OF RATED THERMAL POWER. ON THIS DATE, IT WAS IDENTIFIED THAT SIX CONTAINMENT MANUAL ISOLATION VALVES WERE NOT BEING SURVEILLANCED IN

ACCORDANCE WITH TECH SPECS BECAUSE OF DEFICIENCIES IN THE SURVEILLANCE PROCEDURE. THESE DEFICIENCIES RESULTED FROM OVERSIGHTS BY PERSONNEL DEVELOPING AND REVIEWING A RECENTLY APPROVED REVISION TO THE SURVEILLANCE PROCEDURE. THE AFFECTED VALVES WERE VERIFIED TO BE CLOSED AND THESE PROCEDURE DEFICIENCIES HAVE BEEN CORRECTED. APPROPRIATE ENGINEERING PERSONNEL WILL BE COUNSELED AS TO THE NECESSITY OF PERFORMING ADEQUATE REVIEW OF MODIFICATION PACKAGES AS THEY IMPACT SURVEILLANCE PROCEDURES. ADDITIONALLY, EMPHASIS WILL BE PLACED ON PERFORMING IMMEDIATE VERIFICATION FOLLOWING SIGNIFICANT PROCEDURE REVISIONS.

[68] DAVIS-BESSE 1 DOCKET 50-346 LER 89-010
CONTROL ROOM EMERGENCY VENTILATION SYSTEM INOPERABLE DUE TO COMPRESSOR HIGH PRESSURE TRIPS.
EVENT DATE: 061289 REPORT DATE: 072689 NSSS: BW TYPE: PWR

(NSIC 214925) ON 6/26/89, DURING THE PERFORMANCE OF THE CONTROL ROOM EMERGENCY VENTILATION SYSTEM (CREVS) MONTHLY TEST, DB-SS-03041, THE COOLING SYSTEM COMPRESSOR WOULD NOT START. THE CREVS TRAIN 1-1 WAS DECLARED INOPERABLE, AND THE STATION ENTERED THE 7-DAY ACTION STATEMENT OF TECH SPEC 3.7.6.1. THE CAUSE WAS A REFRIGERANT PRESSURE SWITCH LOCATED INSIDE PANEL C6706 THAT HAD TRIPPED ON HIGH PRESSURE. THE SWITCH WAS RESET AND THE SYSTEM RETESTED SUCCESSFULLY AND DECLARED OPERABLE AT 1730 HRS ON 6/27/89. FURTHER EVALUATION CONCLUDED THAT THE PRESSURE SWITCH HAD MOST PROBABLY TRIPPED ON 6/12/89, WHEN THE SYSTEM HAD BEEN OPERATED. THE SYSTEM HAD MOST PROBABLY BEEN IN A DEGRADED CONDITION FOR A PERIOD OF TIME LONGER THAN ALLOWED BY THE ACTION STATEMENT. THIS IS REPORTABLE UNDER 10CFR50.73(A)(2)(I)(B) AS A CONDITION PROHIBITED BY THE TECH SPEC. IT IS ALSO RECOGNIZED THAT SINCE CREVS TRAIN 1-2 HAD BEEN DECLARED INOPERABLE ON 6/8/89, TO SUPPORT MAINTENANCE ON THE BACKUP AIR COOLED CONDENSER DAMPER AND NOT DECLARED OPERABLE UNTIL 1455 HOURS ON 6/12/89, THAT BOTH CREVS TRAINS WERE MOST PROBABLY INOPERABLE AT THE SAME TIME. THIS IS ALSO A CONDITION PROHIBITED BY THE TECH SPECS. CREVS OPERATING PROCEDURE, SP1104.69, WILL BE CHANGED TO ADD THE LOCAL VERIFICATION OF THE COMPRESSOR OPERATION IN THE WATER COOLED MODE WHEN STARTING AND SHUTTING DOWN THE SYSTEM.

[69] DAVIS-BESSE 1 DOCKET 50-346 LER 89-007
ONE OF FOUR STRONG MOTION TRIAXIAL ACCELEROMETERS INOPERABLE FOR MORE THAN 30 DAYS.
EVENT DATE: 061489 REPORT DATE: 072489 NSSS: BW TYPE: PWR
VENDOR: TERRA TECHNOLOGY CORP.

(NSIC 214389) ON JUNE 14, 1989, DURING THE PERFORMANCE OF THE SEISMIC MONITORING SYSTEM CHANNEL CHECK, ST 5034.01, THE SYSTEM FAILED WHEN A +12VDC POWER SUPPLY FUSE BLEW. THE SOURCE OF THE PROBLEM WAS DETERMINED TO BE ZT2951, A STRONG MOTION TRIAXIAL ACCELEROMETER MOUNTED INSIDE CONTAINMENT AT THE 653 FT. ELEVATION. DUE TO ALARA CONCERNS, THIS ACCELEROMETER WILL BE REPAIRED DURING THE NEXT OUTAGE OF SUFFICIENT DURATION OR AT LEAST NO LATER THAN THE END OF THE SIXTH REFUELING OUTAGE. THE CAUSE WILL BE FURTHER EVALUATED AT THAT TIME. ZT2951 HAS BEEN ISOLATED, AND THE REMAINDER OF THE SEISMIC MONITORING SYSTEM HAS BEEN RE-ENERGIZED AND IS FUNCTIONAL WITH THREE OF THE FOUR TRIAXIAL ACCELEROMETERS. THREE OF THREE PEAK ACCELEROMETERS ARE OPERABLE. THIS REPORT COMPLETES THE TECHNICAL SPECIFICATION 3.3.3.3 ACTION STATEMENT 'A', WHICH REQUIRES THAT IF A SEISMIC INSTRUMENT HAS BEEN INOPERABLE FOR MORE THAN 30 DAYS, SUBMIT A SPECIAL REPORT PURSUANT TO SPECIFICATION 6.9.2 WITHIN THE NEXT 10 DAYS.

[70] DAVIS-BESSE 1 DOCKET 50-346 LER 89-008
MISSED FIRE SYSTEM MONTHLY VALVE INSPECTION.
EVENT DATE: 062289 REPORT DATE: 072489 NSSS: BW TYPE: PWR

(NSIC 214924) ON 7/8/89, AT 0825 HOURS, THE SHIFT SUPERVISOR NOTED THAT ST

5016.09, THE FIRE SYSTEM VALVE MONTHLY INSPECTION, APPEARED TO BE BEYOND ITS SURVEILLANCE REQUIREMENT 4.7.9.1.1.C LATE DATE OF 6/22/89. AS A PRECAUTION, THE TEST WAS PROMPTLY RUN, AND BY 1252 HOURS THE TEST HAD VERIFIED AN OPERABLE FLOWPATH. ON 7/11/89, IT WAS CONFIRMED THAT THE SURVEILLANCE FOR JUNE HAD BEEN MISSED AND DETERMINED THAT TECH SPEC 3.7.9.1 ACTION STATEMENT B SHOULD STILL BE COMPLETED EVEN THOUGH THE CONDITION THAT CAUSED THE ENTRY INTO THE ACTION STATEMENT WAS RESOLVED IN LESS THAN 5 HOURS. THE 24-HOUR REPORT SHOULD HAVE BEEN MADE BY 7/9/89, AT 0825 HOURS. THE CAUSE OF THE MISSED SURVEILLANCE TEST WAS DUE TO ADMINISTRATIVE AND PERSONNEL ERROR. A REVISED TEST, DP-FP-03003, IS NOW AVAILABLE WHICH PROVIDES MORE FLEXIBILITY IN DETERMINING AN OPERABLE FLOWPATH. A MEETING WAS HELD ON 7/17/89, WITH ALL APPROPRIATE PERSONNEL EMPHASIZING THE INTERDEPARTMENTAL RESPONSIBILITIES FOR ENSURING TECH SPEC REQUIREMENTS ARE MET. THE CAUSES AND LESSONS LEARNED FROM THE SPECIFIC OCCURRENCE WERE ADDRESSED. THIS REPORT SATISFIES BOTH THE SPECIAL REPORT REQUIRED BY TECH SPEC 3.7.9.1 ACTION STATEMENT B.2.C AND THE LER REQUIRED BY 10CFR50.73(A)(2)(I)(B).

[71] DAVIS-BESSE 1 DOCKET 50-346 LER 89-009
IMPROPER OPERATION OF THE HIGH PRESSURE INJECTION SYSTEM LINE ISOLATION VALVES.
EVENT DATE: 062989 REPORT DATE: 073189 NSSS: BW TYPE: PWR

(NSIC 214890) ON JUNE 29, 1989, THE PLANT WAS OPERATING IN MODE 1 AT 100 PERCENT REACTOR POWER. DURING REVIEW OF A QUESTION CONCERNING OPERATION OF THE DIESEL GENERATOR LOAD SEQUENCER IT WAS DETERMINED THAT THE HIGH PRESSURE INJECTION (HPI) SYSTEM LINE ISOLATION VALVES IN BOTH HPI TRAINS WOULD NOT FULLY OPEN WITHIN 30 SECONDS WHEN RESPONDING TO A SAFETY FEATURES ACTUATION SYSTEM SIGNAL IN CONJUNCTION WITH A LOSS OF OFFSITE POWER AS REQUIRED BY TECHNICAL SPECIFICATION TABLE 3.3-5. THE SHIFT SUPERVISOR WAS INFORMED AT 1630 HOURS AND THE PLANT ENTERED TECHNICAL SPECIFICATION 3.0.3 DUE TO TWO INOPERABLE HPI SYSTEMS. AT 1700 HOURS TECHNICAL SPECIFICATION 3.0.3 WAS EXITED WHEN HPI TRAIN 1 WAS RESTORED TO OPERABILITY BY OPENING THE ISOLATION VALVES, HP2C AND HP2D, AND THE PLANT ENTERED TECHNICAL SPECIFICATION 3.5.2 FOR ONE INOPERABLE HPI SYSTEM. TEMPORARY MODIFICATIONS 89-0022 THROUGH 89-0025 TO MODIFY THE VALVE CIRCUITRY TO ENSURE COMPLIANCE WITH THE RESPONSE TIME REQUIREMENT WERE IMPLEMENTED ON BOTH TRAINS OF THE HIGH PRESSURE INJECTION SYSTEM. THESE MODIFICATIONS WERE COMPLETED AT 0722 HOURS ON JULY 1, 1989, AND TECHNICAL SPECIFICATION 3.5.2 WAS EXITED. THIS CONDITION WAS CAUSED BY INADEQUATE DESIGN. THIS CONDITION IS BEING REPORTED IN ACCORDANCE WITH 10CFR50.73(A)(2)(I)(B).

[72] DAVIS-BESSE 1 DOCKET 50-346 LER 89-011
TESTING OF DECAY HEAT COOLER VALVES DID NOT SATISFY ASME REQUIREMENTS.
EVENT DATE: 071389 REPORT DATE: 081489 NSSS: BW TYPE: PWR

(NSIC 215072) ON JULY 13, 1989, DURING THE EVALUATION OF SAFETY SYSTEM FUNCTIONAL INSPECTION (SSFI) FINDINGS ON THE PLANT AIR SYSTEM, IT WAS DETERMINED THAT THE ASME SECTION XI TEST CONDITIONS WERE NOT BEING SATISFIED DURING TESTING OF THE DECAY HEAT COOLER VALVES, DH13A, DH13B, DH14A AND DH14B. SUBARTICLE IWV-3415 REQUIRES TESTING OF THE VALVES FOR FAIL-SAFE OPERATION UPON LOSS OF ACTUATOR POWER. HOWEVER, THE TESTING OF THESE VALVES SINCE THEIR INSTALLATION DURING THE FIFTH REFUELING OUTAGE (1988) ISOLATED AIR TO THE VALVE POSITIONER BUT NOT THE ACTUATOR. THEREFORE, TESTING DID NOT FULLY VERIFY FAIL-SAFE OPERATION OF THE VALVES WITH ONLY THE ACTUATOR SPRING AS THE MOTIVE FORCE. RESPONSIBLE PERSONNEL FAILED TO RECOGNIZE THAT THE NEW DESIGN WOULD NECESSITATE A CHANGE IN THE TEST PROCEDURES IN ORDER TO MEET ASME TEST REQUIREMENTS. THE PLANT MODIFICATIONS PROCEDURE, EN-DP-01200, WILL BE REVISED TO ADD SPECIFIC QUESTIONS TO BE ADDRESSED THAT WILL DETERMINE WHETHER THE ASME SECTION XI TEST REQUIREMENT SHEET NEEDS TO BE INCLUDED IN THE MODIFICATION PACKAGE. THE VALVES WERE PROPERLY RETESTED ON JULY 28, 1989, WITH MODIFIED PROCEDURES, AND THE CAPABILITY OF THE VALVES TO REPOSITION TO THEIR FAIL-SAFE POSITION UPON LOSS OF ACTUATOR POWER (AIR) WAS VERIFIED.

[73] DIABLO CANYON 1 DOCKET 50-275 LER 89-006
CONTROL ROOM VENTILATION SYSTEM SHIFT DUE TO INADEQUATE PROCEDURAL GUIDANCE.
EVENT DATE: 072189 REPORT DATE: 082189 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: DIABLO CANYON 2 (PWR)

(NSIC 215061) ON JULY 21, 1989, AT 0227 PDT, WITH UNIT 1 IN MODE 1 (POWER OPERATION), AND UNIT 2 IN MODE 2 STARTUP , THE CONTROL ROOM VENTILATION SYSTEM AUTOMATICALLY SHIFTED FROM ITS NORMAL VENTILATION MODE TO THE PRESSURIZATION MODE, AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION. THIS SHIFT IN MODES WAS DUE TO THE RECEIPT OF A HIGH RADIATION SIGNAL FROM UNIT 1 RADIATION DETECTORS RE-25 AND 26. RE-25 AND 26 WERE EXPOSED TO RADIATION DURING RADIOGRAPHIC EXAMINATION OF THE UNIT 1 10 PERCENT STEAM DUMP VALVES, PCVS 21 AND 22. VENTILATION SYSTEM WAS RESTORED TO ITS NORMAL MODE AFTER ENSURING THAT THE ESF A CTUATION WAS NOT THE RESULT OF AN ACTUAL EMERGENCY CONDITION. IN ACCORDANCE WITH 10 CFR 50.72 B (2) II), A 4-HOUR NON-EMERGENCY REPORT WAS COMPLETED AT 0345 PDT, JULY 21, 19 89. A PARTIAL CALIBRATION AND A COMPLETE FUNCTIONAL TEST PERFORMED ON RE-25 AND 26 ON JULY 21, AND 22, 1989, RESPECTIVELY, DEMONSTRATED THAT BOTH RADIATION MONITORS WERE FUNCT IONING PROPERLY. THIS EVENT WAS CAUSED BY INADEQUATE PROCEDURAL GUIDANCE, IN THAT THE PROCEDURES BEING USED TO PERFORM THE RADIOGRAPHY DID NOT CONSIDER THE PRESENCE OF RADIATION DETECTORS IN THE PROXIMITY OF THE STEAM DUMP VALVES BEING EXAMINED AND DID NOT ADEQUATELY DEFINE DEPARTMENTAL RESPONSIBILITIES FOR THE PERFORMANCE OF THE RADIOGRAPHIC EXAMINATIONS.

[74] DIABLO CANYON 2 DOCKET 50-323 LER 87-020 REV 01
UPDATE ON ENTRY INTO TECH SPEC 3.0.3 DUE TO BOTH TRAINS OF AUXILIARY BUILDING VENTILATION BEING INOPERABLE.
EVENT DATE: 081887 REPORT DATE: 072889 NSSS: WE TYPE: PWR
VENDOR: AMERICAN WARMING & VENTILATING INC.

(NSIC 214859) 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. 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. SUBSEQUENT INVESTIGATION DETERMINED THAT THE CAUSE OF THESE EVENTS WAS THE FAILURE OF THE S-34 FAN BACKDRAFT DAMPER TO CLOSE, WHICH WAS DUE TO THE LACK OF A PREVENTIVE MAINTENANCE PROGRAM FOR THE DAMPER. A CONTRIBUTARY CAUSE WAS AN INADEQUATE FLOW SWITCH DESIGN. AS A RESULT OF THIS EVENT, A PREVENTIVE MAINTENANCE PROGRAM FOR THE DAMPERS WAS DEVELOPED IN 1987. MECHANICAL MAINTENANCE WILL ENSURE THAT FUTURE PREVENTIVE MAINTENANCE IS ADEQUATE, AND WILL RECOMMEND IMPROVEMENTS IN THE PROGRAM IF NECESSARY. A DESIGN CHANGE HAS BEEN INITIATED TO EVALUATE AND RELOCATE THE FLOW SWITCH SENSING POINTS.

[75] DIABLO CANYON 2 DOCKET 50-323 LER 88-025 REV 01
UPDATE ON SEISMIC BRACING MISSING FROM INSTRUMENT PANEL DUE TO INADEQUATE CONFIGURATION CONTROL.
EVENT DATE: 120188 REPORT DATE: 072889 NSSS: WE TYPE: PWR

(NSIC 214860) THIS VOLUNTARY LER IS BEING SUBMITTED FOR INFORMATIONAL PURPOSES ONLY, AS DESCRIBED IN ITEM 19 OF SUPPLEMENT 1 TO NUREG 1022. ON FEBRUARY 3, 1989, AT 0900 PST I&C TECHNICIANS DISCOVERED THAT SEISMIC BRACING ON PANEL RRM WAS MISSING. CONTAINMENT WIDE RANGE LEVEL CHANNELS 942A AND 943A WERE DECLARED INOPERABLE AT 1450 PST AND ACTION STATEMENT B OF TECHNICAL SPECIFICATION 3.3.3.6 WAS ENTERED. THE BRACING WAS REINSTALLED AND ON FEBRUARY 4, AT 0026 THE WIDE RANGE CHANNELS WERE DECLARED OPERABLE AND THE ACTION STATEMENT WAS EXITED. AS

THE TIME AT WHICH THE BRACING WAS REMOVED COULD NOT BE DETERMINED, IT WAS CONSERVATIVELY ASSUMED THAT THE CHANNELS WERE INOPERABLE FROM THE TIME THE UNIT ENTERED MODE 3 ON NOVEMBER 29, 1988. THE ROOT CAUSE WAS DETERMINED TO BE INADEQUATE CONFIGURATION CONTROL. ACTIONS TO PREVENT RECURRENCE INCLUDE ISSUANCE OF A MAINTENANCE BULLETIN ADDRESSING CONFIGURATION CONTROL DURING MAINTENANCE ACTIVITIES, REVISIONS TO APPLICABLE PROCEDURES TO INCLUDE CONFIGURATION CONTROL AND INCORPORATE CONFIGURATION CONTROL POLICIES INTO NUCLEAR POWER GENERATION, NUCLEAR ENGINEERING AND CONSTRUCTION SERVICES (NECS), AND GENERAL CONSTRUCTION TRAINING SYLLABUS. A PG&E CALCULATION HAS SUBSEQUENTLY SHOWN THE CHANNELS TO BE OPERABLE IN THE EVENT OF A DESIGN BASIS SEISMIC EVENT WITH THE BRACING MISSING. CONSEQUENTLY, THIS EVENT IS NO LONGER CONSIDERED REPORTABLE.

[76] DIABLO CANYON 2 DOCKET 50-323 LER 89-007
MANUAL REACTOR TRIP DUE TO SEAWATER INLEAKAGE FROM A CONDENSER TUBE SHEET PLUG FAILURE.
EVENT DATE: 071689 REPORT DATE: 081589 NSSS: WE TYPE: PWR

(NSIC 215069) ON 7/16/89, AT 0258 PDT, DIABLO CANYON POWER PLANT (DCPP) UNIT 2 WAS MANUALLY TRIPPED IN ACCORDANCE WITH DCPP ABNORMAL OPERATING PROCEDURE AP-20, "CONDENSER TUBE LEAK." A 4 HOUR NON-EMERGENCY REPORT WAS MADE TO THE NRC AT 0358 PDT ON 7/16/89, IN ACCORDANCE WITH 10 CFR 50.72(B)(2)(II). DCPP UNIT 2 WAS IN A PLANNED MAINTENANCE CURTAILMENT AT ABOUT 50 PERCENT POWER FOR CLEANING MAIN CONDENSER TUBE SHEETS. ON 7/16/89 A SHORT TIME AFTER RESTARTING CIRCULATING WATER PUMP 2-1, A CONDENSATE PUMP DISCHARGE HIGH CATION CONDUCTIVITY ALARM ANNUNCIATED IN THE CONTROL ROOM. THE SHIFT FOREMAN DIRECTED THE CONTROL ROOM OPERATORS TO ENTER PROCEDURE AP-20, WHICH REQUIRES POWER TO BE REDUCED. AT 27 PERCENT POWER, FEEDWATER CONDUCTIVITY INCREASED AND THE REACTOR WAS MANUALLY TRIPPED PER PROCEDURE. THE CAUSE OF THIS EVENT WAS FAILURE OF A CONDENSER TUBE SHEET PLUG. THE PLUG WAS NOT RECOVERABLE FOR EXAMINATION, AND THEREFORE THE REASON FOR FAILURE COULD NOT BE DETERMINED. THE MOST LIKELY ROOT CAUSE FOR THE FAILURE WAS EITHER IMPROPER PLUG INSTALLATION OR USE OF A MATERIAL SUSCEPTIBLE TO CORROSION. OTHER TUBE SHEET PLUGS WERE INSPECTED AND FOUND TO BE INSTALLED CORRECTLY AND OF THE PROPER MATERIAL. A NEW PLUG WAS INSTALLED IN THE TUBE SHEET.

[77] DRESDEN 2 DOCKET 50-237 LER 89-016
HPCI PIPING FOUND IN VIOLATION OF FSAR DESIGN CRITERIA DUE TO MANAGEMENT DEFICIENCY.
EVENT DATE: 051889 REPORT DATE: 061989 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: DRESDEN 3 (BWR)

(NSIC 214388) AT 1530 HOURS ON 5/18/89, WITH UNIT 2 OPERATING AT 100% RATED CORE THERMAL POWER AND UNIT 3 IN THE SHUTDOWN MODE FOR A SCHEDULED MAINTENANCE OUTAGE, STATION MANAGEMENT WAS NOTIFIED BY THE COMMONWEALTH EDISON BOILING WATER REACTOR ENGINEERING DEPARTMENT (BWRED) THAT THE UNIT 2 AND UNIT 3 HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM TURBINE STEAM SUPPLY VALVE 2(3)-2301-3 DRAIN POT PIPING WAS IN VIOLATION OF THE FINAL SAFETY ANALYSIS REPORT (FSAR) SEISMIC DESIGN CRITERIA. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO AN ENGINEERING AND TECHNICAL SUPPORT MANAGEMENT DEFICIENCY. THESE DISCREPANCIES WERE IDENTIFIED DURING A HPCI PIPING SEISMIC ANALYSIS PERFORMED BY A CONSULTING FIRM UNDER BWRED DIRECTION AS PART OF AN ONGOING QUALITY ASSURANCE DEPARTMENT HPCI SAFETY SYSTEM FUNCTIONAL INSPECTION (SSFI) PROGRAM. THE SAFETY SIGNIFICANCE OF THIS EVENT WAS MINIMAL AS ENGINEERING ANALYSIS INDICATES THAT THE PIPING INVOLVED WOULD REMAIN OPERABLE UNDER ALL DESIGN BASIS ACCIDENT (DBA) CONDITIONS. COMPREHENSIVE IMPROVEMENTS TO THE MODIFICATION PROCESS HAVE BEEN RECENTLY IMPLEMENTED IN ORDER TO PRECLUDE THIS TYPE OF EVENT. A PREVIOUS SIMILAR EVENT INVOLVING FSAR COMPLIANCE WAS REPORTED BY LER 88-1/050237.

[78] DRESDEN 2 DOCKET 50-237 LER 89-018
AUTO START OF STANDBY GAS TREATMENT SYSTEM DUE TO SPURIOUS VENTILATION RADIATION MONITOR TRIP.
EVENT DATE: 070789 REPORT DATE: 080289 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 214943) ON 4/7/89 DURING NORMAL UNIT 2 POWER OPERATION AT 72% RATED CORE THERMAL POWER, DRESDEN INSTRUMENT SURVEILLANCE (DIS) 1700-7, REACTOR BUILDING VENTILATION RADIATION MONITOR FUNCTIONAL TEST, WAS BEING PERFORMED. AT 1410 HOURS, THE UNIT 2 REACTOR BUILDING VENTILATION (RBV) SYSTEM TRIPPED AND THE STANDBY GAS TREATMENT (SBGT) SYSTEM AUTO INITIATED. AT THE TIME OF THE EVENT, AN INSTRUMENT MECHANIC WAS REMOVING THE CHANNEL B RBV RADIATION MONITOR INDICATOR AND TRIP UNIT FROM ITS HOLDER FOR A VOLTAGE CHECK. AFTER VERIFYING THAT HIGH RADIATION WAS NOT PRESENT IN THE RBV SYSTEM, SBGT WAS SECURED AND RBV WAS RETURNED TO NORMAL. THE CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO A NICK IN THE INSULATION ON ONE OF THE RBV RADIATION MONITOR TRIP UNIT WIRES. WHILE THE UNIT WAS BEING REMOVED FROM ITS HOLDER, THE EXPOSED WIRE CAME IN CONTACT WITH THE CHASSIS AND CAUSED THE TRIP UNIT TO SPIKE HIGH. THE EXPOSED WIRE WAS WRAPPED WITH ELECTRICAL TAPE AND THE PROCEDURE WAS SATISFACTORILY COMPLETED WITHOUT ANY FURTHER PROBLEMS. AS FURTHER CORRECTIVE ACTION, WORK REQUEST 85823 WAS WRITTEN TO REPAIR THE WIRE, AND THE INSTRUMENT MAINTENANCE STAFF WILL REVISE DIS 1700-7 TO INCLUDE CHECKS OF THE WIRING FOR INSULATION DAMAGE. A PREVIOUS EVENT INVOLVING AUTO INITIATION OF SBGT DUE TO A FAULTY REFUEL FLOOR RADIATION MONITOR TEST SWITCH WAS REPORTED BY LER 88-19/050237.

[79] DRESDEN 2 DOCKET 50-237 LER 89-019
SCRAM/GROUP I ISOLATION DUE TO MAIN STEAM RADIATION MONITOR LOCKUP AND SPURIOUS STEAM TUNNEL TEMPERATURE TRIP.
EVENT DATE: 071289 REPORT DATE: 080989 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.
UNITED ELECTRIC CONTROLS COMPANY

(NSIC 214977) ON 7/12/89 AT 1049 HOURS WITH UNIT 2 OPERATING AT 63% POWER WHILE PERFORMING DRESDEN TECHNICAL STAFF SURVEILLANCE (DTS) 500-2, FUNCTIONAL TESTING OF REACTOR PROTECTION SYSTEM (RPS) MOTOR GENERATOR (MG) SET AND RPS RESERVE POWER SUPPLY, CHANNEL 'A' HALF PRIMARY CONTAINMENT GROUP I ISOLATION AND HALF SCRAM SIGNALS COULD NOT BE RESET PROMPTLY DUE TO DIFFICULTIES ENCOUNTERED IN RESETTING THE 'A' MAIN STEAM LINE (MSL) LOGRITHMIC PRIMARY CONTAINMENT GROUP I ISOLATION AND REACTOR SCRAM. THE CAUSE OF THE MSL AREA HIGH TEMPERATURE TRIP WAS ATTRIBUTED TO SETPOINT DRIFT. THE ROOT CAUSE OF THE MSL LRM RESET DIFFICULTIES (WHICH OCCURRED DURING TRANSFER OF POWER SUPPLIES IN ACCORDANCE WITH DTS 500-2) IS UNDER INVESTIGATION. AS CORRECTIVE ACTION, THE "A" MSL LRM WAS REPLACED, IN ADDITION TO ALL OF THE CHANNEL "B" MSL TUNNEL AREA TEMPERATURE SWITCHES. THE SAFETY SIGNIFICANCE OF THIS EVENT WAS MINIMAL SINCE THE PRIMARY CONTAINMENT ISOLATION AND REACTOR SCRAM FUNCTIONS OCCURRED AS DESIGNED. OPERATIONS DEPARTMENT PERSONNEL RESPONDED IMMEDIATELY BY VERIFYING THAT A VALID GROUP I CONDITION HAD NOT OCCURRED AND PROCEEDED WITH A CONTROLLED SCRAM RECOVERY. A PREVIOUS EVENT INVOLVING A SCRAM DURING THE PERFORMANCE OF DTS 500-2 WAS REPORTED BY LER 89-02/050249.

[80] DRESDEN 2 DOCKET 50-237 LER 89-020
POTENTIAL VIOLATION OF SECONDARY CONTAINMENT INTEGRITY DUE TO INTERLOCK DOOR STRIKE FAILURE.
EVENT DATE: 071989 REPORT DATE: 081189 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: DRESDEN 3 (BWR)

(NSIC 215042) AT 0830 HOURS ON 7/19/89, THE OPERATIONS SHIFT SUPERVISOR WAS NOTIFIED THAT A SECONDARY CONTAINMENT INTERLOCK DOOR TO THE UNIT 2/3 DIESEL GENERATOR (D/G) ROOM WAS NOT OPERATING PROPERLY. CONTROL ROOM PERSONNEL ALSO

RECEIVED CONTROL ROOM PANEL 902-4 ANNUNCIATOR E-[illegible], "REACTOR BUILDING TO UNIT 2/3 DG ROOM INTERLOCK DOORS INOPERATIVE OR BYPASSED." IMMEDIATE INVESTIGATION BY OPERATIONS DEPARTMENT PERSONNEL DETERMINED THAT THE LATCH ON THE UNIT 2/3 DG ROOM DOOR HAD FAILED, AND THAT OPENING OF THE DOOR FROM THE REACTOR BUILDING INTO THE INTERLOCK CORRIDOR WOULD CAUSE THE UNIT 2/3 DG ROOM DOOR TO OPEN SLIGHTLY. AN INDIVIDUAL WAS POSTED IN THE AREA TO MAINTAIN ONE OF THE INTERLOCK DOORS IN THE CLOSED POSITION UNTIL REPAIRS WERE COMPLETED BY ELECTRICAL MAINTENANCE. THE SAFETY SIGNIFICANCE OF THIS EVENT WAS MINIMAL AS AN OVERALL REACTOR BUILDING TO ATMOSPHERE DIFFERENTIAL PRESSURE OF >/= 0.25 INCHES OF WATER VACUUM WAS MAINTAINED AT ALL TIMES. A PREVIOUS EVENT INVOLVING FAILURE OF A SECONDARY CONTAINMENT INTERLOCK DOOR WAS REPORTED BY LER 88-8/050249.

[81] FERMI 2 DOCKET 50-341 LER 89-009
OPERATION OF THE CONTAINMENT NITROGEN INERTING AND PURGING SYSTEM ISOLATION VALVES DID NOT MEET THE DESIGN CRITERIA.
EVENT DATE: 042189 REPORT DATE: 061489 NSSS: GE TYPE: BWR

(NSIC 214219) THE CONTAINMENT NITROGEN INERTING AND PURGING SYSTEM ISOLATION VALVES ARE AIR OPERATED AND HAVE TWO SOLENOID VALVES CONTROLLING THEIR INTERRUPTABLE AIR SUPPLY. ONE OF THE TWO SOLENOID VALVES IS QUALIFIED AND THE OTHER IS NOT. MANUAL INITIATION OF THE ISOLATION VALVES WAS ACCOMPLISHED USING THE CONTROL ROOM PUSHBUTTON SWITCHES FOR THE NON QUALIFIED SOLENOIDS. TEST SWITCHES THAT OPERATE THE QUALIFIED SOLENOIDS ARE LOCATED IN THE CONTROL CENTER RELAY ROOM. THE DESIGN OF THESE VALVES WAS UPGRADED IN 1984 AFTER A 10CFR50.55(E) WAS ISSUED. THE 10CFR50.55(E) ITEM WAS CLOSED AS DOCUMENTED IN INSPECTION REPORT 84-46. RE-EVALUATION IN APRIL 1989 CONCLUDED THAT QUALIFIED MANUAL ISOLATION CAPABILITY WAS NEEDED TO FULLY MEET THE DESIGN CRITERIA DESCRIBED IN THE UPDATED FINAL SAFETY ANALYSIS REPORT. IT WAS ALSO DETERMINED THE SURVEILLANCE TESTING WAS BEING PERFORMED SUCH THAT IT DID NOT CONCLUSIVELY PROVE THE QUALIFIED SOLENOID ACTUATED ON RECEIPT OF THE AUTOMATIC ISOLATION SIGNAL. TESTING DID PROVE THE ISOLATION VALVES CLOSED. SUBSEQUENTLY, EACH OF THE SOLENOIDS WAS SATISFACTORILY TESTED. ACCORDINGLY, THE ISOLATION VALVES WERE THEN DECLARED OPERABLE.

[82] FERMI 2 DOCKET 50-341 LER 89-011
THE "B" BACKUP MANUAL SCRAM BREAKER FAILED DUE TO MECHANICAL BINDING.
EVENT DATE: 060289 REPORT DATE: 070389 NSSS: GE TYPE: BWR
VENDOR: I-T-E CIRCUIT BREAKER

(NSIC 214550) ON JUNE 2, 1989, AT 0720 HOURS, A TRANSFER OF REACTOR PROTECTION SYSTEM BUS "B" FROM ITS ALTERNATE POWER SUPPLY TO ITS NORMAL POWER SUPPLY WAS PERFORMED. DURING THIS TRANSFER, THE "B" BACKUP MANUAL SCRAM (BUMS) BREAKER FAILED TO TRIP. THE BREAKER WAS DECLARED INOPERABLE. A HALF SCRAM WAS LEFT INSERTED TO COMPLY WITH THE TECH SPEC ACTION STATEMENT. THE BUMS BREAKER WAS REPLACED. POST MAINTENANCE TESTING WAS SUCCESSFULLY COMPLETED. AT 2020 HOURS, BUMS "B" BREAKER WAS DECLARED OPERABLE. THE FAILED BUMS "B" BREAKER WAS SENT TO DETROIT EDISON'S ENGINEERING RESEARCH DEPARTMENT (ERD) FOR FAILURE ANALYSIS. ERD CONCLUDED THAT THE TOLERANCES OF THE MECHANICAL PARTS THAT INTERACT BETWEEN THE UNDERVOLTAGE TRIP UNIT AND THE CIRCUIT BREAKER SWITCH MAY HAVE CAUSED THE FAILURE OF THIS BREAKER. A DESIGN CHANGE TO REMOVE THE BUMS BREAKERS HAD ALREADY BEEN SCHEDULED FOR THE FIRST REFUELING OUTAGE.

[83] FERMI 2 DOCKET 50-341 LER 89-013
ACTUATION OF THE STANDBY GAS TREATMENT SYSTEM AND ISOLATION OF REACTOR BUILDING HEATING VENTILATION AND AIR CONDITIONING DUE TO PERSONNEL ERROR.
EVENT DATE: 062489 REPORT DATE: 072489 NSSS: GE TYPE: BWR

(NSIC 214868) AT 1500 HOURS, ON JUNE 24, 1989, DIVISION 1 OF THE REACTOR BUILDING

HEATING VENTILATION AND AIR CONDITIONING (RBHVAC) ISOLATED, AND STANDBY GAS TREATMENT SYSTEM (SGTS) AUTOMATICALLY STARTED. THIS OCCURRED WHEN AN INSTRUMENT AND CONTROLS TECHNICIAN WAS INSTALLING A JUMPER TO PREVENT INITIATION OF THE REACTOR BUILDING EXHAUST RADIATION MONITOR DURING REPLACEMENT OF A FLOW SWITCH. OPERATIONS PERSONNEL RESTORED RBHVAC AND SGTS TO THEIR NORMAL LINEUP AT 1510 HOURS. THE FLOW SWITCH WAS INSTALLED AND OPERATIONAL AT 2312 HOURS. THE EVENT WAS CAUSED BY PERSONNEL ERROR. THE TECHNICIAN INVOLVED WAS A UTILITY NON-LICENSED PERSON. THE PLANT MANAGER AND I&C SUPERVISION REVIEWED THE EVENT IN DETAIL, THEN AN ACCOUNTABILITY MEETING WAS HELD. AS A RESULT OF THE REVIEW, A POTENTIAL DESIGN CHANGE HAS BEEN WRITTEN TO IMPROVE TESTABILITY AT THIS PANEL. THIS LICENSEE EVENT REPORT AND ITS ASSOCIATED DEVIATION EVENT REPORT WILL BE GIVEN TO I&C PERSONNEL AS REQUIRED READING.

[84] FERMI 2 DOCKET 50-341 LER 89-012 REV 01
UPDATE ON INADEQUATE TECHNICAL REVIEW OF A TEMPORARY PROCEDURE CHANGE CAUSES DIVISION I ACTUATION.
EVENT DATE: 063089 REPORT DATE: 071789 NSSS: GE TYPE: BWR

(NSIC 214887) ON JUNE 3, 1989, AN INSTRUMENTATION AND CONTROL TECHNICIAN WAS PERFORMING A SURVEILLANCE PROCEDURE ON DIVISION I ISOLATION LOGIC. THE PROCEDURE HAD BEEN CHANGED ON A ONE-TIME BASIS TO ALLOW THE INSTALLATION OF "STAR LUGS" AT SPECIFIC TEST POINTS TO ENHANCE TESTABILITY. "STAR LUGS" HAVE BEEN INSTALLED UNDER APPROVED DESIGN CONTROL MEASURES IN CIRCUITS REQUIRING PERIODIC SURVEILLANCE. UPON LIFTING A LEAD AS DIRECTED BY THE CHANGE TO THE PROCEDURE, VARIOUS DIVISION I ISOLATIONS/ACTUATIONS WERE EXPERIENCED. IT WAS DETERMINED THAT LIFTING THE HEAD HAD REMOVED POWER TO THE CABINET. THIS CAUSED THE ISOLATIONS/ACTUATIONS EXPERIENCED PER DESIGN. THE OPERATORS TOOK THE APPROPRIATE ACTIONS TO RESTORE THE AFFECTED COMPONENTS. THE SURVEILLANCE WAS COMPLETED WITHOUT FURTHER DIFFICULTY. THE EVENT WAS CAUSED BY INADEQUATE TECHNICAL REVIEW OF THE TEMPORARY CHANGE TO THE PROCEDURE. AN ACCOUNTABILITY MEETING WAS HELD BETWEEN MANAGEMENT AND THE PERSONNEL INVOLVED. A LESSONS LEARNED DOCUMENT HAS BEEN DEVELOPED. AN IN-DEPTH TECHNICAL REVIEW WILL BE PERFORMED ON TEMPORARY CHANGES TO I&C PROCEDURES USED TO INSTALL STAR LUGS IN THE FUTURE.

[85] FERMI 2 DOCKET 50-341 LER 89-015
LOSS OF POWER TO DIVISION I REACTOR PROTECTION SYSTEM DUE TO OVERVOLTAGE ON THE MOTOR GENERATOR.
EVENT DATE: 070789 REPORT DATE: 080789 NSSS: GE TYPE: BWR

(NSIC 214997) ON JULY 7, 1989, POWER WAS LOST TO THE DIVISION I REACTOR PROTECTION SYSTEM BUS A WHEN ITS MOTOR GENERATOR SET EXPERIENCED AN OVERVOLTAGE CONDITION. INVESTIGATION REVEALED THAT VOLTAGE METER READINGS HAD DRIFTED UP FROM 120 VOLTS TO 128 VOLTS FOLLOWING ADJUSTMENT OF THE POTENTIOMETER. ADDITIONALLY, A PROBLEM WITH THE STABILITY OF THE INSTALLED VOLTAGE METER INDICATION WAS IDENTIFIED. BOTH THE POTENTIOMETER AND THE VOLTAGE METER WERE REPLACED PRIOR TO PLACING THE MG SET BACK IN SERVICE. ON JULY 18, 1989, A SIMILAR EVENT OCCURRED. TROUBLESHOOTING WAS COMMENCED, BUT A CAUSE FOR THIS PROBLEM HAS NOT YET BEEN IDENTIFIED. THE VOLTAGE REGULATOR WAS REPLACED ON JULY 18, 1989. WHILE THE MG SET WAS RUNNING FOR TEST, A VOLTAGE FLUCTUATION OCCURRED ON JULY 19, 1989. TESTING IS IN PROGRESS IN ORDER TO DETERMINE THE CAUSE OF THIS EVENT. SEVERAL MG SET PARAMETERS ARE BEING CONTINUALLY MONITORED. FOLLOWING DETERMINATION OF THE ROOT CAUSE FOR THESE EVENTS, A SUPPLEMENT TO THIS LER WILL BE SUBMITTED WITHIN THIRTY DAYS.

[86] FERMI 2 DOCKET 50-341 LER 89-016
RESIDUAL HEAT REMOVAL SERVICE WATER COOLING TOWER FAN BRAKE INOPERABLE DUE TO LOW NITROGEN PRESSURE.
EVENT DATE: 071189 REPORT DATE: 081089 NSSS: GE TYPE: BWR

VENDOR: MARLEY CO., THE

(NSIC 214998) ON 7/11/89, DIVISION I OF THE RESIDUAL HEAT REMOVAL SERVICE WATER SYSTEM WAS CONSERVATIVELY DECLARED INOPERABLE DUE TO LOW NITROGEN PRESSURE FOR ONE OF THE MECHNICAL DRAFT COOLING TOWER FAN BRAKES. THE BRAKES WERE INSTALLED TO PROTECT THE FANS FROM OVERSPEEDING IN THE EVENT OF A DESIGN-BASIS TORNADO. DURING THE INVESTIGATION AND REPAIR OF THE CONDITION, IT WAS NOTED THAT CONTROL POWER TO THE BRAKES WAS BEING MAINTAINED IN THE AC SUPPLY POSITION RATHER THAN THE DC SUPPLY POSITION AS SHOWN ON THE DRAWING. LACK OF NITROGEN PRESSURE WOULD HAVE PREVENTED THE BRAKE FROM FULFILLING ITS DESIGN FUNCTION IN THE EVENT A TORNADO HAD OCCURRED. A REVIEW OF THE ORIGINAL ANALYSIS TO DETERMINE IF THE BRAKES ARE ACTUALLY NECESSARY IS BEING CONDUCTED. THE NITROGEN SUPPLY WAS RESTORED AND A LEAKY HOSE REPLACED. THE SYSTEM OPERATING PROCEDURE WAS REVISED AND THE POWER SUPPLY PROPERLY ALIGNED.

[87] FITZPATRICK DOCKET 50-333 LER 89-012
UNANALYZED DIESEL GENERATOR TEST MODE AND HYPOTHETICAL 3-PHASE FAULT CREATE POTENTIAL TO EXCEED 4KV BREAKER CURRENT RATINGS AND POTENTIAL LOSS OF EMERGENCY CORE COOLING CAPABILITY.
EVENT DATE: 071489 REPORT DATE: 081189 NSSS: GE TYPE: BWR

(NSIC 214995) TECHNICAL SPECIFICATIONS REQUIRE A MONTHLY 1 HR FULL LOAD TEST FOR EACH OF THE 2 SETS OF EMERGENCY DIESEL GENERATORS (EDGS)(EK). THIS IS ACCOMPLISHED BY CONNECTING THE EDGS IN PARALLEL ELECTRICALLY WITH THE NORMAL STATION SERVICE TRANSFORMER (NSST)(EA) TO SUPPLY LOADING TO THE GENERATOR. AN ANALYSIS REQUESTED BY AN NRC SAFETY SYSTEM FUNCTIONAL INSPECTION (SSFI) FOUND THAT IN THIS ELECTRICAL CONFIGURATION, THE OCCURRENCE OF A HYPOTHETICAL 3-PHASE BOLTED BUS FAULT COULD RESULT IN CURRENTS WHICH WOULD EXCEED THE MOMENTARY DUTY RATINGS FOR CERTAIN 52 EMERGENCY SWITCHGEAR 4160 VOLT CIRCUIT BREAKERS (EB)(EA) BY 23% TO 27% DEPENDING ON THE BUS IN WHICH THE FAULT OCCURRED. THIS POSTULATED EVENT COULD RESULT IN POTENTIAL FAILURE OF THE PROTECTIVE FUNCTIONS OF THE SWITCHGEAR AND LOSS OF POWER TO THE EMERGENCY BUS AND EMERGENCY CORE COOLING SYSTEMS REQUIRED FOR SAFE SHUTDOWN OF THE PLANT. THE PROBABILITY OF OCCURRENCE OF A BOLTED BUS FAULT COMBINED WITH SWITCHGEAR FAILURE DURING THE 2 HR TEST PERIOD HAS BEEN CALCULATED TO BE LESS THAN 9.24 E-7 BY A SIMPLIFIED BUT CONSERVATIVE PRELIMINARY ANALYSIS. AN EVALUATION AND JUSTIFICATION FOR CONTINUED OPERATION UNTIL THE MARCH 1990 REFUEL OUTAGE HAS BEEN PREPARED. AS A SHORT-TERM CORRECTIVE ACTION THE SURVEILLANC? TO REDUCE RISK OF ACCIDENTAL FAULT.

[88] FT. CALHOUN 1 DOCKET 50-285 LER 89-016 REV 01
UPDATE ON AUXILIARY FEEDWATER PUMP FW-10 OUTSIDE DESIGN BASIS.
EVENT DATE: 061689 REPORT DATE: 090189 NSSS: CE TYPE: PWR
VENDOR: MOORE PRODUCTS COMPANY

(NSIC 215125) ON 6/16/89, ENGINEERING EVALUATION OF TEST RESULTS REVEALED THAT, FOR AN UNKNOWN PERIOD PRIOR TO 6/14/89, AUXILIARY FEEDWATER PUMP FW-10 OPERABILITY WAS OUTSIDE THE DESIGN BASIS FOR CERTAIN ACCIDENT CONDITIONS DUE TO INOPERABILITY OF THE PUMP PNEUMATIC SPEED CONTROL LOOP. PROBLEMS WITH THE CONTROL LOOP HAD BEEN FOUND ON 6/13/89, AFTER FW-10 WAS TAKEN OUT OF SERVICE FOR SPECIAL TESTING. THESE PROBLEMS WOULD HAVE LIMITED PUMP SPEED AND DISCHARGE PRESSURE BELOW THAT NEEDED TO INJECT WATER INTO THE STEAM GENERATORS UNDER SOME ACCIDENT CONDITIONS. BECAUSE THE CONTROL LOOP COULD NOT BE REPAIRED AND TESTED EXPEDITIOUSLY, THE AIR SUPPLY TO THE CONTROL LOOP WAS VALVED OUT ON 6/14/89. THIS ALLOWED PUMP SPEED AND DISCHARGE PRESSURE TO BE CONTROLLED BY THE MECHANICAL SPEED LIMITER ON THE MAIN GOVERNOR AS DESIGNED TO MEET ACCIDENT MITIGATION CRITERIA. THE PUMP WAS DETERMINED TO BE OPERABLE AND RETURNED TO SERVICE. AN INVESTIGATION WAS INITIATED TO DETERMINE THE DURATION AND CAUSE OF THE CONTROL LOOP INOPERABILITY. THE NRC REGIONAL OFFICE WAS BRIEFED ON 6/13/89, AND, FOLLOWING THE DETERMINATION OF REPORTABILITY, THE NRC OPERATIONS CENTER WAS

NOTIFIED ON 6/16/89 AT 1646 HOURS, PURSUANT TO 10 CFR 50.72(B)(1)(II). FORT CALHOUN STATION UNIT NO. 1 WAS AT APPROX. 100% POWER AT THE TIME OF DETERMINATION.

[89] FT. CALHOUN 1 DOCKET 50-285 LER 89-017
RAW WATER SYSTEM OUTSIDE ITS DESIGN BASIS.
EVENT DATE: 063089 REPORT DATE: 073189 NSSS: CE TYPE: PWR
VENDOR: MISSION MANUFACTURING COMPANY

(NSIC 214948) ON JUNE 24, 1989, DURING THE REMOVAL OF THE PUMP ASSEMBLY ON RAW WATER PUMP AC-10A FOR MAINTENANCE, AN INTERNAL VALVE COMPONENT FROM CHECK VALVE RW-125 WAS FOUND LYING ON THE PUMP DISCHARGE VANE. REPAIR OR REPLACEMENT OF THE VALVE INTERNALS COULD NOT BE ACCOMPLISHED WITHIN THE TIME REQUIREMENT OF THE TECHNICAL SPECIFICATION LCO. ON JUNE 29, 1989 PLANT MANAGEMENT DECIDED TO PLACE THE CHECK VALVE BODY BACK INTO THE SYSTEM WITH THE INTERNALS REMOVED. A SAFETY ANALYSIS FOR OPERABILITY WAS PERFORMED TO JUSTIFY CONTINUED OPERATION OUTSIDE THE SYSTEM DESIGN BASIS WITH PRESCRIBED COMPENSATORY ACTIONS. ON JUNE 30, 1989, THE AFFECTED TRAIN OF THE RAW WATER SYSTEM WAS RETURNED TO SERVICE. PURSUANT TO 10CFR50.72 (B)(1)(II)(B), THE NRC WAS NOTIFIED AT APPROXIMATELY 1610 ON JUNE 30, 1989. THIS LER IS SUBMITTED PURSUANT TO 10CFR50.73(B)(2)(II)(B). THE PLANT WAS OPERATING AT 100 PERCENT POWER DURING THIS PERIOD. THIS EVENT IS DUE PRIMARILY TO LACK OF LONG TERM PREVENTIVE MAINTENANCE AND INSERVICE TESTING PROGRAMS FOR THE RAW WATER CHECK VALVES. PREVIOUSLY COMMITTED ELEMENTS OF THE FORT CALHOUN STATION SAFETY ENHANCEMENT PROGRAM SHOULD PRECLUDE RECURRENCE. RW-125 AND CORRESPONDING CHECK VALVES IN THE OTHER TRAINS OF THE RAW WATER SYSTEM WILL BE REPLACED AT THE EARLIEST OPPORTUNITIES.

[90] FT. ST. VRAIN DOCKET 50-267 LER 89-003
STANDBY DIESEL ENGINE COOLANT FLOW DIRECTORS FOUND DISLODGED DUE TO IMPROPER INSTALLATION.
EVENT DATE: 031789 REPORT DATE: 041789 NSSS: GA TYPE: HTGR

(NSIC 215088) INADEQUATE INSTRUCTIONS IN O&M MANUAL. ON 2/27/89, WHILE PERFORMING MAINTENANCE ON THE '1D' STANDBY DIESEL GENERATOR ENGINE, IT WAS DISCOVERED THAT AN ENGINE HEAD COOLANT FLOW DIRECTOR HAD BECOME DISLODGED FROM ITS PORT. THESE FLOW DIRECTORS ARE LOCATED IN THE ENGINE CYLINDER HEAD AND FUNCTION TO DIRECT COOLING WATER FLOW ACROSS PORTIONS OF THE ENGINE HEAD DIRECTLY ABOVE THE COMBUSTION CHAMBER. FURTHER INSPECTION OF THE '1D' ENGINE AND THE OTHER 3 DIESEL ENGINES IDENTIFIED ADDITIONAL FLOW DIRECTORS (8 TOTAL) HAD BECOME DISLODGED FROM THEIR PORTS WITHIN THE CYLINDER HEADS. THE REACTOR WAS IN A SHUTDOWN CONDITION AT THE TIME OF THESE DISCOVERIES. ON 3/17/89, FOLLOWING SEVERAL DISCUSSIONS WITH THE MANUFACTURER, IT WAS DETERMINED THAT THIS CONDITION ALONE COULD HAVE DEGRADED DIESEL ENGINE PERFORMANCE. NRC NOTIFICATION WAS MADE 3/17/89 PER THE REQUIREMENTS OF 10 CFR 50.72(B)(2)(III)(A). THE MANUFACTURERS "OPERATIONS AND MAINTENANCE" MANUAL DOES NOT GIVE ADEQUATE INSTRUCTION ON THE INSTALLATION OF FLOW DIRECTORS INTO THE CYLINDER HEADS. THIS RESULTED IN A TOTAL OF EIGHT FLOW DIRECTORS INTO THE CYLINDER HEADS. THIS RESULTED IN A TOTAL OF EIGHT FLOW DIRECTORS (8 OF 192) BECOMING DISLODGED FROM THEIR COOLING PORTS. PSC HAS REPLACED THE CYLINDER HEADS ON ALL FOUR STANDBY DIESEL GENERATOR ENGINES WITH HEADS CONTAINING PROPERLY INSTALLED FLOW DIRECTORS.

[91] FT. ST. VRAIN DOCKET 50-267 LER 89-004
WIDE RANGE NUCLEAR CHANNEL UPSCALED FROM NOISE SOURCE CAUSING SPURIUOS SCRAM.
EVENT DATE: 032189 REPORT DATE: 042189 NSSS: GA TYPE: HTGR

(NSIC 215087) CAUSE - DESIGN DEFICIENCY OF WIDE RANGE CHANNELS. DURING THE TIME PERIOD FROM 3/21/89 THROUGH 3/24/89, WITH THE REACTOR SHUTDOWN FOR OUTAGE ACTIVITIES, THE PLANT PROTECTIVE SYSTEM (PPS) REACTOR SCRAM CIRCUITRY WAS SPRUIOUSLY ACTUATED 13 SEPARATE TIMES. IN THE MAJORITY OF THESE EVENTS, ALL 37

CONTROL RODS WERE ALREADY FULLY INSERTED IN THE CORE AND NO CONTROL ROD MOVEMENT OCCURRED AS A RESULT OF THESE ACTUATIONS. HOWEVER, THERE WERE SEVERAL OCCURRENCES WHEN A SINGLE CONTROL ROD PAIR WAS WITHDRAWN FOR SURVEILLANCE TESTING AND INSERTED PROPERLY UPON SCRAM ACTUATION. THE CAUSE FOR THE 13 SPURIOUS REACTOR SCRAMS WAS ELECTRONIC NOISE INDUCTION INTO WIDE RANGE CHANNELS III AND V. INVESTIGATIVE EFFORTS HAVE IDENTIFIED SOME BUT NOT ALL SOURCES OF NOISE IN THE WIDE RANGE CHANNELS. IT HAS BEEN DETERMINED THAT, FOR THE DURATION OF THE PLANT OPERATIONAL PHASE, THE MODIFICATIONS NECESSARY TO CORRECT THE WRC NOISE PROBLEM ARE NO LONGER JUSTIFIED. THIS DECISION IS BASED ON THE TIME REQUIRED TO COMPLETE THE CHANGES, THE ABSENCE OF WRC NOISE RELATED TRANSIENTS DURING REACTOR OPERATION (GREATER THAN 5%), THE NON-SAFETY SIGNIFICANCE NATURE OF THE NOISE PROBLEM, AND THE SHORT DURATION OF FUTURE PLANT OPERATION. PSC IS ANALYZING WHETHER THE WRCS WILL BE USED DURING THE PLANT DEFUELING PHASE TO DETERMINE THE NEED FOR FUTURE WRC MODIFICATIONS.

[92] FT. ST. VRAIN DOCKET 50-267 LER 89-005
OPERATION IN VIOLATION OF LCO 4.2.7.
EVENT DATE: 032389 REPORT DATE: 042489 NSSS: GA TYPE: HTGR

(NSIC 215086) CAUSE - INADEQUATE PROCEDURAL CONTROLS. AT 2218 HOURS ON 3/23/89, DURING PREPARATIONS TO RESTART THE PLANT FOLLOWING AN EXTENDED OUTAGE, THE PRIMARY COOLANT SYSTEM WAS PRESSURIZED TO GREATER THAN 100 PSIA WITH THE REGION 27 CONTROL ROD DRIVE (CRD) PENETRATION INTERSPACE ISOLATED (I.E., NOT PRESSURIZED). PER THE REQUIREMENTS OF LCO 4.2.7, THE PCRV (I.E., PRIMARY COOLANT SYSTEM) SHALL NOT BE PRESSURIZED TO MORE THAN 100 PSIA UNLESS THE INTERSPACES BETWEEN THE PRIMARY AND SECONDARY PENETRATION CLOSURES ARE MAINTAINED AT A PRESSURE GREATER THAN PRIMARY COOLANT PRESSURE. THEREFORE, THIS EVENT CONSTITUTES OPERATION PROHIBITED BY THE TECH SPECS AND IS BEING REPORTED PURSUANT TO THE REQUIREMENTS OF 10 CFR 50.73(A)(2)(I)(B). AT 0102 HOURS ON 3/24/89, WITH THE PRIMARY COOLANT SYSTEM PRESSURIZED TO APPROXIMATELY 107 PSIA, THE REGION 27 PENETRATION INTERSPACE ISOLATION VALVE WAS OPENED THEREBY PRESSURIZING THE INTERSPACE TO GREATER THAN PRIMARY COOLANT PRESSURE AS REQUIRED. THIS EVENT IS ATTRIBUTED TO INADEQUATE PROCEDURAL CONTROLS ON PRIMARY COOLANT SYSTEM PRESSURIZATION. IN ADDITION, PSC HAS INITIATED A HUMAN PERFORMANCE EVALUATION SYSTEM INVESTIGATION TO ADDRESS OTHER CONTRIBUTING FACTORS FOR THIS EVENT.

[93] FT. ST. VRAIN DOCKET 50-267 LER 89-006
TECH SPEC SURVEILLANCES NOT PERFORMED AS REQUIRED.
EVENT DATE: 032989 REPORT DATE: 042889 NSSS: GA TYPE: HTGR

(NSIC 215085) CAUSE - INADEQUATE ADMINISTRATIVE CONTROLS. ON 3/29/89 AND AGAIN ON 4/5/89 WITH THE REACTOR OPERATING AT POWER, IT WAS DISCOVERED THAT TWO RESCHEDULED TECH SPEC SURVEILLANCE REQUIREMENTS HAD NOT BEEN COMPLETED AS REQUIRED. THESE TWO SURVEILLANCES ARE ASSOCIATED WITH (1) DEMONSTRATING OPERABILITY OF THE STANDBY DIESEL GENERATOR SHUTDOWN AND DECLUTCH FUNCTIONS AND (2) DEMONSTRATING OPERABILITY OF THE GASEOUS REACTOR BUILDING VENTILATION EXHAUST STACK MONITORS. FAILURE TO PERFORM THESE SURVEILLANCES IN ACCORDANCE WITH THE REQUIREMENTS ESTABLISHED IN THE FSV TECH SPECS CONSTITUTES A CONDITION PROHIBITED BY THE TECH SPECS AND IS BEING REPORTED HEREIN PER 10 CFR 50.73(A)(2)(I)(B). THIS EVENT WAS CAUSED BY INADEQUATE CONTROL OF RESCHEDULED TECH SPEC SURVEILLANCE PROCEDURES. PSC IS ADDRESSING THIS DEFICIENCY. UPON DISCOVERING THAT THESE SURVEILLANCE TESTS HAD NOT BEEN PERFORMED AS REQUIRED, BOTH TESTS WERE ISSUED AND COMPLETED. ALL EQUIPMENT TESTED WAS FOUND TO BE OPERABLE AS DESIGNED.

[94] GINNA DOCKET 50-244 LER 89-007
SAFETY INJECTION PUMPS INOPERABILITY DUE TO FLOW METER CALIBRATION ERRORS.
EVENT DATE: 061989 REPORT DATE: 071989 NSSS: WE TYPE: PWR

(NSIC 214868) ON JUNE 19, 1989 AT 1440 EDST WITH THE REACTOR AT APPROXIMATELY 99% FULL POWER THE "B" AND "C" SAFETY INJECTION (SI) PUMPS WERE DECLARED INOPERABLE DUE TO ASSESSED DESIGN FLOW DELIVERY CONCERNS. DECLARING TWO (2) SI PUMPS INOPERABLE PLACED THE PLANT OUTSIDE THE TECHNICAL SPECIFICATIONS REQUIRING A PLANT SHUTDOWN. WHILE IN THE PROCESS OF PLANT SHUTDOWN THE SI PUMP FLOWS WERE RETURNED TO THE REQUIRED FLOW RATES BY PUMP MINIMUM FLOW RECIRCULATION LINE VALVE THROTTLING. ON JUNE 21, 1989 AT 1401 A SIMILAR PROBLEM OCCURRED WITH THE "B" AND "C" SI PUMPS AND PLANT MANAGEMENT DECIDED TO SHUT THE PLANT DOWN UNTIL THE SI PUMP FLOW CONCERNS WERE RESOLVED. THE PLANT WAS SHUTDOWN AND SUBSEQUENTLY COOLED DOWN TO LESS THAN 350F. ORIGINAL CALIBRATION DATA PROVIDED BY THE PLANT DESIGN WAS INCORRECT FOR THE INSTALLED APPLICATION. THE UNDERLYING CAUSE OF THE EVENT RESULTED FROM INCORRECT ORIGINAL CALIBRATION DATA, PROVIDED BY THE PLANT DESIGNER FOR THE INSTALLED SYSTEM.

[95] GINNA DOCKET 50-244 LER 89-008
DROPPED CONTROL ROD DURING ROD CONTROL SYSTEM EXERCISES CAUSES AUTOMATIC TURBINE RUNBACK.
EVENT DATE: 070689 REPORT DATE: 080789 NSSS: WE TYPE: PWR

(NSIC 214978) ON JULY 6, 1989 AT 1557 EDST WITH THE REACTOR AT APPROXIMATELY 99% FULL POWER, A TURBINE RUNBACK TO APPROXIMATELY 75% REACTOR POWER OCCURRED DUE TO A DROPPED CONTROL ROD. THE CONTROL ROOM OPERATORS PERFORMED THE APPLICABLE STEPS OF ABNORMAL PROCEDURE AP-TURB.2 (AUTOMATIC TURBINE RUNBACK) AND STABILIZED THE PLANT. SUBSEQUENTLY THE CONTROL ROOM OPERATORS REDUCED REACTOR POWER TO LESS THAN 50% AND RETRIEVED THE DROPPED CONTROL ROD PER EQUIPMENT RESTORATION PROCEDURE ER-RCC.1 (RETRIEVAL OF A DROPPED RCC). THE UNDERLYING CAUSE OF THE DROPPED CONTROL ROD WAS A SPURIOUS EVENT.

[96] GINNA DOCKET 50-244 LER 89-009
FAILURE OF CONTROL ROD POSITION INDICATION SYSTEM DUE TO A GROUNDED COIL STACK CAUSES PLANT SHUTDOWN PER TECH SPEC.
EVENT DATE: 072989 REPORT DATE: 082889 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 215108) ON JULY 29, 1989 AT 1843 EDST WITH THE REACTOR AT APPROXIMATELY 100% FULL POWER THE CONTROL ROD POSITION INDICATION SYSTEM WAS RENDERED INOPERABLE DUE TO A CONTROL ROD POSITION INDICATION COIL STACK SHORTING TO GROUND. PLANT SHUTDOWN TO HOT SHUTDOWN WAS COMMENCED WITHIN ONE HOUR TO COMPLY WITH PLANT TECHNICAL SPECIFICATIONS ON CONTROL ROD POSITION INDICATION OPERABILITY REQUIREMENTS. HOT SHUTDOWN WAS ACHIEVED AND ALL CONTROL RODS WERE INSERTED IN THE CORE AND THE REACTOR TRIP BREAKERS WERE OPENED. THE FAULTY CONTROL ROD POSITION INDICATION COIL STACK WAS REPLACED WITH A QUALIFIED SPARE, TESTED SATISFACTORILY AND RETURNED TO SERVICE.

[97] GINNA DOCKET 50-244 LER 89-010
SAFEGUARDS BUS UNDERVOLTAGE RELAY ACTUATION DUE TO A LOOSE FUSE CONNECTION CAUSES AUTOMATIC START OF THE "B" EMERGENCY DIESEL GENERATOR.
EVENT DATE: 073089 REPORT DATE: 082989 NSSS: WE TYPE: PWR

(NSIC 215109) ON JULY 30, 1989, AT 0806 EDST WITH THE REACTOR IN THE HOT SHUTDOWN CONDITION, THE "B" EMERGENCY DIESEL GENERATOR STARTED AUTOMATICALLY DUE TO AN INITIATION SIGNAL FROM THE BUS 16 UNDERVOLTAGE MONITORING SYSTEM. THE "B" EMERGENCY DIESEL GENERATOR, AFTER STARTING, ATTAINED PROPER VOLTAGE AND FREQUENCY, BUT, BY DESIGN, DID NOT TIE INTO BUS 16, BECAUSE BUS 16 WAS AT ITS PROPER VOLTAGE, FED FROM ITS NORMAL POWER SUPPLY. IMMEDIATE OPERATOR ACTION WAS TO VERIFY THAT BUS 16 WAS ENERGIZED AND THAT THE "B" EMERGENCY DIESEL GENERATOR WAS OPERATING PROPERLY. THE CAUSE OF THE EVENT WAS DETERMINED TO BE A LOOSE CONNECTION ON THE BUS 16 UNDERVOLTAGE MONITORING SYSTEM. CORRECTIVE ACTION TAKEN

WAS TO TIGHTEN THE LOOSE CONNECTION FOLLOWED BY A SYSTEM CHECKOUT AND SATISFACTORY TEST AND RETURN TO SERVICE.

[98]   GRAND GULF 1   DOCKET 50-416   LER 89-008 REV 01
UPDATE ON TWO REDUNDANT SECONDARY CONTAINMENT ISOLATION VALVES FAILED TO CLOSE.
EVENT DATE: 052389   REPORT DATE: 073189   NSSS: GE   TYPE: BWR

(NSIC 214901) ON MAY 23, 1989 TWO REDUNDANT SECONDARY CONTAINMENT ISOLATION DAMPERS, QLT42F019 AND QLT42F020, FAILED TO CLOSE WITHIN THE 4 SECOND TIME LIMIT OF TECHNICAL SPECIFICATION 3.6.6.2. IT WAS SUSPECTED THAT THE INTERNAL DISK OF EACH EXHAUST VALVE, WHICH SHUTTLES TO VENT OR ADMIT AIR ON THE ACTUATOR CYLINDER FAILED TO PROPERLY LIFT TO VENT AIR PRESSURE DURING INITIAL ATTEMPTS TO CLOSE THE DAMPERS. THE EXHAUST VALVE ASSOCIATED WITH THE ACTUATOR OF EACH DAMPER WAS REPLACED. A VENDOR ASSESSMENT OF A REMOVED EXHAUST VALVE FOR POSSIBLE MALFUNCTION WILL BE PERFORMED. ON JUNE 16, 1989, THE QLT42F020 DAMPER AGAIN FAILED TO CLOSE WITHIN THE 4 SECOND TIME LIMIT. THE DAMPER ACTUATOR WAS DISASSEMBLED AND NO OBVIOUS FAILURES OR DEFECTS WERE OBSERVED. THE ACTUATOR CONTAINED MOBILGREASE 28 WHICH WAS PREVIOUSLY IDENTIFIED IN A 10CFR21 REPORT AS HAVING A TENDENCY TO CAUSE SEALS TO SWELL AND PRODUCE SLOWER STROKE TIMES. HOWEVER, THE MODEL OF ACTUATOR INSTALLED ON THESE DAMPERS WAS NOT PROJECTED TO DEGRADE IN STROKING TIME. THIS WAS NOT CONSIDERED A POTENTIAL CONTRIBUTOR AT THE TIME OF THE FIRST INCIDENT SINCE THE QLT42F019 HAD BEEN REFURBISHED AND LUBRICATED WITH THE APPROVED MOLYKOTE 44 DURING THE THIRD REFUELING OUTAGE. GRAND GULF HAS A 5 YEAR PROGRAM TO REFURBISH BETTIS ACTUATORS WHICH ENSURES THAT APPROVED MOLYKOTE 44 IS USED AS A LUBRICANT DURING THE REBUILD.

[99]   GRAND GULF 1   DOCKET 50-416   LER 89-009
RWCU/RCIC ISOLATIONS AND DELINQUENT LCO ACTION DUE TO PERSONNEL ERROR.
EVENT DATE: 062089   REPORT DATE: 072089   NSSS: GE   TYPE: BWR

(NSIC 214830) ON JUNE 20, 1989 RELATED ACTIVITIES AND A COMMON ERROR LED TO THE OCCURRENCE OF THREE REPORTABLE EVENTS WITHIN A SHORT PERIOD OF TIME. AT 0500 THE REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM ISOLATED ON A HIGH STEAM LINE FLOW SIGNAL. AT 0530 HOURS, THE REACTOR WATER CLEANUP (RWCU) SYSTEM ISOLATED ON HIGH DIFFERENTIAL TEMPERATURE IN THE RWCU PUMP ROOM. AT 0415, THE DIVISION 1 DIESEL GENERATOR WAS REMOVED FROM SERVICE FOR SCHEDULED MAINTENANCE REQUIRING THE OFFSITE A.C. POWER SOURCES TO BE DEMONSTRATED OPERABLE WITHIN ONE HOUR. THIS TECHNICAL SPECIFICATION ACTION WAS NOT PERFORMED UNTIL APPROXIMATELY 3 HOURS AFTER THE DIESEL GENERATOR WAS MADE OPERABLE. THE OVERALL CAUSE OF THE THREE INCIDENTS WAS PERSONNEL ERROR BY THE LICENSED SHIFT SUPERINTENDENT. RESTORATION FROM A PLANNED MAINTENANCE OUTAGE ON THE RCIC SYSTEM AND PREPARATION FOR A DIVISION 1 DIESEL GENERATOR MAINTENANCE OUTAGE WERE IN PROGRESS. THE SHIFT SUPERINTENDENT BECAME SO INVOLVED WITH THE ACTUAL WORK DETAILS IN RESTORING RCIC AND PREPARING FOR THE DIVISION 1 DIESEL GENERATOR MAINTENANCE THAT HE FAILED TO MAINTAIN HIS OVERVIEW FUNCTION. ALL SHIFT SUPERINTENDENTS WERE INSTRUCTED TO CONTROL THE AMOUNT OF WORK PERFORMED AT ONE TIME TO A LEVEL THEY CAN PROPERLY OVERSEE. THE RESPONSIBLE SHIFT SUPERINTENDENT WAS INSTRUCTED ON HIS RESPONSIBILITY TO MAINTAIN AN OVERVIEW OF ACTIVITIES OCCURRING ON HIS SHIFT.

[100]   GRAND GULF 1   DOCKET 50-416   LER 89-010
REACTOR SCRAM CAUSED BY LIGHTNING STRIKE.
EVENT DATE: 072289   REPORT DATE: 082189   NSSS: GE   TYPE: BWR

(NSIC 215076) ON JULY 22, 1989, A SEVERE ELECTRICAL STORM PASSED OVER GRAND GULF NUCLEAR STATION. DURING THE STORM, THE REACTOR AUTOMATICALLY SCRAMMED DUE TO A HIGH NEUTRON FLUX SIGNAL ON THE AVERAGE POWER RANGE MONITORS (APRMS). ADDITIONALLY, A SPIKE TO THE REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM LOGIC CAUSED RCIC TO AUTOMATICALLY ACTUATE AND INJECT INTO THE REACTOR VESSEL. REACTOR

WATER LEVEL DECREASED TO -15 INCHES AND WAS RAISED TO THE LEVEL 8 HIGH LEVEL TRIP (+53.5 INCHES) IN APPROXIMATELY 2 MINUTES. A CHANNEL CHECK OF APRM INDICATIONS WAS PERFORMED DURING PLANT RESTART ON JULY 2 3, 1989 TO CONFIRM PROPER RESPONSE. NO ABNORMALITIES WERE OBSERVED. SYSTEM ENERGY REQUISITIONED A SPECIALIST IN LIGHTNING PROTECTION TO PERFORM A PLANT SURVEY AND STUDY OF THE EXISTING PLANT LIGHTNING PROTECTION SYSTEM. THE CONTRACTOR HAS SUBMITTED PROPOSALS TO PROVIDE LIGHTNING DISSIPATION ARRAYS ON VULNERABLE STRUCTURES. SYSTEM ENERGY IS PRESENTLY EVALUATING IMPLEMENTATION METHODS AND CONSTRUCTING A SCHEDULE FOR IMPLEMENTATION . CURRENT PROGRESS INDICATES THAT IMPLEMENTATION WILL LIKELY BE COMPLETED BY DECEMBER 31, 1989. IN ANY CASE, IMPLEMENTATION WILL BE COMPLETED NO LATER THAN THE STARTUP FROM THE FOURTH REFUELING OUTAGE (RF04).

[101] GRAND GULF 1 DOCKET 50-416 LER 89-011
MISSED CHEMISTRY SURVEILLANCE DUE TO PERSONNEL ERROR.
EVENT DATE: 072689 REPORT DATE: 082289 NSSS: GE TYPE: BWR

(NSIC 215074) ON JULY 26, 1989 IT WAS DISCOVERED THAT A FILTER AND CARTRIDGE HOLDER USED FOR A SAMPLE COLLECTION OF THE FUEL HANDLING AREA VENTILATION (FHAV) EFFLUENT HAD BEEN INSTALLED ON JULY 19, 1989 WITHOUT THE PARTICULATE FILTER AND IODINE CARTRIDGE INSTALLED. HENCE NO PARTICULATE OR IODINE SAMPLE FOR THE FHAV EXHAUST WAS COLLECTED AND ANALYZED FOR THE PERIOD JULY 19 - 26, 1989 AS REQUIRED BY TECHNICAL SPECIFICATION 4.11.2.1.2. PERSONNEL ERROR BY THE CHEMISTRY TECHNICIAN IN INSTALLING AN EMPTY FILTER AND CARTRIDGE HOLDER IN THE FHAV SAMPLE PANEL WAS THE CAUSE OF NOT MEETING THE SURVEILLANCE REQUIREMENT. THE SURVEILLANCE PROCEDURE DID NOT REQUIRE INSPECTION OF THE HOLDER PRIOR TO ITS INSTALLATION. THE SURVEILLANCE PROCEDURE HAS BEEN CHANGED TO REQUIRE THE INDIVIDUAL TO CONFIRM BY INSPECTION THAT THE SAMPLE HOLDER IS LOADED WITH A FILTER AND CARTRIDGE PRIOR TO ITS INSTALLATION. OTHER EVOLUTIONS INVOLVING THE CHANGING OF FILTERS WERE REVIEWED TO ENSURE THE POTENTIAL FOR SIMILAR ERRORS DID NOT EXIST. BASED ON AN EVALUATION OF AVAILABLE DATA IT HAS BEEN CONCLUDED THAT THE FHAV EFFLUENT DURING THE WEEK OF JULY 19-26 POSED NO THREAT TO THE HEALTH AND SAFETY OF THE PUBLIC.

[102] HATCH 2 DOCKET 50-366 LER 89-003
ANALYSIS ON LIQUID EFFLUENT NOT PERFORMED PER TECH SPECS.
EVENT DATE: 072889 REPORT DATE: 082289 NSSS: GE TYPE: BWR

(NSIC 215140) ON 7/28/89, AT APPROXIMATELY 0920 CDT, UNIT 2 WAS IN THE RUN MODE AT AN APPROXIMATE POWER LEVEL OF 1905 CMWT (APPROXIMATELY 78 PERCENT OF RATED THERMAL POWER). AT THAT TIME, A DAYSHIFT NON-LICENSED RADWASTE OPERATOR DISCOVERED CHEMICAL WASTE SAMPLE TANK (CWST) "A" WAS BEING DISCHARGED WITHOUT THE ANALYSIS REQUIRED BY UNIT 2 TECHNICAL SPECIFICATIONS SECTIONS 4.11.1.1.1 AND 4.11.1.1.2 HAVING BEEN PERFORMED. THE "B" CWST HAD BEEN SAMPLED AND ANALYZED AS REQUIRED BY THE UNIT 2 TECHNICAL SPECIFICATIONS, BUT, AT APPROXIMATELY 0821 CDT, THE NIGHTSHIFT NON- LICENSED RADWASTE OPERATOR INADVERTENTLY BEGAN RELEASING THE "A" CWST. SUBSEQUENT ANALYSIS DETERMINED THE "A" CWST CONTENTS SUITABLE FOR DISCHARGE. THE ROOT CAUSE OF THIS EVENT WAS COGNITIVE PERSONNEL ERROR BY NON-LICENSED OPERATIONS PERSONNEL. DIRECTLY CONTRIBUTING TO THE ERROR WAS A LACK OF PHYSICAL CONTROLS TO HELP PREVENT OPERATION OF THE WRONG DISCHARGE VALVE SWITCH. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDED THE IMMEDIATE TERMINATION OF THE DISCHARGE OF THE "A" CWST, THE COMPLETION OF THE REQUIRED ANALYSIS OF THE "A" CWST, THE IMPLEMENTATION OF PHYSICAL CONTROLS ON THE USE OF THE TANKS' DISCHARGE VALVES, AND THE REVISION OF RADWASTE DISCHARGE PROCEDURES TO REFLECT THE NEW PHYSICAL CONTROLS.

[103] HOPE CREEK 1 DOCKET 50-354 LER 89-013
FAILURE TO PERFORM REQUIRED LOCAL LEAK RATE TESTING ON HPCI/RCIC TURBINE EXHAUST VACUUM RELIEF PIPING DUE TO LACK OF INCLUSION IN THE INSERVICE TEST PROGRAM MANUAL.
EVENT DATE: 062989 REPORT DATE: 072689 NSSS: GE TYPE: BWR

(NSIC 214928) ON 6/29/89 DURING A PRE-REFUEL OUTAGE AUDIT OF THE LOCAL LEAK RATE TEST (LLRT) PROGRAM, AN INSERVICE INSPECTION (ISI) SUPERVISOR DETERMINED THAT THE HIGH PRESSURE COOLANT INJECTION (HPCI) AND REACTOR CORE ISOLATION COOLING (RCIC) TURBINE EXHAUST VACUUM RELIEF NETWORK OF VALVES AND PIPING SHOULD BE INCLUDED WITHIN THE SCOPE OF THE LLRT PROGRAM. THIS CONCLUSION WAS REACHED FOLLOWING RECEIPT OF CORRESPONDENCE FROM PSE&G LICENSING AND REGULATION CONFIRMING TESTING REQUIREMENTS OF EXTENDED PRIMARY CONTAINMENT BOUNDARIES. AFTER RESEARCHING SYSTEM MAINTENANCE HISTORY, IT WAS DETERMINED THAT THE PIPING HAD BEEN BREACHED FOR VALVE TESTING DURING THE STATIONS FIRST REFUELING OUTAGE IN 1988. THE RETEST FOLLOWING A RESTORATION OF THE SYSTEM WAS NOT EQUIVALENT TO THAT REQUIRED FOR TYPE B LLRT TESTING. THE ISI SUPERVISOR THEN INFORMED THE NUCLEAR SHIFT SUPERVISOR (NSS, SRO LICENSED) OF THESE FINDINGS. FAILURE TO PERFORM AN APPROPRIATE RETEST CONSTITUTED A MISSED SURVEILLANCE TEST. APPROPRIATE TECH SPEC SECTION ACTION STATEMENTS REGARDING PRIMARY CONTAINMENT INTEGRITY WERE ENTERED. THIS EVENT RESULTED FROM THE SUBJECT VALVES NOT BEING INCLUDED WITHIN THE SCOPE OF THE INSERVICE TEST PROGRAM (IST) MANUAL DUE TO LACK OF SPECIFIC GUIDANCE IN THE UFSAR, AND THUS, THE CORRECT RETEST REQUIREMENTS WERE NOT INCLUDED IN THE WORK ORDER.

[104] HOPE CREEK 1 DOCKET 50-354 LER 89-014
BLOWN SCRAM SOLENOID FUSE RESULTS IN SINGLE ROD SCRAM DUE TO DESIGN DEFICIENCY.
EVENT DATE: 071089 REPORT DATE: 080989 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 214999) ON 7/10/89 AT 0639, HOPE CREEK EXPERIENCED A SINGLE CONTROL ROD (46-31) SCRAM DURING I&C SURVEILLANCE TESTING ON REACTOR PROTECTION SYSTEM (RPS) CHANNEL "B". INVESTIGATION DETERMINED THAT A BLOWN FUSE EXISTED ON THE RPS CHANNEL "A" SIDE OF THE SCRAM SOLENOID AT THE HYDRAULIC CONTROL UNIT (HCU) FOR CONTROL ROD 46-31 PRIOR TO PERFORMING THE SURVEILLANCE. WHEN RPS CHANNEL "B" WAS TRIPPED AS REQUIRED BY THE SURVEILLANCE PROCEDURE, BOTH SIDES OF THE SCRAM SOLENOID WERE THEREFORE DE-ENERGIZED, CAUSING THE ROD TO SCRAM. THE ROOT CAUSE OF THIS OCCURRENCE IS THE LACK OF A MECHANISM TO MONITOR THE STATUS OF SCRAM SOLENOID CONTROL POWER FUSES. IMMEDIATE CORRECTIVE ACTIONS CONSISTED OF REPLACING THE BLOWN SCRAM SOLENOID FUSE, ASSESSING THE IMPACT OF THE SCRAMMED ROD ON CORE THERMAL LIMITS, AND RETURNING THE ROD TO ITS ORIGINAL FULLY WITHDRAWN POSITION. LONG TERM CORRECTIVE ACTIONS INCLUDE INVESTIGATING THE FEASIBILITY OF INSTALLING MONITORING HARDWARE FOR ALL SCRAM SOLENOID FUSES.

[105] HOPE CREEK 1 DOCKET 50-354 LER 89-015
MISSED SURVEILLANCE TEST DUE TO SCHEDULING PROGRAM MALFUNCTION FOLLOWING MAINFRAME COMPUTER FAILURE.
EVENT DATE: 071389 REPORT DATE: 081389 NSSS: GE TYPE: BWR

(NSIC 215025) ON JULY 13, 1989, DURING A PRE-OUTAGE SURVEILLANCE REVIEW, AN I&C DEPARTMENT SUPERVISOR DISCOVERED THAT A MONTHLY FUNCTIONAL TEST OF A HIGH PRESSURE COOLANT INJECTION SYSTEM (HPCI) DIVISION 1 STEAM LEAK DETECTION TEMPERATURE SWITCH HAD NOT BEEN PERFORMED SINCE 3/17/89. THIS DISCREPANCY WAS IMMEDIATELY REPORTED TO THE SENIOR NUCLEAR SHIFT SUPERVISOR, AND TECHNICAL SPECIFICATION ACTION 4.0.3 WAS ENTERED AT 1400. THE SURVEILLANCE WAS IMMEDIATELY PERFORMED WITH SATISFACTORY RESULTS, AND T.S. ACTION STATEMENT 4.0.3 WAS EXITED AT 1500. INVESTIGATION SUBSEQUENT TO THE INCIDENT DETERMINED THE MOST LIKELY CAUSE TO BE PROBLEMS ENCOUNTERED DURING RECOVERY FROM A COMPUTER MALFUNCTION ON THE DAY THE WORKORDER WAS COMPLETED. WHEN THE STATUS OF THE SURVEILLANCE WAS

ENTERED INTO THE COMPUTER FOLLOWING COMPLETION IN MARCH, THE COMPUTER PROGRAM DID NOT SELF-UPDATE DUE TO THE MALFUNCTION, THUS, THE PROGRAM DID NOT SCHEDULE THE SURVEILLANCE FOR THE NEXT TIME IT WAS DUE. CORRECTIVE ACTIONS CONSIST OF DEVELOPING A PROCESS FOR NOTIFYING APPROPRIATE DEPARTMENT MANAGERS IN THE EVENT A SIMILAR COMPUTER MALFUNCTION OCCURS, AND MODIFYING THE COMPUTER PROGRAM TO ALLOW THE COMPUTER SYSTEM ADMINISTRATORS TO BROADCAST MESSAGES TO ALL USERS SHOULD A SIMILAR MALFUNCTION OCCUR.

[106] INDIAN POINT 2 DOCKET 50-247 LER 89-009
ACTUATION OF PRESSURIZER SPRAY DUE TO TEMPORARY PROCEDURE CHANGE.
EVENT DATE: 062689 REPORT DATE: 082189 NSSS: WE TYPE: PWR

(NSIC 215079) ON 6/26/89, DURING COOLDOWN FROM A HYDROTEST OF THE REACTOR COOLANT SYSTEM, THE PRESSURIZER SPRAY LINE VALVE BY-PASS VALVE WAS OPENED. AT THE TIME OF THE VALVE MANIPULATION A TEMPERATURE DIFFERENTIAL OF 478F EXISTED ACROSS THE SPRAY NOZZLE. THE TECH SPECS PROHIBIT ACTUATION OF THE PRESSURIZER SPRAY IF THE SPRAY NOZZLE TEMPERATURE DIFFERENTIAL EXCEEDS 320F. THE CAUSE OF THE EVENT WAS A TEMPORARY PROCEDURE CHANGE TO THE HYDROSTATIC TEST PROCEDURE WHICH PERMITTED OPENING OF THE SPRAY BY-PASS VALVES. THE EFFECT, IF ANY, OF THE THERMAL TRANSIENT IS CURRENTLY UNDER ENGINEERING EVALUATION. THIS EVENT WAS NOT DISCOVERED UNTIL 7/21/89 DURING AN EVALUATION OF PLANT COMPUTER DATA.

[107] INDIAN POINT 2 DOCKET 50-247 LER 89-010
ULTIMATE HEAT SINK TEMPERATURE IN EXCESS OF TECH SPEC LIMITS.
EVENT DATE: 072589 REPORT DATE: 082489 NSSS: WE TYPE: PWR

(NSIC 215111) ON 7/25/89, AT 1850 HOURS, WHILE THE PLANT WAS AT FULL POWER, THE MEASURED HUDSON RIVER WATER TEMPERATURE AT THE PLANT INLET EXCEEDED 85F. THE SERVICE WATER SYSTEM (SWS) WHICH RECEIVES WATER FROM THE HUDSON RIVER, WAS EVALUATED FOR OPERATION WITH A MAXIMUM TEMPERATURE OF 85F. THE COOLING REQUIREMENTS FOR EQUIPMENT SPECIFIED TO BE OPERABLE IN THE TECHNICAL SPECIFICATIONS ARE IN PART BASED ON A MAXIMUM SWS INLET TEMPERATURE OF 85F. ACCORDINGLY, THE PLANT ENTERED TECH SPEC PARAGRAPH 3.0.1 WHICH APPLIES WHEN CIRCUMSTANCES IN EXCESS OF THE SPECIFICATION EXIST. THE PLANT ENTERED TECH SPEC 3.0.1 AGAIN ON JULY 26, 1989 AT 1945 HOURS, WHEN THE SERVICE WATER TEMPERATURE EXCEEDED THE TECH SPEC LIMITATIONS. PRIOR NRC REVIEW AND APPROVAL OF LICENSE AMENDMENT NO. 135, ISSUED ON 8/25/88 FOR ESSENTIALLY THE SAME SITUATION (SERVICE WATER INLET TEMPERATURE EXCEEDING 85F) PROVIDED THE BASIS FOR ISSUANCE OF A TEMPORARY WAIVER OF COMPLIANCE ON 7/27/89. THE WAIVER PERMITTED OPERATION OF INDIAN POINT UNIT 2 AT UP TO 100% POWER WITH A SERVICE WATER INLET TEMPERATURE UP TO 90F. THIS WAS FOLLOWED BY EMERGENCY AMENDMENT NO. 143 TO PERMANENTLY INCORPORATE THE 90F TEMPERATURE LIMIT INTO TECHNICAL SPECIFICATIONS. THE NRC STAFF IS CONTINUING ITS REVIEW OF A 7/13/89 REQUEST AND SUPPORTING ANALYSES WHICH WOULD PERMIT OPERATION OF INDIAN POINT UNIT 2 WITH SERVICE WATER INLET TEMPERATURE OF UP TO 95F.

[108] INDIAN POINT 3 DOCKET 50-286 LER 89-002 REV 01
UPDATE ON MAIN STEAM ISOLATION VALVE FAILS TO CLOSE DUE TO DEFICIENCIES IN THE PREVENTIVE MAINTENANCE REPACK PROCEDURE.
EVENT DATE: 020489 REPORT DATE: 081489 NSSS: WE TYPE: PWR
VENDOR: ATWOOD & MORRILL CO., INC.

(NSIC 215063) ON FEBRUARY 4, 1989 WITH THE REACTOR IN HOT SHUTDOWN IN PREPARATION FOR A STEAM GENERATOR REPLACEMENT OUTAGE, ONE MAIN STEAM ISOLATION VALVE (MSIV) (MS-1-34) FAILED TO CLOSE FROM THE CONTROL ROOM. OPERATORS LOCALLY CLOSED MS-1-34. THE ROOT CAUSE OF THIS EVENT HAS BEEN IDENTIFIED AS DEFICIENCIES IN THE PREVENTATIVE MAINTENANCE REPACK PROCEDURE FOR THE MAIN STEAM ISOLATION VALVES.

ALL FOUR VALVES HAVE BEEN REBUILT WITH PROCEDURES UPGRADED PER MANUFACTURER RECOMMENDATIONS AND SUCCESSFULLY RETESTED.

[109] INDIAN POINT 3 DOCKET 50-286 LER 89-012
CROSSWIRING OF A HOT LEG AND COLD LEG CHANNEL TEST SWITCH DURING A MODIFICATION.
EVENT DATE: 062489 REPORT DATE: 080489 NSSS: WE TYPE: PWR

(NSIC 214983) ON JUNE 24, 1989 AT 1400 HOURS WHILE COMMENCING LOAD ESCALATION FROM TWENTY-FIVE PERCENT REACTOR POWER, CONTROL ROOM OPERATORS OBSERVED 34 LOOP DELTA TEMPERATURE INDICATION READING LOW AND 34 LOOP AVERAGE REACTOR COOLANT TEMPERATURE READING APPROXIMATELY THREE (3) DEGREES HIGHER THAN NORMAL. THE INSTRUMENT AND CONTROL DEPARTMENT INSPECTED THE LOOP 34 INSTRUMENTATION PANEL AND FOUND ONE HOT LEG TEMPERATURE CHANNEL TEST SWITCH AND THE COLD LEG TEMPERATURE CHANNEL TEST SWITCH CROSS-WIRED. THE WIRES WERE RECONNECTED TO THE CORRECT SWITCHES AND THE INSTRUMENTS WERE RETURNED TO NORMAL OPERATION. THE ROOT CAUSES FOR THIS EVENT ARE IDENTIFIED AS SEVERAL PERSONNEL ERRORS. THESE ERRORS INCLUDE INADEQUATE INSTALLATION WORKMANSHIP, INADEQUATELY PERFORMED POST INSTALLATION TESTING AND INADEQUATE QUALITY CONTROL. SINCE THIS EVENT IS ONE OF SEVERAL INVOLVING SIMILAR OCCURRENCES, AN ADDENDUM TO THIS LER WILL BE SUBMITTED BY OCTOBER 31, 1989. THE ADDENDUM WILL DETAIL AN EVALUATION OF THE EQUIPMENT INSTALLATION PROCESS AND ANY RESULTANT CORRECTIVE ACTIONS. THE SPECIFIC CORRECTIVE ACTIONS FOR THIS EVENT INCLUDE REEMPHASIZING TO QUALITY CONTROL AND MAINTENANCE PERSONNEL THE IMPORTANCE OF CORRECTLY CONDUCTING POST MAINTENANCE TESTING AND INDEPENDENT QUALITY CONTROL INSPECTIONS.

[110] KEWAUNEE DOCKET 50-305 LER 89-012
FAILURE TO SATISFY SAFEGUARD CABLE SEPARATION REQUIREMENTS RESULTS IN THE PLANT BEING IN A CONDITION OUTSIDE OF THE DESIGN BASIS.
EVENT DATE: 062289 REPORT DATE: 072489 NSSS: WE TYPE: PWR

(NSIC 214921) ON 6/22/89, AT 1600, WHILE THE PLANT WAS OPERATING AT 10% POWER, A DESIGN DEFICIENCY WAS IDENTIFIED THAT CAUSED BOTH MOTOR DRIVEN AUXILIARY FEEDWATER (MDAFW) PUMPS, ALTHOUGH REMAINING FUNCTIONAL, TO BE DECLARED INOPERABLE. THE LO-LO STEAM GENERATOR SIGNAL FOR BOTH MDAFW PUMPS START CIRCUITS WERE FOUND TO USE CONTROL CABLES THAT WERE ROUTED IN DIFFERENT SECTIONS OF THE SAME CABLE TRAY RACEWAY, THEREFORE NOT MEETING SEPARATION CRITERIA. THEY ALSO WERE DESIGNATED AS NORMAL CABLE AND ROUTED THROUGH NORMAL CABLE TRAYS. THIS CONDITION WAS NOT CONSISTENT WITH THE REQUIREMENTS SPECIFIED IN THE PLANT'S UPDATED SAFETY ANALYSIS REPORT (USAR). PREVIOUSLY, ON 6/14/89, AT 1400, WHILE THE PLANT WAS AT 100% POWER, A DESIGN DEFICIENCY WAS IDENTIFIED IN THE TURBINE DRIVEN AUXILIARY FEEDWATER (TDAFW) PUMP START CIRCUITRY. THE DC MOTOR OPERATED VALVE WHICH ADMITS STEAM TO THE TDAFW PUMP WAS IDENTIFIED AS CONTAINING A NORMAL CABLE IN THE SAFEGUARD CONTROL CIRCUITRY. ON 6/17/89, AT 0000, THE PLANT WAS SHUT DOWN TO REPAIR A REACTOR COOLANT PUMP SEAL. AT THIS TIME, THE CIRCUIT WAS MODIFIED TO USE A SAFEGUARD CABLE AS SCHEDULED. THE ROOT CAUSE OF THE EVENTS HAS BEEN DETERMINED TO BE A DESIGN DEFICIENCY AS THE ORIGINAL CABLES WERE DESIGNATED NORMAL. IF THE CABLES HAD BEEN PROPERLY DESIGNATED AS SAFEGUARD, THEY WOULD HAVE BEEN INSTALLED IN SAFEGUARD TRAYS.

[111] KEWAUNEE DOCKET 50-305 LER 89-013
A BROKEN INPUT SIGNAL WIRE CAUSES A DOWNSCALE FAILURE OF A RADIATION MONITOR RESULTING IN AN ENGINEERED SAFETY FEATURE ACTUATION.
EVENT DATE: 072089 REPORT DATE: 082189 NSSS: WE TYPE: PWR
VENDOR: VICTOREEN INSTRUMENT DIVISION

(NSIC 215067) ON JULY 20, 1989, AT 0917, WHILE THE PLANT WAS OPERATING AT 100% POWER AN ENGINEERED SAFETY FEATURE (ESF) ACTUATION OCCURRED. WORK WAS BEING PERFORMED ON THE FOUR PEN RECORDER INPUT FROM THE AUXILIARY BUILDING VENT MONITOR

A (R-13). THE R-13 DRAWER WAS RETRACTED FOR APPROXIMATELY ONE MINUTE TO VERIFY A POTENTIOMETER SETTING RELATING TO THE RECORDER INPUT. UPON INSERTION OF THE DRAWER, TRAIN A OF THE AUXILIARY BUILDING SPECIAL VENTILATION (ASV) SYSTEM WAS ACTUATED. THE ROOT CAUSE OF THE EVENT WAS DETERMINED TO BE THE FAILURE OF THE INPUT SIGNAL WIRE DUE TO STRESS APPLIED DURING DRAWER REMOVAL AND INSTALLATION. THIS LOSS OF SIGNAL CAUSED A DOWNSCALE FAILURE OF THE R-13 DRAWER. CORRECTIVE ACTIONS INCLUDED A DETERMINATION OF THE WIRE TO THE TERMINAL LUG ALONG WITH REPLACEMENT OF THE TERMINAL LUG. IN ADDITION, A DESIGN CHANGE TO UPGRADE THE RADIATION MONITOR SYSTEM, INCLUDING THE REPLACEMENT OF SEVERAL RADIATION MONITOR DRAWERS, IS IN PROGRESS AND TENTATIVELY SCHEDULED FOR INSTALLATION IN 1992.

[112] LA SALLE 2 DOCKET 50-374 LER 89-008
HPCS DG DAMAGED DUE TO THE UNEXPECTED CLOSURE OF THE SYSTEM AUX. TRANSFORMER.
EVENT DATE: 061489 REPORT DATE: 071489 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 214846) ON 6/14/89 AT APPROX. 1200 HRS, WITH UNIT 2 AT 95.8% POWER IN OPERATIONAL CONDITION 1, THE UNIT 2 SYSTEM AUX. TRANSFORMER (SAT) HAD JUST BEEN RETURNED TO SERVICE FOLLOWING THE REPLACEMENT OF THE PRIMARY "A" PHASE BUSHING. WHILE ATTEMPTING TO PARALLEL THE 2B DG WITH ITS OFFSITE POWER SOURCE (SAT), THE SAT FEED BREAKER (ACB 2432) TO BUS 243 CLOSED UNEXPECTEDLY. PERSONNEL IN THE CONTROL ROOM DID NOT OBSERVE ANY SIGNIFICANT DG DEVIATIONS FROM NORMAL OPERATION. THE 2B DG WAS SUBSEQUENTLY SHUTDOWN AND AN INSPECTION OF THE GENERATOR STATOR WINDINGS REVEALED SOME MINOR DAMAGE HAD OCCURRED. DAMAGE WAS DUE TO THE INADVERTENT OUT OF PHASE CLOSURE OF THE SAT FEED BREAKER. THE AUTOMATIC CLOSURE OF THE SAT FEED BREAKER WAS DUE TO ONE OF ITS SECONDARY STABS BEING BENT CAUSING A SHORT WHICH EFFECTIVELY JUMPERED THE BREAKER CLOSING CIRCUIT PERMISSIVES. DAMAGED STAB RESULTED FROM PROCEDURAL DEFICIENCIES. THE CONSEQUENCES OF THE EVENT WERE MINIMAL WITH RESPECT TO PLANT SAFETY. THE 2B DG AND THE HIGH PRESSURE CORE SPRAY (HPCS) SYSTEM WERE ALREADY INOPERABLE PRIOR TO THIS EVENT (SEE LER 89-007-00). THIS EVENT, HOWEVER, DAMAGED A MAJOR PIECE OF PLANT EQUIPMENT. BECAUSE THE EFFECTS OF THE WINDING DAMAGE COULD NOT BE DETERMINED, THE ENTIRE GENERATOR WAS REPLACED. HPCS SYSTEM AND THE 2B DG WERE DECLARED OPERABLE ON 6/27/89 AT APPROX. 0535 HOURS AFTER BEING INOPERABLE FOR APPROX. 305 HRS FOR THIS EVENT.

[113] LA SALLE 2 DOCKET 50-374 LER 89-010
HIGH PRESSURE CORE SPRAY INOPERABLE DUE TO DIVISION III BATTERY CHARGER OSCILLATIONS.
EVENT DATE: 071589 REPORT DATE: 081489 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LA SALLE 1 (BWR)
VENDOR: FURNAS ELECTRIC CO.

(NSIC 215052) ON 7/15/89 WITH UNIT 2 IN OPERATIONAL CONDITION 1 (RUN) AT 95% POWER, THE NUCLEAR STATION OPERATOR (NSO) NOTICED VOLTAGE OSCILLATIONS (120-131VDC) ON THE UNIT 2 DIVISION III BATTERY CHARGER. THESE OSCILLATIONS WERE NOT CONSTANT, BUT WOULD OCCUR FOR 5-10 SECONDS AT INDETERMINATE INTERVALS. THE BATTERY CHARGER WAS DECLARED INOPERABLE AT 2119 HOURS ON 7/15/89. THIS ALSO RENDERED THE UNIT 2 HIGH PRESSURE CORE SPRAY (HCS) SYSTEM INOPERABLE AND A 14-DAY TIMECLOCK WAS INITIATED IN ACCORDANCE WITH TECH SPECS. SIMILAR OCCURRENCES OF THESE OSCILLATIONS HAVE BEEN OBSERVED SINCE THIS EVENT. BECAUSE OF THE INTERMITTENT NATURE OF THE PROBLEM, THE ROOT CAUSE HAS NOT BEEN DETERMINED. THE CAUSE OF THESE EVENTS APPEARS TO BE RELATED TO THE CHARGER HIGH VOLTAGE SHUTDOWN RELAY (HVSR). THE EXACT FAILURE MODE OF THIS RELAY HAS NOT BEEN DETERMINED. THIS REPORT WILL BE SUPPLEMENED ONCE THE FAILURE MODE IS DETERMINED. THE CONSEQUENCES OF THIS EVENT(S) WAS MINIMAL. DIVISION I AND II EMERGENCY CORE COOLING SYSTEMS (ECCS) WERE FULLY OPERABLE DURING THIS PERIOD. IN ADDITION, THE REACTOR CORE ISOLATION COOLING SYSTEM WAS FULLY OPERABLE AS AN ALTERNATE HIGH PRESSURE INJECTION SYSTEM.

[114] LIMERICK 1 DOCKET 50-352 LER 89-040
THE INOPERABILITY OF THE SEISMIC MONITORING SYSTEM DUE TO A DESIGN DEFICIENCY OF THE TRIAXIAL RESPONSE SPECTRUM ANALYZER PRINTER CONTROLLER CIRCUIT BOARD.
EVENT DATE: 051289 REPORT DATE: 062189 NSSS: GE TYPE: BWR

(NSIC 214464) ON 5/12/89, WITH LIMERICK GENERATING STATION UNIT 1 IN COLD SHUTDOWN, THE SEISMIC MONITORING SYSTEM TRIAXIAL RESPONSE SPECTRUM ANALYZER PRINTER FAILED TO OPERATE DURING PERFORMANCE OF ITS CALIBRATION FUNCTIONAL TEST. TEST PERSONNEL IMMEDIATELY NOTIFIED THE MAIN CONTROL ROOM OPERATORS, AND THE SEISMIC MONITORING SYSTEM WAS DECLARED INOPERABLE IN ACCORDANCE WITH TECHNICAL SPECIFICATION SECTION 3.3.7.2. THERE WERE NO ADVERSE CONSEQUENCES ASSOCIATED WITH THIS EVENT SINCE NO SEISMIC EVENTS OCCURRED DURING THE SECOND UNIT 1 CYCLE OF REACTOR OPERATION. THERE WAS NO RELEASE OF RADIOACTIVE MATERIAL AS A RESULT OF THIS EVENT. THE CAUSE OF THE FAILURE OF THE PRINTER WAS A DESIGN DEFICIENCY OF THE PRINTER CONTROLLER CIRCUIT BOARD. AN UPGRADED CONTROLLER CIRCUIT BOARD AND PRINTER WERE INSTALLED AND SUCCESSFULLY TESTED ON JUNE 20, 1989 AND THE SEISMIC MONITORING SYSTEM WAS THEN DECLARED OPERABLE. SINCE THIS CIRCUIT BOARD IS ONLY USED IN THE TRIAXIAL RESPONSE SPECTRUM ANALYZER, THIS CONDITION CAN NOT OCCUR IN ANY OTHER PLANT EQUIPMENT.

[115] LIMERICK 1 DOCKET 50-352 LER 89-035 REV 01
UPDATE ON CONTROL ROD UNCOUPLING DUE TO UNCOUPLING ROD MISALIGNMENT DURING ORIGINAL INSTALLATION.
EVENT DATE: 052489 REPORT DATE: 081189 NSSS: GE TYPE: BWR

(NSIC 215022) ON MAY 6, 1989, IT WAS IDENTIFIED THAT TWO CONTROL RODS, 22-35 AND 30-15, HAD THEIR UNCOUPLING RODS MISALIGNED. SUBSEQUENT EVALUATION OF THIS CONDITION REVEALED THAT THESE CONTROL RODS HAD BEEN IMPROPERLY VERIFIED AS COUPLED TO THEIR DRIVES DURING THE LIMERICK UNIT 1 SECOND FUEL CYCLE. ON MAY 24, 1989, BECAUSE OF LACK OF COMPLIANCE WITH EITHER TECHNICAL SPECIFICATION (TS) 3.1.3.6 OR ITS ASSOCIATED ACTION STATEMENTS, THIS WAS DETERMINED TO BE A CONDITION PROHIBITED BY TS AND, THEREFORE, REPORTABLE IN ACCORDANCE WITH 10CFR 50.73(A)(2)(I)(B). DURING THE UNIT 1 FIRST REFUELING OUTAGE, THE UNCOUPLING RODS FOR THE 2 CONTROL RODS WERE MISALIGNED DURING INSTALLATION DUE TO PERSONNEL ERROR. A PROCEDURE INADEQUACY AND DESIGN DEFICIENCY ALSO CONTRIBUTED TO THE OCCURRENCE OF THIS EVENT. THE RODS WOULD HAVE SCRAMMED IF REQUIRED AND, IF ONE HAD DROPPED, WOULD HAVE BEEN BOUNDED BY EXISTING ACCIDENT ANALYSES. THE CONTROL ROD DRIVES WERE REPLACED WITH ONES CONTAINING AN IMPROVED UNCOUPLING ROD DURING THE SECOND REFUELING OUTAGE. THE INSTALLATION PROCEDURE WAS ALSO REVISED TO PREVENT RECURRENCE OF THE EVENT.

[116] LIMERICK 1 DOCKET 50-352 LER 89-036
PRESSURE SETPOINT DRIFT OF THE MAIN STEAM SYSTEM SAFETY RELIEF VALVES DUE TO CORROSION INDUCED BONDING WITHIN THE VALVES, WHICH HAS BEEN A REPETITIVE PROBLEM.
EVENT DATE: 052589 REPORT DATE: 062489 NSSS: GE TYPE: BWR
VENDOR: TARGET ROCK CORP.

(NSIC 214411) ON 5/23/89, LIMERICK GENERATING STATION (LGS) PERSONNEL WERE INFORMED THAT PRESSURE SETPOINT TESTING OF THE 14 REACTOR MAIN STEAM SYSTEM TARGET ROCK CORP., MODEL 7567 F, PILOT OPERATED TWO-STAGE SAFETY RELIEF VALVES (SRVS) REVEALED THAT ONLY ONE SRV LIFTED WITHIN THE TECH SPECS (TS) REQUIRED LIMIT OF +/- 1% OF THE NAMEPLATE SETPOINT AS SPECIFIED IN TS SECTION 3.4.2. THE ROOT CAUSE FOR THE SETPOINT DRIFT OF 11 SRVS WAS PRIMARILY CORROSION INDUCED BONDING BETWEEN THE STELLITE PILOT DISC AND STELLITE SEAT. LABYRINTH SEAL INDUCED FRICTION CONTRIBUTED TO THE SETPOINT DRIFT OF 2 SRVS. UNDER THE GUIDANCE OF THE BOILING WATER REACTOR OWNERS GROUP SRV SETPOINT DRIFT FIX PROGRAM, 7 OF THE 14 SRVS WERE REPLACED WITH VALVES HAVING PILOT DISCS MADE OF PH13-8MO (STAINLESS STEEL) DURING THE SECOND UNIT 1 REFUELING OUTAGE. THE SRV'S ORIGINALLY HAD STELLITE PILOT DISCS. THE 7 MODIFIED AND 7 UNMODIFIED SRVS WERE REFURBISHED,

PRESSURE TESTED, AND RECEIVED RECERTIFICATION PRIOR TO BEING INSTALLED. AT PRESENT, SEVERAL PLANTS HAVE REPORTED SETPOINT SURVEILLANCE TEST DATA THAT INDICATE A MARKED SET PRESSURE PERFORMANCE IMPROVEMENT USING THE PILOT DISCS MADE OF PH13-8MO. THERE WERE NO ADVERSE CONSEQUENCES OR RELEASE OF RADIOACTIVE MATERIAL AS A RESULT OF THIS EVENT. LER 87-034 WAS SUBMITTED ON 8/17/87.

[117] LIMERICK 1 DOCKET 50-352 LER 89-037
TWO NSSSS GROUP VI A&B ISOLATIONS AND 'B' STANDBY GAS TREATMENT SYSTEM STARTS DUE TO TRIPPING OF AN UNDERFREQUENCY RELAY TO A UNIT 2 REACTOR PROTECTION SYSTEM.
EVENT DATE: 052589 REPORT DATE: 062389 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LIMERICK 2 (BWR)

(NSIC 214463) ON 5/25-26/89, 2 UNIT 1 NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NSSSS) GROUP VI A&B ISOLATIONS AND INITIATIONS OF THE 'B' STANDBY GAS TREATMENT SYSTEM FAN, AN ENGINEERED SAFETY FEATURE, OCCURRED DUE TO A LOSS OF POWER TO A UNIT 2 REACTOR PROTECTION SYSTEM (RPS) DISTRIBUTION PANEL. THE LOSS OF POWER TO THE UNIT 2 RPS DISTRIBUTION PANEL IN BOTH INSTANCES RESULTED FROM THE SPURIOUS TRIPPING OF AN UNDERFREQUENCY RELAY. AFTER BOTH EVENTS, THE 'B' SGTS FAN WAS SECURED, THE ISOLATION WAS RESET AND THE UNDERFREQUENCY RELAY WAS RESET, RESTORING POWER TO THE RPS DISTRIBUTION PANEL. AFTER THE FIRST EVENT, POWER TO THE RPS DISTRIBUTION PANEL WAS SWITCHED FROM THE NORMAL SOURCE TO THE ALTERNATE SOURCE. AFTER THE SECOND EVENT, THE UNDERFREQUENCY RELAY WAS REPLACED WITH AN IDENTICAL SPARE. THE CAUSE OF THE RELAY TRIPPING WAS INDETERMINATE. WE HAVE CONCLUDED THAT A TRUE UNDERFREQUENCY CONDITION DID NOT EXIST FROM BOTH POWER SUPPLIES YET THE RELAY WAS TESTED AND FOUND TO BE FUNCTIONING PROPERLY. WE HAVE POSTULATED THAT THE RELAY PANEL WAS BUMPED BY A WORKER IN THE AREA AND SIGNS WILL BE POSTED ON THE PANELS CAUTIONING WORKERS NOT TO JAR THE PANELS WHEN WORKING IN THE AREA. THESE EVENTS ARE REPORTABLE IN ACCORDANCE WITH 10CFR 50.73(A)(2)(IV).

[118] LIMERICK 1 DOCKET 50-352 LER 89-043
REFUEL FLOOR HVAC ISOLATIONS DUE TO A PROCEDURAL ERROR AND PERSONNEL ERROR.
EVENT DATE: 062889 REPORT DATE: 072889 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: LIMERICK 2 (BWR)

(NSIC 214926) ON JUNE 28, 1989 AND JUNE 29, 1989 THE REFUEL FLOOR HEATING, VENTILATION, AND AIR CONDITIONING (HVAC) SYSTEM ISOLATED DUE TO INADEQUATE DIFFERENTIAL PRESSURE (DP) BETWEEN THE OUTSIDE ATMOSPHERE AND THE REFUEL FLOOR ATMOSPHERE. THIS ISOLATION GENERATED A SIGNAL THAT ISOLATED REFUEL AREA SECONDARY CONTAINMENT AND INITIATED THE STANDBY GAS TREATMENT SYSTEM, ENGINEERED SAFETY FEATURES (ESF). THE CONSEQUENCES OF THESE EVENTS WERE MINIMAL AND THERE WAS NO RELEASE OF RADIOACTIVE MATERIAL TO THE ENVIRONMENT. THE CAUSES OF THESE EVENTS WERE 1) PROCEDURAL DEFICIENCY THAT DIRECTED THE REFUEL FLOOR HVAC EXHAUST FANS TO BE SHUTDOWN PRIOR TO THE SUPPLY FANS 2) AN UNTESTED SYSTEM CONFIGURATION IN OPERATION, 3) AN INACCURATE MAIN CONTROL ROOM ANNUNCIATOR SETPOINT AND 4) A PERSONNEL ERROR THAT RESULTED IN THE INADVERTENT ACTIVATION OF A REFUEL AREA SUPPLY ISOLATION VALVE LIMIT SWITCH(ES). FOR BOTH EVENTS, FOUR HOUR NOTIFICATIONS WERE MADE IN ACCORDANCE WITH THE REQUIREMENTS OF 10 CFR 50.72 (B)(2)(II) SINCE THE EVENTS RESULTED IN THE AUTOMATIC ACTUATION OF AN ESF. THESE EVENTS ARE BEING REPORTED IN ACCORDANCE WITH THE REQUIREMENTS OF 10 CFR 50.73 (A)(2)(IV) AND 10 CFR 50.73(A)(2)(I)(B).

[119] LIMERICK 1 DOCKET 50-352 LER 89-044
TECH SPEC VIOLATION DUE TO MISSED SURVEILLANCE OF 'C' REACTOR HIGH LEVEL TRIP BECAUSE OF PERSONNEL ERROR.
EVENT DATE: 062889 REPORT DATE: 072789 NSSS: GE TYPE: BWR

(NSIC 214927) ON JUNE 29, 1989 AT 2110 THE SURVEILLANCE TEST COORDINATOR (STC) DETERMINED THAT THE SURVEILLANCE REQUIREMENTS FOR UNIT 1 TECHNICAL SPECIFICATION

(TS) 3.3.9 "FEEDWATER/MAIN TURBINE TRIP ACTUATION SYSTEM INSTRUMENTATION" WERE NOT MET AND THE ASSOCIATED TS ACTION HAD NOT BEEN TAKEN IN THE REQUIRED TIME RESULTING IN A VIOLATION OF TS. THE UNIT 1 'C' REACTOR HIGH LEVEL MONTHLY CHANNEL FUNCTIONAL SURVEILLANCE TEST WAS NOT PERFORMED WITHIN THE REQUIRED INTERVAL. THE CONSEQUENCES OF THIS EVENT WERE MINIMAL AS THE REMAINING CHANNELS WERE SUFFICIENT TO PROVIDE THE TRIP FUNCTION. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR IN THAT SURVEILLANCE TEST PROGRAM PERSONNEL DID NOT RECOGNIZE THAT THE TEST HAD NOT BEEN RESCHEDULED AFTER ITS LAST PERFORMANCE. SURVEILLANCE TEST PERSONNEL WERE COUNSELED. A CROSS CHECK BETWEEN OUT OF SURVEILLANCE AND UNSCHEDULED TESTS WILL BE ROUTINELY PERFORMED TO ENSURE THAT UNSCHEDULED TESTS ARE ADDRESSED BEFORE WHAT WOULD HAVE BEEN THE NEXT SCHEDULED PERFORMANCE DATE.

[120] LIMERICK 2 DOCKET 50-353 LER 89-002
INOPERABILITY OF PLANT SYSTEMS DUE TO UNACCEPTABLE PHYSICAL SEPARATION BETWEEN CLASS 1E AND NON-CLASS 1E CABLES RESULTING FROM PERSONNEL ERROR.
EVENT DATE: 071389 REPORT DATE: 081489 NSSS: GE TYPE: BWR

(NSIC 215027) ON JULY 13, 1989, PLANT STAFF DETERMINED THAT THE UNIT 2 D-22 EMERGENCY DIESEL GENERATOR AND THE 'B' LOOP OF CORE SPRAY WERE INOPERABLE DUE TO AN ELECTRICAL PHYSICAL SEPARATION DEFICIENCY. THIS DEFICIENCY EXISTED BETWEEN A PARTIALLY SLEEVED BUNDLE OF NON-CLASS 1E DROPOUT CABLES EXITING FROM A CABLE GUTTER AND CLASS 1E DROPOUT CABLES EXITING FROM A CABLE TRAY AND ENTERING THE AUXILIARY EQUIPMENT ROOM PANEL 20-C792. THE APPROPRIATE CABLES WERE WRAPPED ON JULY 13, 1989, PROVIDING PHYSICAL SEPARATION BETWEEN THE AFFECTED CABLES. WE HAVE CONCLUDED THAT THIS CONDITION HAS EXISTED SINCE JUNE 22, 1989, THE DATE OF THE ISSUANCE OF THE UNIT 2 FUEL LOAD OPERATING LICENSE. THE CONSEQUENCES OF THIS CONDITION WERE MINIMAL BECAUSE NO FIRE OR FAULT CONDITION OCCURRED. THERE WAS NO RELEASE OF RADIOACTIVE MATERIAL AS A RESULT OF THIS CONDITION. THIS CONDITION RESULTED FROM A PERSONNEL ERROR WHERE CONSTRUCTION AND QUALITY CONTROL PERSONNEL FAILED TO IDENTIFY A PHYSICAL SEPARATION DEFICIENCY DURING THE UNIT 2 FACILITY TURNOVER INSPECTION PROCESS. A QUALITY CONTROL WALKDOWN INSPECTION OF UNIT 2 ELECTRICAL EQUIPMENT WAS COMPLETED AND ONLY THIS ONE MINOR DEFICIENCY WAS IDENTIFIED. THIS DEFICIENCY IS BEING REPORTED IN ACCORDANCE WITH 10 CFR 50.73(A)(2)(I)(B).

[121] MCGUIRE 1 DOCKET 50-369 LER 88-040 REV 01
UPDATE ON CONTAINMENT INTEGRITY BREACHED AND FUEL MOVEMENT SUSPENDED WHEN 3 TEMPORARY PENETRATIONS WERE FOUND LEAKING.
EVENT DATE: 102588 REPORT DATE: 071489 NSSS: WE TYPE: PWR
VENDOR: DOW CORNING CORP.

(NSIC 214969) ON 10/25/88 AT ABOUT 1100, CONSTRUCTION AND MAINTENANCE DEPARTMENT (CMD) PERSONNEL, WORKING ON EQUIPMENT LOCATED INSIDE UNIT 1 CONTAINMENT AND NEAR TEMPORARY REFUELING PENETRATION E461, NOTICED AIR WAS LEAKING THROUGH THE PENETRATION INTO CONTAINMENT BUILDING. AT ABOUT 1315, AFTER FINISHING WORK AND EXITING CONTAINMENT, CMD PERSONNEL NOTIFIED MECHANICAL MAINTENANCE (MNT) TECH SUPPORT PERSONNEL OF THE LEAKING PENETRATION. UNIT 1 FUEL UNLOADING OPERATIONS WERE SUSPENDED. AT 1330, MNT CONTACTED CMD TO RESEAL PENETRATION E461. AT 1528 MNT REPORTED PENETRATION E461 WAS NO LONGER LEAKING AND FUEL UNLOADING WAS RESUMED. ABOUT 30 MIN. LATER, CMD REPORTED PENETRATIONS M260, E429, AND E461 HAD FAILED LEAK TEST, AND FUEL MOVEMENT WAS SUSPENDED. AT 1723, MNT REPORTED TO OPERATIONS (OPS) THAT THE 3 PENETRATIONS WERE RESEALED AND LEAK TESTED SATISFACTORILY. FUEL UNLOADING WAS RESUMED. ON 11/13/88 AT ABOUT 1335, CMD PERFORMED A LEAK TEST OF PENETRATIONS AND NO LEAKAGE WAS NOTED. AT ABOUT 1530, HEALTH PHYSICIST NOTICED AIR LEAKAGE IN A 4" PIPE SLEEVE IN PENETRATION M260. AT 1543, FUEL MOVEMENT WAS SUSPENDED. AT 1820, SHIFT ENGINEER WAS NOTIFIED THAT PENETRATION E429 WAS LEAKING. AT 1850, THE SHIFT SUPERVISOR REQUESTED THAT ALL 3 PENETRATIONS BE LEAK TESTED TO RE-VERIFY CONTAINMENT INTEGRITY. CAUSES INCLUDE

DEFECTIVE PROCEDURE, MANAGEMENT/QUALITY ASSURANCE DEFICIENCY AND INAPPROPRIATE ACTION.

[122] MCGUIRE 1 DOCKET 50-369 LER 89-006 REV 01
UPDATE ON NON-SAFETY RELATED COMPONENTS WERE INSTALLED BETWEEN SAFETY RELATED SOLENOIDS AND VALVE OPERATORS ON A CONTAINMENT ISOLATION VALVE BECAUSE OF A DESIGN DEFICIENCY.
EVENT DATE: 033089 REPORT DATE: 050889 NSSS: WE TYPE: PWR
VENDOR: FISHER FLOW CONTROL DIV (ROCKWELL INT)

(NSIC 215029) ON MARCH 30, 1989, INSTRUMENTATION AND ELECTRICAL PERSONNEL, INSPECTED THE AIR SUPPLY CONFIGURATION TO SAFETY RELATED VALVE 1NV-459A, CHEMICAL AND VOLUME CONTROL (NV) SYSTEM LETDOWN ORIFICE CONTAINMENT ISOLATION. IT WAS DISCOVERED THAT THE NONSAFETY RELATED VALVE POSITIONER WAS PLACED BETWEEN THE SAFETY RELATED SOLENOID AND THE VALVE ACTUATOR . VALVE 1NV-459A WAS BEING INSPECTED TO DETERMINE THE CAUSE OF THE VALVE LEAKING BY THE SEAT. ON APRIL 10, 1989, IAE PERSONNEL COMPLETED WORK REQUEST NO. 137545 WHICH INCLUDED RETUBING THE NONSAFETY RELATED POSITIONER FOR VALVE 1NV-459A SO THAT THE POSITIONER WOULD NOT BE BETWEEN THE SAFETY RELATED SOLENOIDS AND THE VALVE ACTUATOR. UNIT 1 WAS IN MODE 5, COLD SHUTDOWN, AT THE TIME THIS EVENT WAS DISCOVERED. THIS EVENT IS ASSIGNED A CAUSE OF A DESIGN DEFICIENCY BECAUSE THE VALVE POSITIONER WAS INSTALLED AS DESCRIBED BY INCORRECT INSTRUMENT DETAIL DRAWINGS PREPARED BY DESIGN ENGINEERING PERSONNEL DURING THE INITIAL CONSTRUCTION OF UNIT 1.

[123] MCGUIRE 1 DOCKET 50-369 LER 89-013
UNIT OPERATED GREATER THAN 100% THERMAL POWER BECAUSE OF INAPPROPRIATE ACTIONS AND PROCEDURAL DEFICIENCIES.
EVENT DATE: 070589 REPORT DATE: 080489 NSSS: WE TYPE: PWR

(NSIC 215000) ON 6/30/89, FLOW INDICATION FOR UNIT 1 STEAM GENERATOR (S/G) 'C' FAILED HIGH. DURING THE SUBSEQUENT REPAIR OF THIS PROBLEM ON 7/2/89, THE INSTRUMENTATION AND ELECTRICAL (IAE) TECHNICIAN LEFT THE COMPUTER POINTS ASSOCIATED WITH STEAM FLOW FOR S/G 'C' LOCKED AT 58% POWER VALUES. ON 7/5/89, OPERATIONS (OPS) PERSONNEL BEGAN POWER ESCALATION AND AT 0715, WITH INDICATED POWER AT 86%, RECEIVED THE ALARM FOR 3% DEVIATION BETWEEN THE NUCLEAR INSTRUMENTATION (NI) POWER INDICATIONS AND THE OPERATOR AID COMPUTER (OAC) THERMAL POWER BEST ESTIMATE INDICATION. THIS REQUIRES RECALIBRATION OF THE NI INDICATIONS. WHILE PERFORMING THIS CALIBRATION, THE IAE TECHNICIAN CONTACTED A REACTOR UNIT ENGINEER (RE) FOR A DETERMINATION OF THE PROPER CALIBRATION DATA. THE RE INSTRUCTED THE IAE TECHNICIAN TO USE THE OAC THERMAL POWER BEST ESTIMATE FIGURE. THIS FIGURE WAS WRONG BECAUSE OF THE LOCKED COMPUTER INPUTS. THE NI INDICATIONS WERE ADJUSTED IN A NON-CONSERVATIVE DIRECTION BY APPROX. 6%. OPS PERSONNEL RESUMED POWER ESCALATION AND CONTINUED TO 95% (ACTUALLY 101%) INDICATED POWER AT WHICH POINT THEY NOTICED THAT THE SECONDARY OUTPUT APPEARED HIGH AND THE CONDENSATE SYSTEMS APPEARED TO BE UNSTABLE. POWER ESCALATION WAS STOPPED. THIS EVENT IS ASSIGNED A CAUSE OF INAPPROPRIATE ACTION WITH A CONTRIBUTORY CAUSE OF PROCEDURE DEFICIENCY.

[124] MCGUIRE 1 DOCKET 50-369 LER 89-014
THE LAND USE CENSUS TECH SPEC REQUIREMENT WAS NOT COMPLETELY FULFILLED BECAUSE OF POOR INTERFACE BETWEEN GROUPS.
EVENT DATE: 071389 REPORT DATE: 081689 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: MCGUIRE 2 (PWR)

(NSIC 215073) DURING CLARIFICATION OF THE RESPONSIBILITIES FOR GENERAL OFFICE RADIATION PROTECTION (RP) PERSONNEL, STATION RP PERSONNEL, AND THE ENVIRONMENTAL SERVICES PERSONNEL ON THE LAND USE CENSUS (LUC), IT WAS DISCOVERED THAT PART OF TECH SPEC (TS) REQUIREMENT 3.12.2 HAD NEVER BEEN FULFILLED. TS 3.12.2 STATES

THAT THE LUC LOCATIONS ARE REQUIRED TO BE COMPARED WITH THE CURRENT RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM (REMP) SAMPLING LOCATIONS OF THE SAME SAMPLE TYPE AND PATHWAY. IF THE LUC LOCATION CALCULATED DOSE OR DOSE COMMITMENT IS 20% GREATER THAN THE REMP SAMPLING LOCATIONS, THE NEW LUC LOCATION MUST BE IDENTIFIED IN THE NEXT SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (SERR). THE SERR SHALL CONTAIN THE FOLLOWING: (1) NEW LUC LOCATION, (2) CHANGES FOR THE OFFSITE DOSE CALCULATION MANUAL REFLECTING THE NEW LUC LOCATION, AND (3) IF SAMPLES CANNOT BE OBTAINED, AN EXPLANATION OF WHY SAMPLES ARE NOT OBTAINABLE. IF CERTAIN CONDITIONS ARE MET FOR SAMPLING (TS 3.12.1) THEN THE NEW LUC LOCATION SHOULD BE ADDED TO THE REMP. UNIT 1 WAS IN MODE 1, POWER OPERATION, AT 100% POWER AND UNIT 2 WAS IN MODE 6, REFUELING, WHEN THIS EVENT WAS DISCOVERED. THIS EVENT IS ASSIGNED A CAUSE OF MANAGEMENT DEFICIENCY BECAUSE OF POOR INTERFACE BETWEEN GROUPS. THE STATION RP PERSONNEL DID NOT FULLY UNDERSTAND THEIR RESPONSIBILITIES.

[125] MCGUIRE 1 DOCKET 50-369 LER 89-016
TECH SPEC REQUIRED FLOW ESTIMATES OF THE UNIT VENT SAMPLE DEVICE WERE NOT PERFORMED BECAUSE OF A FAILURE TO FOLLOW PROCEDURE.
EVENT DATE: 072089 REPORT DATE: 082189 NSSS: WE TYPE: PWR

(NSIC 215075) ON 7/14/89 AT 1115, OPERATIONS PERSONNEL INFORMED RADIATION PROTECTION (RP) PERSONNEL THAT A LOSS OF FLOW ALARM FOR THE UNIT 1 UNIT VENT RADIATION MONITOR HAD BEEN RECEIVED IN THE CONTROL ROOM. RP PERSONNEL REFERENCED THE PROCEDURE FOR INOPERABLE UNIT VENT SAMPLING REQUIREMENTS AND DETERMINED THAT THE SURVEILLANCE REQUIREMENT REQUIRED BY TECHNICAL SPECIFICATIONS WAS TO OBTAIN A RADIOACTIVE GAS SAMPLE ONCE EVERY 12 HOURS. HOWEVER, RP PERSONNEL A AND B DID NOT PROCEED THROUGH THE PROCEDURE STEPS AND FAILED TO DETERMINE THAT A 4 HOUR ESTIMATION OF FLOW THROUGH THE CONTINUOUS RADIOACTIVE PARTICULATE AND RADIOACTIVE IODINE SAMPLE WAS ALSO REQUIRED. AT 1600 AND 2000, RP PERSONNEL DID NOT PERFORM THE REQUIRED 4 HOUR ESTIMATION OF FLOW THROUGH THE CONTINUOUS SAMPLE DEVICE. THE ESTIMATION OF FLOW THROUGH THE SAMPLE DEVICE WAS PERFORMED ONCE PER 12 HOURS FROM JULY 14, 1989 TO JULY 20, 1989. ON JULY 20TH AT 0800, 1200, AND 1600, RP PERSONNEL PERFORMED THE 4 HOUR ESTIMATE OF FLOW AFTER DISCOVERING THE ERROR. AT 1627, THE UNIT VENT RADIATION MONITOR WAS REPAIRED AND DECLARED OPERABLE. THIS EVENT IS ASSIGNED A CAUSE OF INAPPROPRIATE ACTION BECAUSE RP PERSONNEL A AND B FAILED TO FOLLOW PROCEDURE. RP STAFF PERSONNEL HAVE COVERED THE DETAILS OF THIS EVENT WITH ALL RP SHIFT SUPERVISORS. UNIT 1 WAS IN MODE 1, POWER OPERATION, AT 100 PERCENT POWER AT THE TIME OF THE EVENT.

[126] MCGUIRE 2 DOCKET 50-370 LER 89-005
PRESSURIZER PORV 2NC-32B WOULD NOT HAVE ACTUATED IN THE LOW PRESSURE MODE OF CONTROL BECAUSE OF AN INAPPROPRIATE ACTION AND A CONTRIBUTING MANAGEMENT DEFICIENCY.
EVENT DATE: 070788 REPORT DATE: 072889 NSSS: WE TYPE: PWR
VENDOR: BUSSMANN MFG (DIV OF MCGRAW-EDISON)

(NSIC 215001) DURING PERFORMANCE OF PROCEDURE PT/2/A/4150/14, PRESSURIZER PORV COLD OVERPRESSURE FUNCTIONAL TEST, ON JULY 5, 1989, INSTRUMENTATION AND ELECTRICAL PERSONNEL DISCOVERED THAT TWO WIRES IN THE CIRCUIT FOR VALVE 2NC-32B, PRESSURIZER POWER OPERATED RELIEF VALVE, HAD BEEN REVERSED. INSTRUMENTATION AND ELECTRICAL PERSONNEL CORRECTED THE REVERSED WIRES AND SUCCESSFULLY COMPLETED THE PERIODIC TEST AND FUNCTIONAL VERIFICATION. UNIT 2 WAS IN MODE 3, HOT STANDBY, IN PREPARATION FOR A REFUELING OUTAGE WHEN THE REVERSED WIRES WERE FOUND. THIS EVENT IS ASSIGNED A CAUSE OF INAPPROPRIATE ACTION BECAUSE OF INATTENTION TO DETAIL BY MAINTENANCE ENGINEERING SERVICES PERSONNEL. THE MAINTENANCE ENGINEERING SERVICES PERSONNEL REVERSED THE WIRE NUMBERS DURING DOCUMENTATION OF A CHANGE IN 1988. THIS EVENT IS ALSO ASSIGNED A CONTRIBUTORY CAUSE OF MANAGEMENT DEFICIENCY BECAUSE OF A BREAKDOWN IN THE FUNCTIONAL VERIFICATION PROGRAM.

[127] MCGUIRE 2 DOCKET 50-370 LER 89-006
TECH SPEC 3.0.3 WAS ENTERED TO PERFORM A REACTOR COOLANT SYSTEM THERMAL MIXING TEST.
EVENT DATE: 070589 REPORT DATE: 080489 NSSS: WE TYPE: PWR

(NSIC 214960) ON JULY 5, 1989, AT 1340, INSTRUMENTATION AND ELECTRICAL PERSONNEL DEFEATED THE STEAM GENERATOR HIGH HIGH LEVEL FEEDWATER ISOLATION AND NEGATIVE STEAM LINE RATE MAIN STEAM ISOLATION PROTECTION CIRCUITS ON UNIT 2. OPERATIONS PERSONNEL LOGGED UNIT 2 INTO TECHNICAL SPECIFICATION 3.0.3. AT 1417, OPERATIONS PERSONNEL BEGAN A RAPID, CONTROLLED COOLDOWN OF THE UNIT 2 REACTOR COOLANT SYSTEM THROUGH D STEAM GENERATOR. AT 1424, THE COOLDOWN WAS STOPPED BY OPERATIONS PERSONNEL. AT 1548, INSTRUMENTATION AND ELECTRICAL PERSONNEL RETURNED THE STEAM GENERATOR HIGH HIGH LEVEL FEEDWATER ISOLATION AND NEGATIVE STEAM LINE RATE MAIN STEAM ISOLATION PROTECTION CIRCUITS TO OPERABLE STATUS AND OPERATIONS PERSONNEL LOGGED UNIT 2 FROM TS 3.0.3. UNIT 2 WAS IN MODE 3, HOT STANDBY, AT THE TIME OF THIS EVENT. THIS TEST WAS BEING PERFORMED USING AN APPROVED PROCEDURE.

[128] MILLSTONE 1 DOCKET 50-245 LER 88-005 REV 01
UPDATE ON INSUFFICIENT CONTAINMENT SPRAY INTERLOCK SETPOINT.
EVENT DATE: 052788 REPORT DATE: 072589 NSSS: GE TYPE: BWR

(NSIC 214937) ON 5/27/88, WHILE OPERATING AT 100% POWER (529F, 1032 PSIG) INVESTIGATIONS RESULTING FROM THE POTENTIAL FOULING OF EMERGENCY CORE COOLING SYSTEM (ECCS) SUCTION STRAINERS (LER 88-004-01) AND DESIGN BASIS DOCUMENTATION WORK ON THE LOW PRESSURE COOLANT INJECTION (LPCI) SYSTEM REVEALED THAT THE CONTAINMENT SPRAY 5 PSIG INTERLOCK SETPOINT WAS INSUFFICIENT TO ENSURE THAT ECCS PUMP CAVITATION WOULD NOT OCCUR. THE EVALUATION CONCLUDED THAT WITH ALL SIX ECCS PUMPS OPERATING AND THE FAILURE OF ONE HPCI HEAT EXCHANGER OUTLET VALVE TO OPEN ON THE SERVICE WATER SIDE, THE AVAILABLE NPSH COULD BE INSUFFICIENT TO PREVENT CAVITATION IF THE 5 PSIG INTERLOCK WAS SOLELY RELIED ON TO TERMINATE SPRAYS. THE EMERGENCY OPERATING PROCEDURES HAVE BEEN REVISED TO RESTRICT USE OF CONTAINMENT SPRAY AND TO ENSURE THAT SUFFICIENT PRESSURE IS MAINTAINED TO PRECLUDE ECCS PUMP CAVITATION. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(V). DURING THE 1989 REFUELING OUTAGE THE ECCS SUCTION STRAINERS WERE REPLACED WITH ONES HAVING A LARGER CROSS SECTIONAL AREA. ADDITIONALLY, THE CONTAINMENT SPRAY INTERLOCK PRESSURE SWITCH WAS REPLACED WITH A SWITCH HAVING A WIDER SETPOINT RANGE. THE 5 PSIG SETPOINT REMAINED UNCHANGED. WHEN GENERAL ELECTRIC COMPLETES THE ANALYSIS TO DETERMINE THE REQUIRED PRESSURE TO ENSURE ADEQUATE NPSH AT THE PUMP SUCTIONS (NOW IN PROGRESS) THE SETPOINT WILL BE ADJUSTED, IF NECESSARY.

[129] MILLSTONE 3 DOCKET 50-423 LER 89-015
INOPERABLE EMERGENCY AC DISTRIBUTION - 480 VAC LOAD CENTER CROSS-TIED DUE TO PERSONNEL ERROR.
EVENT DATE: 062289 REPORT DATE: 072489 NSSS: WE TYPE: PWR

(NSIC 214932) ON 6/22/89 AT 0900, WHILE IN MODE 5 (COLD SHUTDOWN) AT 0% POWER, 83 DEGREES ATMOSPHERIC PRESSURE, THE SHIFT SUPERVISOR NOTED THAT THE ELECTRIC PLANT LINE-UP WAS NOT PER THE REQUIREMENTS OF TECH SPEC 3.8.3.2, "ON SITE POWER DISTRIBUTION - SHUTDOWN". ON 6/20/89 AT 1445, THE 'B' EMERGENCY DIESEL GENERATOR WAS REMOVED FROM SERVICE DUE TO A FAILED SURVEILLANCE TEST. THE 'A' EMERGENCY DIESEL GENERATOR AND ITS ASSOCIATED 'A' TRAIN 4160/480 VAC EMERGENCY BUSSES WERE REQUIRED TO BE IN SERVICE TO SATISFY TECH SPEC REQUIREMENTS. LOAD CENTER 32S WAS BEING SUPPLIED POWER VIA THE 32R CROSS-TIE BREAKER. FULL LOAD CAPACITY WAS NOT AVAILABLE FOR THE 32S EMERGENCY BUS. THE ROOT CAUSE OF THIS INCIDENT WAS THE OPERATING SHIFT DID NOT REVIEW THE PLANT ELECTRICAL LINE-UP ADEQUATELY ENOUGH TO ENSURE THAT ALL TECH SPEC REQUIREMENTS WERE MET. ALSO THE OPERATING SHIFT DID NOT HAVE ADEQUATE PROCEDURAL GUIDANCE TO ENSURE THAT THE TECH SPEC REQUIREMENTS WERE MET WHILE IN MODES 5 AND 6. ON 6/22/89, AT 0900, THE PLANT ENTERED THE ASSOCIATED LIMITING CONDITION FOR OPERATION (LCO) AND ACTION REQUIREMENTS WERE

MET. THE LCO WAS EXITED AT 1159, WHEN THE 'B' EMERGENCY DIESEL GENERATOR AND ITS ASSOCIATED EMERGENCY BUSSES WERE RETURNED TO SERVICE. OPERATING PROCEDURES ASSOCIATED WITH THE ELECTRIC PLANT LINE-UP, REQUIRED DURING MODE 5 AND MODE 6, WILL BE REVISED AS NECESSARY.

[130] MILLSTONE 3 DOCKET 50-423 LER 89-016
MISSED FIRE DETECTION TECH SPEC SURVEILLANCE DUE TO PERSONNEL ERROR.
EVENT DATE: 070389 REPORT DATE: 080289 NSSS: WE TYPE: PWR

(NSIC 215004) ON 7/3/89 AT APPROXIMATELY 1200 HOURS AT 0% POWER IN MODE 5 (COLD SHUTDOWN) AT 81F AND ATMOSPHERIC PRESSURE, OPERATIONS DEPARTMENT SUPERVISORY PERSONNEL DISCOVERED THAT THE REQUIRED FIRE DETECTION SURVEILLANCE FOR THE CONTROL BUILDING HEATING, VENTILATION AND AIR CONDITIONING (HVAC), CHILLER AND COMPUTER ROOMS HAD NOT BEEN COMPLETED WITHIN ITS REQUIRED FREQUENCY. ROOT CAUSE OF THE EVENT WAS PERSONNEL ERROR. THE OPERATIONS DEPARTMENT SHIFT SUPERVISORS DID NOT INVESTIGATE TO DETERMINE THE REQUIRED COMPLETION DATE OF THE SURVEILLANCE TO ENSURE CONTINUED OPERABILITY OF THE FIRE DETECTION SYSTEM FOR THE ABOVE MENTIONED AREAS. AS A CONTRIBUTORY CAUSE OF THE EVENT, THE REQUIRED COMPLETION DATES WERE NOT READILY ACCESSIBLE TO THE SHIFT SUPERVISORS. IMMEDIATE CORRECTIVE ACTION WAS TO ESTABLISH AN HOURLY FIRE WATCH PATROL FOR THE AFFECTED AREAS. THE SURVEILLANCE WAS SUCCESSFULLY COMPLETED ON 7/24/89. THE SHIFT SUPERVISOR TURNOVER LOG AND THE PERSONAL COMPUTER SURVEILLANCE TRACKING SYSTEM HAVE BEEN UPDATED TO INCLUDE REQUIRED COMPLETION DATES FOR SURVEILLANCES.

[131] MILLSTONE 3 DOCKET 50-423 LER 89-017
NONCOMPLIANCE WITH ACTION STATEMENT DUE TO INADEQUATE ADMINISTRATIVE GUIDANCE.
EVENT DATE: 071889 REPORT DATE: 081789 NSSS: WE TYPE: PWR

(NSIC 215048) ON 7/18/89, AT 0902 HOURS, WHILE OPERATING IN MODE 1 AT 97% POWER, 2250 PSIA AND 585F, THE SHIFT SUPERVISOR (SS) DETERMINED THAT A MOTOR-OPERATED ISOLATION VALVE FOR THE CONTAINMENT RECIRCULATION SPRAY HEADER HAD BEEN INOPERABLE FOR 24 HOURS. ON JULY 17, AT 0618 HOURS, A LICENSED OPERATOR (CO) WHILE PERFORMING THE QUARTERLY VALVE STROKE SURVEILLANCE TEST, PLACED THE HAND SWITCH IN THE CLOSE POSITION. DUAL POSITION INDICATION WAS OBSERVED AT THE CONTROL BOARD. THE CO AND THE SS REVIEWED THE PLANT TECH SPECS, AND DETERMINED THAT THE VALVE WAS OPERABLE IN THE OPEN (ACCIDENT) POSITION. THE NEW SHIFT OF OPERATORS ON THE DAY SHIFT OF JULY 18, MADE THE DETERMINATION THAT THE VALVE WAS INOPERABLE AS A CONTAINMENT ISOLATION VALVE. THE REQUIRED TECH SPEC ACTION HAD NOT BEEN PERFORMED. THE ROOT CAUSE OF THE EVENT WAS INADEQUATE ADMINISTRATIVE GUIDANCE ON THE DEFINITION OF CONTAINMENT ISOLATION VALVES. THIS LED TO A MISINTERPRETATION OF THE TECH SPECS AND THE FSAR BY THE OPERATORS AND THE OPERATIONS DEPARTMENT MANAGEMENT. THE IMMEDIATE ACTION WAS TO TEST THE VALVE TO ENSURE THAT IT WAS FULLY SHUT, FULFILLING ITS CONTAINMENT ISOLATION FUNCTION. THE VALVE LIMIT SWITCHES WERE ADJUSTED, RETESTED AND THE VALVE DECLARED OPERABLE. INTERIM GUIDANCE ON CONTAINMENT ISOLATION VALVES WERE PROVIDED. FINAL GUIDANCE WILL BE INCLUDED IN PERMANENT PLANT PROCEDURES BY 2/28/90.

[132] MONTICELLO DOCKET 50-263 LER 89-011
EXCESSIVE CHECK VALVE LEAKAGE CONSTITUTES POTENTIAL DEGRADATION OF HIGH PRESSURE COOLANT INJECTION SYSTEM.
EVENT DATE: 062289 REPORT DATE: 072189 NSSS: GE TYPE: BWR
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 214874) DURING COLD SHUTDOWN A HYDROSTATIC TEST OF FEEDWATER (FW) PIPING REVEALED EXCESSIVE LEAKAGE PAST REACTOR WATER CLEANUP (RWCU) RETURN CHECK VALVE RC 6.1. RWCU RETURN FLOW IS ROUTED TO THE "B" FEEDWATER LINE VIA THE HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM INJECTION LINE. IN THE EVENT OF A BREACH IN THE NON-SAFETY RELATED, NON-SEISMICALLY DESIGNED PORTION OF RWCU PIPING

UPSTREAM OF THE LEAKING CHECK VALVE RC 6.1, THE REACTOR VESSEL INJECTION FLOW CAPABILITY OF HPCI WOULD BE REDUCED BY THE AMOUNT OF THE LEAKAGE. CHECK VALVE RC 6.1 WILL BE REPLACED DURING THE UPCOMING REFUELING OUTAGE UNDER AN EXISTING, PRESCHEDULED MODIFICATION. AT THAT TIME THE VALVE WILL BE DISASSEMBLED AND INSPECTED TO DETERMINE THE CAUSE OF THE LEAKAGE. A SUPPLEMENTAL REPORT WILL BE SUBMITTED THAT DISCUSSES THE RESULTS OF THE INVESTIGATION, THE CAUSES FOUND, AND PLANNED CORRECTIVE ACTIONS. FOR SUBSEQUENT PLANT OPERATION, THE LINE HAS BEEN ISOLATED WITH ALL RWCU RETURN FLOW GOING TO "A" FW LINE. THE CHECK VALVE MAINTENANCE PROGRAM IS BEING REVIEWED IN LIGHT OF THIS EVENT AND NECESSARY IMPROVEMENTS WILL BE MADE.

[133] MONTICELLO DOCKET 50-263 LER 89-013
FIRE BARRIER PENETRATION INOPERABILITY AS A RESULT OF FAILURE TO ADEQUATELY ASSESS ORIGINAL SEAL CONFIGURATION.
EVENT DATE: 063089 REPORT DATE: 073089 NSSS: GE TYPE: BWR

(NSIC 214945) SEVERAL UNSEALED PENETRATIONS WERE DISCOVERED IN BARRIERS SEPARATING DIVISION I AND DIVISION II FIRE AREAS. THIS IS A CONDITION WHICH IS CONTRARY TO THE REQUIREMENTS OF TECHNICAL SPECIFICATION 3.13.G. TEMPORARY SEALS WERE INSTALLED IMMEDIATELY. DUE TO COGNITIVE PERSONNEL ERROR. INSPECTION BY CONTRACT PERSONNEL OF APPENDIX R FIRE AREA BOUNDARY PENETRATIONS FAILED TO IDENTIFY SEVERAL INADEQUATE SEALS. INTERFERENCE FROM PIPE INSULATION JACKETING AND EXISTING PLANT EQUIPMENT PREVENTED DIRECT VISUAL EXAMINATION OF THE PENETRATIONS RESULTING IN ACCEPTANCE OF AN INADEQUATE SEAL. A WALKDOWN OF ALL FIRE AREA BARRIERS TO IDENTIFY ANY DEFICIENT PENETRATION SEALS WAS INITIATED. ALL ACCESSIBLE FIRE AREA BARRIERS HAVE BEEN INSPECTED. INACCESSIBLE AREAS WILL BE INSPECTED DURING THE UPCOMING OUTAGE. THIS INVESTIGATION IS ONGIONG AND A SUPPLEMENTAL REPORT WILL BE SUBMITTED DETAILING THE RESULTS OF THIS WALKDOWN. THE EXPECTED COMPLETION DATE OF THE WALKDOWNS IS AUGUST 25, 1989.

[134] NINE MILE POINT 1 DOCKET 50-220 LER 89-010
DESIGN DEFICIENCY RESULTING IN POSSIBLE FAILURE OF SAFETY SYSTEM TO PERFORM ITS INTENDED FUNCTION.
EVENT DATE: 063089 REPORT DATE: 073189 NSSS: GE TYPE: BWR

(NSIC 214867) ON JUNE 30, 1989, AT 0959 DURING SURVEILLANCE TESTING OF THE SERVICE WATER EFFLUENT RADIATION MONITOR IT WAS NOTED THAT CERTAIN EQUIPMENT FAILURES WOULD NOT RESULT IN DOWNSCALE INDICATION AND, THEREFORE, ANNUNCIATION IN THE CONTROL ROOM. THE ROOT CAUSE HAS BEEN DETERMINED TO BE AN INADEQUATE DESIGN REVIEW OF VENDOR SUPPLIED INFORMATION WHEN THE SERVICE WATER MONITOR WAS REPLACED IN 1986. CORRECTIVE ACTIONS TAKEN FOR THIS EVENT INCLUDE A COMPENSATORY SURVEILLANCE TO VERIFY OPERABILITY AND INITIATION OF MODIFICATION TO PROVIDE ANNUNCIATION CONSISTENT WITH THE TECHNICAL SPECIFICATION INTENT. THE SCHEDULE FOR COMPLETION OF THE MODIFICATION WILL BE PROVIDED IN A SUPPLEMENT TO THE LER WHEN THE SCOPE OF THE MODIFICATION IS DETERMINED. ADDITIONALLY, RADIATION MONITORS WHOSE TECHNICAL SPECIFICATION CHANNEL FUNCTIONAL TEST REQUIRES SPECIFIC ANNUNCIATION WILL BE REVIEWED FOR ADEQUACY OF ANNUNCIATING AN INOPERABLE CONDITION BY SEPTEMBER 30, 1989. SUBSEQUENT TO THIS MODIFICATION, PROCEDURES HAVE BEEN ESTABLISHED TO ENSURE PROPER REVIEWS OF VENDOR INFORMATION ARE PERFORMED.

[135] NINE MILE POINT 2 DOCKET 50-410 LER 89-002 REV 01
UPDATE ON AN UNANALYZED CONDITION DUE TO DESIGN AND INSTALLATION DEFICIENCIES DUE TO HUMAN PERFORMANCE DEFICIENCIES.
EVENT DATE: 012589 REPORT DATE: 062289 NSSS: GE TYPE: BWR

(NSIC 214473) ON 1/25/89 AT 1503 HRS, THE NINE MILE POINT UNIT 2 (NMP2) OPERATORS WERE NOTIFIED OF A DEFICIENCY THAT PLACED THE STATION IN AN UNANALYZED CONDITION. SPECIFICALLY, AN UNIDENTIFIED ELECTRICAL BLOCKOUT (PENETRATION) IN A WALL

(DESIGNATED AS A PRESSURE BOUNDARY) WAS FOUND NOT TO HAVE BEEN SEALED. IT WAS POSTULATED THAT STRUCTURAL DAMAGE TO THE CONTROL BUILDING COULD OCCUR DURING AN INTERNAL FLOODING EVENT AS A RESULT OF THIS UNSEALED PENETRATION. SUBSEQUENT TO THIS DISCOVERY, AN EVALUATION WAS PERFORMED WHICH IDENTIFIED SEVERAL OTHER DEFICIENT WATERTIGHT SEALS. DURING THIS TIME, THE REACTOR MODE SWITCH WAS IN SHUTDOWN AND THE REACTOR WAS IN A COLD SHUTDOWN CONDITION (OPERATIONAL CONDITION 4). THE IMMEDIATE CAUSE FOR THIS EVENT WAS A COMBINATION OF INSTALLATION AND DESIGN DEFICIENCIES. THE ROOT CAUSES FOR THIS EVENT WERE A COMBINATION OF TWO HUMAN PERFORMANCE DEFICIENCIES: (1) POOR WORK PRACTICES, AND (2) INADEQUATE CHANGE MANAGEMENT. THE CORRECTIVE ACTIONS TAKEN ARE AS FOLLOWS: (1) ENGINEERING DESIGN CHANGE DOCUMENTS WERE ISSUED TO CORRECT THE IDENTIFIED DESIGN DEFICIENCIES, (2) WORK REQUESTS WERE ISSUED TO CORRECT THE INSTALLATION DEFICIENCIES; (3) AN EVALUATION OF OTHER PENETRATION SEALS IN PRESSURE BOUNDARIES WAS PERFORMED; (4) A PROBLEM REPORT HAS BEEN INITIATED.

[136] NINE MILE POINT 2 DOCKET 50-410 LER 89-019
TECH SPEC VIOLATION AS A RESULT OF A FAILURE TO ADMINISTRATIVELY CONTROL AN INOPERABLE COMPONENT DUE TO A PERSONNEL ERROR.
EVENT DATE: 060489 REPORT DATE: 071389 NSSS: GE TYPE: BWR

(NSIC 214647) ON 6/13/89, AT 2147 HOURS, IT WAS DISCOVERED THAT AN AREA UNIT COOLER WHICH IS REQUIRED FOR THE OPERABILITY OF THE DIVISION II SERVICE WATER SYSTEM HAD BEEN RETURNED TO SERVICE WITH A DEFECT WHICH WOULD HAVE PREVENTED IT FROM PERFORMING ITS SAFETY FUNCTION. AS A RESULT NINE MILE POINT UNIT NO. 2 (NMP2) WAS IN VIOLATION OF TECHNICAL SPECIFICATION (T.S.) 3.7.1.1 "SERVICE WATER SYSTEM OPERATING". AT THE TIME OF THE DISCOVERY NMP2 WAS IN OPERATIONAL MODE 1 "RUN" AT 100 PERCENT RATED POWER. THE ROOT CAUSE OF THE EVENT WAS DETERMINED TO BE PERSONNEL ERROR ON THE PART OF NMP2 CONTROL ROOM SUPERVISION WHO DID NOT PROPERLY TAG AN INOPERABLE PIECE OF EQUIPMENT IN ACCORDANCE WITH SITE ADMINISTRATIVE PROCEDURES. IMMEDIATE CORRECTIVE ACTION WAS TO VERIFY THE OPERABILITY OF THE REDUNDANT COMPONENT AND TO PLACE A TAG-OUT ON THE INOPERABLE COMPONENT. LONG TERM CORRECTIVE ACTION INCLUDES REVISION TO THE ADMINISTRATIVE CONTROLS FOR THE EQUIPMENT STATUS LOG TO MORE CLEARLY DEFINE OUTSTANDING ENTRIES AND REQUIRING CONTROL ROOM SUPERVISION TO INITIAL ALL TAG-OUTS PRIOR TO THEIR BEING REMOVED.

[137] NINE MILE POINT 2 DOCKET 50-410 LER 89-020
PUMP VIBRATION ACCEPTANCE CRITERIA ABOVE ALLOWABLE LIMITS DUE TO PROCEDURE DEFICIENCY.
EVENT DATE: 061989 REPORT DATE: 071789 NSSS: GE TYPE: BWR

(NSIC 214828) ABSTRACT ON JUNE 19, 1989, OPERATIONS PERSONNEL WERE NOTIFIED THAT OPERATIONS SURVEILLANCE PROCEDURE N2-OSP-EGF-Q001 DID NOT CONTAIN THE CORRECT PUMP VIBRATION LIMITING ACCEPTANCE CRITERIA AS REQUIRED BY THE NINE MILE POINT UNIT 2 (NMP2) INSERVICE TEST (IST) PROGRAM. TWO PUMPS IN EACH OF DIVISION I, DIVISION II, AND DIVISION III EMERGENCY DIESEL GENERATOR FUEL OIL TRANSFER SYSTEM WERE AFFECTED BY THIS PROCEDURE DEFICIENCY. NMP2 WAS OPERATING AT 100 PERCENT POWER AT THE TIME OF THE DISCOVERY OF THIS EVENT. THE ROOT CAUSE OF THE EVENT HAS BEEN DETERMINED TO BE A PROCEDURAL DEFICIENCY IN IMPLEMENTING THE IST PROGRAM WHICH RESULTED IN ERRONEOUS ACCEPTANCE CRITERIA BEING INCORPORATED INTO AN OPERATIONS SURVEILLANCE PROCEDURE. INITIAL CORRECTIVE ACTION INCLUDED A REVIEW OF THE LATEST PUMP VIBRATION TEST RESULTS. THE CORRECT ACCEPTANCE CRITERIA WAS APPLIED AND ALL SIX PUMPS WERE DETERMINED TO BE OPERABLE WITH FOUR OF THE SIX PUMPS IN THE ALERT (INCREASED FREQUENCY TESTING) RANGE. ADDITIONAL CORRECTIVE ACTION INCLUDED ISSUING A REVISION TO OPERATIONS SURVEILLANCE PROCEDURE N2-OSP-EGF-Q001 TO INCORPORATE CORRECTED ACCEPTANCE CRITERIA. A PROBLEM REPORT (PR) HAD BEEN ISSUED IDENTIFYING PREVIOUS PUMP VIBRATION TEST DATA AND REQUESTED ENGINEERING TO APPROVE NEW REFERENCE VALUES AND ACCEPTANCE CRITERIA.

[138] NORTH ANNA 1 DOCKET 50-338 LER 88-027 REV 01
UPDATE ON INCONSISTENCY BETWEEN THE TECH SPECS AND THE UFSAR ON ESF SLAVE RELAY TESTING.
EVENT DATE: 111788 REPORT DATE: 072689 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: NORTH ANNA 2 (PWR)

(NSIC 214862) THIS VOLUNTARY LICENSEE EVENT REPORT (LER) IS BEING SUBMITTED AS A RESULT OF DISCOVERING AN INCONSISTENCY BETWEEN THE UPDATED FINAL SAFETY ANALYSIS REVIEW (UFSAR) AND THE TECHNICAL SPECIFICATIONS FOR THE SURVEILLANCE REQUIREMENTS OF THE SOLID STATE PROTECTION SYSTEM (SSPS) SLAVE RELAYS. AT 1609 HOURS ON NOVEMBER 17, 1988, ENGINEERING PERSONNEL DISCOVERED THAT THE SURVEILLANCE REQUIREMENTS IN SECTION 3.3.2.1 OF THE TECHNICAL SPECIFICATIONS DO NOT SPECIFICALLY REFER TO TESTING OF THE ENGINEERING SAFETY FEATURES (ESF) FINAL ACTUATED DEVICES THROUGH THE ESF SLAVE RELAYS DURING ON-LINE OPERATION AS STATED IN SECTION 7.3.2.1.5 OF THE UFSAR. IMMEDIATELY AFTER DISCOVERING THE INCONSISTENCY A JUSTIFICATION FOR CONTINUED OPERATION WAS WRITTEN TO JUSTIFY CONTINUED OPERATION WITHOUT FUNCTIONALLY TESTING THE ESF SLAVE RELAYS DURING ON-LINE OPERATION. AN ENGINEERING REVIEW OF THE ESF TESTING PROGRAM HAS BEEN COMPLETED AND REVISED TECHNICAL SPECIFICATIONS HAVE BEEN SUBMITTED TO THE NRC. THE SAFETY CONSEQUENCES OF THIS EVENT ARE MINIMAL BASED ON THE FACT THAT EQUIPMENT WHICH IS REQUIRED TO PERFORM ESF FUNCTIONS IS VERIFIED TO BE OPERABLE DURING PLANT OPERATION THROUGH SUCCESSFUL COMPLETION OF OTHER REQUIRED TECHNICAL SPECIFICATION SURVEILLANCES. ALSO, REDUNDANCY IS PROVIDED FOR EACH ESF FUNCTION. THE HEALTH AND SAFETY OF THE GENERAL PUBLIC HAS NOT BEEN AFFECTED AS A RESULT OF THIS. EVENT.

[139] NORTH ANNA 1 DOCKET 50-338 LER 89-008 REV 01
UPDATE ON SERVICE WATER FLOW TO REACTOR HEAT EXCHANGER LESS THAN DESIGN.
EVENT DATE: 041489 REPORT DATE: 062389 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: NORTH ANNA 2 (PWR)

(NSIC 214459) AT 1835 HOURS ON 4/14/89, WITH UNIT 1 IN MODE 5 (COLD SHUTDOWN), IT WAS DETERMINE THAT THERE WAS LESS THAN DESIGN SERVICE WATER SYSTEM FLOW AVAILABLE TO THE RECIRCULATION SPRAY HEAT EXCHANGERS (RSHXS). THE UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR) ASSUMES 18,000 GPM SERVICE WATER FLOW TO THE RSHXS FOR CONTAINMENT HEAT REMOVAL CAPABILITY. THE PERFORMANCE OF 1-PT-75.6 INDICATED THAT THERE WAS 15,990 GPM TOTAL FLOW TO THE RSHXS. SERVICE WATER FLOW TO THE RSHXS LESS THAN THE DESIGN ASSUMPTIONS IN THE UFSAR IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(II). 1-SW-MOV-103A, B, C AND D WERE MANUALLY REPOSITIONED TO ALLOW A BALANCED FLOW TO EACH RSHX AND A TOTAL SERVICE WATER FLOW TO THE RSHXS OF GREATER THAN OR EQUAL TO 18,000 GPM. A SIMILAR FLOW TEST (2-PT-75.6) WAS PERFORMED ON UNIT 2. A TOTAL SERVICE WATER FLOW OF GREATER THAN 18,000 GPM WAS ACHIEVED. THE UNIT 2 SERVICE WATER SYSTEM INLET MOTOR OPERATED VALVES TO THE RSHXS WERE ALSO MANUALLY REPOSITIONED TO ALLOW A BALANCED FLOW TO EACH RSHX. NO SIGNIFICANT SAFETY CONSEQUENCES RESULTED FROM THIS EVENT BECAUSE THE DESIGN BASIS WAS MET FOR MINIMUM SERVICE WATER FLOW, HIGHEST FOULING FACTOR AND THE WORST CASE KEY PARAMETERS ACTUALLY EXPERIENCED AT NORTH ANNA. THE HEALTH AND SAFETY OF THE GENERAL PUBLIC WERE NOT AFFECTED BY THIS EVENT.

[140] NORTH ANNA 1 DOCKET 50-338 LER 89-012
INADVERTENT START OF 1J EDG AND 2-SW-P-1A DURING SSPS TESTING.
EVENT DATE: 062689 REPORT DATE: 072089 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: NORTH ANNA 2 (PWR)

(NSIC 214838) AT 2100 HOURS ON 6/26/89, WITH UNIT 1 IN MODE 5 (COLD SHUTDOWN), THE 1J EMERGENCY DIESEL GENERATOR (EDG) WAS INADVERTENTLY STARTED DURING THE PERFORMANCE OF REACTOR PROTECTION AND ENGINEERED SAFETY FEATURES RESPONSE TIME TESTING - SLAVE RELAYS. AT 0940 HOURS ON 7/7/89, WITH UNIT 2 AT 100% POWER (MODE 1), THE UNIT 2 'A' SERVICE WATER PUMP (2-SW-P-1A) WAS ALSO INADVERTENTLY STARTED

DURING THE PERFORMANCE OF UNIT 1 REACTOR PROTECTION AND ENGINEERED SAFETY FEATURES (ESF) RESPONSE TIME TESTING - SLAVE RELAYS. SINCE THE 1J EDG AND 2-SW-P-1A ARE PART OF A ENGINEERED SAFETY FEATURES SYSTEM AND STARTING THESE COMPONENTS WAS NOT PREPLANNED, THESE EVENTS ARE REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(IV). A FOUR HOUR REPORT WAS MADE FOR EACH EVENT IN ACCORDANCE WITH 10CFR50.72(B)(2)(II). THE INADVERTENT START OF THE 1J EDG WAS CAUSED BY PERSONNEL ERROR. THE INADVERTENT START OF 2-SW-P-1A WAS CAUSED BY PROCEDURAL ERROR. AS AN IMMEDIATE CORRECTIVE ACTION, THE 1J EDG AND 2-SW-P-1A WERE SECURED AND RETURNED TO NORMAL STANDBY STATUS. TO PREVENT RECURRENCE OF SIMILAR EVENTS, PERSONNEL WERE COUNSELLED AND THE UNIT 1 AND UNIT 2 REACTOR PROTECTION AND ENGINEERED SAFETY FEATURES RESPONSE TIME TESTING - SLAVE RELAYS PERIODIC TESTS WILL BE REVISED PRIOR TO THE NEXT SCHEDULED REFUELING OUTAGE.

[141] NORTH ANNA 1 DOCKET 50-338 LER 89-013
INCORE FLUX MAPPING FRAME ASSEMBLY DISCOVERED TO BE UNRESTRAINED.
EVENT DATE: 062689 REPORT DATE: 072189 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: NORTH ANNA 2 (PWR)
VENDOR: TELEFLEX, INC.

(NSIC 214923) AT 1100 HOURS ON 6/26/89, WITH UNIT 1 IN MODE 5 (COLD SHUTDOWN), STATION PERSONNEL PERFORMING A WALKDOWN IN THE INCORE INSTRUMENTATION DRIVE ROOM DISCOVERED THAT THE INCORE FLUX MAPPING FRAME ASSEMBLY WAS UNRESTRAINED. CONSEQUENTLY, THERE IS A POTENTIAL THAT DURING A SEISMIC EVENT THE UNRESTRAINED FRAME COULD DAMAGE THE INCORE FLUX MAPPING TUBES LOCATED DIRECTLY ABOVE THE SEAL TABLE. BREAKAGE OF THESE TUBES COULD RESULT IN AN UNANALYZED SMALL BREAK LOSS OF COOLANT ACCIDENT. FURTHER INVESTIGATION REVEALED THAT THE UNIT 2 INCORE FLUX MAPPING FRAME ASSEMBLY WAS ALSO NOT RESTRAINED. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(II)(B). THE CAUSE OF THIS EVENT WAS INADEQUATE INSTALLATION. THE INCORE FLUX MAPPING FRAME ASSEMBLY WAS NOT BOLTED TO THE CONCRETE FLOOR DURING INITIAL INSTALLATION AS STATED IN THE VENDOR TECHNICAL MANUAL. AS A CORRECTIVE ACTION, AN ENGINEERING WORK REQUEST AND SUPPORTING 10 CFR 50.59 SAFETY EVALUATION WERE WRITTEN AND APPROVED BY THE STATION NUCLEAR SAFETY AND OPERATING COMMITTEE TO SEISMICALLY RESTRAIN THE MOVABLE FRAME ASSEMBLY. THE UNIT 1 AND UNIT 2 INCORE FLUX MAPPING FRAME ASSEMBLIES WERE SEISMICALLY MOUNTED ON 7/7/89 AND 7/6/89, RESPECTIVELY. NO SIGNIFICANT SAFETY CONSEQUENCES RESULTED FROM THIS EVENT BECAUSE THE PRESSURE TUBES ABOVE THE SEAL TABLE WERE NOT DAMAGED AND THE INCORE INSTRUMENTATION SYSTEM REMAINED OPERABLE.

[142] NORTH ANNA 1 DOCKET 50-338 LER 89-014
REACTOR TRIP DUE TO A LOSS OF EHC SYSTEM PRESSURE.
EVENT DATE: 071989 REPORT DATE: 081089 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 214996) AT 1740 HOURS, ON 7/19/89, UNIT 1 EXPERIENCED AN AUTOMATIC REACTOR TRIP FROM 90% POWER. THE REACTOR TRIP SIGNAL OCCURRED DUE TO A LOSS OF ELECTRO HYDRAULIC CONTROL (EHC) SYSTEM PRESSURE WHICH GENERATED A TURBINE TRIP AND A SUBSEQUENT REACTOR TRIP SINCE POWER WAS GREATER THAN 10%. THIS EVENT IS REPORTABLE PURSUANT TO 10CFR50.73(A)(2)(IV). A 4 HOUR REPORT WAS MADE IN ACCORDANCE WITH 10CFR50.72(B)(2)(II). THE LOSS OF EHC SYSTEM PRESSURE WAS DUE TO A FAILED O-RING ON THE TURBINE TRIP SOLENOID OPERATED VALVE (SOV) 20-ET. THE LOSS OF EHC SYSTEM PRESSURE RESULTED IN THE CLOSURE OF THE TURBINE STOP VALVES WHICH GENERATED THE TURBINE TRIP SIGNAL. A REACTOR TRIP SIGNAL WAS AUTOMATICALLY INITIATED, AS DESIGNED, SINCE POWER WAS GREATER THAN 10%. NO SAFETY INJECTION SIGNAL (MANUAL OR AUTOMATIC) WAS INITIATED AS REQUIRED DURING THIS EVENT. UNIT 1 WAS PLACED ON LINE AT 2136 HOURS ON 7/20/89. A ROOT CAUSE EVALUATION IS BEING PERFORMED TO DETERMINE THE ROOT CAUSE OF THE TURBINE TRIP SOLENOID OPERATED VALVE 20-ET O-RING FAILURE. RECOMMENDATIONS RESULTING FROM THE ROOT CAUSE EVALUATION ON THE O-RING FAILURE WILL BE EVALUATED AND IMPLEMENTED AS NECESSARY. THIS EVENT POSED NO SIGNIFICANT SAFETY IMPLICATIONS BECAUSE SAFETY RELATED EQUIPMENT

FUNCTIONED AS DESIGNED AND KEY REACTOR PARAMETERS STABILIZED FOLLOWING THE REACTOR TRIP.

[143] OCONEE 1 DOCKET 50-269 LER 89-009 REV 01
UPDATE ON A MANAGEMENT DEFICIENCY RESULTED IN AN INCORRECT TECH SPEC WHICH ALLOWED A SINGLE BREAKER FAILURE TO PREVENT THE EMERGENCY POWER SWITCHING LOGIC FROM FUNCTIONING.
EVENT DATE: 060789 REPORT DATE: 080189 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: OCONEE 2 (PWR)
OCONEE 3 (PWR)

(NSIC 214946) ON JUNE 7, 1989, AT 1400 HOURS, WITH UNITS 1 AND 3 OPERATING AT 100% FULL POWER AND UNIT 2 IN A REFUELING OUTAGE, IT WAS DISCOVERED THAT TECHNICAL SPECIFICATION 3.7.1.B.1 ALLOWED UNRESTRICTED PLANT OPERATION IN A CONFIGURATION WHICH COULD ALLOW A SINGLE FAILURE OF ONE STANDBY BREAKER TO PREVENT THE EMERGENCY POWER SWITCHING LOGIC (EPSL) SYSTEM FROM PERFORMING ITS FUNCTION UNDER CERTAIN ACCIDENT SCENARIOS. THIS DISCOVERY WAS MADE DURING DESIGN ENGINEERING REVIEWS FOR THE DESIGN BASIS DOCUMENTATION ANALYSIS. THE ROOT CAUSE OF THIS EVENT WAS MANAGEMENT DEFICIENCY WHICH RESULTED IN AN INADEQUATE TECHNICAL SPECIFICATION. THE IMMEDIATE CORRECTIVE ACTIONS CONSISTED OF MAINTAINING BOTH STANDBY BUSES OPERABLE. IN THE EVENT THAT ONE OF THE STANDBY BUSES BECAME INOPERABLE, THEN TECHNICAL SPECIFICATION 3.7.2.A WOULD APPLY AND ALL OCONEE UNITS WOULD ENTER A 72 HOUR LIMITING CONDITION OF OPERATION. SUBSEQUENT INVESTIGATION FOUND PROCEDURAL DEFICIENCIES WHICH WOULD PREVENT EPSL FROM PERFORMING CERTAIN FUNCTIONS. THE ROOT CAUSES OF THIS WAS DEFECTIVE PROCEDURE. PROCEDURE CHANGES WERE MADE TO CORRECT THIS PROBLEM.

[144] OCONEE 1 DOCKET 50-269 LER 89-010 REV 02
UPDATE ON CENTRAL SWITCHYARD WAS USED AS AN UNACCEPTABLE OFFSITE POWER SOURCE AS A RESULT OF A MANAGEMENT DEFICIENCY.
EVENT DATE: 060889 REPORT DATE: 071089 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: OCONEE 2 (PWR)
OCONEE 3 (PWR)

(NSIC 215116) ON JUNE 8, 1989, WITH UNIT 1 AND 3 AT 100% FULL POWER AND UNIT 2 IN A REFUELING OUTAGE, IT WAS DETERMINED THAT THE CENTRAL SWITCHYARD COULD NOT BE USED AS AN OFFSITE POWER SOURCE BECAUSE IT DOES NOT MEET THE NUCLEAR REGULATORY COMMISSION'S REQUIREMENTS FOR DEGRADED GRID PROTECTION. THIS CONDITION WAS DISCOVERED BY DESIGN ENGINEERING DURING A FOLLOW-UP ITEM FOR THE LEE GAS TURBINE REVIEW IN WHICH THE CENTRAL SWITCHYARD WAS ANALYZED AS AN OFFSITE POWER SOURCE. THE ROOT CAUSE OF THIS INCIDENT WAS DETERMINED TO THE A MANAGEMENT DEFICIENCY, DEFICIENT COMMUNICATION. IMMEDIATE CORRECTIVE ACTIONS INCLUDED ISSUING A TRAINING PACKAGE TO ALL OPERATIONS PERSONNEL AND REVISING OP/0/A/1107/03 (100 KV POWER SUPPLY) PROCEDURE ELIMINATING CENTRAL SWITCHYARD AS A POWER SOURCE TO THE STANDBY BUSES.

[145] OCONEE 1 DOCKET 50-269 LER 89-011 REV 01
UPDATE ON TECH SPEC 3.7 WAS VIOLATED AS A RESULT OF A DEFECTIVE PROCEDURE.
EVENT DATE: 061889 REPORT DATE: 072889 NSSS: BW TYPE: PWR
OTHER UNITS INVOLVED: OCONEE 2 (PWR)
OCONEE 3 (PWR)

(NSIC 214947) ON JUNE 18, 1989, AT 1300 HOURS, WITH UNIT 1 AND 3 AT 100% FULL POWER AND UNIT 2 IN A REFUELING OUTAGE, BOTH INDEPENDENT EMERGENCY ON-SITE POWER PATHS FROM KEOWEE WERE UNINTENTIONALLY REMOVED FROM SERVICE FOR APPROXIMATELY 20 MINUTES WHILE PERFORMING PT/2/A/0610/01J, "EPSL ES ACTUATION KEOWEE EMERGENCY START TEST". THIS WAS A VIOLATION OF TECHNICAL SPECIFICATION 3.7. THIS CONDITION WAS DISCOVERED BY A SHIFT SUPERVISOR QUESTIONING ACTION STEPS IN THE

PROCEDURE THAT APPEARED TO BE IN ERROR. THE ROOT CAUSE OF THIS EVENT WAS DETERMINED TO BE A DEFECTIVE PROCEDURE DUE TO ERRONEOUS INFORMATION. CORRECTIVE ACTIONS INCLUDED CORRECTING THE DEFECTIVE PROCEDURE AND COMPLETING THE TEST.

[146] OYSTER CREEK DOCKET 50-219 LER 89-010 REV 01
UPDATE ON DESIGN DEFICIENCY CAUSES NON-COMPLIANCE WITH 10CFR50 APPENDIX R CRITERIA.
EVENT DATE: 031689 REPORT DATE: 080989 NSSS: GE TYPE: BWR

(NSIC 215055) TWO CONCERNS HAVE BEEN IDENTIFIED WITH RESPECT TO THE ISOLATION CONDENSER SYSTEM AVAILABILITY DURING A FIRE CONDITION. THIS NON-COMPLIANCE WITH 10CFR50 APPENDIX R IS CONSIDERED REPORTABLE IN ACCORDANCE WITH 10CFR50.73(A)(2)(II)(B). AS A RESULT OF AN UNRELATED TASK, AN ENGINEER IDENTIFIED A POSTULATED FAILURE MECHANISM FOR CABLING ASSOCIATED WITH THE ISOLATION CONDENSER VALVES. WHILE REVIEWING THE CIRCUITRY FOR THE ABOVE CONCERN, MORE CONCERNS WERE IDENTIFIED REGARDING ISOLATION CONDENSER (IC) SYSTEM AVAILABILITY. A FIRE CONDITION IN ANY 1 OF 4 AREAS CAN CAUSE DAMAGE TO THE HIGH FLOW SENSORS OR ASSOCIATED LOGIC CIRCUITS CAUSING SPURIOUS CLOSURE OF ALL THE IC VALVES. THE POWER AND CONTROL CABLES TO THE AC POWERED VALVES WERE NOT PROTECTED SINCE IT WAS DETERMINED THEY DID NOT HAVE TO OPERATE TO ACHIEVE SAFE SHUTDOWN. THE SUBSEQUENT FAILURE OF POWER OR CONTROL CABLES FOR THE AC POWERED VALVES AFTER THEIR SPURIOUS CLOSURE PREVENTS REOPENING OF THE VALVES. THE CAUSE OF THIS OCCURRENCE HAS BEEN DETERMINED TO BE A DESIGN DEFICIENCY - FAILURE TO IDENTIFY SPURIOUS CLOSURE OF THE AC POWERED VALVES DUE TO HIGH FLOW SENSOR LOGIC DAMAGE. CORRECTIVE ACTIONS INCLUDE A MODIFICATION WHICH ADDED A COORDINATED FUSE ARRANGEMENT TO PROTECT THE CONTROL CIRCUITRY FOR THE AFFECTED VALVE AND HOURLY FIRE WATCHES WHILE THE PLANT IS OPERATED UNTIL FINAL RESOLUTION IS DETERMINED.

[147] OYSTER CREEK DOCKET 50-219 LER 89-016
MAIN TRANSFORMER FAILURE CAUSES AUTOMATIC REACTOR SHUTDOWN.
EVENT DATE: 062589 REPORT DATE: 072589 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 214866) ON JUNE 25, 1989 AT 0008 HOURS, THE MAIN GENERATOR TRIPPED DUE TO A PHASE DIFFERENTIAL CONDITION CAUSED BY A FAULT IN ONE OF THE MAIN OUTPUT TRANSFORMERS. WHEN THE GENERATOR TRIPPED A TURBINE TRIP SIGNAL WAS GENERATED WHICH RESULTED IN AN ANTICIPATORY REACTOR SCRAM. THE PLANT WAS COOLED DOWN WITH THE MAIN CONDENSERS AND THE SHUTDOWN COOLING SYSTEM AND REACHED A COLD SHUTDOWN CONDITION AT 1100 HOURS. THE CAUSE OF THIS EVENT WAS EQUIPMENT FAILURE. EXAMINATION OF THE TRANSFORMER DETERMINED THAT AN INTERNAL WINDING HAD FAILED THEREBY CAUSING THE PHASE DIFFERENTIAL CONDITION WHICH CAUSED THE GENERATOR TRIP. THIS TRANSIENT WAS WITHIN THE DESIGN BASIS OF THE PLANT AND HAD NO SAFETY SIGNIFICANCE. PREPARATIONS HAVE BEEN MADE TO REMOVE THE FAILED TRANSFORMER AND INSTALL A SPARE. UNTIL THE SPARE IS INSTALLED THE PLANT WILL BE OPERATED AT HALF LOAD. NO OTHER CORRECTIVE ACTION WAS DETERMINED NECESSARY.

[148] OYSTER CREEK DOCKET 50-219 LER 89-017
MAIN TRANSFORMER FAILURES CAUSES AUTOMATIC REACTOR SHUTDOWN.
EVENT DATE: 071189 REPORT DATE: 080389 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 214975) ON JULY 11, 1989 AT 0055 HOURS, THE MAIN GENERATOR TRIPPED DUE TO A PHASE DIFFERENTIAL CONDITION CAUSED BY A FAULT IN THE OPERATING MAIN OUTPUT TRANSFORMER (THE OTHER MAIN TRANSFORMER FAILED ON JUNE 25, 1989). WHEN THE GENERATOR TRIPPED, A TURBINE TRIP SIGNAL WAS GENERATED WHICH RESULTED IN AN ANTICIPATORY REACTOR SCRAM. THE PLANT WAS COOLED DOWN UTILIZING THE MAIN CONDENSER AND THE SHUTDOWN COOLING SYSTEM AND REACHED THE COLD SHUTDOWN CONDITION AT 0950 HOURS. THE CAUSE OF THIS EVENT WAS EQUIPMENT FAILURE. EXAMINATION OF

THE TRANSFORMER DETERMINED THAT AN INTERNAL WINDING HAD FAILED, CAUSING THE PHASE DIFFERENTIAL CONDITION WHICH CAUSED THE GENERATOR TRIP. THE PLANT RESPONDED AS DESIGNED AND OPERATOR ACTION WAS PROMPT AND APPROPRIATE. THIS TRANSIENT WAS WITHIN THE DESIGN BASIS OF THE PLANT AND HAD NO SAFETY SIGNIFICANCE. THE ENTIRE ELECTRICAL SYSTEM WAS EVALUATED FOR ADVERSE EFFECTS DUE TO THE FAULT, AND AN EVALUATION WAS COMPLETED TO DETERMINE IF A GENERIC PROBLEM EXISTED BETWEEN THE TWO FAILED TRANSFORMERS AND THE OTHER POWER TRANSFORMERS AT THE STATION. AS A RESULT, IT WAS DETERMINED THE FAILURE WAS CONFINED TO THE TWO MAIN OUTPUT TRANSFORMERS AND DID NOT INVOLVE ANY OTHER TRANSFORMERS. THE PLANT WAS RETURNED TO POWER OPERATION ON JULY 19, 1989 USING THE SPARE TRANSFORMER INSTALLED DUE TO THE PREVIOUS FAILURE OF THE OTHER MAIN OUTPUT TRANSFORMER.

[149] OYSTER CREEK DOCKET 50-219 LER 89-018
ELECTROMATIC RELIEF VALVE HIGH PRESSURE FUNCTION INOPERABLE DUE TO LOOSE WIRES.
EVENT DATE: 072989 REPORT DATE: 082589 NSSS: GE TYPE: BWR

(NSIC 215107) ON 7 29/89 AT APPROXIMATELY 2215 HOURS, DURING A ROUTINE SURVEILLANCE, A LOOSE WIRE WAS FOUND IN THE HIGH PRESSURE ACTUATION CIRCUIT FOR THE "D" ELECTROMATIC RELIEF VALVE (EMRV). THE SENIOR TECHNICIAN PERFORMING THE SURVEILLANCE VERIFIED THAT THE WIRE WAS NOT PROPERLY LANDED, DETERMINED THE PROPER CONNECTION POINT FOR THE WIRE USING PLANT DRAWINGS, AND RELANDED THE WIRE. ALTHOUGH THE WIRE WAS MAKING SOME CONTACT, ADEQUATE CONTINUITY FOR VALVE OEPRATION COULD NOT BE VERIFIED. IT WAS CONSERVATIVELY DETERMINED THAT THE "B" EMRV COULD NOT HAVE BEEN RELIED UPON TO OPEN DURING A HIGH PRESSURE CONDITION AND THAT THE VALVE HAD BEEN INOPERABLE DUE TO THE LOOSE WIRE. THIS EVENT WAS POSSIBLY CAUSED THE LAST TIME THE SURVEILLANCE WAS PERFORMED ON 6/28/89 SINCE THE "D" EMRV ACTUATED NORMALLY ON HIGH PRESSURE ON 6/25/89 FOLLOWING A REACTOR SCRAM. THE SAFETY SIGNIFICANCE OF THIS EVENT IS CONSIDERED TO BE MINIMAL BECAUSE THE "D" EMRV ACTUATED NORMALLY ON JUNE 25 AND NO WORK OTHER THAN THE SURVEILLANCE ON JUNE 28 WAS PERFORMED ON THE EMRV'S BETWEEN JUNE 25 AND THE DATE OF THIS EVENT. ADDITIONALLY, ONLY THE HIGH PRESSURE RELIEF FUNCTION FOR THE "D" EMRV WAS INHIBITED. THE ADS AND MANUAL FUNCTIONS FOR THE VALVE WERE UNAFFECTED AND THE VALVE WOULD HAVE OPERATED AS DESIGNED IF EITHER OF THESE FUNCTIONS HAD BEEN REQUIRED.

[150] PALISADES DOCKET 50-255 LER 84-021 REV 02
UPDATE ON FAILED PRIMARY COOLANT PUMP P-50C.
EVENT DATE: 091684 REPORT DATE: 082489 NSSS: CE TYPE: PWR
VENDOR: BYRON JACKSON PUMPS, INC.

(NSIC 215092) ON SEPTEMBER 15, 1984, WITH THE PLANT AT 60 PERCENT POWER PRIMARY COOLANT PUMP P-50C WAS SHUTDOWN DUE TO FAILURE OF ALL SEALS EXCEPT THE VAPOR SEAL. SUBSEQUENT PUMP DISASSEMBLY ON OCTOBER 1, 1984, DETERMINED THAT THE BOLTS AND PINS COUPLING THE PUMP SHAFT TO THE IMPELLER HAD FAILED AS WELL AS SIGNIFICANT WEARING OF PUMP INTERNAL STATIONERY AND ROTATING COMPONENTS THEREBY CAUSING THE SEAL FAILURE. THE PARTS WERE SENT TO BATTELLE FOR EXTENSIVE EXAMINATION. THE ROOT CAUSE WAS FAILURE TO TORQUE THE IMPELLER BOLTS PROPERLY AND THEREFORE NOT HAVING THE CORRECT PRELOAD CAUSED THE SUBSEQUENT FAILURE OF THE BOLTS. THE PROCEDURES FOR ASSEMBLY OF THE PUMP WERE UPGRADED TO PREVENT RECURRENCE OF THIS PROBLEM. THE PLANT WAS AT 60% POWER PRIOR TO THE SEAL FAILURE.

[151] PALISADES DOCKET 50-255 LER 89-010
BREAKER FAILURE RESULTS IN TWO COINCIDENTALLY INOPERABLE SAFETY INJECTION COMPONENTS.
EVENT DATE: 060289 REPORT DATE: 062989 NSSS: CE TYPE: PWR
VENDOR: ALLIS CHALMERS

(NSIC 214443) AT 1845 ON JUNE 2, 1989 OPERATIONS PERSONNEL IDENTIFIED THAT HIGH

PRESSURE SAFETY INJECTION (HPSI) PUMP P-66A (BQ;P) WOULD NOT START ON DEMAND FROM THE CONTROL ROOM. P-66A WAS TO BE UTILIZED TO REFILL SAFETY INJECTION TANK T-82A (BP;TK) AFTER A DRAINING AND BORON SAMPLING EVOLUTION. AT THE TIME P-66A FAILED TO START, T-82A WAS INOPERABLE PER LEVEL REQUIREMENTS SPECIFIED IN PLANT TECHNICAL SPECIFICATION (TS) 3.3.1.A. WITH BOTH P-66A AND T-82A INOPERABLE, THE LIMITING CONDITIONS OF OPERATION FOR TS 3.3.2 WERE EXCEEDED AND THE PLANT ENTERED TS 3.0.3. THE REACTOR WAS CRITICAL WITH THE PLANT OPERATING AT 80 PERCENT OF RATED POWER WHEN THE EVENT OCCURRED. T-82A LEVEL WAS RETURNED TO ITS NORMAL OPERATING LEVEL BY USE OF HPSI PUMP P-66B. WHEN ITS LEVEL WAS RESTORED, THE PLANT EXITED TS 3.0.3. THE FAILURE OF P-66A TO START WAS DUE TO A LOOSE PLUNGER ASSEMBLY BOLT ON ITS ASSOCIATED SWITCHGEAR BREAKER 152-207 (EB;52). THE PLUNGER ASSEMBLY BOLT WAS LOOSE DUE TO A FAILED LOCK WASHER. THIS ITEM IS BELIEVED TO BE AN ISOLATED OCCURRENCE BY BOTH CONSUMERS POWER AND THE VENDOR.

[152] PALISADES DOCKET 50-255 LER 89-015
POTENTIAL BREAKER FAILURE RESULTS IN OPERATION OUTSIDE DESIGN BASIS.
EVENT DATE: 071889 REPORT DATE: 081789 NSSS: CE TYPE: PWR

(NSIC 215058) AT 1230 ON JULY 18, 1989, WITH THE REACTOR CRITICAL AND THE PLANT OPERATING AT 80 PERCENT OF RATED POWER, AN INVESTIGATION DISCOVERED A POTENTIAL FAILURE THAT WOULD PLACE THE PLANT OUTSIDE THE CURRANT MAIN STEAM LINE BREAK (MSLB) CONTAINMENT ANALYSIS. A FAILURE OF 2400 VOLT BREAKER 152-105 (EB;SWGR) (STATION POWER TO SAFEGUARDS BUS 1C) TO OPEN ON A FAST TRANSFER SIGNAL FOLLOWING A UNIT TRIP WOULD PREVENT THE AUTOMATIC REENERGIZATION OF BUS IC (EB;BU). THIS WOULD CAUSE A FAILURE OF TWO OF THREE CONTAINMENT SPRAY PUMPS. FOR A MSLB INSIDE CONTAINMENT, THIS RESULTS IN A CALCULATED PEAK CONTAINMENT PRESSURE GREATER THAN THE DESIGN LIMIT AND A CALCULATED PEAK CONTAINMENT TEMPERATURE GREATER THAN THE CURRENT EEQ PROFILE. AFTER REVIEW OF THE EVENT, SAFEGUARDS BUS 1C WAS TRANSFERRED TO THE STARTUP POWER SUPPLY AT 1254 ON JULY 18, 1989. THIS ELIMINATES THE NEED FOR A FAST TRANSFER OF THIS BUS. AN ONGOING OFFSITE POWER MODIFICATION WILL PERMANENTLY ELIMINATE THE NEED FOR A FAST TRANSFER BY POWERING THE SAFEGUARDS BUSES DIRECTLY FROM THE SWITCHYARD. THE POTENTIAL EFFECT ON THE MSLB CONTAINMENT ANALYSES OF A FAILURE OF BREAKER 152-105 TO OPEN WAS NOT RECOGNIZED PREVIOUSLY BECAUSE IT WAS ASSUMED THAT REDUNDANCY AND SEPARATION IN THE CONTAINMENT HEAT REMOVAL SYSTEMS, WITH THE AVAILABILITY OF OFFSITE POWER, WOULD NEGATE THE EFFECTS OF INDIVIDUAL ELECTRICAL COMPONENT FAILURES.

[153] PALISADES DOCKET 50-255 LER 89-016
UNTESTED PIPING ASSOCIATED WITH THE CONTAINMENT BOUNDARY.
EVENT DATE: 071989 REPORT DATE: 081889 NSSS: CE TYPE: PWR
VENDOR: R-P & C VALVE

(NSIC 215059) AT APPROXIMATELY 1000 HOURS ON JULY 19, 1989 TECH SPEC (TS) SURVEILLANCE TEST MI-5, WAS INITIATED TO TEST PRESSURE SWITCHES (JM;PS) ASSOCIATED WITH THE CONTAINMENT BUILDING HIGH PRESSURE INITIATION LOGIC. DURING THE CONDUCT OF THE TEST, AT APPROXIMATELY 1000 HOURS, A LEAK WAS IDENTIFIED ON MV-1805 (JM;RTV) BY THE PLANT INSTRUMENTATION AND CONTROL (I&C) TECHNICIAN PERFORMING THE TEST. AT THIS TIME THE TEST WAS STOPPED. SUBSEQUENTLY, IT WAS DETERMINED THAT SMALL SECTIONS OF PIPING ASSOCIATED WITH CONTAINMENT PENETRATIONS 17 AND 48 HAD NOT BEEN TESTED DURING EITHER LLRTS OR PAST ILRTS. AT 1500 THE PLANT REVIEW COMMITTEE DIRECTED THAT TESTS BE DEVELOPED AND IMMEDIATELY PERFORMED TO VERIFY CONTAINMENT INTEGRITY. AT 1045 ON JULY 20, 1989 AFTER IDENTIFYING THAT TESTING WOULD NOT BE COMPLETED WITHIN THE 24 HOUR TIME IMPOSED UNDER THE INTENT OF GENERIC LETTER 89-07, THE NRC RESIDENT INSPECTOR WAS NOTIFIED OF OUR INTENT THAT ENFORCEMENT DISCRETION BE GRANTED TO COMPLETE THE TESTING AND AT APPROXIMATELY 1445 IT WAS GRANTED BY THE NRC REGION III ADMINISTRATOR. THE TOTAL LEAKAGE RATE FOR ALL EIGHT PENETRATIONS WAS APPROXIMAT ELY 1,000 CC/MIN. THE FAILURE TO SUBSEQUENTLY IDENTIFY THE UNTESTED PIPING UNDER THE CURRENT LLRT OR

ILRT PROGRAM HAS BEEN ATTRIBUTED TO A FAILURE TO IDENTIFY THE LACK OF TESTING OVERLAP BETWEEN THE ILRT AND LLRTS PERFORMED.

[154] PALISADES DOCKET 50-255 LER 89-017
INADEQUATE PROCEDURE RESULTS IN TESTING OF CONTAINMENT AIR COOLERS DURING PROHIBITED CONDITIONS.
EVENT DATE: 072189 REPORT DATE: 082189 NSSS: CE TYPE: PWR

(NSIC 215112) ON JULY 21, 1989, DURING PREPARATION FOR PERFORMANCE OF TECHNICAL SPECIFICATION (TS) SURVEILLANCE TEST QO-1, "SAFETY INJECTION SYSTEM", IT WAS DISCOVERED A PORTION OF THE TEST PROCEDURE REQUIRED THREE OF THE FOUR NORMALLY OPERATING CONTAINMENT AIR COOLER (CAC) "A" FANS [BK;FAN] BE SECURED. THESE FANS ARE DESIGNED TO COOL CONTAINMENT AFTER A DESIGN BASIS ACCIDENT (DBA). WITH THREE OF THE FANS SECURED, TS 3.4.2 AND 3..43 WERE VIOLATED. THESE TS ALLOW NO MORE THAN TWO CACS TO BE OUT OF SERVICE CONTINGENT ON VERIFICATION OF OPERATION OF OTHER EQUIPMENT. WITH THREE OF THE CAC "A" FANS SECURED, THE ABILITY OF ONE TRAIN OF SAFETY EQUIPMENT TO PROVIDE ADEQUATE CONTAINMENT COOLING IN AN EMERGENCY COULD POTENTIALLY HAVE BEEN ADVERSELY AFFECTED. THE OTHER CHANNEL OF CONTAINMENT COOLING EQUIPMENT, COMPRISED OF TWO CONTAINMENT SPRAY PUMPS, WOULD STILL HAVE BEEN AVAILABLE TO PERFORM THIS SAFETY FUNCTION. THE REACTOR WAS CRITICAL WITH THE PLANT OPERATING AT 80 PERCENT OF RATED POWER WHEN THIS CONDITION WAS IDENTIFIED. PERFORMANCE OF QO-1 WAS IMMEDIATELY SUSPENDED WHEN THE TEST INADEQUACY WAS DISCOVERED. AT THE TIME OF DISCOVERY, NO EQUIPMENT HAD YET BEEN TAKEN OUT OF SERVICE FOR THE TEST. THE TEST PROCEDURE WAS REVISED TO PERMIT ONLY ONE FAN TO BE STOPPED AND STARTED AT A TIME. THIS REVISED TEST METHOD FULLY SATISFIES TS REQUIREMENTS REGARDING CAC FAN OPERATION.

[155] PALISADES DOCKET 50-255 LER 89-018
COINCIDENT EQUIPMENT INOPERABILITY RESULTS IN OPERATIONAL CONDITION PROHIBITED BY TECH SPECS.
EVENT DATE: 072589 REPORT DATE: 082489 NSSS: CE TYPE: PWR
VENDOR: AMOT CONTROL CORP.
BAILEY CONTROLS CO.

(NSIC 215113) AT 1400 HRS ON 7/25/89 TECH SPEC (TS) 3.0.3 WAS ENTERED DUE TO THE COINCIDENT INOPERABILITY OF AUX. FEEDWATER FLOW INDICATING CONTROLLER FIC-0727 (BA;TC) AND EMERGENCY DIESEL GENERATOR 1-2 (EK;DG). FIC-0727 IS ASSOCIATED WITH EMERGENCY DIESEL GENERATOR 1-1. TS 3.7.2.I ALLOWS ONE OF THE DIESEL GENERATORS TO BE INOPERABLE PROVIDED THE OTHER IS TEST STARTED, CONTROLS ARE LEFT IN THE AUTOMATIC MODE AND THERE ARE NO INOPERABLE ESF FEATURE COMPONENTS ASSOCIATED WITH THE OPERABLE DIESEL GENERATOR. FIC-0727 WAS RETURNED TO SERVICE AND TS 3.0.3 EXITED AT 1427 HRS. THE REACTOR WAS CRITICAL WITH THE PLANT OPERATING AT 80% OF RATED POWER WHEN THIS CONDITION WAS IDENTIFIED. FIC-0727 HAD BEEN DECLARED INOPERABLE TO AFFECT PLANNED REPAIRS OF DRIFT PROBLEMS BEING EXPERIENCED WITH ITS ASSOCIATED FLOW CONTROL VALVE. DIESEL GENERATOR 1-2 WAS BEING RUN FOR IN-PROGRESS WORK ON ITS AIR INTAKE MANIFOLD PRESSURE GAUGE PULSATION DAMPENER. WHILE BEING RUN, A LOW LUBE OIL PRESSURE INDICATION WAS IDENTIFIED. THE DIESEL GENERATOR WAS SECURED AND DECLARED INOPERABLE AT 1333. AFTER CONSULTATION WITH PLANT LICENSING PERSONNEL, TS 3.0.3 WAS ENTERED AT 1400. FIC-0727 WAS RETURNED TO SERVICE AT 1427 AND TS 3.0.3 EXITED. THE CAUSE OF THE LOW LUBE OIL PRESSURE EXPERIENCED WITH DIESEL GENERATOR 1-2 REMAINS SOMEWHAT INDETERMINATE, IT PRELIMINARILY HAS BEEN ATTRIBUTED TO A STICKING PRESSURE REGULATING VALVE.

[156] PALISADES DOCKET 50-255 LER 89-019
ANALYZED BORON DILUTION INCIDENT NOT BOUNDING FOR NEWLY IDENTIFIED POTENTIAL SINGLE FAILURE.
EVENT DATE: 073189 REPORT DATE: 083089 NSSS: CE TYPE: PWR

(NSIC 215114) ON 7/31/89, IT WAS DETERMINED THAT THE "BORON DILUTION INCIDENT", AS PRESENTED IN PALISADES FINAL SAFETY ANALYSIS REPORT (FSAR) SECTION 14.3, WAS NOT BOUNDING FOR A RECENTLY HYPOTHESIZED EVENT. IN ORDER FOR THIS HYPOTHISIZED EVENT TO OCCUR, THE PLANT WOULD HAVE TO BE UTILIZING THE SHUTDOWN COOLING SYSTEM (BP) WITH THE PRIMARY COOLANT SYSTEM (PCS)(AB) DEPRESSURIZED, ALL THREE CHARGING PUMPS (CB;P) WOULD HAVE TO BE SECURED, A PRIMARY MAKEUP WATER (PMW) PUMP (CA;P) WOULD HAVE TO BE INSERVICE AND THE MANUAL PMW BYPASS VALVE MV-2167 (CA;V) TO THE CHARGING PUMP SUCTION WOULD HAVE TO FAIL OR BE MIS-POSITONED. WITH THESE CONDITIONS AND FAILURE, IT HAS BEEN DETERMINED THAT AN OPERATING PMW PUMP WOULD BE ABLE TO PUMP APPROX. 90 GALLONS/MINUTE OF UNBORATED WATER INTO THE PCS. THIS FLOW RATE IS GREATER THAN THAT RESULTING FROM OTHER FAILURES AND THAT PREVIOUSLY ANALYZED. THIS FLOW RATE COULD SUBSEQUENTLY CAUSE A RETURN TO CRITICALITY WHEN IN THE COLD SHUTDOWN CONDITION. THE FAILURE TO IDENTIFY THIS HYPOTHESIZED EVENT HAS BEEN ATTRIBUTED TO PERSONNEL INVOLVED IN ANALYSES NOT BEING AWARE OF THE POTENTIAL UNBORATED WATER FLOW PATH THROUGH THE IDLED CHARGING PUMPS. APPLICABLE OPERATING PROCEDURES ARE BEING REVISED TO ENSURE A SINGLE FAILURE OR PERSONNEL ERROR DOES NOT PROVIDE A DILUTION PATH. SYSTEM ENHANCEMENTS ARE BEING EVALUATED WHICH WILL PHYSICALLY PRECLUDE THIS HYPOTHESIZED EVENT.

[157] PALO VERDE 1 DOCKET 50-528 LER 88-010 REV 01
UPDATE ON GROUND FAULT IN 13.8 KV BUS CAUSES FIRE IN UNIT AUXILIARY TRANSFORMER AND REACTOR TRIP.
EVENT DATE: 070688 REPORT DATE: 082889 NSSS: CE TYPE: PWR
VENDOR: DUFF-NORTON COMPANY
GENERAL ELECTRIC CO.
L & S MACHINE CO., INC.
WAGNER ELECTRIC CORP.

(NSIC 215101) ON JULY 6, 1988, PALO VERDE UNIT I WAS IN MODE 1 (POWER OPERATION) AT APPROXIMATELY 100 PERCENT POWER WHEN THE REACTOR TRIPPED ON A LOW DEPARTURE FROM NUCLEATE BOILING RATIO (DNBR). THE NON-CLASS IE 13.8 KV BUS IE-NAN-S02 FAULTED TO GROUND EXACERBATING AN EXISTING WEAKNESS IN THE UNIT AUXILIARY TRANSFORMER CAUSING THE UNIT AUXILIARY TRANSFORMER TO RUPTURE AND CATCH FIRE. THIS CAUSED A LOSS OF ELECTRICAL POWER TO THE REACTOR COOLANT PUMPS INITIATING THE LOW DNBR TRIP. THE REACTOR WAS STABILIZED IN MODE 3 (HOT STANDBY) ON NATURAL CIRCULATION. A NOTIFICATION OF UNUSUAL EVENT (NUE) WAS DECLARED AT 1215 MST DUE TO THE FIRE. THE FIRE WAS EXTINGUISHED AND THE NUE TERMINATED AT 1221 MST. AT 1303 MST AN ATTEMPT WAS MADE TO REENERGIZE THE FAULTED BUS 1E-NAN-S02 AND A FIRE STARTED IN THE SWITCHGEAR. AN NUE WAS DECLARED DUE TO THE FIRE AND LOSS OF NON-CLASS 13.8 KV ELECTRICAL POWER. THIS FIRE WAS EXTINGUISHED AT 1322 MST. FOLLOWING VISUAL INSPECTIONS, CLEANING, AND MEGGERING, IE-NAN-SOI WAS REENERGIZED AT 1749 MST. REACTOR COOLANT PUMP IA WAS STARTED AT 0033 MST ON JULY 7, 1988, AND FORCED CIRCULATION WAS REESTABLISHED. THE NUE WAS TERMINATED AT 0102 MST ON JULY 7, 1988 AND A COOLDOWN WAS COMMENCED. CORRECTIVE ACTIONS TAKEN TO PREVENT RECURRENCE INCLUDE THE REWORK OF BUS IE-NAN-SO2 AND THE REPLACEMENT OF THE UNIT AUXILIARY TRANSFORMER.

[158] PALO VERDE 1 DOCKET 50-528 LER 89-008
LOW LEVEL RADIOACTIVITY FOUND IN ON-SITE LANDFILL.
EVENT DATE: 042989 REPORT DATE: 081489 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: PALO VERDE 3 (PWR)

(NSIC 215038) ON 7/14/89 AN INTERNAL AUDIT REVEALED THAT COOLING TOWER SLUDGE CONTAINED LOW LEVEL AMOUNTS OF RADIOACTIVITY. AN INVESTIGATION DETERMINED THAT COOLING TOWER SLUDGE HAD BEEN REMOVED FROM UNITS 1 AND 3 FOR DISPOSAL IN AN ONSITE LANDFILL. AS CORRECTIVE ACTION THE COMPANY HAS REQUESTED APPROVAL FROM A STATE AGENCY TO DISPOSE OF COOLING TOWER SLUDGE IN THE LANDFILL. FURTHER THE COMPANY PROPOSES TO SEEK AUTHORITY UNDER 10 CFR 20.302 FOR ONSITE DISPOSAL OF LICENSED MATERIAL.

[159] PALO VERDE 1 DOCKET 50-528 LER 89-014
SPECIAL REPORT MISSED ON SEISMIC MONITORING SYSTEM.
EVENT DATE: 060189 REPORT DATE: 072189 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: PALO VERDE 2 (PWR)
PALO VERDE 3 (PWR)

(NSIC 214909) ON JUNE 22, 1989 IT WAS DISCOVERED THAT THE SEISMIC MONITOR LOCATED ON THE REACTOR COOLANT PUMP HAD BEEN DISCONNECTED AND THUS, INOPERABLE SINCE APRIL 23, 1989. A SPECIAL REPORT WAS REQUIRED ON JUNE 1, 1989 AND WAS NOT SUBMITTED. A HISTORICAL REVIEW SHOWED THE SAME SEISMIC MONITOR HAD ALSO BEEN DISCONNECTED BETWEEN OCTOBER 16, 1987 AND DECEMBER 17, 1987 WITHOUT THE SUBMISSION OF A SPECIAL REPORT. THE ROOT CAUSE WAS DETERMINED TO BE A PROCEDURAL FAILURE WHICH ALLOWED DISCONNECTING OF THE LEADS TO THE SEISMIC MONITOR WITHOUT THE APPROPRIATE APPROVALS. AS CORRECTIVE ACTION, THE PROCEDURE FOR REMOVAL OF THE REACTOR COOLANT PUMP MOTOR IS BEING REVISED TO ALERT PERSONNEL THAT REMOVAL OF THE 2B REACTOR COOLANT PUMP IN UNIT 1 ALSO RENDERS THE SEISMIC MONITOR SYSTEM INOPERABLE. A PREVIOUS SIMILAR EVENT WAS IDENTIFIED IN LER 528/88-001-00.

[160] PALO VERDE 1 DOCKET 50-528 LER 89-013
POTENTIALLY UNQUALIFIED CONTAINMENT PURGE ISOLATION VALVES.
EVENT DATE: 072689 REPORT DATE: 082589 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: PALO VERDE 2 (PWR)
PALO VERDE 3 (PWR)
VENDOR: BETTIS CORPORATION

(NSIC 215133) ON JULY 26, 1989, AT APPROXIMATELY 1803 MST, PALO VERDE UNIT 1 WAS IN A REFUELING OUTAGE WITH THE CORE (AC) OFF-LOADED TO THE SPENT FUEL POOL, PALO VERDE UNIT 2 WAS IN MODE 1 (POWER OPERATIONS) AT 100 PERCENT POWER, AND PALO VERDE UNIT 3 WAS IN MODE 6 (REFUELING) WHEN APS ENGINEERING DEPARTMENT PERSONNEL DETERMINED THAT THE CONTAINMENT PURGE POWER ACCESS VALVES WERE NOT INSTALLED IN ACCORDANCE WITH THE CONFIGURATION SPECIFIED IN THE ENVIRONMENTAL QUALIFICATION REPORT FOR THOSE VALVES. THE CONTAINMENT PURGE POWER ACCESS VALVES HAD A HANDWHEEL ON THE MANUAL JACKING SCREW WHICH WAS NOT INCLUDED IN THE SEISMIC ANALYSIS FOR THE VALVES. THESE HANDWHEELS WERE INSTALLED IN ALL THREE UNITS. AS IMMEDIATE CORRECTIVE ACTION, THE CONTAINMENT PURGE POWER ACCESS VALVES WERE DECLARED INOPERABLE UNTIL THE HANDWHEELS WERE REMOVED. AS CORRECTIVE ACTION TO PREVENT RECURRENCE, THE HANDWHEEL KIT (I.E. HANDWHEEL, NUT AND WASHER) HAVE BEEN REMOVED FROM POWER ACCESS PURGE VALVES IN UNIT 1, 2, AND 3 IN ACCORDANCE WITH APPROVED DESIGN CHANGE DOCUMENTS. AN INVESTIGATION OF THIS EVENT IS IN PROGRESS IN ACCORDANCE WITH THE APS INCIDENT INVESTIGATION PROGRAM. THE INVESTIGATION IS EXPECTED TO BE COMPLETED BY FEBRUARY 28, 1990. FOLLOWING THE COMPLETION OF THE INVESTIGATION, A SUPPLEMENT TO THIS REPORT WILL BE ISSUED. NO PREVIOUS SIMILAR EVENTS HAVE BEEN REPORTED.

[161] PALO VERDE 2 DOCKET 50-529 LER 89-001 REV 01
UPDATE ON ESF ACTUATION CAUSED BY LOSS OF POWER TO CLASS 1E 4.16 KV BUSSES.
EVENT DATE: 010389 REPORT DATE: 072789 NSSS: CE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 214910) AT 1940 MST ON JANUARY 3, 1989, PALO VERDE UNIT 2 WAS IN MODE 1 (POWER OPERATION) AT APPROXIMATELY 100 PERCENT POWER WHEN A TOTAL LOSS OF OFFSITE POWER TO THE CLASS IE 4.16 KV BUSSES 2E-PBA-S03 AND 2E-PBB-S04 OCCURRED. THE LOSS OF POWER (LOP) TO BUSSES 2E-PBA-S03 AND 2E-PBB-S04 GENERATED AN ENGINEERED SAFETY FEATURES (ESF) SIGNAL WHICH AUTOMATICALLY STARTED BOTH "A" AND "B" DIESEL GENERATORS. THE DIESEL GENERATORS STARTED SATISFACTORILY AND ASSUMED THE LOADS ON THE 4.16 KV CLASS IE BUSSES. AT APPROXIMATELY 1959 MST ON JANUARY 3, 1989 A NOTIFICATION OF UNUSUAL EVENT (NUE) WAS DECLARED DUE TO THE LOSS OF BOTH OFFSITE POWER SOURCES TO THE IN-PLANT CLASS IE BUSSES. THE NUE WAS TERMINATED AT APPROXIMATELY 1648 MST ON JANUARY 4, 1989 AFTER OFFSITE POWER WAS RESTORED TO ONE

OF THE CLASS IE 4.16 KV BUSSES. THE LOSS OF POWER (LOP) WAS CAUSED BY A REDUCED IMPULSE WITHSTAND VOLTAGE RATING DUE TO RAIN SATURATION OF CONTAMINATION ON THE ESF TRANSFORMER BUSHINGS, AND A POSSIBLE GROUND POTENTIAL RISE ATTRIBUTED TO LIGHTNING IN THE AREA. THIS ALLOWED A FAULT TO INITIALIZE AT THE ESF TRANSFORMER BUSHINGS. THE CONTAMINATION IS A BUILD UP OF MINERAL DEPOSITS FROM MISTING OF THE COOLING TOWERS. TWO BUSHINGS ON EACH TRANSFORMER FAILED DUE TO THE FAULT AND WERE REPLACED.

[162] PALO VERDE 2 DOCKET 50-529 LER 89-004
TECH SPEC ACTION REQUIREMENT PERFORMED LATE DUE TO PERSONNEL ERROR.
EVENT DATE: 061789 REPORT DATE: 071789 NSSS: CE TYPE: PWR

(NSIC 214835) ON JUNE 7, 1989, AT APPROXIMATELY 0320 MST, PALO VERDE UNIT 2 WAS IN MODE 3 (HOT STANDBY) WHEN THE SHIFT SUPERVISOR DISCOVERED THAT THE OPERABILITY AND ACTION REQUIREMENTS OF TECHNICAL SPECIFICATION 3.8.1.1.A HAD NOT BEEN MET AS A RESULT OF THE ONSITE CLASS 1E DISTRIBUTION SYSTEM NOT BEING SUPPLIED BY TWO PHYSICALLY INDEPENDENT CIRCUITS. BOTH CLASS 1E BUSSES WERE BEING SUPPLIED BY A SINGLE STARTUP TRANSFORMER. THE CAUSE OF THE MISSED ACTION WAS A COGNITIVE PERSONNEL ERROR BY THE OPERATIONS PERSONNEL. AS IMMEDIATE CORRECTIVE ACTION, ONE OF THE 4.16 KV CLASS 1E BUSSES WAS SHIFTED TO ITS NORMAL POWER SUPPLY. AS CORRECTIVE ACTION TO PREVENT RECURRENCE, PERSONNEL INVOLVED WERE COUNSELED. UNIT 2 PERSONNEL WERE BRIEFED ON THE NEED FOR THOROUGH SHIFT TURNOVERS AND THAT THEY BE COGNIZANT OF BUS ALIGNMENT, AND A PRECAUTION WAS ADDED TO THE 13.8 KV ELECTRICAL SYSTEM OPERATING PROCEDURE. A PREVIOUS SIMILAR EVENT WAS REPORTED IN UNIT 2 LER 529/88-003.

[163] PALO VERDE 2 DOCKET 50-529 LER 89-009
REACTOR TRIP DUE TO PARTIAL LOSS OF FORCED FLOW.
EVENT DATE: 071289 REPORT DATE: 081489 NSSS: CE TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.
PACIFIC VALVES, INC.

(NSIC 215039) ON JULY 12, 1989 AT APPROXIMATELY 2212 MST PALO VERDE UNIT 2 WAS OPERATING AT APPROXIMATELY 100 PERCENT POWER WHEN 2 OF THE 4 REACTOR COOLANT PUMPS WERE LOAD SHED FROM THEIR POWER SUPPLY (BUS 2E-NAN-S02), RESULTING IN A REACTOR TRIP ON CALCULATED LOW DNBR DUE TO LOW REACTOR COOLANT FLOW. IMMEDIATELY FOLLOWING THE TRIP, A SAFETY INJECTION ACTUATION SIGNAL (SIAS) AND CONTAINMENT ISOLATION ACTUATION SIGNAL (CIAS) ENGINEERED SAFETY FEATURES OCCURRED ON LOW REACTOR COOLANT SYSTEM (RCS) PRESSURE. FOLLOWING THE EVENT, AT APPROXIMATELY 1529 MST ON JULY 13, 1989, A PORTION OF THE MAIN FEEDWATER SYSTEM (MFWS) WAS OVERPRESSURIZED. THE CAUSE OF THE LOAD SHED WAS A FAILED FUSE IN THE BUS POTENTIAL TRANSFORMER. THE CAUSE OF THE SIAS/CIAS WAS RCS DEPRESSURIZATION DUE TO IMPROPER STEAM BYPASS CONTROL SYSTEM (SBCS) RESPONSE AND LEAKING PRESSURIZER SPRAY VALVES. THE CAUSE OF THE MFWS OVERPRESSURIZATION WAS A FAILED CHECK VALVE. IMMEDIATE CORRECTIVE ACTION TAKEN WAS TO REPLACE THE FUSE. AN INDEPENDENT INVESTIGATION IS BEING CONDUCTED TO DETERMINE THE CAUSES OF THE INCIDENTS WHICH OCCURRED DURING THIS EVENT. THIS SUBMITTAL ALSO PROVIDES A SPECIAL REPORT IN ACCORDANCE WITH TECHNICAL SPECIFICATION 3.5.2 ACTION B.

[164] PALO VERDE 3 DOCKET 50-530 LER 89-007 REV 01
UPDATE ON POTTER AND BRUMFIELD RELAY MALFUNCTIONS.
EVENT DATE: 050389 REPORT DATE: 072589 NSSS: CE TYPE: PWR
VENDOR: POTTER & BRUMFIELD

(NSIC 214912) ON MAY 3, 1989 AT APPROXIMATELY 0730 MST, PALO VERDE UNIT 3 WAS IN A REFUELING OUTAGE WITH THE CORE OFF-LOADED WHEN APS DETERMINED THAT DEFICIENCIES DISCOVERED DURING THE INSTALLATION OF POTTER AND BRUMFIELD (P&B) RELAYS CONSTITUTED A REPORTABLE CONDITION PURSUANT TO 10CFR21 AND 10CFR50.73. THE P&B

RELAYS ARE UTILIZED IN THE PVNGS ENGINEERED SAFETY FEATURES ACTUATION SYSTEMS AND CAUSE SAFETY-RELATED COMPONENTS TO ACTUATE WHEN DE-ENERGIZED. ON AUGUST 3, 1988, APS REPORTED A DEFICIENCY IN THE P&B MDR SERIES RELAYS (REFERENCE LER 528/88-018). AS A RESULT, APS AND P&B RE-DESIGNED THE RELAYS FOR INSTALLATION DURING THE PVNGS UNIT 1, 2, AND 3 REFUELING OUTAGES. DURING POST INSTALLATION TESTING OF THE RELAYS IN UNIT 3 ON APRIL 24 AND 25, 1989 AND PRIOR TO DECLARING THE RELAYS OPERABLE, IT WAS DISCOVERED THAT APPROXIMATELY TWENTY-FIVE PERCENT OF THE NEW MODEL RELAYS MALFUNCTIONED. THE CAUSE OF THE RELAY MALFUNCTIONS HAS BEEN DETERMINED TO BE AN INADEQUATE METHODOLOGY OF APPLYING AN EPOXY MATERIAL TO THE RELAY COILS TO PRECLUDE CONTAMINATION OF THE ROTOR AND STATOR MATING SURFACES IN THE RELAY INTERNALS. THE EPOXY CAUSES THE ROTOR AND STATOR TO BOND WHICH RESULTS IN THE RELAY FAILING TO OPERATE.

[165] PALO VERDE 3 DOCKET 50-530 LER 89-006
MISSED SHIFT SURVEILLANCE TEST.
EVENT DATE: 060689 REPORT DATE: 070589 NSSS: CE TYPE: PWR

(NSIC 214586) AT APPROXIMATELY 1530 MST ON JUNE 6, 1989, PALO VERDE UNIT 3 WAS IN A REFUELING OUTAGE WITH THE CORE OFF-LOADED TO THE SPENT FUEL POOL WHEN OPERATIONS PERSONNEL DISCOVERED THAT TWO REQUIRED DAYSHIFT, SHIFTLY SURVEILLANCES HAD NOT BEEN PERFORMED WITHIN THE REQUIRED INTERVAL. THE SHIFTLY CHANNEL CHECK FOR THE FUEL POOL AREA MONITOR (RU-31) AND THE SHIFTLY VERIFICATION THAT STEAM GENERATOR PRESSURE WAS WITHIN TECHNICAL SPECIFICATION LIMITS WERE NOT PERFORMED WITHIN THE PRESCRIBED SURVEILLANCE INTERVALS. THE APPROPRIATE SURVEILLANCES WERE COMPLETED AT APPROXIMATELY 550 MST ON JUNE 6, 1989. THE CAUSE OF THIS EVENT WAS A COGNITIVE PERSONNEL ERROR BY THE DAY-SHIFT OPERATIONS PERSONNEL RESPONSIBLE FOR PERFORMING THE SHIFTLY SURVEILLANCES. AS CORRECTIVE ACTION, THE RESPONSIBLE PERSONNEL HAVE BEEN COUNSELED, NIGHT ORDERS WERE WRITTEN TO REMIND PERSONNEL IN UNITS 1, 2, AND 3 TO PERFORM SHIFTLY SURVEILLANCES, AND PROCEDURE ENHANCEMENTS WILL BE MADE. PREVIOUS SIMILAR EVENTS WERE REPORTED IN LER'S 529/86-001 AND 529/86-029.

[166] PALO VERDE 3 DOCKET 50-530 LER 89-005
PLANT VENT LOW RANGE EFFLUENT MONITOR ALARM NOT PROPERLY INVESTIGATED.
EVENT DATE: 062889 REPORT DATE: 072889 NSSS: CE TYPE: PWR

(NSIC 214911) AT APPROXIMATELY 1115 MST ON JUNE 28, 1989, PALO VERDE UNIT 3 WAS IN A REFUELING OUTAGE WITH THE CORE OFF-LOADED TO THE SPENT FUEL POOL WHEN A UNIT 3 TECHNICIAN DISCOVERED THAT THE SAMPLE FLOW RATE FOR THE PLANT VENT LOW RANGE RADIOACTIVE EFFLUENT MONITOR (RU-143) WAS BELOW THE LOW FLOW ALARM SETPOINT RENDERING THE MONITOR INOPERABLE. INVESTIGATION DETERMINED THAT THE LOW FLOW ALARM HAD OCCURRED AT APPROXIMATELY 0531 MST ON JUNE 28, 1989; HOWEVER, THE ALARM WAS NOT PROPERLY INVESTIGATED. THIS RESULTED IN NOT MEETING ACTION REQUIREMENTS 36 AND 40 OF TECHNICAL SPECIFICATION (T.S.) 3.3.3.8.1 THE CAUSE OF THE LOW FLOW CONDITION ON RU-143 WAS A LOOSE SET SCREW ON THE COUPLING BETWEEN THE MONITOR'S SAMPLE PUMP AND ITS DRIVE MOTOR. THE CAUSE OF THE IMPROPER FOLLOW-UP ACTION FOR THE LOW FLOW ALARM IS UNDER INVESTIGATION IN ACCORDANCE WITH THE PVNGS INCIDENT INVESTIGATION PROGRAM AND WILL BE DESCRIBED IN A SUPPLEMENT TO THIS REPORT. AS CORRECTIVE ACTION, THE PRE-PLANNED ALTERNATE SAMPLING PROGRAM WAS IMPLEMENTED BY 1150 MST ON JUNE 28, 1989, FULFILLING T.S. 3.3.3.8 ACTION REQUIREMENTS. THE LOOSE SET SCREW WAS TIGHTENED AND RU-143 WAS RETURNED TO SERVICE AT APPROXIMATELY 1558 MST ON JUNE 24, 1989. A PREVIOUS SIMILAR EVENT WAS REPORTED IN UNIT 1 LER 528/85-067.

[167] PALO VERDE 3 DOCKET 50-530 LER 89-008
SURVEILLANCE REQUIREMENTS NOT PERFORMED SATISFACTORILY.
EVENT DATE: 063089 REPORT DATE: 072789 NSSS: CE TYPE: PWR

(NSIC 214913) ON JUNE 30, 1989, PALO VERDE UNIT 3 WAS IN A REFUELING OUTAGE WITH THE CORE OFF-LOADED TO THE SPENT FUEL POOL WHEN MAINTENANCE PERSONNEL IDENTIFIED A SIXTH TRIP LEVER ARM ON THE TEN (10) TON FUEL HANDLING CRANE RAILS. THESE TRIP LEVER ARMS ROTATE A STAR WHEEL WHICH ACTUATES LIMIT SWITCHES TO PREVENT LOADS IN EXCESS OF 2000 POUNDS FROM TRAVEL OVER FUEL ASSEMBLIES IN THE SPENT FUEL STORAGE POOL. THE SIXTH TRIP LEVER ARM ROTATED THE STAR WHEEL A SIXTH TIME ACTIVATING THE LIMIT SWITCHES WHICH ALLOW THE CRANE TROLLEY/HOOK TO TRAVEL OVER THE SPENT FUEL STORAGE POOL WHEN IN THE CONTAINER MODE. IN ACCORDANCE WITH A FIELD CHANGE REQUEST IN AN AUTHORIZED DESIGN CHANGE PACKAGE THERE SHOULD ONLY BE FIVE (5) TRIP LEVER ARMS FOR THE CRANE INTERLOCKS. THIS DESIGN CHANGE PACKAGE WAS COMPLETED IN MARCH OF 1986. AN INVESTIGATION OF THIS EVENT IS BEING CONDUCTED IN ACCORDANCE WITH THE PVNGS INCIDENT INVESTIGATION PROGRAM. THE RESULTS OF THE INVESTIGATION WILL BE REPORTED IN A REVISED LER ONLY IF IT WOULD SIGNIFICANTLY CHANGE THE READER'S PERCEPTION OF THE COURSE, SIGNIFICANCE, IMPLICATIONS, OR CONSEQUENCES OF THE EVENT; OR RESULTS IN SUBSTANTIAL CHANGES IN THE CORRECTIVE ACTION PLANNED BY THE LICENSEE. IMMEDIATE CORRECTIVE ACTION WAS TO MOVE THE CRANE TROLLEY/HOOK FROM OVER THE SPENT FUEL STORAGE POOL AND DECLARE THE CRANE INOPERABLE. AS CORRECTIVE ACTION TO PREVENT RECURRENCE, THE SIXTH TRIP LEVER ARM WAS REMOVED.

[168] PALO VERDE 3 DOCKET 50-530 LER 89-009
INADVERTENT FUEL BUILDING ESSENTIAL VENTILATION ESF ACTUATION.
EVENT DATE: 072889 REPORT DATE: 082289 NSSS: CE TYPE: PWR
VENDOR: GOULD BROWN BOVERI COMPANY

(NSIC 215154) AT APPROXIMATELY 0430 MST ON JULY 28, 1989, PALO VERDE UNIT 3 WAS IN MODE 6 (REFUELING) WITH THE REACTOR COOLANT SYSTEM AT AMBIENT TEMPERATURE AND CORE RELOADING IN PROGRESS WHEN AN INADVERTENT TRAIN "A" FUEL BUILDING ESSENTIAL VENTILATION ACTUATION SIGNAL (FBEVAS) WAS INITIATED ON THE BALANCE OF PLANT ENGINEERED SAFETY FEATURE ACTUATION SYSTEM. THE TRAIN "A" FBEVAS RESULTED IN THE DESIGNED CROSS-TRIPS OF TRAIN "B" FBEVAS AND TRAIN "A" AND "B" CONTROL ROOM ESSENTIAL FILTRATION ACTUATION SIGNALS (CREFAS). THE ACTUATION OCCURRED WHEN A MAINTENANCE INDIVIDUAL RESET THE SPENT FUEL POOL AREA RADIATION MONITOR (RU-31) REMOTE INDICATING AND CONTROL UNIT WITHOUT ENSURING THAT THE CHANNEL WAS PLACED IN "BYPASS." FOLLOWING THE ACTUATION, CONTROL ROOM ESSENTIAL VENTILATION SYSTEM TRAIN "B" FAN TRIPPED. ALL OTHER COMPONENTS OPERATED AS DESIGNED. RADIATION PROTECTION PERSONNEL VERIFIED THAT NO ACTUAL HIGH RADIATION LEVELS EXISTED IN THE AREA OF THE SPENT FUEL POOL AREA MONITOR. THE ROOT CAUSE OF THIS EVENT WAS A COGNITIVE PERSONNEL ERROR BY AN APS MAINTENANCE INDIVIDUAL WHO DID NOT ENSURE THAT RU-31 WAS PLACED IN BYPASS IN ACCORDANCE WITH APPROVED PROCEDURES. AS CORRECTIVE ACTION, THE INDIVIDUAL HAS BEEN COUNSELED. PREVIOUS SIMILAR EVENTS WERE REPORTED IN LER'S 528/85-033 AND 528/87-026.

[169] PEACH BOTTOM 2 DOCKET 50-277 LER 88-012 REV 02
UPDATE ON REPORT REQUIRED BY TECH SPECS ON INOPERABLE DRYWELL RADIATION MONITORS NOT SUBMITTED.
EVENT DATE: 040287 REPORT DATE: 081589 NSSS: GE TYPE: BWR
OTHER UNITS INVOLVED: PEACH BOTTOM 3 (BWR)

(NSIC 215010) ON 05/24/88, IT WAS DISCOVERED THAT THE UNIT 2 AND UNIT 3 DRYWELL HIGH RANGE RADIATION MONITORS WERE OUT OF SERVICE WITHOUT BEING REPORTED TO THE NRC AS REQUIRED BY THE TECH SPECS (TS). TS 3.2.F REQUIRES AT LEAST 2 OF THE 4 INSTRUMENT CHANNELS (ON EACH UNIT) TO BE OPERABLE. AN ALTERNATE METHOD OF MONITORING DRYWELL RADIATION, WEEKLY MANUAL SURVEYS, WAS IN PLACE IN ACCORDANCE WITH TS 3.2.F WHEN THE MONITORS WERE REMOVED FROM SERVICE. THE RECORDERS FOR UNIT 2 MONITORS (RE-8103 A, B, C AND D) WERE TURNED OFF ON 03/19/87. THE UNIT 3 RE-9103 A AND C MONITORS WERE REMOVED FROM SERVICE ON 11/07/87 WHILE THE RECORDERS FOR THE B AND D CHANNELS WERE TURNED OFF. THESE EVENTS WERE NOT REPORTED BECAUSE REMOVAL OF THE MONITORS FROM SERVICE WAS NOT PROPERLY DOCUMENTED AS PROCEDURE REQUIRED. RE 9103 B AND D AND RE 8103 B AND D WERE RETURNED TO

SERVICE ON 06/16/88 AND 08/30/88 RESPECTIVELY. THE REMAINING MONITORS WILL BE RETURNED TO SERVICE PRIOR TO RESTART. AN OPERATIONS MANAGEMENT MANUAL WAS DEVELOPED AND ESTABLISHED CONTROLS FOR TRACKING LIMITING CONDITIONS FOR OPERATIONS. THIS MANUAL ALONG WITH THE COMPLETED TRAINING SHOULD PREVENT RECURRENCE OF THESE EVENTS. A PLANT OPERATIONS REVIEW COMMITTEE POSITION HAS BEEN ISSUED WHICH CLARIFIES THE APPLICABILITY REQUIREMENTS OF THE DRYWELL HIGH RANGE RADIATION MONITORS.

[170] PEACH BOTTOM 2 DOCKET 50-277 LER 89-013
INEFFECTIVE TEST STATUS COMMUNICATION CAUSES MISSED TECH SPEC SURVEILLANCE.
EVENT DATE: 041189 REPORT DATE: 072189 NSSS: GE TYPE: BWR

(NSIC 214877) ON 6/3/89, DURING REVIEW OF THE SURVEILLANCE TEST AND RECORDS SYSTEM (STARS) DATABASE, IT WAS NOTED THAT A REQUIRED SURVEILLANCE TEST FOR THE DIESEL GENERATOR (DG) CARDOX ROOM SMOKE DETECTORS HAD BECOME "OVERDUE" (EXCEEDED ITS GRACE PERIOD) ON 1/11/89. THE FUNCTIONAL TEST WAS NOT ABLE TO BE PERFORMED BECAUSE THE DETECTORS WERE LOCKED OUT OF SERVICE. FURTHER REVIEW REVEALED THE TEST HAS NOT BEEN PERFORMED SINCE RETURNING THE DETECTORS TO SERVICE AND SECURING THE FIREWATCH ON 4/11/89. THE DETECTORS HAD BEEN REMOVED FROM SERVICE AS PART OF A "BLOCK" PROVIDED TO ALLOW MODIFICATION (2390) OF FIRE DETECTORS IN THE ADJACENT DG ROOMS. NO SAFETY CONSEQUENCES OCCURRED AS A RESULT OF THIS EVENT. THE REQUIRED SURVEILLANCE TESTS WERE PERFORMED IMMEDIATELY WITH SATISFACTORILY RESULTS AND THE DETECTORS DECLARED OPERABLE AT 1221 ON 6/3/89. THE ROOT CAUSE OF THIS EVENT WAS LACK OF AN EFFECTIVE METHOD FOR COMMUNICATING SURVEILLANCE TEST STATUS TO SHIFT MANAGEMENT. AS A RESULT, HE SHIFT MANAGER HAS BEEN PLACED ON DIRECT DISTRIBUTION FOR SURVEILLANCE TEST STATUS REPORTS. PRIOR TO DECLARING A TECHNICAL SPECIFICATION SYSTEM OPERABLE THESE REPORTS HALL BE REVIEWED TO ENSURE THE REQUIRED TESTS HAVE BEEN COMPLETED. NO PREVIOUS SIMILAR LERS WERE IDENTIFIED.

[171] PEACH BOTTOM 2 DOCKET 50-277 LER 89-012 REV 01
UPDATE ON FEEDWATER CONTROL MALFUNCTION CAUSES LOW LEVEL REACTOR SCRAM.
EVENT DATE: 051989 REPORT DATE: 081589 NSSS: GE TYPE: BWR
VENDOR: CUTLER-HAMMER

(NSIC 215015) AT 0721 ON 5/19/89 WITH UNIT 2 AT 24% THERMAL POWER THE FEEDWATER LEVEL CONTROL SYSTEM WAS TRANSFERRED FROM SINGLE ELEMENT TO THREE ELEMENT CONTROL. IMMEDIATELY, THE "REACTOR HI-LO LEVEL" ALARM ANNUNCIATED, FOLLOWED BY BOTH "B" AND "C" REACTOR FEED PUMPS (RFP) TRIPPING SIMULTANEOUSLY. FEEDWATER LEVEL CONTROL WAS RETURNED TO SINGLE ELEMENT AND A RESTART OF "C" RFP ATTEMPTED. BEFORE FEED FLOW COULD BE RESTABLISHED, LEVEL DECREASED BELOW 0 INCHES RESULTING IN AN AUTOMATIC SCRAM AND GROUP II AND III ISOLATIONS. THE "C" RFP WAS RESTARTED AND LEVEL DECREASE STOPPED ABOVE 48 INCHES. AT THIS LEVEL, ALTERNATE ROD INSERTION BACKUP SCRAM INITIATED AND BOTH REACTOR RECIRCULATION PUMPS TRIPPED. THE HIGH PRESSURE COOLANT INJECTION AND REACTOR CORE ISOLATION COOLING SYSTEMS ALSO RECEIVE INITIATION SIGNALS AT THIS LEVEL, BUT DID NOT ACTUATE AS THE LOGIC WAS NOT SATISFIED. REACTOR WATER LEVEL WAS RESTORED TO < 0 INCHES AND THE UNIT STABILIZED IN THE HOT SHUTDOWN CONDITION. THE SCRAM AND GROUP II AND III ISOLATIONS WERE RESET AND ESSENTIAL SYSTEMS RETURNED TO SERVICE. THE ROOT CAUSE OF THIS EVENT WAS FAILURE OF THE FEEDWATER LEVEL CONTROL ELECTOR SWITCH. THE FAILED SWITCH WAS REPLACED, FEEDWATER CONTROL AMPLIFIERS WERE CALIBRATED AND PROCEDURES WERE ENHANCED. THIS EVENT WILL BE REVIEWED WITH THE APPROPRIATE PLANT PERSONNEL. THREE SIMILAR LERS WERE IDENTIFIED.

[172] PEACH BOTTOM 2 DOCKET 50-277 LER 89-016
LOCAL POWER RANGE MONITOR SPIKING HIGH CAUSES SCRAM WHILE SHUTDOWN.
EVENT DATE: 072289 REPORT DATE: 081889 NSSS: GE TYPE: BWR

(NSIC 215117) AT 1110 ON 7/22/89 WITH UNIT 2 IN COLD SHUTDOWN THE REACTOR

PROTECTION SYSTEM (RPS) INITIATED A FULL REACTOR SCRAM SIGNAL. THE FULL SCRAM WAS THE RESULT OF A CHANNEL "B" RPS SCRAM SIGNAL BEING RECEIVED IN CONJUNCTION WITH A "A" RPS SCRAM SIGNAL ALREADY INSERTED. THE "B" RPS SCRAM SIGNAL WAS THE RESULT OF AN "F" AVERAGE POWER RANGE MONITOR (APRM) HI-HI SIGNAL CAUSED BY LOCAL POWER RANGE MONITOR (LPRM) DETECTOR 4B-40- 33 SPIKING UPSCALE. THE "A" RPS SCRAM SIGNAL HAD PREVIOUSLY BEEN MANUALLY INSERTED AS REQUIRED BY TECHNICAL SPECIFICATIONS (LESS THAN THE REQUIRED NUMBER OF INTERMEDIATE RANGE MONITORS (IRM) OPERABLE). IMMEDIATE CORRECTIVE ACTION WAS TO BYPASS LPRM 4B-40-33. THE "B" RPS SCRAM WAS RESET AFTER THE CAUSE OF THE SCRAM SIGNAL WAS DETERMINED. THE PROXIMATE CAUSE OF THIS EVENT WAS THE OUTPUT SIGNAL FOR LPRM DETECTOR 4B-40-33 SPIKING HIGH. NO ACTUAL SAFETY CONSEQUENCES OCCURRED AS A RESULT OF THIS EVENT. IT IS PLANNED TO PERFORM AN INVESTIGATION AND EVALUATE THE PERFORMANCE OF THE BYPASSED LPRM. NO PREVIOUS SIMILAR LERS WERE IDENTIFIED.

[173] PEACH BOTTOM 3 DOCKET 50-278 LER 89-001
TECH SPEC LIMITING CONDITION FOR OPERATION WAS NOT SATISFIED DUE TO A PROCEDURAL DEFICIENCY.
EVENT DATE: 072089 REPORT DATE: 081889 NSSS: GE TYPE: BWR

(NSIC 215118) ON 7/20/89 THE REACTOR BLDG (RB) EXHAUST VENT FLOW RECORDER WAS MADE INOPERABLE IN ACCORDANCE WITH A SPECIAL PROCEDURE BY DE-ENERGIZING ITS INSTRUMENT PANEL. HOWEVER, A FLOW RATE ESTIMATE WAS NOT PERFORMED WITHIN 4 HOURS AS REQUIRED BY TECH SPECS (TECH. SPECS.) SECTION 3.8.C.4.D. THE ONCOMING SHIFT REALIZED THE DISCREPANCY, ESTIMATED THE RB EXHAUST VENT FLOW RATE, AND DETERMINED THE EFFLUENT RELEASE RATE WAS WITHIN THE LIMITS AS IDENTIFIED IN THE OFF SITE DOSE CALCULATION MANUAL. THERE WERE NO SAFETY CONSEQUENCES AS A RESULT OF THIS EVENT SINCE THE RB VENTILATION SYSTEM WAS WITHIN THE LIMITS AS IDENTIFIED IN THE OFF SITE DOSE CALCULATION MANUAL. THERE WERE NO SAFETY CONSEQUENCES AS A RESULT OF THIS EVENT SINCE THE RB VENTILATION SYSTEM WAS SHUT DOWN PRIOR TO DE-ENERGIZING THE INSTRUMENT PANEL AND ACTIVITY WAS LOW. THE RB EXHAUST VENT GAS MONITOR REMAINED OPERABLE THROUGHOUT THE EVENT. THE ROOT CAUSE OF THE EVENT WAS AN INADEQUATE SPECIAL PROCEDURE IN THAT IT DID NOT IDENTIFY THE TECH SPEC REQUIREMENT TO ESTIMATE RB EXHAUST VENT FLOW RATE WITHIN 4 HOURS OF REMOVING THE RECORDER FROM SERVICE. THE SHIFT TEAM WAS COUNSELED ON THE IMPORTANCE OF ATTENTION TO DETAIL, AND THE ADMINISTRATIVE PROCEDURE FOR WRITING SPECIAL PROCEDURES WILL BE REVISED TO PROVIDE ADDITIONAL GUIDANCE TO THE OPERATORS WHEN THE SPECIAL PROCEDURE HAS ASSOCIATED TECH SPEC REQUIREMENTS. THERE WERE NO PREVIOUS SIMIL

[174] PILGRIM 1 DOCKET 50-293 LER 81-036 REV 01
UPDATE ON AS-FOUND TRIP POINTS FOR PRESSURE SWITCH HIGHER THAN LIMITS.
EVENT DATE: 070981 REPORT DATE: 082389 NSSS: GE TYPE: BWR
VENDOR: BARKSDALE COMPANY

(NSIC 215091) WHILE PERFORMING SURVEILLANCE TESTS, THE AS-FOUND TRIP POINTS FOR PRESSURE SWITCHES PS-263-53A AND PS-263-52A WERE HIGHER THAN THE LIMIT OF THE TECH SPECS (TABLE 3.2.B) BY 12 PSI AND 17 PSI RESPECTIVELY. THESE PRESSURE SWITCHES CONTRIBUTE TO THE AUTOMATIC INITIATION SIGNALS AND THE LOW PRESSURE INJECTION VALVE PERMISSIVES FOR THE LPCI MODE OF THE RHR AND THE CORE SPRAY SYSTEMS. LPCI AND CORE SPRAY REMAINED FUNCTIONAL. THE OUT OF SPECIFICATION PRESSURE SWITCH TRIP POINTS HAD NO ADVERSE IMPACT ON THE PUBLIC HEALTH AND SAFETY. THE SWITCHES WERE RECALIBRATED TO BE WITHIN THE TECH SPEC LIMIT. INVESTIGATIONS SHOWED THAT THE TRIP POINT OF THESE BARKSDALE PRESSURE SWITCHES MAY DRIFT HIGH OR LOW DEPENDING UPON WHETHER THE SWITCHES WERE CALIBRATED WHEN THE REACTOR VESSEL WAS PRESSURIZED (AT POWER) OR DEPRESSURIZED (SHUTDOWN). TO PREVENT RECURRENCE, THE SWITCHES HAVE BEEN REPLACED WITH ROSEMOUNT TRANSMITTERS.

[175] PILGRIM 1 DOCKET 50-293 LER 89-014 REV 01
UPDATE ON INADVERTENT OVERPRESSURIZATION OF THE REACTOR CORE ISOLATION COOLING SYSTEM SUCTION PIPING DURING SURVEILLANCE TESTING AND SUBSEQUENT COMPLETION OF A SHUTDOWN.
EVENT DATE: 041289 REPORT DATE: 080389 NSSS: GE TYPE: BWR
VENDOR: ANCHOR/DARLING INDUSTRIES

(NSIC 214984) ON 4/12/89 AT 1000 HOURS, THE REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM WAS DECLARED INOPERABLE AND A 7 DAY LIMITING CONDITION FOR OPERATION BEGAN. THE SYSTEM WAS DECLARED INOPERABLE AS A RESULT OF A SCHEDULED RCIC SYSTEM LOGIC SYSTEM FUNCTIONAL TEST THAT RESULTED IN AN OVERPRESSURIZATION OF THE SYSTEM'S SUCTION PIPING AND THE DISCHARGE OF WATER FROM THE SUCTION PRESSURE RELIEF VALVE. THE SYSTEM'S TURBINE-PUMP WAS INTENTIONALLY REMOVED FROM ACTIVE SERVICE FOR THE TEST. THE EVENT OCCURRED DURING 25% REACTOR POWER OPERATION WITH THE REACTOR MODE SELECTOR SWITCH IN THE RUN POSITION. THE DIRECT CAUSE FOR THE EVENT WAS UTILITY OPERATOR (ONE LICENSED AND ONE NON-LICENSED) ERROR. THE OPERATORS INCORRECTLY POSITIONED AND/OR INACCURATELY VERIFIED THE POSITIONS OF 6 CIRCUIT BREAKERS TO MOTOR OPERATED VALVES PRIOR TO (AND FOR) THE TEST. THE RESPONSIBLE OPERATORS WERE SUSPENDED BECAUSE THE BREAKERS WERE INCORRECTLY POSITIONED FOR THE TEST. THE OPERATORS HAVE BEEN REASSIGNED TO NON-OPERATOR DUTIES UNDER SEPARATE ADMINISTRATIVE ACTION. SIGNIFICANT CORRECTIVE ACTIONS TAKEN INCLUDE REVISION OF PROCEDURES (REGARDING TAGGING AND INDEPENDENT VERIFICATION METHODS), ANALYSIS AND EXAMINATION OF THE SYSTEM PIPING, CHECK VALVE REPAIR, REPLACEMENT OF A DAMAGED SUCTION PRESSURE SWITCH AND PRESSURE GAUGE, AND RCIC SYSTEM TESTING.

[176] PILGRIM 1 DOCKET 50-293 LER 89-021
AUTOMATIC CLOSING OF THE INBOARD AND OUTBOARD PRIMARY CONTAINMENT SYSTEM GROUP 6 ISOLATION VALVES.
EVENT DATE: 061589 REPORT DATE: 071789 NSSS: GE TYPE: BWR

(NSIC 214880) ON JUNE 15, 1989 AT 1540 HOURS, AN AUTOMATIC ACTUATION OF THE INBOARD AND OUTBOARD REACTOR WATER CLEANUP (RWCU) SYSTEM PORTION OF THE PRIMARY CONTAINMENT ISOLATION CONTROL SYSTEM (PCIS) OCCURRED. THE ACTUATION RESULTED IN THE AUTOMATIC CLOSING OF THE INBOARD AND OUTBOARD PRIMARY CONTAINMENT SYSTEM GROUP 6 (SIX)/RWCU SYSTEM ISOLATION VALVES AND A TEMPORARY INTERRUPTION IN RWCU SYSTEM OPERATION. THE PCIS LOGIC CIRCUITRY WAS RESET AND THE RWCU SYSTEM WAS RETURNED TO SERVICE ON JUNE 15, 1989 AT 1715 HOURS AFTER VERIFYING THE CAUSE FOR THE ACTUATION. THE DIRECT CAUSE FOR THE ACTUATION WAS AIR (GAS) IN THE INSTRUMENTATION SENSING LINES OF THE RWCU SYSTEM FLOW SENSORS. THE MOST LIKELY CAUSE FOR THE AIR WAS DISSOLVED GASSES THAT CAME OUT OF SOLUTION WHEN THE REACTOR VESSEL WAS PREVIOUSLY DEPRESSURIZED, AND/OR WHEN THE SENSING LINES WERE PREVIOUSLY BACKFILLED. THE STEPS TO BE TAKEN FOR BACKFILLING WERE NOT IDENTIFIED IN A FORMAL PROCEDURE. THE SENSING LINES WERE BLED TO ELIMINATE THE AIR. A NEW PROCEDURE IS BEING DEVELOPED TO FORMALIZE THE STEPS FOR BACKFILLING (OR BLEEDING) THE FLOW SENSING LINES. THE STARTUP PROCEDURE IS BEING REVISED TO ADD A STEP(S) FOR PURGING AIR THAT MIGHT BE TRAPPED IN THE SENSING LINES FROM DISSOLVED GASSES. THIS EVENT OCCURRED DURING A STARTUP WITH THE REACTOR MODE SELECTOR SWITCH IN THE STARTUP POSITION. THE CONTROL RODS WERE IN A PARTIALLY WITHDRAWN POSITION.

[177] PILGRIM 1 DOCKET 50-293 LER 89-022
AUTOMATIC CLOSING OF THE INBOARD PRIMARY CONTAINMENT SYSTEM GROUP 6 ISOLATION VALVE.
EVENT DATE: 071189 REPORT DATE: 081089 NSSS: GE TYPE: BWR

(NSIC 214985) ON JULY 11, 1989 AT 1609 HOURS, AN AUTOMATIC ACTUATION OF THE INBOARD REACTOR WATER CLEANUP (RWCU) SYSTEM PORTION OF THE PRIMARY CONTAINMENT ISOLATION CONTROL SYSTEM (PCIS) OCCURRED. THE ACTUATION RESULTED IN THE CLOSING OF THE INBOARD PRIMARY CONTAINMENT SYSTEM GROUP 6/RWCU SYSTEM ISOLATION VALVE

(MO-1201-2) AND A TEMPORARY INTERRUPTION IN RWCU SYSTEM OPERATION. INITIAL ACTION TAKEN WAS TO VERIFY THE INTEGRITY OF THE RWCU SYSTEM PIPING. THE PCIS LOGIC CIRCUITRY WAS RESET AND THE RWCU SYSTEM WAS RETURNED TO SERVICE ON JULY 11, 1989 AT 2130 HOURS. THE ROOT CAUSE FOR THE ACTUATION WAS UTILITY NON-LICENSED MAINTENANCE PLANNING PERSONNEL ERROR. DURING A WORK PLANNING INSPECTION, THE PLANNER INADVERTENTLY CAUSED A MOMENTARY ELECTRICAL DISCONTINUITY WHILE INSPECTING A DAMAGED TERMINAL BLOCK SCREW. THE DISCONTINUITY CAUSED THE RELATED CIRCUITRY TO ACTUATE. CORRECTIVE MEASURES BEING TAKEN INCLUDE REVISING THE MAINTENANCE WORK PLANNING PROCEDURE TO EMPHASIZE THOSE ACTIONS THAT ARE ALLOWED AND PROHIBITED DURING WORK PLAN WALKDOWNS. THIS EVENT OCCURRED DURING POWER OPERATION WITH THE REACTOR MODE SELECTOR SWITCH IN THE RUN POSITION. THE REACTOR VESSEL (RV) WATER TEMPERATURE WAS APPROXIMATELY 540 DEGREES FAHRENHEIT AND THE RV PRESSURE WAS 970 PSIG. THE REACTOR POWER LEVEL WAS 50 PERCENT. THIS REPORT IS SUBMITTED IN ACCORDANCE WITH 10 CFR 50.73(A)(2)(IV).

[178] PILGRIM 1 DOCKET 50-293 LER 89-023
DECREASING MAIN CONDENSER VACUUM RESULTING IN THE INITIATION OF A MANUAL SCRAM DUE TO PROCEDURE INADEQUACY.
EVENT DATE: 071889 REPORT DATE: 081789 NSSS: GE TYPE: BWR

(NSIC 215018) ON JULY 18, 1989 AT 1038 HOURS, A MANUALLY INITIATED REACTOR PROTECTION SYSTEM (RPS) SCRAM SIGNAL AND REACTOR SCRAM OCCURRED. THE SCRAM SIGNAL RESULTED IN AN AUTOMATIC SEQUENCE OF EXPECTED DESIGNED RESPONSES THAT INCLUDED A TURBINE-GENERATOR TRIP, AUTOMATIC OPENING OF TWO 345 KV SWITCHYARD AIR CIRCUIT BREAKERS, AND AN AUTOMATIC TRANSFER OF STATION ELECTRICAL LOADS. THE DIRECT CAUSE FOR THE SCRAM WAS THE DELIBERATE MOVEMENT OF THE REACTOR MODE SELECTOR SWITCH (RMSS) FROM THE RUN POSITION WHILE AT 35 PERCENT REACTOR POWER. THIS CONSERVATIVE ACTION WAS TAKEN DUE TO DECREASING MAIN CONDENSER VACUUM. THE PRIMARY CAUSE FOR THE DECREASING VACUUM WAS ATTRIBUTED TO AN INADEQUACY IN THE APPROVED PROCEDURE THAT WAS BEING USED TO RECONFIGURE THE STEAM JET AIR EJECTORS OF THE MAIN CONDENSER GAS REMOVAL SYSTEM. CORRECTIVE ACTIONS TAKEN INCLUDED THE FOLLOWING: REVISION OF THE PROCEDURE TO INCLUDE SPECIFIC STEPS FOR INTERCHANGING A STEAM JET AIR EJECTORS DURING OPERATION, AND CLEANING THE SALTWATER PORTIONS OF THE CIRCULATING WATER SYSTEM TO REMOVE MACRO-FOULING DUE TO MARINE ORGANISMS. THE UNIT WAS RETURNED TO COMMERCIAL SERVICE ON JULY 26, 1989 AT 0936 HOURS. THIS EVENT OCCURRED WHEN THE REACTOR VESSEL (RV) PRESSURE WAS 980 PSIG AND THE RV WATER TEMPERATURE WAS 320 DEGREES FAHRENHEIT. THIS REPORT IS SUBMITTED IN ACCORDANCE WITH 10 CFR 50.73(A)(2)(IV).

[179] POINT BEACH 1 DOCKET 50-266 LER 89-007
ATWS MITIGATING SYSTEM ACTUATION CIRCUITRY BISTABLE FAILURE.
EVENT DATE: 062589 REPORT DATE: 072589 NSSS: WE TYPE: PWR
VENDOR: ACTION INSTRUMENT CO., INC.

(NSIC 214915) ON 6/25/89, DURING NORMAL 100% POWER CONDITIONS, OPERATORS NOTED THE "LOSS OF FEEDWATER TURBINE TRIP BLOCKED" STATUS LIGHT ON CONTROL BOARD 1-C03 WAS ILLUMINATED. THIS INSTRUMENT CHANNEL WAS RECENTLY INSTALLED TO MEET THE REQUIREMENTS OF 10 CFR 50.62 - ANTICIPATED TRANSIENT WITHOUT SCRAM MITIGATING SYSTEM ACTUATION CIRCUITRY (AMSAC). THE P-20 BISTABLE ALARM MODULE WITHIN THE AMSAC INSTRUMENT CHANNEL WAS FOUND TO BE FAULTY BY INSTRUMENT TECHNICIANS. UPON DETERMINING THAT THE AMSAC CHANNEL WAS IN A DISABLED MODE, A POWER REDUCTION WAS INITIATED IN ACCORDANCE WITH PLANT TECH SPEC LIMITING CONDITION FOR OPERATION. THE P-20 BISTABLE ALARM MODULE WAS REPLACED AND THE AMSAC CHANNEL WAS RESTORED TO SERVICE. THE UNIT ACHIEVED 100% POWER OPERATION AT 9:00 A.M.; APPROXIMATELY 7 HOURS 46 MINUTES AFTER THE STATUS LIGHT WAS DISCOVERED LIT.

[180] POINT BEACH 2 DOCKET 50-301 LER 89-003
SAFETY INJECTION ACCUMULATOR LEVEL DETECTOR INSTRUMENT FAILURE.
EVENT DATE: 071289 REPORT DATE: 080989 NSSS: WE TYPE: PWR
VENDOR: MAGNETROL, INC.

(NSIC 215065) ON 7/12/89, DURING NORMAL 100% POWER CONDITIONS, OPERATORS WERE COMPLETING A ROUTINE LEVEL FILL OF THE "B" SAFETY INJECTION ACCUMULATOR. ACCUMULATOR TANK "B" LEVEL INDICATING CHANNEL 2LE-935 WAS OUT OF SERVICE, PENDING CORRECTIVE MAINTENANCE. REDUNDANT ACCUMULATOR LEVEL INDICATING CHANNEL 2LE-934 WAS IN OPERATION, ALONG WITH TWO OTHER CHANNELS FROM THE "A" SAFETY INJECTION ACCUMULATOR. DURING THE FILL OPERATION, THE REMAINING OPERABLE "B" TANK LEVEL CHANNEL 2LE-934 BEGAN TO INDICATE SPURIOUSLY (INDICATIONS OF MINOR LEVEL CHANGES RESULTED IN RELATIVELY LARGE PRESSURE CHANGES). THE FILL OPERATION WAS SUSPENDED AND LEVEL CHANNEL 2LE-934 WAS DECLARED INOPERABLE. OPERATORS GAINED LEVEL INDICATION FOR THE "B" ACCUMULATOR TANK BY CROSS-CONNECTING THE VENT AND FILL LINES BETWEEN THE "A" & "B" TANKS. AS THE PRESSURES AND LEVELS EQUALIZED IN BOTH TANKS, LEVEL COULD BE READ FROM THE OPERABLE LEVEL INDICATOR 2LE-939 ON THE "A" TANK. DURING THE EQUALIZATION PROCESS, LEVEL IN THE "A" ACCUMULATOR APPROACHED THE TECH SPEC HIGH LEVEL LIMIT. THE LEVEL WAS ADJUSTED TO MAINTAIN THE PROPER LEVEL BAND. LEVELS WOULD HAVE EQUALIZED ABOVE TECH SPEC HIGH LIMIT HAD THE "A" ACCUMULATOR NOT BEEN ADJUSTED. IT WAS THEREFORE CONCLUDED THAT THE "B" ACCUMULATOR TANK LEVEL HAD EXCEEDED THE HIGH LEVEL LIMIT CITED IN TECH SPEC 15.3.3.A.1.B.

[181] PRAIRIE ISLAND 1 DOCKET 50-282 LER 89-005
AUTOMATIC START OF AN AUXILIARY FEEDWATER PUMP DUE TO FAILURE OF AN UNDERVOLTAGE SENSOR.
EVENT DATE: 052289 REPORT DATE: 062189 NSSS: WE TYPE: PWR
VENDOR: MOTOROLA
RUSS ELECTRIC

(NSIC 214399) ON MAY 22, 1989, BOTH UNITS WERE AT 100%. AT 0216 ALARMS INDICATING TROUBLE WITH THE DC CONTROL POWER TO NONSAFEGUARDS 4160V BUSES 11 AND 12 AND INDICATION OF AN AUTOMATIC START OF NO. 12 MOTOR DRIVEN AUXILIARY FEEDWATER (MDAFW) PUMP WERE RECEIVED IN THE CONTROL ROOM. LOCAL INVESTIGATION BY OPERATIONS PERSONNEL FOUND THAT THE DC CONTROL POWER TO 4160V BUS 11 WAS NORMAL. HOWEVER, THE AUTOMATIC BUS TRANSFER SWITCH (ABT) FOR DC CONTROL POWER TO 4160V BUS 12 HAD TRANSFERRED TO ITS ALTERNATE SOURCE. THIS TRANSFER RESULTED IN A MOMENTARY LOSS OF DC CONTROL POWER TO 4160V BUS 12 WHICH CAUSED THE NO. 12 MDAFW PUMP TO START. CONTROL ROOM OPERATORS STOPPED NO. 12 MDAFW PUMP AT 0219. THIS WAS A NON-ESF ACTUATION OF ESF EQUIPMENT. INVESTIGATION BY PLANT ELECTRICIANS FOUND THAT THE ABT DC UNDERVOLTAGE SENSOR HAD FAILED TO THE DE-ENERGIZED CONDITION, WHICH IN TURN CAUSED THE ABT TO TRANSFER TO THE ALTERNATE SOURCE. FURTHER INVESTIGATION FOUND THAT A MOTOROLA 2N4927 TRANSISTOR INSIDE THE DC UNDERVOLTAGE SENSOR HAD FAILED. A REPLACEMENT DC UNDERVOLTAGE SENSOR WAS OBTAINED FROM THE MANUFACTURER. TESTED AND INSTALLED ON MAY 25, 1989.

[182] PRAIRIE ISLAND 1 DOCKET 50-282 LER 89-007
LACK OF CIRCUIT PROTECTION COORDINATION FOR ASSOCIATED CIRCUITS ON TWO APPENDIX R RELATED MOTOR CONTROL CENTERS.
EVENT DATE: 060689 REPORT DATE: 081889 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: PRAIRIE ISLAND 2 (PWR)

(NSIC 215062) ON JUNE 6, 1989, IT WAS DETERMINED THAT TWO CIRCUIT BREAKERS ON APPENDIX R RELATED MOTOR CONTROL CENTERS LACKED ADEQUATE COORDINATION FOR CIRCUIT PROTECTION. "ADEQUATE COORDINATION" ENSURES THAT IN THE EVENT OF FIRE, THE LOADS REQUIRED FOR SAFE SHUTDOWN WILL BE UNAFFECTED BY THOSE NOT REQUIRED FOR SAFE SHUTDOWN. THAT IS, THE LOADS NOT REQUIRED FOR SAFE SHUTDOWN MUST HAVE PROTECTED CIRCUITS SO THAT IN THE EVENT OF A FAULT DURING A FIRE, THE FAULT WILL BE CLEARED

BY THEIR ... BREAKER AND NOT BY AN UPSTREAM BREAKER THAT FEEDS SAFE SHUTDOWN LOADS. UPON DISCOVERY, FIRE WATCHES WERE ESTABLISHED AS COMPENSATORY MEASURES. THE BREAKERS WERE REPLACED. THE PROCUREMENT AND THE RECEIVING PROCESSES HAVE BEEN REVISED TO PREVENT RECURRENCE.

[183] PRAIRIE ISLAND 1 DOCKET 50-282 LER 89-009
AUTOMATIC CONTROL ROOM ISOLATION AND START OF CONTROL ROOM CLEANUP FAN DUE TO FAILURE OF A CHLORINE GAS MONITOR.
EVENT DATE: 071889 REPORT DATE: 081789 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: PRAIRIE ISLAND 2 (PWR)
VENDOR: M D A SCIENTIFIC, INC.

(NSIC 215023) ON JULY 18, 1989, BOTH UNITS WERE OPERATING AT 100% POWER. AT 1932 AN ALARM INDICATING CHLORINE GAS PRESENT IN THE CONTROL ROOM VENTILATION INTAKE DUCT WAS RECEIVED, AND NO. 122 CONTROL ROOM CLEANUP FAN AUTOMATICALLY STARTED AND OUTSIDE AIR TO THE CONTROL ROOM WAS AUTOMATICALLY ISOLATED. INVESTIGATION SHOWED THAT NO. 122 CONTROL ROOM CHLORINE MONITOR HAD GENERATED A FALSE HIGH CHLORINE SIGNAL. INVESTIGATIVE EFFORTS FAILED TO REVEAL THE CAUSE OF THE FALSE SIGNAL. ALL FOUR CHLORINE MONITORS WERE OPERABLE THROUGHOUT THE EVENT. THE CONTROL ROOM VENTILATION SYSTEM ENTERED ITS SAFEGUARDS MODE OF OPERATION AS DESIGNED.

[184] PRAIRIE ISLAND 1 DOCKET 50-282 LER 89-010
REACTOR TRIP RESULTING FROM LOSS OF ONE REACTOR COOLANT PUMP DUE TO PERSONNEL ERROR.
EVENT DATE: 072189 REPORT DATE: 081889 NSSS: WE TYPE: PWR

(NSIC 215124) ON JULY 21, 1989 UNIT 1 WAS AT 100% POWER. DURING THE AFTERNOON, A "HOT LACQUER" SMELL WAS NOTICED COMING FROM 4160V BUS 11. 4160V BUS 11 SUPPLIES NO. 11 REACTOR COOLANT PUMP AND NO. 11 FEEDWATER PUMP. THE PROBLEM WAS INVESTIGATED AND DETERMINED TO BE OF NO IMMEDIATE CONCERN BUT WORTHY OF INCREASED AWARENESS. AN "OPERATIONS NOTE" WAS ISSUED TO ALERT OPERATORS OF THE PROBLEM. DURING A SUBSEQUENT INVESTIGATION FOR THE SOURCE OF THE SMELL, AN OPERATOR PULLED OPEN THE POTENTIAL FUSE DRAWER FOR 4160V BUS 11, CAUSING UNDERVOLTAGE RELAYS TO TRIP. AFTER A 5 SECOND TIME DELAY TIMED OUT, THE BREAKER FOR NO. 11 REACTOR COOLANT PUMP TRIPPED AND THE REACTOR TRIPPED AT 2345 ON JULY 21, 1989 DUE TO SINGLE LOOP LOSS OF FLOW REACTOR TRIP SIGNAL. THE UNIT WAS RETURNED TO SERVICE AT 2204 ON JULY 22, 1989. 4160V BUS DOORS HAVE BEEN LABELED, CAUTIONING PERSONNEL OF THE CONSEQUENCES OF OPENING THE POTENTIAL FUSE DRAWERS. POTENTIAL FUSE DRAWER FRONTS WILL ALSO BE LABELED.

[185] QUAD CITIES 2 DOCKET 50-265 LER 88-022
LOSS OF CHIMNEY MONITORS WHEN POWER SUPPLY DEENERGIZED FOR MAINTENANCE DUE TO PROCEDURE INADEQUACY.
EVENT DATE: 060188 REPORT DATE: 081288 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 214304) ON 5/31/88, AT 1123 HOURS, UNIT 1 WAS IN THE RUN MODE AT 100% POWER AND UNIT 2 WAS IN THE SHUTDOWN MODE. AT THIS TIME, MOTOR CONTROL CENTER (MCC) 27-1 WAS INTENTIONALLY DE-ENERGIZED TO ALLOW MAINTENANCE TO BE PERFORMED. ALTHOUGH POWER WAS SUBSEQUENTLY RESTORED TO MCC 27-1, NOBLE GAS MONITORING EQUIPMENT WAS UNKNOWINGLY LOST FOR A PERIOD OF TIME IN EXCESS OF TECH SPEC TABLE 3.2-6 REQUIREMENTS. SAMPLING IS REQUIRED EVERY 8 HOURS WHEN THE NOBLE GAS MONITORS ARE UNAVAILABLE. THE NOBLE GAS MONITOR WAS NOT FUNCTIONING FOR A TOTAL OF 14 HOURS 11 MINUTES. THIS OPERATION OUTSIDE OF THE TECH SPECS WAS REALIZED ON 7/18/88, DURING REVIEW OF THE EVENT AND TESTING PERFORMED AS A RESULT OF THAT REVIEW. THE CAUSE OF THIS EVENT WAS PROCEDURE INADEQUACIES BECAUSE THE REFERENCE DOCUMENTS CONSULTED FOR THE ORIGINAL OUT-OF-SERVICE DID NOT CLEARLY IDENTIFY THE AFFECTED EQUIPMENT. IN ADDITION, THE SAMPLE SYSTEM LOW FLOW ALARM THAT INITIATED

IN THE CONTROL ROOM WHEN POWER WAS LOST WAS NOT RECOGNIZED AS SIGNIFICANT BECAUSE OF A DEFICIENCY IN ADMINISTRATIVE PROCEDURE QAP 300-13 (OFF NORMAL INSTRUMENT STATUS). CORRECTIVE ACTIONS INCLUDE PROCEDURE DEVELOPMENT AND REVISIONS, AND EQUIPMENT MODIFICATION TO REROUTE THE POWER SUPPLY TO THE SAMPLING SYSTEM TO ENSURE IMPROVED RELIABILITY. THIS REPORT IS PROVIDED TO COMPLY WITH 10CFR50.73(A)(2)(I).

[186] RANCHO SECO DOCKET 50-312 LER 88-005 REV 01
UPDATE ON LETDOWN SYSTEM RELIEF VALVE LIFTS DUE TO STEAM/WATER HYDRAULIC TRANSIENT.
EVENT DATE: 031588 REPORT DATE: 082389 NSSS: BW TYPE: PWR
VENDOR: DRESSER INDUSTRIAL VALVE & INST DIV

(NSIC 215096) WITH THE PLANT SUBCRITICAL AND IN HOT SHUTDOWN, STEAM/WATER HYDRAULIC TRANSIENTS CAUSED RELIEF VALVES PSV-22021 (MARCH 15, 1988) AND PSV-22024 (MARCH 22, 1988) TO LIFT. EACH LIFT RESULTED IN A DISCHARGE OF WATER FROM THE LETDOWN SYSTEM TO THE REACTOR BUILDING SUMP. THE DIRECT CAUSE OF THE EVENTS WAS A PRESSURE SPIKE DUE TO ACCELERATION OF A SLUG OF WATER INTO THE RELIEF VALVE. THE ACCELERATION RESULTED FROM THE RAPID EXPANSION OF HIGH ENERGY WATER FLASHING TO STEAM IN A LOW PRESSURE AREA. THIS RAPID EXPANSION CAUSED TRAPPED WATER TO UNSEAT THE SUBJECT RELIEF VALVES. ON MARCH 15 (EVENT #1), THE WATER IN THE LINE BETWEEN SFV-22005 AND RELIEF VALVE PSV-22021 CAUSED PSV-22021 TO LIFT. ON MARCH 22 (EVENT #2), THE WATER IN THE LINE BETWEEN SFV-22025 AND RELIEF VALVE PSV-22024 CAUSED PSV-22024 TO LIFT. AFTER EVENT #1 THE DISTRICT PERFORMED A TECHNICAL ANALYSIS TO DETERMINE THE CAUSE. PROCEDURAL CONTROLS WERE IMPLEMENTED TO PREVENT RECURRENCE OF THE EVENT. AFTER EVENT #2 THE DISTRICT FORMED A TEAM UNDER THE DIRECTION OF SYSTEM ENGINEERING DETERMINE AND IMPLEMENT EFFECTIVE IMMEDIATE AND LONG TERM CORRECTIVE ACTIONS. AS A LONG-TERM CORRECTIVE ACTION THE DISTRICT COMPLETED AN ENGINEERING EVALUATION OF THE LETDOWN SYSTEM. THE DISTRICT SUBMITS THIS VOLUNTARY REPORT TO INFORM THE NRC AND THE INDUSTRY OF AN ADVERSE PHENOMENON IN THE LETDOWN SYSTEM AND THE DISTRICT'S CORRECTIVE ACTIONS

[187] RANCHO SECO DOCKET 50-312 LER 89-001 REV 01
UPDATE ON TECH SPEC REQUIRED SHUTDOWN DUE TO INOPERABLE AUX FEEDWATER SYSTEM.
EVENT DATE: 013189 REPORT DATE: 072489 NSSS: BW TYPE: PWR
VENDOR: BAILEY CONTROLS CO.
GIMPEL MACHINE WORKS INC.
TERRY STEAM TURBINE COMPANY
VELAN VALVE CORP.
WOODWARD GOVERNOR COMPANY

(NSIC 214882) AT 1725 HOURS ON 1/31/89, A POST MAINTENANCE TEST OF AUXILIARY FEEDWATER (AFW) PUMP P-318 WAS IN PROGRESS. DURING THE PERFORMANCE OF THIS TEST, THE PUMP REACHED AN OVERSPEED CONDITION RESULTING IN AN OVERPRESSURIZATION OF BOTH AFW TRAINS. AT 2156 HRS, BOTH AFW TRAINS WERE DECLARED INOPERABLE. CONTROL ROOM OPERATORS BEGAN A PLANT SHUTDOWN AT 2212 HOURS AND TRANSITION TO DECAY HEAT COOLING. THE REACTOR WAS IN HOT STANDBY AT 0146 HOURS ON 2/1/89. AT 0155 HRS, THE REACTOR WAS MANUALLY TRIPPED TO ASSURE THAT A GREATER THAN 1% SHUTDOWN MARGIN WAS ACHIEVED WITHIN THE TECHNICAL SPECIFICATION (TS) IMPOSED 4 HR TIME LIMIT. AT 1554 HRS, CONTROL ROOM OPERATORS ESTABLISHED DECAY HEAT COOLING 1 HR AND 59 MIN AFTER THE SPECIFIED TS TIME REQUIREMENT. THE MANDATORY SHUTDOWN OF THE PLANT AS REQUIRED BY TS IS REPORTABLE PURSUANT TO 10 CFR 50.73(A)(2)(I)(A). THE FAILURE TO ESTABLISH DECAY HEAT COOLING WITHIN 12 HRS SUBSEQUENT TO THE REACTOR TRIP AS REQUIRED BY TS IS REPORTABLE PURSUANT TO 10 CFR 50.73(A)(2)(I)(B). A SITE TEAM CONDUCTED A ROOT CAUSE INVESTIGATION OF THE PUMP OVERSPEED AND ASSOCIATED OVERPRESSURIZATION OF THE AFW SYSTEM. THE INVESTIGATION DISCLOSED THAT THE TURBINE GOVERNOR FAILED TO CONTROL TURBINE SPEED AND THE MECHANICAL OVERSPEED TRIP MECHANISM FAILED TO CLOSE THE TURBINE STEAM INLET VALVE.

[188] RANCHO SECO DOCKET 50-312 LER 89-005 REV 01
UPDATE ON POTENTIAL INOPERABILITY OF FLOW TRANSMITTERS DUE TO AN UNQUALIFIED CONFIGURATION WITH PLASTIC CONDUIT OPENING PLUGS.
EVENT DATE: 021889 REPORT DATE: 080789 NSSS: BW TYPE: PWR
VENDOR: ROSEMOUNT, INC.

(NSIC 214988) ON 2/18/89, WHILE THE PLANT WAS IN COLD SHUTDOWN, THE DISTRICT DISCOVERED THAT 6 ESSENTIAL FLOW TRANSMITTERS IN THE AUX. FEEDWATER (AFW) SYSTEM HAD PLASTIC PLUGS INSTALLED IN THEIR SPARE CONDUIT CONNECTION OPENING. IN THIS CONFIGURATION THE ENVIRONMENTAL QUALIFICATION DOCUMENTATION IS NOT VALID. A CORRECTIVE ACTION PLAN WAS PREPARED ON 2/22/89, WHICH REQUIRED REPLACEMENT OF PLASTIC PLUGS WITH METAL PIPE PLUGS. THIS WAS ACCOMPLISHED PRIOR TO THE 3/9/89 RESTART OF RANCHO SECO. THE DISTRICT CHECKED ALL OTHER SAFETY RELATED TRANSMITTERS FOR PLASTIC SHIPPING PLUGS, BUT DID NOT FIND ADDITIONAL ONES. A REPORTABILITY REVIEW ON 7/18/89, DETERMINED THE ENGINEERING CHANGE NOTICE (ECN) THAT INSTALLED 4 OF THE 6 FLOW TRANSMITTERS WAS INADEQUATE, AS IT EXCLUDED A VENDOR DRAWING WHICH CONTAINED INFORMATION VITAL FOR ENVIRONMENTAL QUALIFICATION OF THE TRANSMITTERS. FOR 1 OF THE 2 ADDITIONAL FLOW TRANSMITTERS, THE CHANGE FROM ORIGINALLY INSTALLED METAL CONDUIT PLUGS TO PLASTIC SHIPPING PLUGS OCCURRED UNDER A WORK REQUEST (WR) FOR CALIBRATION WHICH REQUIRED THE TRANSMITTER TO BE REPLACED. THE TIME OF THE CHANGE OF PLUGS IN THE REMAINING TRANSMITTER HAS NOT BEEN DETERMINED. INSTALLATION INSTRUCTIONS SUPPLIED WITH THE TRANSMITTERS ALSO CONTAINED INFORMATION WITH REGARD TO THE REQUIRED CONDUIT PLUG, BUT WAS NOT UTILIZED.

[189] RANCHO SECO DOCKET 50-312 LER 89-008
MISSED GASEOUS EFFLUENT GRAB SAMPLE DUE TO PERSONNEL ERROR.
EVENT DATE: 071989 REPORT DATE: 081789 NSSS: BW TYPE: PWR

(NSIC 215084) ON 7/17/89, OPERATIONS DECLARED THE AUX. BLDG STACK (ABS) WIDE RANGE GAS MONITOR (WRGM) R-15045 INOPERABLE WHEN IT FAILED SURVEILLANCE PROCEDURE SP.450B. CHEMISTRY INITIATED ALTERNATE SAMPLING OF THE ABS AS REQUIRED BY TECH SPEC 3.16. ON 7/19/89, OPERATIONS DECLARED REACTOR BLDG DUCT (RBD) WRGM R-15044 INOPERABLE WHILE I&C TECHNICIANS PERFORMED PROCEDURE SP.447. CHEMISTRY INITIATED ALTERNATE SAMPLING OF THE RBD AS REQUIRED BY TECH SPEC 3.16. AT APPROX. 1600 HRS ON 7/19/89, TESTING WAS COMPLETED AND RBD RADIATION MONITOR R-15044 WAS RETURNED TO SERVICE. A DISTRICT CHEMISTRY TECHNICIAN PROCEEDED TO THE LOCAL SAMPLE AREA TO SECURE THE ALTERNATE SAMPLING EQUIPMENT FOR THE RBD. THE TECHNICIAN INADVERTENTLY DISCONNECTED THE ALTERNATE SAMPLING EQUIPMENT FOR THE ABS, RESULTING IN THE REQUIRED ALTERNATE SAMPLING NOT BEING PERFORMED. FURTHER, WHEN THE SAME CHEMISTRY TECHICIAN ATTEMPTED TO OBTAIN THE REQUIRED 12-HR NOBLE GAS GRAB SAMPLE FROM THE ABS, HE ACTUALLY OBTAINED THE GRAB SAMPLE FROM THE RBD. AT APPROX. 0800 HOURS ON 7/20/89, A SECOND CHEMISTRY TECHNICIAN DISCOVERED THAT THE REQUIRED ALTERNATE SAMPLING EQUIPMENT WAS CONNECTED TO THE OPERABLE RBD AND NOT THE INOPERABLE ABS. THE TECHNICIAN VERIFIED THAT THE ABS PORTABLE SAMPLER WAS OUT OF SERVICE AND TOOK APPROPRIATE ACTIONS TO REINSTATE ALTERNATE SAMPLING TO THE ABS.

[190] RANCHO SECO DOCKET 50-312 LER 89-009
DECAY HEAT REMOVAL SYSTEM INOPERABLE IN EXCESS OF TECH SPEC LIMIT DUE TO PERSONNEL ERROR.
EVENT DATE: 072589 REPORT DATE: 082389 NSSS: BW TYPE: PWR

(NSIC 215129) ON JULY 25, 1989, WITH THE PLANT IN COLD SHUTDOWN, DISTRICT LICENSED CONTROL ROOM OPERATORS PERFORMED SURVEILLANCE PROCEDURES SP.30B "QUARTERLY DECAY HEAT REMOVAL B LOOP VALVE INSPECTION AND TESTS" AND SP.32B "QUARTERLY DECAY HEAT REMOVAL SYSTEM (DHS) LOOP "B" SURVEILLANCE P-261B." THE "B" DHS WAS . OPERATION, THE "A" DHS WAS OPERABLE AND IN NORMAL STANDBY. THE B2 DIESEL GENERATOR WAS INOPERABLE, OUT OF SERVICE FOR PLANNED MAINTENANCE. DECAY

HEAT REMOVAL WAS STOPPED FOR 1 HOUR, 26 MINUTES (FROM 2232 HOURS TO 2358 HOURS) WHILE SP.30B AND SP.32B WERE PERFORMED AND DHS PURIFICATION WAS ALIGNED WITH THE PURIFICATION AND LETDOWN SYSTEM. DURING THE PERIOD THE DHS WAS INOPERABLE, INCORE TEMPERATURE INCREASED APPROXIMATELY 3.8 DEGREES F TO 112.8 DEGREES F. NO OPERATIONS WERE CONDUCTED WHICH DILUTED THE CONCENTRATION OF BORON IN THE REACTOR COOLANT SYSTEM, AND A SUBCOOLING MARGIN OF AT LEAST 150 DEGREES F WAS MAINTAINED. THE INOPERABILITY OF THE DHS FOR A PERIOD IN EXCESS OF THE 1-HOUR TECHNICAL SPECIFICATION LIMIT CONSTITUTES A CONDITION REPORTABLE IN ACCORDANCE WITH 10 CFR 50.73(A)(2)(I)(B).

[191] RIVERBEND 1 DOCKET 50-458 LER 88-011 REV 03
UPDATE ON TECH SPEC VIOLATION DUE TO A FAILURE TO RECOGNIZE THE AS-FOUND CONDITION OF A CONTAINMENT ISOLATION VALVE AS INOPERABLE.
EVENT DATE: 050388 REPORT DATE: 072889 NSSS: GE TYPE: BWR
VENDOR: VELAN VALVE CORP.

(NSIC 214864) ON 5/3/88 AT APPROXIMATELY 2230, WITH THE UNIT IN OPERATIONAL CONDITION 1 (APPROXIMATELY 100 PERCENT POWER), CONTAINMENT ISOLATION VALVE 1E51*MOVF078 WAS DISCOVERED TO BE INOPERABLE. THE TORQUE ARM KEY HAD FALLEN OUT OF ITS KEYWAY AS DOCUMENTED ON A MAINTENANCE WORK ORDER REQUEST DATED 3/28/88. THE PERSONNEL THAT INITIALLY DISCOVERED THIS CONDITION WERE PERFORMING A VALVE LINE-UP VERIFICATION PROCEDURE AND DID NOT RECOGNIZE THIS CONDITION AS AFFECTING VALVE OPERABILITY. THIS CONDITION IS REPORTED PURSUANT TO 10CFR50.73 (A)(2)(1), AS A CONDITION PROHIBITED BY RIVER BEND STATION TECHNICAL SPECIFICATION 3.6.4 BECAUSE THE VALVE WAS INOPERABLE FOR A TIME GREATER THAN ALLOWED BY TECHNICAL SPECIFICATIONS. THE AFFECTED PENETRATION (FOR THE REACTOR CORE ISOLATION COOLING TURBINE EXHAUST VACUUM BREAKER LINE) WAS DECLARED INOPERABLE ON 5/3/88, THE LINE WAS ISOLATED, AND VALVE 1E51*MOVF078 WAS REPAIRED AND RESTORED TO OPERABLE STATUS. ALL OPERATIONS PERSONNEL WILL BE INSTRUCTED ON THIS OCCURRENCE VIA MEMORANDUM. THIS EVENT WILL BE ADDRESSED IN A FUTURE OPERATOR REQUALIFICATION TRAINING MODULE. NO EVENTS OCCURRED WHICH WOULD HAVE REQUIRED CONTAINMENT ISOLATION AND AUTOMATIC CONTAINMENT ISOLATION WAS STILL AVAILABLE VIA ISOLATION VALVE 1E51*MOVF077.

[192] RIVERBEND 1 DOCKET 50-458 LER 88-018 REV 02
UPDATE ON REACTOR SCRAM DUE TO MAIN GENERATOR EXCITER BRUSH FAILURE.
EVENT DATE: 082588 REPORT DATE: 073189 NSSS: GE TYPE: BWR
VENDOR: GEN ELEC CO (STEAM TURB/ENGRD PROD)
INGERSOLL-RAND CO.
ROSEMOUNT, INC.
WESTINGHOUSE ELECTRIC CORP.

(NSIC 214941) AT 1232 ON 8/25/88 WITH THE UNIT AT 100 PERCENT POWER (OPERATIONAL CONDITION 1), THE REACTOR AUTOMATICALLY SCRAMMED DUE TO A TURBINE CONTROL VALVE FAST CLOSURE CAUSED BY A LOSS OF MAIN GENERATOR FIELD EXCITATION RESULTING IN AUTOMATIC MAIN GENERATOR AND TURBINE TRIPS. IMMEDIATELY FOLLOWING THE SCRAM, REACTOR PRESSURE SPIKED TO A PEAK BETWEEN 1100 AND 1117 PSIG CAUSING THE FIVE LOW-LOW SET SAFETY RELIEF VALVES TO CYCLE PER DESIGN. THE TURBINE BYPASS VALVES OPENED AS REQUIRED AND THE REACTOR RECIRCULATION PUMPS TRANSFERRED TO SLOW SPEED PER DESIGN. REACTOR WATER LEVEL INITIALLY DECREASED TO +4 INCHES AS INDICATED BY THE WIDE RANGE INSTRUMENTS DUE TO THE REACTOR PRESSURE SPIKE. THE HIGH PRESSURE CORE SPRAY (HPCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEMS INJECTED AS A RESULT OF A SPURIOUS LOW REACTOR WATER LEVEL 2 SIGNAL CAUSED BY A HYDRAULIC PERTURBATION IN THE REACTOR WATER LEVEL INSTRUMENT REFERENCE LINES. AS A RESULT OF THE FEEDWATER FLOW CONTINUING (DUE TO THE "A" FEEDWATER CONTROL VALVE BEING IN THE MANUAL MODE AT 50 PERCENT OPEN) IN CONJUNCTION WITH THE HPCS AND RCIC INJECTIONS, REACTOR WATER LEVEL RAPIDLY INCREASED TO LEVEL 8 CAUSING THE HPCS INJECTION VALVE AND THE RCIC STEAM SUPPLY VALVE TO CLOSE AND THE REACTOR FEEDWATER PUMPS TO TRIP PER DESIGN. THERE WAS NO SIGNIFICANT ADVERSE IMPACT ON

THE SAFE OPERATION OF THE PLANT OR TO THE HEALTH AND SAFETY OF THE PUBLIC AS A RESULT OF THIS EVENT.

[193] RIVERBEND 1 DOCKET 50-458 LER 89-026 REV 01
UPDATE ON HIGH PIPING VIBRATION OUTSIDE VALVE SSEMBLIES DESIGN BASIS.
EVENT DATE: 051289 REPORT DATE: 063089 NSSS: GE TYPE: BWR

(NSIC 214660) AT 1000 ON 5/12/89 WITH THE UNIT IN OPERATIONAL CONDITION 5 (REFUELING), GSU DETERMINED THAT THREE SAFETY-RELATED MOTOR OPERATED VALVES (TWO IN THE RESIDUAL HEAT REMOVAL (RHR) SYSTEM AND ONE IN THE LOW PRESSURE CORE SPRAY SYSTEM) HAD OPERATED IN A CONDITION THAT WAS OUTSIDE THEIR DESIGN BASIS DUE TO HIGH STEADY-STATE PIPING VIBRATION LEVELS. SINCE THE ORIGINAL DESIGN BASIS REQUIRES DYNAMIC QUALIFICATION OF THESE VALVES FOR 100 DAYS POST-ACCIDENT AND THEIR SUBSEQUENT OPERATION WOULD HAVE RESULTED IN THE VALVE ASSEMBLIES EXCEEDING THEIR FATIGUE LIFE, THIS CONDITION IS OUTSIDE THE DESIGN BASIS. THE HIGH STEADY-STATE VIBRATION LEVELS WERE EXPERIENCED WHEN FLOW WAS DIRECTED THROUGH THE RHR TEST RETURN LINES AND WERE CAUSED BY THE SIZE AND LOCATION OF RESTRICTING ORIFICES. GSU HAD NOTED THE HIGH STEADY-STATE VIBRATIONS DURING PLANT START-UP TESTING AND EVALUATED CONTINUED OPERATION BASED ON PIPE STRESS REQUIREMENTS. HOWEVER, THE ORIGINAL EVALUATION DID NOT CONSIDER THE EFFECT ON THE DYNAMIC QUALIFICATION OF THE VALVE ASSEMBLIES. THIS EFFECT WAS NOT REALIZED UNTIL RECENTLY WHEN ADDITIONAL MEASUREMENTS WERE TAKEN DURING TESTING TO VERIFY CORRECTION OF THE VIBRATION. THE HIGH STEADY-STATE VIBRATION LEVELS HAVE BEEN CORRECTED BY RESIZING AND RELOCATING THE RESTRICTING ORIFICES. THERE WAS NO SIGNIFICANT IMPACT ON THE HEALTH AND SAFETY OF THE PUBLIC AS A RESULT OF THIS CONDITION.

[194] RIVERBEND 1 DOCKET 50-458 LER 89-030
MISPOSITIONED PRESSURE TRANSMITTER ISOLATION VALVE FOUND MISALIGNED CAUSING INABILITY TO SENSE DRYWELL PRESSURE, A CONDITION PROHIBITED BY TECH SPEC 3.0.4.
EVENT DATE: 061789 REPORT DATE: 071789 NSSS: GE TYPE: BWR

(NSIC 214831) ON 6/17/89 WITH THE UNIT IN OPERATIONAL CONDITION 4 (COLD SHUTDOWN), WHILE PERFORMING A SAFETY SYSTEM VALVE LINE-UP, A PRESSURE TRANSMITTER ROOT VALVE FOR THE PENETRATION VALVE LEAKAGE CONTROL SYSTEM (PVLCS) WAS FOUND TO BE CLOSED. THIS CAUSED ONE DIVISION OF PVLCS TO BE INOPERABLE. INVESTIGATION DETERMINED THAT THIS VALVE HAD PROBABLY BEEN MISPOSITIONED SINCE THE CONCLUSION OF THE PRIMARY CONTAINMENT INTEGRATED LEAK RATE TEST ON 5/30/89. AT 2230 ON 6/15/89, THE PLANT ENTERED A MODE OF OPERATION IN WHICH BOTH DIVISIONS OF PVLCS ARE REQUIRED TO BE OPERABLE. THIS WAS A VIOLATION OF TECHNICAL SPECIFICATION 3.0.4. THIS CONDITION WOULD NOT HAVE PREVENTED THE PVLCS SYSTEM FROM BEING INITIATED. HOWEVER, THIS MAY HAVE PREVENTED THIS DIVISION FROM SUPPLYING ADEQUATE SEALING PRESSURE TO THE OPERABLE BRANCH LINES IN THE EVENT A BRANCH LINE EXPERIENCED A LOW PRESSURE CONDITION. THE PROBABLE CAUSE FOR THIS MISALIGNMENT WAS A PERSONNEL ERROR IN CONJUNCTION WITH A MISAPPLICATION OF A TEST TAGGING PROCEDURE. CORRECTIVE ACTION WAS TAKEN ON 6/17/89 TO OPEN THE ROOT VALVE AND RESTORE THE PVLCS DIVISION TO OPERABLE STATUS. DURING THE SAFETY SYSTEM VALVE LINE-UP OF 6/17/89, NO OTHER SAFETY RELATED VALVE MISPOSITIONINGS WERE IDENTIFIED. HENCE, THIS SHOULD BE CONSIDERED AN ISOLATED INCIDENT.

[195] RIVERBEND 1 DOCKET 50-458 LER 89-031
FAILURE TO PERFORM STP PRIOR TO EXCEEDING 25% REACTOR POWER IS IN VIOLATION OF TECH SPEC AND IS DUE TO HUMAN ERROR.
EVENT DATE: 062489 REPORT DATE: 072489 NSSS: GE TYPE: BWR

(NSIC 214904) ON 6/24/89 AT 1715 WITH THE UNIT AT APPROXIMATELY 39 PERCENT POWER IN OPERATIONAL CONDITION 1 (POWER OPERATION) DURING INITIAL PLANT STARTUP FOLLOWING THE SECOND REFUELING OUTAGE, GSU DISCOVERED THAT SURVEILLANCE TEST

PROCEDURE (STP)-509-0101, "TURBINE BYPASS VALVE OPERABILITY," WAS NOT PERFORMED PRIOR TO EXCEEDING 25% POWER AS REQUIRED BY TECHNICAL SPECIFICATION (TS) SURVEILLANCE REQUIREMENT 4.7.9.A. FAILURE TO PERFORM THIS SURVEILLANCE PRIOR TO EXCEEDING 25 PERCENT REACTOR POWER CONSTITUTED A VIOLATION OF TS 4.0.4. OPERABILITY OF THE TURBINE BYPASS VALVES (*PCV*) WAS PREVIOUSLY OBSERVED DURING TESTING OF THE SAFETY RELIEF VALVES (*RV*) AT 920 PSIG REACTOR PRESSURE DURING THE REACTOR STARTUP. HOWEVER, THIS WAS NOT DOCUMENTED WITH STP-509-0101. THE STP WAS SUCCESSFULLY PERFORMED AT 35 PERCENT REACTOR POWER. THE CONTROL OPERATING FOREMAN (COF) FAILED TO FOLLOW TECHNICAL SPECIFICATION 3/4.7.9 AND VIOLATED TECHNICAL SPECIFICATION 4.0.4. THIS PERSONNEL ERROR WAS CAUSED BY AN INTERPRETATIONAL MISTAKE BY THE COF. AS CORRECTIVE ACTION TO HELP PREVENT RECURRENCE, THE CONTROL OPERATING FOREMAN WAS COUNSELLED ON THIS EVENT. THERE WAS NO IMPACT ON THE SAFE OPERATION OF THE PLANT OR TO THE HEALTH AND SAFETY OF THE PUBLIC AS A RESULT OF THIS EVENT SINCE SUBSEQUENT PERFORMANCE OF THIS SURVEILLANCE DEMONSTRATED THAT THE MAIN TURBINE BYPASS SYSTEM WAS OPERABLE THROUGHOUT THIS TIME.

[196] ROBINSON 2 DOCKET 50-261 LER 88-022 REV 01
UPDATE ON CORRECTED CONTAINMENT FLOOD LEVEL RESULTS IN SUBMERGENCE OF ADDITIONAL COMPONENTS.
EVENT DATE: 100788 REPORT DATE: 080289 NSSS: WE TYPE: PWR
VENDOR: ANACONDA WIRE AND CABLE CO.
ASCO VALVES
BRAND REX CO.
CONTINENTAL WIRE & CABLE CORP.
EATON METAL PRODUCTS CO.
SAMUEL & MOORE COMPANY

(NSIC 214971) OCTOBER 6, 1988 THE LICENSEE DISCOVERED AN ERROR IN THE CONTAINMENT VESSEL (CV) FLOOD LEVEL CALCULATION USED FOR THE 90-DAY RESPONSE TO INFORMATION BULLETIN NO. 79-01B. THE CORRECTED FLOOD LEVEL WOULD BE APPROXIMATELY THREE FEET HIGHER THAN ORIGINALLY STATED THE LICENSEE JUSTIFIED CONTINUED REACTOR OPERATION BY ENGINEERING EVALUATION WHICH AT ENVIRONMENTALLY QUALIFIED EQUIPMENT SUBJECT TO IMMERSION WITH THE REVISED CV FLOOD LEVEL WOULD MEET PLANT TECHNICAL SPECIFICATIONS OPERABILITY REQUIREMENTS. AN INADEQUATE METHODOLOGY FOR PERFORMING THE ORIGINAL FLOOD LEVEL CALCULATION HAD RESULTED IN THE ERROR. A CONTINUING LICENSEE EVALUATION OF THE PLANT ENVIRONMENTAL QUALIFICATION PROGRAM HAD DISCOVERED THE ERROR. PLANNED CORRECTIVE ACTIONS INCLUDE REVISION OF APPLICABLE ENVIRONMENTAL QUALIFICATION FILES FOR SUBMERGENCE, PLANT MODIFICATION OF A SPECIFIC CV ELECTRICAL PENETRATION FOR REGULATORY GUIDE 1.97 COMPONENTS, RESOLUTION OF CERTAIN ST-ACCIDENT MONITORING COMPONENT OPERABILITY CONCERNS, AND CLARIFICATION OF THE CV FLOW LEVEL AS STATED IN THE RESPONSE TO INFORMATION BULLETIN NO. 79-01B. THIS LER IS SUBMITTED TO IDENTIFY AN APPARENT IMPACT TO COMPLIANCE WITH 10CFR50.49.

[197] SALEM 1 DOCKET 50-272 LER 89-026
CONTROLLED SHUTDOWN PER TECH SPEC 3.3.2.1 ACTION 13; 1A SEC INOPERABLE DUE TO EQUIPMENT FAILURE.
EVENT DATE: 061889 REPORT DATE: 071789 NSSS: WE TYPE: PWR

(NSIC 214843) ON 6/28/89, A CONTROLLED SHUTDOWN FROM MODE 1 (POWER OPERATION) TO MODE 3 (HOT STANDBY) WAS COMPLETED IN ACCORDANCE WITH THE REQUIREMENTS OF TECH SPEC 3.3.2.1 TABLE 3.3-3 ACTION 13. THIS TECH SPEC ACTION STATEMENT WAS ENTERED DUE TO INOPERABILITY OF 1A SEC. 1A SEC HAD BEEN DECLARED INOPERABLE DUE TO AN AUTO TEST FEATURE LOCAL ALARM. THE 1A SEC CHASSIS WAS REPLACED; HOWEVER, THE REPLACEMENT CHASSIS ALSO INDICATED AN AUTO TEST FEATURE ALARM. SINCE ANOTHER SPARE SEC CHASSIS WAS NOT AVAILABLE, THE ORIGINAL FAILED CHASSIS WAS PLACED BACK INTO THE SEC CABINET. INVESTIGATION AS TO THE CAUSE OF ITS FAILURE THEN COMMENCED. TROUBLESHOOTING AND REPAIR OF THE CHASSIS COULD NOT BE COMPLETED

WITHIN THE REQUIRED TIME OF THE ACTION STATEMENT; THEREFORE, THE UNIT WAS BROUGHT TO MODE 3 AS PER THE ACTION STATEMENT. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO AN EQUIPMENT FAILURE. THE 1A SEC CHASSIS COULD NOT BE REPAIRED WITHIN THE TIME CONSTRAINT AS SPECIFIED BY THE TECH SPEC ACTION STATEMENT DUE TO FAILED PC BOARDS AND AN IMPROPERLY SET SEQUENCE TIMING SWITCH. THE FAILED PC BOARDS WERE REPLACED AND THE SEQUENCE TIMING SWITCH WAS RESET. 1A SEC WAS SUBSEQUENTLY TESTED AND FOUND TO BE OPERABLE. THE ACTION STATEMENT WAS EXITED AT 1859 HOURS ON 6/18/89. THE UNIT WAS RETURNED TO SERVICE ON 6/19/89.

[198] SALEM 1 DOCKET 50-272 LER 89-027
REACTOR TRIP ON #13 STEAM GENERATOR LO-LO LEVEL DUE TO AN EQUIPMENT DESIGN CONCERN.
EVENT DATE: 061989 REPORT DATE: 071789 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 2 (PWR)

(NSIC 214844) ON 6/19/89 AT 2100 HOURS, A REACTOR TRIP ON NO. 13 STEAM GENERATOR (S/G) "LOW-LOW LEVEL" OCCURRED. THE NO. 13 MAIN STEAMLINE ISOLATION VALVE, 13MS167, HAD CLOSED. PRIOR TO THE EVENT, REACTOR POWER WAS BEING INCREASED 3% PER HOUR. AT THE TIME OF THE EVENT, A POST MAINTENANCE SURVEILLANCE FOR THE 12MS18 MAIN STEAMLINE BYPASS STOP VALVE WAS IN PROGRESS. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO INADEQUATE DESIGN OF THE CONTINUITY CHECK CIRCUITRY FOR THE MS167 VALVES. SURVEILLANCE TESTING OF VALVE 12MS18 WAS IN PROGRESS PRIOR TO THE EVENT. WHEN THE SOLID STATE PROTECTION SYSTEM (SSPS) TRAIN "A" OUTPUT INTERFACE CABINET SWITCH WAS TURNED TO "OPERATE OUTPUT", THE 13MS167 VALVE CLOSED FOLLOWED BY THE TRIP. THE "OPERATE OUTPUT" FUNCTION CAUSES CLOSURE OF THE MS18 VALVE WHILE CHECKING CONTINUITY OF THE MS167 VALVE CLOSURE CIRCUIT. THIS DESIGN CAN CAUSE INADVERTENT CLOSURE OF THE MS167 VALVES AS OCCURRED DURING THIS EVENT AND A SIMILAR UNIT 2 EVENT ON 4/11/89 (REF. LER 311/89-008-00). A DESIGN CHANGE HAS BEEN IMPLEMENTED WHICH CORRECTS THE CIRCUIT DESIGN CONCERN BY ADDING A CONTACT WHICH PREVENTS THE 74-3A RELAY FROM RESETTING DURING THE TESTING OF THE MS18 VALVES. THIS CONTACT DOES NOT PREVENT THE MS167 VALVES FROM FUNCTIONING IN THE EVENT OF A VALID MAIN STEAM ISOLATION SIGNAL.

[199] SALEM 2 DOCKET 50-311 LER 89-011 REV 01
UPDATE ON TECH SPEC SURVEILLANCE 4.3.2.1.1 HISTORICAL NON-COMPLIANCE DUE TO INADEQUATE DESIGN REVIEW.
EVENT DATE: 052289 REPORT DATE: 080489 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SALEM 1 (PWR)

(NSIC 214987) ON 5/22/89, IT WAS DETERMINED THAT TECH SPEC 4.3.2.1.1 HAD NOT BEEN FULLY COMPLIED WITH. THE UFSAR STATES THAT EQUIPMENT WHICH CANNOT BE ACTUATED AT POWER WILL BE CHECKED FOR CIRCUIT CONTINUITY UP THROUGH THE ACTUATING DEVICE. THIS REQUIREMENT APPLIES TO THE MS167 VALVE (MAIN STEAM ISOLATION VALVE) ACTUATION CIRCUITRY IN THE SOLID STATE PROTECTION SYSTEM (SSPS). THE TECH SPEC REQUIRES FUNCTIONAL TESTING; HOWEVER, DUE TO THE TEST CIRCUIT DESIGN, USE OF THE PROCEDURE DID NOT ASSURE COMPLETE CIRCUIT CONTINUITY. THE CIRCUIT DESIGN WAS SUCH THAT A CURRENT WAS BEING APPLIED TO PARALLEL COMPONENTS WHICH COULD MASK FAILURE OF THE SOLENOID VALVE COIL BEING TESTED. ADDITIONALLY, THE SOLENOID VALVES, WHICH ACTUATE TO PROVIDE FAST CLOSURE OF THE MS167 VALVES, ARE NOT CHECKED FOR COIL CONTINUITY SINCE THE SOLENOID VALVE CIRCUITRY IS ISOLATED FROM THE TEST CIRCUITRY. THE ROOT CAUSE HAS BEEN ATTRIBUTED TO INADEQUATE DESIGN REVIEW. DESIGN CHANGES, IMPLEMENTED IN 1983 AND 1982 FOR UNITS 2 AND 1 RESPECTIVELY, ADDED A RELAY TO THE ACTUATION CIRCUITRY. THIS RESULTED IN REMOVAL OF THE SOLENOID VALVE FROM THE CONTINUITY TEST CIRCUIT. ALSO, THIS PROBLEM EXISTED BEFORE DUE TO THE EXISTENCE OF ANOTHER RELAY IN PARALLEL TO THE SOLENOID VALVE. PROCEDURAL MODIFICATIONS HAVE BEEN IMPEMENTED THAT MEASURE THE CIRCUIT RESISTANCES THEREBY VERIFYING CIRCUIT CONTINUITY.

[200] SAN ONOFRE 1 DOCKET 50-206 LER 89-014
BATTERY CHARGER CROSS-TRAIN ALIGNMENT PERMITTED BY TECH SPECS FOR UNLIMITED DURATIONS.
EVENT DATE: 040589 REPORT DATE: 0/2689 NSSS: WE TYPE: PWR

(NSIC 214865) ON 1977, A SECOND 125VDC BATTERY AND TWO ASSOCIATED CHARGERS WERE ADDED TO THE UNIT 1 ELECTRICAL SYSTEM. THE DESIGN FOR THE BATTERY AND CHARGERS WAS SIMILAR TO THE EXISTING SYSTEM IN THAT ONE OF THE TWO NEW CHARGERS WAS FED FROM THE OPPOSITE TRAIN POWER SOURCE AND THE OTHER WAS FED FROM ITS RESPECTIVE TRAIN POWER SUPPLY. THE TECHNICAL SPECIFICATION (TS) CHANGE ISSUED FOR THE MODIFICATION POSED NO RESTRICTIONS ON THE USE OF THE CHARGERS (BOTH EXISTING AND NEW) BEING FED FROM THE OPPOSITE TRAIN POWER SOURCE. THEREFORE, UNLIMITED OPERATION OF THE DC BUSES ON THEIR CROSS-TRAIN CHARGERS WAS ALLOWED. IN EARLY 1989, A DESIGN CHANGE WAS COMPLETED TO TRAIN-ALIGN THE BATTERY CHARGERS, THEREBY PREVENTING CROSS-TRAIN OPERATION. SCE NOW BELIEVES THAT THE TS ALLOWANCE OF UNLIMITED USE OF THE CHARGER CROSS-TRAIN ALIGNMENT REPRESENTED AN UNWARRANTED CONDITION WHICH COULD HAVE RESULTED IN REDUCED RELIABILITY OF THE ONSITE EMERGENCY ELECTRICAL SYSTEM DURING ACCIDENT SCENARIOS. THIS CONDITION IS BEING REPORTED AS A VOLUNTARY LER FOR INDUSTRY INFORMATION. THE UNLIMITED OPERATION OF THE DC BUSES ON THEIR CROSS-TRAIN CHARGERS RESULTED FROM A FAILURE TO DEVELOP APPROPRIATE TS WHICH RECOGNIZE THE IMPACT OF THE MODIFICATION ON POTENTIAL ACCIDENT SCENARIOS. THIS INADEQUATE ENGINEERING AND TECHNICAL WORK IS SIMILAR TO THAT DESCRIBED IN SCE'S 10/3/88 SUBMITTAL TO THE NRC.

[201] SAN ONOFRE 1 DOCKET 50-206 LER 89-016
FIRE PROTECTION SPRAY SYSTEM PLUGGED NOZZLES DUE TO BALL DRIP VALVE FAILURE.
EVENT DATE: 070389 REPORT DATE: 080289 NSSS: WE TYPE: PWR

(NSIC 214942) ON 6/29/89 WITH UNIT 1 AT 40% POWER, THE AIR FLOW TEST OF THE LUBE OIL RESERVOIR AND CONDITIONER FIRE PROTECTION SYSTEM (FPS) REVEALED THAT APPROX. 20 OF THE 78 FPS NOZZLES WERE PLUGGED. IT WAS DETERMINED ON 7/3/89 THAT THIS CONDITION COULD HAVE PRECLUDED ADEQUATE FIRE SUPPRESSION CAPABILITY TO CERTAIN AREAS. THE TECHNICAL SPECIFICATIONS (TSS) REQUIRE THAT A CONTINUOUS FIRE WATCH BE ESTABLISHED FOR SUCH A CASE, WHICH EXISTED PRIOR TO THE SURVEILLANCE. SINCE THIS CONDITION WAS UNKNOWN AT THE TIME, A FIRE WATCH HAD NOT BEEN PREVIOUSLY ESTABLISHED, CONSTITUTING A VIOLATION OF TSS. THE NOZZLE PLUGGING WAS DUE TO THE ACCUMULATION OF PIPING CORROSION MATERIAL INSIDE THE FIRE SPRAY HEADER AND NOZZLES, WHICH HAS BEEN ATTRIBUTED TO THE FAILURE OF THE BALL DRIP VALVE ASSOCIATED WITH THE HEADER DELUGE VALVE. THE BALL DRIP VALVE IS DESIGNED TO DRAIN NORMAL, EXPECTED LEAKAGE OF WATER PAST THE DELUGE VALVE TO PRECLUDE THE ACCUMULATION OF WATER IN THE HEADER. THE CAUSE OF THE BALL DRIP VALVE FAILURE HAS BEEN ATTRIBUTED TO THE SLOW BUILDUP OF CORROSION PRODUCTS ON ITS OPERATING MECHANISM. BALL DRIP VALVES HAVE NOT BEEN INCLUDED IN ANY SURVEILLANCES SUCH THAT THEY WOULD BE PERIODICALLY CHECKED. AIR WAS SYSTEMATICALLY BLOWN THROUGH THE LUBE OIL RESERVOIR AND CONDITIONER AREA FPS PIPING AND NOZZLE ATTACHMENT POINTS TO ENSURE ANY CORROSION DEBRIS AND BLOCKAGE WERE COMPLETELY REMOVED. THE BALL DRIP WAS REPLACED.

[202] SAN ONOFRE 1 DOCKET 50-206 LER 89-020
FUSE BLOCKS FOR THE EMERGENCY SIREN TRANSFER SWITCH INSTALLED CONTRARY TO TECH SPECS.
EVENT DATE: 072389 REPORT DATE: 082289 NSSS: WE TYPE: PWR

(NSIC 215105) AT APPROX. 0500 ON 7/23/89, WITH UNIT 1 AT 75% POWER, DURING PERFORMANCE OF THE MONTHLY EMERGENCY SIREN FUSE BLOCK REMOVAL VERIFICATION, IN ACCORDANCE WITH TECH SPEC (TS) 4.1.1, THE FUSE BLOCKS FOR BOTH BREAKER 8-1145 AND BREAKER 8-1293A, WERE FOUND INSTALLED. SINCE THE TS SURVEILLANCE REQUIRES THAT ONE FUSE BE REMOVED, THIS WAS A CONDITION PROHIBITED BY TS. THE FUSE BLOCK TO BREAKER 8-1293A WAS IMMEDIATELY REMOVED. THERE IS NO SAFETY SIGNIFICANCE TO THIS

EVENT SINCE DURING THE TIME THE FUSE BLOCKS FOR BOTH BREAKERS 8-1145 AND 81293A WERE INSTALLED, BREAKER 8-1293A REMAINED OPEN. THE ROOT CAUSE OF THIS EVENT WAS DETERMINED TO BE THE FAILURE TO DOCUMENT THE MANIPULATION AND VERIFICATION OF SAFETY RELATED EQUIPMENT, AS REQUIRED PER PROCEDURES. INVESTIGATION DETERMINED THAT THE FUSE BLOCK TO BREAKER 8-1293A HAD BEEN INAPPROPRIATELY LEFT INSTALLED FOLLOWING MAINTENANCE ON THE FUSE BLOCK FOR BREAKER 8-1145 ON 7/10/89. PRIOR TO THE MAINTENANCE ON 7/10/89, PROPER DOCUMENTATION FOR THE REMOVAL OF THE FUSE BLOCK HAD NOT BEEN IMPLEMENTED. THEREFORE, UPON COMPLETION OF THE MAINTENANCE, THE CONTROL OPERATOR (CO) ONLY PROVIDED GENERAL SYSTEM RESTORATION GUIDANCE TO THE PLANT EQUIPMENT OPERATOR (PEO). THE CO DID NOT SPECIFY THE SEQUENCE OF STEPS FOR THE PEO TO PERFORM IN ORDER TO ENSURE PROPER SYSTEM RESTORATION. THIS RESULTED IN THE PEO LEAVING THE FUSE BLOCK FOR BREAKER 8-8908310129.

[203] SAN ONOFRE 1 DOCKET 50-206 LER 89-019
MANUAL REACTOR TRIP FOLLOWING A LOSS OF FEEDWATER TO ONE STEAM GENERATOR AS A RESULT OF MIS-COMMUNICATION.
EVENT DATE: 072489 REPORT DATE: 082389 NSSS: WE TYPE: PWR

(NSIC 215104) ON 7/24/89, AT 1216 WHILE AT 76% POWER, UNIT 1 WAS MANUALLY TRIPPED DUE TO A LOSS OF FEEDWATER FLOW TO STEAM GENERATOR (SG) "A" AND RESULTANT LOW LEVEL ALARM AT 1215. OPERATORS AUTHORIZED MAINTENANCE TECHNICIANS TO PERFORM TESTING OF THE SG "A" HIGH LEVEL ALARM AND STEAM FLOW/FEED FLOW MISMATCH REACTOR TRIP ALARM. DURING THE HIGH LEVEL ALARM PORTION OF THE TEST, THE SG "A" HIGH LEVEL ALARM ANNUNCIATED AS ANTICIPATED. THE HIGH LEVEL ALARM WAS PROMPTLY FOLLOWED BY THE SG "A" STEAM/FEED MISMATCH ALARM, WHICH WAS NOT ANTICIPATED AT THIS POINT IN THE TEST SEQUENCE. AFTER OBSERVING THAT SG "A" LEVELS WERE RAPIDLY DECREASING AND THE SG "A" MAIN FEEDWATER FLOW CONTROL VALVE (FCV) HAD TRIPPED CLOSED OPERATORS UNSUCCESSFULLY ATTEMPTED TO OPEN THE FCV AND, IN ACCORDANCE WITH PROCEDURES, MANUALLY TRIPPED THE REACTOR. SG "A" MAY HAVE DRIED OUT FOR A BRIEF PERIOD SHORTLY AFTER THE REACTOR HAD BEEN TRIPPED UNTIL THE AUXILIARY FEEDWATER SYSTEM ACTUATED AT 1217. ALL REQUIRED SYSTEMS FUNCTIONED NORMALLY. THE EFFECTS OF A RECENT DESIGN CHANGE ON THE FCV CIRCUITRY WERE NOT RECOGNIZED TO RESULT IN A LOSS OF FEEDWATER FLOW AND WERE, THEREFORE, NOT TRANSFERRED INTO STATION PROCEDURES OR OPERATOR TRAINING. THE ROOT CAUSE IS ATTRIBUTED TO WEAKNESSES IN SCE'S PROCESSES FOR ENSURING THAT DESIGN CHANGE INFORMATION IS ADEQUATELY INCORPORATED INTO PROCEDURES.

[204] SAN ONOFRE 2 DOCKET 50-361 LER 88-005 REV 01
UPDATE ON SPURIOUS TRAIN "A" TOXIC GAS ISOLATION SYSTEM ACTUATIONS DUE TO EQUIPMENT FAILURE.
EVENT DATE: 022088 REPORT DATE: 072589 NSSS: CE TYPE: PWR
OTHER UNITS INVOLVED: SAN ONOFRE 3 (PWR)
VENDOR: BECKMAN INSTRUMENTS, INC.

(NSIC 214863) AT 1905 ON FEBRUARY 20, 1988, WITH UNIT 2 IN MODE 1 AT 80% REACTOR POWER AND UNIT 3 IN MODE 3, TRAIN 'A' OF THE TOXIC GAS ISOLATION SYSTEM (TGIS) INITIATED BOTH TRAINS OF THE CONTROL ROOM EMERGENCY AIR CLEANUP SYSTEM (CREACUS) ON A SPURIOUS HIGH AMMONIA GAS LEVEL. SIMILARLY, AT 2348 ON FEBRUARY 21, 1988, WITH UNIT 2 IN MODE 1 AT 100% REACTOR POWER AND UNIT 3 IN MODE 2; AND AT 0241 ON FEBRUARY 22, 1988, WITH UNITS 2 AND 3 IN MODE 1 AT 100% AND 6% REACTOR POWER, RESPECTIVELY, TRAIN 'A' OF TGIS INITIATED BOTH TRAINS OF CREACUS ON A SPURIOUS HIGH AMMONIA GAS LEVEL. CREACUS OPERATED IN THE ISOLATION MODE AS DESIGNED FOLLOWING EACH OF THE ACTUATIONS, WHILE IT WAS DETERMINED THAT NO AMMONIA WAS PRESENT. THERE IS NO SAFETY SIGNIFICANCE TO THIS EVENT SINCE ALL TGIS AND CREACUS COMPONENTS OPERATED AS DESIGNED. INTERMITTENT SPIKING OF THE AMMONIA ANALYZER WAS DETERMINED TO BE THE CAUSE OF THE ACTUATIONS. THE DEFICIENT AMMONIA ANALYZER WAS REPLACED. FOLLOWING SUCCESSFUL TESTING AND MONITORING OF THE ANALYZER, TGIS TRAIN 'A WAS RETURNED TO SERVICE AT 0130 ON FEBRUARY 29, 1988. SUBSEQUENT INVESTIGATION DETERMINED THAT THE ANALYZER TEMPERATURE CONTROL PRINTED

CIRCUIT BOARD HAD FAILED, DE-ENERGIZING THE ANALYZER HEATER AND RESULTING IN THE OBSERVED SPIKING.

[205] SAN ONOFRE 2 DOCKET 50-361 LER 88-028 REV 01
UPDATE ON PLANT OPERATION ABOVE 100% POWER DUE TO DECREASE IN INDICATED PLANT POWER RELATIVE TO ACTUAL PLANT POWER.
EVENT DATE: 103188 REPORT DATE: 081589 NSSS: CE TYPE: PWR
VENDOR: PERMUTIT COMPANY, THE

(NSIC 215098) ON 10/31/88, IT WAS DETERMINED THAT UNIT 2 HAD BEEN OPERATED AT AN ESTIMATED ACTUAL SLIGHTLY IN EXCESS OF 100% FOR A PORTION OF THE TIME BETWEEN 8/27/88 AND 10/21/88 DUE TO: APPARENT DEFOULING OF FEEDWATER FLOW VENTURIS FOLLOWING OUTAGES; PLACING INTO SERVICE OF FEEDWATER FLOW TRANSMITTER FT1122, WHOSE OUTPUT INDICATED A LOWER FEEDWATER FLOW THAN THE TRANSMITTER PREVIOUSLY IN SERVICE; AND APPARENT DEGRADATION OF PORTIONS OF FLOW ELEMENT FE1111 THAT PROVIDE A DIFFERENTIAL PRESSURE INPUT TO FLOW TRANSMITTER FT1112. AS CORRECTIVE ACTION, A NEW METHODOLOGY FOR APPROXIMATING REACTOR POWER WAS DEVELOPED AND APPLIED TO PLANT HISTORICAL DATA. AS A RESULT, IT WAS DETERMINED THAT UNITS 2 AND 3 HAD EXCEEDED 100% ESTIMATED ACTUAL POWER (BUT NEVER EXCEEDED 102%, EXCEPT FOR THE EVENT DESCRIBED IN LER 88-035 (DOCKET NO. 50-361)) FOR SHORT PERIODS DIRECTLY FOLLOWING SOME OUTAGES DUE TO THE APPARENT PHENOMENON OF FEEDWATER FLOW VENTURIS DEFOULING. A LATER EVALUATION OF THE DEFOULING PHENOMENON, ASSUMING MAXIMUM EXPECTED INSTRUMENT ERROR AND COMPLETE DEFOULING OF THE VENTURIS (NEITHER OF WHICH HAS BEEN OBSERVED), CONCLUDED THAT THE POSSIBILITY HAD EXISTED FOR UNITS 2 AND 3 TO HAVE POTENTIALLY EXCEEDED 102% BY A SMALL AMOUNT (LESS THAN APPROX. 0.35%).

[206] SAN ONOFRE 2 DOCKET 50-361 LER 89-011
VOLUNTARY ENTRY INTO TECH SPECS 3.0.3 TO REPLACE PPS MATRIX POWER SUPPLY.
EVENT DATE: 062689 REPORT DATE: 072689 NSSS: CE TYPE: PWR

(NSIC 214891) AT 0755 ON 6/26/89, MAINTENANCE WAS GIVEN AN APPROVAL TO PERFORM THE UNIT 2 PLANT PROTECTION SYSTEM (PPS) MATRIX TEST WHICH IS A 31-DAY SURVEILLANCE REQUIRED BY TECH SPECS (TS). A PREREQUISITE STEP TO PERFORMING THE MATRIX TESTING IS TO CHECK ASSOCIATED PPS POWER SUPPLIES TO DETECT ANY VOLTAGE ANOMALIES. WHEN CHECKING PPS POWER SUPPLIES, MAINTENANCE TECHNICIANS DISCOVERED PPS MATRIX AD POWER SUPPLY, WHICH SUPPLIES POWER TO ONE HALF OF THE EMERGENCY FEEDWATER ACTUATION SYSTEM (EFAS) AD LOGIC MATRIX, EXHIBITING AN EXCESSIVE AC RIPPLE VOLTAGE OUTPUT OF 400 MILLIVOLTS, BUT STILL WITHIN SCE'S REPLACEMENT GUIDELINE OF 500 MILLIVOLTS RIPPLE. RATHER THAN LEAVING THE POWER SUPPLY IN SERVICE AND RISKING FAILURE AT A FUTURE TIME, THE POWER SUPPLY WAS REPLACED. OPERATIONS PERSONNEL RECOGNIZED THAT THE POWER SUPPLY REPLACEMENT WOULD CONSTITUTE A TS 3.0.3 ENTRY, DUE TO 3 AUXILIARY FEEDWATER VALVES (AFW) RECEIVING AN EFAS SIGNAL, THEREBY RESULTING IN THE VALVES OPENING AND BEING UNABLE TO CLOSE UPON RECEIVING A MAIN STEAM ISOLATION SIGNAL (MSIS) AS REQUIRED BY TS. BETWEEN 1322 AND 1343 ON 6/26/89, TS 3.0.3 WAS ENTERED TO REPLACE THE POWER SUPPLY. THE POWER SUPPLY WAS REMOVED AND A REPLACEMENT POWER SUPPLY WAS INSTALLED AND PLACED IN SERVICE. THE CAUSE OF THE EVENT WAS THAT EXISTING TSS DO NOT INCLUDE A LIMITING CONDITION FOR OPERATION AND ACCOMPANYING ACTION STATEMENT SPECIFICALLY FOR THESE AWF COMPONENTS.

[207] SAN ONOFRE 3 DOCKET 50-362 LER 89-007
SPURIOUS RECIRCULATION ACTUATION SIGNAL DURING SURVEILLANCE TESTING.
EVENT DATE: 062989 REPORT DATE: 073189 NSSS: CE TYPE: PWR
VENDOR: CUTLER-HAMMER

(NSIC 214958) ON 6/29/89 AT 1722, WITH UNIT 3 IN MODE 1 AT 75% POWER TO PERFORM MAINTENANCE ON THE MAIN CONDENSER, A SPURIOUS TRAIN "A" AND "B" RECIRCULATION

ACTUATION SIGNAL (RAS) OCCURRED DURING PERFORMANCE OF THE 31-DAY ENGINEERED SAFETY FEATURE ACTUATION SIGNAL (ESFAS) MATRIX TESTING. ALL COMPONENTS ACTUATED AS DESIGNED. BASED UPON THE RESULTS OF INVESTIGATIONAL TESTING, THE CAUSE OF THE RAS ACTUATION HAS BEEN ATTRIBUTED TO MECHANICAL AND/OR ELECTRICAL MALFUNCTION OF THE MATRIX "CD" RELAY HOLD PUSHBUTTON SWITCH WHICH RESULTED IN LOSS OF HOLD VOLTAGE TO THE MATRIX RELAY HOLD COILS. THE HOLD VOLTAGE IS DESIGNED TO BE APPLIED PRIOR TO DE-ENERGIZATION OF ALL OF THE MATRIX RELAYS BY THE SEQUENCING OF CONTACTS WITHIN THE HOLD PUSHBUTTON. THE INTENDED SEQUENCING PRECLUDES AN ACTUATION BY HOLDING ALL THE RELAYS IN THEIR UNACTUATED STATE EXCEPT FOR THE RELAY BEING TESTED. THE PUSHBUTTON SWITCH HAS BEEN REPLACED, FUNCTIONALLY TESTED, AND THE 31-DAY ESFAS SURVEILLANCE COMPLETED. THE FAULTED SWITCH IS BEING ANALYZED AT AN INDEPENDENT LABORATORY IN AN ATTEMPT TO DETERMINE A MORE DETAILED FAILURE MECHANISM. SCE IS WORKING WITH THE ESFAS VENDOR TO DETERMINE APPROPRIATE DESIGN ENHANCEMENTS TO THE SWITCH AND/OR TEST CIRCUITRY TO PREVENT SIMILAR SPURIOUS ACTUATIONS.

[208] SAN ONOFRE 3 DOCKET 50-362 LER 89-008
PLANT SHUTDOWN REQUIRED BY TECH SPECS DUE TO LPSI PUMP MECHANICAL SEAL FAILURE.
EVENT DATE: 063089 REPORT DATE: 073189 NSSS: CE TYPE: PWR
VENDOR: DURAMETALLIC CORP.

(NSIC 214959) AT 0806 ON 6/27/89, WITH UNIT 3 AT 75% POWER, LOW PRESSURE SAFETY INJECTION (LPSI) PUMP 3P015 WAS REMOVED FROM SERVICE FOR PREVENTIVE MAINTENANCE (PM). AT 0855 ON 6/29/89, AN INSERVICE TEST (IST) WAS CONDUCTED ON 3P015. EXCESSIVE MECHANICAL SEAL BREAKAGE WAS OBSERVED NECESSITATING SEAL REPLACEMENT. SINCE SEAL REPLACEMENT WOULD REQUIRE MORE TIME THAN REMAINED IN THE TECH SPEC 3.5.2 ON 6/30/89, UNIT SHUTDOWN WAS INITIATED. AN UNUSUAL EVENT (UE) WAS DECLARED. AT 0745 ON 6/30/89, THE UE WAS TERMINATED. AT 1134, THE UNIT ENTERED MODE 3 AND AT 1830, ENTERED MODE 4. THE MECHANICAL SEAL WAS REPLACED AND THE PUMP WAS RETURNED TO SERVICE ON 7/8/89. INVESTIGATION HAS DETERMINED THAT THE SEAL FAILURE WAS CAUSED BY OIL IN CONTACT WITH THE SEAL O-RING. THE SOURCE OF THE OIL IS UNKNOWN, HOWEVER, IT IS BELIEVED TO BE FROM EITHER AN OVERFILLING OF THE LOWER MOTOR BEARING OIL RESERVOIR OR LEAKAGE FROM THE THREADED CONNECTION BETWEEN THE LOWER BEARING OIL GAUGE FILL TUBE AND THE LOWER BEARING CARTRIDGE. THE FOLLOWING ACTIONS HAVE BEEN OR WILL BE TAKEN: 1) A SPRAY DEFLECTOR HAS BEEN INSTALLED ON THE LPSI PUMP WHICH WILL PREVENT OIL FROM ENTERING THE SEAL IF LEAKAGE FROM OR OVERFILL OF THE LOWER MOTOR BEARING OCCURS; 2) SIMILAR SPRAY DEFLECTORS WILL BE INSTALLED; 3) REVIEW EVENT WITH PERSONNEL; AND 4) REVIEW AND REVISE PROCEDURES FOR ADDING OIL TO LPSI.

[209] SEABROOK 1 DOCKET 50-443 LER 89-008
MANUAL REACTOR TRIP DURING NATURAL CIRCULATION TEST.
EVENT DATE: 062289 REPORT DATE: 072489 NSSS: WE TYPE: PWR
VENDOR: COPES-VULCAN, INC.

(NSIC 214903) ON 6/22/89, A MANUAL REACTOR TRIP WAS INITIATED WHILE CONDUCTING A NATURAL CIRCULATION TEST. SHORTLY AFTER TRIPPING THE REACTOR COOLANT PUMPS, AS PER THE NATURAL CIRCULATION TEST PROCEDURE, ONE OF THE CONDENSER STEAM DUMP VALVES BEING USED TO CONTROL TEMPERATURE FAILED TO THE FULL OPEN POSITION. THIS OPEN VALVE CAUSED AN INCREASED STEAM DEMAND WHICH INITIATED AN UNPLANNED PLANT COOLDOWN. THIS COOLDOWN CAUSED THE PRESSURIZER LEVEL TO DECREASE BELOW THE 17% MANUAL TRIP CRITERIA SPECIFIED IN THE STARTUP TEST PROCEDURE. DUE TO THE UNIT SHIFT SUPERVISOR'S MISINTERPRETATION OF THE TEST PROCEDURE TRIP CRITERIA, THE PLANT WAS NOT MANUALLY TRIPPED AT THAT TIME. WHEN THE STEAM DUMP VALVE, MS-PV-3011, WAS CLOSED, THE PREVIOUSLY DECREASING PRESSURIZER PRESSURE AND PRESSURIZER LEVEL BEGAN INCREASING. BOTH PRESSURIZER PRESSURE AND LEVEL CONTINUED TO INCREASE UNTIL THE PRESSURIZER PRESSURE, AT 2310 PSIG, APPROACHED THE TEST PROCEDURE MANUAL TRIP CRITERION OF 2340 PSIG AND A MANUAL REACTOR TRIP WAS INITIATED AT 12:36PM. THE CAUSE OF MS-PV-3011 FAILING TO THE FULL OPEN

POSITION HAS BEEN DETERMINED TO BE THE POSITIONER FEEDBACK LINKAGE WHICH BECAME DISCONNECTED DURING THE TEST. CORRECTIVE ACTION FOR THIS VALVE AND SIMILAR VALVES IS BEING SCHEDULED. THE STARTUP TEST PROGRAM WILL BE REVISED TO REQUIRE THAT A MORE COMPREHENSIVE PRE-TEST BRIEFING BE PROVIDED AND A DETERMINATION BE MADE OF WHICH TESTS REQUIRE SPECIAL CLASSROOM AND REVIEWED.

[210] SEQUOYAH 1 DOCKET 50-327 LER 87-030 REV 02
UPDATE ON BLOWN FUSE IN EMERGENCY START CIRCUITS RESULT IN SPURIOUS EMERGENCY DIESEL GENERATOR STARTS ON TWO OCCASIONS.
EVENT DATE: 062087 REPORT DATE: 082389 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)
VENDOR: LITTLEFUSE INC

(NSIC 215094) THIS LER IS RESERVED TO PROVIDE ADDITIONAL INFORMATION ON COMPLETED CORRECTIVE ACTIONS. ON 6/20/87, AT 0055 EST, ALL 4 STANDBY DIESEL GENERATORS (D/GS) STARTED DUE TO A BLOWN FUSE IN A REMOTE EMERGENCY START CIRCUITRY FOR 2A-A D/G. THE BLOWN FUSE (FLAS-5, LOT NO. 3) CAUSED LOSS OF VOLTAGE TO THE 2A-A REMOTE EMERGENCY START CIRCUITRY. STANDBY D/GS 1A-A, 1B-B, 2A-A, AND 2B-B STARTED AS REQUIRED. ON 7/4/87, AT 1212 EST, ALL 4 STANDBY D/GS STARTED DUE TO A BLOWN FUSE IN 125V DC VITAL BATTERY BOARD II, CIRCUIT C-16. THE BLOWN FUSE (FLAS-5, LOT NO. 2) CAUSED LOSS OF POWER TO THE LOGIC RELAY PANEL FOR 1B-B D/G. UPON LOSS OF VOLTAGE TO THE 1B-B D/G REMOTE EMERGENCY START CIRCUITRY, AND BY D/GS 1A-A, 1B-B, 2A-A, AND 2B-B STARTED AS REQUIRED. FOR BOTH EVENTS DESCRIBED ABOVE, NO DAMAGE OCCURRED TO THE D/GS, AND NONE OF THE D/GS LOADED BECAUSE A DEGRADED STAGE CONDITION DID NOT EXIST ON THE 6.9 KV SHUTDOWN BOARDS. TVA HAS CONTRACTED WITH LITTLEFUSE INC. TO SUPPLY 15,000 FLAS-5 FUSES. APPROXIMATELY 3,200 OUT OF 3,702 (DELIVERED) FUSES WERE INSTALLED IN MARCH AND APRIL 1987, 1,683 OF WHICH WERE FROM LOTS 2 AND 3. AS OF JULY 13, 1987, 69 FLAS-5 FUSE FAILURES HAVE OCCURRED. OF THESE, 67 FAILED FUSES WERE SUPPLIED IN LOTS 2 AND 3. THE LOT NUMBER OF THE REMAINING TWO FAILED FUSES COULD NOT BE DETERMINED.

[211] SEQUOYAH 1 DOCKET 50-327 LER 88-007 REV 04
UPDATE ON OPENING OF CONTAINMENT RESULTS IN SECONDARY CONTAINMENT ENVELOPE OUTSIDE THE BOUNDARY SET FOR SURVEILLANCE TESTING OF AUXILIARY BLDG. GAS TREATMENT SYSTEM.
EVENT DATE: 012488 REPORT DATE: 090189 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 215097) THIS LER HAS BEEN REVISED TO DESCRIBE THE RESULTS OF THE DESIGN EVALUATION CONSIDERING ALTERNATIVES TO ENSURE THAT THE AUXILIARY BUILDING GAS TREATMENT SYSTEM CAN PERFORM ITS DESIGN FUNCTION DURING VARIOUS MODES OF 2-UNIT OPERATION. ON 1/24/88, WITH UNITS 1 AND 2 IN MODE 5 (COLD SHUTDOWN), IT WAS DISCOVERED THAT THE AUXILIARY BUILDING SECONDARY CONTAINMENT ENCLOSURE (ABSCE) WAS NOT BEING MAINTAINED WITHIN THE CONFIGURATION SET DURING THE TECHNICAL SPECIFICATION (TS) SURVEILLANCE TESTING USED TO VERIFY ABGTS OPERABILITY. ON 8/24/88, WITH UNIT 1 IN MODE 5 AND UNIT 2 IN MODE 1 (APPROXIMATELY 98 PERCENT POWER), IT WAS DETERMINED THAT THE UNIT 1 CONTAINMENT PURGE SYSTEM WAS IN OPERATION WITHOUT THE REQUIRED COMPENSATORY MEASURES BEING PROPERLY DOCUMENTED. THESE CONDITIONS WERE CAUSED BY (1) THE LACK OF ADEQUATE CONTROLS TO ENSURE THE ABSCE BOUNDARY WAS MAINTAINED WITHIN THE CONDITION SET BY SURVEILLANCE TESTING, (2) AN INAPPROPRIATE DESIGN ASSUMPTION MADE DURING PLANT CONSTRUCTION ON HOW ABSCE BEACHES WOULD BE CONTROLLED, AND (3) AN INCOMPLETE COMPENSATORY MEASURES PROGRAM. AS SHORT-TERM CORRECTIVE ACTIONS, THE BLAST DOOR WAS CLOSED (BEFORE UNIT 2 ENTERED MODE 4 ON FEBRUARY 6, 1988), THE PROCEDURE GOVERNING ABSCE BREACHES WAS CHANGED, AND THE UNIT CONTAINMENT PURGE SYSTEM WAS TAGGED OUT OF SERVICE.

[212] SEQUOYAH 1 DOCKET 50-327 LER 88-047 REV 01
UPDATE ON ERRATIC FEEDWATER CONTROLS CAUSED A FEEDWATER ISOLATION ON HI-HI SG LEVEL RESULTING IN A REACTOR TRIP ON LO-LO SG LEVEL.
EVENT DATE: 122588 REPORT DATE: 072589 NSSS: WE TYPE: PWR

(NSIC 214861) ON 12/25/88, AT 0051 EST, UNIT 1 REACTOR TRIPPED FROM 7% POWER ON LO-LO LEVEL IN S/G LOOP 4 FOLLOWING A FEEDWATER ISOLATION (FWI) ON HI-HI LEVEL IN S/G LOOP 2. BEFORE THIS EVENT, THE MAIN TURBINE WAS ROLLED AT 0004 EST AND REACHED 1700 RPM AT 0037 ST. DURING THIS TIME, SPARKS WERE NOTED IN THE NO. 10 BEARING AREA ON THE TURBINE/GENERATOR. AT 0040 EST, THE SHIFT OPERATIONS SUPERVISOR ELECTED TO MANUALLY TRIP THE TURBINE AND THEN INITIATE A CONTROLLED REDUCTION IN POWER. APPROX. 10 MIN LATER, A FWI OCCURRED ON HI-HI LEVEL IN LOOP 2. WITH THE REACTOR AT APPROX. 2% POWER, THE OPERATOR ATTEMPTED TO REDUCE REACTOR POWER TO LESS THAN 5% TO BE WITHIN THE CAPABILITIES OF AUX. FEEDWATER (AFW) SINCE MAIN FEEDWATER WAS ISOLATED. AS A RESULT OF A COMBINATION OF THE FWI, THE COLD AFW, AND THE RAPID REDUCTION IN POWER, S/G LEVELS DROPPED SHARPLY AND THE UNIT TRIPPED ON LO-LO LEVEL IN LOOP 4. FOLLOWING THE TRIP, THE BALANCE OF PLANT OPERATOR TOOK MANUAL CONTROL OF AFW TO LIMIT THE REACTOR COOLANT SYSTEM (RCS) COOLDOWN. AS AVERAGE TEMPERATURE (TAVG) DROPPED TO 540F, THE LEAD OPERATOR STARTED AN EMERGENCY BORATION AT 75 GPM AS REQUIRED. TAVG DECREASED TO 538F BEFORE TURNING AROUND AND SLOWLY INCREASING. THE TRIP WAS DUE TO ERRATIC LEVEL CONTROL ON ALL S/G'S MAIN AND BYPASS REGULATOR VALVES FOR FEEDWATER.

[213] SEQUOYAH 1 DOCKET 50-327 LER 89-006 REV 01
UPDATE ON AN AUXILIARY BLDG FIRE DOOR WAS BREACHED WITHOUT APPROPRIATE COMPENSATORY MEASURES IN PLACE DUE TO INADEQUATE TRAINING.
EVENT DATE: 021689 REPORT DATE: 072589 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 214886) THIS LER IS BEING REVISED TO PROVIDE THE RESULTS OF AN EVALUATION OF THE TRAINING AND PROGRAMMATIC CONTROLS ASSOCIATED WITH THE REQUIREMENTS FOR BREACHING OF FIRE AND AUX. BLDG SECONDARY CONTAINMENT ENCLOSURE (ABSCE) BOUNDARY DOORS. ON 2/16/89 WITH UNIT 1 MODE 1, IT WAS DISCOVERED THAT BLDG WASTE PACKAGING AREA DOOR A111 (ELEVATION 706) HAD BEEN BREACHED WITHOUT ISSUANCE OF A BREACHING PERMIT. DOOR A111 FUNCTIONS BOTH AS A TECH SPEC (TS) FIRE BARRIER DOOR AND AS PART OF THE ABSCE. SUBSEQUENT INVESTIGATION TO THE INCIDENT FOUND THAT THE DOOR HAD BEEN BREACHED OPEN FROM 1741 EST TO 1847 EST AND 1900 EST TO 2130 EST ON 2/15/89, AND FOR AN UNDETERMINED PERIOD OF TIME ON 2/16/89. THE CONTRIBUTING CAUSES OF THIS EVENT ARE INCOMPLETE DOOR VENTIFICATION, INADEQUATE PROGRAMMATIC CONTROLS TO ENSURE THAT DOORS REMAIN UNIQUELY VENTIFIED, AND PERSONNEL OVERSIGHT. FOR IMMEDIATE CORRECTIVE ACTION, DOOR A111 WAS USED ON 2/16/89. PERSONNEL RESPONSIBLE FOR THE BREACHES ON 2/15/89 WERE COUNSELLED ON THE REQUIREMENTS FOR ABSCE BOUNDARY AND FIRE DOORS. A PLANT-WIDE DISPATCH WAS ISSUED ON 2/23/89, EXPLAINING TO PERSONNEL THE IMPORTANCE FOR ADHERENCE TO REQUIREMENTS ASSOCIATED WITH FIRE DOORS, SUCH THAT, COMPLIANCE WITH REGULATIONS CAN BE ENSURED. THIS DISPATCH ALSO EXPLAINED TO PLANT PERSONNEL THAT A FIRE DOOR MAY BE IDENTIFIED EITHER BY ITS RED COLOR OR A SIGN ON THE DOOR.

[214] SEQUOYAH 1 DOCKET 50-327 LER 89-003 REV 01
UPDATE ON BRIEF INTERRUPTION OF CONTROL POWER TO 6.9KV SHUTDOWN BOARD RESULTED IN AUTO START OF A MOTOR-DRIVEN AUXILIARY FEEDWATER PUMP.
EVENT DATE: 021889 REPORT DATE: 080989 NSSS: WE TYPE: PWR
VENDOR: GOULD INC. (POWER SYS DIV)

(NSIC 214991) ON 2/10/89, AT 0004 EST, WITH UNIT 1 IN MODE 1 (100% REACTOR POWER, 578F, AND 2235 PSIG), THE 1A-A MOTOR-DRIVEN AUXILIARY FEEDWATER PUMP (MDAFWP) STARTED FOLLOWING THE MOMENTARY INTERRUPTION OF CONTROL POWER TO THE 6.9KV SHUTDOWN BOARD 1A-A. DURING PERFORMANCE OF SURVEILLANCE INSTRUCTION (SI)-621, "PERIODIC FUNCTIONAL TEST OF LOSS OF VOLTAGE RELAYS ON SHUTDOWN BOARD," THE

NORMAL FEEDER BREAKER 1718 FOR 6.9KV SHUTDOWN BOARD 1A-A DID NOT TRIP AS REQUIRED RESULTING IN SI-621 BEING STOPPED AT 2300 EST ON 2/9/89. A GROUND WAS DISCOVERED ON THE POSITIVE SIDE OF THE 125VDC ALTERNATE CONTROL BUS BY TRANSMISSION AND CUSTOMER SERVICES PERSONNEL, WHICH WAS VERIFIED BY THE OPERATOR BY CHECKING THE GROUND INDICATOR ON 125VDC VITAL BATTERY BOARD III. WHILE ATTEMPTING TO ISOLATE THE GROUND, BREAKER 204 ON 125VDC VITAL BATTERY BOARD III WAS OPENED WHICH REMOVED POWER FROM THE ALTERNATE CONTROL BUS. BREAKER 204 WAS CLOSED AFTER VERIFYING THAT THE GROUND WAS NOT ON THE SHUTDOWN BOARD. WHEN POWER WAS RESTORED ON THE ALTERNATE CONTROL BUS, THE BOY RELAYS OPERATED AS THEY WOULD IN BLACKOUT (BO) CONDITION AND THE UVY RELAY REENERGIZED. THIS COMPLETED THE NEEDED PERMISSIVES THAT CAUSED THE MDAFWP BREAKER TO CLOSE. OPERATIONS PERSONNEL ATTEMPTED TO ENSURE THAT OPERATING BREAKER 204 WOULD NOT ADVERSELY IMPACT PLANT EQUIPMENT.

[215] SEQUOYAH 1 DOCKET 50-327 LER 89-018 REV 01
UPDATE ON FOUR EVENTS CONCERNING ABGTS OPERABILITY WITH RADIATION MONITOR O-RM-90-101 REMOVED FROM SERVICE.
EVENT DATE: 061989 REPORT DATE: 082889 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SEQUOYAH 2 (PWR)

(NSIC 215080) THIS REVISED REPORT DESCRIBES FURTHER EVENTS CONCERNING THE OPERABILITY OF THE AUXILIARY BUILDING GAS TREATMENT SYSTEM (ABGTS) DURING PERIODS WHEN RADIATION MONITOR (RM) O-RM-90-101 WAS REMOVED FROM SERVICE. THE FIRST EVENT WAS THE DISCOVERY THAT PROCEDURES DIRECT REMOVING THE RM FROM SERVICE FOR SURVEILLANCE TESTING, THEREBY RENDERING BOTH TRAINS OF ABGTS INOPERABLE. THE SECOND AND THIRD EVENTS WERE CASES WHERE THE RM WAS REMOVED FROM SERVICE FOR REPAIRS, THEREBY RENDERING BOTH TRAINS OF ABGTS INOPERABLE, FOR LONGER DURATIONS THAN ALLOWED BY LIMITING CONDITION FOR OPERATION (LCO) 3.0.3. THE FOURTH AND SUBSEQUENT EVENTS WERE CASES WHERE THE RM WAS REMOVED FROM SERVICE FOR SERVICING OR REPAIR FOR A PERIOD OF TIME WITHIN THE LIMITS ALLOWED BY LCO 3.0.3. THE ROOT CAUSE OF ALL THE EVENTS WAS THE INAPPROPRIATE INCLUSION OF RM O-RM-90-101 WITHIN THE SCOPE OF EQUIPMENT NEEDED TO DEMONSTRATE ABGTS OPERABILITY. A CONTRIBUTING CAUSE OF THE SECOND AND THIRD EVENTS WAS A FAILURE TO RECOGNIZE THE NECESSITY TO ENTER LCO 3.0.3 WHEN RM O-RM-90-101 IS REMOVED FROM SERVICE. TECH SPEC CHANGE 88-34 WAS APPROVED BY NRC ON 8/3/89, WHICH REMOVED RM O-RM-90-101 FROM THE SCOPE OF EQUIPMENT NEEDED TO DEMONSTRATE ABGTS OPERABILITY.

[216] SEQUOYAH 1 DOCKET 50-327 LER 89-020
FAILURE TO PERFORM A PROPER MONTHLY CHANNEL CHECK FOR THE REACTOR VESSEL LEVEL INSTRUMENTATION SYSTEM UPPER-RANGE INDICATORS.
EVENT DATE: 070589 REPORT DATE: 080389 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 214953) JULY 5, 1989, WITH UNITS 1 AND 2 IN MODE 1 (100 PERCENT POWER, AT 2235 PSIG AND DEGREES F), THE SHIFT TECHNICAL ADVISOR (STA) OBSERVED THAT THE REACTOR VESSEL LEVEL INDICATING SYSTEM (RVLIS) UPPER-RANGE INDICATOR (1-LI-68-369) WAS INDICATING HIGH AS COMPARED TO THE REDUNDANT CHANNEL. OPERATIONS PERSONNEL DECLARED 1-LI-68-369 INOPERABLE AND ENTERED TECHNICAL SPECIFICATION LIMITING CONDITION FOR OPERATION (LCO) 3.3.3.7 AT 1837 EDT. THE CAUSE OF THE INSTRUMENT FAILURE WAS DETERMINED TO BE DIRTY CONNECTORS. INSTRUMENT WAS REPAIRED AND RETURNED TO OPERABLE STATUS ON JULY 5, 1989, AT 2337. LCO 3.3.3.7 WAS EXITED AT THIS TIME. UPON INVESTIGATION OF THE INCIDENT, IT WAS DETERMINED THAT SURVEILLANCE INSTRUCTION (SI) 3, "DAILY, WEEKLY, AND MONTHLY LOGS," DID NOT HAVE ADEQUATE CHANNEL CHECK CRITERIA FOR THE INSTRUMENTS IN QUESTION. THE ADEQUATE PROCEDURE RESULTED IN FAILURE TO PERFORM PROPER CHANNEL CHECKS OF THE RVLIS UPPER-RANGE INSTRUMENTS SINCE APRIL 7, 1989, WHEN THE INITIAL CHANNEL CHECK FOR THESE INSTRUMENTS WAS PERFORMED. IMMEDIATE CORRECTIVE ACTION TAKEN WAS TO REVISE SI-3 TO REQUIRE THE CORRECT CHANNEL CHECK ACCEPTANCE CRITERIA. AS A RESULT OF A SIMILAR EVENT, PLANT MANAGEMENT ISSUED A MEMORANDUM

TO LICENSED PERSONNEL EMPHASIZING THE IMPORTANCE AND SIGNIFICANCE OF THEIR ROLE IN REVIEWING SIS AND IDENTIFYING POTENTIAL PROBLEMS.

[217] SEQUOYAH 1 DOCKET 50-327 LER 89-019
A SPURIOUS CONTAINMENT VENTILATION ISOLATION OCCURRED DURING UNBLOCKING OF HANDSWITCH HS-90-136A2.
EVENT DATE: 071389 REPORT DATE: 081089 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 214992) AT APPROXIMATELY 1000 EDT, JULY 13, 1989, WITH UNIT 1 IN MODE 1 (100 PERCENT POWER, 2,235 PSIG, AND 578 DEGREES F), A CONTAINMENT VENTILATION ISOLATION (CVI) OCCURRED ON UNIT 1. THE TRAIN B CVI WAS INITIATED FROM RADIATION MONITOR (RM) 1-RM-90-131 WHEN THE RM WAS RETURNED TO THE NORMAL (UNBLOCKED) POSITION FOLLOWING AN ACCEPTABLE FUNCTIONAL TEST OF A MODIFICATION TO ADD TEST POINTS TO PANEL 0-M-12. AS IMMEDIATE ACTIONS SUBSEQUENT TO THE CVI, THE ASSISTANT SHIFT OPERATIONS SUPERVISOR VERIFIED THE ALARM WAS FALSE, AND THE UNIT OPERATOR RECOVERED FROM THE CVI IN ACCORDANCE WITH SYSTEM OPERATING INSTRUCTION 88, "CONTAINMENT ISOLATION SYSTEM," AT APPROXIMATELY 1025 EDT. CORRECTIVE ACTIONS INCLUDED REPERFORMING THE FUNCTIONAL TEST ACTIVITIES OUTLINED IN THE WORKPLAN TO VERIFY NO INTERNAL SHORTS OR GROUNDS EXISTED IN THE WIRING OR TERMINATIONS INSTALLED BY THE MODIFICATION. THE FUNCTIONAL TEST WAS REPEATED TWICE, INCLUDING ACTUATION OF THE BLOCKING HANDSWITCH HS-90-136A2, AND NO CVIS OCCURRED.

[218] SEQUOYAH 1 DOCKET 50-327 LER 89-022
AN EVENT WHERE LCO 3.0.3 WAS ENTERED AS A RESULT OF EXCEEDING THE TIME LIMIT OF ACTION 2.D OF LCO 3.3.1.1 FOLLOWING THE LOSS OF ONE EXCORE DETECTOR.
EVENT DATE: 072289 REPORT DATE: 082189 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 215070) AT 0258 EDT ON 7/22/89, WITH BOTH UNITS IN MODE 1, AT 100% POWER, 2,235 LBS. GAUGE, AND 578 DEGREES FAHRENHEIT, POWER RANGE EXCORE DETECTOR N-43 ON UNIT 1 FAILED DURING OPERATION AND WAS DECLARED INOPERABLE. LIMITING CONDITION FOR OPERATION (LCO) 3.3.1.1 WAS ENTERED WHICH REQUIRED, IN PART, THAT THE QUADRANT POWER TILT RATIO BE MONITORED WITH THE REMAINING THREE EXCORE DETECTORS AND VERIFIED CONSISTENT WITH A NORMALIZED SYMMETRIC POWER DISTRIBUTION USING THE MOVEABLE INCORE DETECTORS IN THE FOUR PAIRS OF SYMMETRIC THIMBLE LOCATIONS AT LEAST ONCE EVERY 2 HOURS. AT 1458, AFTER FAILING TO MEET THE LCO 3.3.1.1 ACTION 2.D ACTION REQUIREMENT DESPITE EXTENSIVE EFFORTS TO PERFORM THE REQUIRED SURVEILLANCE TESTS, LCO 3.0.3 WAS ENTERED. ENTRY INTO LCO 3.0.3 IS CONSIDERED TO BE AN OPERATION PROHIBITED BY TECHNICAL SPECIFICATIONS AND, THUS, IS REPORTABLE UNDER 10 CFR 50.73, PARAGRAPH A.2.I.B. AT 1930, THE REQUIRED SURVEILLANCE TEST WAS COMPLETED, AND LCO 3.0.3 WAS EXITED, RETURNING TO THE ACTION STATEMENTS OF LCO 3.3.1.1. THE ROOT CAUSE OF THIS EVENT WAS DETERMINED TO BE AN INADEQUATE PROCEDURE. SURVEILLANCE INSTRUCTION (SI) 178, "MOVEABLE DETECTOR DETERMINATION QUADRANT POWER TILT," CONTAINED NO GUIDANCE TO HELP THE USER CHOOSE WHICH OF THE THREE AVAILABLE METHODS IS APPROPRIATE FOR CURRENT OPERATING CONDITIONS. THREE CONTRIBUTING CAUSES WERE IDENTIFIED.

[219] SEQUOYAH 2 DOCKET 50-328 LER 89-009
FAILURE TO DETERMINE THE ICE BED TEMPERATURES EVERY 12 HRS AS SPECIFIED IN THE TECH SPEC LCO 3.6.5.2.A.
EVENT DATE: 062989 REPORT DATE: 073189 NSSS: WE TYPE: PWR

(NSIC 214954) 6/29/89, AT 2301, WITH UNITS 1 AND 2 AT 100% POWER, LIMITING CONDITION FOR OPERATION (LCO) 3.0.3 WAS ENTERED BECAUSE OF A FAILURE TO COMPLY WITH TECH SPEC (TS) 3.6.5.2.A. ON 5/4/89, IT WAS DISCOVERED THAT THE ICE CONDENSER BED TEMPERATURE RECORDER IN MAIN CONTROL ROOM (MCR) WAS NOT PRINTING.

LCO 3.6.5.2 WAS ENTERED. A TEMPORARY RECORDER WAS INSTALLED IN THE MCR TO MONITOR THE ICE BED TEMPERATURE, AND LCO 3.6.5.2 WAS EXITED. ON 6/25/89, AN NRC INCIDENT INSPECTOR QUESTIONED VALIDITY OF THIS CORRECTIVE ACTION WITHOUT BENEFIT OF A TEMPORARY ALTERATION CONTROL FORM (TACF). IT WAS DETERMINED THAT EITHER A TACF WAS APPROPRIATE OR THAT LCO 3.6.5.2 SHOULD BE ENTERED AND READINGS TAKEN AT THE LOCAL ICE CONDENSER MONITORING PANEL INSIDE CONTAINMENT. ON 6/29/89, TVA MANAGEMENT CONCURRED WITH NRC'S POSITION, AND LCO 3.6.5.2 WAS ENTERED. PERSONNEL ENTERED CONTAINMENT AND RECORDED THE ICE CONDENSER TEMPERATURES AS SPECIFIED IN TS 3.6.5.2.A. HOWEVER, THE PERSONNEL FAILED TO RECORD THE TEMPERATURE WITHIN THE TIME INTERVAL SPECIFIED IN TS 3.6.5.2.A, AND LCO 3.0.3 WAS ENTERED AT 2301. AFTER ACCEPTABLE ICE BED TEMPERATURES WERE VERIFIED AND RECORDED, LCO 3.0.3 WAS EXITED AT 0027 ON 6/30/89. THE ROOT CAUSE OF THE EVENT WAS PERSONNEL MISINTERPRETATION OF THE TS AND IMPLEMENTING SURVEILLANCE INSTRUCTIONS CONTRIBUTING CAUSE WAS INADEQUATE SURVEILLANCE INSTRUCTION.

[220] SEQUOYAH 2 DOCKET 50-328 LER 89-008
REACTOR TRIP BECAUSE OF A SUSPECTED DROPPED ROD.
EVENT DATE: 071089 REPORT DATE: 080989 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 214994) ON 7/10/89, AT 1134 EDT, WITH UNIT 2 IN MODE 1 (100% REACTOR POWER, APPROX. 2,235 PSIG, AND TAVG AT 578F), A REACTOR TRIP OCCURRED. THE TRIP OCCURRED ON HIGH NEGATIVE FLUX AS NOTED ON THE FIRST-OUT-ANNUNCIATOR. PLANT SHUTDOWN PROCEEDED IN AN ORDERLY MANNER CONSISTENT WITH PROCEDURES WITH NO OVERCOOLING TRANSIENT. A POSTTRIP REVIEW TEAM WAS FORMED THAT CONDUCTED PERSONNEL INTERVIEWS AND DEVELOPED RECOMMENDATIONS FOR IMMEDIATE CORRECTIVE/INVESTIGATIVE ACTIONS. THESE ACTIONS ARE DETAILED IN THE REPORT AND INCLUDE TROUBLESHOOTING ON THE ROD CONTROL SYSTEM AS WELL AS OTHER DEFICIENCIES NOTED BY THE OPERATORS. PERSONNEL STATEMENTS, STRIP CHART RECORDERS, TROUBLESHOOTING WORK REQUESTS, PREVIOUS TRIP REPORTS, WESTINGHOUSE OWNER'S GROUP TRIP DATA BASE, AND A TRIP MODELING ROUTINE OF THE WATTS BAR SIMULATOR WERE PART OF THE RESOURCES UTILIZED IN THE TEAM'S EVALUATION. ALSO, PLANT MANAGEMENT REQUESTED WESTINGHOUSE TO ASSIST IN PROVIDING EXPERTISE IN TROUBESHOOTING THE ROD CONTROL SYSTEM. THE TEAM'S CONCLUSION IS THAT THE TRIP WAS A DROPPED ROD EVENT BECAUSE OF A SPURIOUS CONTROL SIGNAL FAULT. NEITHER THE TRIP ITSELF NOR THE TEAM'S RECOMMENDED RESTART PLAN POSE ANY COMPROMISE TO THE SAFE OPERATION OF THE UNIT. AT 1857 WITH UNIT 2 IN MODE 3 (0% REACTOR POWER, 2235 PSIG, AND 547F), A SECOND REACTOR TRIP SIGNAL WAS GENERATED BY A SOURCE RANGE SPIKE APPROX. 7 HRS AFTER THE HIGH FLUX TRIP. SOURCE RANGE CHANNEL HAD NOT BEEN BYPASSED.

[221] SHEARON HARRIS 1 DOCKET 50-400 LER 89-012
TECH SPEC VIOLATION DUE TO FAILURE TO INITIATE CONTINUOUS SAMPLE OF SECONDARY WASTE CONTINUOUS RELEASE CAUSED BY PERSONNEL ERROR.
EVENT DATE: 063089 REPORT DATE: 073189 NSSS: WE TYPE: PWR

(NSIC 214930) WHILE AT 100% REACTOR POWER ON 6/30/89, A CONTINUOUS RELEASE OF THE SECONDARY WASTE SAMPLE TANK (SWST) (EIIS:WD) WAS INITIATED AT 1720 HRS ON 6/30/89, AND TERMINATED AT 0602 HRS ON 7/1/89. TECH SPECS 3.11.1.1 TABLE 4.11-1 ACTION 2.B REQUIRES A CONTINUOUS COMPOSITE SAMPLE PROPORTIONAL TO THE DISCHARGE FLOW RATE WHEN THE SWST IS OPERATED IN THE CONTINUOUS RELEASE MODE. THE REQUIRED CONTINUOUS SAMPLING OF THE RELEASE WAS NOT PERFORMED. THIS WAS DISCOVERED AT 0620 HRS ON 7/1/89, WHEN A CHEMISTRY TECHNICIAN WENT TO SECURE THE SAMPLER FOLLOWING THE RELEASE AND FOUND IT HAD NOT BEEN PREVIOUSLY TURNED ON. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR AS CHEMISTRY PERSONNEL STATED THEY WERE NOT NOTIFIED BY RADWASTE PERSONNEL THAT A CONTINUOUS RELEASE OF THE SWST WAS BEING INITIATED. THE EXISTING PROCEDURE HAD A SIGN OFF FOR NOTIFYING THE MAIN CONTROL ROOM AND HEALTH PHYSICS, BUT ONLY CONTAINED A NOTE FOR NOTIFYING CHEMISTRY WHEN THE FLOW RATE WAS CHANGED. CORRECTIVE ACTIONS INCLUDE: A GRAB SAMPLE OF SECONDARY WASTE WAS TAKEN AT 0640 HRS ON 7/1/89, AND ANALYZED WITH NO

RADIOACTIVITY BEING DETECTED, OPERATING PROCEDURE, SECONDARY WASTE SAMPLE TANK, (OP)-120.01.02 HAS BEEN REVISED TO INCLUDE AN OPERATOR SIGN-OFF TO NOTIFY CHEMISTRY WHEN A SWST RELEASE HAS COMMENCED. SAMPLES TAKEN PRIOR TO AND AFTER THE RELEASE REVEALED NO RADIOACTIVITY PRESENT IN SWST.

[222] SHOREHAM DOCKET 50-322 LER 89-006
AUTOMATIC REACTOR TRIP AND RHR SYSTEM ISOLATION DUE TO PERSONNEL ERROR DURING PRESSURE TRANSMITTER VALVE OPERATIONS.
EVENT DATE: 070789 REPORT DATE: 080489 NSSS: GE TYPE: BWR

(NSIC 214951) AT 1437 HOURS ON 7/7/89 SHOREHAM NUCLEAR POWER STATION WAS IN OPERATIONAL CONDITION 4 (COLD SHUTDOWN) WITH THE REACTOR AT 104F, 0 PSIG, AND ALL CONTROL RODS FULLY INSERTED. WHILE RETURNING A LEVEL TRANSMITTER TO SERVICE A HYDRAULIC TRANSIENT WAS INADVERTENTLY INITIATED ON A COMMON PROCESS LEG FOR THE "B" RPV INSTRUMENT RACK. THE TRANSIENT RESULTED IN AUTOMATIC ESF AND RPS ACTUATION SIGNALS. THESE ACTUATION SIGNALS RESULTED IN A TEMPORARY LOSS OF SHUTDOWN COOLING WHEN THE RESIDUAL HEAT REMOVAL (RHR) SYSTEM ISOLATED AND A FULL REACTOR TRIP OCCURRED. THIS EVENT WAS CAUSED BY AN I&C TECHNICIAN NOT RECOGNIZING THAT THE TRANSMITTER EQUALIZING VALVE WAS ONLY PARTIALLY SEATED DUE TO DIFFICULTY RELATED TO MANUAL OPERATION OF THE EQUALIZING VALVE. AS CORRECTIVE ACTIONS, THE REACTOR TRIP AND ANALOG TRIP "GROSS FAILURE" ALARMS WERE RESET IMMEDIATELY, THE RHR PUMP WAS RESTARTED, AND THE EQUALIZING VALVE WAS SEATED. FOR PREVENTIVE ACTIONS, THE EVENT WILL BE DISCUSSED WITH I&C TECHNICIANS AND FOREMEN TO ENSURE THAT ALL PERSONNEL ARE AWARE OF THE POTENTIAL FOR THIS TYPE OF INCIDENT, AND AN ENGINEERING EVALUATION OF THE VALVE WILL BE PERFORMED TO DETERMINE ANY CORRECTIVE MEASURES THAT WILL ENABLE THE VALVE TO BE OPERATED MORE EASILY. PLANT MANAGEMENT WAS NOTIFIED AND THE NRC WAS NOTIFIED OF THE EVENT AT 1544 PER 10 CFR 50.72.

[223] SOUTH TEXAS 1 DOCKET 50-498 LER 89-002 REV 02
UPDATE ON INADEQUATE DESIGN AND TESTING OF CONTAINMENT PENETRATION OVERCURRENT PROTECTIVE DEVICES DUE TO A DESIGN ERROR.
EVENT DATE: 010689 REPORT DATE: 080489 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SOUTH TEXAS 2 (PWR)

(NSIC 215077) ON 1/6/89, UNIT 1 WAS IN MODE 1 AT 50% POWER. DURING A REVIEW OF A CALCULATION, IT WAS DISCOVERED THAT TWO CONTAINMENT ELECTRICAL PENETRATIONS WERE NOT PROVIDED WITH BACKUP OVERCURRENT PROTECTION AS COMMITTED TO IN THE FSAR. THE PENETRATION CIRCUITS WERE DEENERGIZED PER TECH SPEC 3.8.4.1. AS PART OF THE CORRECTIVE ACTION FOR THIS EVENT, FOLLOW UP CALCULATION REVIEWS WERE PERFORMED WHICH IDENTIFIED ADDITIONAL CIRCUITS FOR WHICH TESTING OF OVERCURRENT PROTECTIVE DEVICES WAS NOT PERFORMED AS REQUIRED BY TECH SPECS. THE CAUSE OF THESE EVENTS WAS A DESIGN ERROR WHICH RESULTED IN THE IMPROPER APPLICATION OF PENETRATION OVERCURRENT PROTECTION CRITERIA AND FAILURE TO PERFORM SURVEILLANCE TESTING. THE PENETRATIONS WHICH REQUIRE ADDITIONAL BACKUP PROTECTION WILL REMAIN DEENERGIZED AS REQUIRED BY TECH SPECS UNTIL BACKUP PROTECTION IS INSTALLED. A DETAILED REVIEW OF THE PENETRATION PROTECTION CALCULATIONS AND A REVIEW OF THE SURVEILLANCE TEST DATABASE HAS BEEN PERFORMED. AN ERROR HAS BEEN DISCOVERED IN THE REVIEW OF ONE CIRCUIT WHICH WAS THE RESULT OF THE FAILURE TO INCORPORATE A DESIGN CHANGE IN THE CALCUALTION REVIEW; THEREFORE, OUTSTANDING DESIGN CHANGES WILL BE REVIEWED FOR FURTHER IMPACT ON THE PENETRATION CALCULATIONS.

[224] SOUTH TEXAS 1 DOCKET 50-498 LER 89-015
REACTOR TRIP DUE TO A FAILED RELAY IN THE GENERATOR BREAKER CONTROL CIRCUIT.
EVENT DATE: 070489 REPORT DATE: 080389 NSSS: WE TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 214968) ON JULY 4, 1989, UNIT 1 WAS IN MODE 1 AT 100 PERCENT POWER. AT

1915 HOURS, THE MAIN GENERATOR OUTPUT CIRCUIT BREAKER OPENED. THE MAIN TURBINE GOVERNOR VALVES CLOSED TO LIMIT TURBINE OVERSPEED AND A REACTOR TRIP SUBSEQUENTLY OCCURRED ON OVERTEMPERATURE-DELTA TEMPERATURE. THE PLANT SAFETY SYSTEMS RESPONDED AS DESIGNED AND NO UNEXPECTED POST-TRIP TRANSIENTS OCCURRED. THE CAUSE OF THIS EVENT WAS THE FAILURE OF A 125 VDC RATED AUXILIARY RELAY WHICH WAS ERRONEOUSLY USED IN THE 250 VDC GENERATOR CIRCUIT BREAKER TRIP CIRCUIT. DESIGN, PROCUREMENT, INSTALLATION AND TESTING DOCUMENTATION WAS REVIEWED IN AN EFFORT TO DETERMINE THE SOURCE OF THE INCORRECT RELAY; HOWEVER, THE SOURCE WAS NOT FOUND. THE MOST PLAUSIBLE CAUSE OF THE USE OF THE 125 VDC RELAY WAS THE FAILURE OF THE RELAY MANUFACTURER TO SUPPLY THE RELAY REQUIRED BY THE PURCHASE ORDER. THE FAILED RELAY AND ONE OTHER 125 VDC RELAY HAVE BEEN REPLACED WITH THE CORRECT MODEL 250 VDC CONTROL CIRCUITS IN UNIT 2 HAVE BEEN INSPECTED FOR SIMILAR RELAYS WHICH ARE NOT OF THE CORRECT RATING. NO OTHERS WERE FOUND. THE INVESTIGATION IS CONTINUING TO ATTEMPT TO DETERMINE THE CAUSE OF THE USE OF THE WRONG RELAY.

[225] SOUTH TEXAS 1 DOCKET 50-498 LER 89-016
TECH SPEC VIOLATION DUE TO INADEQUATE PROCEDURAL CONTROL OVER A PLANT MODIFICATION.
EVENT DATE: 071389 REPORT DATE: 081489 NSSS: WE TYPE: PWR

(NSIC 215036) ON JULY 13, 1989 WITH UNIT 1 IN MODE 1 AT 100 PERCENT POWER, IT WAS DISCOVERED THAT THE ROD POSITION DEVIATION MONITOR, WHICH IS DERIVED FROM THE PLANT COMPUTER, HAD NOT BEEN CAPABLE OF ALARMING IN THE CONTROL ROOM SINCE MAY 11, 1989. THE ROD POSITION DEVIATION MONITOR WAS IMMEDIATELY RESTORED TO THE CONTROL ROOM ALARM CRT. WHEN THIS MONITOR IS INOPERABLE THE SURVEILLANCE FREQUENCY OF THE DIGITAL ROD POSITION INDICATION SYSTEM AND THE DEMAND POSITION INDICATION SYSTEM MUST BE INCREASED. BECAUSE THE INOPERABLE STATUS OF THE MONITOR WAS NOT KNOWN, THE INCREASED SURVEILLANCE FREQUENCIES WERE NOT APPLIED WHICH RESULTED IN A VIOLATION OF TECHNICAL SPECIFICATIONS. THE CAUSE OF THIS EVENT WAS INADEQUATE PROCEDURAL CONTROL OF A DESIGN CHANGE TO THE PLANT COMPUTER SOFTWARE. ADDITIONALLY, NO EXPLICIT METHOD EXISTED TO DETERMINE THE OPERABILITY OF THE ROD POSITION DEVIATION MONITOR. THE PROCEDURE WHICH CONTROLS CHANGES TO THE PLANT COMPUTER WILL BE REVISED. A METHOD TO VERIFY OPERABILITY OF THE ROD POSITION DEVIATION MONITOR WILL BE DEVELOPED. OTHER PLANT COMPUTER POINTS WHICH ARE REQUIRED TO SATISFY TECHNICAL SPECIFICATION WILL BE REVIEWED TO ENSURE THAT THEIR OPERABILITY IS ADEQUATELY VERIFIED.

[226] SOUTH TEXAS 2 DOCKET 50-499 LER 89-017
REACTOR TRIP AND PARTIAL LOSS OF OFFSITE POWER DUE TO A MAIN TRANSFORMER FAILURE.
EVENT DATE: 071389 REPORT DATE: 081189 NSSS: WE TYPE: PWR
VENDOR: MCGRAW EDISON CO., POWER SYSTEMS DIV

(NSIC 215037) ON JULY 13, 1989, UNIT 2 WAS IN MODE 1 AT 99 PERCENT POWER. AT 2002 HOURS, AN INTERNAL FAULT OCCURRED IN THE UNIT 2 MAIN STEP-UP TRANSFORMER 2A. THE PROTECTIVE RELAYS TRIPPED THE TURBINE AND ACTUATED THE SWITCHYARD AND GENERATOR BREAKERS TO CLEAR THE FAULT. THE REACTOR TRIPPED ON THE TURBINE TRIP. THE PLANT WAS BROUGHT TO AN ORDERLY COOLDOWN WITH NO UNEXPECTED PRIMARY SYSTEM POST TRIP TRANSIENTS. THE INVESTIGATION OF THE TRANSFORMER FAILURE IS ONGOING. THE MOST PROBABLE CAUSE WAS FAILURE OF THE HIGH SIDE, PHASE A BUSHING. THE TRANSFORMER WILL BE RETURNED TO THE MANUFACTURER FOR REPAIR AND THE BUSHING AND TRANSFORMER WILL BE ANALYZED FURTHER TO DETERMINE THE CAUSE OF THIS EVENT.

[227] ST. LUCIE 1 DOCKET 50-335 LER 89-004
CONTAINMENT FAN COOLER FILTERS LEFT IN PLACE DURING UNIT POWER OPERATION DUE TO INADEQUATE PROCEDURES.
EVENT DATE: 071589 REPORT DATE: 081489 NSSS: CE TYPE: PWR

(NSIC 215020) ON 7/15/89, WHILE ST. LUCIE UNIT ONE WAS SHUTDOWN, IT WAS

DISCOVERED THAT THE REACTOR CONTAINMENT FAN COOLERS (RCFC) HAD FILTER MEDIA IN PLACE DURING POWER OPERATIONS. THE RCFC FILTERS REDUCE FOULING OF THE SYSTEM'S COOLING COILS WHEN THE UNIT IS SHUTDOWN. FILTERS WERE FILTERS WERE INSTALLED DURING THE FEBRUARY 1987 REFUELING OUTAGE, WERE REPLACED DURING THE JULY 1988 REFUELING OUTAGE AND LEFT INSTALLED UNTIL JULY, 1989. THE CAUSE OF THIS EVENT WAS DUE TO INADEQUATE MAINTENANCE INSTRUCTIONS. THE FILTERS WERE REMOVED, AND AN ENGINEERING EVALUATION IS BEING PREPARED TO VERIFY FILTER PERFORMANCE UNDER ACCIDENT CONDITIONS. ST. LUCIE UNIT TWO RCFC WERE VERIFIED TO HAVE FILTERS REMOVED, MAINTENANCE INSTRUCTIONS WILL BE REVISED, AND OTHER SAFETY RELATED COMPONENTS WERE CHECKED FOR IMPROPERLY INSTALLED FILTERS. THIS EVENT WAS DETERMINED NOT TO BE REPORTABLE UNDER 10CFR50.72 OR 50.73 AND IS BEING SUBMITTED FOR INFORMATIONAL PURPOSES.

[228] ST. LUCIE 1 DOCKET 50-335 LER 89-003
AUTOMATIC REACTOR TRIP ON LOW STEAM GENERATOR WATER LEVEL DURING STARTUP DUE TO PROCEDURAL DEFICIENCY.
EVENT DATE: 071789 REPORT DATE: 081489 NSSS: CE TYPE: PWR
VENDOR: ROCHESTER INSTRUMENT SYSTEMS, INC.

(NSIC 215024) ON 7/17/89, ST. LUCIE UNIT 1 WAS IN MODE 2 AND PERFORMING A TURBINE STARTUP. STEAM GENERATOR (SG) LEVELS WERE MAINTAINED WITH THE 1A AND 1B AUXILIARY FEEDWATER (AFW) PUMPS. AT 0254, THE 1B MAIN FEEDWATER (MFW) PUMP WAS STARTED, AFW WAS SECURED AND THE TURBINE WAS ROLLED SHORTLY THEREAFTER SG LEVELS WERE OBSERVED TO BE DECREASING, AND AFW WAS RESTORED. THE UNIT AUTOMATICALLY TRIPPED AT 0300 ON LOW SG LEVEL DUE TO MISMATCHES BETWEEN STEAM LOADS AND FEED FLOW RATES. THE AUDIBLE ANNUNCIATOR FOR THE LOW SG LEVELS PRE-TRIP ALARMS FAILED BEFORE THE TRIP. STANDARD POST TRIP ACTIONS WERE PERFORMED AND THE UNIT WAS STABILIZED IN MODE 3. MFW FLOW TO THE SGS WAS PREVENTED BECAUSE THE MFW BLOCK VALVES HAD NOT BEEN OPENED. PROCEDURES WILL BE CHANGED TO VERIFY THEIR PROPER LINE UP BEFORE STARTING THE TURBINE. AFFECTED AUDIBLE ANNUNCIATORS WERE RETURNED TO SERVICE. A COGNITIVE PERSONNEL ERROR AMONG THE UTILITY LICENSED OPERATORS LED TO THE AUTOMATIC REACTOR TRIP.

[229] ST. LUCIE 2 DOCKET 50-389 LER 89-005
LOSS OF LOAD REACTOR TRIP ON HIGH STEAM GENERATOR LEVEL DUE TO PERSONNEL ERROR.
EVENT DATE: 062689 REPORT DATE: 072689 NSSS: CE TYPE: PWR

(NSIC 214894) ON JUNE 26, 1989 AT 2347 HOURS, ST. LUCIE UNIT 2 REACTOR TRIPPED ON LOSS OF LOAD FROM 22% POWER DURING TURBINE STARTUP. THE LOSS OF LOAD REACTOR TRIP WAS A RESULT OF A TURBINE TRIP DUE TO HIGH STEAM GENERATOR LEVEL IN THE 2A STEAM GENERATOR. THE TRIP WAS UNCOMPLICATED AND THE UNIT WAS QUICKLY STABILIZED IN MODE 3, HOT STANDBY. THE ROOT CAUSE OF THE EVENT WAS COGNITIVE PERSONNEL ERROR BY UTILITY- LICENSED OPERATORS DUE TO LESS THAN ADEQUATE COMMUNICATION BETWEEN THE SHIFT CREW PERFORMING THE TURBINE STARTUP EVOLUTION. THE UTILITY-LICENSED OPERATORS WERE COUNSELED ON THE NEED FOR WELL CONTROLLED EVOLUTIONS, IN WHICH GOOD COMMUNICATION IS OF THE UTMOST IMPORTANCE BY ALL PARTICIPANTS AS A SHORT TERM CORRECTIVE ACTION. THE PLANT TRAINING GROUP WILL EVALUATE THIS EVENT TO DETERMINE APPROPRIATE TRAINING REQUIREMENTS AND METHODS.

[230] SUMMER 1 DOCKET 50-395 LER 89-012
TURBINE TRIP/REACTOR TRIP DUE TO INADVERTENT SHORTING OF STATOR WATER COOLING SIGNAL.
EVENT DATE: 071189 REPORT DATE: 081089 NSSS: WE TYPE: PWR
VENDOR: GENERAL ELECTRIC CO

(NSIC 215002) ON JULY 11, 1989, A TURBINE TRIP/REACTOR TRIP OCCURRED WHILE OPERATING AT 100% REACTOR POWER. TECHNICIANS WORKING INSIDE THE "GENERATOR STATOR COOLING WATER" CABINET INADVERTENTLY SHORTED THE POWER LEADS ON THE

TEMPERATURE CONVERTER CAUSING THE AC POWER FUSE TO BLOW. THIS GAVE A FALSE INDICATION OF LOSS OF GENERATOR STATOR COOLING WATER WHICH CAUSED A TURBINE TRIP AND A REACTOR TRIP DUE TO THE TURBINE TRIPPING ABOVE 50% REACTOR POWER. IN ADDITION TO THE AFOREMENTIONED LOSS, THREE OTHER GENERATING STATIONS TRIPPED WHILE ATTEMPTING TO COMPENSATE FOR THE VARS LOST ON THE GRID WITH THE TURBINE TRIP/REACTOR TRIP. AS A RESULT OF THE LOSS OF FOUR GENERATING STATIONS, THE OFFSITE VOLTAGE TO THE ENGINEERING SAFETY FEATURE (ESF) BUSSES DECREASED BELOW THE MINIMUM ACCEPTABLE VALUE AND A NOTIFICATION OF UNUSUAL EVENT (NUE) WAS DECLARED AT 1510 HOURS. THE LICENSEE'S DISPATCHERS IMMEDIATELY TOOK ACTION TO PLACE ADDITIONAL GENERATING UNITS ON-LINE. BOTH EMERGENCY DIESEL GENERATORS CAME ON-LINE AND SUPPLIED THEIR RESPECTIVE ESF BUSSES. THE OFFSITE VOLTAGE TO THE ESF BUSSES WAS RETURNED TO AN ACCEPTABLE LEVEL AT APPROXIMATELY 1645 HOURS AND THE NUE WAS TERMINATED AT APPROXIMATELY 1710 HOURS. THERE WERE NO PERSONNEL INJURIES OR RELEASES OF RADIOACTIVE MATERIALS AS A RESULT OF THIS EVENT.

[231] SURRY 1 DOCKET 50-280 LER 89-025
UNPLANNED ESF ACTUATION OCCURS WHEN AUXILIARY VENT DAMPERS REALIGNED DURING TESTING DUE TO IMPROPER JUMPER INSTALLATION.
EVENT DATE: 062289 REPORT DATE: 072089 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)

(NSIC 214879) ON JUNE 22, 1989 AT 1131 HOURS, WITH UNITS 1 AND 2 AT COLD SHUTDOWN, A FAULT OCCURRED ON THE 120 VOLT VITAL BUS I-1, CIRCUIT 19, DURING THE PERFORMANCE OF A SPECIAL TEST. THE FAULT TRIPPED THE CIRCUIT'S SUPPLY BREAKER WHICH RESULTED IN THE CLOSURE OF THE FUEL BUILDING NON-FILTERED EXHAUST DAMPER (AOD-VS-101A) AND CLOSURE OF THE VENTILATION VENT STACK ISOLATION DAMPER (AOD-VS-112A). THIS EVENT WAS REPORTED TO THE NUCLEAR REGULATORY COMMISSION AS A FOUR HOUR NON-EMERGENCY REPORT AS AN UNPLANNED ACTUATION OF AN ENGINEERED SAFETY FEATURES COMPONENT PER 10CFR50.72(B)(2)II. THE FAULT WAS CAUSED BY A JUMPER INSTALLED TO FACILITATE PERFORMANCE OF THE TEST. ONE LEAD OF THE JUMPER HAD CONTACTED A NEUTRAL TERMINAL IN THE CIRCUIT CREATING THE FAULT. THE JUMPER WAS REMOVED AND PLACED IN A NEW LOCATION, AND THE TEST WAS SATISFACTORILY COMPLETED. THE PERSON WHO INSTALLED THE JUMPER WAS REMINDED OF THE NEED TO EXERCISE CAUTION WHEN INSTALLING JUMPERS TO ENSURE THEIR PROPER INSTALLATION.

[232] SURRY 1 DOCKET 50-280 LER 89-024
CONTROL ROOM/RELAY ROOM CHILLER TRIP DUE TO INADEQUATE SW SUPPLY.
EVENT DATE: 062489 REPORT DATE: 072089 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)

(NSIC 214878) ON JUNE 24, 1989 AT 0928 HOURS, WITH UNITS 1 AND 2 AT COLD SHUTDOWN (CSD), ONE OF THE THREE CONTROL ROOM/RELAY ROOM (CR/RR) CHILLERS 1-VS-E- LC, TRIPPED ON HIGH DISCHARGE PRESSURE. THE "B CHILLER WAS NOT RECEIVING AN ADEQUATE SUPPLY OF SERVICE WATER (SW) AND WAS MANUALLY STOPPED. THIS OCCURRED AFTER ONE OF THE TWO SIX INCH SW SUPPLY LINES TO THE CR/RR CHILLERS WAS ISOLATED IN SUPPORT OF MAINTENANCE ACTIVITIES. THE "A" CHILLER WAS OPERABLE, BUT NOT OPERATING AT THE TIME. THIS EVENT IS CONTRARY TO TECHNICAL SPECIFICATION (T.S.) 3.14.C. THE CAUSE OF THIS EVENT WAS AN INADEQUATE SUPPLY OF SW CAUSED BY THE ISOLATION OF ONE OF THE TWO SIX INCH SW SUPPLY LINES TO THE CHILLERS. IT HAS SINCE BEEN DETERMINED THAT ONE SIX INCH SW LINE CANNOT PROVIDE SUFFICIENT FLOW TO MAINTAIN TWO OPERATING CR/RR CHILLERS. A DESIGN CHANGE IS BEING IMPLEMENTED TO INSTALL THREE EIGHT INCH SW SUPPLY LINES TO THE CR/RR CHILLERS.

[233] SURRY 1 DOCKET 50-280 LER 89-026
TURBINE TRIP/REACTOR TRIP ON HI-HI STEAM GENERATOR LEVEL FOLLOWING TURBINE RUNBACK CAUSED BY A BLOWN FUSE IN NI-41.
EVENT DATE: 070989 REPORT DATE: 080789 NSSS: WE TYPE: PWR
VENDOR: COPES-VULCAN, INC.

(NSIC 214982) ON 7/9/89 WITH UNIT 1 AT 63% POWER, INSTRUMENT TECHNICIANS WERE CALIBRATING POWER RANGE NUCLEAR INSTRUMENTATION (NI) WHEN A FAILURE OF THE NI OCCURRED. THIS RESULTED IN A SPURIOUS NI DROPPED ROD SIGNAL AND AUTOMATIC INITIATION OF A 30% POWER TURBINE RUNBACK. FOLLOWING THE TURBINE RUNBACK OPERATORS HAD DIFFICULTY STABILIZING STEAM GENERATOR LEVELS DUE TO INAPPROPRIATE OPERATOR RESPONSE AND EQUIPMENT MALFUNCTION. A TURBINE TRIP/REACTOR TRIP OCCURRED DUE TO A HIGH-HIGH LEVEL CONDITION IN THE "B" STEAM GENERATOR TWO MINUTES AND 45 SECONDS AFTER INITIATION OF THE TURBINE RUNBACK. THE FAILURE OF THE POWER RANGE NI WAS DUE TO A BLOWN CONTROL POWER FUSE CAUSED BY THE USE OF A VOLTMETER WITH AN UNGROUNDED POWER SUPPLY. THE PERSONNEL INVOLVED WERE DISCIPLINED. THIS EVENT AND SIMILAR EVENTS RELATED TO IMPROPER USE OF METERS WILL BE DISCUSSED DURING THE INSTRUMENT TECHNICIANS CONTINUING TRAINING CLASSES. INSTRUMENT TECHNICIANS HAVE BEEN ISSUED INSTRUCTIONS TO USE GROUNDED METERS FOR TESTING UNLESS SPECIFICALLY DIRECTED OTHERWISE.

[234] SURRY 1 DOCKET 50-280 LER 89-028
EDG UNDERGROUND FUEL OIL TANKS ACCESS PLUGS REMOVED WITH NO ADMINISTRATIVE CONTROL DUE TO INADEQUATE PROCEDURE.
EVENT DATE: 071389 REPORT DATE: 081289 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)

(NSIC 215016) ON JULY 13, 1989 AT 1500 HOURS, WITH UNIT 1 AT 65% POWER AND UNIT 2 IN COLD SHUT DOWN, AN OPERATOR DISCOVERED THE ACCESS PLUG REMOVED FROM THE ENTRANCE TO EACH OF THE TWO UNDERGROUND FUEL OIL TANKS (EIIS-TK). THE PLUGS HAD BEEN REMOVED EARLIER THAT DAY AT 0900 HOURS TO PERMIT TESTING OF THE FUEL OIL TANKS. WITH THE PLUGS REMOVED, BOTH UNDERGROUND FUEL OIL (EIIS-TK) TANKS WERE CONSIDERED TECHNICALLY INOPERABLE DUE TO DEGRADED MISSILE PROTECTION. THIS CONDITION IS CONTRARY TO TECHNICAL SPECIFICATION 3.16.A.1. THE FUEL OIL SAMPLE PROCEDURE DID NOT PROVIDE GUIDANCE FOR ESTABLISHING ADMINISTRATIVE CONTROL WHEN REMOVING THE ACCESS PLUGS. THE ACCESS PLUGS WERE REINSTALLED UPON DISCOVERY. THE SAMPLE PROCEDURE WILL BE REVISED TO PROVIDE SPECIFIC INSTRUCTIONS REGARDING ADMINISTRATIVE CONTROL REQUIREMENTS WHEN REMOVING THE UNDERGROUND FUEL OIL TANK ACCESS PLUGS.

[235] SURRY 1 DOCKET 50-280 LER 89-029
INTAKE CANAL LEVEL INSTRUMENTATION INOPERABLE DUE TO INSTALLATION OF STOP LOGS IN THE INTAKE STRUCTURE AND CHANNEL NOT PLACED IN TRIP.
EVENT DATE: 071489 REPORT DATE: 081389 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)

(NSIC 215017) ON JULY 14, 1989, UNIT 1 WAS OPERATING AT 18% POWER AND UNIT 2 WAS IN COLD SHUTDOWN. AT 1210 HOURS, STOP LOGS WERE INSTALLED IN THE UNIT 2 "A". CIRCULATING WATER (CW) HIGH LEVEL INTAKE STRUCTURE. ISOLATION OF THE INTAKE STRUCTURE RESULTED IN THE INOPERABILITY OF AN INTAKE CANAL LEVEL SENSOR CHANNEL. TECHNICAL SPECIFICATION 3.7 REQUIRES THAT AN INOPERABLE CHANNEL BE PLACED IN "TRIP" CONDITION WITHIN ONE HOUR. CONTRARY TO THE ABOVE REQUIREMENT, THE INOPERABLE CHANNEL WAS NOT PLACED IN "TRIP" WITHIN ONE HOUR. FURTHER REVIEW OF THIS EVENT DETERMINED THAT A SIMILAR EVENT HAD OCCURRED ON JULY 2, 1989. THE CAUSE OF THESE EVENTS HAS BEEN ATTRIBUTED TO PERSONNEL ERROR. OPERATORS FAILED TO RECOGNIZE THAT INSTALLATION OF THE STOP LOGS DEFEATED A CANAL LEVEL CHANNEL. THE AFFECTED CHANNEL WAS IDENTIFIED AS BEING INOPERABLE AND WAS PLACED IN TRIP. THE STOP LOGS WERE SUBSEQUENTLY REMOVED FROM THE 2A INTAKE STRUCTURE AND THE LEVEL SENSOR CHANNEL WAS RESTORED TO NORMAL OPERATION. PROCEDURES WILL BE DEVELOPED FOR STOP LOG INSTALLATION.

[236] SURRY 1 DOCKET 50-280 LER 89-030
CHARGING PUMP SERVICE WATER PUMP AIR BOUND AFTER AIR ENTERED THE SERVICE WATER SYSTEM.
EVENT DATE: 071889 REPORT DATE: 081789 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)

(NSIC 215119) ON JULY 18, 1989 WITH UNIT 1 AT 100% POWER AND UNIT 2 IN COLD SHUTDOWN, FOLLOWING THE MANIPULATION OF A SERVICE WATER (SW) VALVE TO THE UNIT 2 BEARING COOLING HEAT EXCHANGERS, THE UNIT 1 CHARGING PUMP SW PUMPS BECAME AIR BOUND AND THE PUMPS WERE DECLARED INOPERABLE. THIS IS CONTRARY TO TECHNICAL SPECIFICATION 3.3.A.7. THE CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO AIR ENTERING THE SW LINES THAT SUPPLY THE CHARGING PUMP SW PUMPS. THE AFFECTED PUMPS WERE VENTED AND RETURNED TO SERVICE. ADDITIONAL HIGH POINT VENTS ARE BEING INSTALLED. A PROCEDURE FOR PERIODIC VENTING HAS BEEN DEVELOPED AND RESTORATION PROCEDURES ENHANCED. IN ADDITION, A DESIGN CHANGE BEING IMPLEMENTED TO INCREASE THE SIZE AND NUMBER OF THE SW SUPPLY LINES TO THE PUMPS HAS APPROPRIATE VENTS TO FACILITATE REMOVAL OF ENTRAPPED AIR. ENGINEERING IS ALSO CONTINUING THEIR INVESTIGATION OF THE EVENT TO DETERMINE IF OTHER ACTIONS ARE REQUIRED.

[237] SURRY 1 DOCKET 50-280 LER 89-031
CHARGING PUMP SERVICE WATER PUMPS AIR BOUND AND CONTROL ROOM CHILLERS TRIP AFTER AIR ENTERED THE SERVICE WATER SYSTEM.
EVENT DATE: 072389 REPORT DATE: 082289 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)

(NSIC 215120) ON JULY 23, 1989 AT 1747 HOURS, WITH UNIT 1 AT 100 POWER AND UNIT 2 AT COLD SHUTDOWN, FOLLOWING MANIPULATION OF A SERVICE WATER (SW) VALVE TO THE UNIT 2 BEARING COOLING WATER HEAT EXCHANGERS, DISCHARGE PRESSURES FOR THE CONTROL ROOM/RELAY ROOM (CR/RR) AIR CONDITIONING CHILLERS' SW PUMPS AND THE UNIT 1 AND UNIT 2 CHARGING PUMP SW PUMPS DECREASED DUE TO AIR BINDING IN THE PUMPS. THE PUMPS WERE DECLARED INOPERABLE. ALSO, THE TWO OPERATING CR/RR CHILLERS TRIPPED ON HIGH CONDENSER DISCHARGE PRESSURE. THIS IS CONTRARY TO TECHNICAL SPECIFICATION 3.3.A.7 AND 3.23.C. THE CAUSE OF THE EVENT HAS BEEN ATTRIBUTED TO AIR ENTERING THE SW LINES THAT SUPPLY THE PUMPS. THE AFFECTED PUMPS WERE VENTED AND RETURNED TO SERVICE AND THE CR/RR CHILLERS WERE RETURNED TO SERVICE. ADDITIONAL HIGH POINT VENTS ARE BEING INSTALLED. A PROCEDURE FOR PERIODIC VENTING HAS BEEN DEVELOPED AND THE ABNORMAL PROCEDURE FOR RESTORATION HAS BEEN ENHANCED. IN ADDITION, A DESIGN CHANGE BEING IMPLEMENTED TO INCREASE THE SIZE AND NUMBER OF SW SUPPLY LINES TO THE PUMPS HAS PROVISIONS FOR APPROPRIATE VENTS TO FACILITATE REMOVAL OF ENTRAPPED AIR. ENGINEERING IS CONTINUING THEIR INVESTIGATION OF THE EVENT TO DETERMINE IF OTHER ACTIONS ARE REQUIRED.

[238] SURRY 1 DOCKET 50-280 LER 89-033
AUXILIARY VENTILATION FANS TRIPPED DUE TO LOW AIR PRESSURE CREATED BY LEAKING AOD ACTUATORS.
EVENT DATE: 073189 REPORT DATE: 083089 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)

(NSIC 215122) ON JULY 31, 1989 AT 1410 HOURS, WITH UNIT 1 AT 100% POWER AND UNIT 2 AT COLD SHUTDOWN, AN OPERATOR WAS OPENING FOUR VENTILATION EXHAUST AIR OPERATED DAMPERS (AOD) WHEN SIX AUXILIARY BUILDING VENTILATION FANS TRIPPED. THESE FANS ARE REQUIRED TO TRIP UPON INITIATION OF AN ENGINEERED SAFETY FEATURES (ESF) SIGNAL. HOWEVER, NO ESF SIGNAL WAS PRESENT AT THE TIME. THE EVENT WAS DUE TO A DECREASE IN PRESSURE IN THE FAN'S PNEUMATIC CONTROL HEADER CAUSED BY THE OPENING OF THE FOUR AIR OPERATED DAMPERS. AIR LEAKAGE FROM THE HEADER AND DEGRADED DAMPER ACTUATORS RESULTED IN THE PRESSURE DROP. THE LEAKING HEADER AND THE DAMPER'S ACTUATORS WILL BE REPAIRED. THIS EVENT WAS REPORTED TO THE NUCLEAR REGULATORY COMMISSION AS A FOUR HOUR NON-EMERGENCY REPORT DUE TO AN UNPLANNED ACTUATION OF AN ESF COMPONENT PER 10CFR50.72(B)(2)(II).

[239] SURRY 1 DOCKET 50-280 LER 89-034
FAILURE TO SAMPLE SERVICE WATER FROM COMPONENT COOLING HEAT EXCHANGERS WITHIN TWELVE HOURS DUE TO PERSONNEL ERROR.
EVENT DATE: 080289 REPORT DATE: 083089 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: SURRY 2 (PWR)

(NSIC 215123) ON 8/2/89 WITH UNIT 1 AT 100% POWER AND UNIT 2 AT COLD SHUTDOWN, IT WAS DISCOVERED THAT THE TECH SPEC FOR SAMPLING THE SERVICE WATER (SW) FROM THE "B" AND "C" COMPONENT COOLING (CC) HEAT EXCHANGERS HAD NOT BEEN PERFORMED WITHIN THE SPECIFIED TIME INTERVAL. THE SAMPLES WERE TAKEN 12 HOURS AND 44 MINUTES APART. TECH SPEC TABLE 3.7-5(A) REQUIRES THAT SW EFFLUENT FROM THE CC HEAT EXCHANGERS BE SAMPLED AND ANALYZED FOR RADIOACTIVE CONTAMINATION EVERY 12 HOURS WHENEVER THE SW RADIATION MONITORS ARE OUT OF SERVICE. THE CAUSE OF THE EVENT WAS INEFFECTIVE COMMUNICATION BETWEEN THE OPERATORS RESPONSIBLE FOR ENSURING THE SAMPLES ARE TAKEN. THE EVENT WAS DISCUSSED WITH THE OPERATORS AND DISCIPLINARY ACTION WAS TAKEN. IN ADDITION, CONTROL ROOM LOGS WILL BE MODIFIED.

[240] SUSQUEHANNA 1 DOCKET 50-387 LER 89-015
IMPROPERLY PLACED JUMPER DURING SURVEILLANCE CAUSES RHR SHUTDOWN COOLING VALVE CLOSURE.
EVENT DATE: 052189 REPORT DATE: 062189 NSSS: GE TYPE: BWR

(NSIC 214469) AT 1357 ON 5/21/89, WITH UNIT 1 IN ITS FOURTH REFUELING OUTAGE, A PRIMARY CONTAINMENT ISOLATION SYSTEM LOGIC FUSE WAS INADVERTENTLY BLOWN WHEN AN ELECTRICIAN ACCIDENTALLY CAUSED A SHORT CIRCUIT TO GROUND WHILE ATTEMPTING TO INSTALL A TEMPORARY JUMPER IN A DIFFICULT LOCATION IN SUPPORT OF A SURVEILLANCE TESTING PROCEDURE. THE BLOWN FUSE CAUSED THE RESIDUAL HEAT REMOVAL SYSTEM SHUTDOWN COOLING (SDC) SUCTION INBOARD ISOLATION VALVE, HV-151-F009, TO CLOSE WHICH RESULTED IN A TEMPORARY LOSS OF SDC. THIS CONSTITUTED AN UNPLANNED ACTUATION OF AN ENGINEERED SAFETY FEATURE AND WAS HENCE REPORTABLE UNDER 10CFR50.73(A)(2)(IV). THE TESTING WAS SUSPENDED, THE BLOWN FUSE WAS REPLACED, HV-151-F009 WAS REOPENED AND SDC WAS RESTORED AT 1457 ON 5/21/89. REACTOR COOLANT TEMPERATURE INCREASE DURING THE EVENT WAS INSIGNIFICANT. THE SURVEILLANCE PROCEDURE WAS REVISED TO FACILITATE INSTALLATION OF THE TEMPORARY JUMPER AND SUBSEQUENTLY COMPLETED WITHOUT FURTHER INCIDENTS. A PROJECT TO MINIMIZE INADVERTENT EQUIPMENT ACTUATIONS DURING TESTING WAS INITIATED IN 1988 AND IS BEING ACTIVELY PURSUED. SINCE THE HV-151-F009 FAILED IN ITS SAFE (CLOSED) POSITION REGARDING ITS PRIMARY CONTAINMENT ISOLATION FUNCTION AND SINCE AN ALTERNATE METHOD OF DECAY HEAT REMOVAL WAS AVAILABLE, THERE WAS NO COMPROMISE TO THE HEALTH OR SAFETY OF THE PUBLIC.

[241] SUSQUEHANNA 1 DOCKET 50-387 LER 89-020
UNPLANNED ESF ACTUATIONS AS A RESULT OF CIRCUIT BREAKER TERMINATION/MOUNTING HARDWARE FAILURES IN RPS DISTRIBUTION PANEL.
EVENT DATE: 070289 REPORT DATE: 073089 NSSS: GE TYPE: BWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 214893) ON 7/2/89 AND 7/5/89, WITH THE UNIT AT 100% POWER, TERMINATION/MOUNTING PROBLEMS WITH REACTOR PROTECTION SYSTEM (RPS) CIRCUIT BREAKER CB8B CAUSED NUMEROUS POWER INTERRUPTIONS IN THE "B" RPS DISTRIBUTION SYSTEM, EACH OF WHICH RESULTED, PER DESIGN, IN ACTUATIONS OF SEVERAL ENGINEERED SAFETY FEATURE (ESF) SYSTEMS AND COMPONENTS. THE ESF ACTUATION TRIP SIGNALS ON 7/2/89, WHICH OCCURRED AT 0804, 0827 AND 0933 AND WERE CAUSED BY CIRCUIT BREAKER CB8B TRIPS, WERE RESET AT 1115 ON 7/2/89 AFTER CB8B WAS REPLACED. THE ESF ACTUATION TRIP SIGNALS ON 7/5/89, WHICH OCCURRED AT 1005 AND WERE CAUSED BY THE "B" RPS BUS ELECTRICAL PROTECTION ASSEMBLY (EPA) SUPPLY BREAKERS TRIPPING, WERE RESET AT 1205 AFTER TERMINATION/MOUNTING FAILURES WITH CIRCUIT BREAKER CB8B WERE REWORKED. DURING THE INVESTIGATION ON 7/2/89 IT WAS REVEALED THAT ONE OF THE TRIPS OF CIRCUIT BREAKER CB8B, WHICH IS RATED FOR 100 AMPS, OCCURRED AT 65 AMPS.

CB8B WAS REPLACED AND IT HAS SINCE BEEN DETERMINED THAT THE REPLACED BREAKER WAS IN PROPER WORKING ORDER. INVESTIGATIONS ON 7/5/89 OF CIRCUIT BREAKER CB8B TERMINATION/MOUNTING CONFIGURATION REVEALED A LOOSE INPUT CONNECTION, WHICH IS SUSPECTED TO HAVE CAUSED THE BREAKER CB8B TRIPS ON 7/2/89, AND A CRACKED GROUND INSULATOR WHICH CAUSED THE EPA BREAKER TRIPS ON 7/5/89.

[242] TROJAN DOCKET 50-344 LER 89-008 REV 01
UPDATE ON SPENT FUEL POOL EXHAUST SYSTEM INOPERABLE WHILE MOVING FUEL.
EVENT DATE: 042189 REPORT DATE: 080189 NSSS: WE TYPE: PWR

(NSIC 214956) ON APRIL 4, 1989 THE PLANT WAS IN MODE 6 (REFUELING) WITH REACTOR COOLANT SYSTEM CONDITIONS OF ZERO PSIG AND 90 DEGREES F. INITIAL TESTING OF THE SPENT FUEL POOL EXHAUST SYSTEM ON APRIL 13, PRIOR TO INITIATING FUEL MOVEMENT, SHOWED THAT CERTAIN DOORS BEING OPEN WOULD NOT ALLOW THE SPENT FUEL POOL EXHAUST SYSTEM TO ACHIEVE THE REQUIRED NEGATIVE 0.10 INCHES WATER GAUGE PRESSURE. AS A RESULT, PLANT PERSONNEL WERE VERBALLY TOLD NOT TO OPEN DOORS TO THE OUTSIDE IN THE FUEL AND AUXILIARY BUILDINGS. FUEL MOVEMENT WAS IN PROGRESS WHEN A DOOR FROM THE AUXILIARY BUILDING TO THE OUTSIDE WAS DISCOVERED OPEN AT 1030 ON APRIL 21, 1989 DURING A ROUTINE TOUR BY OPERATIONS PERSONNEL. WHEN TESTED WITH THIS DOOR OPEN, THE SPENT FUEL POOL EXHAUST SYSTEM COULD ONLY ACHIEVE A NEGATIVE PRESSURE OF 0.09 INCHES WATER GAUGE. IMMEDIATE CORRECTIVE ACTION WAS TO CLOSE THE DOOR. INTERIM ACTION WAS TO PLACE SIGNS ON OUTSIDE DOORS IN THE FUEL AND AUXILIARY BUILDINGS THAT THE DOORS WERE NOT TO BE OPENED WITHOUT PERMISSION FROM THE SHIFT SUPERVISOR. PERMANENT CORRECTIVE ACTION WILL BE TO ESTABLISH FORMAL ADMINISTRATIVE CONTROLS FOR DOORS THAT EFFECT THE OPERABILITY OF THE SPENT FUEL POOL EXHAUST SYSTEM. THIS EVENT HAD NO EFFECT ON THE HEALTH AND SAFETY OF THE PUBLIC AS THE SPENT FUEL POOL EXHAUST SYSTEM WAS NOT NEEDED TO MITIGATE THE CONSEQUENCES OF AN ACCIDENT.

[243] TROJAN DOCKET 50-344 LER 89-013
COGNITIVE PERSONNEL ERRORS IN DIRECTING WORK RESULTED IN AN INADVERTENT REACTOR TRIP SIGNAL WHILE SHUT DOWN.
EVENT DATE: 071389 REPORT DATE: 081489 NSSS: WE TYPE: PWR

(NSIC 215021) ON 7/13/89, FINAL PREPARATIONS WERE BEING MADE FOR ENTERING MODE 4 FOLLOWING THE REFUELING OUTAGE. THE STEAM GENERATOR LEVELS WERE BELOW THE LOW-LOW LEVEL TRIP DUE TO A SPECIAL EVOLUTION IN PROGRESS TO CONDUCT BORIC ACID CREVICE FLUSHING OF THE STEAM GENERATORS. TO ALLOW THE REACTOR TRIP BREAKERS TO BE RESET FOR BREAKER AND COLD ROD DROP TESTING, THE DOORS TO SEVERAL OF THE REACTOR PROTECTION SYSTEM RACKS WERE OPEN TO ALLOW INSTALLATION OF ADJUSTMENT POTENTIOMETERS THAT PROVIDED SIMULATED STEAM GENERATOR LEVELS. BECAUSE TRIP BREAKER TESTING WAS COMPLETED, ROD DROP TESTING HAD BEEN POSTPONED, AND IT WAS DESIRED THAT THE RACK DOORS BE SHUT FOR ENTRY INTO MODE 4, THE MANAGER OF OPERATIONS AND MAINTENANCE (O&M) DIRECTED THE POTENTIOMETERS BE REMOVED AND THE DOORS SHUT. WHEN THE TEST SIGNALS WERE REMOVED THE ACTUAL LIVE SIGNALS RESULTED IN A STEAM GENERATOR LOW-LOW LEVEL REACTOR TRIP SIGNAL CAUSING THE REACTOR TRIP BREAKERS TO OPEN. THE RODS WERE NOT LATCHED NOR WERE THE ROD DRIVE MOTOR GENERATORS RUNNING. THIS EVENT HAD NO EFFECT ON THE PUBLIC HEALTH AND SAFETY. THE CAUSES OF THIS EVENT WERE THAT BOTH MANAGEMENT AND THE TECHNICIANS BYPASSED NORMAL WORK CONTROL SYSTEMS, AND THAT INADEQUATE CONTROLS WERE IN PLACE FOR THE REACTOR PROTECTION RACKS. CORRECTIVE ACTIONS WILL BE TO DEVELOP CONTROLS FOR THE REACTOR PROTECTION RACKS, AND TO ISSUE A MEMORANDUM FOR ALL PLANT PERSONNEL.

[244] TROJAN DOCKET 50-344 LER 89-015
HYDROGEN GAS SUPPLY SYSTEM ISSUES.
EVENT DATE: 071389 REPORT DATE: 081489 NSSS: WE TYPE: PWR

(NSIC 215047) A REVIEW WAS PERFORMED ON THE HYDROGEN GAS SUPPLY SYSTEM AS A

RESULT OF NRC INFORMATION NOTICE 87-20. THIS REVIEW NOTED PGE-1012 DOES NOT ADEQUATELY ADDRESS THE POSSIBILITY OF HYDROGEN GAS LEAKS IN THE AUX. BLDG. ALSO, IT WAS DETERMINED THAT EXISTING VENTILATION FLOW RATES AND HIGH CLOSURE SETPOINTS ON HYDROGEN GAS SUPPLY EXCESS FLOW VALVES (GS037, GS038, AND GS162) COULD RESULT IN EXCESSIVE ACCUMULATION OF HYDROGEN IN CERTAIN AREAS OF THE AUX. BLDG. IF THE HYDROGEN LINE TO THE VOLUME CONTROL TANK (VCT) BREAKS. FINALLY, THE NRC NOTED DURING CHEMISTRY TEAM INSPECTION 89-07, DURING THE WEEK OF APRIL 17, 1989, THE PLACEMENT OF THE HYDROGEN GAS BULK STORAGE TANKS ON THE ROOF OF THE CONTROL BUILDING AND THE HYDROGEN GAS BULK STORAGE TANKS ON THE ROOF OF THE CONTROL BUILDING AND THE HYDROGEN FILL STATION ON THE NORTH WALL FOR THE AUXILIARY BUILDING, WHILE IN COMPLIANCE WITH THE TROJAN FSAR, DID NOT CONFORM TO THE GUIDE LINES PROVIDED IN ELECTRIC POWER RESEARCH INSTITUTE (EPRI) NP-5283-SR-A, SPECIAL REPORT, SEPTEMBER 1987 "GUIDELINES FOR PERMANENT BWR HYDROGEN WATER CHEMISTRY INSTALLATIONS - 1987 REVISION." ACTIONS HAVE BEEN TAKEN TO RELO[illegible] THE STORAGE TANKS AND FILL STATION, AND TO INSTALL NEW EXCESS FLOW VALVE IN [illegible] TO THE VCT. CONFIGURATIONS HAVE BEEN EVALUATED AND CONFIRMED TO MEET T[illegible] GU[illegible] OF EPRI NP-5283-SR-A.

[245] TROJAN DOCKET 50[illegible] [illegible]9-01[illegible]
INADEQUATE PROCEDURE AND PERSONNEL ERRORS RESULT I[illegible] WITH [illegible]
CONTAINMENT RECIRCULATION SUMP PROTECTIVE SCREEN N[illegible]
EVENT DATE: 071789 REPORT DATE: 081689 NSSS[illegible]

(NSIC 215071) ON JULY 17, 1989 THE PLANT WAS IN [illegible] WITH REACTOR COOLANT SYSTEM (RCS) CONDITIONS OF 360 [illegible] WHEN AN INSPECTION OF THE CONTAINMENT RECIRCULATION SUMP [illegible]ED [illegible] MESH SCREEN ON THE TOP OF THE CONTAINMENT RECIRCULATI[illegible] TRASH [illegible] NOT INSTALLED. THIS INSPECTION WAS BEING CONDUCTED DU[illegible] SIGNI[illegible] AMOUNTS OF DEBRIS DISCOVERED WITHIN THE SCREENED AREA OF THE CONTAINMENT [illegible]RCULATION SUMP. THE SCREEN IS PART OF THE DESIGN OF THE CONTAINMENT RECIRCULATION SUMP AND THEREFORE THE PLANT WAS OUTSIDE ITS DESIGN BASIS. THE REASON FOR THE MESH SCREEN NOT BEING INSTALLED COULD NOT BE SPECIFICALLY DETERMINED BUT A CONSTRUCTION ERROR (NOT INSTALLED) OR A FAILURE TO REINSTALL IT EARLY IN THE PLANT LIFE, AFTER REMOVAL FOR VORTEX TESTING OF THE RECIRCULATION SUCTION PIPING INLET, APPEAR TO BE LIKELY CAUSES FAILURE TO DETECT THIS CONDITION EARLIER THAN 1989 IS ATTRIBUTED TO AN INADEQUATE INSPECTION OF THE CONTAINMENT RECIRCULATION SUMP. CORRECTIVE ACTIONS WERE TO INSTALL THE MISSING SCREEN, REPAIR DAMAGED PORTIONS OF THE EXISTING SCREEN, REMOVE THE FOREIGN MATERIALS FROM THE AREA, REVISE THE INSPECTION PROCEDURE, AND INSPECT CONTAINMENT USING THE REVISED PROCEDURE. THE EFFECTS OF FOREIGN MATERIAL IN THE CONTAINMENT RECIRCULATION SUMP IS UNDER EVALUATION.

[246] TURKEY POINT 3 DOCKET 50-250 LER 88-026 REV 01
UPDATE ON UNITS 3 AND 4 OUTSIDE THE FINAL SAFETY ANALYSIS REPORT DESIGN BASIS WITH REGARD TO HURRICANE FLOOD PROTECTION.
EVENT DATE: 110788 REPORT DATE: 082189 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: TURKEY POINT 4 (PWR)

(NSIC 215056) ON NOVEMBER 17, 1988, AT APPROXIMATELY 1900, WITH UNIT 3 IN COLD SHUTDOWN AND UNIT 4 DEFUELED, IT WAS DETERMINED THAT UNITS 3 AND 4 WERE OUTSIDE THEIR DESIGN BASIS WITH REGARD TO HURRICANE FLOOD PROTECTION. THE FINAL SAFETY ANALYSIS REPORT STATES, "THE UNIT IS DESIGNED FOR A HURRICANE TIDE TO AN ELEVATION OF +20 FEET WITH WAVE RUN-UP TO AN ELEVATION OF 22.5 FEET ON THE EAST SIDE OF THE UNIT. THE FOLLOWING CONDITIONS WERE IDENTIFIED BY A QUALITY ASSURANCE AUDITOR: 1) THE DIESEL OIL TRANSFER PUMPS ARE MOUNTED AT ELEVATION 19.0 FEET WITHOUT FLOOD PROTECTION. 2) A SECTION OF A FLOOD PROTECTION WALL, APPROXIMATELY 8 FEET IN LENGTH, BETWEEN THE EMERGENCY DIESEL GENERATOR (EDG) BUILDING AND THE UNIT 3 SWI[illegible] ENCLOSURE HAS BEEN TEMPORARILY REMOVED AS A PORTION OF A PLANT MODIFI[illegible] THE STOPLOGS ON THE EAST FACE OF THE

AUXILIARY BUILDING PROVIDE PROTECTION ONLY TO ELEVATION 20 FEET. ITEMS 1 AND 2 ABOVE WERE CAUSED BY DESIGN ERROR. ITEM 3 WAS CAUSED BY THE LACK OF PLANT DRAWINGS THAT CLEARLY IDENTIFY STOP LOG DETAILS. AS AN INTERIM CORRECTIVE ACTION IN THE EVENT OF A HURRICANE WARNING, A TEMPORARY FLOOD PROTECTION DIKE WILL BE ERECTED USING SANDBAGS AND POLYETHYLENE SHEET. AS LONG TERM CORRECTIVE ACTIONS, PLANT MODIFICATIONS WILL BE PERFORMED TO PROVIDE FLOOD PROTECTION FOR THE ABOVE CONCERNS.

[247] TURKEY POINT 3 DOCKET 50-250 LER 89-010
OVERLOADED BREAKER RESULTS IN LOSS OF POWER TO VITAL INSTRUMENT RACK AND AUTOMATIC ISOLATION OF CONTROL ROOM AND CONTAINMENT VENTILATION.
EVENT DATE: 061689 REPORT DATE: 071789 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: TURKEY POINT 4 (PWR)

(NSIC 214914) ON 6/16/89, AT 1115, WITH UNIT 3 IN COLD SHUTDOWN AND UNIT 4 IN STARTUP (MODE 2 AT 2% POWER), INSTRUMENTATION AND CONTROL PERSONNEL PLUGGED AN OSCILLOSCOPE INTO A CONVENIENCE RECEPTABLE LOCATED IN RACK NO. 67. UPON ENERGIZING THE OSCILLOSCOPE, A BREAKER SUPPLYING VITAL A/C TO PROCESS RADIATION MONITORING (PRM) RACK NO. 66 TRIPPED. THIS RESULTED IN THE LOSS OF POWER TO THE RACK NO. 66, AND AUTOMATIC ISOLATION OF THE CONTAINMENT VENTILATION AND CONTROL ROOM VENTILATION SYSTEMS. THE BREAKER WAS CLOSED AT APPROX. 1122. AND THE CONTROL ROOM VENTILATION SYSTEM WAS RETURNED TO ITS NORMAL ALIGNMENT AT APPROX. 1205. THE CAUSE OF THE EVENT APPEARS TO BE OVERLOADING OF BREAKER EP08-19. AN ENGINEERING PACKAGE HAS BEEN PREPARED THAT WILL PROVIDE THE PRMS RACK WITH AN APPROPRIATELY SIZED VITAL A/C POWER SOURCE. THE SUBJECT RECEPTABLE HAS BEEN REMOVED FROM SERVICE. A CLEARANCE TAG HAS BEEN TAPED ACROSS THE A/C OUTLET TO RENDER IT UNUSUABLE, AND THE TAG STATES "DO NOT USE THIS RECEPTABLE." THE OSCILLOSCOPE WIRING WAS CONFIRMED TO BE CORRECT. A STUDY WILL BE CONDUCTED TO IDENTIFY OTHER RECEPTACLES POWERED BY VITAL A/C, AND TO DETERMINE IF DETERMINATING OR REWIRING TO ANOTHER POWER SOURCE IS APPROPRIATE.

[248] TURKEY POINT 3 DOCKET 50-250 LER 89-011
TWO SAFEGUARDS ACTUATIONS DUE TO PERSONNEL ERROR.
EVENT DATE: 061789 REPORT DATE: 071789 NSSS: WE TYPE: PWR

(NSIC 214808) ON JUNE 16, 1989, WITH UNIT 3 IN MODE 5, THE "A" TRAIN SAFEGUARDS LOGIC WAS BEING RE-ENERGIZED. AT APPROXIMATELY 1613, WHEN THE MANUAL BLOCK BUTTONS WERE RELEASED, IN ACCORDANCE WITH PROCEDURE, AN "A" TRAIN SAFETY INJECTION (SI) SIGNAL WAS PRODUCED. THE INSTRUMENT AND CONTROLS TECHNICIANS INVESTIGATING THE EVENT IDENTIFIED AND REPLACED TWO SWITCH CONTACT BLOCKS WHICH SHOWED SIGNS OF PHYSICAL DAMAGE. THESE DAMAGED SWITCH CONTACT BLOCKS WERE CONSIDERED TO BE THE CAUSE OF THE ACTUATION. ACTIVITIES TO RETURN POWER TO THE SAFEGUARD LOGIC WERE RESUMED, THIS TIME STARTING WITH THE "B" SAFEGUARDS TRAIN. AT APPROXIMATELY 2149, WHEN THE MANUAL BLOCK BUTTONS WERE RELEASED A "B" TRAIN SI SIGNAL WAS PRODUCED. THE ROOT CAUSE OF THIS EVENT IS PERSONNEL ERROR BY NON-LICENSED UTILITY PERSONNEL. APPROXIMATELY ONE MONTH PRIOR TO THESE EVENTS, A NEW LABEL WAS INSTALLED ON THE SAFETY INJECTION BLOCK SWITCH WITH THE "BLOCKED" AND "UNBLOCKED" POSITIONS REVERSED. THIS RESULTED IN THE SWITCH BEING IN THE "UNBLOCKED" POSITION INSTEAD OF THE "BLOCKED" POSITION WHEN THE MANUAL BLOCK BUTTONS WERE RELEASED. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDE COUNSELING INVOLVED PERSONNEL, REPLACING THE INCORRECT LABEL, REVISING PROCEDURE 0-ADM-209, EQUIPMENT TAGGING AND LABELING, AND REVISING DRAWING 5610-M-430-171 SHEET 5.

[249] TURKEY POINT 4 DOCKET 50-251 LER 89-002 REV 01
UPDATE ON "A" TRAIN SAFEGUARDS ACTUATION DUE TO INADEQUATE ADMINISTRATIVE CONTROLS.
EVENT DATE: 041289 REPORT DATE: 072189 NSSS: WE TYPE: PWR

(NSIC 214869) ON APRIL 12, 1989, AT APPROXIMATELY 1140, WITH UNIT 4 IN COLD SHUTDOWN, A REACTOR CONTROL OPERATOR (RCO) WAS ATTEMPTING TO RELEASE A CLEARANCE ON THE UNIT 4 SAFEGUARDS RACKS POWER FUSES. WHEN THE RCO INSTALLED FUSES FU3 AND FU4, HE RETURNED THE POWER TO THE TRAIN SAFETY INJECTION (SI) LOGIC GENERATING AN "A" TRAIN SI SIGNAL. THE CAUSE OF THIS EVENT WAS INADEQUATE ADMINISTRATIVE CONTROLS. THE RCO DID NOT RECOGNIZE THE NEED TO USE PROCEDURE 4-ONOP-049, "RE-ENERGIZING SAFEGUARDS RACKS AFTER LOSS OF SINGLE POWER SUPPLY," TO INSTALL THESE FUSES BECAUSE THE STANDARD CLEARANCE FOR THIS ACTIVITY DID NOT SPECIFY ITS USE, NOR WAS THERE ANY WARNING SPECIFYING ITS USE IN THE SAFEGUARDS RACKS. USE OF PROCEDURE 4-ONOP-049 PREVENTS A SAFEGUARDS ACTUATION WHILE INSTALLING THESE FUSES. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDED ADDING TO THE STANDARD CLEARANCES IN THE PLANT CLEARANCE ORDER NETWORK THE INSTRUCTION TO USE PROCEDURE 3/4-ONOP-049 WHEN RELEASING CLEARANCES ON THESE FUSES, PLACING SIGNS IN THE SAFEGUARDS RACKS AND AT THE POWER SUPPLY BREAKERS FOR THE SAFEGUARDS LOGIC OF BOTH UNITS WITH THE CAUTION TO USE PROCEDURE 3/4-ONOP-049 WHEN REPLACING THESE FUSES.

[250] TURKEY POINT 4 DOCKET 50-251 LER 89-005
4A ACCUMULATOR LEVEL TECH SPEC NOT MET DUE TO WRONG REDUNDANT LEVEL TRANSMITTER IDENTIFIED AS MALFUNCTIONING.
EVENT DATE: 062889 REPORT DATE: 080989 NSSS: WE TYPE: PWR
VENDOR: ROSEMOUNT, INC.

(NSIC 214979) ON JULY 11, 1989, IT WAS DETERMINED THAT THE UNIT 4 SAFETY INJECTION ACCUMULATOR LEVEL TECHNICAL SPECIFICATION (TS) LIMITS WERE EXCEEDED DURING OPERATION. TS 3.4.1.A.3 REQUIRES EACH ACCUMULATOR TO CONTAIN 875 - 891 CUBIC FEET OF WATER. TS 3.4.1.B.1 STATES THAT ONE ACCUMULATOR MAY BE TAKEN OUT OF SERVICE FOR FOUR HOURS IF TS 3.4.1.A.3 CANNOT BE MET. DUE TO THE INCORRECT INITIAL DETERMINATION OF WHICH OF TWO REDUNDANT ACCUMULATOR LEVEL TRANSMITTERS WAS INDICATING CORRECTLY USING NORMAL DIAGNOSTIC TECHNIQUES, THE WRONG LEVEL TRANSMITTER ON 4A ACCUMULATOR WAS TAKEN OUT OF SERVICE FOR REPAIR. DURING THE TIME TAKEN TO DETERMINE THAT THE WRONG LEVEL TRANSMITTER WAS OUT OF SERVICE, THE 4A ACCUMULATOR ACTUALLY EXCEEDED THE 891 CUBIC FOOT TS LIMIT (6664 GALLONS) BY A MAXIMUM VALUE OF 20.4 GALLONS FOR APPROXIMATELY NINE HOURS AND THIRTY-EIGHT MINUTES. THIS CONDITION WAS ALARMED BUT CONSIDERED INVALID SINCE IT OCCURRED ON THE CHANNEL OUT OF SERVICE AND THE REDUNDANT CHANNEL INDICATED WITHIN NORMAL LIMITS. THIS EVENT IS WITHIN THE BOUNDS OF AN EVALUATION PREVIOUSLY PERFORMED BY WESTINGHOUSE WHICH CONCLUDED THAT THE HEALTH AND SAFETY OF THE PUBLIC WAS NOT AFFECTED. PROCEDURES NOW CONTAIN A MAXIMUM DEVIATION VALUE WHICH WILL INITIATE AN INVESTIGATION TO IDENTIFY ANY MALFUNCTIONS PRIOR TO EXCEEDING ANY ADMINISTRATIVE OR TS LIMITS.

[251] TURKEY POINT 4 DOCKET 50-251 LER 89-006
DEGRADED INTAKE COOLING WATER FLOW CONDITION TO COMPONENT COOLING WATER HEAT EXCHANGERS AND UNIT SHUTDOWN REQUIRED BY TECH SPECS DUE TO VALVE FAILURE.
EVENT DATE: 071389 REPORT DATE: 081489 NSSS: WE TYPE: PWR
VENDOR: PRATT, HENRY COMPANY

(NSIC 215043) ON 7/13/89, INTAKE COOLING WATER (ICW) FLOW TO THE UNIT 4 COMPONENT COOLING WATER (CCW) HEAT EXCHANGERS DEGRADED TO BELOW THE DESIGN BASIS FLOW RATE FOR FIFTEEN MINUTES WHILE BACKWASHING THE 4A ICW/CCW BASKET STRAINER. ICW FLOW WAS RESTORED WITHIN FIVE MINUTES OF RECOGNIZING THE DEGRADED FLOW CONDITION. ON 7/15/89 THE SAME CONDITION OCCURRED BUT ICW FLOW WAS MAINTAINED ABOVE THE DESIGN BASIS FLOW. ON 7/16/89, THE 4B ICW HEADER WAS TAKEN OUT OF SERVICE FOR INVESTIGATION. THE CAUSE OF THESE CONDITIONS WAS INITIALLY BELIEVED TO BE CLOGGING OF THE 4B ICW/CCW BASKET STRAINER AND/OR A MALFUNCTION OF ITS DIFFERENTIAL RPESSURE GAUGE. SUBSEQUENT EVALUATION INDICATED FLOW OBSTRUCTION IN OR AROUND 4B ICW HEADER ISOLATION VALVE 4-50-308. DUE TO THE FREQUENCY OF CLEANING ICW/CCW BASKET STRAINERS, A MANAGEMENT DECISION WAS MADE NOT TO CONTINUE

POWER OPERATIONS WITH ONLY ONE HEADER OF ICW OPERABLE. UNIT 4 WAS PLACED IN MODE 3 ON 7/17/89. VALVE 4-50-308 WAS REPLACED AND THE 4B ICW HEADER WAS RETURNED TO SERVICE ON 7/20/89. THE INITIAL DEGRADED ICW FLOW CONDITION DID NOT AFFECT ANY OTHER PLANT EQUIPMENT.

[252] TURKEY POINT 4 DOCKET 50-251 LER 89-007
MISSED VISUAL EXAMINATION ON REPAIRED CONTAINMENT PENETRATIONS DUE TO PERSONNEL ERROR.
EVENT DATE: 071389 REPORT DATE: 081489 NSSS: WE TYPE: PWR

(NSIC 215057) ON JULY 13, 1989, WITH UNIT 4 AT 25% POWER (MODE 1), IT WAS DETERMINED THAT A REQUIRED VISUAL EXAMINATION FOR LEAKAGE HAD BEEN MISSED. THREE CONTAINMENT ISOLATION VALVES HAD BEEN REPLACED, REQUIRING A STEAM GENERATOR HYDROSTATIC TEST. TO ALLOW DELAY TO THE NEXT SCHEDULED STEAM GENERATOR HYDROSTATIC TEST, ASME CODE CASE N-416 WAS INVOKED, REQUIRING A SURFACE EXAMININATION (NDE), AND A VISUAL EXAMINATION FOR LEAKAGE AFTER SYSTEM PRESSURIZATION. SINCE THE TESTING WAS NOT COMPLETED IMMEDIATELY UPON REACHING CONDITIONS ALLOWING IT, THE 3 ASSOCIATED CONTAINMENT PENETRATIONS WERE TECHNICALLY INOPERABLE. THE BECAUSE WAS COGNITIVE PERSONNEL ERROR IN NOT PERFORMING THE TEST IMMEDIATELY UPON RETURN TO SERVICE OF THE REPAIRED PIPING. ALTHOUGH THE 3 PENETRATIONS WERE TECHNICALLY INOPERABLE, NO DEGRADATION OF A SAFETY BARRIER EXISTED, AS DEMONSTRATED BY THE SATISFACTORY COMPLETION OF THE VISUAL EXAMINATION WITHIN ONE HOUR OF DISCOVERY OF THE CONDITION. THE PERSONNEL INVOLVED HAVE BEEN COUNSELED. PROCEDURES AND PRACTICES GOVERNING WORK AND FOLLOW-ON TESTING BY CONSTRUCTION ARE BEING REVIEWED TO DETERMINE IF CHANGES CAN BE MADE TO AID IN PREVENTING RECURRENCE.

[253] VERMONT YANKEE DOCKET 50-271 LER 89-009
LACK OF REDUNDANCY IN RESIDUAL HEAT REMOVAL SERVICE WATER SYSTEMS.
EVENT DATE: 062889 REPORT DATE: 072889 NSSS: GE TYPE: BWR

(NSIC 214916) ON 6/28/89, WITH THE REACTOR OPERATING AT 99.9% STEADY-STATE POWER, DURING A REVIEW OF LOADS POWERED FROM BUS 9, IT WAS IDENTIFIED THAT THE POWER SUPPLY FEEDING THE INSTRUMENTATION CIRCUITRY FOR THE RESIDUAL HEAT REMOVAL SYSTEM (RHR) (EIIS=BO) HEAT EXCHANGER SERVICE WATER OUTLET VALVE V10-89B WAS SUPPLIED FROM MOTOR CONTROL CENTER (MCC) 9D (EIIS=ED). UPON A LOSS OF OFF-SITE POWER, MCC 9D IS POWERED BY THE SII DIVISION OF THE EMERGENCY POWER SYSTEM (EIIS=EK), WHICH ALSO SUPPLIES CONTROL AND 480 VOLT POWER TO THE "A" TRAIN RHR VALVE V10=89A. A FAILURE OF THE "A" EMERGENCY DIESEL GENERATOR (SII POWER DIVISION) (EIIS=EK) TO START UPON LOSS OF OFF-SITE POWER COULD HAVE RENDERED BOTH RHR SERVICE WATER (RHR/SW) (EIIS=BI) TRAINS INOPERABLE. AT 1749 HOURS, THE "B" RHR/SW SUBSYSTEM WAS DECLARED INOPERABLE TO ALLOW THE INSTALLATION OF A TEMPORARY MODIFICATION (89-038) TO TRANSFER THE V10-89B INSTRUMENTATION POWER TO A REDUNDANT POWER SUPPLY. AT 1823 HOURS, THE "B" RHR/SW SUBSYSTEM WAS RETURNED TO SERVICE. ALL OTHER LOADS POWERED FROM THE POWER PANEL 89, MCC 8E, AND MCC 9D WERE IMMEDIATELY REVIEWED TO ENSURE NO SIMILAR CONDITIONS EXISTED. THE ROOT CAUSE ANALYSIS AND THE CONTRIBUTING FACTORS ARE LOCATED IN THE BODY OF THE LER.

[254] VERMONT YANKEE DOCKET 50-271 LER 89-014
REACTOR CORE ISOLATION COOLING SYSTEM INOPERABLE DUE TO MOTOR BURN OUT ON RCIC-21 VALVE.
EVENT DATE: 071889 REPORT DATE: 072589 NSSS: GE TYPE: BWR
VENDOR: LIMITORQUE CORP.

(NSIC 214876) ON 6/7/89 WITH THE PLANT OPERATING AT 92% POWER, A VALVE OPERABILITY TEST OF THE REACTOR CORE ISOLATION (RCIC) VALVE V13-21 WAS INITIATED AT 0200 HRS. AFTER A SUCCESSFUL OPEN STROKE THE VALVE OPERATOR FAILED WHEN ATTEMPTS TO STROKE CLOSE WERE INITIATED. RCIC WAS THEN DECLARED INOPERABLE AT

0220 HRS. INVESTIGATIONS REVEALED A FAILED DC MOTOR ON THE RCIC VALVE OPERATOR (EIIS=BN). THE ARMATURE WINDINGS OF THE MOTOR HAD OVERHEATED AND SHORT CIRCUITED. THE FAILED MOTOR WAS REPLACED AND THE VALVE RE-TESTED. THE RCIC SYSTEM WAS DECLARED OPERABLE AT 1545 HRS ON 6/8/89. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO A PRIOR OVERCURRENT CONDITION AND REPEATED CYCLIC OPERATIONS DURING TESTING WHICH RESULTED IN A PREMATURE END OF LIFE FAILURE. TO PREVENT RECURRENCE OF THE EVENT PLANT PROCEDURES WILL BE REVISED TO PROVIDE INFORMATION RELATIVE TO ACCEPTABLE NUMBER OF CONSERVATIVE OPERATIONS AND REQUIRED COOLDOWN TIMES BETWEEN OPERATIONS. ALL VARIATIONS FROM STANDARD PARAMETERS (I.E., OVERCURRENT CONDITION) WILL CONTINUE TO RECEIVE A THOROUGH ENGINEERING EVALUATION TO ACCESS FOR OPERABILITY.

[255] VOGTLE 1 DOCKET 50-424 LER 89-015 REV 01
UPDATE ON PERSONNEL ERROR LEADS TO INADEQUATELY PERFORMED SURVEILLANCE.
EVENT DATE: 070289 REPORT DATE: 082589 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 215147) ON JULY 2, 1989, AT 1400 CST, DURING THE REGULAR SHIFT TURNOVER BRIEFING, THE EVENING-SHIFT REACTOR OPERATOR (RO) DISCOVERED A FAILURE TO ADEQUATELY PERFORM THE REQUIREMENTS OF TECHNICAL SPECIFICATION (TS) 4.2.1.1.B. TECHNICAL SPECIFICATION 4.2.1.1.B REQUIRES MONITORING AND LOGGING OF THE AXIAL FLUX DIFFERENCE (AFD) FOR EACH OPERABLE EXCORE CHANNEL AT LEAST ONCE PER 30 MINUTES FOLLOWING THE FIRST 24 HOURS AFTER THE AFD ALARM BECOMES INOPERABLE. INSTEAD, THIS SURVEILLANCE WAS PERFORMED AT ONE-HOUR INTERVALS FOR THE FIRST 30 HOURS. THIS CONDITION EXISTED FOR APPROXIMATELY 6 HOURS (FROM 0808 CST TO 1400 CST). THE ROOT CAUSE OF THIS EVENT WAS COGNITIVE PERSONNEL ERROR. AFTER REVIEWING THE TS REQUIREMENTS AND INITIATING THE APPROPRIATE DATA SHEETS ON HIS SHIFT THE PREVIOUS DAY, THE DAY-SHIFT SHIFT SUPERVISOR FAILED TO CHANGE THE FREQUENCY OF MONITORING AND LOGGING FROM ONE HOUR TO 30 MINUTES AS REQUIRED BY TS 4.2.1.1.B AND PLANT PROCEDURE. THE CORRECTIVE ACTIONS INCLUDED INCREASING THE FREQUENCY FOR MONITORING AND LOGGING TO BE IN COMPLIANCE WITH TS 4.2.1.1.B REQUIREMENTS, COUNSELING OF INVOLVED PERSONNEL, AND PLACING A COPY OF THE LER IN THE OPERATIONS REQUIRED READING BOOK FOR ALL ON-SHIFT LICENSED OPERATORS.

[256] VOGTLE 1 DOCKET 50-424 LER 89-016
MANUAL REACTOR TRIP DUE TO FAILURE OF MAIN FEEDWATER ISOLATION VALVE.
EVENT DATE: 070889 REPORT DATE: 080289 NSSS: WE TYPE: PWR
VENDOR: ANCHOR/DARLING VALVE CO.

(NSIC 214965) ON JULY 8, 1989, AT 0327 CDT, A MANUALLY INITIATED REACTOR TRIP OCCURRED ON UNIT 1 WITH THE REACTOR AT 100% OF RATED THERMAL POWER. THE MANUAL TRIP WAS INITIATED BECAUSE THE LOOP 4 MAIN FEEDWATER ISOLATION VALVE (MFIV) FAILED CLOSED RESULTING IN A DECREASE IN THE NO. 4 STEAM GENERATOR LEVEL. THE PLANT WAS STABILIZED IN MODE 3 FOLLOWING THE REACTOR TRIP. TROUBLESHOOTING FOLLOWING THE MANUAL TRIP FAILED TO IDENTIFY THE EXACT CAUSE OF THE SPURIOUS CLOSURE; HOWEVER, TWO POTENTIAL FAILURE MECHANISMS ARE BEING EVALUATED; SOLENOID FAILURE AND AUXILIARY RELAY FAILURE. A DOCUMENT REVIEW OF PREVIOUS TRIPS IDENTIFIED A SIMILAR EVENT ON APRIL 24, 1988, WHICH INVOLVED A SPURIOUS CLOSURE OF THE SAME MFIV AND A MANUALLY INITIATED REACTOR TRIP. FOLLOWING THAT EVENT, VARIOUS COMPONENTS WERE REPLACED; HOWEVER, BENCH TESTING DID NOT PROVIDE ANY CONCLUSIVE EVIDENCE OF THE CAUSE OF VALVE FAILURE. CORRECTIVE ACTIONS FOR THIS EVENT INCLUDED TROUBLESHOOTING OF THE VALVE CONTROL CIRCUITRY, REPAIR OF THE ASSOCIATED HANDSWITCH FOR A PROBLEM NOTED DURING TROUBLESHOOTING BUT UNRELATED TO THE FAILURE, AND CONTINUED CONTROL LOOP MONITORING WITH A MULTI-CHANNEL RECORDER.

[257] VOGTLE 1 DOCKET 50-424 LER 38-046
INACCURATE DATABASE INFORMATION LEADS TO TECH SPEC VIOLATION.
EVENT DATE: 071389 REPORT DATE: 080889 NSSS: WE TYPE: PWR

(NSIC 215005) ON 7-13-89, A REVIEW OF SURVEILLANCE REQUIREMENTS FOR THE FUEL HANDLING BUILDING (FHB) RADIATION MONITORS DETERMINED THAT THE ANALOG CHANNEL OPERATIONAL TEST (ACOT) HAD NOT BEEN PERFORMED FOR RADIATION MONITORS ARE-2533 A & B SINCE 10-22-88. TECHNICAL SPECIFICATIONS (TS) TABLE 4.3-2, ITEM 11.C, REQUIRES A MONTHLY ACOT OF THESE MONITORS WHENEVER IRRADIATED FUEL IS IN EITHER STORAGE POOL. THE SHIFT SUPERVISOR WAS INFORMED OF THIS CONDITION AND AN "INFORMATION ONLY" LIMITING CONDITION FOR OPERATION (LCO) WAS ENTERED FOR MONITORS ARE-2533 A & B. MONITORS ARE-2532 A & B WERE OPERABLE AT THIS TIME, WHICH SATISFIED TS REQUIREMENTS. SUBSEQUENT INVESTIGATION REVEALED PERIODS SINCE 10-22-88 WHEN MONITORS ARE-2533 A & B WERE CONSIDERED THE OPERABLE CHANNEL FOR FHB POST ACCIDENT VENTILATION ACTUATION. DURING THOSE PERIODS BEYOND 11-28-88, THE TS OPERABILITY REQUIREMENTS WERE NOT SATISFIED SINCE THE SURVEILLANCE REQUIREMENTS WERE NOT MET. THE CAUSE OF THE EVENT WAS PERSONNEL ERROR. THE DUE DATE FOR THE SURVEILLANCE WAS IMPROPERLY ENTERED AS NOVEMBER, 1990, INSTEAD OF NOVEMBER, 1988, IN THE COMPUTERIZED SURVEILLANCE TRACKING PROGRAM. A COMPUTERIZED AUDIT PROGRAM WHICH REVIEWS SURVEILLANCES AND IDENTIFIES FOR ACTION ANY WHICH ARE NOT SCHEDULED TO BE PERFORMED WITHIN THEIR PERIODIC INTERVALS WILL BE USED IN AN EFFORT TO PREVENT SIMILAR OCCURRENCES.

[258] VOGTLE 1 DOCKET 50-424 LER 89-017
RADIATION MONITORS HIGH ALARM RESULTS IN FUEL HANDLING BUILDING ISOLATION.
EVENT DATE: 072889 REPORT DATE: 082489 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: VOGTLE 2 (PWR)

(NSIC 215148) ON 7-28-89, AT 1720 CDT, RADIATION MONITOR ARE-2532B WENT TO ITS HIGH ALARM SETPOINT, CAUSING A FUEL HANDLING BUILDING (FHB) ISOLATION. THE FHB POST ACCIDENT FILTRATION UNITS STARTED AND THE APPROPRIATE VENTS AND DAMPERS ACTUATED. CONTROL ROOM OPERATORS ANNOUNCED THE FHB ISOLATION OVER THE PUBLIC ADDRESS SYSTEM AND BEGAN TO SURVEY OTHER AREA MONITORS TO CHECK FOR THE POSSIBLE SPREAD OF RADIOACTIVE CONTAMINANTS, BUT FOUND NO OTHER MONITOR WITH AN ABNORMAL READING. AT APPROXIMATELY 1730 CDT, THE ACTIVITY LEVEL ON ARE-2532B DROPPED BELOW THE HIGH ALARM SETPOINT. AT 1923 CDT, THE FHB ISOLATION SIGNAL WAS RESET, AND AT 1924 CDT, THE FHB POST ACCIDENT FILTRATION UNITS WERE STOPPED. AN INITIAL REVIEW INDICATED THAT RCS COOLANT HAD LEAKED INSIDE THE POST ACCIDENT SAMPLING SYSTEM PANEL, WHICH WAS IN USE AT THE TIME. IT WAS BELIEVED THAT A PORTION OF THE RCS HAD EVAPORATED AND INITIATED THE ARE- 2532B HIGH ALARM CONDITION. HOWEVER, SMEARS TAKEN INSIDE THE PANEL FOUND NO LOOSE CONTAMINATION WHICH WOULD INDICATE THAT RCS LEAKAGE OCCURRED. ADDITIONAL SMEARS FOUND SIGNS OF CONTAMINATION ON A FLOOR DRAIN BUT POTENTIAL SOURCES OF THIS CONTAMINATION HAVE NOT BEEN PROVEN.

[259] VOGTLE 2 DOCKET 50-425 LER 89-021 REV 01
UPDATE ON FAILURE OF INTERCEPT VALVES TO OPEN RESULTS IN REACTOR TRIP ON SG LO-LO LEVEL.
EVENT DATE: 052289 REPORT DATE: 080789 NSSS: WE TYPE: PWR

(NSIC 215006) ON 5-22-89, PREPARATIONS WERE UNDERWAY TO START UP THE MAIN TURBINE. AT APPROXIMATELY 1547 CDT, STEPS WERE IN TO BRING THE TURBINE SPEED UP TO 180 AT 1605 CDT, INDICATIONS OF A STEAM/FEED MISMATCH PROBLEM WERE SEEN ON STEAM GENERATOR (SG) #2 AND THE REACTOR OPERATOR (RO) OBSERVED A DECREASE IN TAVE GREATER THAN EXPECTED. BECAUSE SG LEVELS AND PRESSURE WERE DECREASING, THE BALANCE OF PLANT OPERATOR TRIPPED THE TURBINE AT 1606 CDT. FEED TO THE STEAM GENERATORS WAS INCREASED AND THE STEAM DUMPS WERE MANUALLY CLOSED. HOWEVER, AT 1607 CDT, AN AUTOMATIC REACTOR TRIP OCCURRED ON LOW-LOW LEVEL IN SG #2. CONTROL ROOM OPERATORS TOOK APPROPRIATE ACTIONS TO STABILIZE THE UNIT IN MODE 3 (HOT

STANDBY). THE DIRECT CAUSE OF THE EVENT WAS THE FAILURE OF THE INTERCEPT VALVES TO OPEN AND A CONTRIBUTING CAUSE WAS OPERATOR OVERSIGHT IN NOT RECOGNIZING THE INTERCEPT VALVES' FAILURE TO OPEN AS TURBINE SPEED WAS INCREASED. WHEN THE MAIN TURBINE REACHED 1800 RPM, THE CONTROL VALVES OPENED FURTHER TO MAINTAIN A CONSTANT TURBINE SPEED CREATING A LARGE STEAM LOAD. THIS CAUSED PRESSURIZATION OF THE HIGH PRESSURE TURBINE AND MOISTURE SEPARATOR REHEATERS (MSR), EVENTUALLY LIFTING THE "B" MSR RELIEF VALVE. THE OPERATOR TRIPPED THE MAIN TURBINE, SG WATER LEVELS LOWERED AND A REACTOR TRIP OCCURRED. CORRECTIVE ACTIONS INCLUDED ADJUSTMENT OF THE CLOSING BIAS VOLTAGE ON THE INTERCEPT VALVE CIRCUIT CARDS.

[260] VOGTLE 2 DOCKET 50-425 LER 89-023
ESF ACTUATION RESULTS WHEN TRANSFERRING OFFSITE POWER SOURCES.
EVENT DATE: 072089 REPORT DATE: 081589 NSSS: WE TYPE: PWR
VENDOR: FISHER CONTROLS CO.

(NSIC 215034) ON JULY 20, 1989, NON-1E 4160 VOLT BUS 2NA05 WAS BEING TRANSFERRED FROM THE RESERVE AUXILIARY TRANSFORMER (RAT) TO THE NORMAL SUPPLY FROM THE UNIT AUXILIARY TRANSFORMER (UAT), WHEN RAT 2A TRIPPED ON A DIFFERENTIAL RELAY LOCKOUT. POWER WAS LOST TO THE TRAIN "A" 1E BUS AND VARIOUS NON-1E BUSES. THE LOSS OF POWER RESULTED IN A DIESEL GENERATOR START AND ACTUATION OF THE AUXILIARY FEEDWATER (AFW) SYSTEM. AN INVESTIGATION FOUND THAT A CURRENT TRANSFORMER IN THE 2NA05 4160V BUS HAD BEEN IMPROPERLY TERMINATED. THE IMPROPER TERMINATION HAD GONE UNDETECTED BECAUSE THE NORMALLY SMALL LOAD CARRIED BY THE BUS WAS INSUFFICIENT TO ACTUATE THE DIFFERENTIAL LOCKOUT RELAY UNDER NORMAL CONDITIONS. HOWEVER, DURING THE TRANSFER, THE MOMENTARY SURGE OF CURRENT AS BOTH THE NORMAL AND ALTERNATE SUPPLY BREAKERS WERE CLOSED WAS SUFFICIENT TO ACTUATE THE RELAY. THE CAUSE OF THIS EVENT IS AN ERROR IN THE ORIGINAL DESIGN. CORRECTIVE ACTION INCLUDES CORRECTING THE IMPROPER TERMINATION AND REVIEWING OTHER BREAKERS IN THE UNITS 1 AND 2 RATS AND UATS FOR SIMILAR WIRING CONDITIONS.

[261] VOGTLE 2 DOCKET 50-425 LER 89-024
FAILURE OF PRESSURE CHANNEL CIRCUIT CARD CAUSES REACTOR TRIP.
EVENT DATE: 072689 REPORT DATE: 082189 NSSS: WE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 215149) ON 7-26-89, AT 0136 CDT, WHILE PERSONNEL WERE PERFORMING CORRECTIVE MAINTENANCE ON POWER RANGE NUCLEAR INSTRUMENTATION CHANNEL 2N43, A 2 OUT OF 4 OVERTEMPERATURE DELTA-T (OTDT) TRIP SIGNAL WAS RECEIVED AND CAUSED AN AUTOMATIC REACTOR TRIP FOR VEGP UNIT 2. THE CORRECTIVE MAINTENANCE ON 2N43 REQUIRED THE CHANNEL 111 OTDT REACTOR TRIP BISTABLE TO BE TRIPPED AS A PART OF THE REMOVAL FROM SERVICE PROCESS. A LOSS OF INPUT FROM PRESSURIZER PRESSURE CHANNEL 2P458 THEN OCCURRED AND CAUSED THE CHANNEL IV OTDT REACTOR TRIP BISTABLE TO TRIP. THIS COMPLETED THE 2 OUT OF 4 LOGIC REQUIRED FOR REACTOR TRIP ON OTDT. BY 0156 CDT, THE PLANT HAD BEEN STABILIZED IN MODE 3. THE FAILURE OF CHANNEL 2P458 WAS CAUSED BY THE FAILURE OF AN OPERATIONAL AMPLIFIER IN THE NON-ISOLATED SECTION OF AN NLP2 PROCESS CARD. THIS CHANNEL HAD SPIKED LOW ON TWO SEPARATE OCCASIONS SEVERAL DAYS EARLIER BUT TROUBLESHOOTING FAILED TO IDENTIFY THE EXACT CAUSE OF THE PROBLEM UNTIL AFTER THE REACTOR TRIP. AN ADDITIONAL SPIKING PROBLEM HAD ALSO BEEN EXPERIENCED ON CHANNEL 2N43 AND WAS STILL BEING INVESTIGATED AT THE TIME OF THE EVENT. CORRECTIVE ACTION CONSISTED OF REPLACING THE DEFECTIVE NLP2 CARD FOR CHANNEL 2P458 AND REPLACING A SUSPECT CARD FOR CHANNEL 2N43.

[262] WATERFORD 3 DOCKET 50-382 LER 88-026 REV 01
UPDATE ON TUBING AND SUPPORTS NOT SEISMICALLY QUALIFIED DUE TO PERSONNEL ERROR.
EVENT DATE: 102888 REPORT DATE: 082889 NSSS: CE TYPE: PWR

(NSIC 215099) AT 1810 HOURS ON OCTOBER 28, 1988, WATERFORD STEAM ELECTRIC STATION UNIT 3 WAS IN COLD SHUTDOWN WHEN MANAGEMENT WAS INFORMED THAT THE WRONG PIPING

CLASS TUBING WAS INSTALLED FOR THE BORIC ACID MAKEUP TANKS (BAMTS) LEVEL INDICATORS. THE TUBING AND SUPPORTS WERE TO HAVE BEEN INSTALLED AS SEISMICALLY QUALIFIED; HOWEVER, WHEN SOME SUPPORTS WERE DISCOVERED NOT INSPECTED, A NON-SEISMIC QUALIFICATION WAS JUSTIFIED. WHEN THE BAMT TECHNICAL SPECIFICATION (TS) REQUIREMENTS WERE CHANGED, THE SUPPORTS WERE INSPECTED TO SEISMICALLY QUALIFY THE INSTALLATION. A RECENT REVIEW DISCOVERED THAT THE INSTALLED TUBING WELDS COULD NOT BE DETERMINED TO BE SEISMICALLY QUALIFIED. THEREFORE, THE PLANT OPERATED IN A CONDITION OUTSIDE THE DESIGN BASIS SINCE JANUARY 8, 1987. THE CAUSE OF THIS EVENT WAS PERSONNEL ERROR IN THAT INADEQUATE REVIEWS OF THE DESIGN AT SEVERAL POINTS IN THIS EVENT ALLOWED THE CONDITION TO EXIST SINCE INITIAL CONSTRUCTION. THE SAFETY EVALUATION PROGRAM IS BEING REVIEWED FOR ENHANCEMENTS. EMPHASIS HAS BEEN INCREASED ON NONCONFORMANCE ITEMS AND THEY ARE TRACKED AT PLAN OF THE DAY MEETINGS. THE SUPPORTS WERE INSPECTED AND THE TUBING REPLACED. DESIGN ENGINEERING INSPECTED OTHER INSTALLATION DRAWINGS TO ENSURE THAT CORRECT CODE BREAKS WERE ESTABLISHED. SINCE THE TUBING WAS ASME CLASS I MATERIAL IT IS UNLIKELY TO HAVE BEEN AFFECTED BY A SEISMIC EVENT.

[263] WATERFORD 3 DOCKET 50-382 LER 89-012
RADIATION MONITOR INOPERABLE DURING DISCHARGE DUE TO INADEQUATE ADMINISTRATIVE CONTROLS.
EVENT DATE: 070689 REPORT DATE: 080389 NSSS: CE TYPE: PWR

(NSIC 214961) AT 0730 HOURS ON JULY 6, 1989, WATERFORD STEAM ELECTRIC STATION UNIT 3 WAS OPERATING AT 100% POWER WHEN A HEALTH PHYSICS (HP) TECHNICIAN NOTED THAT THE LIQUID WASTE MANAGEMENT (LWM) SYSTEM EFFLUENT RADIATION MONITOR SAMPLE PUMP WAS NOT OPERATING DURING A RADIOACTIVE RELEASE OF WASTE CONDENSATE TANK 'A'. THE RADIATION MONITOR'S ABILITY TO AUTOMATICALLY TERMINATE THE RELEASE ON A LOW SAMPLE FLOW WAS NOT AVAILABLE, RENDERING THE MONITOR TECHNICALLY INOPERABLE. TECHNICAL SPECIFICATION (TS) 3.3.3.10 ACTION REQUIREMENT 28 SPECIFIES THAT WITH AN INOPERABLE MONITOR, TWO INDEPENDENT ACTIVITY SAMPLES MUST BE ANALYZED AND TWO INDEPENDENT DISCHARGE VALVE LINEUP CHECKS MUST BE COMPLETED PRIOR TO INITIATING A RELEASE. BECAUSE ONLY THE PRE-RELEASE ACTIVITY SAMPLE WAS ANALYZED PRIOR TO INITIATING THE DISCHARGE, THE PLANT OPERATED IN A CONDITION PROHIBITED BY TSS. THE ROOT CAUSE OF THIS EVENT IS THE INADEQUATE REVIEW OF A COMPUTER FIRMWARE/SOFTWARE REVISION THAT WAS INSTALLED ON JUNE 29, 1989. ALTHOUGH THE LOW SAMPLE FLOW TRIP WAS NOT AVAILABLE, THE HIGH ACTIVITY TRIP REMAINED OPERABLE. THE RELEASE WAS TERMINATED WHEN THE MONITOR'S OPERABILITY BECAME QUESTIONABLE. SAMPLES TAKEN BEFORE AND AFTER THE RELEASE INDICATED THAT NO UNEXPECTED ACTIVITY WAS DISCHARGED DURING THE RELEASE; THEREFORE, THIS EVENT DID NOT POSE A THREAT TO THE HEALTH AND SAFETY OF THE GENERAL PUBLIC OR TO PLANT PERSONNEL.

[264] WATERFORD 3 DOCKET 50-382 LER 89-013
MANUAL REACTOR TRIP DUE TO LOSS OF FEED FLOW TO STEAM GENERATOR NUMBER 1.
EVENT DATE: 071589 REPORT DATE: 081489 NSSS: CE TYPE: PWR
VENDOR: WESTINGHOUSE ELECTRIC CORP.

(NSIC 215030) ON JULY 19, 1989, WATERFORD STEAM ELECTRIC STATION UNIT 3 WAS OPERATING AT 100% POWER WHEN A RAPIDLY DECREASING LEVEL WAS OBSERVED IN STEAM GENERATOR NUMBER 1. THE MAIN FEEDWATER REGULATING VALVE FOR THIS STEAM GENERATOR WAS FOUND CLOSED. THE CONTROLLER FOR THE VALVE WAS SWITCHED TO MANUAL AND THE VALVE WAS OPENED IN AN ATTEMPT TO RESTORE THE STEAM GENERATOR LEVEL. WHEN IT BECAME APPARENT THAT THE LEVEL COULD NOT BE RESTORED BEFORE RECEIVING A LOW STEAM GENERATOR LEVEL REACTOR TRIP, A MANUAL REACTOR TRIP WAS INITIATED. THE ROOT CAUSE FOR THE EVENT WAS ATTRIBUTED TO A CIRCUIT FAILURE IN THE POSITION CONTROL CIRCUITRY FOR THE MAIN AND STARTUP FEEDWATER REGULATING VALVES FOR STEAM GENERATOR NUMBER 1. THE CIRCUIT CARD CONTAINING THE FAILED COMPONENT WAS REPLACED AND THE SYSTEM RESTORED TO NORMAL. DURING THE COURSE OF THIS EVENT NO THREAT TO THE HEALTH OR SAFETY OF THE PUBLIC OR PLANT PERSONNEL EXISTED.

[265] WATERFORD 3 DOCKET 50-382 LER 89-014
ACC-116A AND 116B NOT INCLUDED IN IN-SERVICE TEST PROGRAM DUE TO MISINTERPRETATION OF REQUIREMENTS.
EVENT DATE: 071889 REPORT DATE: 081689 NSSS: CE TYPE: PWR

(NSIC 215031) AT 1321 HOURS ON JULY 18, 1989, WATERFORD STEAM ELECTRIC STATION UNIT 3 WAS OPERATING AT 100% POWER WHEN LICENSING PERSONNEL RAISED THE QUESTION OF INSERVICE TEST (IST) REQUIREMENTS FOR AUXILIARY COMPONENT COOLING WATER (ACCW) VALVES ACC-116A AND ACC-116B. ACC-116A AND B ARE ASME CODE CLASS 3 VALVES CONNECTING THE WET COOLING TOWER (WCT) BASINS TO THE EMERGENCY FEEDWATER (EFW) PUMP SUCTION AND ARE REQUIRED TO MITIGATE CERTAIN DESIGN BASIS ACCIDENT (DBA) CONDITIONS. THESE VALVES WERE NOT INCLUDED IN THE PUMP AND VALVE IST PROGRAM CONTRARY TO ASME SECTION XI AND TECHNICAL SPECIFICATION (TS) SURVEILLANCE 4.0.5; THEREFORE, THE PLANT OPERATED IN A CONDITION PROHIBITED BY TSS. THE ROOT CAUSE OF THIS EVENT IS INADEQUATE REVIEW OF ASME SECTION XI IST REQUIREMENTS RESULTING FROM A MISINTERPRETATION OF THE ASME SECTION XI DEFINITION OF "ACTIVE" VALVES. BECAUSE ACC-116A AND 116B WERE JUDGED TO BE "PASSIVE" VALVES IN RELATION TO THE ASME SECTION XI DEFINITION, THEY WERE NEVER INCLUDED IN THE IST PROGRAM. AN ENGINEERING EVALUATION CONCLUDED THAT THE VALVES ARE OPERABLE AND THAT THE ABILITY TO MITIGATE THE CONSEQUENCES OF EITHER DBA CONSIDERED IS NOT ADVERSELY AFFECTED; THEREFORE, THIS EVENT DID NOT THREATEN THE HEALTH AND SAFETY OF THE GENERAL PUBLIC OR PLANT PERSONNEL.

[266] WOLF CREEK 1 DOCKET 50-482 LER 89-010 REV 01
UPDATE ON INADEQUATE PROGRAMMATIC CONTROLS LEADS TO PERSONNEL ERROR RESULTING IN TECH SPEC VIOLATION.
EVENT DATE: 061589 REPORT DATE: 072689 NSSS: WE TYPE: PWR

(NSIC 214934) ON 6/16/89, AT APPROX. 1000 CDT, IT WAS DISCOVERED THAT TECH SPEC SURVEILLANCE REQUIREMENT 4.11.2.6 HAD NOT BEEN MET WHEN THE QUANTITY OF RADIOACTIVE MATERIAL WAS NOT DETERMINED WITHIN 7 DAYS OF AN ADDITION TO THE NUMBER 6 WASTE GAS DECAY TANK (GDT) THAT OCCURRED ON 6/6/89. CHEMISTRY PERSONNEL WERE NOTIFIED OF THE DISCOVERY AND THE QUANTITY OF RADIOACTIVE MATERIAL WAS DETERMINED TO BE BELOW TECH SPEC 3.11.2.6 LIMITS AT APPROX. 1015 CDT. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO INADEQUATE PROGRAMMATIC CONTROLS WHICH ALLOWED A PERSONNEL ERROR BY NON-LICENSED OPERATIONS PERSONNEL TO RESULT IN THIS FAILURE TO MEET THE TECH SPEC SURVEILLANCE REQUIREMENT. TO PREVENT RECURRENCE OF THIS EVENT, ALL PROCEDURES INVOLVING ADDITIONS TO THE GDTS WILL BE REVISED TO REQUIRE OPERATIONS PERSONNEL TO NOTIFY CHEMISTRY WHEN ANY ADDITIONS ARE MADE TO THE GDTS. A RADWASTE TURNOVER CHECKLIST IS BEING DEVELOPED WHICH WILL REMIND THE OPERATORS TO NOTIFY CHEMISTRY WHEN ANY ADDITIONS ARE MADE TO THE GDTS. AN ADDITION HAS BEEN MADE TO PROCEDURE ADM 04-020, "CHEMISTRY SURVEILLANCE PROGRAM", TO PROVIDE CHEMISTRY PERSONNEL A MECHANISM FOR DOCUMENTING THE GDT NUMBER, THE DATE AND THE TIME OF ANY ADDITIONS TO THE GDTS WHEN INFORMATION IS RECEIVED FROM OPERATIONS PERSONNEL.

[267] WOLF CREEK 1 DOCKET 50-482 LER 89-011
TECH SPEC 3.0.3 ENTRY PER ADMINISTRATIVE GUIDELINES CAUSED BY INOPERABILITY OF CLASS 1E AIR CONDITIONING UNITS.
EVENT DATE: 062389 REPORT DATE: 072489 NSSS: WE TYPE: PWR

(NSIC 214935) ON 6/23/89, A CONDITION OCCURRED IN WHICH BOTH CLASS 1E AIR CONDITIONING UNITS, SGK05 A & B, WERE SIMULTANEOUSLY INOPERABLE. ALTHOUGH THIS SITUATION IS NOT EXPLICITLY ADDRESSED BY THE TECH SPECS, ENTRY INTO TECH SPEC 3.0.3 WAS DECLARED IN ACCORDANCE WITH PREVIOUSLY ESTABLISHED ADMINISTRATIVE GUIDELINES. ON 6/21/89, AT 1900 CDT, SGK05B WAS REMOVED FROM SERVICE FOR AN EDDY CURRENT INSPECTION AND POTENTIAL REPAIR. ON 6/23/89, AT 1135 CDT, SGK05A TRIPPED AND WOLF CREEK GENERATING STATION ENTERED TECH SPEC 3.0.3. AT 1155 CDT, SGK05A WAS RESTORED TO OPERABLE STATUS FOLLOWING RECRIMPING A WIRE ASSOCIATED WITH THE

UNIT'S PUMP-DOWN SWITCH ON THE LOCAL CONTROL PANEL AND TECH SPEC 3.0.3 WAS EXITED. SKG05A TRIPPED BECAUSE A LOOSE WIRE HAD DISCONNECTED FROM THE LUG ON THE PUMP-DOWN SWITCH. DURING THE TIME BOTH UNITS WERE OUT OF SERVICE, THE MAXIMUM TEMPERATURE REACHED IN THE AFFECTED ELECTRICAL EQUIPMENT ROOMS WAS LESS THAN 90F POSING NO THREAT TO CONTINUED ELECTRICAL EQUIPMENT OPERABILITY. SUBSEQUENTLY, ON 6/25/89, AT 1500 CDT, SGK05B WAS RESTORED TO OPERABLE STATUS FOLLOWING EDDY CURRENT INSPECTION AND REPAIR.

[268] WOLF CREEK 1 DOCKET 50-482 LER 89-012
CONTROL ROOM VENTILATION ISOLATION SIGNAL CAUSED BY LOSS OF POWER TO CONTROL ROOM AIR INTAKE RADIATION MONITOR.
EVENT DATE: 062689 REPORT DATE: 072589 NSSS: WE TYPE: PWR

(NSIC 214936) ON 6/26/89, AT APPROX. 1615 CDT, A CONTROL ROOM VENTILATION ISOLATION SIGNAL WAS INITIATED WHEN CONTROL ROOM AIR INTAKE RADIATION MONITOR GKRE-04 EXPERIENCED A LOSS OF POWER. ALL AVAILABLE ENGINEERED SAFETY FEATURES EQUIPMENT RESPONDED PROPERLY. THE 'B' TRAIN CONTROL ROOM AIR CONDITIONING UNIT HAD PREVIOUSLY BEEN REMOVED FROM SERVICE FOR MAINTENANCE. GKRE-04 WAS PLACED IN BYPASS FOR TROUBLESHOOTING AND THE AFFECTED VENTILATION SYSTEMS WERE RESTORED TO NORMAL CONFIGURATION AT APPROX. 1710 CDT. DURING TROUBLESHOOTING, A BLOWN FUSE IN THE POWER SUPPLY TO THE RM-80 MICROPROCESSOR ASSOCIATED WITH GKRE-04 WAS DISCOVERED. WHEN THE FUSE BLEW, THE MONITOR EXPERIENCED A LOSS OF POWER. IT WAS DETERMINED THAT A VOLTAGE FLUCTUATION HAD LIKELY CAUSED THE BLOWN FUSE, BUT IDENTIFICATION OF THE ROOT CAUSE WAS NOT POSSIBLE. THE FUSE WAS REPLACED, AND GKRE-04 WAS RETURNED TO SERVICE ON 6/29/89, AT APPROX. 1504 CDT FOLLOWING SATISFACTORY COMPLETION OF SURVEILLANCE TESTING.

[269] WOLF CREEK 1 DOCKET 50-482 LER 89-013
VALVE LEAKAGE AND INCREASED STEAM DEMAND CAUSE ENGINEERED SAFETY FEATURES ACTUATIONS FROM STEAM GENERATOR HIGH-HIGH LEVEL.
EVENT DATE: 071189 REPORT DATE: 080989 NSSS: WE TYPE: PWR

(NSIC 215009) ON 7/11/89, AT APPROXIMATEY 0224 CDT, WITH THE UNIT IN MODE 2, STARTUP, WITH REACTOR POWER AT APPROXIMATELY 1%, A FEEDWATER ISOLATION SIGNAL (FWIS) OCCURRED WHEN THE WATER IN STEAM GENERATOR (S/G) 'A' INCREASED TO THE HIGH-HIGH LEVEL SETPOINT. CONSEQUENTLY, A MAIN FEEDWATER PUMP TRIP, A MAIN TURBINE TRIP SIGNAL, S/G BLOWDOWN AND SAMPLE ISOLATION SIGNAL AND AN AUXILIARY FEEDWATER ACTUATION SIGNAL WERE INITIATED. ALL ENGINEERED SAFETY FEATURES EQUIPMENT REQUIRED TO OPERATE RESPONDED PROPERLY. THE WATER LEVEL IN S/G 'A' WAS DECREASED BELOW THE HIGH-HIGH LEVEL SETPOINT AND THE ACTUATED SYSTEMS WERE RETURNED TO NORMAL CONFIGURATIONS AT APPROXIMATELY 0400 CDT. THE ANALYSIS OF THIS EVENT CONCLUDED THAT LEAKAGE THROUGH THE MAIN FEEDWATER BYPASS CONTROL VALVES COMBINED WITH THE S/G "SWELL" INDUCED BY THE OPENING OF THE STEAM DUMP VALVES CAUSED S/G 'A' LEVEL TO REACH THE HIGH-HIGH LEVEL SETPOINT, THEREBY INITIATING THE FWIS. TO PREVENT RECURRENCE OF THIS EVENT, THIS LICENSEE EVENT REPORT WILL BE ADDED TO REQUIRED READING FOR ALL OPERATIONS PERSONNEL TO ALERT THE OPERATORS TO THE POTENTIAL FOR LEAKAGE THROUGH THE MAIN FEEDWATER BYPASS CONTROL VALVES. A CAUTION IS BEING ADDED TO THE APPROPRIATE PROCEDURE TO SERVE AS A REMINDER OF THIS POTENTIAL FOR LEAKAGE.

[270] WOLF CREEK 1 DOCKET 50-482 LER 89-015
PERSONNEL ERROR CAUSES INCORRECT VALVE LINEUP RESULTING IN A FAILURE TO OBTAIN REQUIRED TURBINE BUILDING SUMP SAMPLE.
EVENT DATE: 071689 REPORT DATE: 081589 NSSS: WE TYPE: PWR

(NSIC 215040) ON JULY 17, 1989, IT WAS DISCOVERED THAT A VIOLATION OF THE SAMPLING REQUIREMENTS OF TECHNICAL SPECIFICATION 3.11.1.1 HAD OCCURRED. ON JULY 16, 1989, A DAILY GRAB SAMPLE WAS NOT OBTAINED FOR A CONTINUOUS LIQUID EFFLUENT

RELEASE TO AN UNRESTRICTED AREA. PREVIOUSLY, ON JULY 15, 1989, A VALVE LINEUP WAS PERFORMED TO DIRECT THE DISCHARGE OF SEVERAL TURBINE BUILDING SUMP DISCHARGE PATHWAY TO ITS NORMAL CONFIGURATION TO THE SECONDARY LIQUID WASTE OIL INTERCEPTOR. THIS ERROR RESULTED IN SIX TURBINE BUILDING SUMPS STILL BEING DISCHARGED TO THE SITE OIL SEPARATOR. UNAWARE OF THIS CONDITION, OPERATIONS PERSONNEL INFORMED CHEMISTRY PERSONNEL THAT THE DISCHARGE HAD BEEN TERMINATED AND THAT NO SAMPLING WAS REQUIRED. ON JULY 17, 1989, THE VALVE LINEUP ERROR WAS DISCOVERED AND PROMPTLY CORRECTED. SUBSEQUENT SAMPLING AND ANALYSIS BY CHEMISTRY PERSONNEL CONFIRMED THAT NO RELEASE LIMITS WERE EXCEEDED. THIS EVENT WAS THE RESULT OF COGNITIVE PERSONNEL ERROR BY UTILITY LICENSED PERSONNEL. AN OVERSIGHT DURING PIPING & INSTRUMENTATION DIAGRAM REVIEW RESULTED IN AN INCORRECT VALVE LINEUP. THIS REPORT WILL BE ADDED TO OPERATIONS REQUIRED READING TO ENSURE OTHER PERSONNEL ARE AWARE OF THIS EVENT AND ITS CONSEQUENCES.

[271] WOLF CREEK 1 DOCKET 50-482 LER 89-014
CONTROL ROOM INTAKE RADIATION MONITOR CAUSES CONTROL ROOM VENTILATION ISOLATION AS A RESULT OF VOLTAGE TRANSIENT.
EVENT DATE: 071889 REPORT DATE: 081789 NSSS: WE TYPE: PWR

(NSIC 215041) ON JULY 18, 1989, AT APPROXIMATELY 2336 CDT, A CONTROL ROOM VENTILATION ISOLATION SIGNAL (CRVIS) WAS INITIATED BY A SPIKE ON CONTROL ROOM AIR INTAKE RADIATION MONITOR GKRE-05. ALL REQUIRED ENGINEERED SAFETY FEATURES EQUIPPED RESPONDED PROPERLY. GKRE-05 WAS PLACED IN BYPASS FOR TROUBLESHOOTING AND THE AFFECTED SYSTEMS WERE RESTORED TO NORMAL CONFIGURATION AT APPROXIMATELY 0013 CDT ON JULY 19, 1989. IT WAS DETERMINED DURING TROUBLESHOOTING THAT A SPIKE ON GKRE-05 COULD BE INITIATED BY PARTIALLY UNSCREWING THE OPERATE LAMP AND MOVING IT SIDE TO SIDE. IT WAS ALSO REVEALED THAT THE LAMP WAS FOUND TO BE BURNED OUT FOLLOWING THE CRVIS BUT HAD BEEN ILLUMINATED PRIOR TO THE CRVIS. IT IS BELIEVED THAT A VOLTAGE TRANSIENT ASSOCIATED WITH THE LAMP BURNING OUT CAUSED THE CRVIS. AN ENGINEERING EVALUATION IS BEING CONDUCTED TO FURTHER EVALUATE THE SENSITIVITY OF GKRE-05 AND OTHER MONITORS TO VOLTAGE TRANSIENTS. GKRE-05 WAS RESTORED TO SERVICE ON JULY 20, 1989, AT 0759 CDT.

[272] WPPSS 2 DOCKET 50-397 LER 89-024
SECONDARY CONTAINMENT BYPASS LEAKAGE FOUND TO BE GREATER THAN ALLOWED BY DESIGN BASIS AS A RESULT OF EQUIPMENT DESIGN DEFICIENCY.
EVENT DATE: 061489 REPORT DATE: 071489 NSSS: GE TYPE: BWR

(NSIC 214825) ON JUNE 14, 1989, DURING A PLANT DESIGN REVIEW FOR UNMONITORED CONTAINMENT RELEASE PATHS, A PLANT DESIGN ENGINEER IDENTIFIED A POTENTIAL PATH THROUGH WHICH RADIOACTIVE LIQUID FROM THE PRIMARY CONTAINMENT COULD BYPASS THE LEAKAGE COLLECTION AND FILTRATION SYSTEMS ASSOCIATED WITH THE SECONDARY CONTAINMENT. THE BYPASS LEAKAGE PATH IDENTIFIED WAS THE CONTROL ROD DRIVE (CRD) SYSTEM HYDRAULIC CONTROL UNIT (HCU) CONTROL VALVES. THIS POSTULATED EVENT WAS EVALUATED AS HAVING THE POTENTIAL TO RESULT IN BYPASS LEAKAGE OF A QUANTITY LARGE ENOUGH TO VIOLATE THE WNP-2 DESIGN BASIS FOR CONTROL ROOM HABITABILITY RADIATION DOSE LIMITS AFTER A LOSS OF COOLANT ACCIDENT (LOCA). AN ENGINEERING ASSESSMENT DETERMINED THAT INSTALLATION OF TWO CHECK VALVES IN SERIES IN THE COMMON DISCHARGE LINE OF THE CRD PUMPS WOULD PROVIDE A BARRIER SUFFICIENT TO PREVENT BYPASS LEAKAGE THROUGH THE CRD SYSTEM. THESE VALVES WERE INSTALLED ON JUNE 24, 1989, PRIOR TO THE END OF THE REFUELING OUTAGE BEFORE THE PLANT STARTUP. THE IMMEDIATE CAUSE OF THIS EVENT WAS EQUIPMENT DESIGN DEFICIENCY IN THAT THE CRD SYSTEM DESIGN WAS POTENTIALLY NOT CAPABLE OF PREVENTING POST LOCA LIQUID BYPASS LEAKAGE FROM EXCEEDING THE DESIGN BASIS LIMIT. THE PLANT DESIGN REVIEW TO VERIFY THAT THERE ARE NO OTHER POTENTIAL UNMONITORED RELEASE PATHS WILL CONTINUE AS ORIGINALLY DESCRIBED IN LER 88-012-00.

[273] WPPSS 2 DOCKET 50-397 LER 89-026
POTENTIAL FAILURE OF PENETRATION SEALS COULD CAUSE FAILURE OF SAFETY EQUIPMENT IN SECONDARY CONTAINMENT FOLLOWING POSTULATED DESIGN BASIS STEAMLINE BREAK IN STEAM TUNNEL.
EVENT DATE: 061989 REPORT DATE: 071989 NSSS: GE TYPE: BWR
VENDOR: DOW CORNING CORP.

(NSIC 214895) ON 6/19/89, TESTING CONFIRMED THAT SELECTED PENETRATIONS IN THE STEAM TUNNEL COULD FAIL TO PERFORM AS A PRESSURE BOUNDARY FOLLOWING A DESIGN BASIS MAIN STEAMLINE BREAK IN THE STEAM TUNNEL. AS A RESULT, QUALIFICATION LIMITS COULD BE EXCEEDED FOR SAFETY EQUIPMENT IN THE REACTOR BUILDING (SECONDARY CONTAINMENT). REVIEW AND IDENTIFICATION OF THIS CONDITION OCCURRED AS PART OF BY THE CURRENT REVIEW PROCESS OF PLANT DESIGN CHANGES. ALSO, THIS EVENT IS REPORTABLE PER 10CFR PART 21 AS A DEFICIENCY IN THE SEAL DESIGN OF THE STEAM TUNNEL PENETRATIONS BY THE PLANT ARCHITECT/ENGINEER BURNS & ROE, INC. DURING REVIEW OF A DESIGN CHANGE TO CORE DRILL AND SEAL TWO NEW 2 AND 1/2-INCH DIAMETER PENETRATIONS FROM THE STEAM TUNNEL TO THE REACTOR BUILDING, IT WAS DETERMINED THAT THE PROPOSED SEALANT WAS NOT PRESSURE RATED FOR DIAMETERS GREATER THAN 3/4-INCH. INSPECTION DETERMINED 11 EXISTING PENETRATIONS IN THE STEAM TUNNEL REQUIRED MODIFICATION TO RESIST THE POSTULATED DESIGN BASIS PRESSURES. AN URGENT PLANT MODIFICATION REQUEST WAS INITIATED TO MODIFY THE 11 PENETRATIONS PRIOR TO PLANT STARTUP. ROOT CAUSES OF INADEQUATE PRESSURE BOUNDARIES IN THE STEAM TUNNEL INCLUDE 1) LESS THAN ADEQUATE DESIGN, AND 2) MANAGEMENT PROGRAMS OF THE AE DID NOT ENSURE THE SEAL DESIGNS WERE COMPATIBLE WITH THE DESIGN BASIS REQUIREMENTS.

[274] WPPSS 2 DOCKET 50-397 LER 89-028
TURBINE THROTTLE VALVE CLOSURE REACTOR SCRAM DURING TURBINE TESTING CAUSED BY INADEQUATE PROCEDURE.
EVENT DATE: 062989 REPORT DATE: 072789 NSSS: GE TYPE: BWR

(NSIC 214963) 6/29/89, AT 0020 HOURS, DURING TESTING OF THE MAIN TURBINE OVERSPEED PROTECTION TROLLER (OPC), A TURBINE THROTTLE VALVE CLOSURE REACTOR SCRAM OCCURRED AT THE TIME OF THE SCRAM, THE TURBINE WAS OPERATING IN THROTTLE VALVE CONTROL COASTING DOWN FROM THE ACTUATION SETPOINT OF 1854 RPM. AFTER 3 UNSUCCESSFUL ATTEMPTS TO INPUT AN 1800 SPEED DEMAND, THE OPERATOR DECIDED TO WAIT UNTIL ACTUAL TURBINE SPEED HAD DROPPED BELOW 1800 RPM TO TRY AGAIN. AFTER TURBINE SPEED DROPPED BELOW 1800 RPM, THE OPERATOR KEYED IN ANOTHER 1800 RPM REFERENCE DEMAND AT 50 RPM PER MINUTE AS SPECIFIED IN THE PROCEDURE. THIS INPUT WAS ACCEPTED BY THE TURBINE CONTROL SYSTEM. THE OPERATOR, AS EXPECTED BY THE PROCEDURE, THEN RETURNED TURBINE CONTROL TO THE "IN SERVICE" POSITION WITH THE OPC KEYLOCK SWITCH. THIS RESULTED IN RAPID OPENING OF THE TURBINE GOVERNOR AND THROTTLE VALVES AND SUBSEQUENT PRESSURIZATION OF THE TURBINE FIRST STAGE CHAMBER SINCE THROTTLE VALVES WERE LESS THAN 95% OPEN AT THIS POINT, THIS CAUSED ACTUATION OF THE TURBINE FIRST STAGE PRESSURE SWITCHES CAUSING THE SCRAM LOGIC TO BE MET RESULTING IN A TURBINE THROTTLE VALVE CLOSURE REACTOR SCRAM. THE ROOT CAUSE OF THE EVENT WAS AN INADEQUATE PROCEDURE IN THAT THE TURBINE OPERATING PROCEDURE DID NOT PREVENT THE OPC TEST FROM BEING DONE WITH THE TURBINE IN THE THROTTLE VALVE CONTROL MODE. CORRECTIVE ACTIONS TAKEN CONSIST OF DEVIATION OF THE OPERATING PROCEDURE.

[275] WPPSS 2 DOCKET 50-397 LER 89-027
INADEQUATE SEISMIC RESTRAINT OF ISOLATION VALVES COULD RESULT IN UNISOLATABLE BREACH OF PRIMARY CONTAINMENT CAUSED BY INADEQUATE WORK PRACTICES.
EVENT DATE: 063089 REPORT DATE: 072889 NSSS: GE TYPE: BWR

(NSIC 214962) ON 6/30/89, PRELIMINARY ENGINEERING EVALUATION DETERMINED THAT TWO SEISMIC SUPPORTS MISSING ON EACH OF TWO POST ACCIDENT SAMPLING SYSTEM (PASS) CONTAINMENT REGULATION VALVES, FOUND BY A DESIGN ENGINEER ON 6/27/89, WOULD PROBABLY RESULT IN LURE OF THE PIPE AT ITS PRIMARY CONTAINMENT PENETRATION DURING

A DESIGN BASIS EARTHQUAKE (DBE). THIS WOULD CREATE AN UNISOLATABLE BREACH OF PRIMARY CONTAINMENT. THE FOUND" CONDITION WAS DISCOVERED WHILE THE DESIGN ENGINEER WAS PERFORMING A VISUAL INSPECTION OF PLANT SUPPORTS AND WHILE THE PLANT WAS AT 3% POWER AND IN MODE 2 (STARTUP). AT 1650 HOURS ON 6/30/89 THE PRIMARY CONTAINMENT TECHNICAL SPECIFICATION ACTION STATEMENT 3.6.1.1 WAS ENTERED AND PREPARATIONS WERE MADE TO RESTORE RESTRAINTS TO THE REQUIRED PLANT CONFIGURATION. AT 1745 HOURS WHEN WORK WAS NOT COMPLETED ON THE RESTRAINTS, A PLANT SHUTDOWN WAS INITIATED. PRIMARY CONTAINMENT TECHNICAL SPECIFICATION STATEMENT WAS EXITED AT 1843 HOURS WHEN THE RESTRAINTS WERE RESTORED. THE ROOT CAUSES OF THE EVENT ARE 1) LESS THAN ADEQUATE WORK PRACTICES TO ENSURE THE PLANT FIGURATION REMAINS WITHIN DESIGN REQUIREMENTS, AND 2) LESS THAN ADEQUATE TRAINING OF SUBJECT PERSONNEL TO IMPLIMENT PLANT MODIFICATIONS. NO FURTHER CORRECTIVE ACTIONS WERE IDENTIFIED THAT WOULD SIGNIFICANTLY MINIMIZE THE OCCURRENCE OF THIS CONDITION IN FUTURE MODIFICATIONS.

[276] WPPSS 2 DOCKET 50-397 LER 89-029
REACTOR WATER CLEANUP AND REACTOR CORE ISOLATION COOLING SYSTEM ISOLATIONS CAUSED BY INADEQUATE TEST/SURVEILLANCE PROCEDURE.
EVENT DATE: 070389 REPORT DATE: 080189 NSSS: GE TYPE: BWR

(NSIC 215003) ON JULY 3, 1989 AT 1022 HOURS A REACTOR WATER CLEANUP (RWCU) SYSTEM ISOLATION OCCURRED WHEN RWCU-V-1 CLOSED AS PART OF A GROUP 7 NUCLEAR STEAM SUPPLY SHUTOFF SYSTEM (NS4) ISOLATION. AT 1026 HOURS A SIMILAR REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM ISOLATION OCCURRED WHEN RCIC-V-63 CLOSED. BOTH THESE ISOLATIONS OCCURRED AS A RESULT OF TECHNICAL SPECIFICATION SURVEILLANCE TESTING PER PLANT PROCEDURE 7.4.3.2.1.6, LEAK DETECTION MONITOR DIVISION II CHANNEL CALIBRATION (CC)/CHANNEL FUNCTIONAL TEST (CFT). THE ROOT CAUSE OF THIS ESF ACTUATION WAS LESS THAN ADEQUATE PREPARATION AND REVIEW OF A SURVEILLANCE TEST PROCEDURE WHICH HAD BEEN REWRITTEN TO SUPPORT A PLANT MODIFICATION. THIS MODIFICATION HAD RECENTLY BEEN COMPLETED DURING A REFUELING OUTAGE. IMMEDIATE CORRECTIVE ACTION CONSISTED OF PROMPT ACTION BY THE PLANT OPERATORS TO RETURN THE IMPACTED SYSTEMS TO SERVICE. THE SURVEILLANCE TEST PROCEDURE WAS ALSO CORRECTED AND DEVIATED. FURTHER CORRECTIVE ACTION WILL BE TAKEN BY PLANT MAINTENANCE TO STRENGTHEN THEIR PROCEDURE MODIFICATION PROCESS. IN ADDITION, TESTING AND PROJECT MANAGEMENT PROCEDURES WILL BE MODIFIED TO REQUIRE, WHEN PRACTICAL, A MORE COMPLETE CHECK OF THE MODIFIED SURVEILLANCE PRIOR TO PLANT OPERATION AND TO ALLOW MORE TIME FOR REVIEW AND PREPARATION OF PROCEDURES ASSOCIATED WITH OUTAGE WORK.

[277] YANKEE ROWE DOCKET 50-029 LER 88-008 REV 01
UPDATE ON REACTOR/TURBINE TRIP ON LOSS OF GENERATOR FIELD EXCITATION.
EVENT DATE: 051788 REPORT DATE: 080489 NSSS: WE TYPE: PWR
VENDOR: GENERAL ELECTRIC CO.

(NSIC 214970) AT 2323 HOURS, 5/17/88, IN MODE 1 AT FULL POWER OPERATION (100%), A LOSS OF GENERATOR FIELD EXCITATION RESULTED IN AUTOMATIC TURBINE GOVERNOR RESPONSE AND SUBSEQUENT RELAY ACTION WHICH TRIPPED BK-1 AND BK-2 (REACTOR TRIP BREAKERS). THE LOSS OF GENERATOR FIELD EXCITATION RESULTED FROM AUTOMATIC TRIPPING OF THE AC FEED TO THE STATIC EXCITER. THE HARRIMAN (Z-126) TRANSMISSION LINE ALSO DENERGIZED DURING THIS EVENT. LOSS OF BOTH GENERATOR EXCITATION AND ONE OF TWO TRANSMISSION LINES RESULTED IN A LOSS OF FLOW FROM 3 OF 4 MAIN COOLANT PUMPS (MCPS). THE OPERATORS SECURED MCP #1 APPROXIMATELY 2 MINUTES AFTER PLANT TRIP AND ESTABLISHED NATURAL CIRCULATION. BY 2330 HOURS THE Z-126 LINE HAD BEEN RE-ENERGIZED. BY 2345 HOURS THE ELECTRICAL BUSSES HAD BEEN CROSS-TIED AND RESTART OF THE MCPS HAD COMMENCED. AT 0005 HOURS, 5/18/88, ALL 4 MCPS WERE OPERATING. ALL AUTOMATIC SAFETY SYSTEMS FUNCTIONED AS DESIGNED: THE PLANT EMERGENCY DIESEL GENERATORS NO. 2 AND 3 STARTED AS REQUIRED. THE ROOT CAUSE OF THIS EVENT WAS A FAILURE OF THE FIELD OVERVOLTAGE PROTECTION UNIT CIRCUIT BAORD. CORRECTIVE ACTION INVOLVES IMPLEMENTING A PLANT DESIGN CHANGE THAT WILL MODIFY THE STATIC EXCITER'S INTERNAL CONTROL AND 2400V FEEDER CIRCUIT BREAKER CONTROL.

THERE WAS NO ADVERSE EFFECT TO THE PUBLIC HEALTH OR SAFETY. THIS IS THE FIRST OCCURRENCE OF THIS NATURE AT THIS FACILITY.

[278] YANKEE ROWE DOCKET 50-029 LER 89-009
TURBINE BUILDING SUMP COMPOSITE SAMPLER INOPERABLE.
EVENT DATE: 071889 REPORT DATE: 081789 NSSS: WE TYPE: PWR

(NSIC 215053) ON 7/18/89, WITH THE PLANT IN MODE 1 AT 100% POWER, THE TURBINE BLDG. SUMP (TBS) COMPOSITE PUMP WAS FOUND ELECTRICALLY UNPLUGGED. THIS PUMP IS USED TO PROVIDE A CONTINUOUS COMPOSITE SAMPLE OF THE TURBINE BLDG. SUMP LIQUID EFFLUENT PER T.S. TABLE 3.3-8, ITEM 2.C TO MEET THE RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS REQUIREMENT OF T.S. TABLE 4.11-1, ITEM B. THE PUMP WAS INOPERABLE BETWEEN 7/10/89 AND 7/18/89. THE CAUSE OF THE EVENT IS ATTRIBUTED TO IMPROPER PERFORMANCE OF THE SURVEILLANCE PROCEDURE PERFORMED 7/10/89, AND A PROCEDURAL INADEQUACY WITH THE DAILY CHANNEL CHECK. THE CHEMIST INVOLVED IN THE SURVEILLANCE PROCEDURE HAS BEEN RE-INSTRUCTED IN PROPER PROCEDURE PERFORMANCE. INSTRUCTIONS FOR THE DAILY CHANNEL CHECK HAVE BEEN REVISED TO REQUIRE PHYSICALLY CHECKING THE PLUG. THIS IS CONSIDERED AN ISOLATED INCIDENT; THEREFORE, NO FURTHER CORRECTIVE ACTION IS DEEMED NECESSARY. TBS COMPOSITE SAMPLE RESULTS PRIOR TO AND FOLLOWING THE EVENT, AS WELL AS RECENT HISTORICAL DATA FOR THIS SAMPLE STATION, INDICATE NO RADIOACTIVITY ABOVE INSTRUMENTATION DETECTION LIMITS REQUIRED BY TECH SPECS. THERE WAS NO IMPACT ON THE HEALTH AND SAFETY OF THE PUBLIC.

[279] YANKEE ROWE DOCKET 50-029 LER 89-010
POTENTIAL FOR EXCEEDING RATED CAPACITY OF NO. 3 EMERGENCY GENERATOR DIESEL.
EVENT DATE: 072489 REPORT DATE: 082489 NSSS: WE TYPE: PWR

(NSIC 215102) ON 7/24/89 IT WAS IDENTIFIED THAT A POTENTIAL HAD PREVIOUSLY EXISTED FOR EXCEEDING THE 400 KW RATED CAPACITY OF NO. 3 EMERGENCY DIESEL GENERATOR (EDG) (EIIS: DG) BY 4.5%. THIS CONDITION WAS DISCOVERED AS A RESULT OF TESTS RUN WITH NO. 3 LOW PRESSURE SAFETY INJECTION PUMP (LPSI) (EIIS: P). THE TESTS, CONDUCTED ON 3/29/89 AND 4/27/89 SHOWED THAT THE LPSI PUMPS WERE OPERATING IN A CAVITATING MODE DURING NORMAL SURVEILLANCE TESTING. IT WAS ALSO DETERMINED THAT THE PUMP POWER REQUIREMENT DURING NON-CAVITATING OPERATION WAS GREATER THAN THAT UTILIZED IN THE EDG LOADING CALCULATIONS. THE INPUT PARAMETERS TO THE ORIGINAL EDG LOADING CALCULATIONS WERE OBTAINED WITH THE LPSI SYSTEM ALIGNED IN THE NORMAL INJECTION MODE. IN THIS ALIGNMENT THE LPSI PUMPS ARE AT RUNOUT FLOW WHICH WAS ASSUMED TO PRODUCE THE MAXIMUM PUMP POWER REQUIREMENT. THE ROOT CAUSE OF THIS EVENT HAS BEEN ATTRIBUTED TO THE FACT THAT THE PUMP WAS OPERATING IN CAVITATION DURING SURVEILLANCE TESTING, WHICH WAS NOT RECOGNIZED, RESULTING IN LOWER THAN EXPECTED PUMPING POWER UTILIZED IN THE EDG LOADING CALCULATION. ADMINISTRATIVE CONTROLS WILL BE ADDED TO EXISTING PROCEDURES TO ENSURE ADHERENCE TO PUMP AND DIESEL GENERATOR OPERATING PARAMETERS. THIS IS THE FIRST EVENT OF THIS KIND.

[280] YANKEE ROWE DOCKET 50-029 LER 89-011
INADEQUATE DEENERGIZATION OF 480V EMERGENCY BUS NO. 1.
EVENT DATE: 072589 REPORT DATE: 082489 NSSS: WE TYPE: PWR

(NSIC 215103) ON 7/25/89, WHILE IN MODE 1 AT 100% POWER, CORRECTIVE MAINTENANCE WAS BEING PERFORMED ON THE REVERSE POWER RELAY FOR EMERGENCY DIESEL GENERATOR (EDG) NO. 1. AT 1119 HOURS, A POTENTIAL TRANSFORMER FUSE WAS REMOVED, RESULTING IN THE INADVERTENT DEENERGIZATION OF THE 480V EMERGENCY BUS NO. 1 AND A SUBSEQUENT LOSS OF POWER TO VARIOUS PLANT EQUIPMENT AND INSTRUMENTATION. EMERGENCY BUS NO. 1 AND ITS ASSOCIATED LOADS WERE REENERGIZED AT 1133 HOURS; IN ACCORDANCE WITH PLANT PROCEDURES, A SIMULTANEOUS DECLARATION AND TERMINATION OF AN UNUSUAL EVENT OCCURRED. THE NRC WAS NOTIFIED VIA ENS AT 1215 HOURS. THE

PRIMARY ROOT CAUSES OF THIS EVENT WERE THAT THE EDG SYSTEM WAS NOT PROPERLY TAGGED OUT OF SERVICE AND AN ADEQUATE, TECHNICAL REVIEW OF A TEMPORARY CHANGE REQUEST (TCR) WAS NOT PERFORMED. CORRECTIVE ACTIONS INCLUDE: (1) A PROCEDURE FOR TAKING THE EDGS OUT OF SERVICE HAS BEEN DEVELOPED, (2) TRAINING OF OPERATIONS PERSONNEL ON THIS PROCEDURE IS PLANNED, (3) THE PROCEDURE THAT CONTROLS TCRS HAS BEEN REVISED TO ENSURE AN INDEPENDENT REVIEW FOR TCRS TO SAFETY-RELATED EQUIPMENT AND (4) TRAINING OF SUPERVISORS ON THIS REVISED PROCEDURE AND 10CFR50.59 REVIEWS IS PLANNED. THIS IS THE FIRST OCCURRENCE OF THIS NATURE AT THIS FACILITY.

[281] ZION 1 DOCKET 50-295 LER 88-006 REV 01
UPDATE ON SERVICE WATER PUMP FLOWRATE BELOW REQUIRED DILUTION FLOW DURING A RADIOACTIVE LIQUID WASTE RELEASE.
EVENT DATE: 032588 REPORT DATE: 071489 NSSS: WE TYPE: PWR
VENDOR: LAYNE-BOWLER, INC.

(NSIC 214972) DURING THE REVIEW OF T.S.S.P. 16-88 (SERVICE WATER PUMP PERFORMANCE TEST) IT WAS NOTED THAT THE MEASURED CAPACITY FOR 2 SERVICE WATER PUMPS WAS APPROXIMATELY 32,000 GALLONS PER MINUTE. THIS FLOWRATE IS INSUFFICIENT TO PROVIDE THE MINIMUM DILUTION FLOW OF 44,000 GALLONS PER MINUTE (GPM) REQUIRED DURING A RADIOACTIVE LIQUID WASTE RELEASE, PER TECH SPEC 3.11.1.B. THE ROOT CAUSE WAS IDENTIFIED AS AN INCORRECT ASSUMPTION ABOUT THE ACTUAL FLOW OUTPUT OF TWO SERVICE WATER PUMPS. TECH SPEC 3.11.1.B WAS ADDED TO ZION'S TECH SPECS IN 1978. AT THAT TIME, IT WAS ASSUMED THAT THE PUMPS WERE OPERATING AT RATED FLOW (22,000 GPM EACH) BUT IN REALITY THEY TYPICALLY OPERATE AT A LOWER POINT ON THEIR PUMP CURVES. A SEARCH OF ALL RADIOACTIVE LIQUID RELEASE FORMS GENERATED SINCE 1978 RESULTED IN ONLY ONE OCCURRENCE WHERE DILUTION FLOW WAS BELOW TECH SPEC MINIMUM. CALCULATIONS PERFORMED BY THE CHEMISTRY DEPARTMENT INDICATED THAT NO 10CFR20 LIMITS FOR ISOTOPE CONCENTRATIONS IN THE LIQUID WASTE STREAM WERE EXCEEDED. A TECH SPEC CHANGE DELETING SECION 3.11 AND TRANSFERRING IT TO THE OFFSITE DOSE CALCULATION MANUAL WILL BE PURSUED.

[282] ZION 1 DOCKET 50-295 LER 89-009
INADVERTENT START OF AUX FEEDWATER PUMP DUE TO PERSONNEL ERROR.
EVENT DATE: 062089 REPORT DATE: 072089 NSSS: WE TYPE: PWR

(NSIC 214917) ON 6/20/89, UNIT 1 WAS AT FULL POWER WITH PROCEDURE PT-7A "STARTING PROCEDURE FOR AUXILIARY FEEDWATER (AFW) PUMP LUBE OIL PUMPS" IN PROGRESS. AT 1020 HOURS, THE EQUIPMENT ATTENDANT (B-MAN) STATIONED AT THE REMOTE SHUTDOWN PANEL (RSP) ATTEMPTED TO START THE AFW PUMP LUBE OIL PUMP FROM THE RPS, AS PER THE PROCEDURE. DUE TO DISTRACTION IN THE ROOM AND INCONSISTENT PUMP SWITCH LAYOUT, THE B-MAN INADVERTENTLY STARTED THE AFW PUMP. THE B-MAN RECOGNIZED HIS ERROR, BUT COULD NOT STOP THE PUMP LOCALLY. AT 1030 HOURS, THE PUMP WAS SECURED FROM THE MAIN CONTROL BOARD. INVESTIGATION SHOWED THAT THE START/STOP LABELLING ON THE PUMP SWITCH WAS REVERSED AT THE RSP. SAFETY SIGNIFICANCE WAS MINIMAL. NO EQUIPMENT FAILED AND CONTROL OF THE PUMP COULD ALWAYS BE TRANSFERRED BACK TO THE MAIN CONTROL BOARD. THE LABELLING AND SWITCH LAYOUT WILL BE CORRECTED.

[283] ZION 1 DOCKET 50-295 LER 89-010
TRAIN A REACTOR TRIP DEFEAT DUE TO INSTALLATION OF UNAUTHORIZED JUMPER.
EVENT DATE: 071189 REPORT DATE: 081089 NSSS: WE TYPE: PWR

(NSIC 214986) ON 7/11/89 AT 1300 WHILE PERFORMING A WALK DOWN IN REACTOR PROTECTION CABINET 1CB30, A TECHNICAL STAFF ENGINEER NOTICIED AN UNMARKED JUMPER. THE JUMPER SIMULATED AN OPEN SIGNAL ON A TURBINE STOP VALVE PREVENTING THE ABILITY TO OBTAIN A 4 OUT OF 4 TURBINE STOP VALVES CLOSED SIGNAL TO THE REACTOR PROTECTION SYSTEM. THIS JUMPER WOULD NOT HAVE PREVENTED A REACTOR TRIP. BOTH AUTO STOP OIL TRIPS AND 4 CLOSED STOP VALVES INDEPENDENTLY CAUSE THE TURBINE TO TRIP. THE TURBINE IN TURN CAUSES THE REACTOR TO TRIP. THE AUTO STOP OIL TRIPS

OCCUR PHYSICALLY PRIOR TO 4 CLOSED STOP VALVES. THE REACTOR WOULD TRIP EVEN IF THE SUBJECT JUMPER WAS INSTALLED. THE AUTO STOP OIL TRIPS ARE TESTED DURING PT-5A REACTOR PROTECTION LOGIC REACTOR AT HOT SHUTDOWN AND WERE ABLE TO FUNCTION IF NEEDED. TRAIN B LOGIC WAS AVAILABLE AND THERE WERE NO JUMPERS INSTALLED IN THE REACTOR PROTECTION TRAIN B CABINETS. THE SAFETY SIGNIFICANCE WAS MINIMAL DUE TO THE FACT THAT A REACTOR TRIP DUE TO TURBINE TRIP WAS ALWAYS POSSIBLE. THE APPARENT CAUSE OF THE EVENT WAS A PERSONNEL ERROR IN THAT A TEST RELATED JUMPER WAS NOT PROPERLY REMOVED FOLLOWING TESTING. THE CORRECTIVE ACTIONS WILL INCLUDE A TEST PROCEDURE REVIEW TO DETERMINE WHICH SAFETY-RELATED PROCEDURES REQUIRE JUMPER INSTALLATION. PROCEDURES IDENTIFIED AS DEFICIENT WILL BE REVISED TO REQUIRE INDEPENDENT VERIFICATION OF INSTALLATION AND REMOVAL OF ALL JUMPERS.

[284] ZION 1 DOCKET 50-295 LER 89-011
INADVERTENT ENGINEERED SAFETY FEATURE (ESF) ACTUATION DURING PT-10 TESTING DUE TO PERSONNEL ERROR.
EVENT DATE: 071289 REPORT DATE: 081489 NSSS: WE TYPE: PWR

(NSIC 215045) ON 7/12/89, WHILE PERFORMING PERIODIC TEST (PT)-10, SAFEGUARDS ACTUATION UNIT 1, SEVERAL CONTAINMENT ISOLATION VALVES WERE INADVERTENTLY ACTUATED WHEN THE TEST SWITCH WAS REMOVED FROM THE TEST POSITION PRIOR TO RESETTING THE PHASE A ISOLATION SIGNAL. THE TEST SWITCH WAS IMMEDIATELY RETURNED TO THE TEST POSITION, THE PHASE A ISOLATION SIGNAL RESET, AND THE TEST SWITCH RETURNED TO THE NORMAL POSITION. THE AFFECTED CONTAINMENT ISOLATION VALVES WERE THEN REPOSITIONED TO THEIR NORMAL POSITION. THE CAUSE OF THE EVENT IS A PERSONNEL ERROR AGGRAVATED BY A PROCEDURE REQUIRING HUMAN FACTORS ENHANCEMENT. THE CORRECTIVE ACTIONS INCLUDE OPERATOR TRAINING AND PROCEDURE REVISIONS. THERE HAVE BEEN THREE PREVIOUS OCCURRENCES INVOLVING IMPROPER OPERATION OF THIS TEST CIRCUIT. IN ALL 3 CASES, THE TEST SWITCH WAS PLACED IN THE NORMAL POSITION PRIOR TO RESETTING THE PHASE A ACTUATION SIGNAL.

[285] ZION 1 DOCKET 50-295 LER 89-012
INADEQUATE AFW FLOW SETTINGS WITH INOPERABLE AFW PUMP DUE TO ADMINISTRATIVE ERRORS AND PROCEDURAL INADEQUACIES.
EVENT DATE: 072389 REPORT DATE: 082289 NSSS: WE TYPE: PWR

(NSIC 215064) AT THE TIME OF THIS EVENT, THE UNIT WAS OPERATING AT 96% POWER, 559 T-AVERAGE, AND REACTOR COOLANT SYSTEM PRESSURE OF 2235 PSIG. AT 1840 HOURS ON 7/23/89 THE 1A AUXILIARY FEEDWATER (AFW) PUMP (BA) WAS DECLARED INOPERABLE DUE TO FAILING A TRIP TEST. IN ACCORDANCE WITH TECH SPEC 3.7.2.C THE MOTOR DRIVEN AFW PUMPS WERE REALIGNED TO PROVIDE TWO OPERABLE FLOW PATHS TO THE STEAM GENERATORS AT 1940 HOURS. THE FLOW RATES PROVIDED TO THE STEAM GENERATORS WERE NOT VERIFIED WITHIN 8 HOURS TO BE 105 GPM TO EACH STEAM GENERATOR (S/G) USING THE NEW PUMP LINEUP, AS REQUIRED BY TECH SPEC 3.7.2.D. AND TECH SPEC 4.7.2.A.1.B. ALL FLOWS WERE PROPERLY RESET TO 105 GPM TP EACH S/G AT 1148 HOURS ON 7/24/89. THE CAUSE OF THE EVENT WAS AN ADMINISTRATIVE ERROR IN THAT THE STATION'S INTERPRETATION OF APPLICABLE TECH SPECS WHEN DECLARING THE 1A AFW PUMP INOPERABLE DID NOT SATISFY TECH SPECS. A CONTRIBUTING FACTOR TO THE EVENT WAS THE REVISION OF SYSTEM OPERATING INSTRUCTION (SOI-10), AUXILIARY FEEDWATER, IN USE WHICH DID NOT ADDRESS THE RESETTING OF AFW THROTTLE VALVE SETTINGS. THE INITIAL COMPUTER ANALYSIS INDICATES THAT THE FLOW RATES EXPERIENCED THROUGHOUT THIS EVENT ARE BOUNDED BY THE UPDATED LOSS OF NORMAL FEEDWATER (LNFW) ANALYSIS CURRENTLY UNDER STATION REVIEW. A SUPPLEMENTAL REPORT WILL BE SUBMITTED FOLLOWING EVALUATION OF THE LNFW REPORT DETAILING THE ANALYSIS PERFORMED.

[286] ZION 2 DOCKET 50-304 LER 84-004 REV 02
UPDATE ON PLANT CONDITION NOT BOUNDED BY SAFETY ANALYSIS.
EVENT DATE: 012084 REPORT DATE: 083089 NSSS: WE TYPE: PWR
OTHER UNITS INVOLVED: ZION 1 (PWR)

(NSIC 215093) THE ORIGINAL WESTINGHOUSE LOSS OF FEEDWATER TRANSIENT ANALYSES RESULTS SHOW ZION UNITS 1 AND 2 WOULD PROVIDE INADEQUATE AUXILIARY FEEDWATER FLOW TO THE STEAM GENERATORS IN THE EVENT OF LOSING ALL FEEDWATER. THIS SHORTFALL IS MAINLY DUE TO THROTTLING VALVES WHICH ARE SET TO 105 GPM FOR EACH STEAM GENERATOR BECAUSE OF WATER HAMMER CONCERNS. A STANDING ORDER REQUIRING OPERATORS TO INSURE ADEQUATE FLOW DURING SUCH AN EVENT WAS ISSUED. ZION SUBSEQUENTLY RECEIVED A NEW LOSS OF FEEDWATER ANALYSIS FROM WESTINGHOUSE. THE ANALYSIS ASSUMED AN INITIAL REACTOR POWER OF 102% RATHER THAN 102% OF 118% ENGINEERED SAFETY FEATURE (ESF) POWER, AND ALSO ASSUMED THAT 308 GPM IS DELIVERED TO THE STEAM GENERATORS. ENOUGH MARGIN EXISTED IN THE ANALYSIS SUCH THAT NO ADDITIONAL ACTIONS WERE REQUIRED TO ASSURE ADEQUATE COOLING AND THE STANDING ORDER WAS CANCELLED. THE METHODOLOGY FOR THE LOSS OF FEEDWATER TRANSIENT ANALYSIS IS THE SAME AS THE ORIGNAL FSAR. THE REDUCTION IN INITIAL REACTOR POWER IS LESS CONSERVATIVE BUT CONSISTENT WITH THE NRC STANDARD REVIEW PLAN (NUREG-0800). IN THE PREVIOUS SUBMITTAL CONCERNING THE ZION LOSS OF NORMAL FEEDWATER ANALYSIS (LONFW), AN OVERSIGHT WAS MADE CONCERNING THE LOSS OF ALL A.C. POWER TO THE STATION AUXILIARIES TRANSIENT. THE LOSS OF ALL A.C. POWER ANALYSIS SHOULD HAVE BEEN INCLUDED IN THOSE SUBMITTALS.

[287] ZION 2 DOCKET 50-304 LER 88-011 REV 01
UPDATE ON SECOND LEVEL UNDERVOLTAGE CONTACT NOT WIRED PER DRAWING IN 480 VOLT TRANSFORMER BREAKER AUTOCLOSE CIRCUIT.
EVENT DATE: 111188 REPORT DATE: 082389 NSSS: WE TYPE: PWR

(NSIC 215095) WHILE PERFORMING A SYSTEM WALKDOWN FOR A MODIFICATION, WORKERS DISCOVERED LOOSE WIRES IN ESSENTIAL SERVICE (ESS) BREAKER 2484. INVESTIGATION SHOWED THAT THE WIRES WERE A SECOND LEVEL UNDERVOLTAGE CONTACT AND SHOULD HAVE BEEN INSTALLED IN 1986 AS PART OF THE SECOND LEVEL UNDERVOLTAGE MODIFICATION. THIS INCORRECT INSTALLATION RESULTED IN INABILITY OF THE 480 VOLT TRANSFORMER FEED BREAKER FOR ESS BUS 238 TO CLOSE IN ON A SECOND LEVEL UNDERVOLTAGE EVENT. IT WAS DETERMINED THAT THIS MADE 2B AUX. FEED PUMP INOPERABLE FOR SECOND LEVEL UNDERVOLTAGE. THE CAUSE OF THIS EVENT WAS IMPROPER INSTALLATION AND POST MODIFICATION TESTING. A REVIEW OF THE FSAR SHOWED THAT INOPERABILITY OF BUS 248 BLACKOUT LOADS ON SECOND LEVEL UNDERVOLTAGE IS BOUNDED BY THE SAFETY ANALYSIS. IN ADDITION, ABNORMAL OPERATING PROCEDURES PROVIDE GUIDANCE TO SEQUENCE BLACKOUT LOADS MANUALLY. SINCE 1986, THERE HAVE BEEN SIGNIFICANT IMPROVEMENTS IN THE MODIFICATION PROCESS WHICH MAKE RECURRENCE UNLIKELY.

[288] ZION 2 DOCKET 50-304 LER 88-013 REV 01
UPDATE ON UNMONITORED VENT DUE TO MISSED VENT PATH RAD MONITOR SURVEILLANCE CAUSED BY MISCOMMUNICATION.
EVENT DATE: 121988 REPORT DATE: 080789 NSSS: WE TYPE: PWR

(NSIC 214973) WITH ZION UNIT 2 AT COLD SHUTDOWN FOR REFUELING, CONTAINMENT VENTILATION RADIATION MONITOR 2RIA-PR40 (PR40) WAS TAKEN OUT OF SERVICE ON 12/7/88. THIS MONITOR IS REQUIRED OPERABLE DURING CONTAINMENT VENTING, OR SHIFTLY GRAB SAMPLES SHALL BE TAKEN. DUE TO INADEQUATE COMMUNICATIONS BETWEEN CONTROL ROOM AND RADIATION PROTECTION (RP) PERSONNEL, SHIFTLY GRAB SAMPLES WERE NOT TAKEN DURING TWO CONTAINMENT VENTS WHICH WERE PERFORMED WHILE PR40 WAS OUT OF SERVICE. DOCUMENTATION OF THE CONTAINMENT VENTS SHOWS THAT THE RELEASES WERE PROPERLY QUANITIFED AND THAT 10CFR20 LIMITS WERE MET. ALTHOUGH ITS OPERABILITY SURVEILLANCE WAS NOT DOCUMENTED, THE MONITOR WAS ACTUALLY CAPABLE OF PERFORMING ITS FUNCTION DURING BOTH RELEASES.

[289] ZION 2 DOCKET 50-304 LER 89-008
LOSS OF NSSS ANNUNCIATION DUE TO BLOWN POWER SUPPLY FUSES.
EVENT DATE: 062489 REPORT DATE: 072489 NSSS: WE TYPE: PWR
VENDOR: GENERAL ELECTRIC CORP. (NUCLEAR ENG DIV)

(NSIC 214919) ON 6/24/89, AT 1330, AN EQUIPMENT OPERATOR WAS GROUND CHECKING ON DC BUS 211 FOR A +90 VOLT GROUND. AT 1345 AFTER THE GROUND WAS DISCOVERED ON BREAKER #25 (MAIN FEED TO BOP ANNUNCIATOR CABINET 2CB50) IN 125 VDC DISTRIBUTION CENTER 211, IT WAS REPORTED THAT WHEN RE-CLOSING BREAKER #25, BREAKER #27 (MAIN FEED TO THE NUCLEAR STEAM SUPPLY SYSTEM (NSSS) ANNUNCIATOR CABINET 2CB74) LOCATED DIRECTLY BELOW BREAKER #25 OPENED SO THAT POWER WAS LOST TO THE NSSS ANNUNCIATORS IN THE CONTROL ROOM. DUE TO THE UNCERTAINTY OF THE ABILITY OF BREAKER #27 TO OPERATE PROPERLY, THE EMERGENCY DC FEED WAS TURNED ON AT 1515. HOWEVER, WHEN THIS FEED WAS ENERGIZED, ALL THE FUSES FED FROM IT WERE BLOWN. IT WAS DISCOVERED THAT THE POLARITY ON THIS FEED WAS REVERSED. AN ALERT LEVEL GSEP WAS INITIATED SINCE NSSS ANNUNCIATION HAD NO DC FEED. THE RESERVE FEED POLARITY OF THE WIRING TO THE FUSES WAS CORRECTED, ANNUNCIATION WAS RESTORED, AND THE EVENT CONCLUDED.

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

1

IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

SYSTEM INDEX

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

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SYSTEM INDEX

KEYWORD INDEX

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

KEYWORD INDEX

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VENDOR CODE INDEX

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BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE

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For month of September 1989

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4. DATE REPORT COMPLETED

MONTH	YEAR
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11a. TYPE OF REPORT

Monthly Report

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October 1989

12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

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

14. DOCUMENT ANALYSIS - a. KEYWORDS/DESCRIPTORS

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

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